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ENCLOSURE 2

SAFETY EVALUATION OF THE REQUEST TO OPERATE THE SHOREHAM NUCLEAR POWER STATION AT 25% POWER ACCIDENT EVALUATION

1 INTRODUCTION

The staff has completed a review of the PRA-based portion of LILCO's request to operate Shoreham Nuclear Power Station at 25 percent of full power (Reference 1). The PRA which forms the basis of the request is an updated version of the original full power PRA, modified to account for operation at 25 percent power. The staff has previously reviewed the original PRA (Reference 2); the results of that review are provided in Reference 3. The objective of the present review was to assess the validity of the major technical arguments upon which the utility's 25 percent power request is based. These arguments can be summarized as follows:

1. Reduced Vulnerability to Core Damage Accidents

   With operation at 25 percent power, decay heat levels are reduced to the extent that (1) certain plant features, such as turbine bypass, are capable of mitigating accidents prior to core melt and (2) accidents will evolve more slowly allowing considerably greater time for recovery actions. These factors, in conjunction with a number of plant upgrades which have been implemented, will result in a reduced vulnerability to severe core melt accidents at Shoreham.

2. Increased Time Interval Available for Emergency Response

   For accidents which are not arrested prior to core melt, reduced decay heat levels associated with 25 percent power operation will result in a significant delay in both core melt progression and onset of releases from containment. This delay represents an increase in the time available for emergency response.

3. Reduced Offsite Consequences

The magnitude of source term releases for accidents initiated from 25 percent power are less than predicted for similar accidents initiated at 100 percent power due to a proportionally smaller initial fission product inventory at the lower power level. The reduced source terms, in conjunction with the delayed times of release mentioned above, translate into reduced offsite consequences.

The staff review was divided into three main parts corresponding to the three utility arguments. These three parts and their objectives are described below:

Part 1 - Comparative Evaluation of Sequences with Potential Early Risk Impact

The objective of this part of the review was to assess the validity of the utility's assertion that the frequency of core melt accidents will be significantly reduced by (1) operation at 25 percent, and (2) a number of plant upgrades which have been implemented. Emphasis of the review was on treatment of: risk-important sequences (e.g., ATWS, station blackout, and interfacing system LOCA), initiating frequencies, time for operator actions, and treatment of external events. The review focused on the differences in these areas at 25 percent and 100 percent power, and not on the estimates of core melt frequency in an absolute, quantitative sense.

Part 2 - Effect of Power Restriction on Timing of Severe Accidents

The objective of this segment of the review was to assess the validity of the utility's calculated results for sequences identified as risk-important, with special emphasis on characterization of the timing of events in the accident progression, i.e., core uncovery, core melt, and vessel failure.

Part 3 - Effect of Power Restriction on Offsite Consequences

The objective of this segment of the review was to assess the adequacy of the utility's treatment of source terms, including initial fission product inventory for reduced power levels, modelling assumptions and calculated results regarding fission product releases and deposition, and treatment of fission product retention in the secondary containment building. A second objective was to assess the reasonableness of the utility's offsite consequence analyses, and to perform independent consequence analyses, as needed.

The Part 1 evaluation allows an assessment of the first LILCO claim regarding the impact of power restriction and plant upgrades on vulnerability to core damage accident likelihood at Shoreham Nuclear Power Station (SNPS). Similarly, the second and third parts provide information necessary to assess the study claims regarding increased times for operator actions and emergency response, and reduced offsite consequences at 25 percent power.

The organization of this report parallels the three major segments of the staff's review described above. Section 2 provides the staff's evaluation of sequences with potential early risk impact. Sections 3 and 4 provide the staff's evaluation of the effect of the power restriction on the timing and consequences of severe accidents, respectively. The summary and conclusions of the review are presented in Section 5.

2 EFFECT OF POWER RESTRICTION ON CORE MELT FREQUENCY

This section summarizes the major results of the staff review of the Shoreham 25 percent PRA evaluation of core melt frequency. The objective of the review was to assess the validity of the utility's assertion that the likelihood of incidents that can potentially result in core melt will be significantly reduced relative to full-power operation. The utility argument was based on a comparison of core melt frequency estimates for 25 percent power with those previously reported in the 1983 Shoreham Nuclear Power Station PRA for full-power operation of the plant. Thus, observed reductions were due to a combination of operating at a reduced power level (25 percent of full-power) and a number of plant upgrades which have been implemented at the plant since the publication of the 1983 full-power PRA.

The following 4 types of sequences were identified as important by the staff, on the basis of their contribution to core melt frequency and risk in the 25 percent PRA. These sequences were also found to be important in the staff's review of the original Shoreham PRA (for 100 percent power) and other PRAs.

1. Anticipated Transient Without Scram (ATWS)

2. Loss of Coolant Accidents (LOCAs)

3. Loss of Offs[illegible]

4. [illegible] of Injection

ATWS and [illegible] LOCA sequences are of interest because of their rapid nature and [illegible] early challenge to operators and offsite response. Loss of off[illegible] power and loss of injection sequences are of interest because they generally represent the major contributors to total core melt frequency for BWRs.

As part of the staff's review of core melt frequency, a focused evaluation was performed of the modelling of several of these sequences in the PRA. The sequences considered were: (1) ATWS sequences, (2) LOCAs outside containment, and (3) station blackout sequences. The staff's assessment of the modelling of these sequences as well as other factors affecting the reported estimates of core melt frequency is summarized in the discussion that follows. Further technical details and discussions of the review are included in Appendix A.

Anticipated Transients Without Scram (ATWS) sequences represent the cases where the plant is challenged by an off normal condition (accident initiator) that requires termination of the fission reaction, and the reactor protection system fails to function. The contribution of these sequences to core melt frequency was reported by the utility to drop by approximately a factor of three for 25 percent operation as compared to the value reported in the 100 percent power PRA.

Restriction of the ormal power level to 25 percent creates a unique situation for the ATWS conditions in that the Turbine Bypass Value (TBV) can deliver 25 percent of rated steam flow to the main condenser. This represents a success path which is not available at full-power operation. In general, the staff agrees with the analysis of ATWS sequences that shows a reduction in core melt frequency contribution as compared to the estimates reported in the full-power PRA.

Loss of Coolant Accidents outside of the reactor containment involve release of primary coolant to the environment. This release is associated with failure of the high pressure to low pressure boundary in systems interfacing the reactors primary cooling piping. The 25 percent power PRA showed the contribution of these sequences to core melt frequency to be reduced by about a factor of three as compared to the Shoreham full-power PRA. This decrease is primarily due to changes in analysis of the pressure boundary failure and not to the effect of power reduction.

Station Blackout, which is complete loss of Alternating Current (AC) electrical power in the plant (both offsite power and onsite emergency AC) represents an important challenge to plant safety. This is due to the dependence of systems required for reactor core cooling and containment heat removal on AC electric power. Station blackout sequences are typically initiated by loss of offsite power. The likelihood of loss of offsite power depends on the reliability of the power grid and its susceptibility to severe weather. Loss of offsite power can also be induced by a seismic event. The contribution of loss of offsite power sequences to core damage frequency depends on the reliability of onsite AC power sources, and on the time period available to recover AC power. Redundant AC power sources exist in Shoreham; these include diesel generators and a gas turbine. The utility study showed a significant reduction in core melt frequency resulting from loss of offsite power relative to the 1983 PRA (with the exception of seismically induced loss of offsite power). The staff concludes that the results are reasonable and the credit given to the additional sources of onsite AC power is justified.

The staff review also assessed the adequacy of the treatment of external events in the PRA, since external events (such as earthquakes, fires and floods) carry the potential for high risk significance due to their ability to induce conditions that initiate accidents and their potential to fail systems that can mitigate these accidents.

The 100 percent power PRA identified flooding from sources inside the plant (internal flooding) as a leading contributor to the Shoreham estimate of core melt frequency. The dominant flood scenario occurred at elevation 8' of the reactor building where all of the plant emergency core cooling system pumps are located. The 25 percent power PRA does not show a significant contribution from internal flood scenarios. The primary reason is the credit given to the CRD pumps which is located above the reactor building flood elevation. The CRD pumps are capable of maintaining reactor vessel inventory for accident initiators occurring during 25 percent power operation. The credit taken for those pumps is judged by the staff to be reasonable and consistent with other sequences in the PRA which took credit for this alternate high pressure injection source.

The staff did not perform a detailed review of the seismic analysis for Shoreham. However, the staff had previously reviewed the seismic hazard calculations performed for the nearby Millstone 3 site by the same subcontractor as used by LILCO. That review indicated that the seismic hazard could be increased by an order of magnitude due to uncertainties. The staff has compared the seismic hazard curves from the Shoreham PRA to preliminary curves available for the Shoreham site from the Seismic Hazard Characterization Project (SHCP). In contrast to Millstone, the Shoreham SHCP curves are closer to those used in the utility PRA. Based on this comparison, it is our judgment that an increase in the utility estimates of seismic hazard by a factor of five would represent a reasonable high estimate of uncertainty for regulatory purposes at Shoreham. This is not to say that this high estimate represents the true upper limit of scientific uncertainty or that the true seismic hazard could not be less than that proposed in the Shoreham study. Certainly there is no compelling evidence in the historic record that would indicate any likelihood of large earthquakes in eastern Long Island. If the increase in seismic hazard where to translate into an equivalent increase in core melt frequency for seismic events at Shoreham, i.e., a factor of five, the frequency of seismically-induced core melt sequences would increase to approximately 1×10^{-5}, which is about one-fifth that for internally-initiated events. It should be pointed out, however, that comparisons between seismic and nonseismic core melt frequency estimates are not completely valid since mean seismic hazard estimates directly reflect modelling uncertainties, whereas internal event estimates do so to a much lesser extent. As a result, comparisons of the means tend to overestimate the relative contribution of the seismic events to core damage and risk. Further, this effect would influence the results in both the 100 percent power PRA and the 25 percent power PRA.

The 25 percent PRA reported more than an order of magnitude reduction in the fire contribution to core melt frequency as compared to the full-power PRA. As detailed in Appendix A, the staff has identified several areas relating to the fire analysis which should be addressed by the utility. However, our judgment is that they would not significantly alter the PRA results.

In summary, operation at the reduced power level results in a reduction in the overall core damage frequency of about a factor of two. This reduction, however, is well within the uncertainties associated with estimating core melt frequency, especially considering that the reported results are in the form of point estimates and that uncertainties can be much larger than a factor of two. External events (seismic and fires) and estimates of human error data are the potential major contributions to these large uncertainties.

Based upon the limited review performed on the systems analysis segment of 25 percent power PRA submittal, the staff concludes that core melt frequency at 25 percent power is not significantly different than at 100 percent power.

3 EFFECT OF POWER RESTRICTION ON TIMING OF SEVERE ACCIDENTS

This section provides the results of the staff's evaluation of the utility's claims regarding the effect of operation at 25 percent power on the time available for operator actions and emergency response. The section is divided into two parts. The first part describes staff analyses performed for a limited number of sequences to determine the effect of the power restriction on severe accident timing. The emphasis of these analyses was on establishing the timing of key events in the core melt progression up to the time of reactor vessel failure. The second part describes the staff's assessment of the time of releases to the environment for broad classes of accidents at 25 percent power.

As such, the information presented provides a basis for identifying which types of severe accident sequences will likely require prompt offsite emergency response, and the amount of time available prior to significant releases from the reactor coolant system and containment.

3.1 Timing of Core Melt Progression

The spectrum of core melt accidents in BWRs can be grouped into five generic accident classes or plant damage states on the basis of similar challenges to the core and containment functions, and similar possibilities for core melt progression. The plant damage states define the boundary conditions for the subsequent containment event tree (CET) analysis, the purpose of which is to systematically assess and quantify the relative probability of successfully mitigating the challenges to core/containmer , or of obtaining a particular release. The product of the CET analysis is a number of quantified radionuclide release end states; these are typically grouped into a smaller set of release bins or categories on the basis of similar release characteristics.

Six release categories were defined by the utility to represent the 25 percent power accident spectrum for Shoreham. The release characteristics for each of these categories are described in Table 2. Additional information is represented in Table 3 for each of six release categories, specifically, the contribution of all sequences assigned to the release category to total core melt frequency, the time to core slump calculated by the Modular Accident Analysis Program (MAAP) code for the sequence chosen to represent the release category, and the time of releases to the environment for the release category estimated based on analyses performed using the MAAP. Statements made in Section II.C.4(c) of Reference 1 indicate that release categories 1 and 2 account for the bulk of the injury-threatening doses.

To assess the effect of the power reduction on the nature and timing of accident progression, the staff performed confirmatory calculations for several of the sequences used to represent release categories. The sequence types considered were: (1) anticipated transient without scram (ATWS), (2) large break LOCA, (3) station blackout, and (4) transient with loss of injection. These calculations modeled only the thermal-hydraulic behavior of the reactor coolant system up to the time of reactor vessel failure.

A brief discussion of the calculations performed for each sequence type and the results is provided in the subsections below. A discussion is then provided of the applicability of the findings to other sequences.

3.1.1 ATWS Sequences

An ATWS is an expected operational transient (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power) which is accompanied by a failure of the reactor trip system to shut down the reactor. As part of the assessment of ATWS sequences at 25 percent power, the following two aspects of the accident were considered: (1) reactor response to sudden reactivity insertion under ATWS conditions, and (2) core melt progression for the ATWS sequence defined for release Category 1 of the utility submittal. These are discussed below.

3.1.1.1 Reactivity Insertion at 25 Percent Power

Detailed studies have demonstrated that successful operator actuation of the standby liquid control system (SLCS), will bring the ATWS sequence in BWRs under control. In the event of failure of the SLCS function, the operators are directed by procedure to lower the water level to the top of the core and to depressurize the reactor vessel. Recent preliminary work at Rensselaer Polytechnic Institute (Reference 4) suggests that the Shoreham reactor would be subcritical in this configuration even without liquid poison injection, that is, with the control blades in their 25 percent power positions, the reactor vessel water levels at the top of the core, and the reactor vessel pressure at 200 psia (or below). Nevertheless, to account for the possibility that the reactor does remain critical in this configuration, analyses were performed of the power and pressure response during an ATWS event.

The transient analyses were performed by Oak Ridge National Laboratory (ORNL) using the BWR-Long Term Accident Simulation (BWR-LTAS) code developed at ORNL and described in Reference 5. The sequence considered was an ATWS initiated by transient-induced closure of the main steam isolation valves (MSIVs). The analyses assumed that the control blades remained stuck in their normal position and that no operator actions were taken. Two cases were considered, one with the blades in the position corresponding to 25 percent power and the other with the blades in the full-power position.

The calculated results for the two cases are shown in Figures 1 and 2. In the analyses, HPCI, RCIC, and CRD injection maintain reactor vessel water level above the top of core until failure (by assumption) of the HPCI turbine at a suppression pool temperature of 210°F. With HPCI system failure, reactor vessel water level decreases, leading to ADS actuation. As the vessel is depressurized into the regime in which the low pressure injection systems are able to pump cold water into the vessel, oscillations in injection flow, core power, and vessel pressure occur as a result of positive reactivity insertion associated with collapse of voids in the core by cold water. Similar trends are observed in both cases but the following key differences should be noted:

1. The time to ADS actuation and core uncovery is significantly later for 25 percent power,

2. The frequency and magnitude of pressure and power oscillations is reduced for 25 percent power, and

3. Drywell pressure remains below the design value for 25 percent power but exceeds it for 100 percent power.

Much of these differences in behavior can be attributed to the fact that the negative reactivity of the core voids relative to that of the control blades is less with the control blades in their 25 percent power configuration. It follows that perturbations that tend to collapse voids in the core region will insert less positive reactivity with the control blades in their 25 percent power positions than with the control blades in their 100 percent power positions. Hence, the core response to positive reactivity insertions caused by uncontrolled cold water injection by the low pressure ECC systems is more sluggish at 25 percent power.

On the basis of these calculations, the staff concludes that adequate mitigation of an ATWS accident sequence, given failure of complete shutdown, is much simpler for the control room operators if the control blades are in their configuration for 25 percent power operation as opposed to their configuration at 100 percent power. While we expect the likelihood of ATWS events to be extremely low at Shoreham due to hardware improvements and mitigative capabilities at 25 percent power, should an ATWS occur the safety concerns at 25 percent power would be substantially less than at full power.

3.1.1.2 Core Melt Progression for an ATWS Sequence

The release Category 1 sequence, identified in LILCO's submittal as Case C9D, is an ATWS initiated by MSIV closure at time = 0. According to the sequence definition, reactor vessel water level is maintained at the top of the active fuel by various combinations of HPCI, RCIC, CRD, and LPCI, while the SRVs vent steam to the pressure suppression pool. The primary containment is assumed to be vented from the wetwell airspace when the primary containment pressure reaches 60 psig in accordance with the Shoreham emergency operating procedures (EOPs). The vent line is assumed to fail at the flexible coupling which joins the vent line with the RBSVS ducting (reactor building elevation 101 ft), and the resulting harsh reactor building environmental conditions are assumed to fail all reactor vessel injection systems. In the utility analysis, the wetwell is vented at 1.5 hours into the event, at which time all injection is assumed to be lost. Core uncovery is predicted to occur at 1.7 hours, followed by onset of cladding relocation at 4.1 hours. Core slump and reactor vessel failure is calculated to occur at 10.4 hours, at which time the drywell vent is opened to maintain primary containment pressure at or below 70 psig.

To confirm the sequence of core degradation and reactor vessel failure events that would occur after loss of all injection, an independent calculation was performed by ORNL for the latter period of the accident. This analysis was performed using the Boiling Water Reactor Severe Accident Response (BWRSAR) code developed by ORNL and documented in Reference 6. The calculation was initiated at time 90 minutes (the time at which venting and loss of injection occurred in the utility analysis) and was run until postulated reactor vessel failure. Core power was controlled by user input and the control blade positions were established so as to approximate the actual 25 percent power configuration. The predicted timing of events is provided in Table 1, along with the results obtained by LILCO using the MAAP code.

Generally good agreement is noted between the BWRSAR and MAAP estimates of the time to start of cladding relocation and to slump of the first major portion of the core into the bottom head of the reactor vessel. However, the ORNL code predicts a much longer time to reactor vessel failure (30.8 h) than does the MAAP code (10.4 h). This is due to different modelling approaches taken in the two codes with regard to (1) the state of the debris which is assumed to slump into the bottom head, and (2) the extent of debris quenching which occurs in the bottom head.

The two different modelling approaches can be summarized as follows. In BWRSAR, radial columns or zones collapse when their average cladding temperature reaches 4250°F, at which time very little of the UO_2 mass in the region is molten (molten Zircaloy is relocated to the bottom head prior to that time). Falling mass is assumed to be quenched by the water in the lower plenum until the time of bottom head dryout. In MAAP, molten core materials are assumed to accumulate in the lower-most node of each radial zone until one of those nodes becomes completely molten; at that time the material in the molten node and any molten material in adjacent nodes falls to the bottom head. The MAAP models provide for only minimal interactions between the molten material and the water in the lower plenum, and hence the debris does not quench. Subsequent heatup and attack of the reactor vessel lower head by the molten debris is calculated, and produces vessel breach within tens of seconds to a few minutes. The MARCH code, discussed later, has the capability of modelling the heat transfer either way (i.e., with or without debris quenching) as a user option. The effect of the modelling differences on the estimated time to vessel failure is accentuated in the subject analyses due to the significant quantity of water in the bottom head of BWRs and the reduced decay heat levels in the core debris at 25 percent power.

The uncertainty in estimates of time to vessel failure, while significant, is reasonably well-bounded. The assumption of minimal debris quenching in the bottom head is considered by the staff to provide lower limit, conservative estimates of failure times, whereas, models which assume complete debris quenching and bottom head dryout prior to thermal attack of the bottom head may be somewhat optimistic and provide an upper limit. In reality, we would expect that the time to vessel failure would lie between the two extremes predicted by the models, but closer to the estimate obtained assuming debris quenching. This view is supported by the results of the TMI-2 core debris examinations performed to date (Reference 7). Hence, reactor vessel failure times for the ATWS sequence at 25 percent power would not occur until after nine hours following initiation of the transient, and may be delayed by as much as a day.

For comparison, results for an ATWS calculation at 100 percent power are presented in Table 5. The 100 percent power values are based on a MARCH 2 calculation performed previously for the Limerick plant which, like Shoreham, is a BWR/4 with a Mark II containment. Although the plant design characteristics, sequence definition, and computer codes are different for the two cases, they are judged to be sufficiently similar to illustrate the approximate effect of the power restriction on the timing of major events. The calculations indicate that the time to initial core slump and potential reactor vessel failure is extended from about two hours at full power to over nine hours at 25 percent power. It should be recognized that the ATWS event would proceed much differently than modelled here if the sequence were more realistically defined to include additional operator actions. However, in either case the 25 percent power restriction would substantially delay core melt progression and afford additional time for operator actions and protective measures.

An additional difference identified in the ORNL analysis concerns the quantity of hydrogen produced in-vessel. The BWRSAR ATWS calculation for 25 percent power indicates that approximately 2400 lbm of hydrogen are generated. (For the LOCA, station blackout, and loss of injection sequences discussed later the staff calculations indicate that about 1300, 1400, and 2100 lbm of hydrogen would be produced at 25 percent power.) The MAAP code consistently produces much less hydrogen than the staff calculations (typically a total of about 250 lbm). The reasons for this are well established and due largely to assumptions in MAAP regarding the formation of blockages in the core and termination of cladding oxidation (and hydrogen production) following cladding relocation; a more detailed discussion of this matter is presented in Appendix J.2 to Reference 8. Since hydrogen is produced as a result of an exothermic reaction (cladding oxidation by steam), production of larger quantities of hydrogen results in greater energy release during the core heatup process, potentially accelerating the core melt progression. However, as evidenced by the generally good agreement between the staff and utility estimates presented in Table 4, the impact of increased hydrogen production on the timing of core melt progression is not significant.

With regard to the effect of reactor power level on hydrogen production, staff calculations indicate that the difference in the total quantity of hydrogen produced at 25 percent and 100 percent power is within 300 lbm. This difference is not a critical consideration because a great deal of hydrogen is predicted to be generated regardless of the initial power level.

3.1.2 Large Break LOCA Sequences

Loss of coolant accidents involve the loss of reactor coolant via a breach in the reactor coolant system pressure boundary. LOCAs can occur either inside containment due to events such as pipe breaks, or outside containment as in the case of a loss of coolant to an interfacing system. Large break LOCA sequences, in general, represent the most rapidly evolving severe accident sequence. As indicated in Table 3, two of the six release categories in the LILCO PRA for 25 percent power are represented by large break LOCA sequences. These are release Categories 2 and 5.

The release Category 2 sequence, identified as Case CADRF, is a seismically-initiated recirculation line LOCA, with a coincident drywell head failure of 3 ft^2. All reactor vessel injection systems are lost. Only the refueling bay is credited for fission product removal, and the Reactor Building Standby Ventilation System (RBSVS) is assumed to be unavailable.

The release Category 5 sequence, identified as Case C3C, is a large LOCA with loss of all injection except that from the CRD hydraulic system. One or more drywell downcomers are assumed to fail upon reactor vessel failure, allowing bypass of the pressure suppression pool and the wetwell air space is assumed to be vented when the primary containment pressure reaches 60 psig. In the utility analysis of this sequence the wetwell air space is vented at 48 hours into the accident to maintain primary containment pressure at or below 60 psig.

In the utility analyses for these two cases, the timing of core degradation and reactor vessel failure events is similar: i.e., the core uncovers within about 30 seconds, begins to melt at approximately an hour, and slumps at approximately four hours. As indicated in Table 3, however, the time of fission product release to the environment is distinctly different for the two sequences; this is because the containment is failed in the release Category 2 sequence and is intact in the release Category 5 sequence.

To confirm the general nature of the timing of core melt progression and vessel failure, three large break LOCA calculations were performed by ORNL using the BWRSAR code. In the first calculation the drywell was assumed to be failed, as modelled in the CADRF sequence (release Category 2). In the second calculation, the containment was assumed to be intact, as modelled in the C3C sequence (release Category 5). It should be noted that this calculation did not fully simulate the C3C sequence in that the injection flow from the CRD hydraulic system was not modelled. This would have only a minimal effect on sequence progression since the injection flow would be expelled from the break without passing through the core. The third calculation was identical to the second except that the initial power level was changed from 25 percent to 100 percent.

The two BWRSAR calculations performed for 25 percent power yielded similar results regarding the timing of core melt progression; this is not surprising since the only difference between the calculations was the containment back-pressure. The calculated times for key events are presented in Table 4 along with the utility's values. The staff's values for the onset of cladding relocation and core slumping are consistent with the utility's, but indicate a somewhat earlier (about one hour) time to slumping. The staff's estimates of the time to vessel failure are considerably longer for the reasons described in Section 3.1.1.

A comparison of the BWRSAR-predicted core melt progression at 25 percent and 100 percent power is presented in Table 5 for the large break LOCA with no injection and intact containment. These time estimates are considered by the staff to be representative for the sequences selected to represent release Categories 2 and 5. The results indicate that the delay in key events afforded by the power reduction is significant: i.e., it shifts the time of onset of cladding relocation from 0.2 hours to 1 hour, and the time of core slumping from 0.7 hours to 3.3 hours. This shift represents additional time for operator actions and emergency response which would not be available if operating at 100 percent power.

3.1.3 Station Blackout Sequences

Station blackout is defined as a loss of all AC power (except vital AC supplied through DC inverters). This is caused by loss of offsite power and the subsequent failure of the diesel and gas turbine generators. The release Category 4 sequence, identified as Case C1A, is a station blackout sequence coupled with a stuck open relief valve and a failure to isolate the drywell equipment and floor drain lines. The RBSVS is not available. HPCI and RCIC (both turbine driven) are initially available, but HPCI is lost due to low HPCI turbine steam flow at 8.5 minutes, followed by loss of RCIC at 45 minutes.

To confirm the timing of accident events at 25 percent power, MARCH 3 calculations were performed by Battelle Columbus Laboratories (BCL) for the same accident sequence. The MARCH 3 modelling assumptions used were in accord with the methodology described in NUREG-0956. The effect of the treatment of debris quenching on time of bottom head failure was investigated in these calculations by considering (1) no debris quenching in the vessel head, consistent with the MAAP models, and (2) debris fragmentation and quenching upon contact with water in the vessel head, consistent with the BWRSAR models.

The predicted timing of key events is compared to the utility results in Table 4. Although significant differences in time to core uncovery are observed (1.5 hours in MARCH 3 versus 4.1 hours in MAAP) estimates of the time to onset of cladding relocation and core slump are in good agreement with the MAAP results, as is the time of vessel failure when no debris quench is assumed. When debris quench is assumed, the time to vessel failure is extended considerably (12 hours with no quench versus 49 hours with quench).

While the time to vessel failure predicted in the BCL calculation with debris quenching is somewhat higher than indicated in the ORNL calculations for comparable sequences (ATWS and Loss of Injection), the interpretation of the result is consistent with the ORNL results, i.e., delays on the order of a day are predicted when quenching is assumed and the initial water inventory in the bottom head is large. We conclude that the BCL calculation adequately confirms the timing of core melt progression events reported by the utility for this sequence at 25 percent power.

To show the effect of the 25 percent power restriction on severe accident event timing for the station blackout sequence, a comparison with the results for a similar calculation at 100 percent power is presented in Table 6. The 100 percent power values are based on a MARCH 2 calculation performed previously for the Limerick plant. The Limerick sequence is defined somewhat differently with coolant boil off initially taking place at high pressure and depressurization assumed after core uncovery, however, the differences in the timing of predicted accident progression illustrates the extent of the delays afforded by operation at 25 percent power.

3.1.4 Loss of Injection Sequences

Loss of injection sequences can be characterized as operational transients in which the reactor is successfully shut down, but reactor coolant injection systems fail to function. The release Category 6 sequence, identified as Case C6A1, is a transient with loss of all injection, i.e., a transient-induced scram, followed by failure of all of the systems that would normally be relied upon to deliver cooling water to the vessel as necessary to keep the core covered (normal feedwater, HPCI, RCIC, RHR core spray, and CRD flow). In the utility analysis of this sequence, core melting begins at 5.8 hours with reactor vessel failure occurring at 11.3 hours. The primary containment is not vented (pressure does not reach 60 psig), nor does it fail during the first

50 hours of the accident. Fission product releases are, therefore, limited to that associated with primary containment design leakage (0.5 volume percent per day).

A confirmatory calculation for the postulated total loss of reactor vessel injection at the Shoreham station was performed using the BWRSAR code. In this calculation it was assumed that the reactor had been operating at 25 percent power at the time of scram and, in spite of the long times involved, no injection source is ever recovered. For conservatism in the analysis, there is no modelling of pressure suppression pool cooling or operation of the drywell coolers. Also, the reactor vessel is assumed to remain at pressure. This sequence definition is consistent with that for the utility's C6A1 sequence. The calculated times for key events are presented in Table 4. Agreement with the MAAP results reported by the utility is good (with the exception of time to vessel failure and quantity of hydrogen produced, as discussed previously).

In order to clearly demonstrate the effects of operation at 25 percent power, the total loss of injection sequence was recalculated with all parameters the same except for the initial power, which was set at 100 percent of rated power. The difference in timing of the major events of the accident sequence are indicated in Table 6. The results indicate that relative to full power operation, delays of about five hours in the onset of cladding relocation, nine hours in the start of core slump and nine to 20 hours in the time of vessel failure would be realized by restricting operation to 25 percent of rated power.

3.1.5 Applicability of Results to Other Sequences

A number of observations can be made concerning the results reported in the previous four sections. First, for the sequences considered, the independent staff analyses approximately confirm the timing of core melt progression reported by the utility for operation at 25 percent power. Second, based on the staff comparison of the core melt progression at 25 percent versus 100 percent power, the delay in key events afforded by the power restriction is significant, i.e., on the order of hours. Finally, a number of differences remain in the modelling of the accident progression. Most notable are the differences between the staff and utility estimates of the time to vessel failure and the quantity of hydrogen produced in-vessel.

While only a limited number of sequences have been evaluated as part of the staff's review of the utility submittal, we believe that the same observations would hold true for the range of accident sequences that are expected to dominate core melt frequency at Shoreham. The underlying reason is that the observed delays in timing are directly attributable to the reduced decay heat level associated with operation at 25 percent power and that this reduced decay heat level will affect all sequences in a manner similar to observed here. Specifically, the time of core melt for sequences in which the reactor coolant system remains intact is characterized by the time required to boiloff the coolant inventory and subsequently heat the core to oxidation temperatures. Sequences of this type will, in the limiting case of loss of all injection, exhibit the same general behavior as observed for the station blackout and loss of injection sequences. If the reactor does not scram, the coolant boiloff is more rapid (due to decay heat plus some fraction of core power) but subsequent core heatup case with scram; core melt progression for such sequences could be approximated by the ATWS sequence considered previously. For sequences in which the coolant inventory is lost due to breach of the reactor coolant system, the delay in core melt afforded by coolant boiloff will be reduced (by an amount depending on break size and available injection flow), but at 25 percent power a considerable amount of time will still be required to heat the core to oxidation temperatures. The limiting case is represented by the large break LOCA sequence described previously. If the break size is smaller or coolant injection is available, core melt would be considerably delayed or averted.

Furthermore, the reasonably good agreement obtained between the staff and utility estimates of the timing of key core melt events suggests that the principal thermal-hydraulic and core heat transfer models which govern reactor coolant blowdown/boiloff, core heatup, and the early stages of core degradation are not fundamentally different in the utility and staff codes; thus, additional comparisons with MAAP results (for timing) would likely result in the same level of agreement as observed here. Similarly, in those areas in which differences between MAAP and the staff's results have been identified, these same differences would be expected to exist for other sequences as well.

3.2 Timing of Releases to the Environment

Estimates of the time of releases to the environment for the spectrum of core melt accidents have been developed by considering the estimated frequency of each of the plant damage states and release categories in the Shoreham 25 percent power PRA, the types of sequences which comprise the various damage states and release categories, and the time progression of these accidents at 25 percent power. Table 7, extracted from Reference 9, provides a description of the types of sequences which comprise each of the plant damage states, as well as the frequency of occurrence of each damage state at 25 percent power. The utility, in Reference 10, has estimated the time from the initiating event to the initial release of radiation to the environment for each release category within each plant damage state. The utility time estimates are reproduced as Table 8. Based on this assessment, the utility claims that approximately 74 percent of the core melt sequence (represented by release Categories 5 and 6) require 48 hours or more to proceed to an offsite release, while an additional 22.7 percent of the sequences (represented by parts of release Categories 1, 2, 3, and 4) require between seven and 14 hours to produce offsite releases. The remaining 3.3 percent of all core melt accidents would produce a release in about one hour.

The staff has performed a limited review of the utility analysis. This review focused on the timing of releases rather than on the fraction of core melt frequency allocated to each plant damage state and release category. An initial observation is that for several of the release categories, the estimated times of release reported in Table 8 are different than those used in the utility offsite consequence analysis (see Table 3). It is our understanding that the Table 8 values were developed by reviewing the fission product release histories calculated by MAAP for the representative sequence for each of the six release categories, and identifying the time at which the releases exceeded some assumed threshold. In contrast, the times used in the offsite consequence calculations are chosen to best represent the release history as a single "puff" release, and are not linked to a threshold. This difference in approach for estimating the times to release would appear to account for the differences between the time estimates in Table 3 and Table 8.

Using an approach similar to the applicant's, the staff has developed a characterization of the time of release for a spectrum of accidents at both 25 percent and 100 percent power. This assessment was performed at the plant damage state level rather than at the release category level. This avoids having to deal with complex issues and assumptions related to the CET analysis, the binning of CET end states into release categories, and the selection of representative sequences for the various release categories.

The approach taken by the staff was to conservatively estimate the time of release for a typical sequence for each damage state, and to couple these estimates with the utility's estimate of the fraction of core melt frequency for the damage state to obtain a distribution of release times. The major limitations of this approach are that (1) the sequence selected to represent a plant damage state may not be the limiting sequence (for timing) within the damage state and (2) the potential for early containment failure may not be adequately reflected in the release time estimates. However, these limitations should not significantly affect the results of the assessment for the following reasons. Foremost, release times are conservatively estimated by assuming reactor vessel failure at core slump and containment pressurization rates based on participation of the entire core in subsequent core concrete interactions. For all plant damage states, the estimated time to release is significantly less than a more realistic estimate of the time of vessel failure. Hence, early containment challenges associated with reactor vessel failure (e.g., in-vessel and ex-vessel steam explosions, direct containment heating, and containment liner melt-through) would realistically occur later than the estimated times of release. Also, while certain sequences within a given plant damage state may have release times shorter than the sequence selected to represent that damage state, it is the staff's judgement that the fraction of the core melt frequency associated with those sequences is not large enough to significantly alter the distribution of release times for the spectrum of accidents.

The results of the staff's assessment of the time of release to the environment for Shoreham is presented in Table 9 for 25 percent and 100 percent power operation. In both cases the frequency of each plant damage state is based on values reported by the utility. (These values are reported in Reference 1 for 25 percent power and References 2 and 11 for 100 percent power.) The estimated time of release for the various damage states is based on either staff estimates or utility estimates as described below. For operation at 25 percent power, the staff estimate of 14 hours for the Class I damage state is based on a loss of injection sequence, such as the station blackout or loss of injection sequences described in Section 3.1 and Table 6. Reactor vessel failure is assumed to occur coincident with slumping of the first radial zone of the core, or approximately 11 hours. Containment failure by venting is assumed to occur three hours later due to releases from core concrete interactions in which the entire core participates. No consideration is given to the more likely situation in which core debris would enter the pool and be quenched, resulting in much later or perhaps no containment failure. The release time of six hours for the Class III damage state was based on the large break LOCA sequence subject to the same assumptions regarding vessel and containment failure. A similar process was followed to estimate the time to release for Class I and III damage states at 100 percent power.

The time of release for the Class II plant damage state is based on analyses performed for a transient sequence with loss of decay heat removal. This sequence is a dominant contributor to the Class II plant damage state frequency at Shoreham. In this sequence, denoted TW, the reactor shuts down and emergency core cooling systems operate, but the suppression pool heat removal system fails. This leads to pool heatup and eventual containment overpressure failure prior to core melt. Because the core is at decay heat power level the time to containment failure is substantial. Calculations performed for a TW sequence in Peach Bottom (Reference 12) indicate that containment failure does not occur (for Peach Bottom at full power) until about 30 hours after sequence initiation, with subsequent core melt at approximately two days. Similarly, the time of release used in the Shoreham full-power PRA for release categories associated with the Class II damage state was 38 hours. The time of release would be even longer for operation at 25 percent power. Accordingly, the staff has estimated the time of release for the Class II plant damage state to be greater than 24 hours for operation at both 25 percent and 100 percent power.

The release times for the Class IV damage state are taken from the utility analysis since the staff did not have independent containment analyses for these cases. For 25 percent power, the time of release (7.0 hours) is based on the utility analysis of the ATWS sequence selected to represent release Category 3. For 100 percent power, the time of release (2.5 hours) is based on time estimates for the Class III damage state reported in the original 100 percent power PRA (Reference 2).

The time of release to the environment for the Class V and the seismically-induced reactor pressure vessel failure (SRPV) damage states at 25 percent and 100 percent power is taken to be the time to the beginning of cladding relocation for a large break LOCA with no injection. The rationale for this assumption is that (1) a significant amount of the noble gases and volatile fission products would have been released from the core by the time the core reaches the temperatures associated with cladding relocation, (2) the dominant sequences associated with these plant damage states involve large LOCAs or rupture of the reactor pressure vessel; hence, the reactor coolant system (RCS) provides little delay in the release of fission products from the core to the containment, and (3) the containment building is bypassed or ruptured by definition of these plant damage states, minimizing its effectiveness in preventing or delaying the release of fission products to the environment. It should be noted that a more realistic analysis which accounts for the actual release history from the core, and delays afforded by the RCS and containment would result in estimated times of release more on the order of one to three hours for 100 percent and 25 percent power operation.

A summary comparison of the utility and staff estimates of the distribution of the time of release for core melt accidents at Shoreham is presented in Table 10. The staff and utility estimates for 25 percent power are not significantly different for the release time windows considered. These results indicate that approximately 80 percent of all core melt sequences require 12 or more hours to proceed to an offsite release and approximately 95 percent of all accidents require six or more hours to produce a release.

Comparison of the time estimates for 25 percent power with those developed by the staff for 100 percent power illustrates the frequency weighted shift in time of release afforded by operation at 25 percent power. Under conservative assumptions regarding reactor vessel failure times and containment performance, the bulk of the releases at full power (approximately 75 percent) occur between two and six hours following accident initiation. The Class I damage state, with release at five hours, is the major contributor; Classes III and IV also contribute, with releases at just over two hours. For the same assumptions at 25 percent power, releases for the Class I damage state are delayed until 12 or more hours following accident initiation, and releases for Classes III and IV are delayed until between six and 12 hours.

A small fraction (three percent) of core melt accidents at Shoreham still result in releases on the order of an hour. These early releases are due almost exclusively to seismic events which induce simultaneous reactor pressure vessel and containment failure. The difference in the fraction of core melt frequency for this contributor at 25 percent and 100 percent power is attributed by the staff to differences in total core melt frequency estimates and rounding error rather than to some artifact of operation at 25 percent versus 100 percent power.

4 EFFECT OF POWER RESTRICTION ON OFFSITE CONSEQUENCES

This section provides the results of the staff's review of the utility's claim regarding reduced offsite consequences at 25 percent power. The factors which contribute to reduced offsite consequences are a smaller source term release at the lower power level, in conjunction with the delayed times of release discussed in Section 3. The staff's evaluation of the fission product inventory at 25 percent power is provided in Section 4.1. Important fission product release and retention mechanisms for Shoreham (namely, core concrete interactions and the Shoreham reactor building) are also discussed. The impact of the power reduction on offsite consequences is assessed in Section 4.2.

4.1 Source Terms at 25 Percent Power

The magnitude of radionuclide releases for accidents initiated from 25 percent power can be expected to be less than for similar accidents initiated at 100 percent power for two reasons. First, the initial fission product inventory would be smaller at the lower power level. Second, the evolution of certain fission products would be inhibited by the lower heatup rates and temperatures associated with the decay heat level at 25 percent power. An assessment of each of these aspects of the source term reduction is provided below.

4.1.1 Fission Product Inventories

In order to verify the expected lower radionuclide inventories for operation at reduced power, ORIGEN2 calculations were performed by BCL for the Shoreham core. Radionuclide inventories were calculated at the end of two, four, and six years of operation at 25 percent power. A comparison of the results is presented in Table 11. These results indicate that significant increases in the radioisotope inventory do not occur after the second year. Although the quantity of certain radioisotopes continues to increase with time, this increase is considered insignificant relative to its impact on core melt progression and offsite consequences.

The BCL results at the end of two years of operation at 25 percent power are compared in Table 12 with the inventories used in the LILCO analyses for 25 percent power operation. The latter were obtained by adjusting the WASH-1400 PWR inventories to account for differences in power and core size. It can be seen that the two sets of results are in reasonable agreement, with the BCL ORIGEN2 results being slightly higher. This is understandable when it is recognized that the WASH-1400 results were derived for the middle of an equilibrium cycle and thus correspond to slightly lower average exposure than the BCL calculation. The current analysis uses a later version of the ORIGEN code than that applied in WASH-1400. The differences between the values used in the LILCO analysis and the BCL ORIGEN2 values is not considered to be significant.

Also shown in Table 12 are the results for the end of equilibrium cycle for the Shoreham core at full power. Comparison of the values for 25 percent power with those for 100 percent power confirms that the power restriction indeed results in an approximate factor of four reduction in fission product inventory.

4.1.2 Fission Product Releases

The source terms used in the Shoreham 25 percent power PRA were obtained directly from MAAP analyses. The staff has reviewed these source terms for reasonableness and consistency with source terms that would be predicted using the staff methodology, i.e., the Source Term Code Package (STCP). The emphasis of the review was on the source terms for release Categories 1 and 2, as these release categories account for the bulk of the injury-threatening doses.

Two major concerns regarding source terms were identified by the staff. The first was that little or no core-concrete attack in the drywell was considered to occur in the MAAP analyses for Shoreham, and that the utility source terms therefore underestimate the contributions from several important fission product groups, e.g., tellurium and strontium. The second was that the credit for fission product retention in the secondary containment building appeared to be overstated in the utility source term estimates for certain release categories. The staff's assessment of core-concrete interactions and secondary containment building performance is provided separately in the two sections which follow. The development of source terms which account for staff concerns in these areas is discussed in Section 4.2.2.

4.1.2.1 Releases from Core-Concrete Interactions

The MAAP analyses for Shoreham assume that debris leaves the reactor vessel in a molten state and immediately flows through the pedestal downcomers into the suppression pool where it is permanently cooled. Although 10 percent of the core debris is assumed to remain in the drywell, the MAAP models do not predict significant core-concrete interactions. In contrast to the treatment in MAAP, both the BWRSAR and MARCH 3 codes predict that a major portion of the core debris will be solid within the bottom head at the time of vessel failure. Thus, the staff calculations do not support the contention that all core debris would exit the vessel in a molten state, and pass into the pressure suppression pool.

The more likely situation in the staff's view is that a substantial fraction of the core debris, e.g., 20 to 50 percent, would rapidly exit the vessel following bottom head failure and that the remaining debris would be released from the vessel over the next several hours. While the bulk of the core material may flow toward and eventually pass through the four steel downcomer pipes located within the reactor pedestal region, some interaction of the debris with the concrete drywell floor would be expected prior to the debris reaching the downcomers. The extent of this core-concrete interaction and associated fission product release is influenced by several factors including (1) the chemical composition (particularly the fraction of unreacted Zircaloy), discharge rate, and temperature of debris exiting the vessel, (2) the state of the debris bed on the drywell floor during subsequent core debris additions, and (3) the length of time which debris remains on the floor before draining into the suppression pool. Given the right combination of the above parameters, interaction of considerably greater than the 10 percent of core debris assumed by the utility would appear likely.

In order to assess the potential for concrete attack by some portion of the core debris, a series of four calculations were performed by Battelle Columbus Laboratories (BCL) using the CORCON portion of MARCH 3. The assumptions and principal results of these calculations are described below:

Case 1 - In the first CORCON case the entire inventory of core and structural debris was assumed to remain on the floor of the pedestal. This is not to imply that the debris would all remain in the pedestal, but to provide a point of reference and comparison with the results of other analyses. The initial conditions of the debris were those predicted by MARCH 3 for the early head failure case i.e., a mixed mean debris temperature at the predicted time of vessel failure of 3550°F. CORCON partitioned the debris into a metal and an oxide layer, with the latter predicted to be on the bottom. The oxide layer was predicted to remain solid over the 10-hour time period considered, even though the oxide layer temperature was predicted to increase to a peak of about 4040°F before declining. Concrete attack was predicted to be predominantly radial with an increase in cavity radius of 3.6 ft and axial penetration of 0.59 ft.

Case 2 - Since the initial mixed mean debris temperature was below the melting points of the oxides but above that of the metals, the second case considered assumed that the molten metallic components were able to flow down the downcomers but that the oxide phases remained on the pedestal floor. The initial debris temperature was again that from the MARCH 3 calculation. In this case the oxide debris remained solid and increased in temperature to a peak value of about 4090°F before declining. In the absence of chemical reactions between metals and the concrete there was relatively little concrete attack. The total radial and axial concrete attack over the time period considered was 0.46 and 0.49 ft, respectively.

Case 3 - The third case considered was similar to Case 2, except that only half of the total oxide inventory was assumed to remain on the pedestal floor; this would imply that the other half of the oxides were able to flow into the suppression pool with the metal phase. With the reduced mass of debris and the absence of chemical interactions the temperature of the debris was predicted to decrease continuously. The predicted radial and axial concrete erosion was 0.30 and 0.43 ft, respectively.

Case 4 - In the fourth case the debris were assumed to be at the effective liquidus temperature of 4130°F used in the in-vessel analysis. This corresponds to approximately the state of the debris exiting the vessel in the MAAP analyses. One fourth of the core was assumed to remain on the floor of the pedestal. For this case CORCON partitioned the debris into two layers, with the denser oxide layer on the bottom. The oxide was again predicted to be solid and remained below the liquidus temperature through the 10 hours of attack considered. In this case the rate of concrete attack was initially rapid and decreased with time; the debris temperature decreased monotonically from its initial value. The predicted extent of concrete erosion for this case was 2.3 ft in the radial and 0.33 ft in the axial direction.

The above analyses indicate that some attack of the pedestal floor by debris released from the reactor vessel is quite likely under a variety of assumptions. In this context, BCL augmented the above analyses with VANESA code calculations to assess the potential fission product releases that could be associated with such core-concrete interactions.

The results of the VANESA analyses for fission product release for the several cases of corium-concrete interactions are summarized in Table 13. Also shown in this table are VANESA results for Limerick which had been obtained in earlier studies at BCL. The types of concrete in the two plants are comparable.

Comparison of the results for the Shoreham full core at reduced power (Case 1) with the Limerick results indicates relatively little difference. This is to be expected since in both cases there is substantial unreacted Zircaloy in the debris: the chemical reaction of this Zircaloy with concrete dominates the behavior once high debris temperatures are attained. The principal effect of the reduced power operation is delay in time of the start of vigorous interactions. The predicted lower releases of ruthenium, lanthanum, and cerium for Shoreham may be attributable to somewhat lower temperatures for the reduced power operation.

If only the oxide phase is available to attack concrete (Case 2 and 3), the predicted results are different from the interaction of the entire core. Since the oxide phase is predicted to remain solid, heat transfer is conduction limited and high debris temperatures are predicted. In the absence of the chemical reactions associated with the metallic phase, however, the production of some of the more volatile oxides appears to be reduced and the predicted releases are simply due to the volatilization of certain elemental species. Thus, the predicted release of tellurium is enhanced, and those of cesium, strontium, lanthanum, cerium, and barium are reduced relative to the full core case. The predicted releases of ruthenium appear to be sensitive to the specific temperature history in each case, but are low under all the conditions considered here.

If it is assumed that only a fraction of the core debris can interact with the drywell floor, but that this fraction is at a temperature comparable to that assumed in the MAAP analysis (Case 4), the predicted fractional releases of radionuclides are only somewhat lower than those indicated for the entire core, and the releases occur rather rapidly.

The CORCON analyses described above indicate significant potential for concrete attack even if only a fraction of the core debris remains on the pedestal floor and interacts with concrete. The extension of the CORCON calculations to the predictions of fission product release by VANESA indicates substantial sensitivity to the assumptions regarding the nature and degree of debris interaction with concrete. For the cases considered, however, the potential for considerable ex-vessel fission product release is indicated. On the basis of these results, the staff concludes that the utility source terms do not adequately reflect the potential for core-concrete interactions. Indepenuent staff calculations which account for significant core concrete interactions are described in Section 4.2.2.

4.1.2.2 Retention in the Secondary Containment Building

An assessment was performed by the staff's contractor, Oak Ridge National Laboratory (ORNL) of the decontamination factors (DFs) for the Shoreham secondary containment building. The sequences of interest for this assessment were cases C9D, CADRF, and C1A, which were used to represent release Categories 1, 2, and 4, respectively. For these sequences, DFs of 10, 10, and 50 were claimed by the utility. No secondary containment DFs were claimed for the other three release categories.

A preliminary assessment of the secondary building DFs was obtained by comparing Shoreham's secondary containment characteristics to those of the Browns Ferry and Peach Bottom plants, which were previously analyzed in detail. This comparison indicated the following:

1. a total secondary containment DF of 10 for Case C9D appears reasonable based on the similarity between Shoreham's and Browns Ferry's volume, and heat sink and sedimentation area characteristics,

2. a refueling bay DF of 10 for case CADRF appears to be higher than can be justified based on previous ORNL calculations for Browns Ferry and Peach Bottom, and

3. a total secondary containment DF of 50 for C1A appears to be somewhat high, albeit this DF is claimed for a sequence in which high containment pressures are never achieved, the point of fission product release is into the reactor building basement, and the reactor building standby ventilation system is not operational -- all factors which would tend to increase DF.

It is important to note, however, that these judgments apply only if hydrogen burns do not occur in the secondary containment. While the utility analyses indicate that deflagration limits were not reached in any of the MAAP simulations performed for the 25 percent power PRA, the absence of hydrogen burns appears to be a result of the low Zircaloy oxidation fractions typically calculated by MAAP. If one considers the estimates of in-vessel hydrogen production obtained from the BWRSAR and MARCH 3 analyses, which are considerably greater than those calculated by MAAP, it is clear that hydrogen burns in the secondary containment cannot be precluded. Hence, a more detailed assessment was made.

Secondary containment hydrogen burn analyses were performed by ORNL for cases C9D and CADRF. These analyses were performed by ORNL using the MELCOR code in conjunction with a 13-cell model of the Shoreham secondary building, and the hydrogen/steam release histories obtained from the BWRSAR analyses discussed in

Section 3. The results of the analyses indicate that the use of the BWRSAR-predicted hydrogen sources would result in hydrogen deflagrations in the Shoreham secondary containment for both sequences analyzed. BWRSAR/MELCOR predictions for the C9D ATWS sequence indicate that a severe global burn would occur at approximately 16 hours into the accident, producing a peak reactor building pressure of six psid. BWRSAR/MELCOR predictions for the CADRF seismic LOCA sequence indicate that refueling bay hydrogen deflagrations would occur at 1.1 and 3.1 hours into the accident, with a peak induced pressure of 0.8 psid. The second burn approximately coincides with the time of postulated reactor vessel failure.

A potentially important observation made as part of the staff's review of the Shoreham secondary building performance is that operation of the Reactor Building Standby Ventilation System (RBSVS) can increase the severity of deflagrations and reduce secondary containment DFs. Operation of the RBSVS can actually increase the severity of secondary containment hydrogen deflagrations by promoting a well mixed secondary containment atmosphere, resulting in severe, global hydrogen deflagrations for cases in which at least 800 lbm of hydrogen are available. Such burns would tend to flush fission products from the secondary containment into the environment. RBSVS operation might also decrease the secondary containment DF for accidents in which the primary containment fails into the lower region of the reactor building, by actively transporting fission products from the lower regions of the building to the refueling bay (which would be the secondary containment failure location in most accidents).

An additional observation is that Shoreham's low RBSVS filter exhaust capacity renders the plant vulnerable to secondary building pressurization from primary containment blowdown. Primary containment blowdown rates as low as 1200 cfm could initiate pressurization of the secondary containment and leakage of fission products to the environment. This is an important consideration, since the utility estimates that primary containment venting procedure employed in most accidents will result in a 3000 cfm steam source to the reactor building.

In summary, while a variety of conservative and non-conservative modelling assumptions were made in the utility analyses, the dominant factors which would affect the calculated DFs are: (1) the absence of hydrogen deflagrations in the utility analyses, (2) the use of a non-conservative aerosol sedimentation area for cases C9D and C1A, and (3) the use of an erroneous (high) heat sink area for case CADRF. Correction of each of these deficiencies would result in a reduction in DF. The extent of the reduction cannot be assessed in the absence of detailed confirmatory calculations, but the staff believes that secondary containment decontamination factors would more likely range from two to five for cases C9D, CADRF, and C1A.

4.2 Offsite Consequences at 25 Percent Power

The approach taken by LILCO to determine the offsite consequences for operation at 25 percent power was to perform a MAAP analyses for each of the six representative sequences (one sequence for each of the six release categories identified in Table 3). The output from each MAAP run, specifically, the calculated fission product release fractions and release histories, was then used as the basis for defining the source term release characteristics for the respective release category. The release characteristics (in terms of time to release, duration of release, and fractions of fission product inventory released) were then input directly to the CRAC2 and the CRACIT codes to determine the offsite consequences for each of the release categories. An overall picture of risk is obtained by multiplying the consequences predicted for each of the release categories by the probability of the respective release category occurring given a core melt accident, (e.g., column 3 of Table 3) and summing over all release categories.

The offsite consequences for Shoreham at 25 percent power have been reported by the utility in the form of dose-distance curves. These curves reflect the contribution from each of the six release curves, weighted by their respective probabilities. The Shoreham dose-distance curves compare quite favorably with those presented in NUREG-0396 (Reference 13), with the utility curves falling typically a decade or more below the NUREG curves.

A limited review of the utility offsite consequences analysis was performed by the staff as described in Section 4.2.1. In addition, the staff performed independent offsite calculations were performed to investigate impact of the increased time for emergency response afforded by operation at 25 percent power. This is described in Section 4.2.2.

4.2.1 Review of Utility Analysis

The following aspects of the utility offsite consequence analysis were reviewed by the staff:

1. Adequacy of the meteorology data used in the analysis,

2. Consistency of reported source terms and release category probabilities with the reported dose-distance curves, and

3. Consistency of the utility CRACIT results with those predicted by the CRAC2 code.

The meteorology data used in the LILCO consequence calculations for Shoreham was reviewed by the Radiation Protection Branch of the Division of Radiation Protection and Emergency Preparedness. Based on this review, the staff concluded that the meteorology data should reflect expected conditions at the site, and therefore is acceptable for use in the Shoreham analysis.

To assure reproducibility of the dose-distance curves reported by LILCO, and consistency with the reported source terms and release category probabilities, a confirmatory CRAC2 calculation was performed by INEL. The CRAC2 input data used for the Shoreham analysis was supplied by LILCO on floppy disk. This input was compared to that listed in Table A-1 in Reference 14, and no substantive differences were identified. Several discrepancies between Table A-1 in Reference 14 and Tables A.5-2 and 3 in the utility submittal (Reference 1) were identified, but these were largely confined to release Category 6, and would not significantly affect offsite consequences.

A CRAC2 analysis was performed by Idaho National Engineering Laboratory (INEL) using the version of CRAC2 installed on the INEL mainframe computer and the utility-supplied code input. The calculated dose-distance probability distributions were compared to those reported by the utility and found to be in agreement. This indicates that the input and code version used by the utility was essentially the same as used by INEL, but does not address the validity of the source term input.

4.2.2 Independent Assessment of Offsite Consequences

The staff has performed an independent assessment of the effect of the power restriction of offsite consequences. The approach taken was to develop source terms for a slowly evolving sequence which represents the bulk of the core melt frequency for Shoreham, as well as a rapidly evolving but less likely sequence, and to focus on the offsite consequences for these source terms. It is recognized that a complete picture of risk is not obtained by focussing on only two types of releases and their consequences. However, it is the staff's view that consideration of the offsite consequences for these sequences provides a perspective on the effect of the power reduction for the range of accidents that can reasonably be expected at Shoreham.

A set of calculations were performed for each source term by the staff's contractor, INEL, to address the effect of reduced source terms and delayed times of release on offsite consequences. The calculations involved modifying the CRAC2 input, and recomputing the dose-versus-distance probability distributions. The CRAC2 isotope subgroup data were modified by increasing the multiplier for the activities by a factor of four, representing an increase in power from 25 percent to 100 percent. The source term input was modified to include revised time of release, duration of release, and release fractions for both 25 percent and 100 percent power cases. The release fractions were further modified by assuming containment/reactor building DFs. One case assumed that the DF was one, or no decontamination, while the other case assumed the containment/reactor building was effective in reducing the source term, except the noble gases, by a factor of five. Key input assumptions in the CRAC2 analyses were that the population: (1) does not evacuate until 24 hours after the release, and (2) continues normal activities until evacuation (i.e., shielding factors of 0.75 and 0.33 were used for cloudshine and groundshine, respectively).

To provide some perspective as to the additional time for protective actions afforded by operation at 25 percent power, dose-versus-time figures were also generated. Although time dependent output is not available with CRAC2, several CRAC2 calculations were linked together to illustrate the influence of ti..

upon the probability of a dose being exceeded. The CRAC2 evacuation input was modified to freeze the d[illegible]e calculations at specific times after the release. T[illegible] CRAC2 calculations were performed by instantaneously evacuating all the [illegible] around the plant at specific times after the release of the radioactive [illegible] By performing several different evacuation time calculations, a dose-[illegible]ime curve was obtained. It was verified that the time-dependent results [illegible]d correctly satisfied the limiting cases at 0 and 24 hours. Instantaneous [illegible]ation at 0 hours resulted in no dose to the public, whereas, instantaneous evacuation at 24 hours produced the base case probability distributions.

4.2.2.1 Slowly Evolving Sequences

The Class I plant damage state accounts for approximately 80 percent of the total core melt frequency in the Shoreham 25 percent power PRA. Accidents in this class can be characterized as transients with reactor scram, coupled with a loss of reactor coolant injection. Such sequences may progress either at high reactor vessel pressure (e.g., failure of high pressure injection and depressurization systems) or at low pressure (e.g, failure of both high and low pressure systems). In either case, mass and energy releases occur over an extended period as the reactor coolant is boiled off due to decay heat. As evidenced by the calculations presented in Table 6, the timing of major events in the core melt progression, up to core slump, are not significantly different for high pressure and low pressure sequences.

Source terms were developed by the staff to represent releases which might occur for typical Class I BWR transients at 25 percent and 100 percent power. Core melt progression and reactor vessel failure times for such transients would be similar to those for the loss of injection and station blackout sequences described in Section 3.1. For these sequences at 25 percent power, reactor vessel failure is assumed to occur coincident with slumping of the first radial zone of the core; this is estimated to occur at 11 hours based on the results presented in Table 6. For full power, the time of vessel failure was estimated to be 3.5 hours. This is about midway between the times of core slump and bottom head failure reported in Table 6. The containment building is initially intact, but is postulated to fail at some time subsequent to reactor vessel failure as a result of ensuing core concrete interactions.

Two source terms were used to address different modes of releases from the containment. The first source term, Table 14, is based on releases occurring as a result of deliberate venting of the containment wetwell at 75 psia, in accordance with the Shoreham Emergency Operating Procedures. The second source term, Table 15, is based on releases occurring as a result of containment overpressure failure in the drywell at 135 psia. Each of these source terms is discussed below.

The rate of containment pressurization for a Class I transient in Shoreham and, hence, the time of containment venting or containment overpressure failure is strongly dependent on assumptions regarding the transport of core debris to the suppression pool following reactor vessel failure. If, as assumed in the utility analysis, essentially all of the debris rapidly enters the suppression pool with minimal interaction with the concrete diaphragm floor, then the containment venting pressure would not be reached for tens of hours following vessel breach if at all. On the other hand, if a large fraction of the core debris remains on the drywell floor long enough to interact with the concrete, then the products of the core concrete interaction (heat and non-condensible gases) could result in containment pressurization sufficient to necessitate venting or to fail the containment within several hours following vessel breach.

The staff estimates that at 25 percent power, core concrete interactions in which the full core participates could result in the containment venting pressure of 75 psia being reached within three hours following vessel failure. At 100 percent power, this pressurization time would be reduced by approximately half, due to the higher decay heat levels at 100 percent power, and correspondingly shorter times required to heat the ex-vessel debris bed to the temperatures at which unoxidized constituents in the debris (e.g., Zircaloy) would begin to react. On the basis of these conservative assumptions regarding reactor vessel failure times and containment performance, the time to release for the wetwell venting case was set to 14 hours and five hours for 25 percent and 100 percent power operation respectively in the staff's consequence calculations. A duration of release of two hours was used as it represents the time required to depressurize the containment with the available vent area.

If the operators do not vent the containment, pressurization will continue until the containment failure pressure is reached. Under the previous assumptions regarding core concrete interactions, containment pressure would increase from the venting pressure (75 psia) to the estimated ultimate pressure capacity of containment (135 psia) within about two hours, for both 25 percent and 100 percent power operation. Hence, in the staff's consequence calculations for the case with containment drywell failure, the time to release was set to 16 hours and seven hours for 25 percent and 100 percent power operation, respectively. A duration of release of two hours was used in these calculations.

The fission product release fractions used in the offsite consequence calculations are based on Source Term Code Package (STCP) calculations performed by BCL. For the wetwell venting case, Table 14, the release fractions are based on analysis of a TBUX sequence for Peach Bottom, as documented in Reference 15. This sequence involves a transient initiating event, immediately followed by reactor scram and loss of all ac and dc power. As a result, all injection to the reactor is lost, leading to eventual reactor vessel and containment failure. In the BCL analysis, the containment is assumed to fail above the water level in the wetwell, at approximately six hours. Hence, releases from the drywell pass thru and are scrubbed by the suppression pool before release to the environment. This fission product transport path is the same as would result if the wetwell were deliberately vented.

For the case with drywell failure, Table 15, the release fractions are based on analysis of a TQUV sequence for Limerick, as documented in Reference 16. This sequence involves a transient with scram, accompanied by complete failure of low pressure and high pressure coolant makeup to the reactor. In the BCL analysis, this sequence leads to containment failure in the drywell at approximately seven hours.

In both of the referenced BCL calculations the suppression pool downcomers are considered to remain intact following reactor vessel failure. In contrast, the utility 25 percent power PRA assigns a 50 percent probability to the potential for downcomer melt-through and subsequent suppression pool bypass. The staff has assessed the effect that downcomer melt-through would have on the release fractions presented in Tables 14 and 15. The approach taken was to assume that the tellurium, strontium, ruthenium, and lanthanum calculated to be retained in the suppression pool in the BCL calculations was instead distributed among the wetwell airspace, drywell, and environment in the same proportion as each fission product was calculated to be retained in these regions without downcomer failure. (Only these species were considered redistributed since they are largely released subsequent to postulated downcomer melt-through). For the wetwell venting case, downcomer melt-through results in an increase in the release fractions for these species of approximately a factor of two to three. For the drywell failure case, melt-through would result in an increase in the release fractions on the order of 50 percent. This is considered to be within the uncertainty in estimating the fission product release fractions.

Figures 3 and 4 show the five rem and 200 rem whole-body dose-versus-distance results for the core melt scenario with wetwell venting. Similar results are shown in Figures 5 and 6 for the scenario with drywell overpressure failure. Unlike the final results presented in the utility submittal (as well as the curves presented in NUREG-0396), the probabilities shown for each scenario are conditional upon that scenario occurring. In contrast, the LILCO leakage categories were weighted by the release category probability given a core melt and the results were summed over all release categories.

In interpreting the dose-versus-distance curves presented in this section, it should be recognized that while containment/reactor building decontamination factors of five or more may be expected for Shoreham, the effect of downcomer melt-through and other uncertainties in estimating fission product release fractions may offset this reduction. These uncertainties are applicable to full power operation as well. Since the mode of release is uncertain, i.e., venting versus containment overpressure, the conclusions presented below are based on the more limiting case.

Several important trends can be noted from the dose-versus-distance curves for the two scenarios. First, the offsite consequences for the drywell overpressure scenario are considerably more severe than the wetwell venting scenario, even though the time to release is later in the former case. Second, reducing the reactor power from 100 percent to 25 percent represents a significant reduction in the probability of exceeding a given dose, particularly for larger doses. Third, the assumption of a containment/reactor building DF of five also provides substantial reduction in the dose probabilities. For the drywell overpressure core-melt scenario the level of reduction is roughly comparable to that associated with restricting operation to 25 percent power; for the wetwell core-melt scenario a reduction in power by a factor of four shows a more significant impact on the dose-versus-distance probabilities than increasing the containment/reactor building DF. This is due largely to the reduced fission product inventory combined with a delayed time of release at 25 percent power.

The dose-versus-distance curves provide insights regarding the distances from the reactor over which either the Protective Action Guides (PAGs) might be exceeded or injury-threatening doses might occur in the more likely core melt sequences. However, because of the limited nature of this assessment and the large uncertainties inherent in estimation of source terms and modelling of offsite consequences, these results should be interpreted in a qualitative manner, i.e., they should not be used to estimate reduced distances over which protective measures may need to be taken in the event of an accident. Suffice it to say that the distance over which a given dose is exceeded would be significantly reduced at 25 percent power (by a factor of about three relative to full power) but that estimation of the absolute distances at which major reductions occur in the probability of dose exceedance would require a further assessment of uncertainties.

An additional staff calculation was performed to assess the sensitivity of offsite consequences to release height. The LILCO submittal and all sensitivities to date were performed with a 10 m release height. A review of the Mark-II design indicated the more probable release height would be 50 m. To determine the effect on consequences, the late drywell overpressure transient at 100 percent power and containment DF of one was performed with the release height increased to 50 m. The results showed no noticeable change in the offsite dose probabilities with the increase in release height.

To provide some perspective as to the additional time for protective actions afforded by operation at 25 percent power, dose-versus-time probability figures were also generated. Figures 7 and 8 show the probability of five rem and 200 rem whole-body doses being exceeded at two miles versus time for the wetwell venting scenario at 25 percent and 100 percent power. Results for the scenario with drywell overpressure failure are shown in Figures 9 and 10. Several important trends can be observed. First, the probability of exceeding smaller doses (i.e., five rem) two miles from the reactor approaches the 24 hour value quite rapidly following the onset of release. Although the probabilities of exceedance of the smaller doses at 25 percent power are not significantly lower than those for 100 percent power, the time required to reach a given probability of exceedance at 25 percent power is about 10 hours longer than at 100 percent power. This represents additional time available to take protective measures at 25 percent power. The amount of time corresponds approximately to the difference between the time of release at 25 percent and 100 percent power.

The dose-versus-time results for 200 rem exposures indicate that at 25 percent power the dose accumulation rates at two miles are sufficiently small that the probability of exceeding a 200 rem dose is insensitive to time of exposure, and remains small even if protective measures are not taken promptly.

4.2.2.2. Rapidly Evolving Sequences

A source term was developed by the staff to represent the type of release which might occur during a rapidly evolving severe accident in which the containment is initially intact but fails at the time of reactor vessel failure. The source term is considered to be a conservative representation of releases which would not likely be exceeded, but is not intended to represent the worst conceivable case. The staff source term is presented in Table 16, along with the most severe source term considered in the utility PRA. The WASH-1400 source term for a BWR 3 release is also included for comparison. A brief discussion of the key differences between the utility and staff source terms is provided below.

The time to release is significantly shorter in the staff source term. The value of 3.5 hours is based on the time of core slump for the large break LOCA sequence. For the 100 percent power calculations, a time to release of 0.8 hours was assumed, consistent with the time to core slump predicted for a large break LOCA at 100 percent power. The time of core slump was used to characterize the time to release for two reasons. First, under the conservative assumption that core debris does not quench in the reactor vessel bottom head, the vessel would be expected to fail at about that time, releasing core debris into the drywell and suppression pool. Containment failure coincident with vessel failure might also be conservatively postulated to occur as a result of steam explosions in the wetwell or some other mechanism. The 3.5 hour time to release for 25 percent power reflects both these conservatisms. Second, a significant amount of the noble gases and volatile fission products are released from the fuel by the time that core slump is predicted to occur.

The time to release is considered to be conservative in that two barriers to the release of fission products are postulated to fail much earlier than would be predicted by mechanistic analyses. It should be recognized, however, that if the containment is failed prior to reactor vessel failure, as it is in the seismic LOCA sequence for release Category 2, releases to the environment can occur earlier than assumed. For the large LOCA in a failed containment, releases (principally noble gases, cesium and iodine) would begin as early as about one hour at 25 percent power, and earlier at full power.

The duration of release is also significantly shorter in the staff source term. The value of one hour is based on the time to release a significant fraction of the non-volatile fission products, e.g., tellurium and strontium, from the core-concrete interactions in the drywell. This value is consistent with the results of the CORCON/VANESA calculations described in Section 4.1.2.1 for Case 4, i.e., a high core debris temperature. The value is believed to be conservative as somewhat lower initial core debris temperatures would actually be expected. Lower debris temperatures would result in a delay in the onset of vigorous core-concrete interactions and a more gradual release of non-volatiles, e.g., over a period of five to 10 hours.

The staff estimates of cesium and iodine release fractions are a factor of five higher than the utility source term. LILCO, however, assumes a secondary containment building decontamination factor (DF) of 10 for this case. If, for the reasons described in Section 4.1.2.2, less credit is taken for the secondary building, such as a DF of two, the staff and utility estimates are equivalent.

The staff estimates of release fractions for non-volatiles, particularly tellurium and strontium, are significantly higher than the utility values.

The staff values are based on the CORCON/VANESA analyses described in Section 4.1.2.1, which indicate significant potential for concrete attack. The utility values are based on analyses which indicate only minimal core-concrete interactions occur.

Based on the staff-developed source term, offsite consequence calculations were performed for operation at 25 percent and 100 percent power using the CRAC2 code. Table 17 lists the release fractions used in these calculations. Figures 11 and 12 show the five and 200 rem dose-versus-distance results for the various cases. As expected, the staff's dose-versus-distance probabilities were higher than those reported by LILCO. Also, the same general trends described in the previous section for slowly evolving transients can be observed here, specifically, that reducing the reactor power from 100 percent to 25 percent represents a significant reduction in the probability of exceeding a given dose, or conversely, a significant reduction in the distance over which a given dose would be exceeded.

To provide some perspective as to the additional time for protective actions afforded by operation at 25 percent power, a set of dose-versus-time release conditional probability figures were generated following the procedure described in Section 4.2.2. Figures 13 and 14 show the probability of five and 200 rem whole-body doses being exceeded at two miles versus time for 25 percent and 100 percent power. The probability of exceeding the five rem dose two miles from the reactor approaches the 24-hour value quite rapidly for both 25 percent and 100 percent power, and the difference in the time required to reach a given probability of exceedance is comparable to the differences in the time to release for the 25 percent and 100 percent power cases. For the 200 rem doses, the results for full power indicate that following the time of release (0.8 hours) the probability of exceedance at two miles rapidly approaches its 24-hour value. For 25 percent power, CRAC2 indicates a much lower dose accumulation rate; specifically, 200 rem doses are not exceeded until about three hours after the time of release (3.5 hours) or six hours after transient initiation. Since there is a significantly shorter time to release for 100 percent power and a high probability that a 200 rem whole-body dose will be exceeded very shortly after the release, less dose savings could be realized for 100 percent power operation.

5 SUMMARY AND CONCLUSIONS

The staff has completed an expedited review of the PRA-based portion of the LILCO request. This review was oriented towards assessing the validity of the major technical arguments upon which the utility submittal is based. These arguments can be summarized as follows:

1. Reduced vulnerability to Core Damage Accidents

   With operation at 25 percent power, decay heat levels are reduced to the extent that (1) certain plant features, such as turbine bypass flow, are capable of mitigating accidents prior to core melt and (2) accidents will evolve more slowly allowing considerably greater time for recovery actions. These factors, in conjunction with a number of plant upgrades which have or will be implemented, will result in a reduced vulnerability to severe core melt accidents at Shoreham.

2. Increased Time for Emergency Response

   For accidents which are not arrested prior to severe core melt, reduced decay heat levels derived from operating at 25 percent power will result in a significant delay in both core melt progression and onset of releases from containment. This delay represents an increase in the time available for emergency response.

3. Reduced Offsite Consequences

   The magnitude of source term releases for accidents initiated from 25 percent power are less than predicted for similar accidents initiated at 100 percent power due to a proportionally smaller initial fission product inventory at the lower power level. The reduced source terms, in conjunction with the delayed times of release mentioned above, translate into reduced offsite consequences.

On the basis of the staff's review of the utility submittal and supporting documentation we have reached the following conclusions:

1. The 25 percent power restriction, in conjunction with the improvements in the plant design and operating procedures, effectively reduces the significance of several specific plant vulnerabilities to core melt. However, the overall core melt frequency is not significantly reduced because of the numerous sequences that are unaffected. Moreover, the seismic-induced contribution to core melt frequency has large uncertainties, and can contribute about one fifth of the internally initiated core melt frequency estimate for both full power and restricted power operation. Such consideration will make the difference between the estimates of core melt frequencies at 25 percent and full power even less significant.

2. The utility claim that operation at 25 percent power results in a significant increase in the time available for accident mitigation and emergency response is valid. Calculations performed by the staff for selected risk-important sequences confirm the estimates of timing provided by the utility for key events. These calculations indicate that the timing of key events in the core melt progression (e.g., start of core melt, core slump) are significantly delayed at 25 percent power. This delay is on the order of a factor of four. For the most rapidly evolving sequences, significant core damage will not occur until after one hour for operation at 25 percent power versus 10 minutes for operation at 100 percent power. For the most likely sequences, the time of significant core damage will be delayed from about two to three hours for 100 percent power to 10 or more hours at 25 percent power.

   Furthermore, the time of release to the environment is significantly delayed at 25 percent power. Under conservative assumptions regarding reactor vessel failure times and containment performance, the bulk of the releases at full power (approximately 80 percent) occur between two and six hours following accident initiation. For the same assumptions at 25 percent power, the majority of releases (approximately 80 percent) are delayed until 12 hours or more.

   Finally, as discussed below, reductions in dose accumulation rates at 25 percent power afford additional time to take protective measures.

3. The utility claim that offsite consequences are reduced by operation at 25 percent power is valid. The staff has confirmed that the power reduction translates approximately into a factor of four reduction in initial fission product inventory, and that the time to release will be significantly delayed at the lower power level, again by approximately a factor of four. These two direct benefits of the power restriction, in conjunction, translate into significant dose savings for all sequences.

Recognizing that an assessment of the remaining uncertainties in source terms as well as relative frequencies for the various release categories was not practicable, the effect of the power restriction on offsite consequences was determined by considering the offsite consequences for two different accident sequences selected to characterize the range of core melt progression timing which could be expected at Shoreham. This involved the specification of source terms for 25 percent and 100 percent power (i.e., fission product inventory and release fractions in conjunction with release time and duration) and a comparison of offsite consequences for each case.

On the basis of staff calculations, restricting operation to 25 percent of rated power reduces the distances over which injury-threatening doses (i.e., 200 rem) would occur. CRAC2 calculations indicate that distances are reduced by approximately a factor of three relative to full-power operation, however, the absolute distances at which major reductions occur in the probability of exceeding a particular dose are dependent on modelling and input assumptions and are an area of remaining uncertainty. The probability of exceeding a five rem whole-body is also reduced by operating at 25 percent power, but significant reductions do not generally occur within the 10 mile EPZ.

CRAC2 calculations indicate that dose accumulation rates alone may yield significant additional time to avoid injury threatening doses at 25 percent power (in addition to the delay in time of release afforded by the power restriction). Dose-versus-time calculations performed for a rapidly evolving sequence using CRAC2 show that at 25 percent power a 200-rem whole-body dose could be averted at a two mile radius by evacuating within three hours following start of the release (or within six hours after accident initiation).

6 REFERENCES

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Table 1 Reported core melt frequency results

Initiator	Full Power		25% of Full Power	
	Frequency	Percentage	Frequency	Percentage
Internal Events	5.5×10^{-5}	85	2.5×10^{-5}	89
External Events				
Fire	7.3×10^{-6}	11	4.6×10^{-7}	1.6
Seismic	2.5×10^{-6}	4	2.7×10^{-6}	9.6
Total	6.5×10^{-5}		2.8×10^{-5}	

Table 2 Release characteristics for Shoreham release categories (25% power)[1]

Release Categories	Qualitative attributes	Representative sequence	Release sequence characteristics
RC1	No pool scrubbing Large leakage size with driving force Low reactor building retention Short duration, early release	ATWS Class IV plant damage state with overpressure failure in the drywell or wetwell with downcomer failure, bypassing the pool with minimum reactor building retention. Suppression pool is saturated providing sustained gas flow rates.	Early, short duration and high energy release. Noble gases and a few percent of particulates are released.
RC2	No pool scrubbing Large leakage size but without driving force Low reactor building retention Moderate duration, early release	Seismic RPV breach Class IIID plant damage state with drywell failure bypassing the pool. Other sequences include interfacing LOCAs Class V Plant Damage State, ATWS Class IV with small containment leakage failures bypassing the pool (e.g. drywell with downcomers failure)	Early, moderate duration and low energy release. Noble gases and tenths of a percent of particulates are released.
RC3	Pool scrubbing Large leakage size Low reactor building retention Short duration, early release	ATWS Class IV plant damage state with failure in the wetwell and downcomer vents intact. The release pathway involves pool scrubbing.	Early, short duration and high energy release. Noble gases and a few hundredths of a percent of particulates are released.
RC4	No pool scrubbing Small leakage size or Large leakage size without driving force Reactor building retention Long duration with containment attenuation, early release	Station Blackout plant damage state Class IB. Slow developing accident where the releases bypass the suppression pool, but reactor building hold up is significant.	Relatively early, long duration release. Noble gases are slowly released, and less than 10^{-3} particulate fractions are released.

Table 2 Release characteristics for Shoreham release categories (25% power) (Continued)[1]

Release Categories	Qualitative attributes	Representative sequence	Release sequence characteristics
RC5	Late release with and without pool scrubbing	Loss of coolant makeup Class IA plant damage state. Late containment failure due to operator venting after 48 hours. Fission product releases are therefore significantly reduced.	Very slow developing with long times to release. Noble gases and less than 10^{-5} particulate fractions are released.
RC6	Design leakage (contained release) Recovered core melt states	Loss of coolant makeup Class IA plant damage state. The containment is not breached or the core melt sequence is recovered.	Contained released where design leakage determines fission products released to the environment.

[1]Taken from reference 1. Release characteristics presented are those reported by the utility.

Table 3 Release categories and their contribution to core melt and early releases[1]

Release category	Risk dominant contributors	% contribution to core melt[2]	Timing for representative sequence (hours after scram) Core slump[3]	Release to environment[4]
1	ATWS with pool bypass	2.3	10.4	10
2	Seismic LOCA (Failed Containment)	2.0	4.6	5
3	ATWS with no pool bypass	8.2	6.8	7
4	Station blackout	13.9	13.9	15
5	Large LOCA (Intact Containment)	47.9	4.3	48
6	Transient with Loss of Injection	25.7	11.3	15

[1]Values presented are those reported by the utility.
[2]Total core melt frequency is 2.8 E-5/Reactor-Year.
[3]In the analyses performed using MAAP, vessel failure occurs within minutes following core slump.
[4]Values presented are those used in the utility offsite consequence calculations.

Table 4 Comparison of utility and staff estimates of core melt progression for 25% power

Time of Event (Hours after scram)								
ATWS[1]		Large Break LOCA[2]		Station Blackout[3]		Loss of Injection[4]		Event
Utility	Staff	Utility	Staff	Utility	Staff	Utility	Staff	
1.7	1.7	.007	.001	4.1	1.5	2.7	2.3	Uncover top of active fuel
4.1	4.9	.6	1.0	7.5	7.2	5.8	6.6	Begin cladding relocation
10.4	9.4	4.6	3.3	13.9	12.3	11.3	10.7	Slump first radial zone of core
10.4	30.8	4.6	7.8	13.9	49.2 [12.4]	11.3	24.6	Fail bottom head[5]

[1]Sequence as defined for Release Category 1.

[2]Sequence as defined for Release Category 5, except flow from CRD hydraulic system not modelled.

[3]Sequence as defined for Release Category 4.

[4]Sequence as defined for Release Category 6.

[5]Utility analyses assume debris does not quench in bottom head; staff analyses assume debris quenchs and reheats prior to failing bottom head. Number in brackets is MARCH 3 result obtained assuming no debris quench.

Table 5 Effect of power restriction on core melt progression for less probable sequences

Time of Event (Hours After Initiation)				
ATWS		Large break LOCA[1]		
25%[1]	100%[2]	25%	100%	Event
1.7	.7	.001	.001	Uncover top of active fuel
4.9	1.1	1.0	.2	Begin cladding relocation
9.4	1.7	3.3	.7	Slump first radial zone of core
22.3	1.9	3.7	1.0	Dry out bottom head
28.7	1.8	5.8	1.0	Slump remainder of core
30.8	2.4	7.8	1.2	Fail bottom head

[1]Based on ORNL calculations for Shoreham using the BWRSAR code.
[2]Based on BCL calculations for Limerick using the MARCH 2 code.

Table 6 Effect of power restriction on core melt progression for more probable sequences

Time of Event (Hours After Scram)				
Station Blackout with SORV		Loss of injection[3]		
25%[1]	100%[2]	25%	100%	Event
1.5	1.0	2.3	.4	Uncover top of active fuel
7.2	2.2	6.6	1.1	Begin cladding relocation
12.3	2.7	8.1	1.2	Uncover core plate
12.3	2.7	10.7	1.8	Slump first radial zone of core
27.6	3.0	19.7	3.9	Dry out bottom head
49.2	4.0	24.6	4.5	Fail bottom head

[1]Based on BCL calculations for Shoreham using the MARCH 3 code.
[2]Based on BCL calculations for Limerick using the MARCH 2 code.
[3]Based on ORNL claculations for Shoreham using the BWRSAR code.

Table 7 Summary of the core-vulnerable accident plant damage states at 25% power

Plant Damage States	Definition	Example	Frequency per reactor year
CLASS IA	Accident sequences involving loss of inventory makeup where the reactor pressure remains high.	TQUX	1.5E-5
B	Accident sequences involving a loss of offsite power and loss of coolant inventory makeup.	T_EQUV	2.3E-6
C	Accident sequences involving a loss of coolant inventory induced by an ATWS situation.	$T_mC_mC_2$U'U'	6.6E-10
D	Accident sequences involving a loss of coolant inventory makeup where reactor pressure has been reduced to 200 psi.	TQUV	4.6E-6
CLASS II	Transient accident sequences involving a loss of containment heat removal.	TW	1.5E-9
CLASS IIIA	Accident sequences leading to core vulnerable conditions initiated by vessel rupture. (Containment integrity is not breached by the initiating event.)	R	
B	Accident sequences initiated by or resulting in small LOCAs for which the reactor cannot be depressurized.	S_1QUX	2.4E-8
C	Accident sequences initiated by or resulting in medium or large LOCAs for which the reactor is at low pressure.	AQUV	7.0E-7
D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity.	AD	1.1E-7
CLASS IV	Accident sequences involving failure to insert negative reactivity leading to a containment vulnerable condition due to high containment pressure.	$T_mC_mC_2$	3.9E-6
CLASS V	LOCAs outside containment	Interfacing LOCA	1.2E-6
SRPV*	Seismically-induced reactor pressure failure and subsequent containment failure.	Seismic AD	8.0E-7

*SRPV represents a seismically-induced reactor pressure vessel breach with subsequent loss of containment integrity. This sequence was combined with plant damage state Class IIID since the core melt progression is similar to the internally-initiated large LOCA sequences with an initially failed containment prior to core melt.

Table 8 Shoreham Nuclear Power Station -- 25% power Plant damage state release category distribution (percent of core melt)[1]

Release Category	Plant Damage State									
	IA	IB	IC	ID	II	IIIB	IIIC	IIID	IV	V
RC1	1.9E-02 (7.0)	1.9E-02 (7.0)	5.0E-06 (7.0)	2.4E-05 (7.0)	1.8E-07 (7.0)	3.1E-05 (7.0)	3.7E-06 (7.0)	8.1E-03 (0.5)	2.3E+00 (7.0)	4.3E-04 (1.0)
RC2	5.3E-02 (7.0)	8.2E-01 (7.0)	1.4E-05 (7.0)	7.3E-03 (7.0)	1.0E-07 (7.0)	8.4E-05 (7.0)	1.1E-03 (7.0)	3.7E-02 (0.5)	1.0E+00 (7.0)	3.4E-02 (1.0)
RC3	2.8E-01 (7.0)	3.8E-02 (7.0)	7.2E-05 (7.0)	3.5E-04 (7.0)	8.2E-07 (7.0)	4.4E-04 (7.0)	5.3E-05 (7.0)	2.8E-02 (0.5)	7.9E+00 (7.0)	
RC4	2.4E-01 (11.0)	7.3E+00 (14.0)	6.3E-05 (11.0)	6.6E-02 (11.0)	4.7E-07 (11.0)	3.9E-04 (11.0)	1.0E-02 (11.0)	3.2E+00 (1.0)	3.0E+00 (11.0)	9.2E-03 (1.0)
RC5	3.1E+01 (48.0)		2.0E-04 (48.0)	1.5E+01 (48.0)		7.8E-03 (48.0)	2.3E+00 (48.0)			
RC6	2.4E+01 (60.0)		2.0E-03 (60.0)	1.8E+00 (60.0)		7.9E-02 (60.0)	2.8E-01 (60.0)			
TOTAL	5.5E+01	8.2E+00	2.4E-03	1.7E+01	1.6E-06	8.7E-02	2.5E+00	3.3E+00	1.4E+01	4.4E-02

NOTES: The bracketed numbers below each value of percent of core melt represent the time (hrs) from the initiating event to the release of radiation to the environment for the representative severe accident sequence of that group.

The summation of the percent contributions of each group total slightly higher than 100% because of round-off.

[1]Taken from Reference 10. Values presented are those reported by the utility.

Table 9 Time of release to environment for Shoreham accident classes

Plant Damage States	Definition	Fraction of total core melt frequency		Time of release to environment (h)	
		25% power[1]	100% power[2]	25% power	100% power
CLASS I	Transients with SCRAM, loss of coolant makeup, core vulnerability prior to containment challenge	.80	.52	14.	5.
CLASS II	Transients with SCRAM, inadequate containment heat removal, containment vulnerability before core melt	<.001	.24	>24.	>24.
CLASS III	LOCAs with inadequate core cooling, core vulnerability prior to containment challenge	.03	.02	6.	2.2
CLASS IV	Transients with failure to SCRAM, inadequate containment heat removal, containment vulnerability before core melt	.14	.21	7.	2.5
CLASS V	LOCAs with containment bypass prior to core melt	<.001	<.001	1.	.2
SRPV	Seismically-induced reactor pressure vessel failure with subsequent containment failure	.03	.01	1.	.2

[1]Total core melt frequency for 25% power operation is 2.8E-5/Reactor-Year.
[2]Total core melt frequency for 100% power operation is 6.5E-5/Reactor-Year.

Table 10 Comparison of utility and staff estimates of time of release for a spectrum of accidents

Time of Release - t (h)	Utility 25% Power	Staff 25% Power	Staff 100% Power
$0 \leq t < 2$	.03	.03	.01
$2 \leq t < 6$	0.	0.	.75
$6 \leq t < 12$	.16	.17	0.
$12 \leq t$	.81	.80	.24

Table 11 Radioisotope inventories for 2 and 6 years of operation at 25% power (10^6 curies)

	2 years[1]	6 years[2]
KR-85	.1473	.3380
KR-85M	5.255	4.264
KR-87	10.31	8.122
KR-88	14.57	11.45
RB-86	.0063	.01609
SR-89	19.37	15.13
SR-90	1.161	2.853
SR-91	23.99	19.37
Y-90	1.173	2.883
Y-91	24.26	19.57
ZR-95	29.93	27.36
ZR-97	28.84	27.57
NB-95	30.01	27.43
MO-99	30.56	30.30
TC-99M	26.75	26.53
RU-103	21.92	26.04
RU-105	12.59	17.71
RU-106	5.069	11.38
RH-105	12.38	17.41
TE-127	1.520	1.828
TE-127M	.1966	.2492
TE-129	4.784	5.385
TE-129M	.7094	.8072
TE-131M	2.259	2.459
TE-132	23.15	23.55
SB-127	1.538	1.839
SB-129	4.867	5.473
I-131	15.13	16.68
I-132	23.44	23.94
I-133	34.23	33.67

Table 11 Radioisotope inventories (Continued)

	2 years[1]	6 years[2]
I-134	37.78	36.73
I 135	31.61	31.38
XE-133	34.28	33.78
XE-135	19.74	19.93
CS-134	.4379	2.437
CS-136	.3755	.8608
CS-137	1.403	4.014
BH-140	30.11	28.77
LH-140	30.30	29.22
CE-141	28.67	27.43
CE-143	27.41	25.23
CE-144	21.03	21.83
PR-143	27.37	25.20
ND-147	11.29	10.90
NP-239	399.6	396.0
PU-238	.003064	.05208
PU-239	.01631	.02729
PU-240	.00742	.02835
PU-241	.9045	6.406
AM-241	.0007529	.01737
CM-242	.03965	1.616
CM-244	-	.01099

[1]Based on 2 years of operation at 25% power.

[2]Based on 6 years of operation at 25% power without refueling.

Table 12 Comparison of radioisotope inventories (10^6 curies)

	Shoreham 25% power[1]	BCL ORIGEN2 25% power[2]	BCL ORIGEN2 Full power[3]
CO-58	.1484		
CO-60	.0552		
KR-85	.1066	.1473	.5232
KR-85M	4.565	5.255	20.06
KR-87	8.942	10.31	38.78
KR-88	12.94	14.57	54.69
RB-86	.0049	.0063	.07791
SR-89	17.88	19.37	72.93
SR-90	.7040	1.161	4.115
SR-91	20.93	23.99	91.40
Y-90	.7420	1.173	4.261
Y-91	22.83	24.26	91.63
ZR-95	28.52	29.93	118.8
ZR-97	2[illegible].52	28.84	121.8
NB-95	28.52	30.01	113.8
MO-99	30.45	30.56	132.0
TC-99M	26.62	26.75	115.6
RU-103	[illegible]0.93	21.92	103.3
RU-105	13.70	12.59	67.75
RU-106	4.755	5.069	25.33
RH-105	9.322	12.38	63.50
TE-127	1.122	1.520	7.012
TE-127M	.2093	.1966	.8283
TE-129	5.898	4.784	21.83
TE-129M	1.008	.7094	3.237
TE-131M	2.473	2.259	10.18
TE-132	22.83	23.15	100.8
SB-127	1.160	1.538	7.196
SB-129	6.278	4.867	22.20
I-131	16.17	16.13	70.51

Table 12 Comparison of radioisotope inventories (Continued)

	Shoreham 25% power[1]	BCL ORIGEN2 25% power[2]	BCL ORIGEN2 Full power[3]
I-132	22.83	23.44	102.4
I-133	32.35	34.23	146.2
I-134	36.15	37.78	160.7
I-135	28.52	31.61	136.4
XE-133	32.35	34.28	143.9
XE-135	6.468	19.74	39.76
CS-134	1.427	.4379	5.481
CS-136	.5708	.3755	2.413
CS-137	.8943	1.403	5.531
BA-140	30.45	30.11	127.2
LA-140	30.45	30.30	131.2
CE-141	28.52	28.67	120.9
CE-143	24.73	27.41	112.8
CE-144	16.17	21.03	70.05
PR-143	24.73	27.37	110.2
ND-147	23.96	11.29	47.[illegible]
NP-239	312.0	399.6	1,471.
PU-238	.001084	.003064	.0717
PU-239	.003995	.01631	.02556
PU-240	.003995	.00742	.02970
PU-241	.6468	.9045	6.534
AM-241	.000324	.0007529	
CM-242	.09512	.03965	
CM-244	.004375	-	

[1]LILCo May 8, 1987 Letter, Table 4C-1 Values Divided by four.

[2]Based on 2 year operation at 25% power.

[3]End of equilibrium cycle with peak burnup of 27,000 MWD/MT.

Table 13 Ex-vessel fission product releases (expressed as fractions of that available at start of concrete attack)

Species	Shoreham 25% Power Case 1 Full core	Case 2 Oxides only	Case 3 50% oxides	Case 4 25% core, 4130° F	Limerick TQUV	TC4
Iodine	1.0	1.0	.98	.91	1.0	1.0
Cesium	1.0	.42	.45	.79	1.0	1.0
Tellurium	.33	1.0	.90	.15	.35	.35
Strontium	.63	.04	.01	.15	.49	.48
Ruthenium	2E-7	5E-6	3E-8	1E-7	1E-6	1E-6
Lanthanum	.01	6E-4	4E-5	.002	[illegible]	.044
Cerium	.036	.001	1E-5	.007		
Barium	.43	.023	.01	.079		

Table 14 Approximate source terms for a BWR transient with wetwell venting

Parameter	DF = 1* 25% Power	DF = 1* 100% Power	DF = 5 25% Power	DF = 5 100% Power
Time to release (h)	14.	5.	14.	5.
Duration of release (h)	2.	2.	2.	2.
Release fractions				
Noble Gases	1.	1.	1.	1.
Cesium	.005	.005	.001	.001
Iodine	.005	.005	.001	.001
Tellurium	.02	.02	.004	.004
Strontium	.006	.006	.001	.001
Ruthenium	5E-7	5E-7	1E-7	1E-7
Lanthanum	5E-4	5E-4	1E-4	1E-4

*Release fractions based on STCP analysis of Peach Bottom TBUX sequence (NUREG/CR-5062). Cesium and Iodine release fractions increased to .005 to reflect uncertainties.

Table 15 Approximate source terms for a BWR transient with late overpressure in drywell

Parameter	DF = 1* 25% Power	DF = 1* 100% Power	DF = 5 25% Power	DF = 5 100% Power
Time to release (h)	16.	7.	16.	7.
Duration of release (h)	2.	2.	2.	2.
Release fractions				
Noble Gases	1.	1.	1.	1.
Cesium	.005	.005	.001	.001
Iodine	.005	.005	.001	.001
Tellurium	.02	.02	.004	.004
Strontium	.05	.05	.01	.01
Ruthenium	6E-8	6E-8	1E-8	1E-8
Lanthanum	.004	.004	8E-4	8E-4

*Release fractions based on STCP analysis of Limerick TQUV sequence (BMI-2104, Vol. 8). Cesium and Iodine release fractions increased to .005 to reflect uncertainties.

Table 16 Comparison of utility and staff source term for early release at 25% power

	Utility[1]	Staff[2]	Wash-1400 (BWR 3)
Time to release (h)	10	3.5	
Duration of release (h)	5	1.	
Release fractions			
Noble Gases	1.	1.	1.
Cesium	.016	.1	.1
Iodine	.02	.1	.1
Tellurium	1E-5	.1	.3
Strontium	3E-4	.1	.01
Ruthenium	8E-5	0.	.02
Lanthanum	0.	.003	.004

[1]Values shown are for Release Category 1 - ATWS with suppression pool bypass and wetwell venting.

[2]Includes the following conservatisms:
- Release initiated at core slump rather than vessel failure
- Full core assumed to participate in concrete attack
- Minimal fission product retention in containment and reactor building

Table 17 Approximate source terms for a BWR sequence with early release

Parameter	DF = 1.0 25% Power	DF = 1.0 100% Power	DF = 5.0 25% Power	DF = 5.0 100% Power
Time to release (h)	3.5	0.8	3.5	0.8
Duration of release (h)	1.	1.	1.	1.
Release fractions				
Noble Gases	1.	1.	1.	1.
Cesium	0.1	0.1	0.02	0.02
Iodine	0.1	0.1	0.02	0.02
Tellurium	0.1	0.1	0.02	0.02
Strontium	0.1	0.1	0.02	0.02
Ruthenium	0.0	0.0	0.0	0.0
Lanthanum	0.003	0.03	0.0006	0.006

[1]All other parameters identical to the PLG-0542 CRAC2 calculations.

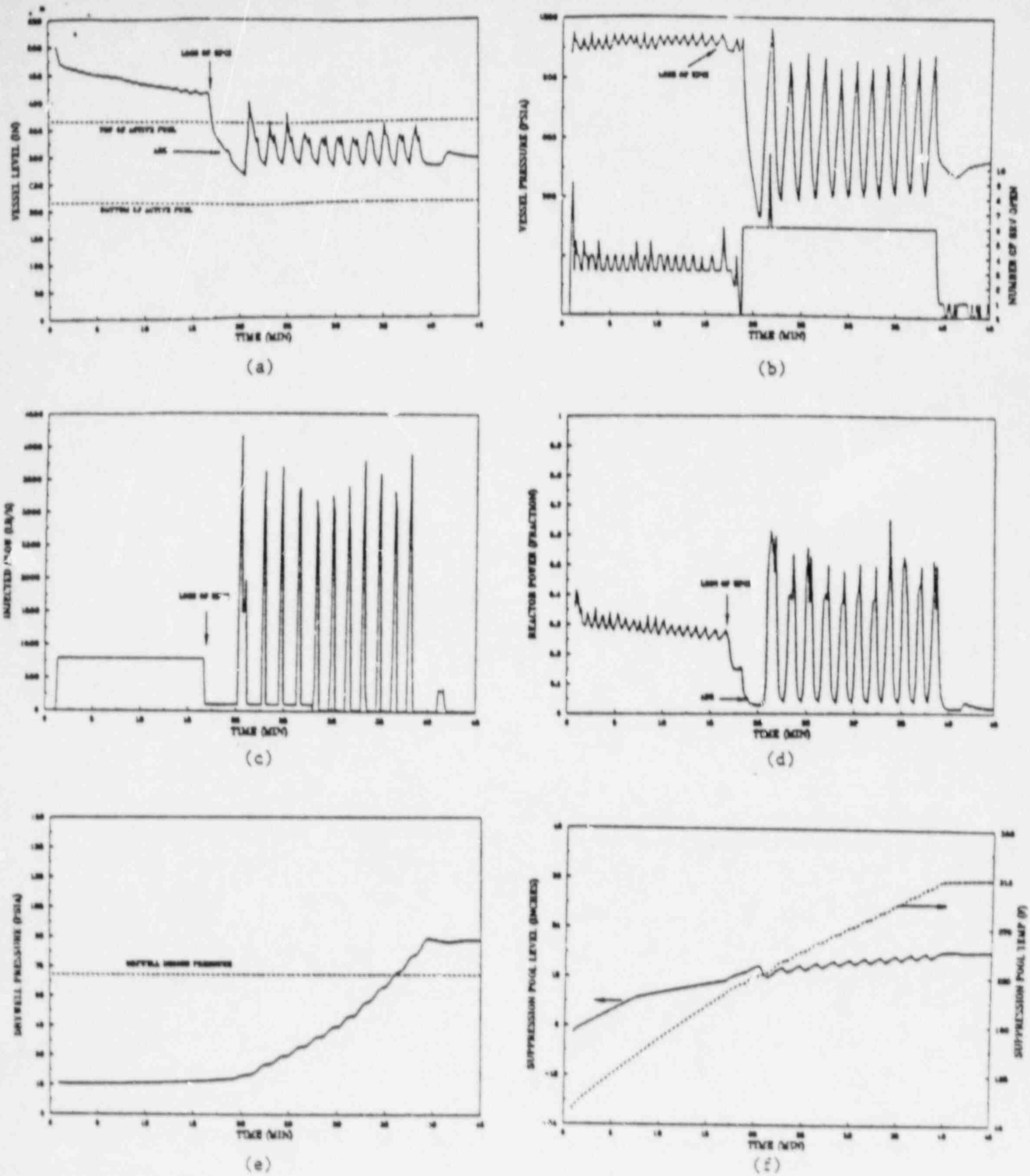
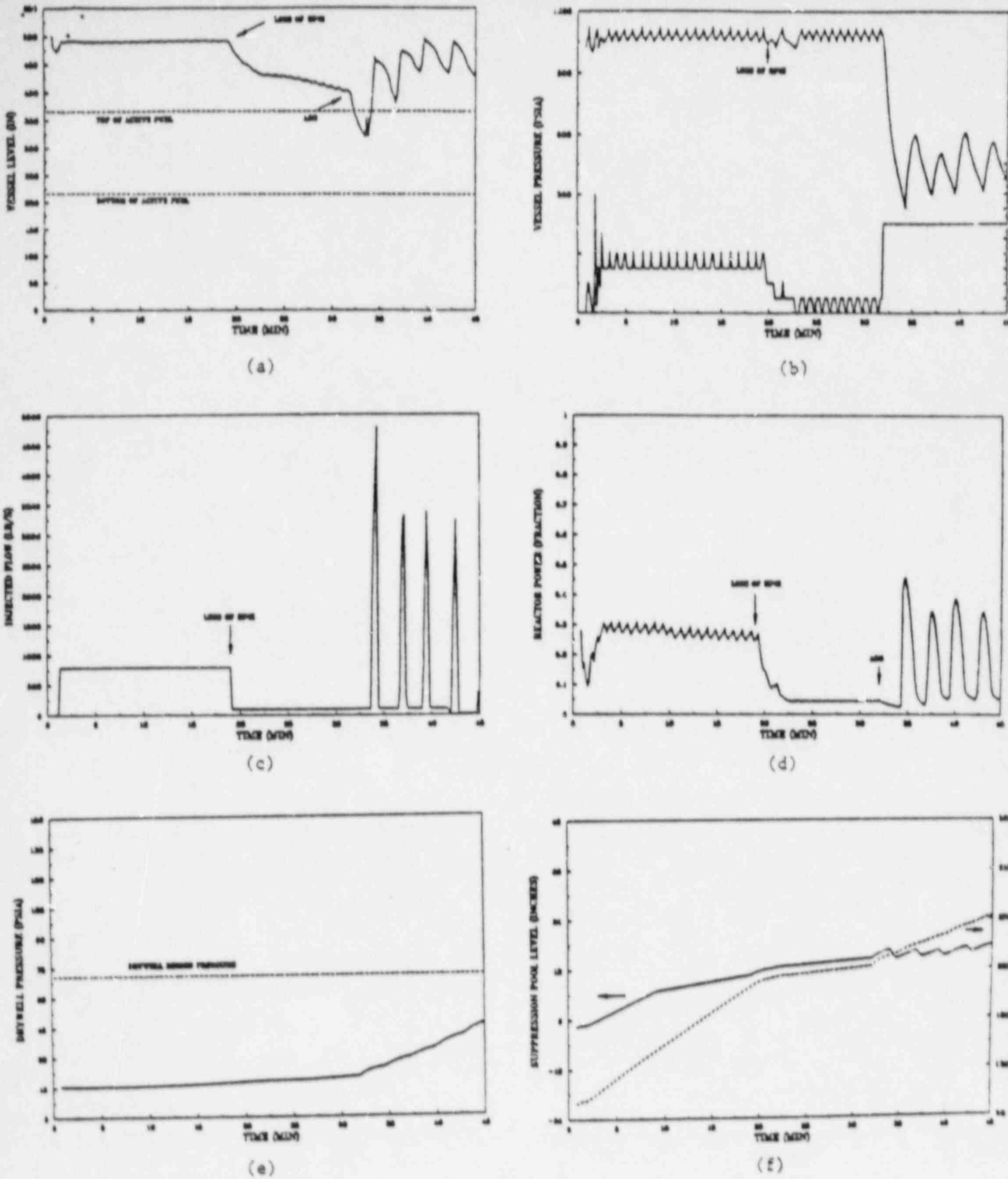


Figure 1 Accident signatures for the MSIV-closure ATWS from 100% power operation, first 45 minutes with no operator action



(a) (b) (c) (d) (e) (f)

Figure 2 Accident signatures for the MSIV-closure ATWS from 25% power operation, first 45 minutes without operator action

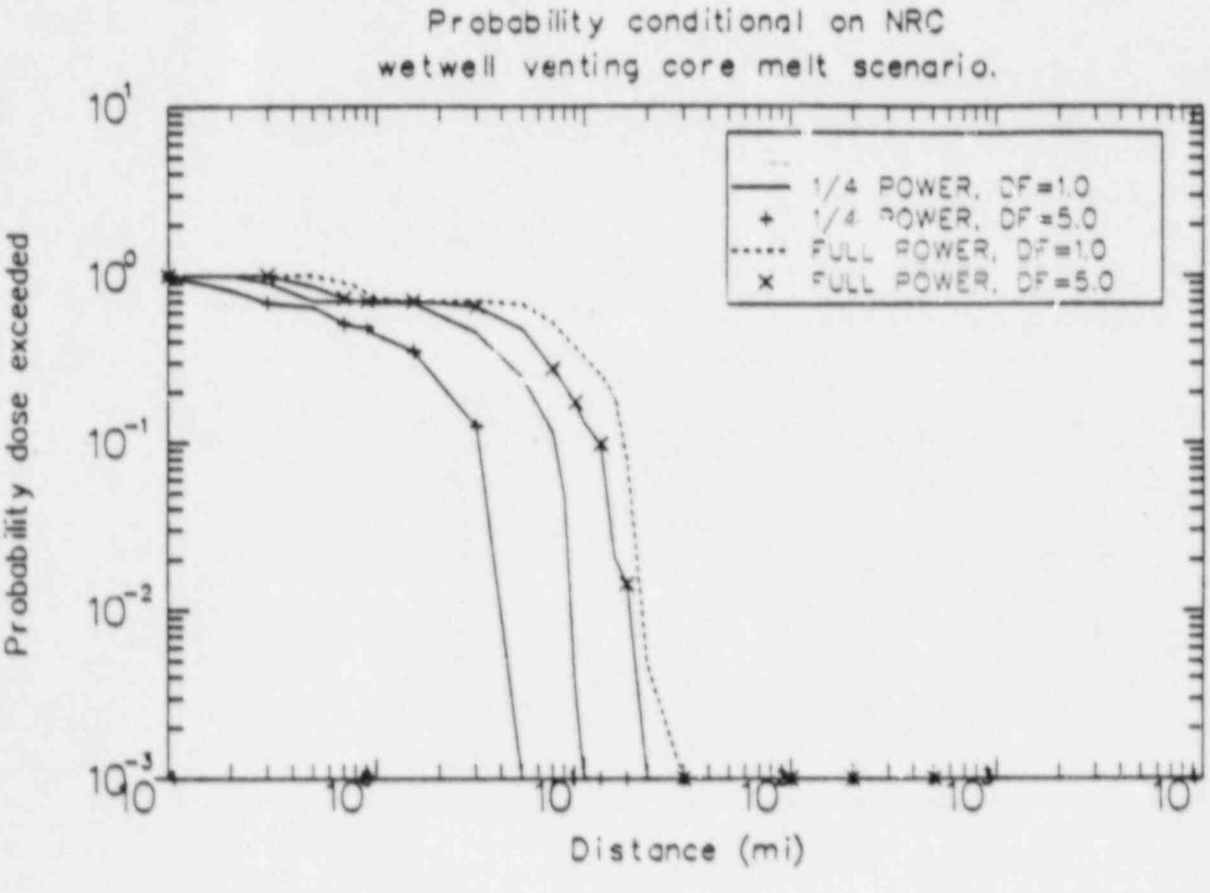


Figure 3 CRAC2 calculated dose-versus-distance probability distributions for whole body dose of 5 rem -- wetwell venting scenario

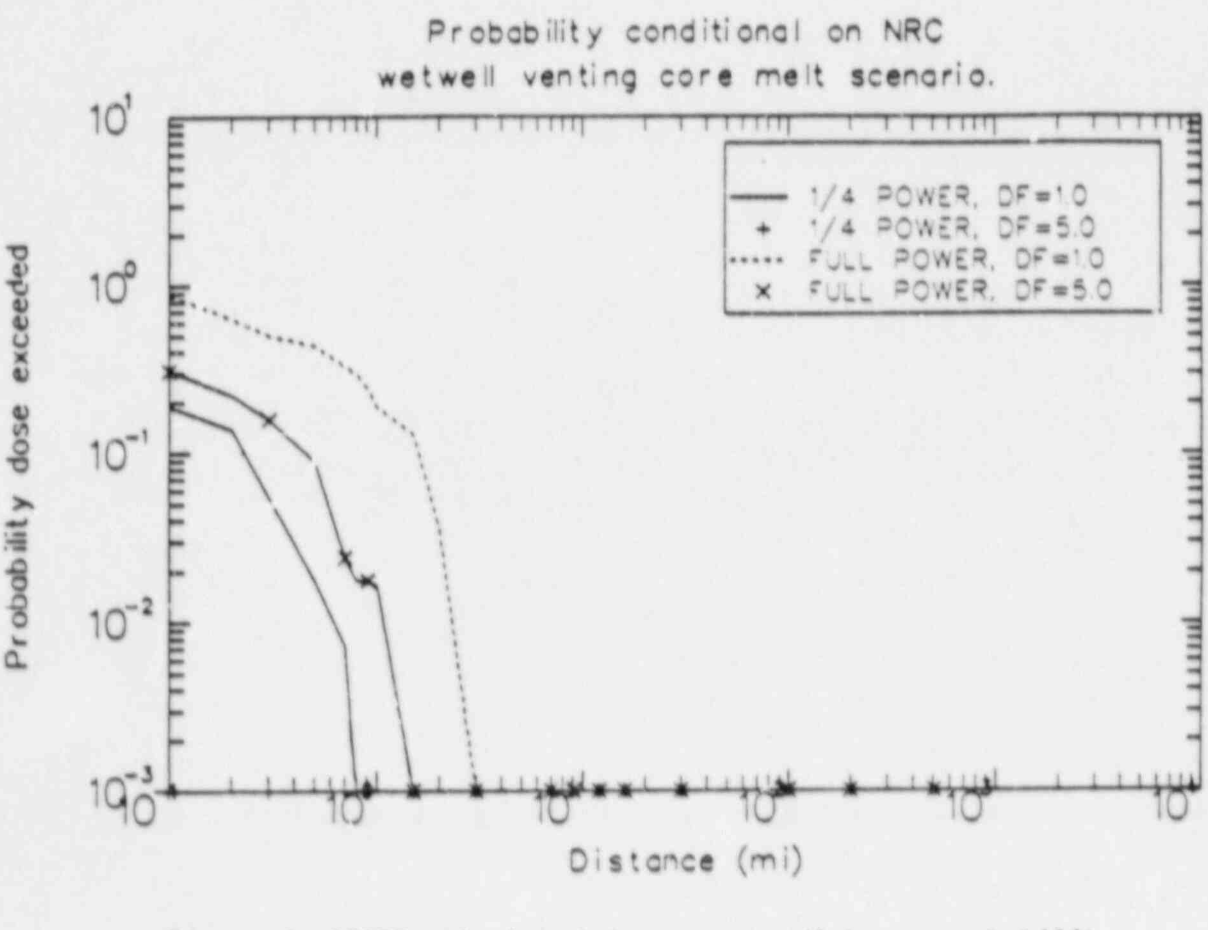


Figure 4 CRAC2 calculated dose-versus-distance probability distributions for whole body dose of 200 rem -- wetwell venting scenario

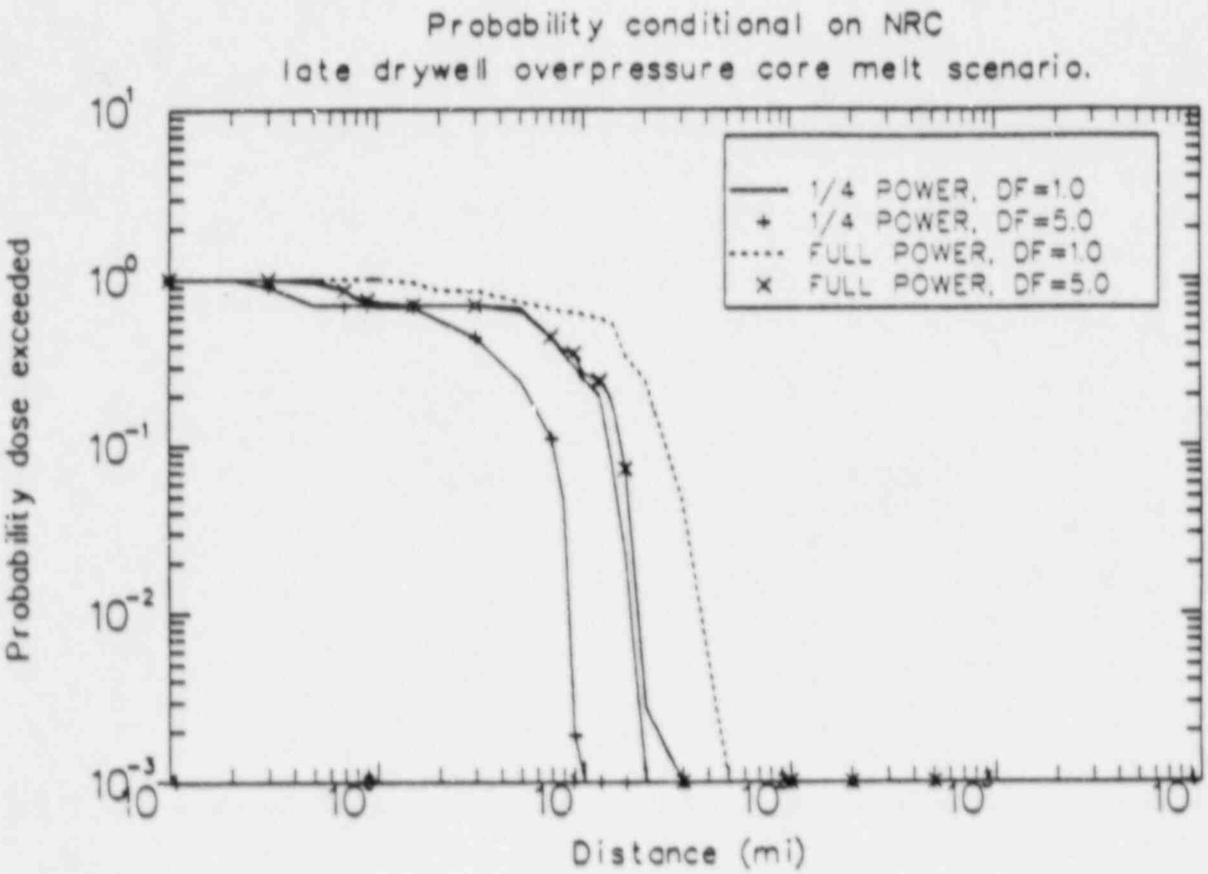


Figure 5 CRAC2 calculated dose-versus-distance probability distributions for whole body dose of 5 rem -- drywell overpressure scenario

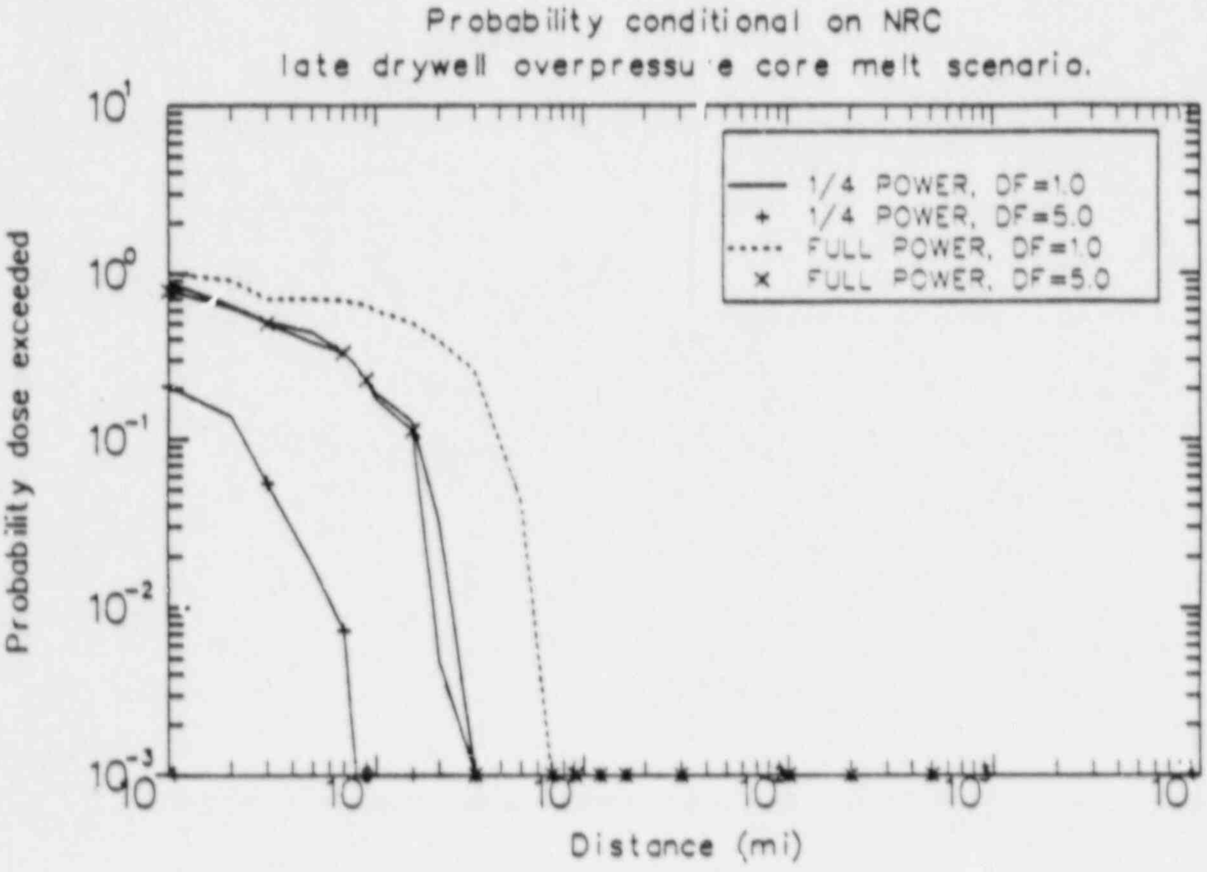


Figure 6 CRAC2 calculated dose-versus-distance probability distributions for whole body dose of 200 rem -- drywell overpressure scenario

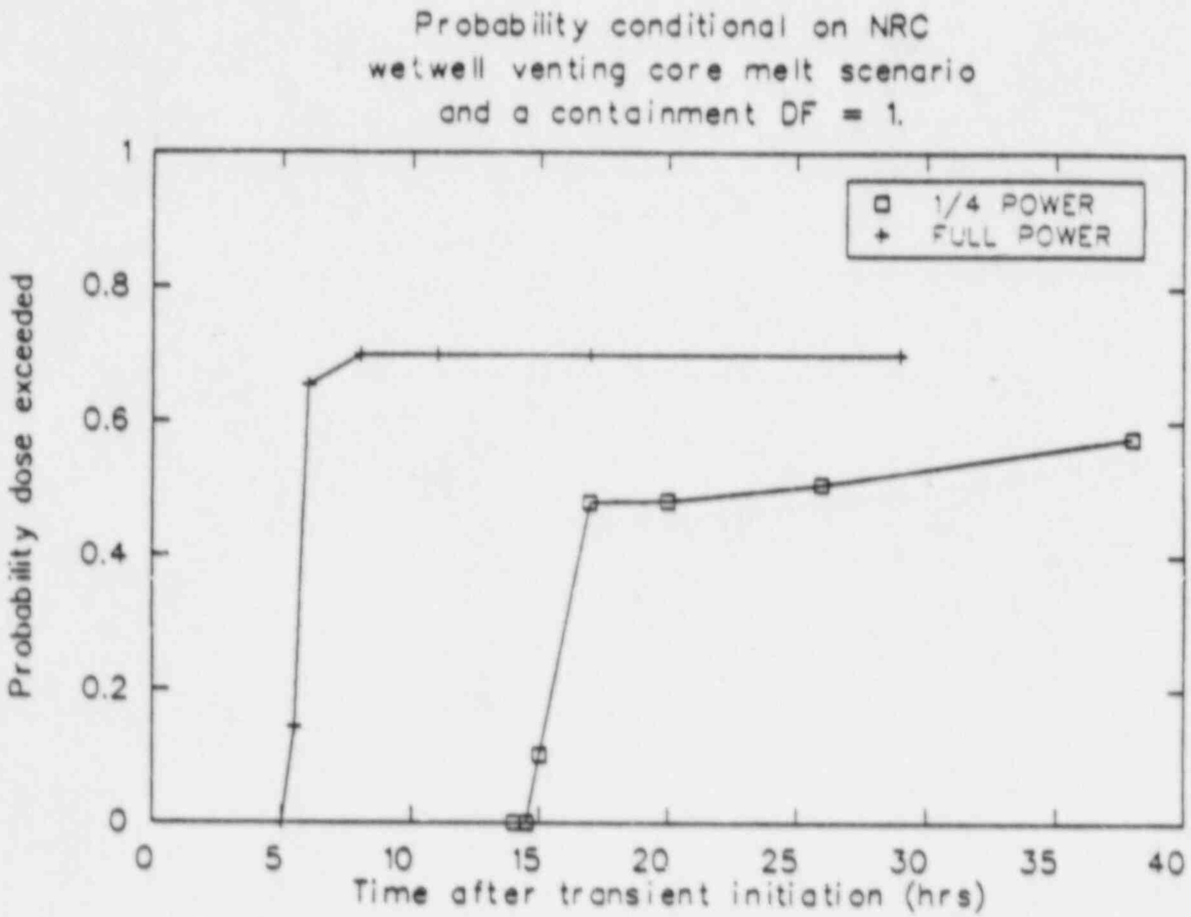


Figure 7 CRAC2 calculated probability of 5 rem whole body dose being exceeded at 2 miles from the plant -- wetwell venting scenario

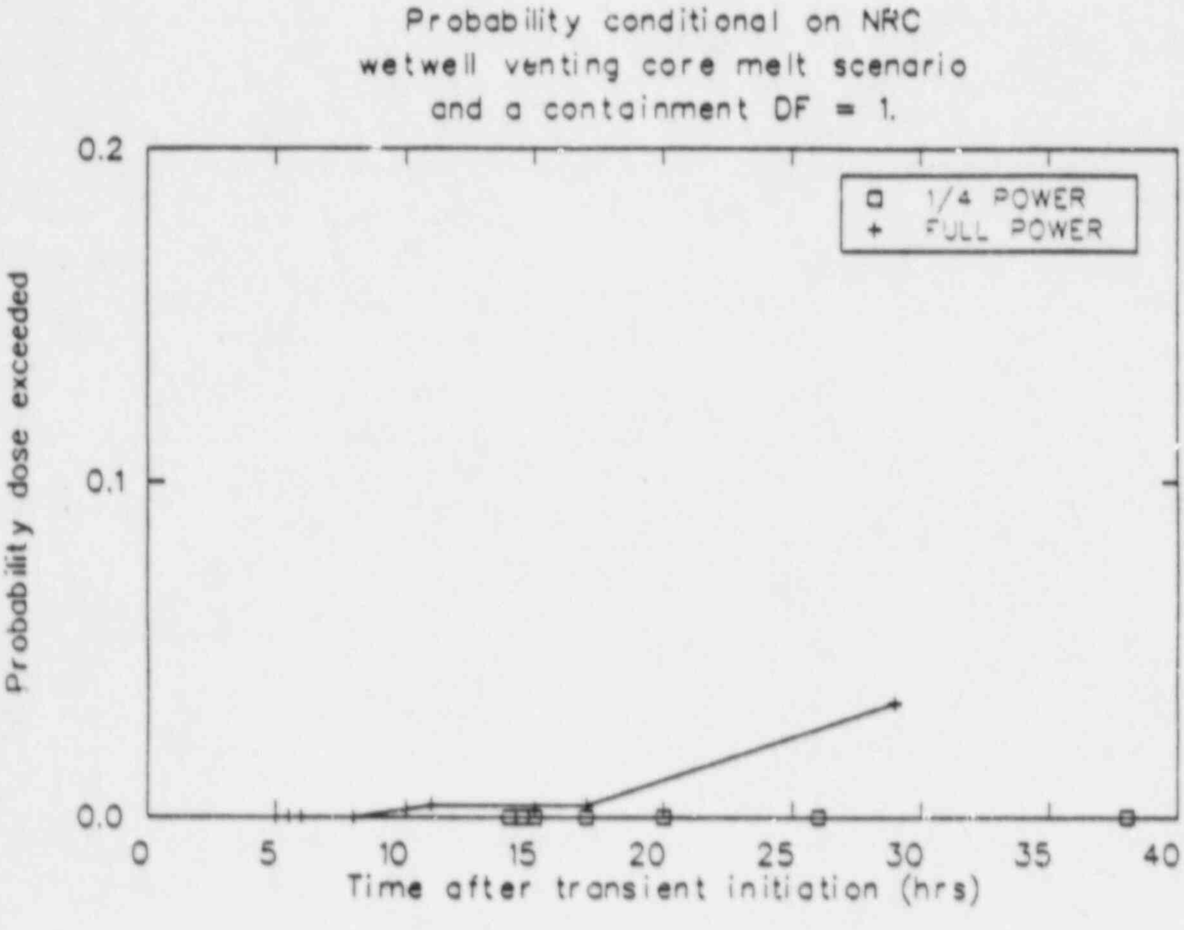


Figure 8 CRAC2 calculated probability of 200 rem whole body dose being exceeded at 2 miles from the plant -- wetwell venting scenario

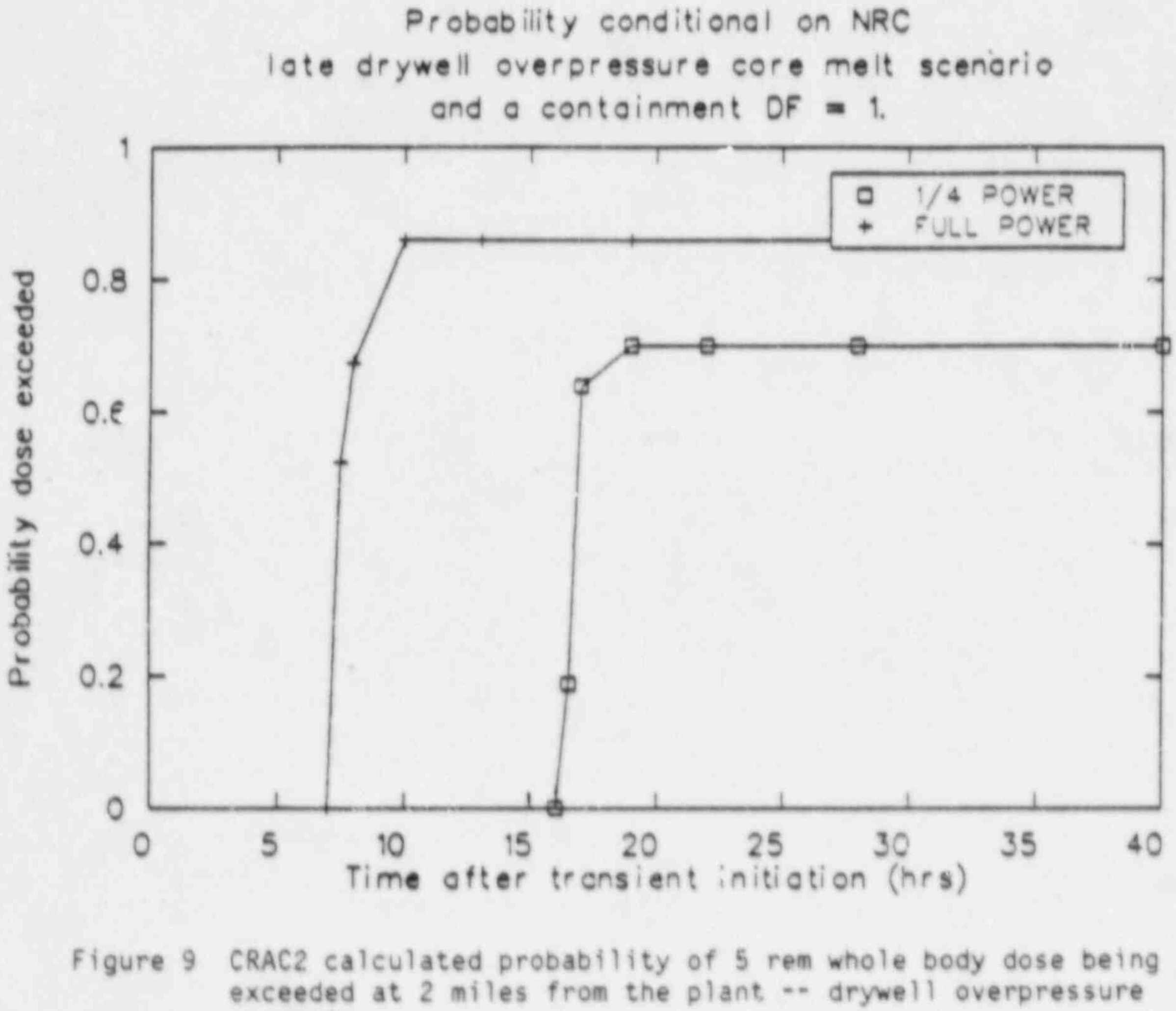


Figure 9 CRAC2 calculated probability of 5 rem whole body dose being exceeded at 2 miles from the plant -- drywell overpressure scenario

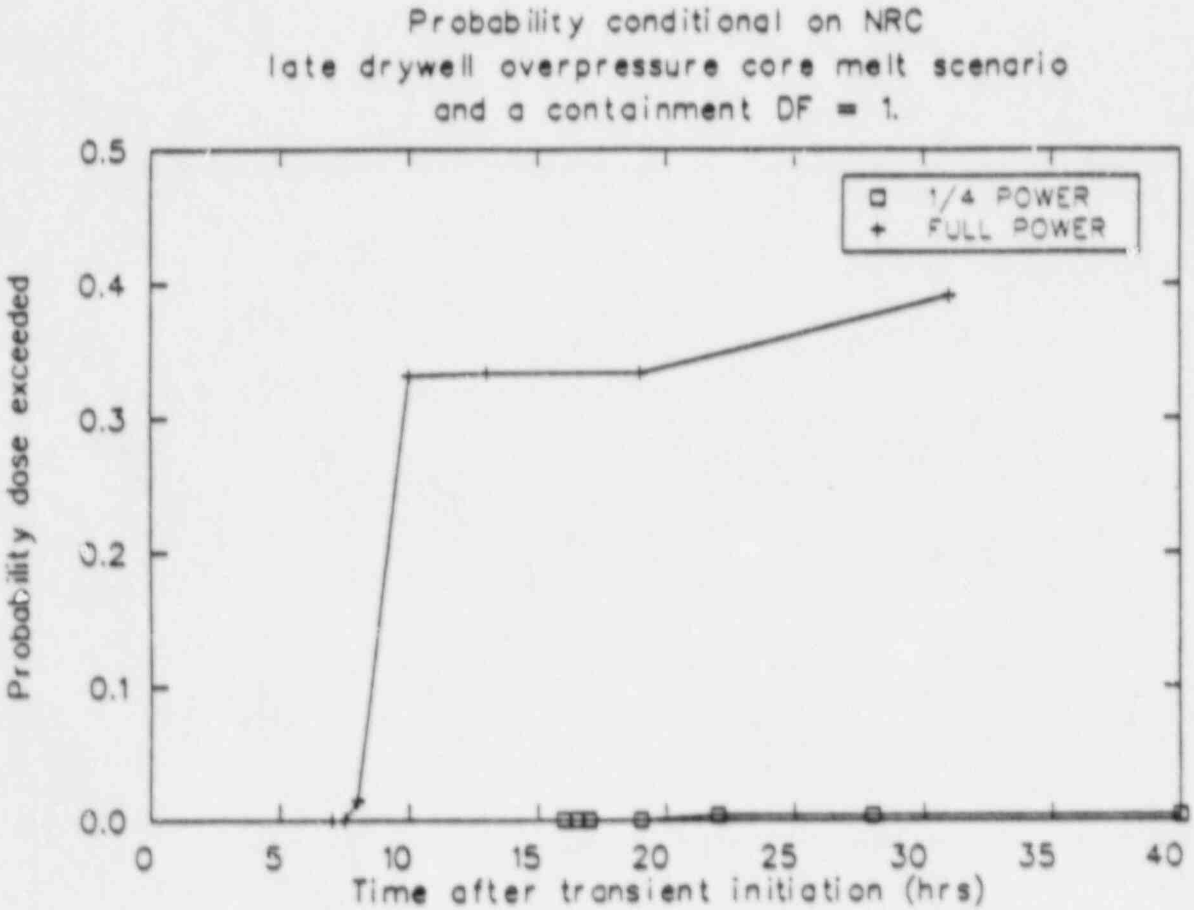


Figure 10 CRAC2 calculated probability of 200 rem whole body dose being exceeded at 2 miles from the plant -- drywell overpressure scenario

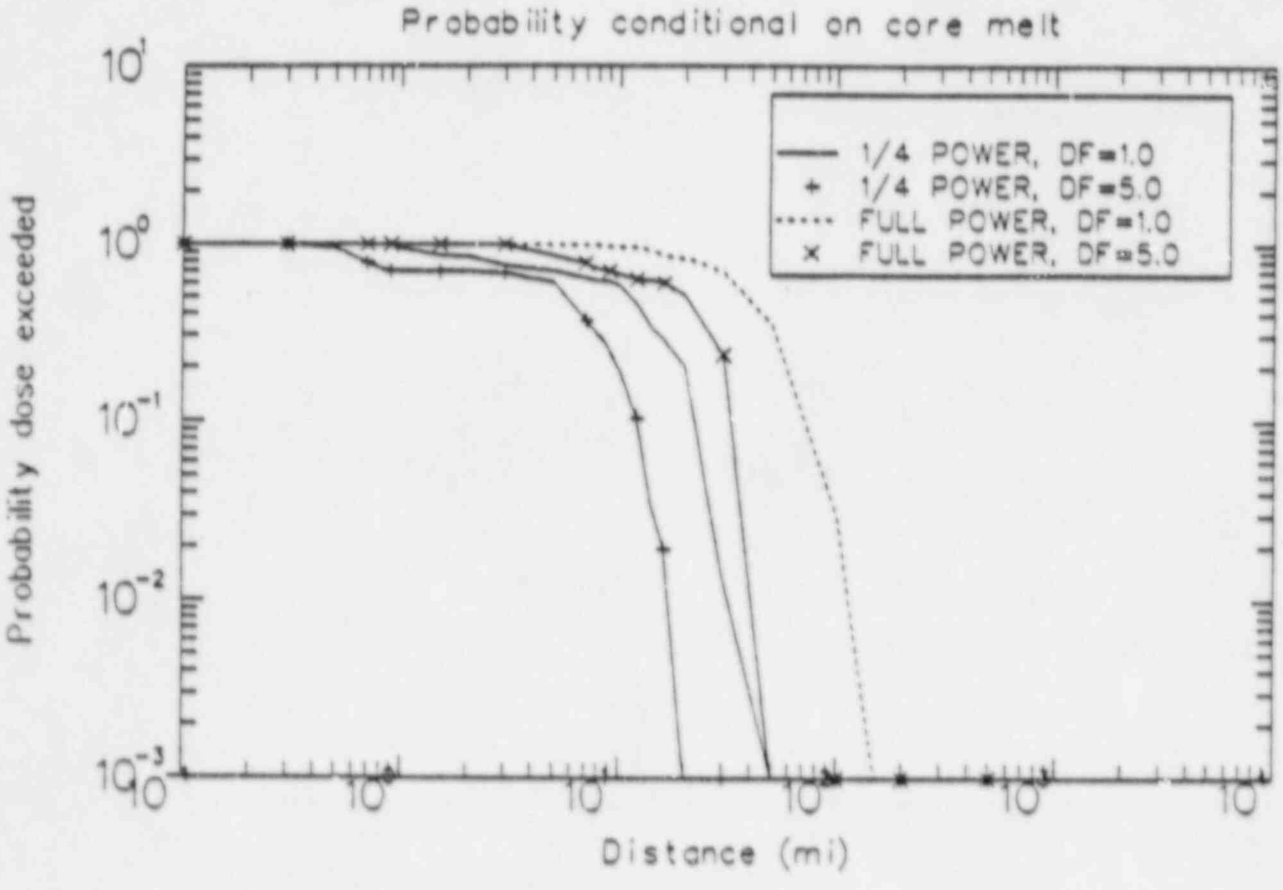


Figure 11 CRAC2 calculated dose-versus-distance probability distributions for whole body dose of 5 rem -- early release scenario

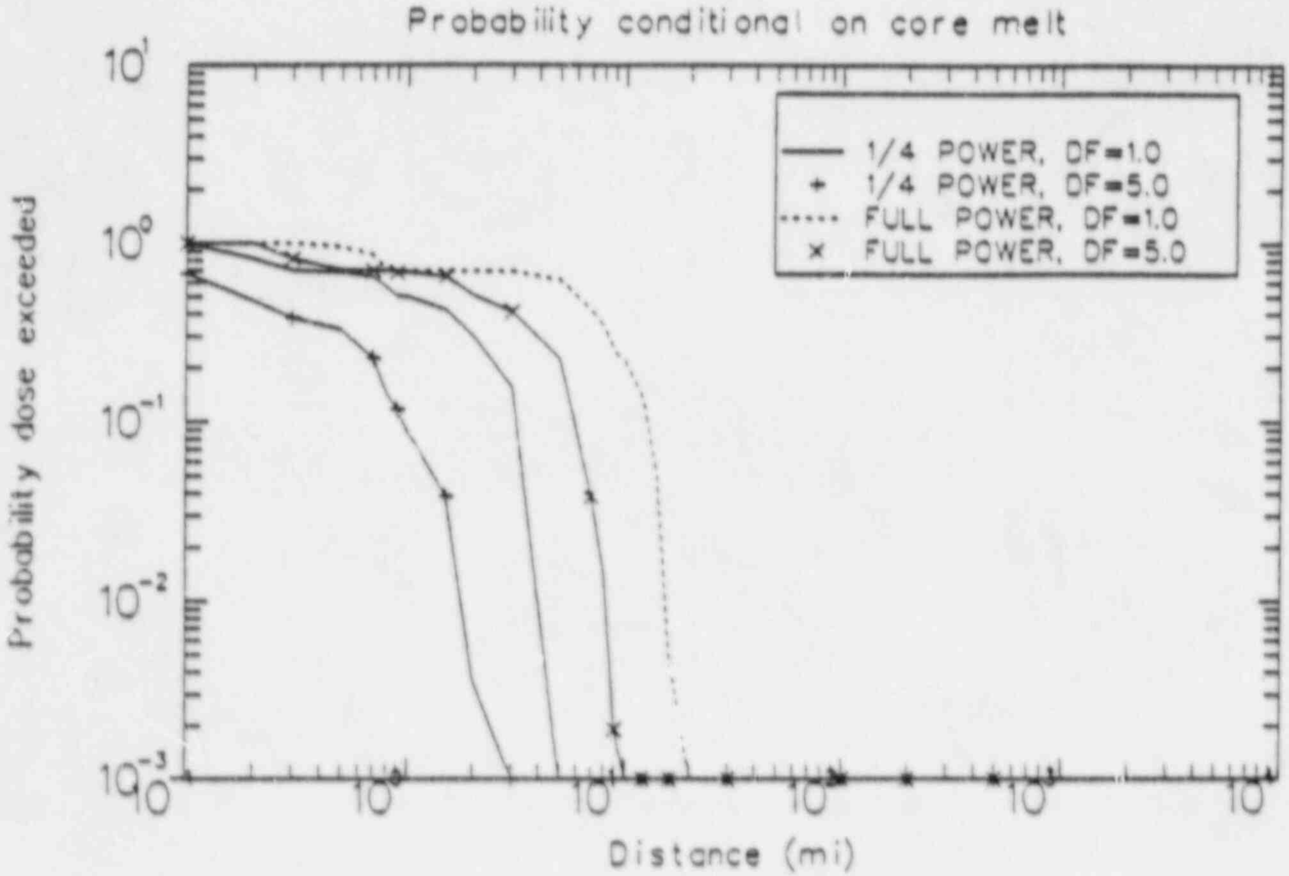


Figure 12 CRAC2 calculated dose-versus-distance probability distributions for whole body dose of 200 rem -- early release scenario

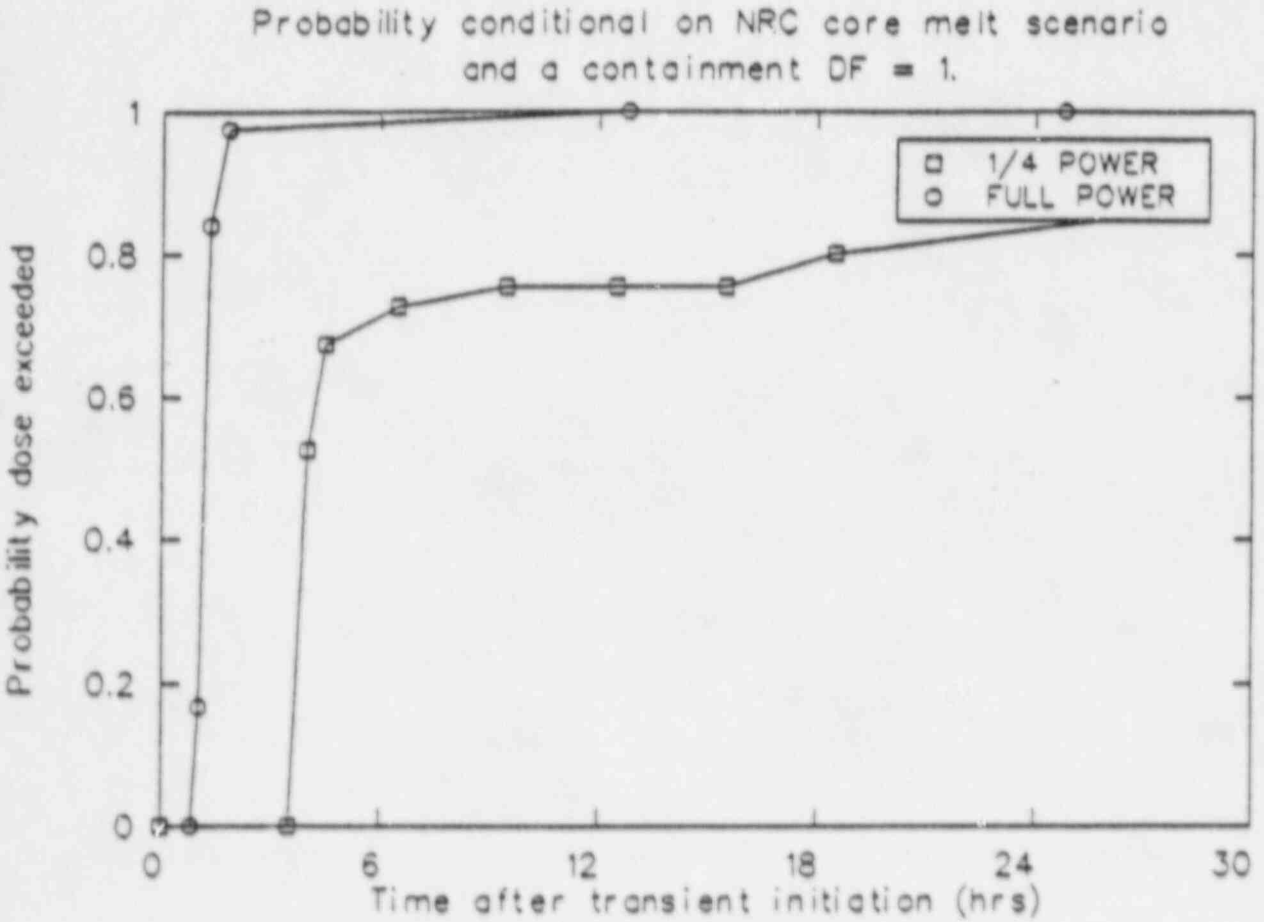


Figure 13 CRAC2 calculated probability of 5 rem whole body dose being exceeded at 2 miles from the plant -- early release scenario

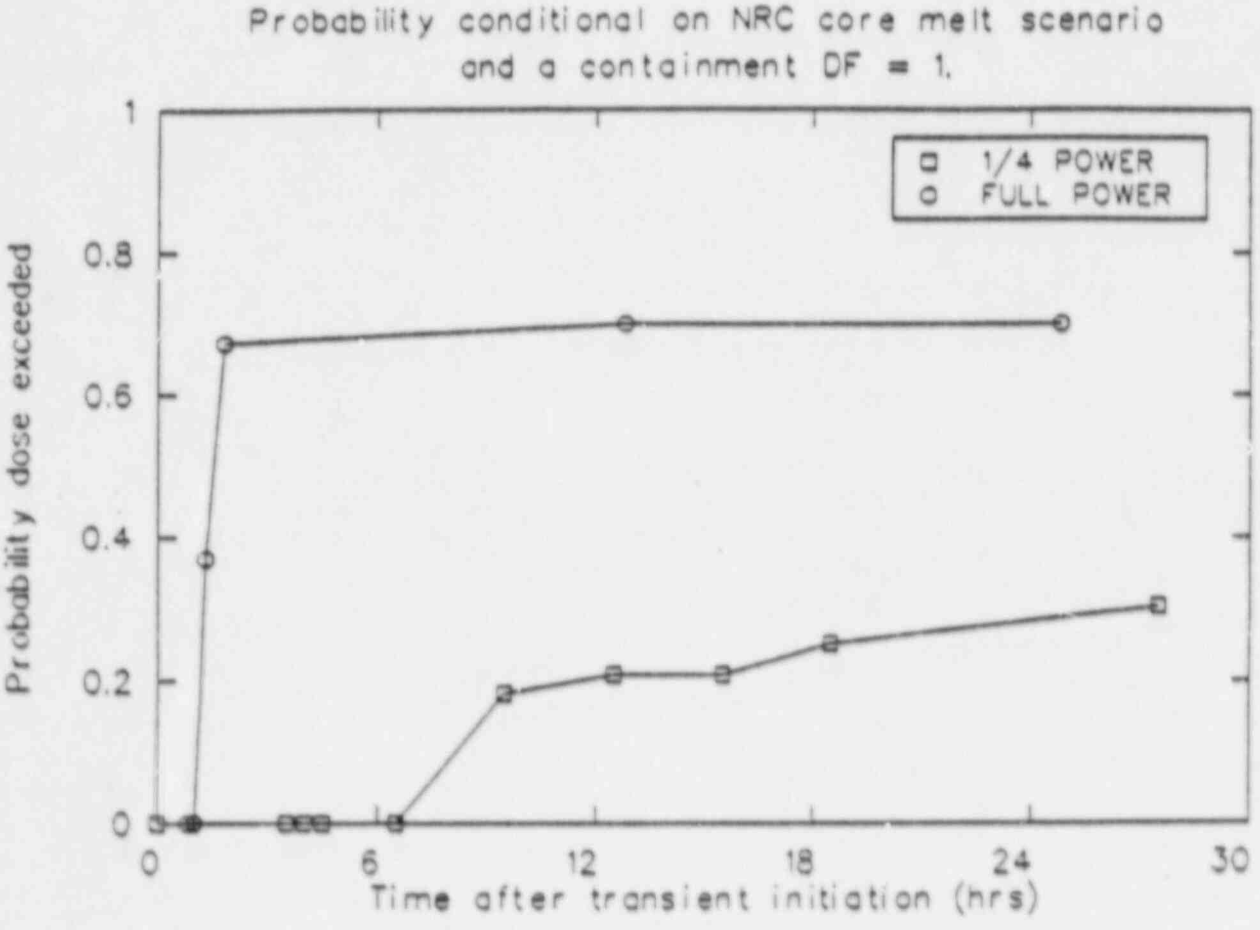


Figure 14 CRAC2 calculated probability of 200 rem whole body dose being exceeded at 2 miles from the plant -- early release scenario

APPENDIX A

EVALUATION OF SNPS CORE MELT FREQUENCY ESTIMATE FOR 25 PERCENT POWER

A.1 Introduction

LILCO claims that the frequency of core melt accidents at Shoreham will be significantly reduced by (1) operation at 25 percent, and (2) a number of plant upgrades which have been implemented since the original PRA. The objective of the staff's review was to assess the validity of the utility's assertion. Emphasis of the review was on treatment of risk-important sequences (e.g., ATWS, station blackout, and interfacing system LOCA), and treatment of external events. The staff's review of the treatment of risk important sequences is discussed in Section A.2 below. The treatment of external events in the PRA is discussed in Section A.3.

A.2 Comparative Evaluation of Risk Important Sequences

Table 1 in the main report shows values reported by LILCO for core melt frequency for 100 percent and 25 percent power operation at SNPS (References A.1 and A.2). The core melt frequency associated with restricting operation to 25 percent of rated power is about a factor of two below that reported for full-power operation. The staff judges this reduction to be well within the range of uncertainty in estimating core melt frequency, especially since the reported results are in the form of point estimates and large uncertainties are usually associated with the contribution from external events.

An evaluation was performed for those sequences triggered by internal or external initiators that may potentially result in early releases. These are:

1. Station Blackout Sequences

2. ATWS Sequences

3. LOCAs Outside the Containment

The review focused on the differences in these sequences at 25 percent and 100 percent power, and not on the estimates of core melt frequency in an absolute, quantitative sense.

A.2.1 Loss of Offsite Power Sequences

The contribution of loss of offsite power sequences to core melt frequency dropped from 10^{-5} per reactor year in the 100 percent PRA to about 3.6×10^{-7} per reactor year in the 25 percent PRA. The seismic contribution to these sequences is reported by the applicant to be about 2.7×10^{-6} per reactor year, and is relatively independent of power level.

The reduction in the contribution of these (non-seismic) sequences to core damage frequency is mainly due to:

1. Existence of redundant means of additional onsite AC power sources, and not considered in the original PRA, and

2. An increased time interval available for recovery actions as a result of the reduced level of decay heat.

Shoreham uses a frequency of occurrence for the loss of offsite power initiating event of 0.082 per reactor year based upon data from their grid. Evidence gathered by EPRI and NSAC and published in several EPRI and NSAC reports (References A.3 through A.6) indicates that loss of offsite power frequency for comparable plants in the Northwest Power Coordinating Council, which includes Shoreham, has a value of 0.13 per reactor year. Shoreham is in a unique geographical situation on Long Island because of the limited number of system interties. For this reason, the staff feels that the treatment of loss of offsite power initiating event frequency may be somewhat optimistic in the 25 percent power license submittal. However, it must be noted that if one uses the latest information available in the NSAC reports, the likelihood of recovery of offsite power is significantly better than the likelihood calculated in the

Shoreham analysis, which was based upon an earlier report (Reference A.6). Considering both issues together, the effect on total core melt frequency will be minimal if the loss of offsite power analysis is modified.

The 25 percent power PRA reported the unavailability of the black-start gas turbine to be 4×10^{-2} per demand based upon analysis of plant data. This value appears reasonable to the staff based upon review of other data sources. Credit is given for the remote start of this device in the event of a sustained loss of offsite power. No operator error is cited, however, given the time available, operator error would not be a significant contributor to failure of this backup source of power.

The study assigned a value of 0.3 per demand for unavailability of the three Colt Industry diesels, and assumed no credit for this source prior to four hours after a sustained loss of AC power. The relatively high unavailability is based primarily upon the method that must be used to connect this source to the in-plant distribution system, which is dominated by operator errors. The value assigned appears reasonable given the procedures that must be followed and the time available.

The on-site mobile power units are assigned a frequency of failure of 3×10^{-2} per demand for the common cause failure of three of four diesels (due primarily to operator errors). This value appears conservative given the time and the procedures that are available.

It is our conclusion that the credit given for the additional sources of AC power in loss of offsite power sequences is justified.

A.2.2 ATWS Sequences

The contribution of ATWS sequences to core melt frequency dropped from about 1.1×10^{-5} per reactor year in the full power PRA to about 4×10^{-6} per reactor year in the 25 percent power PRA. This reduction is credited to design changes as well as some procedural changes. The most important of these are:

1. Improvement in the standby liquid control system (SLCS) to include sodium pentaborate with a high enrichment in boron 10 isotopic content. This improvement is claimed to extend the time available for the operator to initiate the SLCS operations, and

2. Addition of a manual inhibit switch to the automatic depressurization system (ADS) to prevent automatic depressurization during an ATWS event and to avoid low pressure injection.

Restriction of the normal power level to 25 percent creates a unique situation for the PRA under ATWS conditions, in that the turbine bypass valve (TBV) can deliver 25 percent of rated steam flow to the main condenser. If this mode of heat transfer remains available, the operator is not under pressure to initiate shutdown by boron injection within a specific time, and for those event sequences the 25 percent power PRA claims that the core melt frequency is determined by hardware only. This claim ignores the possibility of operator errors of commission which could, for example, interrupt the 25 percent power absorption capability of the TBV and condenser. Nevertheless, the staff agrees that the 25 percent power bypass capability provides an additional success path that is not available at full power.

The event sequences in the 25 percent power PRA cover many cases where heat transfer to the main condenser would not be available and where operator actions would be required for attaining shutdown and decay heat removal. The study uses a period of 43 minutes as being available for SLCS initiation. In addition, for certain event sequences, operator manipulation of the reactor water level is assumed in the PRA, either to promote boron mixing by raising the water level or to reduce the reactor power level by lowering the water level. The dependence of the PRA upon operator reliability in these event sequences involves two considerations. First, the human error probability (HEP) values are derived from the HEP model or correlation of Reference A.7. The applicability of this generic correlation to the very specific unique actions involved in these event sequences is a source of uncertainty. Second, the PRA credits procedures and training, especially simulator based training, for limiting the HEP values and for preventing the inducement of operator stress that could increase the HEP values or increase the variability of operator behavior and consequently the uncertainties in these values.

The degree of implicit credit in the PRA for operator actions during the ATWS requires validation of the procedures and training for these actions and, also, some empirical confirmation of the HEP values for specific events. The credit given to timely operator action in case of the ATWS sequences remains to be a source of uncertainty in PRA studies in general. However, it is the staff's view that the ATWS sequence frequency and concerns related to credit for operator actions are reduced at 25 percent power due to the greater time available for operator actions relative to operation at full power.

A.2.3 LOCAs Outside the Containment

Large LOCAs outside of containment were estimated in the Shoreham full-power PRA to contribute 3.6×10^{-8} per reactor year to core melt frequency. In the 25 percent power PRA, the frequency of occurrence of these events has decreased to about 1.2×10^{-8} per reactor year. This decrease is primarily due to changes in the analysis of the high pressure/low pressure boundary failures and not to the effect of the power restriction. The staff considers this result to be reasonable.

A.3 Treatment of External Events

The original SNPS PRA (Reference A.2) scope included analysis of internal floods. This study was followed by the February 1985 Major Common-Cause Initiating (MCCI) Events Study (Reference A.8), which covered the remainder of external events. As part of the 25 percent power license submittal, the MCCI study was modified (Reference A.9) to reflect the current status of SNPS design and procedures, as well as relevant plant characteristics associated with the 25 percent power operation. The following subsections describe the results of the staff's review of the external events segment of the PRA studies.

A.3.1 Internal Flood Analy

In the 100 percent PRA internal flooding was identified as a leading contributor to the core damage frequency calculated for Shoreham. The Brookhaven review (Reference A.10) prepared an alternative analysis that indicated the frequency of core damage calculated in the Shoreham PRA for the internal flood initiators may be low by an order of magnitude. The dominant flood scenarios in both analyses were those that occurred at elevation 8' of the reactor building. All of the plant emergency core cooling system pumps are located at this elevation.

In the 25 percent power PRA, the internal flooding scenarios do not contribute significantly to either core damage or risk to the public. The primary reason for this is that credit is given to the operation of the CRD pumps in the 25 percent power PRA. These pumps are located above the reactor building flood elevation and are expected to be unaffected by floods in the reactor building. The CRD pumps are capable of maintaining reactor vessel inventory for initiating events which occur from 25 percent power. Based upon the review of the information provided in the license submittal, the use of the CRD pumps in the internal flooding scenarios appears reasonable and is consistent with the other sequences in the PRA which took credit for this alternate high pressure injection source.

A.3.2 Analysis of Seismic Events

The analysis of seismic events at Shoreham was performed for LILCO by Dames and Moore corporation (D&M). Within the same approximate time period, D&M also performed the seismic analyses for Millstone 3 (which is located within 30 miles from Shoreham) and Seabrook.

The staff did not perform a detailed review of the seismic analysis for Shoreham. However, References A.11 and A.12 describe a detailed review of the seismic issues for Millstone 3. A key issue identified in that review is that the seismic hazard assumed for the Millstone site may be an order of magnitude too low. The staff has compared the seismic hazard curves from the Shoreham PRA to preliminary curves available for the Shoreham site from the Seismic Hazard Characterization Project (SHCP). In contrast to Millstone, the

Shoreham SHCP curves are closer to those used in the utility PRA. Based on this comparison, it is our judgment that an increase in the utility estimates of seismic hazard by a factor of five would represent a reasonable high estimate of uncertainty for regulatory purposes at Shoreham. This is not to say that this high estimate represents the true upper limit of scientific uncertainty or that the true seismic hazard could not be less than that proposed in the Shoreham study. Certainly there is no compelling evidence in the historic record that would indicate any likelihood of large earthquakes in eastern Long Island. If the increase in seismic hazard were to translate into an equivalent increase in core melt frequency for seismic events at Shoreham, i.e., a factor of five, the frequency of seismically-induced core melt sequences would increase to approximately 1×10^{-5}, which is about one fifth that for internally-initiated events. It should be pointed out, however, that comparisons between seismic and non-seismic core melt frequency estimates are not completely valid since mean seismic hazard estimates directly reflect modelling uncertainties, whereas internal event estimates do so to a much lesser extent. As a result, comparisons of the means tend to overestimate the relative contribution of the seismic events to core damage and risk. Furthermore, this effect would influence the results in both the 100 percent power PRA and the 25 percent power PRA.

Additional seismic concerns include:

- The effects of a seismic event on non-safety related equipment, other than offsite power and reactor recirculation pumps, was not evaluated in the seismic analysis. Other reviews of seismic analysis have indicated that this omission may have significant effects on the results of the seismic analysis (especially the effects of seismically induced fires due to failures in non-safety equipment). This effect should be evaluated for the Shoreham PRA including the 25 percent power PRA.

- Relay chatter was identified in the Structural Mechanics Associates study (performed for LILCO) as a seismic failure mode. However, this failure mode was assumed not to cause system failure. Without investigating the likelihood of successful operator action after relay chatter has occurred, this assumption appears optimistic.

A.3.3 Fire Analysis

The MCCI studies performed for Shoreham include a fire analysis of selected areas. The 100 percent power MCCI study (Reference A.8) concluded that fires contributed 7.3×10^{-6} to total core damage frequency (approximately 10 percent). The MCCI study for 25 percent power (Reference A.9) indicates that core damage frequency contribution from fires is 4.6×10^{-7} (approximately two percent).

The original fire study performed bounding calculations for fire areas in the plant and refined the bounding analysis for the fires considered to be risk important. Three fire zones were analyzed in detail as the major contributors to fire damage potential.

The 25 percent power MCCI study only reanalyzed the three dominant fire zones from the original analysis. All other fire zone damage frequencies are less than that calculated for the 25 percent power analysis.

We have identified several areas relating to the fire analysis which should be addressed by the applicant, however, our judgment is that they would not significantly affect the PRA results. These are as follows:

- Operator recovery of fires: The values quoted for operator recovery (Event Q) in Table 3-2 of the 25 percent power MCCI study is 1×10^{-2} for operator actions within 30 minutes. The original analysis used a value of 0.7 for the same event for actions within 10 minutes. The change in timing is reasonable based upon the plants limited power level but the value assigned for recovery appears optimistic when one considers the confusion inherent in the fire scenarios analyzed in the 25 percent power MCCI study. The effect of changing this operator recovery value has not been evaluated for this review. However, changing this operator recovery value to its original value would not significantly change the core damage frequency from that calculated in the 25 percent power MCCI study.

- Fires inside the containment: The original MCCI fire analysis screened out a majority of the fire initiating events in the data base that occurred in the containment building of PWRs on the basis that the BWR containment is nitrogen inerted during power operation. The MCCI update reevaluated fires in the containment because at power levels less than 15 percent, the containment need not be inerted. However, those fire events that were screened out in the original MCCI study were not reintroduced into the data base. The fires that were screened out were caused by oil leakage from PWR reactor coolant pumps. The recirculation pumps at Shoreham are also oil lubricated, therefore, we feel that the events are indicative of events which could occur inside a BWR non-inerted drywell. Including these events would increase the frequency of fires inside the non-inerted drywell by a factor of six, which does not significantly affect the core damage frequency calculated for fires.

- Fires Involving the Fuel Oil Storage Tank: The effects of a fire involving the contents of the gas turbine fuel oil storage tank were included in the original MCCI study. However, only the effects on safety-related structures were shown. Several offsite power lines (135 and 69 kV) pass near this storage tank. It is not clear whether the effects of a fuel oil storage tank fire on offsite power distribution were evaluated. This tank is also located on a small hill above th major site structures. It is also not clear whether the effect of a fire and a dike breech or excessive smoke in the vicinity of the safety related structures (primarily diesel generator buildings and control room) was evaluated.

- Other fires: Several fires induced by welding were screened out of the fire data base in the 100 percent power MCCI study. Welding, per se, is not precluded during power operation at most operating reactors. Without further justification of the reasons for excluding these fire events, we feel that these events should remain in the data base. However, keeping these fire occurrences in the data base will not significantly change the results of the fire analysis performed for the 25 percent power PRA.

A.3.4 Other External Events Analysis

The original MCCI report presented analysis of other external initiating events such as high wind, external flood, turbine missile, and aircraft crash. The other external event initiators did not contribute significantly to either core damage or the risk to the public. The 25 percent power MCCI study did not re-examine these other initiators but based upon the results obtained in the 100 percent power PRA determined that the frequency of core damage due to these events was significantly less than the seismic and fire events included in the analysis.

The original MCCI study of these other external initiating events was reviewed and compared with the results of other similar studies (Reference A.11). Based upon these reviews and comparisons, the conclusions stated in the original MCCI study and the 25 percent power MCCI study are reasonable.

A.4 Summary

Comparison of reported core damage frequency results as shown in Table 1 indicated that SNPS operation at the reduced power level results in a reduction in the overall core damage frequency of about a factor of two. This is well within the uncertainties associated with estimating core melt frequency, especially considering that the reported results are in the form of point estimates and that uncertainties can be much larger than a factor of two. External events (seismic and fires) and estimates of human error data are the potential major contributions to these large uncertainties.

A review of seismic hazard calculations for Shoreham indicates that the uncertainty could increase the hazard by a factor of five. A similar increase in core melt frequency for seismic events would place seismically-induced core melt at about one-fifth the frequency presented for the sum of the internal initiating events. This effect, however, would influence the results in both the 100 percent power PRA and the 25 percent power PRA. Some additional concerns were raised about the treatment of fires, however, they remain a minor component of total core damage frequency for the 25 percent power PRA. Also, they may have a greater effect on the 100 percent power PRA results than on the 25 percent power PRA.

Based upon the limited review performed on the systems analysis segment of the 25 percent power PRA submittal, the staff concludes that core melt frequency at 25 percent power is not significantly different than at 100 percent power.

A.6 References

A.1. "Request for Authorization to Increase Power to 25% and Motion for Expedited Commission Consideration," Long Island Lighting Company, Docket No. 50-322, April 14, 1987.

A.2. Probabilistic Risk Assessment of the Shoreham Nuclear Power Station, Docket 50-322, Long Island Lighting Company, June 1983.

A.3. Losses of Offsite Power at U.S. Nuclear Power Plants - All Years Through 1983, NSAC 80, July 1984.

A.4. Losses of Offsite Power at U.S. Nuclear Power Plants - All Years Through 1984, NSAC 85, June 1985.

A.5. Losses of Offsite Power at U.S. Nuclear Power Plants - All Years Through 1985, NSAC 103, May 1986.

A.6. Loss of Offsite Power at Nuclear Power Plants: Data and Analysis, EPRI NP-2301, March 1982.

A.7. Delian Corp., "Probabilistic Risk Assessment of the Shoreham Nuclear Power Station: Initial Power Operation Limited to 25% of Full Power", April 1987.

A.8. NUS Corp., "Major Common-Cause Initiating Events Study -- Shoreham Nuclear Power Station", NUS-4617, February 1985.

A.9. NUS Corp., "Major Common-Cause Initiating Events (MCCI) Contribution to Shoreham Nuclear Power Station Core Damage Frequency -- Early Plant Operation at 25% Rated Power", NUS-4842, March 1987.

A.10. NUREG/CR-4050, "A Review of the Shoreham Nuclear Power Station Probabilistic Risk Assessment", Brookhaven National Laboratory, November 1985.

A.11. NUREG/CR-4142, "A Review of the Millstone 3 PRA, Lawrence Livermore National Laboratory, April 1986.

A.12. NUREG-1152, "Millstone 3 Task Evaluation Report", June 1986.