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June 12, 1987
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Mr. S. Singh Bajwa
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
MS P522
Washington, DC 20555

Dear Mr. Bajwa:

This is to forward the final letter report

"Considerations Regarding Certain Aspects of Severe Accident Mitigation Afforded by Operation of Shoreham at Reduced Power"

by S. A. Hodge and R. M. Harrington. The report provides the results of various calculations done at Oak Ridge at the request of Mr. Robert L. Palla of the Risk Applications Branch of NRR. The calculations address Total Loss of Injection, Anticipated Transient Without Scram (ATWS), and Large-Break LOCA Accident Sequences with emphasis on the differences in event timing for sequences initiated from 25% power as opposed to 100% power.

Please let me know if you desire additional information concerning these matters.

Sincerely,

A. P. Malinauskas, Director
NRC Programs

APM:SAH:jcm

Encl: Letter Report

cc: R. M. Harrington
J. B. Henderson, NRR
S. A. Hodge
R. L. Palla, NRR

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CONSIDERATIONS REGARDING CERTAIN ASPECTS OF SEVERE ACCIDENT MITIGATION AFFORDED BY OPERATION OF SHOREHAM AT REDUCED POWER

S. A. Hodge
R. M. Harrington

Boiling Water Reactor Severe Accident Technology Program
Oak Ridge National Laboratory
Oak Ridge, Tennessee

Letter Report
June 12, 1987

Research sponsored by the U.S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation under Interagency Agreement DOE 0554-0554-A1 with the U.S. Department of Energy under contract DOE-AC05-84OR21400 with the Martin Marietta Energy Systems, Inc.

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CONTENTS

1. INTRODUCTION

This report describes the results of analyses performed at Oak Ridge National Laboratory (ORNL) to assess certain aspects of the effectiveness of operation of the Shoreham Nuclear Power Station at reduced power as a Severe Accident mitigation technique. This work was performed at the request of the Risk Applications Branch of the Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation in response to a current request of the Long Island Lighting Company (LILCO) to operate the Shoreham facility at 25% of full rated power. (The facility would be so operated until certain emergency planning issues related to the granting of a full power license are resolved.)

The specific calculations performed in this work are those specified by the NRC technical sponsor. These include representation of both Loss of Injection and Anticipated Transient Without Scram (ATWS), the two accident classes that in general dominate the risk of core melt at boiling water reactor facilities. This general status applies also to Shoreham, for which LILCO has indicated, in its request for 25% power operation, that loss-of-injection type accident sequences contribute about 70% and ATWS type sequences contribute about 14% of the total estimated core melt frequency.

Other calculations have been performed to estimate the timing of events for the large-break LOCA accident sequence compounded by loss of all reactor vessel injection. Although this combination of initiating events represents a much lower fraction of the overall estimated core melt frequency, it is of interest because it leads to the shortest estimated time to the onset of core degradation. All accident sequence studies include, where appropriate, a calculation of the event timing for the case of initial operation at 100% of rated power, for comparison with the results for 25% power operation.

The authors of this report have been involved with severe accident calculations for boiling water reactor facilities since late 1980. These previous efforts were performed as part of the Severe Accident Sequence Analysis (SASA) Program from October 1980 until September 1986 and for the follow-on Boiling Water Reactor Severe Accident Technology (BWRSAT) Program from October 1986 to the present. Both of these programs have been conducted at ORNL under the auspices of the Office of Nuclear Regulatory Research. A list of the SASA and BWRSAT program reports that the authors of this letter report either wrote entirely or co-authored is provided as Refs. 1—15.

All of the calculations whose results are represented in this letter report were carried out on a best-estimate basis. This includes the models employed for representation of the decay heat power as a function of time.

2. MAJOR POTENTIAL SEVERE ACCIDENT MITIGATION ASPECTS OF OPERATION AT 25% POWER

The purpose of this chapter is to provide a brief discussion of each of three major areas in which operation of the Shoreham Nuclear Power Station at 25% of rated power would be expected to decrease the severity of postulated severe accident sequences.

2.1 Delay in the Timing of Severe Accident Events

Reactor scram from 25% power operation as opposed to scram from 100% power operation translates directly into a factor-of-four reduction in the subsequent decay heat power levels. This cannot, however, be expected to simply be reflected in a factor-of-four increase in the time periods between successive major events of a severe accident sequence. Because of the effects of chemical energy release caused by oxidation of the metals within the core region and the complicated action of the safety/relief valves in releasing steam from the vessel, code calculations are necessary to provide estimates of the extended event timing.

Nevertheless, there can be no question that the reduced level of decay heat power will have a significant effect upon extending the time available for opera r action, equipment repair, or personnel evacuation. Results of calculations of the severe accident sequence event timing for total loss-of-injection sequences are provided in Chap. 3, Anticipated Transient Without Scram (ATWS) event timing is discussed in Chap. 4, and event timing for large-break LOCA with failure of ECCS is provided in Chap. 5.

2.2 Reactor Response to Sudden Reactivity Insertion Under ATWS Conditions

The extent of control blade insertion into the core is greater for operation at 25% power than it is for operation at 100% power. Accordingly, under ATWS conditions, less of the total negative reactivity, which serves in the overall reactivity balance to maintain the core critical, is due to voids. It follows that perturbations to the critical state that tend to collapse voids in the core region will insert less positive reactivity with the control blades in their 25% power positions than with the control blades in their 100% power positions.

This matter is important because required operator actions in the unlikely event of failure of the sodium pentaborate neutron-absorbing poison function are complicated and difficult to accomplish properly. The operators may lose control of the situation resulting in the sudden injection of large amounts of cold water or in pressure excursions, either of which causes sudden void collapse. The more sluggish response of the core to void collapse with the control blades in their 25% power position is discussed in Chap. 4.

2.3 Reduced Opportunity for Core Debris-Concrete Interaction

The Shoreham Nuclear Power Station is fitted with four downcomers immediately beneath the reactor vessel within the pedestal region of the drywell. All potential flow pathways from the inpedestal region to the expedestal region of the drywell are blocked to a level about two feet above the floor. Since the upper surface of the inpedestal downcomers is flush with the floor, the majority of any molten material emanating from the reactor vessel would be expected to flow through the downcomers and be quenched in the pool.

If there is no significant buildup of core debris on the drywell floor, there will be no significant degree of core debris-concrete interaction. It is important to recognize that in order for fission products other than noble gases to reach the outside-plant environment, they must be propelled from the plant by some motive force. This motive force might be provided by the pressure stored within the reactor vessel before vessel bottom head failure, by the pressure stored within the primary containment before drywell or wetwell pressure boundary failure, by flashing of an overheated pressure suppression pool upon primary containment depressurization, or by the gases released from concrete by the ablation process driven by the hot core debris. In general, BWR severe accident sequences proceed in such a manner that most of the volatile fission products are trapped in the pressure suppression pool or are condensed upon reactor vessel surfaces; furthermore, the reactor vessel and containment pressure sources have been dissipated by the time the core debris begins to leave the reactor vessel. In these cases, it is the core debris-concrete interaction that provides the motive gases, which sparge through the core debris and carry a small fraction of the remaining fission products to the outside environment, bypassing the pressure suppression pool via the previously-failed primary containment boundary.

The lower decay heat power of core debris resulting from 25% power operation would, of course, slow the rate of concrete ablation. On the other hand, if all of the core debris were to leave the reactor vessel as molten liquid and flow through the inpedestal downcomers to the pool, there would be no core debris-concrete interaction at all. These matters are discussed in connection with the total loss-of-injection calculation results in Chap. 3.

3. CALCULATIONS FOR LOSS OF INJECTION (TQUV) SEQUENCES

In the TQUV [Transient-induced scram, failure of normal feedwater system to provide core make-up water, failure of high-pressure (HPCI, RCIC) and low-pressure (RHR, Core Spray) systems to provide reactor vessel injection] accident sequence, there is a transient followed by reactor scram, followed by failure of function of all of the systems that would normally be relied upon to deliver cooling water into the reactor vessel as necessary to keep the core covered. If the reactor scram occurs from 100% power, the uncovering of active fuel begins in <1/2 h.[6] The TQUV accident sequence has been among the dominant sequences in almost all BWR probabilistic risk assessments (PRAs).[6] Indeed, in the LILCO submission for operation of the Shoreham Station at 25% of full power, the loss-of-injection class of accident sequences is indicated to constitute ~70% of the total risk of core melt.

It is important to recognize that only a small injection source is necessary to keep the core covered when the only energy release within the reactor vessel is that due to decay heat. This point is discussed in Chap. 1 of Ref. 6 for the case of scram from 100% power, and, of course, even less injection is required in the case of scram from 25% power. For example, with the reactor vessel remaining at pressure, an injection rate of just 77.5 gpm taken from the condensate storage tank at a temperature of 90°F would be sufficient to maintain a constant reactor vessel water level above the core for times greater than 10 min after scram. Since the core would not be uncovered during the 10 min period before injection is assumed to be initiated, an injection rate of 77.5 gpm, which is less than the capacity of the control rod drive (CRD) hydraulic system when the reactor is scrammed, would preclude core uncovery.* Thus, we must include loss of the CRD hydraulic system in addition to the loss of function of the standard injection systems specified by the TQUV definition if we are to define a basis for consideration of a severe accident situation. We will term this heightened level of pumping system degradation "total loss of injection."

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*Plant tests at Shoreham have demonstrated a CRD hydraulic system injection capacity of 112 gpm with the reactor scrammed and the reactor vessel at pressure. However, if the control room operator should reset the scram, this injection rate would automatically decrease to about 57 gpm. This demonstrates why plant procedures and operator training should emphasize the relation between the condition of the shutdown reactor (scrammed or scram-reset) and the available CRD hydraulic system injection flow.

3.1 BWRSAT Program Severe Accident Models

The Boiling Water Reactor Severe Accident Technology (BWRSAT) Program at Oak Ridge National Laboratory employs the severe accident modeling strategy outlined in Table 3.1. The models described in this table were developed at ORNL by L. J. Ott and have been incorporated into the Boiling Water Reactor Severe Accident Response (BWRSAR) code. It should be carefully noted that this methodology results in significantly longer times to reactor vessel bottom head penetration failure than other methods because of the contention that the large amount of water in the BWR bottom head must first be boiled away and that the debris must then reheat to about 1800°F before the reactor vessel bottom head penetrations can fail. It should also be noted that, in general, the debris in the bottom head is not molten at the time of penetration failure and therefore the debris begins to pour from the vessel sometime after penetration failure. If the reactor vessel is pressurized at the time of penetration failure, then the gas blowdown through the bottom head debris has a cooling effect upon the debris.

3.2 The Total Loss-of-Injection Sequence

The sequence and timing of events following a postulated total loss of reactor vessel injection at the Shoreham station as calculated by the BWRSAR code are provided in Table 3.2. It is assumed that the reactor had been operating at 25% power at the time of scram and, in spite of the long times involved, it is assumed that no injection source is ever recovered. For conservatism in the analysis, there is no modeling of pressure suppression pool cooling or operation of the drywell coolers. The reactor vessel is assumed to remain at pressure.

Plots of certain key parameters representing events within the reactor vessel are provided in Fig. 3.1. These plots represent events from just after core uncovery, which occurs at time 138 min, until 1500 min, which is some 24 min after reactor vessel bottom head penetration failure and the beginning of reactor vessel blowdown to the drywell. The effect of bottom head penetration failure at time 1476 min upon reactor vessel pressure can be seen on the extreme right side of Fig. 3.1a, which also shows the effect of earlier safety/relief valve actuations that occurred before bottom head dryout puts an end to reactor vessel steam generation at time 1181 min.

The swollen reactor vessel water level, which includes consideration of the effect of voids, is shown in Fig. 3.1b. The calculated level in general follows a smoothly descending curve, with perturbations caused by safety/relief valve actuations, as water is boiled off by decay heat. However, after time 395 min, molten core debris begins to fall into the remaining water above the core plate, accelerating the rate of decrease in level. After core plate dryout at time 483 min, the water level remains relatively constant, just below the core plate (whose upper surface is at 199.6 in.). However, the BWRSAR code does recognize the displacement of water in the bottom head whenever large debris masses are introduced into it; this is manifested most clearly in Fig. 3.1b by the temporary water level increase shown when radial region two of the core collapses into the bottom head at time 644 min. This temporary level increase is followed by a rapid major decrease of almost 100 in. caused by steam generation as the mass of radial column two is quenched. Subsequently, the decay heat associated with the fuel pellets of radial column two causes a boiloff of the remaining water in the bottom head and bottom head dryout occurs at 1181 min.

Figure 3.1c represents the gradual heatup of the fuel rods in radial column three and the subsequent large reduction in temperature that follows collapse of radial column (zone) two at time 644 min. The quenching of the mass of radial column two in the bottom head releases a large amount of steam, which causes accelerated zirconium-water reaction and a concomitant temperature increase in the outer regions of the core where zirconium metal still exists in the cladding. However, all of the remaining clad structure in radial column three is zirconium oxide at this time, so the steam release associated with the collapse of radial column two produces only rapid cooling there. This effect is important, because without it, radial column three would soon reach temperatures that would cause its remaining zirconium oxide clad structure to lose strength and it would also fall into the bottom head, instead of remaining in place.

Plots (d), (e), and (f) of Fig. 3.1 pertain to the extent of hydrogen generation by the zirconium-steam reaction. Some 51% of the clad and 39% of the channel box wall is predicted to be oxidized during the accident sequence, producing about 1840 lbs of hydrogen. (This demonstrates the wisdom of maintaining an inerted primary containment atmosphere during reactor operation.)

Selected primary containment response characteristics are displayed in the individual plots provided in Fig. 3.2. The primary containment pressure (Fig. 3.2a) shows significant increase in response to the collapse of radial column two into the reactor vessel bottom head at time 644 min and the associated discharge of noncondensible gases. The concomitant drywell atmosphere temperature response (Fig. 3.2b) is slight because this discharge from the reactor vessel is via the safety/relief valves and is cooled during its passage through the pressure suppression pool. Both the pressure and temperature response are marked, however, following the inception of reactor vessel blowdown at time 1476 min when the release is directly into the drywell atmosphere. Nevertheless, the peak pressure is not of a magnitude to threaten primary containment pressure boundary integrity and the temperature excursion is rapidly terminated as the heat sinks absorb the released energy. This does not threaten the integrity of the drywell liner, however, since its peak calculated surface temperature in response to the reactor vessel blowdown is only about 270°F, as shown in Fig. 3.2c.

Certain calculated conditions within the wetwell are shown in Fig. 3.2 plots (d) through (f). As indicated in Fig. 3.2e, a large mass of hydrogen is collected in the wetwell airspace when the reactor vessel is flushed by the large release of steam from the bottom head at the time of collapse of radial column two. However, it should be recognized that this is not all of the hydrogen in the containment; some 450 lbs are predicted to have been passed back into the drywell via the wetwell-to-drywell vacuum breakers.

Figure 3.3 indicates the integrated mass of molten material predicted by the BWRSAR code to have been released from the Shoreham reactor vessel as a function of time following reactor vessel pressure boundary failure at time 1476 min. There is a small initial release of molten stainless steel immediately upon bottom head penetration failure, but the passage of blowdown gases cools the debris and significant additional release does not occur until about time 1910 min. The calculation is carried out to time 2880 min (48 h after scram), although the calculated reactor vessel wall temperature at the end of the calculation (about 2210°F at the connection with the vessel skirt) is much too high for the wall structure to have survived. Indeed, it must be expected that the reactor vessel wall would fail shortly after time 2100 min, when the average wall temperature in the vicinity of the vessel skirt is about 1600°F.

For accident sequence time 2100 min, the BWRSAR code predicts that some 101,000 lbs of molten metal eutectics have left the reactor vessel whereas about 556,000 lbs of solid debris remain within the vessel, including all of the original inventory of UO_2 fuel. Should the reactor vessel bottom head wall fail by creep-rupture at about this time, a great deal of solid debris would be deposited onto the drywell floor. The results of these calculations do not support the contention of the LILCO submission that all debris falling onto the Shoreham drywell floor would be molten, subject to immediate passage through the downcomers into the pressure suppression pool.

3.3 Comparison with Total Loss of Injection From 100% Power

In order to clearly demonstrate the very significant delay in severe accident event timing associated with operation at 25% of full power, the total loss of injection sequence was recalculated with all parameters the same except for the initial power, which was set at 100% of rated (8.312×10^9 Btu/h). The differences in timing of the major events of the accident sequence are indicated in Table 3.3.

In addition to the large delays in severe accident event timing associated with operation at 25% of full power, there are two other significant differences in the calculated plant response. First, more hydrogen is generated in the case of total loss of injection from 25% power, 1840 lbs as opposed to 1620 lbs for the 100% power case. The relative fractions of clad, channel box, and control blade stainless steel sheath oxidation are provided in Table 3.4. The major source of the additional hydrogen is seen to be the additional oxidation of the channel box walls, while they are slowly heated at oxidizing temperature in the 25% power case. (In the 100% power case, the channel box wall reaches its melting temperature more rapidly and more of the zirconium metal is relocated downward without oxidizing.) However, a great deal of hydrogen is predicted to be generated in either case so the additional 14% in the 25% power case, while of interest, does not constitute a special concern.

The second significant difference in calculated plant response is also derived from the much slower heatup rate in the 25% power case. After bottom head dryout, the debris in the reactor vessel bottom head is heated by its internal decay heat generation. For the 100% power case, the debris heats rapidly and its temperature greatly exceeds that of the reactor vessel wall, which is heated only by conduction from the debris. For the 25% power case, however, the debris temperature increases much more slowly and the average reactor vessel wall temperature does not lag so far behind. The upshot of this is that relatively little of the debris has become molten and left the vessel at the time that the average wall temperature reaches the creep-rupture failure range in the 25% power case whereas most (but not all) of the debris has left the vessel by this time in the 100% power case. A comparison of the calculated quantities is provided in Table 3.5. This matter is important because the debris leaving the reactor vessel in the molten liquid phase is expected to immediately enter the pressure suppression pool via the downcomers whereas most solid material falling on the drywell inpedestal region floor upon creep-rupture failure of the bottom head would not.

It should be noted that the models now employed for the BWRSAR calculation for definition of the formation of eutectic mixtures within the core debris and the melting temperatures of these mixtures are current best-estimate and subject to revision upon completion of certain pertinent experiments now planned by Dr. Dana Powers at Sandia National Laboratories.

Table 3.1. BWRSAT program methodology employed to represent events between onset of core degradation and reactor vessel failure for BWRs

1. As canister and control blade material becomes molten, it is relocated onto the core plate. This causes:
   a. a temporarily increased steaming rate
   b. core plate dryout and cessation of steaming
   c. buildup of mass on the core plate and core plate heatup.
2. Each radial region of the core plate fails due to loss of strength when its calculated temperature has increased to 1275°F (964 K). Each core plate region and its accumulated debris falls into the lower plenum, producing a burst of steam and lowering the water level there as the fallen material is quenched.
3. Molten Zr metal flows downward over the lower core fuel rod nodes, leaving the UO_2 fuel pellets encased in thin ZrO_2 sheaths. Steam rising from the lower plenum cools the core nodes from which all unoxidized Zr has been removed. On the other hand, the rising steam causes energy release in the core peripheral nodes where Zr metal at elevated temperature still remains.
4. The standing portions of the core fall into the lower plenum by radial column. Each core column collapses when its average clad temperature reaches 4250°F (2616 K), at which time very little of the UO_2 mass in the region has become molten. (The actual failure mechanism is weakening, by overtemperature, of the ZrO_2 sheaths surrounding the UO_2 fuel pellets.) The failing mass is quenched by the water in the lower plenum until the time of bottom head dryout. After bottom head dryout, the debris begins to reheat.
5. The structure of the control rod guide tubes in the lower plenum is heated by the surrounding core debris and is weakened to the point of failure when its temperature reaches 1400°F (1033 K). Failure of the control rod guide tubes causes all remaining standing portions of the core to immediately collapse.
6. Bottom head penetrations fail by a simulated creep-rupture mechanism as the debris mass in their vicinity is reheated to about 1800°F (1255 K). The reactor vessel depressurizes and equalizes with drywell pressure.
7. The individual components of the debris mass leave the vessel only after they have reheated to their melting points and thereby become liquid.

Table 3.2. Major events of the total-loss-of-injection accident sequence for scram from 25% power

Event	Time After scram (mins)
MSIV closure and total loss of injection	0
The swollen water level falls below the top of active fuel	138
Structural relocation of core material begins	395
Core plate dryout	483
Core plate region 1 fails	522
Core plate region 4 fails	566
Core plate region 2 fails	641
Radial region two of core collapses	644
Core plate region 3 fails	644
Core plate region 5 fails	856
Bottom head dryout	1,181
Radial region one of core collapses	1,476
Bottom head penetrations fail and blowdown into drywell begins	1,476
High temperature weakens control rod guide tube structure and remaining portions of the core collapse. A small amount (13,500 lbs) of molten steel leaves the vessel. The remainder of the core debris is frozen	1,476
Beginning of continuous pour of molten material from the reactor vessel	1,910
Average reactor vessel wall temperature exceeds 1600°F in vicinity of attachment to supporting skirt. Creep rupture failure of vessel wall and partial separation of lower portion of bottom head is probable	2,100

Table 3.3. Comparison of the timing of major events in the total-loss-of-injection accident sequences for initial powers of 100% and 25%

Event	Time after scram (mins)	
	100% power	25% power
MSIV closure and total loss of injection	0	0
Swollen water level falls below top of active fuel	25	138
Structural relocation of core material begins	68	395
Core plate dryout	74	483
Radial region two of core collapses	110	644
Bottom head dryout	231	1181
Bottom head penetrations fail, blowdown into drywell begins	267	1476
Beginning of continuous pour of molten material from the reactor vessel	267	1910
Average reactor vessel wall temperature exceeds 1600°F in vicinity of attachment to supporting skirt. Creep rupture failure of vessel wall and partial separation of lower portion of bottom head is probable	610	2100

Table 3.4. Comparison of the calculated hydrogen generation parameters in the total-loss-of-injection sequences for initial powers of 100% and 25%

Parameter	100% power	25% power
Fraction of clad oxidized	0.48	0.51
Fraction of channel box wall oxidized	0.23	0.39
Fraction of control blade stainless steel sheath oxidized	0.06	0.09
Total hydrogen generated, lbs	1620	1843

Table 3.5. Comparison of the calculated quantities of core debris invessel and exvessel at the time that the reactor vessel wall temperature exceeds 1600°F

Debris constituent	100% power case time 610 min		25% power case time 2100 min	
	Invessel (lb)	Exvessel (lb)	Invessel (lb)	Exvessel (lb)
Zr	22,600	48,700	44,400	18,000
Fe	27,200	156,200	127,200	55,900
ZrO_2	35,600	17,100	65,200	0
UO_2	42,600	235,600	278,800	0
Other[a]	14,100	53,600	39,300	27,600
Total	142,100	511,200	554,900	101,500

[a]Cr, Ni, B_4C, FeO, Fe_3O_4, Cr_2O_3, NiO, and B_2O_3.

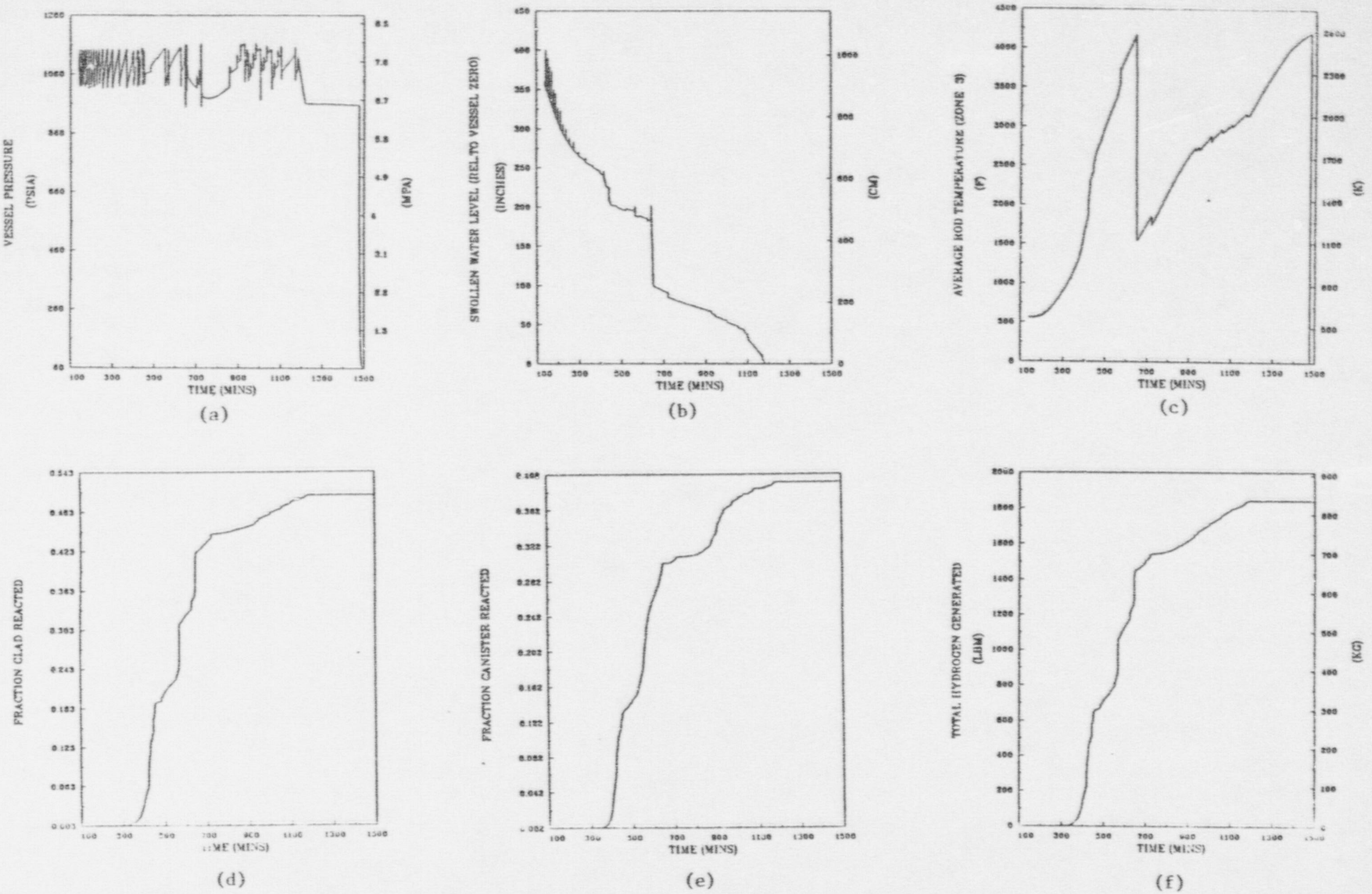


Fig. 3.1. Accident signatures for events within the reactor vessel as predicted by the BWRSAR code for total loss of injection following scram from 25% power at the Shoreham nuclear facility.

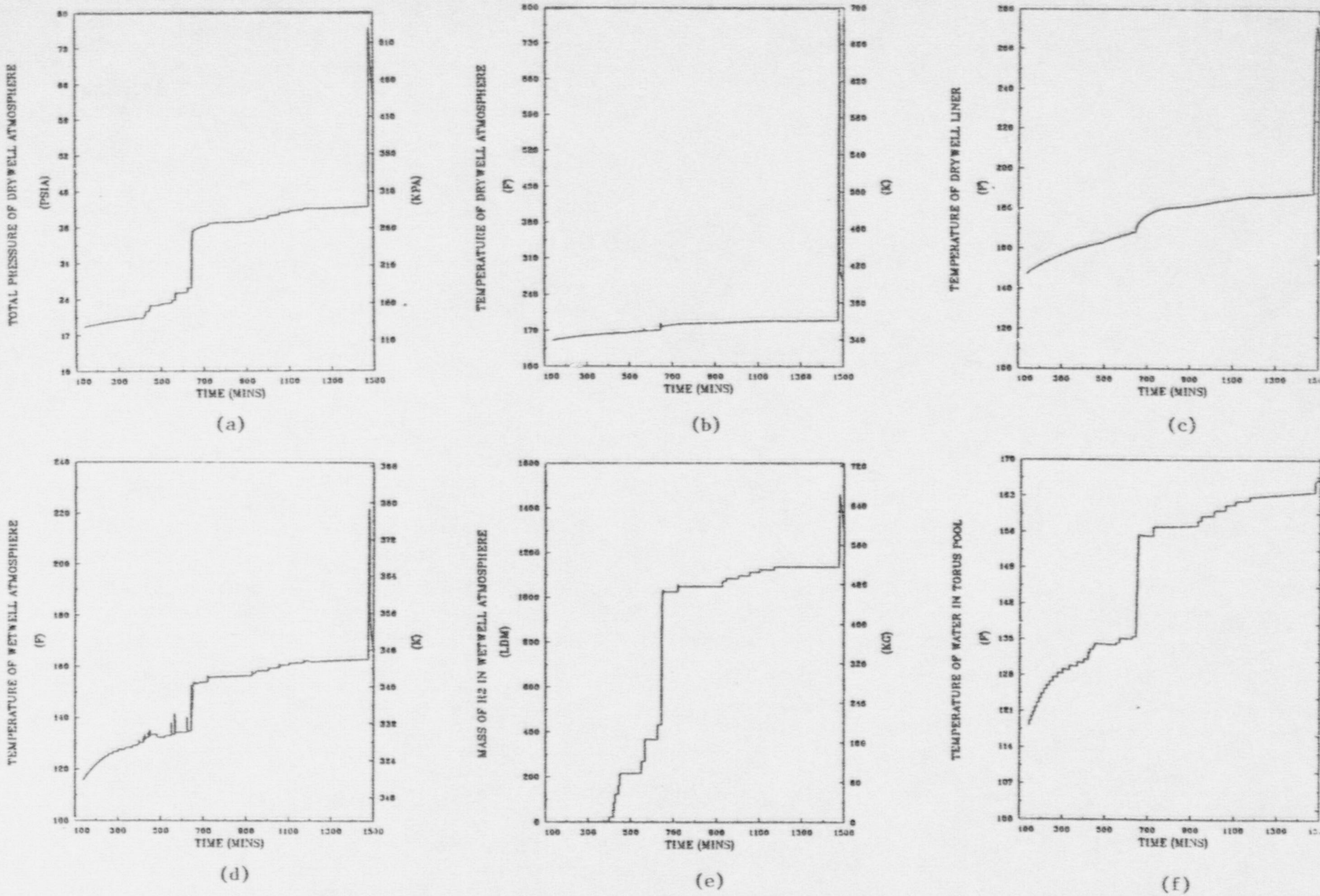


Fig. 3.2. Accident signatures for the response of the drywell and wetwell as predicted by the BWRSAR code for total loss of injection following scram from 25% power at the Shoreham nuclear facility.

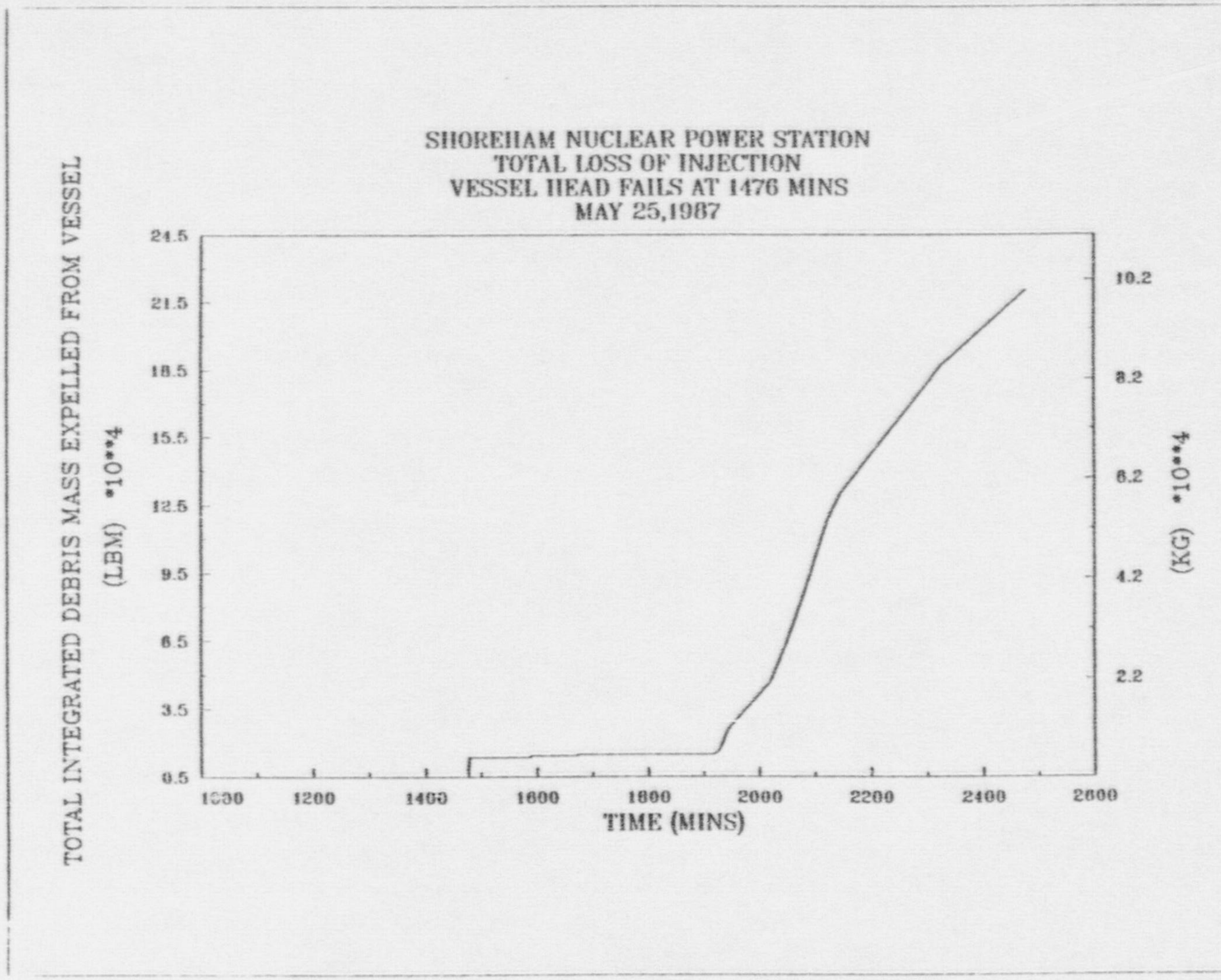


Fig. 3.3. Total mass of molten core debris having left the reactor vessel as a function of time during the latter phase (to 2480 minutes) of a total loss of injection accident sequence from 25% power at the Shoreham nuclear facility. The BWRSAR code calculation was terminated at time 2880 minutes (48 hours). All of the original UO_2 inventory is predicted to remain frozen within the vessel bottom head at this time; however, the calculated wall temperature is so high that the lower portion of the bottom head must have separated from the reactor vessel before this time.

4. ATWS CALCULATIONS

This chapter describes the predicted response of the Shoreham Nuclear Power Station to a postulated complete failure to scram following a transient event that has caused closure of all main steam isolation valves (MSIVs). This accident sequence is the most severe of a class of sequences commonly denoted "ATWS", the acronym for "Anticipated Transient Without Scram." With the MSIVs closed, almost all of the steam exiting the reactor vessel would be passed into the pressure suppression pool through the safety/relief valves (SRVs); the remainder would be used to drive the High Pressure Coolant Injection (HPCI) or Reactor Core Isolation Cooling (RCIC) system turbines during their periods of operation and then, as turbine exhaust, would also enter the pressure suppression pool. Since the rate of energy deposition into the pool can greatly exceed the capacity of the pool cooling equipment, the possibility of excessive pressure suppression pool temperatures leading to primary containment failure by overpressurization is of primary concern during the analysis of ATWS accident sequences.

As in all designs, the criticality of the Shoreham nuclear reactor depends upon a complicated set of factors that simultaneously introduce positive or negative reactivity. Whether there is a power increase, constant power, or a power decrease at a given point in time depends upon the particular reactivity balance at that instant. In BWR studies, it is necessary to recognize the importance of the void coefficient of reactivity. In the BWR, boiling takes place within the core and "voids" are created by the steam bubbles formed within the core volume. The moderation or slowing-down of neutrons is much less in steam than in liquid water so increased voiding has the effect of reducing the supply of thermal neutrons. Therefore, an increase in voids introduces negative reactivity and a decrease in voids introduces positive reactivity. Since the BWR operates with the water moderator at saturation conditions within the core, negative or positive reactivity insertions caused by the creation or elimination of voids are a natural and important result of reactor vessel pressure changes.

Provision is made for rapid reactor shutdown under emergency conditions by neutron-absorbing control blades that can quickly and automatically be inserted (scrammed) into the core upon the demand of the reactor protection system logic. When inserted, the control blades introduce enough negative reactivity to ensure that the reactor is maintained subcritical even with the moderator at room temperature and with zero voids in the core.

Although all transient-initiated accident sequences can most easily be brought under control and terminated by successful scram of the control blades, they can also be brought under control and terminated by appropriate operator action. In other words, given properly trained operators and properly functioning equipment, the failure of the scram function can be considered to be merely a nuisance requiring a more complicated and time-consuming method of achieving shutdown.

One important action that the operators are directed to take in the event of ATWS is to initiate the standby liquid control system (SLCS) if the pressure suppression pool temperature reaches 110°F. This system injects a neutron-absorbing poison of sodium pentaborate solution into the reactor vessel by means of positive displacement pumps. Previous studies[11] at ORNL have demonstrated that with operator action limited to successful initiation of the SLCS, the ATWS accident sequence can be brought under control. If a postulated ATWS accident sequence is to degenerate into a severe accident, then, we must also assume failure of the SLCS.

4.1 Operator Actions if the Poison Injection Function Fails

The BWR Owners Group Emergency Procedures Guidelines (EPGs) provide a strategy for operator actions to deal with the MSIV-closure initiated ATWS that can be summarized as follows: Attempt manual scram and, if not successful, begin manual insertion of control blades. Initiate the SLCS and pressure suppression pool cooling. Reduce core power by taking manual control of the reactor vessel injection systems and lowering the reactor vessel water level to the top of the core; this reduces core inlet flow by interrupting the natural circulation path from the core through the steam separators and back through the jet pumps in the downcomer region. The result is increased voiding in the core and increased temperature of the flow at the core inlet; the latter because the lowered water level uncovers the feedwater spargers through which the HPCI and RCIC systems inject, thereby providing heating of the injected droplets as they all through a steam environment.

The EPGs are intended to be symptom-oriented instructions to the control room operator that are comprehensive and cover every eventuality. To maintain assurance that the thermal energy released from the primary system can be condensed in the pressure suppression pool, there is a requirement that reactor vessel pressure be reduced as the pressure suppression pool temperature increases. For the Shoreham reactor, pressure reduction must begin when the suppression pool temperature exceeds 150°F and continue in accordance with a graph of permissible maximum reactor vessel pressure vs suppression pool temperature. (Previous studies at ORNL[7] demonstrate for Browns Ferry that once reactor vessel depressurization is begun, it must be continuous because each increment of energy deposited in the pool during depressurization increases the suppression pool temperature to the extent that, following the graph, further depressurization would be required.) There is no suggestion in the EPGs that the graphical schedule for reactor vessel depressurization as pressure suppression pool temperature increases should not be followed in the event of ATWS.

Recent preliminary work at Rensselaer Polytechnic Institute (RPI) indicates that the Shoreham nuclear reactor would be subcritical with the reactor vessel water level lowered to the top of the core and the reactor vessel depressurized to 200 psi or less, given the control blades stuck in their normal 25% power operating positions.[16] In other words, the small rate of water injection to the reactor vessel required to maintain the vessel water level at the top of the core would be converted to steam by decay heat alone, and the associated voids within the core region would keep the core subcritical. If subsequent analyses confirm this finding, then, should the SLCS system fail, the Shoreham reactor could be brought subcritical simply by lowering the reactor vessel water level and pressure, while keeping the core covered. It is emphasized that this would not be the case if the control blades were stuck in their (more withdrawn) normal 100% power operating positions.

4.2 Consequences of Low-Pressure Criticality

The question as to whether or not the Shoreham reactor would maintain criticality with the water level lowered to the top of the core and the reactor vessel depressurized is of some importance because of the potential for rapid insertion of positive reactivity under such conditions. This might occur either by uncontrolled injection of cold water by the large-capacity low-pressure injection systems or by any small action that initiates a reactor vessel pressure increase. As indicated by the data provided in the steam tables, the change in steam vapor specific volume for a given increment in pressure is much greater at low pressures.

If it is considered that the Shoreham reactor would remain critical in a depressurized state with the water level lowered to the top of the core, then it can be demonstrated, in general, that the power and pressure transients resulting from sudden insertions of positive reactivity would be less severe with the control blades stuck in their normal position for 25% power operation. To this end, two sets of demonstration calculations with the BWR-LTAS code[9] have been performed for ATWS situations identical except for the control blade positions. The determination of the additional control blade reactivity associated with 25% power operation is described in the Appendix. In both cases, no operator action is assumed and the calculations are based upon Browns Ferry parameters.

The calculated results for the two cases are shown in Figs. 4.1 and 4.2. Without operator action, HPCI, RCIC, and CRD injection maintain reactor vessel water level high above the top of the core until the HPCI turbine is lost upon high lube oil temperature (the system lube oil is cooled by the water being pumped, and without operator action, the system suction would automatically be shifted from the condensate storage tank to the pressure suppression pool; HPCI system failure is assumed at a pool temperature of 210°F).

With HPCI system failure, reactor vessel water level decreases even though the RCIC and CRD hydraulic systems continue to inject. The automatic depressurization system (ADS) is enacted when the combination of low reactor vessel water level and high drywell pressure is sensed.

This is much later for the case of ATWS from 25% power (Fig. 4.2b vs. 4.1b, where six open SRVs represent ADS actuation).

As the reactor vessel is depressurized into the regime in which the large-capacity low-pressure injection systems are able to pump cold water into the vessel, large cycles of injection — no injection begin (Figs. 4.1c and 4.2c). Each period of cold water injection produces void collapse within the core and subsequent reactor power spikes (Figs. 4.1d and 4.2d). However, the power spikes produced for the case with the control blades in their normal 25% power position are much less severe and less numerous than for the case with the normal 100% power control blade positions.

Each power spike produces a pressure increase within the reactor vessel that quickly terminates further water injection by the low-pressure systems. Without injection, the core rapidly becomes uncovered and power drops to decay heat levels. However, as long as the ADS valves remain open, the pressure then rapidly decreases and another cycle of cold-water injection begins.

If the increasing drywell pressure approaches the available control air pressure, the safety/relief valves will close, the reactor vessel will repressurize in a permanent fashion, and no additional low-pressure injection cycles will occur. As shown in Figs. 4.1c and 4.2c, this happens at about time 39 min in the case of an ATWS from 100% power, but not within 45 min for ATWS from 25% power. The associated drywell pressure traces are shown in Figs. 4.1e and 4.2e.

The conclusions obtained by this discussion and the associated calculations are 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% power operation as opposed to their configuration at 100% power. Since the relative negative reactivity contribution of the core voids is less with the blades in their 25% power configuration, any subsequent change in voids also has less effect. Perhaps the best evidence of the overall effect of this is the difference in predicted pressure suppression pool temperatures at time 40 min in Figs. 4.1f and 4.2f.

It should be noted that the results presented in this section for ATWS from 25% power operation were calculated on the basis of a negative 0.015 delta-K/K control rod reactivity differential associated with operation at 25% power as opposed to 100% power. This is considered to be a conservative basis since, as discussed in the Appendix, the best-estimate negative reactivity differential is -0.019. A test BWR-LTAS calculation was performed using -0.019 delta-K/K control rod reactivity, but the calculated reactor power was so low following loss of the HPCI system that the low reactor vessel water level required for ADS actuation was not reached within the 45 min period represented by the calculation.

4.3 The C9D ATWS Sequence

Together with their application for Shoreham to operate at 25% of rated power, LILCO submitted the results of several accident sequence calculations in which events were assumed to occur in such a manner that the accident proceeded beyond core melting and reactor vessel pressure boundary failure. Among these is the C9D ATWS accident sequence, in which the control blades are assumed stuck in their 25% power operating configuration.

The sequence of events for the C9D accident sequence up to the time of loss of injection as described by the LILCO submission is described in Table 4.1. Several conservative assumptions were employed by the LILCO subcontractors in determining this event sequence including failure of the SLCS function and loss of all reactor vessel injection systems upon the inception of wetwell venting, as well as the basic assumption that the reactor would be critical when depressurized with the water level at the top of the core. There would, of course, be no severe accident if any one of these assumptions is not valid.

To determine the sequence of events involving core degradation and reactor vessel failure that would occur after loss of all injection to the reactor vessel, a BWRSAR code calculation has been performed at ORNL for the period of the accident sequence after time 90 min. Core power was controlled by user input and the control blade positions were established so as to approximate the actual 25% power configuration. The predicted timing of events is provided in Table 4.2. Calculated hydrogen generation is 1866 lbs, resulting from oxidation of 57% of the clad and 41% of the channel box walls.

The ORNL best-estimate methods of calculating degraded core and reactor vessel failure events (Table 3.1) predict a much longer time to reactor vessel pressure boundary failure (30.8 h vs. 10.4 h) than does the MAAP code method used in calculating the results of the LILCO submission. (This is primarily because of the BWRSAT Program contention that core debris relocated into the reactor vessel bottom head is quenched as long as there is sufficient water remaining to do it.) Also, the LILCO submission predicts that molten core debris begins to leave the reactor vessel at the time of pressure boundary failure whereas the ORNL method delays the initial pour until the debris in the bottom head has reheated to its melting point. Thus, in the ORNL results calculated with the BWRSAR code, the initial pour of molten metal from the reactor vessel does not occur until time 37.5 h.

If it is desired to compare the event timing given in Table 4.2 for the ATWS case to the timing provided in Table 3.4 for the loss-of-injection cases, one must remember that only about 24% of the total control blade mass is represented as being in the core for the ATWS case; this has a significant effect upon the calculated results.

As a final comment in regard to the lessons learned by review of the C9D ATWS severe accident sequence, it should be noted that the primary containment atmosphere would be depleted of noncondensible gases after several hours of wetwell venting. If the situation within the reactor vessel were subsequently brought under control and the wetwell vent was then closed, a vacuum greater than the design basis might be drawn within the primary containment as the steam therein condensed. This is why plant emergency procedures should ensure that the composition of the primary containment atmosphere is ascertained and that nitrogen is introduced as necessary before the vent is closed.

Table 4.1 Major events of the C9D Anticipated Transient Without Scram (ATWS) accident sequence up to the time of loss of injection

Event	Time after MSIV closure (mins)
MSIV closure and failure of scram function	0.0
HPCI/RCIC/CRD maintain reactor vessel level between Level 8 and Level 2	0.0-18.0
Two RHR cooling loops aligned for pressure suppression pool cooling	10.0
Pressure suppression pool temperature reaches 200°F, ADS is manually actuated, vessel depressurization causes loss of HPCI operational capability. Operators lower reactor vessel water level to the top of the core and maintain it there by use of RCIC and CRD. Reactor power is 10% of full power. The ADS valves automatically open and close to maintain the reactor vessel pressure between 50 and 75 psi higher than that of the wetwell.	18.0
Containment pressure reaches 75 psia. Operators vent the wetwell airspace causing harsh environmental conditions in the secondary containment. All reactor vessel injection capability is assumed lost. Also, since the drywell pressure exceeds the available control air pressure, the ADS function is no longer operable and the reactor vessel repressurizes. Hereafter, the SRVs will open only automatically, on high reactor vessel pressure.	90.0

Table 4.2. Major events of the C9D Anticipated Transient Without Scram (ATWS) accident sequence after loss of injection

Event	Time after scram (mins)
The reactor vessel repressurizes quickly and the core is uncovered rapidly as core power is restored to 10% of full power by void collapse	90
The swollen water level falls below the top of active fuel	100
Low water level and increasing voids restore core power to decay heat levels	103
Structural relocation of core material begins	292
Core plate dryout	518
Radial region two of core collapses	565
Radial region three of core collapses	920
Core plate region 5 fails	923
Core plate region 1 fails	1062
Bottom head dryout	1337
Overtemperature weakens control rod guide tube structure and remaining portions of core collapse	1723
Bottom head penetrations fail and blowdown into drywell begins	1850
Molten core debris begins to pour from the reactor vessel	2250
Average reactor vessel wall temperature exceeds 1600°F in recently of attachment to supporting skirt. Creep rupture failure of vessel wall and separation of lower portion of bottom head is probable	2700

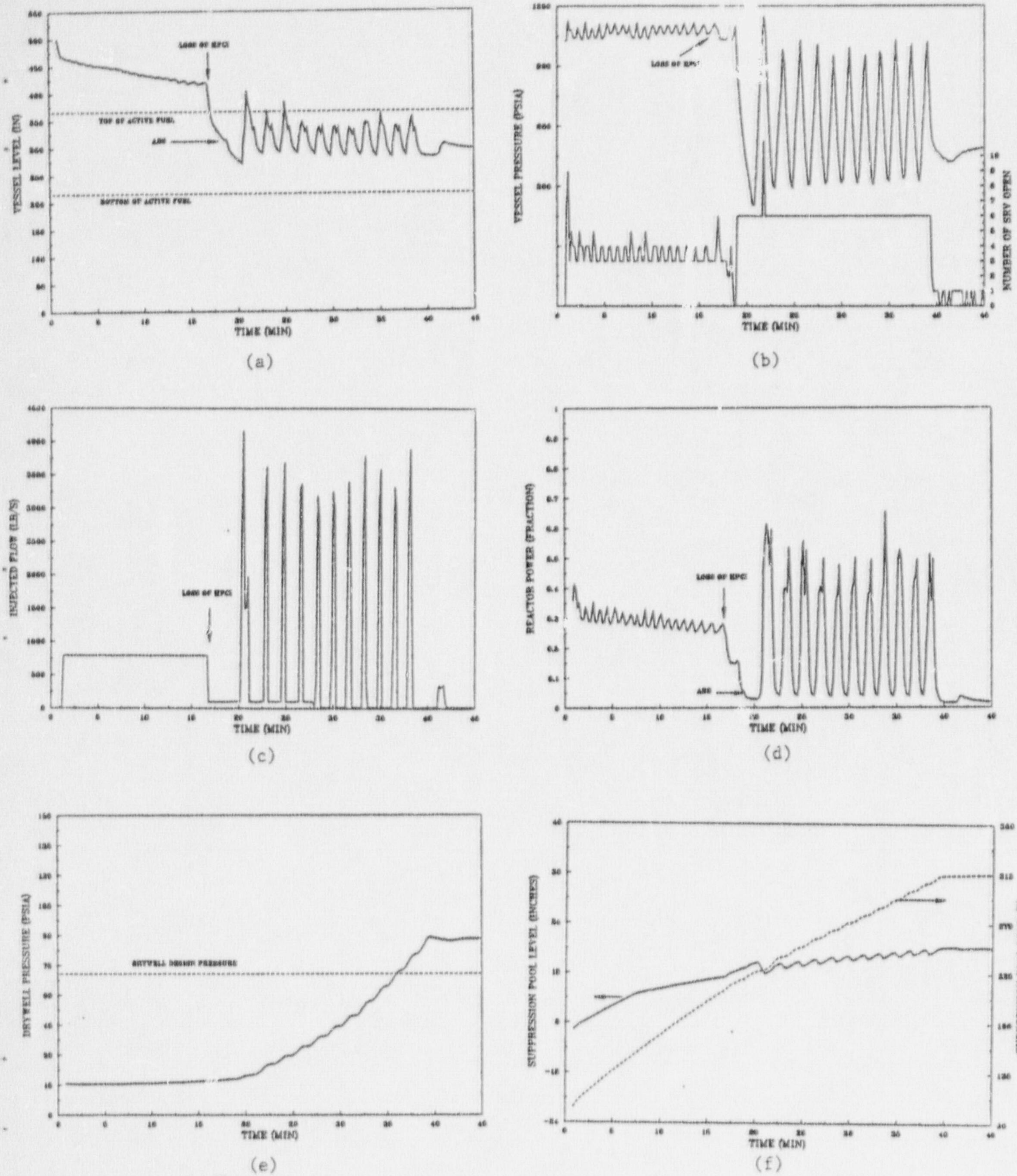


Fig. 4.1. Accident signatures for the MSIV-closure ATWS from 100% power operation, first 45 minutes with no operator action.

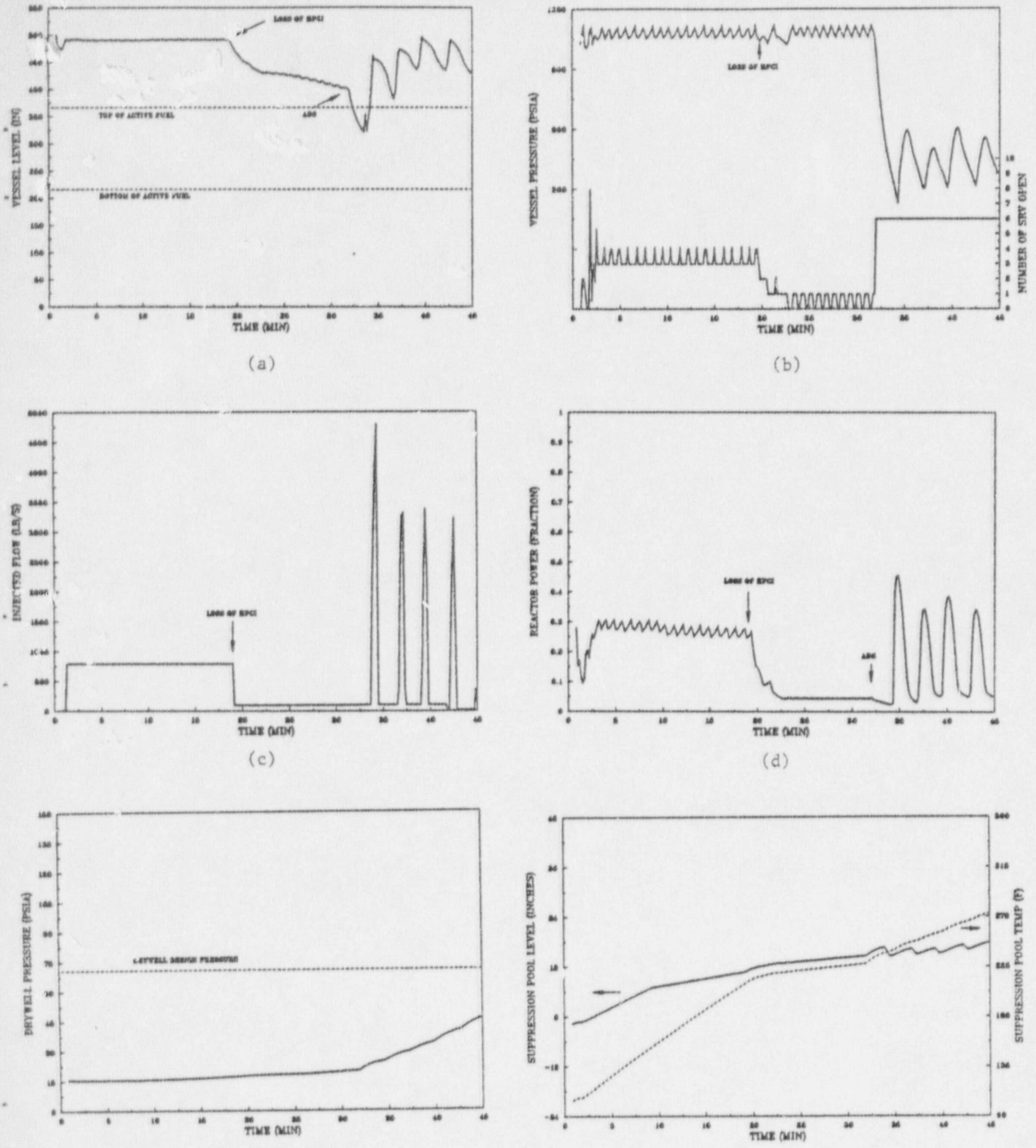


Fig. 4.2. Accident signatures for MSIV-closure ATWS from 25% power operation, first 45 minutes without operator action.

5. CALCULATIONS FOR LARGE-BREAK LOCA WITH LOSS OF ALL ECCS (AE) SEQUENCES

The large-break LOCA initiated by double-ended rupture of a 28-in. (outside-diameter) recirculation pump suction line serves as the design basis accident to establish the adequacy of the Shoreham containment with regard to drywell internal pressure, suppression chamber internal pressure, drywell floor differential pressure, suppression chamber temperature, and secondary containment pressure. The configuration of the break and the location of the reactor vessel within the containment are shown in Fig. 5.1. The response of drywell pressure, suppression chamber pressure, and drywell floor differential pressure and the associated drywell and suppression chamber atmosphere temperatures within the first 30 s after the break are shown in Fig. 5.2. All of the components of these figures were taken from the Shoreham Updated Safety Analysis Report (USAR).

At the 30-s point, ECCS injection to the reactor vessel would begin (via core spray) and the accident sequence would be brought under control and terminated without degradation of the original core geometry. Nevertheless, in the very unlikely case of failure of the ECCS function, the resulting accident sequence (termed AE) provides the shortest period of time to the onset of core degradation of any. Calculations with the BWRSAR models to estimate these times for operation of the Shoreham reactor have been performed for both 100% power operation and 25% power operation. Although the BWRSAR models were never intended to calculate the complex processes of large-break reactor vessel blowdown, their results for the first 30 s of blowdown are in good agreement with those shown in Fig. 5.2. The calculated event timing for the AE sequence taken from the BWRSAR results for both 100% and 25% power operation is provided in Table 5.1.

Less hydrogen is generated in the AE sequences than in the total-loss-of-injection sequences because less steam is retained in the reactor vessel to fuel the metal-water reactions. BWRSAR program results pertaining to hydrogen generation are provided in Table 5.2.

With recognition of the importance of the calculated reactor vessel wall temperature in estimating the time at which the remaining solid core debris could no longer be supported by the reactor vessel bottom head, a finer bottom head debris nodalization was developed and employed for the large-break LOCA calculations. This involved implementation of a small node structure near the reactor vessel wall to represent the effect of debris in close contact with the wall; the effect is to produce higher calculated wall temperatures.

The predicted distribution of core debris invessel and exvessel at the time of probable bottom head failure is provided in Table 5.3. More of the debris remains within the reactor vessel bottom head in the 25% power case, but the difference is not as marked as in the calculation for total loss of injection, which used the original debris noding structure to produce the results shown in Table 3.5. It is intended that all future calculations will use the new debris noding structure.

Table 5.1. Comparison of the timing of major events in the large-break LOCA accident sequences for initial powers of 100% and 25%

Event	Time after scram (mins)	
	100% power	25% power
MSIV closure and total loss of injection	0.0	0.0
Recirculation pump suction line break (area 4.22 ft^2)	0.002	0.002
Swollen water level falls below top of active fuel	0.05	0.06
Core plate dryout	0.09	0.10
Structural relocation of core material begins	10.9	59.5
First collapse of a core radial region	44.8	200
Second collapse of a core radial region	45.3	201
Bottom head dryout	59.2	220
High temperature weakens control rod guide tube structure and remaining portions of the core collapse	59.3	349
First failure of reactor vessel bottom head penetrations	70.4	469
Beginning of continuous pour of molten material from the reactor vessel	80.0	570

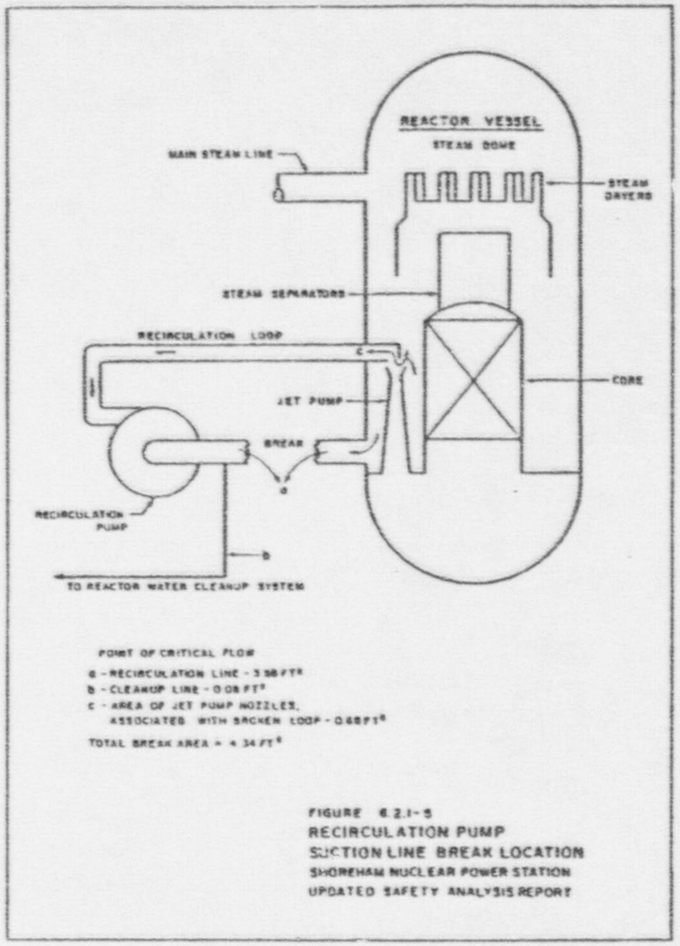
Table 5.2. Comparison of the calculated hydrogen generation parameters in the large-break LOCA sequences for initial powers of 100% and 25%

Parameter	100% power	25% power
Fraction of clad oxidized	0.38	0.35
Fraction of channel box wall oxidized	0.28	0.17
Fraction of control blade stainless steel sheath oxidized	0.08	0.04
Total hydrogen generated, lbs	1326	1157

Table 5.3. Comparison of the calculated quantities of core debris invessel and exvessel at the time that the reactor vessel wall temperature exceeds 1600°F

Debris constituent	100% power case time 200 min		25% power case time 810 min	
	Invessel (lb)	Exvessel (lb)	Invessel (lb)	Exvessel (lb)
Zr	58,600	15,700	75,200	6,400
Fe	77,900	105,300	163,900	19,800
ZrO_2	47,900	500	38,500	0
UO_2	270,400	8,400	278,800	0
Other[a]	43,300	24,700	58,500	8,800
Total	498,100	154,600	614,900	35,000

[a] Cr, Ni, B_4C, FeO, Fe_3O_4, Cr_2O_3, NiO, and B_2O_3.



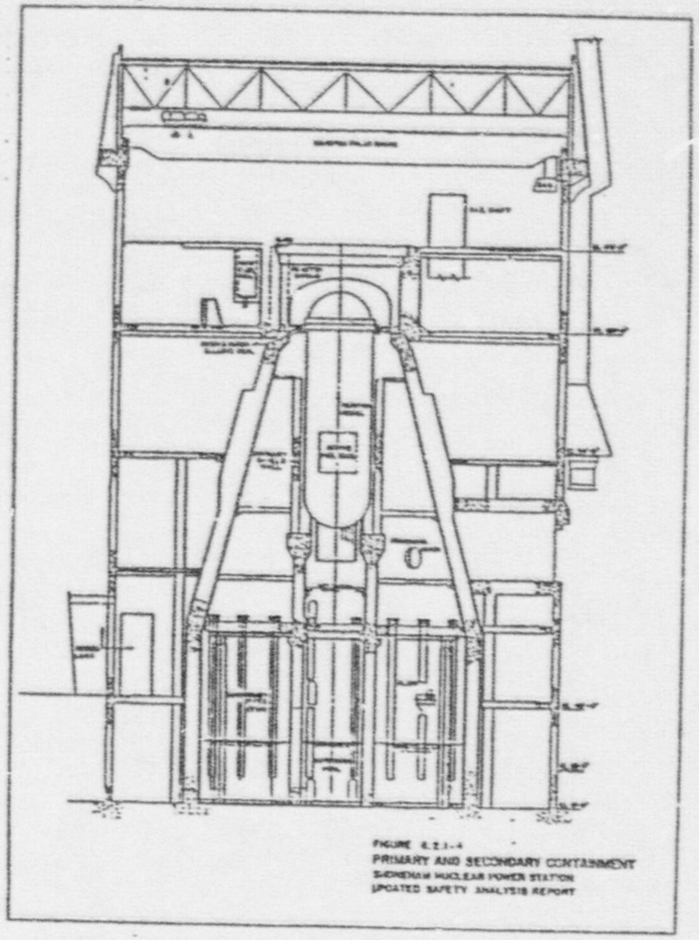
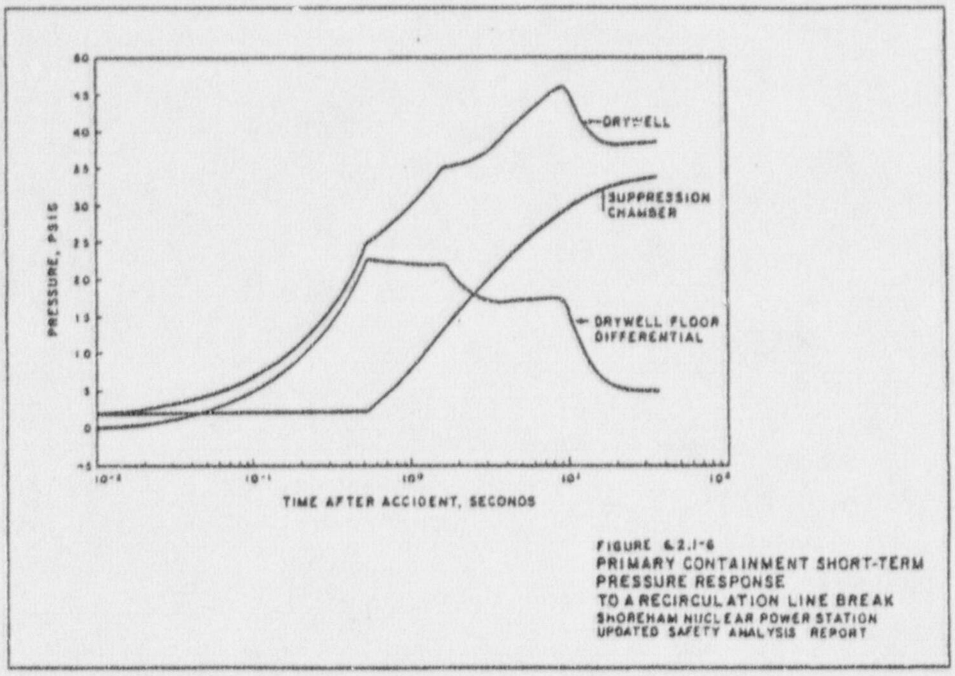


Fig. 5.1. Location of the assumed recirculation pump suction line break for the design basis accident discussed in the Shoreham USAR; the location of the reactor vessel within the containment system is also shown.



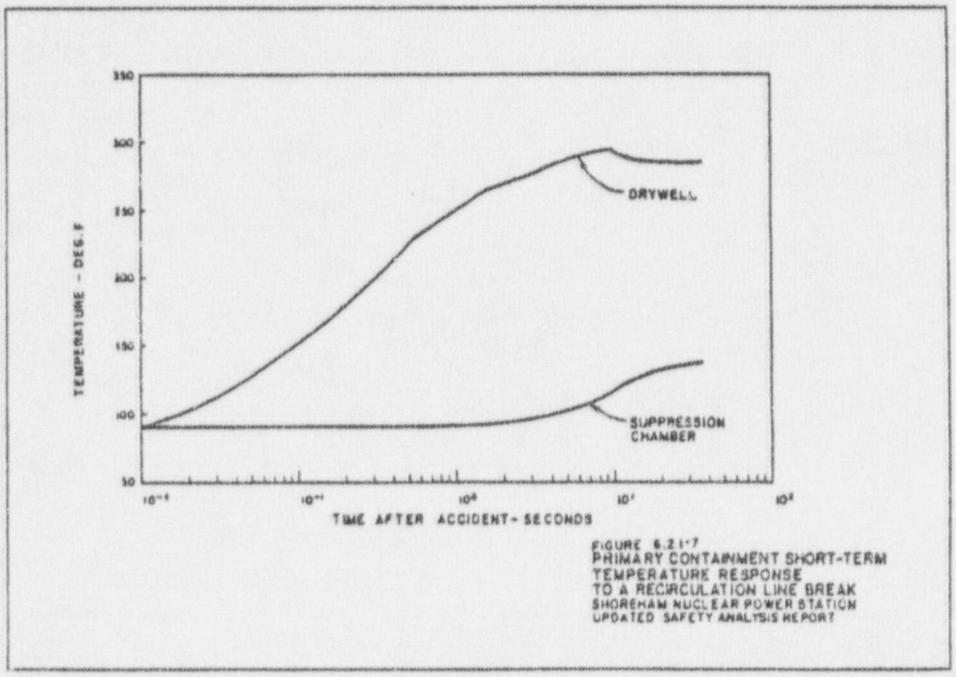


Fig. 5.2. Calculated primary containment pressures and temperatures for the first 30 s after double-ended rupture of a recirculation pump suction line. The results shown here are taken from the Shoreham USAR.

6. SUMMARY AND FINDINGS

This letter report presents the results of calculations performed by personnel of the Boiling Water Reactor Severe Accident Technology (BWRSAT) Program at Oak Ridge National Laboratory. The purpose of these calculations is to examine certain aspects of the potential for mitigation of severe accident consequences afforded by operation of the Shoreham Nuclear Power Station at 25% of rated power. The areas examined are those requested by the NRC-NRR technical sponsor of this work.

Degraded core response calculations have been performed for total loss of injection accident sequences using the Boiling Water Reactor Severe Accident Response (BWRSAR) models developed by the BWRSAT program at Oak Ridge. To establish the delay in sequence event timing afforded by the lower decay heat power levels derived from operation at 25% power, two accident sequence calculations were performed using identical input parameters except for the initial power levels, which were equivalent to 100% of rated power in one case and to 25% of rated power in the other. Both calculations were carried out well beyond the time of predicted failure of the reactor vessel bottom head penetrations so that the pouring of molten constituents of core debris from the vessel could be analyzed.

Anticipated Transient Without Scram (ATWS) calculations have been performed with the BWR-Long Term Accident Simulation (BWR-LTAS) code to examine the more sluggish reaction of a nuclear reactor to positive reactivity insertions with the control blades in the 25% power configuration as opposed to the 100% power configuration.

Other ATWS accident sequence calculations were performed with the BWRSAR program models. These calculations were based upon the C9D accident sequence described in the LILCO submission regarding the operation of the Shoreham Nuclear Power Station at 25% power. Their purpose is to examine the difference in predicted timing of severe accident events introduced by the BWRSAT program methodology (described in Table 3.1) as opposed to that used for the calculations cited by LILCO.

A final set of calculations with the BWRSAR degraded core response models was made to determine the shortest possible period of time between reactor scram and the onset of core degradation. This occurs for the very low probability large-break LOCA with total loss of ECCS (AE) accident sequence. These calculations were performed both for scram from 100% power and for scram from 25% power, and were carried out through the period of release of molten material from the reactor vessel.

The major findings derived from the efforts discussed above are as follows:

1. Given the unlikely occurrence of an accident sequence that proceeds through core degradation and reactor vessel bottom head penetration failure, there would be a significantly longer period of time between the events of the severe accident sequence following operation at 25% power than following operation at 100% power. This conclusion is based upon calculations for the total loss of injection sequence and the comparison of event timing provided in Table 3.3, as well as calculations for the large-break LOCA with loss of all ECCS sequence with its comparison of event timing provided in Table 5.1. There is no reason to believe that a similar conclusion can not be drawn for any other accident sequence for which comparison of event timing might be made.

2. About 14% additional in-core hydrogen generation is predicted for the total loss of injection accident sequence initiated from 25% power as opposed to the hydrogen generation predicted for the identical sequence initiated from 100% power. The additional hydrogen generation is primarily caused by increased oxidation of the channel box walls, which occurs during the time that they are slowly heated within the temperature range at which oxidation occurs. This matter is discussed in detail in Sect. 3.3. The generation of this additional hydrogen, while of interest, is not a critical consideration because a great deal of hydrogen (>1600 lbs) is predicted to be generated regardless of the assumed initial power.

3. Given the continuance of a severe accident sequence beyond the point at which molten core debris would begin to pour from the reactor vessel, then consideration must be given to the temperature of the reactor vessel wall in the vicinity of the bottom head. With the core debris at the decay heat power level derived from previous operation at 100% power, the temperature of the debris is predicted to increase rapidly, with the reactor vessel wall temperature lagging far behind. Under these conditions, much of the debris is predicted to have become molten and left the vessel by the time the wall temperature reaches levels at which creep rupture is expected to occur.

On the other hand, with the core debris heated by the lower decay heat power level derived from previous operation at 25% power, the debris temperature increases more slowly, and the vessel wall temperature more closely follows the debris temperature. Under these conditions, the wall temperature is predicted to reach temperatures at which creep rupture failure would be expected while a major portion of the core debris is still solid within the bottom head. Thus, the results of the calculations described in this report do not support the contention of the LILCO submission that all core debris would exit the reactor vessel in a molten state.

4. ATWS accident sequences initiated by transient-induced closure of the main steam isolation valves from operation at 25% power should not be of concern. Successful operation of the Standby Liquid Control System (SLCS) to inject neutron-absorbing poison solution would provide safe reactor shutdown. In the unlikely 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 (Ref. 16) 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% power positions, the reactor vessel water level at the top of the core, and the reactor vessel pressure at 200 psia (or below). However, if the reactor does remain critical in this configuration, then the results of the BWR-LTAS code calculations discussed in Sect. 4.2 demonstrate the sluggish reaction of the core to possible positive reactivity insertions caused by uncontrolled cold-water injection by the low-pressure ECCS systems.

5. A shadow calculation performed for the severe accident phase of the C9D ATWS accident sequence described in the LILCO submission indicates that the ORNL BWRSAT program best-estimate method for calculating degraded core and reactor vessel failure events (Table 3.1) predicts a much longer time to reactor vessel pressure boundary failure (30.8 h vs. 10.4 h) than does the MAAP code method used in calculating the results provided by LILCO. This is because of the approach taken at ORNL that the water in the reactor vessel bottom head must be boiled away before the bottom head penetration welds can become heated to failure temperatures.

6. For operation at 25% of rated power, the very earliest that core degradation might begin following the initiating event of an accident sequence is calculated to be 59.5 min. This occurs in the large-break LOCA with total loss of injection accident sequence (AE), which has a much lower probability than the loss-of-injection or ATWS sequences. Failure of the reactor vessel bottom head penetrations is not predicted to occur until 469 min (7.8 h). Additional information regarding event timing for this accident sequence is provided in Table 5.1.

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APPENDIX

Memo to: S. A. Hodge May 28, 1987

From: R. M. Harrington

Subject: Control Rod Reactivity Input for BWR-LTAS

25% Initial Power ATWS Runs

Three runs were made recently to scope the effect of a 25% initial power vs. 100% initial power on the reactor response during a hypothetical ATWS accident without operator action: a full power run, with zero for parameter DKRODS and two runs with different negative control rod reactivities. DKRODS is the amount of reactivity in the core due to control rods, and is the only parameter affecting dynamic behavior that is different for differing initial power levels. The recirculation pump speed varies with power level, but during an ATWS, the recirculation pumps are typically tripped very early on high vessel pressure (or slightly later on low vessel water level). Since BWR-LTAS is initialized 50 s after the beginning of the accident, the recirculation pumps would be tripped for all cases, so there would be no need for different input numbers expressing different recirculation flow between 100% and 25% initial power levels.

To determine what amount of reactivity to use to simulate the reactor at 25% power, the steady state version of the BWR-LTAS reactor power calculating routine was run. This program finds the steady state power level based on user input for reactor pressure, downcomer water level, jet pump pressure rise (a function of recirculation pump speed), core inlet enthalpy, and control rod reactivity. The input parameters for void and doppler coefficients are based on Browns Ferry information. Multiple runs were made to search for the combination of jet pump pressure rise (i.e. recirculation pump speed) and negative control rod reactivity that would be required to match the desired 25% core thermal power, and the 7% core outlet quality specified by Shoreham in Lap Cheng's May 18, 1987 telecopy message (Attachment 1). Other parameters such as reactor vessel pressure were also specified by Mr. Lap Cheng, and were the same for each run. The result shows that the control rods must add a negative 0.019 delta-K/K to reach the desired 25% power operating condition; the code is set up to reproduce 100% power operating conditions without any control rod reactivity. Therefore, it is concluded that a negative 0.019 delta-K/K control rod reactivity is required to reduce the reactor power from 100% to 25%.

The BWR-LTAS transient case with -0.019 control rod reactivity represents a best estimate of response with 25% initial power. In order to test the sensitivity of the results to this input, the negative control rod reactivity was decreased by 20%, to -0.015 control rod reactivity.

As mentioned above, an input of 0.0 control rod reactivity is used for the 100% initial power case.

In order to test the conservatism of the BWR-LTAS based inputs for negative control rod reactivity at 25% power, an independent estimate was based on the results of detailed steady state neutronic codes, also supplied in Attachment 1. At full power the total control rod insertion has been calculated by Shoreham to be 514 notches; at 25% power the corresponding insertion is 1536 notches. Since one notch is 1/48 of the total travel of one control rod, and there are 137 control rods, the net total control rod insertion is 7.8% at full power and 23.3% at 25% power. The total control rod worth is 24.968 dollars, or 0.18 delta-K/K; therefore, the extra negative reactivity required to decrease power from full power to 25% power is:

DKRODS = -(0.233-0.078) * 0.18 = -0.0279 delta-K/K

This shows that the -0.019 and -0.015 numbers used in the BWR-LTAS runs are well on the conservative side.

x OX TELECOPIER 495;18- 5-87; 4:23PM ; Attach 1 → 2770767;# 2

MAY 18 '87 15:27 SNPS NED P.2

Long Island Lighting Company
Nuclear Engineering Department

TELEPHONE MEMORANDUM

Call Date 5-18-87 Time 2:00 pm Incoming ______ Outgoing X

Between S.A. Hodge / R.M. Harrington of Oak Ridge National Lab.

and L. Y. Chang of NED.

Subject Response to request from ORNL.

SUMMARY: The following information were given to ORNL over the phone.

At 25% power: (609 MWt)

Core flow = 42% of rated = 32.34×10^6 lbm/hr.
Steam dome pressure = 946 psia
Feedwater enthalpy = 277 BTU/lbm (using all FW heaters)
Core exit quality = 0.07
Feedwater flow = 2.26×10^6 lbm/hr

} MAAP inputs

Scram reactivity = -\$24.968 (GEZ-6410, SNPS Transient Analysis Design Report Supplement 1, 1979)

Delayed neutron fraction = β = 0.007199 (BOL) (GEZ-6893, Shoreham Control Systems Design Report, 1979)

Control rod insertion at 100% power = 514 notches (3"/notch) (C-NFD-047)

" " " " 25% " = 1536 notches (TSO Job #07173 5/03/87)

Number of separators = 163. Downcomer submergence = 9 ft.

Equipment and floor drain lines are isolated on low reactor water level (L2) and high drywell pressure (1.69 psig). The isolation valves for these drain lines are located outside but close to the primary containment wall.

Routing:

JUV

JAR

EPS

MMB

B. Germano (NOSD)

Telephone Memoranda File