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ATWS at Browns Ferry Unit One—Accident Sequence Analysis

R. M. Harrington
S. A. Hodge

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
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SUMMARY

This study describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant 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 major concern during ATWS accident sequences.

The ATWS accident sequences have been selected for the Severe Accident Sequence Analysis (SASA) study presented in this report because these sequences have always been included among the dominant accident sequences leading to core melt identified by the BWR probabilistic risk assessments (PRAs) conducted to date. The function of a PRA is to attempt to consider all possible accident sequences at a nuclear plant using event tree and fault tree methodology for the purpose of identifying the more probable, or dominant, accident sequences. The SASA approach, on the other hand, is to examine the limited range of dominant accident sequences identified by the PRA in much greater depth than would be possible in a PRA study.

The purpose of the SASA program ATWS studies presented in this report is first, to determine the probable course of the accident progression and thereby establish the timing and the sequence of events for use in planning for the unlikely case that one of these accidents might actually occur. The important second purpose of these studies is to produce recommendations concerning the implementation of better system design and improved emergency operating instructions and methods of operator training so that the probability of severe consequences, should one of these sequences be initiated, is further reduced.

The MSIV-closure initiated ATWS accident sequence is initiated by a transient such as main steamline space high temperature or high main steam line radiation that causes MSIV closure. The reactor protection system logic is designed to recognize the beginning of MSIV closure and to produce an immediate scram, effective before the MSIVs have completely closed.* The accident sequences analyzed in this report are

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*Actually, the event of MSIV closure would result in a series of four scram signals. In order of receipt these are MSIV position <90% open, high reactor power, high reactor vessel pressure, and low reactor vessel water level.

based upon an assumption that MSIV closure is successful, but there is a complete failure of the scram function; that is, the control rods remain in the withdrawal pattern that existed before the inception of the transient. Total failure of rod movement constitutes the most severe ATWS case, but is also the most improbable of the possible scram system failures. Thus the results of this study are intended to provide an upper bounding estimate of the consequences of these very unlikely events.

As in all reactor designs, the criticality of the Boiling Water Reactor (BWR) 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.* It is easy to imagine that there must be many dangerous situations that might arise during reactor power operation that would require quick shutdown by reactor scram. However, careful review reveals that there is only one transient that might actually require control blade scram to prevent the occurrence of a Severe Accident, which by definition involves fuel damage and fission product release.

The one transient for which it is possible that only the rapid shutdown from power operation that is provided by scram could preclude severe fuel damage is a closure of all MSIVs compounded by failure of automatic recirculation pump trip. This is an "unanticipated" transient or, in other words, it is not expected to occur during the operating lifetime of the plant. Before considering the ramifications of failure of recirculation pump trip, it is instructive to examine the progression of the accident without scram but with recirculation pump trip.

*This is true even with as many as five control rods stuck in the fully withdrawn position.

During the period while the MSIVs are closing,* the reactor vessel is progressively isolated and, because the reactor is at power, the reactor vessel pressure rapidly increases. The pressure increase causes the collapse of some of the voids in the core, inserting positive reactivity and increasing reactor power, which in turn causes increased steam generation and further increases pressure. All of this happens in a matter of seconds. The cycle is interrupted when the reactor vessel pressure reaches the level of the safety relief valve (SRV) setpoints; the SRVs open to reduce the rate of pressure increase and the recirculation pumps are automatically tripped.† With the tripping of the recirculation pumps, the core flow is reduced to between 20 and 30 percent of its former value as the driving mechanism is shifted from forced circulation to natural circulation. With reduced flow, the temperature of the moderator in the core region is increased, producing voids, and introducing a significant amount of negative reactivity. The rapid increase of reactor power is terminated and the power then rapidly decreases to about 30 percent of that at normal full power operation.

If failure of installed logic caused the recirculation pumps to continue operation after the reactor vessel pressure had exceeded their trip setpoint,‡ then there are two possible outcomes. Since the total capacity of the SRVs is about 85% of normal full power steam generation, an increasing spiral of reactor power and reactor vessel pressure might continue to the point of overpressurization failure of the primary system boundary,§ inducing a large-break LOCA. On the other hand, the LOCA might be avoided because with all of the SRVs open, the loss of coolant through these valves would cause core uncovery and a concomitant reactor shutdown by loss of moderator before the pressure could reach the level necessary to cause rupture of the pressure boundary.

The question of the outcome of the extremely unlikely accident sequence involving MSIV closure followed by failure of both scram and recirculation pump trip is beyond the scope of the work presented in this report. Nevertheless, this question is being addressed within the overall scope of the ongoing NRC-sponsored SASA Program effort to study the

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*Plant Technical Specifications require that the MSIV closing time be not less than 3 nor more than 5 seconds.

†Normal operating pressure is 1020 psia (7.03 MPa). The 13 SRVs have setpoints between 1120 and 1140 psia (7.72 and 7.86 MPa). Automatic recirculation pump trip occurs when the reactor vessel pressure reaches 1135 psia (7.83 MPa).

‡It should be noted that provision is also made for automatic recircultion pump trip upon low reactor vessel water level at 470" (11.94 m) above vessel zero.

§It should be recalled that two independent protection system failures are involved here: failure of scram upon MSIV closure or high reactor vessel pressure [setpoint 1070 psia (7.38 MPa)] and failure of recirculation pump trip.

BWR ATWS. Specifically, current work at Brookhaven National Laboratory using the RAMONA code and at INEL using RELAP5 is intended to address this question.

Assuming that the recirculation pump trip does function as designed, and this will be the basis for all future discussion in this report, then it can be stated that although all transient-initiated accident sequences could most easily be brought under control and terminated by scram, 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-to-scram can be considered to be merely a nuisance requiring a more complicated and time-consuming method of achieving shutdown.

ATWS, or failure of the automatic scram function, requires that the operator manually take the actions necessary to introduce enough negative reactivity into the core to produce shutdown. The operator might do this by manual scram, in case the ATWS were caused by failure of the protective system logic. Otherwise, the operator could manually drive in the control blades, one at a time. This procedure, for the most part, involves different piping and valves than are used for scram, and therefore, although relatively slow, has a significant probability of success. In the meantime, the operators could initiate the standby liquid control system (SLCS); this system injects a neutron-absorbing solution of sodium pentaborate solution into the reactor vessel by means of positive displacement pumps.

Unfortunately, although unlikely, it is possible that manual rod insertion might also fail in the event of an ATWS. Also, the SLCS was not designed to provide a quick backup for use in an ATWS situation. The injected sodium pentaborate solution has a specific gravity of about 1.1 and is injected at a single point near the reactor vessel wall in the lower plenum. Therefore, it is expected that this heavy solution would settle in the reactor vessel lower plenum and that significant amounts would not enter the core region unless there were a large sweeping flow into the core from the lower plenum. As part of the automatic ATWS protection logic, the recirculation pumps are tripped at the inception of the accident, reducing the core inlet flow.

Studies performed in support of this report indicate that with operator action limited to initiation of the SLCS five minutes after the inception of the accident, the accident could be brought under control. Operator-provided pressure suppression pool cooling would be essential over the long term.

The recently developed 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 rods. 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 separators and back through the jet pumps in the downcomer region. The result is increased voiding in the core.*

The instructions continue: With the reactor power lowered so that the rate of pressure suppression pool heatup is relatively slow, wait until the pre-determined amount of sodium pentaborate solution necessary to achieve hot shutdown has been injected. Then, increase the rate of reactor vessel injection so that normal reactor vessel water level is restored; this action is to sweep the sodium pentaborate solution up into the core and restores natural circulation, which promotes mixing. Since the SLCS continues to inject, the reactor can subsequently be brought to cold shutdown.

The results of this study show that the instructions provided by the EPGs, if properly interpreted and implemented by the operators, would provide a satisfactory reactor shutdown and accident termination of the MSIV-closure initiated ATWS event. Nevertheless, there are three areas that require careful consideration. First, unless everything proceeds very smoothly, the operator will find that he or she is directed by the EPGs to take action to manually depressurize the reactor vessel during the period in which the reactor remains at significant power and as will be explained, this can lead to significant difficulties with plant control. Second, there might be secondary and independent equipment failures during the accident such as the occurrence of one or two stuck-open relief valves or failure of manual rod insertion or SLCS injection that would have a significant effect on the sequence of events. Third, it is difficult to extract the necessary instructions from the EPGs, even under stress-free and unlimited time situations. Each of these problem areas will be addressed in turn in the following paragraphs.

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 exceeds 165°F (347 K). This instruction is in the form of a graph of permissible maximum reactor vessel pressure vs. pressure suppression pool temperature. However, calculations performed in support of this study show that once 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.

There can be little question that manual attempts to reduce reactor vessel pressure under ATWS conditions would be extremely difficult and

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*This step also increases the temperature of the core inlet flow by uncovering the feedwater spargers through which the HPCI and RCIC systems inject, thereby restoring effective feedwater heating.

could lead to loss of operator control of the situation. Two points support these conclusions: First, an attempt to lower reactor vessel pressure would be initiated in confusion since the operator would not know which SRVs were already open when he or she attempted to take control. If the operator attempted to manually open a valve that was already open, nothing would happen. But if the operator opened a previously closed valve, the reactor vessel pressure would only drop slightly until one of the previously open valves went shut. Thus there would be only a negligible effect of operator action until the operator had manually opened as many valves as had previously been automatically open. Then, as the operator opened the next valve, the pressure would suddenly and rapidly fall because the initial pressure decrease would increase the voids in the core, reducing reactor power and steam generation and thereby further reducing the pressure. Unless the operator is very quick to shut the SRVs when the sudden pressure reduction begins, the reactor vessel pressure will drop to levels permitting vessel flooding by the low-pressure injection systems, thereby initiating very undesirable reactor power and vessel pressure fluctuations.

The second point in support of the conclusion that reactor vessel depressurization under ATWS conditions should be avoided is provided by the data provided in the steam tables, which show that the change in steam vapor specific volume for a given change in pressure is much greater at low pressures. Therefore, even if the operators were successful in smoothly lowering reactor vessel pressure, when they attempted to control pressure at the lower level, they would find that such control was impossible because of severe power and pressure oscillations.

For the case in which the operator actions are in accordance with the EPGs and all equipment operates as designed, calculations indicate that the difficulties associated with attempted manual depressurization of the reactor vessel would be avoided. This is because the power reduction obtained by the combined effects of manual rod insertion, reactor vessel water level reduction, and sodium pentaborate injection, together with the heat removal afforded by maximum pressure suppression pool cooling result in a predicted peak suppression pool temperature of only 157°F (343 K), less than the 165°F (347 K) at which reactor vessel depressurization is required by the EPGs. However, the second area associated with the EPGs that requires careful investigation involves the necessity to consider secondary equipment failures such as the occurrence of SORVs. This is particularly important for the MSIV-closure initiated ATWS accident sequence because repeated automatic cycling of SRVs can cause these valves to become stuck-open. It is shown in this study that stuck-open relief valves have little effect until the latter stages of an ATWS transient.

Consideration of other, independent, failures such as failure of manual rod insertion, failure of SLCS, and failure of pressure suppression pooling cooling is also provided in this study. Calculations indicate that the effect of failure of manual rod insertion would be to increase the peak pressure suppression pool temperature over that for the case with manual rod insertion by only 7°F (3.9 K), so the requirement for manual reactor vessel depressurization could also be avoided in this eventuality. For the case in which manual rod insertion is performed, but the SLC system does not operate, the pressure suppression pool temperature is predicted to reach 165°F (347 K) at 23 min after the inception of the accident sequence. Since the operator would begin the required depressurization at this time with the reactor vessel water level near the top of the core, a large fraction of the available reactor vessel water inventory would be vaporized during the depressurization and total core uncovery is predicted. Subsequently, the operator could restore vessel water level without core power spikes because, by this time, sufficient negative reactivity to ensure hot shutdown would have been achieved by manual rod insertion. Peak suppression pool temperature for this case is 180°F (356 K).

For the most severe (but least likely) case in which both SLC injection and manual rod insertion are failed, the operators cannot insert poison into the core, but their actions to lower the reactor vessel water level and maintain pressure suppression pool cooling would delay the ultimate overpressurization failure of the primary containment. The pressure suppression pool heat capacity limit would be exceeded in 19 min, and the operators would subsequently depressurize the reactor vessel in accordance with the EPGs, causing total core uncovery and subcriticality in the process. When the operator acts to recover the core using the low-pressure injection systems, power spikes would ensue. The subsequent accident sequence involves a series of power and pressure cycles, compounded by the fact that the manually open SRVs will close without recourse whenever the reactor vessel pressure is within 20 psi (0.138 MPa) of the drywell pressure. In the unlikely event that some form of poison injection capability is not restored in the interim, primary containment failure by overpressurization is predicted to occur 12 h after accident initiation.

The effect of one or two stuck-open relief valves upon the sequence of events for the cases previously discussed has been considered in this study. In general, the effect is small because several SRVs are open anyway during the early part of the accident sequence so that the occurrence of an SORV would not be recognized until the reactor power had been lowered to within the capacity of the stuck-open valves.

The third area associated with the EPGs that requires careful consideration of their efficacy when applied to the ATWS accident sequences involves their bases. These instructions are symptom-oriented. In other words, the operator is not expected to understand the accident sequence but is expected to respond to symptoms. This approach might be successful in dealing with a group of accidents that have similar symptoms and require similar corrective actions by the operator. But mitigation of the ATWS accident sequence requires the operator to reduce core inlet flow and to intentionally reduce the reactor vessel water level to the top of the core. This is to increase the voids in the core and thereby reduce core power and the rate of pressure suppression pool heatup and is the proper thing to do when confronted with ATWS, but no other accident sequence would require these actions.

It is the opinion of the authors of this report that the operator actions required to deal with ATWS do not fit into the envelope of operator actions required to deal with other BWR accident sequences, in which scram is effective. We believe that the symptom-oriented procedures for operator control of BWR accident sequences should be limited to situations in which reactor scram is successful. Separate procedures should be developed for ATWS control.

ATWS AT BROWNS FERRY UNIT ONE - ACCIDENT SEQUENCE ANALYSIS

R. M. Harrington
S. A. Hodge

ABSTRACT

This study describes the predicted response of Unit One at the Browns Ferry Nuclear Plant to a postulated complete failure to scram following a transient occurrence that has caused closure of all Main Steam Isolation Valves (MSIVs). This hypothetical event constitutes the most severe example of the type of accident classified as Anticipated Transient Without Scram (ATWS). Without the automatic control rod insertion provided by scram, the void coefficient of reactivity and the mechanisms by which voids are formed in the moderator/coolant play a dominant role in the progression of the accident. Actions taken by the operator greatly influence the quantity of voids in the coolant and the effect is analyzed in this report. The progression of the accident sequence under existing and under recommended procedures is discussed. For the extremely unlikely cases in which equipment failure and wrongful operator actions might lead to severe core damage, the sequence of emergency action levels and the associated timing of events are presented.

1. INTRODUCTION

This is the fifth report in a series of accident studies concerning the BWR 4 - MK I containment plant design.* These studies have been conducted by the Severe Accident Sequence Analysis (SASA) Program at Oak Ridge National Laboratory with the full cooperation of the Tennessee Valley Authority (TVA), using Unit 1 at the Browns Ferry Nuclear Plant as the model design. The SASA Program is sponsored by the Containment Systems Research Branch of the Division of Accident Evaluation within the Nuclear Regulatory Research arm of the Nuclear Regulatory Commission. The purpose is to determine the probable course of each of a series of severe accidents so as to establish the timing and the sequence of events; this information would be of use in the unlikely event that one of these accidents might actually occur. These studies also

*Previous reports concern Station Blackout (NUREG/CR-2181), Scram Discharge Volume Break (NUREG/CR-2672), Loss of Decay Heat Removal (NUREG/CR-2973), and Loss of Injection (NUREG/CR-3179) accident sequences.

provide recommendations concerning the implementation of better system design and better emergency operating instructions and operator training to further decrease the probability of such an event.

The Browns Ferry Nuclear Plant is located on the Tennessee River between Athens and Decatur, Alabama. Each unit of this three-unit plant comprises a Boiling Water Reactor (BWR) steam supply system designed by the General Electric Company with a maximum power authorized by the operating license of 3293 MW(t) or 1067 net MW(e). The General Electric Company and the TVA performed the construction. Unit 1 began commercial operation in August 1974, followed by Unit 2 in March 1975, and Unit 3 in March 1977. The primary containments are of the Mark I pressure suppression pool type and the three units share a secondary containment of the controlled leakage, elevated release design. Each unit occupies a separate reactor building located in one structure underneath a common refueling floor.

This report presents a study of the predicted sequence of events during a postulated Anticipated Transient Without Scram (ATWS) accident sequence at Unit 1 of the Browns Ferry Nuclear Plant. This accident category was selected for analysis because it has been identified as a dominant contributor to the overall calculated core melt frequency in every BWR Probabilistic Risk Assessment (PRA) performed to date.* By definition, the ATWS accident sequence involves failure-to-scram following an anticipated plant transient† that would normally result in a scram. Since there are a large number of anticipated transients that might be used as the initiating event for the ATWS accident sequence, it was important to the efficacy of this study to select the transient leading to the most severe consequences. The subject of possible initiating events is discussed in Chap. 2, where the Main Steam Isolation Valve (MSIV) closure transient is selected for major emphasis in this report.

Previous SASA studies have shown that the determination of the effect of operator actions upon the progression of an accident sequence is facilitated if the accident sequence of events is first established for the case without operator action. This procedure is also followed in this study and the MSIV-closure initiated ATWS accident sequence without operator action is the subject of Chapter 3.

The effects of possible operator actions in both mitigation and exacerbation of the MSIV-closure initiated ATWS is discussed in Chap. 4. The basic principles of reactivity control, reactor vessel level and pressure control, and pressure suppression pool temperataure control are explained in Sect. 4.1, together with a description of the associated

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*See, for example, The Reactor Safety Study (WASH 1400) and the Interim Reliability Evaluation Program (IREP) analysis for Browns Ferry Unit 1 (NUREG/CR-2802). ATWS has also been identified as a dominant contributor in the PRAs that have been conducted for BWRs of advanced design.

†An anticipated transient is a transient event that is expected to occur at least once during the plant operating lifetime.

plant instrumentation and control equipment and operating procedures. The progression of the accident sequence in which the plant operators follow the BWR Owner's Group Emergency Procedure Guidelines exactly is discussed in Sects. 4.2 and 4.3. Consideration of the effect of equipment failures including stuck-open relief valves and the loss of pressure suppression pool cooling is provided in Sect. 4.2 and the consequences of failure of manual control rod insertion or the sodium pentaborate injection function of the Standby Liquid Control System (SLCS) are discussed in Sect. 4.3.

For the extremely unlikely case in which manual rod insertion, sodium pentoborate injection, and pressure suppression pool cooling all fail, the accident progresses to the point of severe core damage. The emergency action levels and timing for this sequence are discussed in subsection 4.3.4.

This study has produced some new insights into the important physical phenomena controlling the plant response to an MSIV-closure initiated ATWS accident sequence. Recommendations concerning mitigating operator actions are provided in Sect. 5.1 of Chap. 5. The sequence of events for the case with successful SLC system operation but without other operator actions is presented in Sect. 5.2. The effect of failure of both manual control rod insertion and the poison injection function of the SLCS is discussed in Sect. 5.3 and the effect of stuck-open relief valves is described in Sect. 5.4.

The uncertainties involved in the calculational model and the uncertainties associated with the assumption of operator actions are discussed in Chap. 6.

The implications of the results of this study are described in Chap. 7. The discussion includes an evaluation of the available instrumentation, the level of operator training, the emergency procedures, and the overall system design from the standpoint of adequacy for use in the mitigation of this accident.

The computer code BWR-LACP developed by R. M. Harrington at ORNL to model operator actions and the associated primary system and containment response during the period before permanent core uncovery in accident sequences at Browns Ferry has been used in all previous SASA studies and was also applied to this study. Primary system calculations for the portion of a severe accident sequence before core uncovery are much simpler for a BWR than for a PWR. The low reactor vessel water level that is common to all BWR severe accident sequences would ensure that the reactor vessel is isolated and that the recirculation pumps would be tripped; thus the core inlet flow would be a function only of the amount of makeup water injection and the effect of natural recirculation circuits within the reactor vessel. Therefore, sophisticated primary system analyses codes such as RELAP5, RETRAN, or TRAC are usually not necessary for BWR severe accident calculations; fundamental modeling of the processes within the reactor vessel in a properly benchmarked relatively simple code such as BWR-LACP is sufficient. Appendix A provides a description of the additions and improvements made to BWR-LACP to provide the special capabilities needed for ATWS calculations and includes a discussion of the benchmarking calculations performed to demonstrate the adequacy of the code.

Depending on the parameter that is known, the calculation of steady state power under ATWS conditions can be either a very simple or a very complicated procedure. It is shown in Appendix B that if the injection rate to the reactor vessel is specified, then the steady state power can be determined by a simple hand calculation. Conversely, if the reactor vessel water level is specified, then the power calculation is much more complicated.

Appendix C was prepared by the Reliability and Human Factors group at ORNL. Their review provides a preliminary assessment of human factors problems related to BWR ATWS and includes an analysis of critical operator actions following the Emergency Procedures Guidelines. The work reported in Appendix C has several cross-references to discussions in the main body of this report.

A listing of acronyms and symbols used in the report is provided, with definitions, in Appendix D.

The primary sources of plant-specific information used in the preparation of this report were the recently issued updated version of the Browns Ferry Nuclear Plant (BFNP) Final Safety Analysis Report (FSAR), the USNRC BWR Systems Manual, the BFNP Hot License Training Program Operator Training Manuals, the BFNP Unit 1 Technical Specifications, the BFNP Emergency Operating Instructions, and various other specific drawings, documents, and manuals obtained from the Tennessee Valley Authority. The experience gained from two plant visits in connection with previous studies and from three working visits to the Browns Ferry Control Room simulator for the modeling of ATWS accident sequences was also applied in this effort.

The setpoints for automatic equipment response used in this study are the actual setpoints specified for instrument adjustment at the plant. These setpoints are established so as to provide margin for the known range of instrument error and therefore differ slightly (in the conservative direction) from the currently established Technical Specification limits.

This study could not have been conducted on a realistic basis without the current plant status and extensive background information provided by the Tennessee Valley Authority. The assistance and cooperation of TVA personnel at the Browns Ferry Nuclear Plant, at the Power Operations Training Center, and at the Engineering Support Offices in Chattanooga and Knoxville are gratefully acknowledged.

2. INITIATING EVENTS

In the United States, nuclear reactor plants are not licensed unless their design includes explicit provision for safe recovery to normal conditions from each of the operating transients that might reasonably be expected to occur at least once during the lifetime of the plant. These expected and designed-for transients are termed "anticipated transients." It is the purpose of this report to examine the effect of loss of the protective scram function upon the outcome of accident sequences initiated by anticipated transients. Such complicated sequences have been considered before and are commonly classified as "Anticipated Transients Without Scram (ATWS)."*

Loss of the scram function might be caused by protection system sensor or other electrical/logic failures, by mechanical failure of the control rod drive hydraulic system or by disruption of the alignment of the control rod drive mechanism assemblies. By whatever means, failure of the scram function is very unlikely. A recent report[2.1] by staff members of the Division of Systems Safety, U.S. Nuclear Regulatory Commission (NRC) provides the estimate that "the probability of the rods failing to insert when called upon is approximately 3×10^{-5} per demand, . . . neglecting the difference between PWRs and BWRs."

Power operation of the Browns Ferry Plant involves control rod patterns that range from a minimum of one-half of the rods withdrawn to, at the end of core life, all of the rods fully withdrawn. As an example, Fig. 2.1 illustrates the middle-of-life control rod pattern used at the Browns Ferry control room simulator and Table 2.1 summarizes its characteristics. The reader should note the symmetry of the pattern.

Given the occurrence of an anticipated transient, the severity of loss of the scram function might vary from a partial ATWS, in which some of the withdrawn control rods insert normally in response to the scram signal but at least one does not,† to a full ATWS in which none of the withdrawn control rods move at all. All of the ATWS accident sequences considered in this report assume the most severe case: a full ATWS with all control rods retained in their normal 100% power operating position after imposition of the scram signals.

Because there are a large number of anticipated transients, it is important to identify those for which a concomitant failure to scram leads to the most severe consequences. Fortunately, the task of separating ATWS sequences into categories of severity has been recently

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*The low probability of the occurrence of a failure to scram combined with the low probability of an unanticipated transient makes the probability of the combination of these independent events too small to be considered.

†Actually, failure of 1 rod to insert does not constitute an ATWS and is not an uncommon event. Conservative GE calculations show that a failure of insertion of five closely grouped control rods might cause local fuel damage.

completed in a separate study[2.2] conducted by the General Electric Company (GE). The basis for that study is discussed in Sect. 2.1 and study results are summarized in Sects. 2.2 through 2.4.

2.1 Systems for Mitigation of ATWS

In February 1979, the NRC staff requested that GE conduct a study to document the response during an ATWS accident sequence of the existing BWR plant designs assuming that they were fitted with the proposed ATWS mitigation systems then under consideration. The resulting study (Ref. 2.2) includes an analysis for a BWR 4 MK I design (representative of, but smaller than, that of the Browns Ferry plants) for which the mitigation systems listed in Table 2.2 were assumed installed and operative. It should be noted that no existing plant has all of these features and only the first item, the recirculation pump trip (on high reactor vessel pressure), is installed at Browns Ferry.

Recirculation pump trip provides an automatic and rapid conversion of core flow from forced circulation to natural circulation. At Browns Ferry, protection against reactor vessel overpressurization during a ATWS accident sequence in which the immediate effect is an increase in primary system pressure is provided by the tripping of the breakers feeding the motor ends of both recirculation pump motor-generator sets on high reactor vessel pressure at 1120 psig (7.82 MPa).* This provides a rapid reduction in core flow as the motor-generator sets coast down, increasing core voids and thereby inserting a large amount of negative reactivity and reducing core power.

Although installed for reasons other than ATWS mitigation, a second recirculation pump trip system available at Browns Ferry would serve to reduce the severity of the power excursion following an ATWS initiated by closure of the main turbine stop valve or by fast closure of the turbine control valves. Circuit breakers located between the generator end of each recirculation pump motor-generator set and the associated recirculation pump motor are automatically opened, provided main turbine first stage pressure corresponds to 30% rated load or larger, within 175 milliseconds of the beginning of turbine stop valve closure or turbine control valve fast closure. The resulting decrease in core flow and increase in core voiding provides an anticipatory reduction of core power in the event of main turbine trip or generator load rejection transients.

The second of the proposed ATWS mitigation systems listed in Table 2.2, alternate rod insertion, will be required for BWRs by forthcoming amendment to 10 CFR 50 but is not currently installed at Browns Ferry. This system, whose exact design has not been specified, will provide a parallel path for actuation of the scram valves and scram discharge volume vent and drain valves, as necessary for control rod insertion. This

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*Reactor vessel overpressurization protection is provided by 4 SRVs set at 1105 psig, 4 set at 1115 psig, and 5 set at 1125 psig.

will be accomplished by the addition of redundant venting valves on the scram valve pilot air headers.

The third item listed in Table 2.2 concerns the rate at which the liquid neutron poison (sodium pentaborate) can be injected into the reactor vessel and whether or not the injection is initiated automatically. At Browns Ferry, the poison solution would be injected at the rate of 56 gpm (0.0035 m^3/s) after manual initiation* of the standby liquid control (SLC) system. It is expected that a future amendment to 10 CFR 50 will require an increased poison injection rate capability for Browns Ferry, either by an increase of the pump capacity to 86 gpm (0.0054 m^3/s) or by an increase of the poison concentration in the injected solution. It is not expected that automatic SLC system actuation will be required for Browns Ferry, although this feature is being provided for several BWRs currently under construction and will be required as a condition for the issuance of future BWR construction permits.

The fourth of the proposed ATWS mitigation systems listed in Table 2.2 concerns the adoption of an improved liquid poison injection device. The need for this can be appreciated by an examination of Fig. 2.2, which shows the existing mechanism, a single injection sparger with horizontal discharge beneath the core plate. (During normal operation, the sparger acts as one of the sensing taps in a system designed to measure the differential pressure across the core plate.) A comparison of Fig. 2.2 and Fig. 2.3 underscores the remote and decidedly unsymmetric location of the injection sparger. This, combined with the high specific gravity of the injected solution (about 1.1), prevents a uniform dispersal of the injected poison upward into the core region unless there is a core inlet flow sufficient to induce turbulent mixing in the reactor vessel lower plenum. On the other hand, a high inlet flow would provide forced circulation to the core and might induce prohibitively high core power during the period before enough poison had been injected to have significant effect.

Various new means of liquid poison injection have been proposed to provide symmetry of poison entrance such as injection through the instrument sensing lines into the throats of the reactor vessel jet pumps. One proposal that accomplishes this goal and at the same time overcomes the disadvantage of the higher specific gravity of the injected solution is to inject through the existing core spray spargers, which are circular and located in the upper plenum (see Fig. 2.3) above and around the outer edge of the core. The latter concept is incorporated in the Limerick and other recent plant designs. No change in the existing design is contemplated for the Browns Ferry plants.

The fifth and final proposed ATWS mitigation system listed in Table 2.2 is automatic feedwater pump runback. Upon a combination of high reactor vessel pressure and sustained high core power, this proposed system would automatically reduce feedwater flow and thereby reduce core power. This system is provided for some late model BWRs, but is not a required ATWS mitigation feature. At Browns Ferry, the feedwater pumps are steam-turbine driven and would therefore be automatically shut down

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*There is no automatic initiation.

if the initiating event for an ATWS were MSIV closure. For other initiating events such as main turbine trip in which the MSIVs remain open, this system, if installed, might have significant effect on the outcome. (The EPGs direct the operators to take manual action to terminate feedwater injection.)

2.2 Sequence Selection

The GE study[2.2] of ATWS with proposed mitigation features for the BWR 4 MK I design cannot be considered directly applicable to the Browns Ferry units because most of the assumed mitigation systems are not installed, as explained in Sect. 2.1. Nevertheless, the study does establish that the severity of all ATWS transients is bounded by failure-to-scram accident sequences initiated by (1) MSIV closure, (2) turbine trip, or (3) an inadvertently-open relief valve (IORV) during power operation.

The results of the GE study for the case of the IORV-initiated ATWS are summarized in Table 2.3. By procedure, the operator initiates pressure suppression pool (PSP) cooling when the pool temperature reaches 95°F (308 K) and attempts a manual scram when the pool temperature reaches 110°F (316 K). The control rods fail to insert; this begins the ATWS phase of the accident sequence.

The plant status during the IORV-initiated ATWS sequence is schematically illustrated in Fig. 2.4. The reactor is at power, steaming both to the pressure suppression pool through the open SRV and to the main turbine, which is continuing to drive the generator and produce electricity. The feedwater (FW) pumps continue to supply water to the reactor vessel, but the water mass lost from the primary system to the pressure suppression pool must be replaced. The makeup water comes from the condensate storage tank (CST) both through the control rod drive (CRD) hydraulic system and via vacuum drag through the standpipe in the CST to the main condenser hotwell.

Sensing the failure of the manual scram, logic initiates the SLC system timer and the SLC pumps start automatically 2 min later. The sodium pentaborate begins to enter the core after 30 s and, as a result, reactor power begins to slowly decrease. The relief valve capacity is equivalent to 6.5% of full reactor power; therefore, the power delivered to the main turbine is the difference between reactor power and 6.5% of reactor power. As the reactor power decreases, the turbine control system will automatically reduce the turbine steam demand (and consequently, the amount of generated electricity) as necessary to maintain reactor pressure in the normal operating range.

When enough sodium pentaborate has been injected to reduce the reactor power to below 6.5%, the main turbine is completely unloaded and all steam flow is to the pressure suppression pool. Since the capacity of the open relief valve is greater than the steam supply being generated, reactor vessel pressure decreases. The main steam isolation valves automatically shut when the pressure has decreased to 800 psig (5.62 MPa), causing loss of the FW pumps, which are turbine-driven. The reactor vessel water level decreases, causing trip of the recirculation pumps at a reactor vessel water level of 470 in. (11.94 m) above vessel zero. This converts core flow from forced to natural circulation which has the effect, with the amount of sodium pentaborate that has been injected up to this time, of reducing the core power to decay heat levels.

The pressure suppression pool temperature continues to increase during the final phase of the accident sequence because the decay-heat generated steam continues to be condensed in the pool. Peak pool temperature [183°F (357 K)] is reached about 1.5 h after the inception of the accident.

The IORV-initiated ATWS sequence does not threaten primary containment integrity because the pressure suppression pool cooling provided by the residual heat removal (RHR) and residual heat removal service water (RHRSW) systems is nearly equal to the heat load introduced to the pool through the open SRV.* Should the study of the IORV-initiated ATWS be repeated specifically for Browns Ferry, there would be differences in event timing because at Browns Ferry there is no automatic SLC system, the setpoint for MSIV closure on low primary system pressure is slightly lower, and there are other differences of plant design that would have a small effect on the results.† Nevertheless, the operators would, by procedure, manually initiate the SLC system and the general outcome of the sequence would be the same (i.e., no threat to containment). Therefore, the IORV-initiated ATWS sequence will not be further considered in this report.

The outcomes of the two other bounding ATWS accident sequences identified by the GE study are expected to be significantly affected by the equipment differences between Browns Ferry and the model plant assumed in the study. The MSIV closure-initiated ATWS sequence is described in Sect. 2.3 and the turbine trip-initiated sequence is discussed in Sect. 2.4.

2.3 Main Steam Isolation Valve Closure — ATWS

The results of the GE study for the case of the MSIV-closure initiated ATWS are summarized in Table 2.4. With the MSIVs shut, all steam generated by the at-power reactor is conveyed into the pressure suppression pool through as many relief valves as are necessary to pass the steam. The plant status is schematically illustrated in Fig. 2.5. The fact that this accident sequence involves multiple SRV discharge into the pressure suppression pool over an extended period of time makes it

*This is the equivalent of 6.5% reactor power until time 24 min, as indicated in Table 2.3.

†The improved sodium pentaborate injection points assumed in the GE study have little effect in this accident sequence because of the large core inlet flow provided by the continued operation of the recirculation pumps.

the most severe of the three bounding ATWS accident sequences identified in the GE study.

As indicated in Table 2.4, the MSIVs require about 4 s to close. As the valves close, automatic reactor scram fails and reactor vessel pressure increases, causing void collapse in the core and thereby inserting positive reactivity. Core power increases rapidly, causing more pressure increase and the opening of all reactor vessel relief valves. The recirculation pumps trip on high reactor vessel pressure [1120 psig (7.82 MPa)] about 5 s after the beginning of MSIV closure, converting core flow from forced to natural circulation. The reduced core flow immediately causes an increased temperature of the water moderator in the core and consequently, increased voids and the insertion of negative reactivity. Reactor power decreases and some of the SRVs close, stabilizing reactor vessel pressure at the relief valve setpoint [about 1120 psig (7.82 MPa)]. Feedwater flow reaches zero shortly thereafter since the steam supply to the feedwater turbines is lost when the MSIVs are shut.

Without feedwater, the reactor vessel water level decreases until the HPCI and RCIC pumps are automatically actuated. It is important to recognize that from this time on, the HPCI and RCIC pumps act as feedwater pumps and that their combined rate of injection determines the reactor power. That this is so can be shown by the following argument: If the reactor power expressed as a percentage of full reactor power is greater than the total injection flow (HPCI, RCIC, and CRD hydraulic system) expressed as a percentage of full feedwater flow, then the mass flow of steam being generated is greater than the mass of water being injected, and the reactor vessel water level will decrease. Decreasing reactor vessel water level causes increased voiding in the core, inserting negative reactivity and reducing reactor power. Conversely, if the mass rate of water injection exceeds the mass rate of steam generation, then reactor water level will increase so that there are fewer voids in the core, inserting positive reactivity and increasing reactor power. Thus the rate of injection by the HPCI and RCIC systems will determine the reactor power in the MSIV-closure initiated ATWS accident sequence.

The term "reactor power" used in the preceding paragraph should be understood to mean the steaming rate from the reactor vessel expressed as a percentage of the steaming rate at normal full power operation. Since the combined injection flow of the HPCI, RCIC, and CRD hydraulic systems is 2.846×10^6 lb/h (358.6 kg/s), then the steaming rate from the reactor vessel for stable reactor vessel water level would also be 2.846×10^6 lb/h (358.6 kg/s) or 21.27% of that at normal full power operation. However, the core thermal power would be higher. This is because the HPCI and RCIC systems inject relatively cold water from the condensate storage tank whereas under normal operating conditions, the feedwater is heated. Thus a significant amount of the total core power under MSIV-closure initiated ATWS conditions would be expressed as sensible heat, raising the temperature of the injected water and not directly contributing to steam generation. The core thermal power, reactor power, and the flows that would produce a stable reactor vessel level at Browns Ferry are indicated on Fig. 2.5. Note that reactor power is 21.27%, while core thermal power is about 28%. (This discussion is presented in more detail in Appendix B.)

For the model plant and pertinent equipment assumed for the GE study, sodium pentaborate solution would begin entering the core 3 min after the initiating event (Table 2.4) and the reactor would be in hot shutdown 17 min after the beginning of MSIV closure. These results lean heavily upon the assumption of improved sodium pentaborate injection points so that the injected solution is readily introduced into the core. For the MSIV-closure initiated accident sequences, in which the recirculation pumps are almost immediately tripped, the core inlet flow is much reduced and dramatic operator actions to properly manage the accident must be taken for plants such as Browns Ferry which have the sodium pentaborate injection sparger shown in Fig. 2.2. Thus the results of the GE study beyond the first 3 min of the MSIV closure-ATWS sequence cannot be considered applicable to Browns Ferry.

The work documented in this report is plant-specific, and concentrated upon the MSIV-closure initiated ATWS sequences for Browns Ferry Unit 1. This is because there is no question that these ATWS sequences, in which all reactor power is deposited into the pressure suppression pool, pose the greatest challenges to containment integrity. As is shown in the following chapters of this report, the operator must take action since the case without operator action (Chap. 3) results in early loss of containment and probable severe core damage. On the other hand, the potential for harmful operator action is high, as discussed in Chaps. 4, 5, and 7.

2.4 Turbine Trip — ATWS

The ATWS initiated by main turbine trip is the third of the three ATWS accident sequences that bound the severity of ATWS accidents as identified by the GE study. The results of the GE study for the first 45 s of this accident sequence are summarized in Table 2.5. Discussion of the GE results is not carried further here because the assumption of quick feedwater injection runback to zero plays such a large role in the outcome and Browns Ferry and similar plants do not have it.

It should be understood that the level of core power in the turbine trip initiated ATWS is established in a totally different way than in the MSIV-closure initiated ATWS. In the turbine trip initiated ATWS, the feedwater pumps continue to function and are automatically adjusted so as to maintain reactor vessel water level in its normal operating range. Thus reactor vessel water level is approximately constant and does not play a role in causing variation of core power. Recirculation pump trip occurs early in this accident sequence, reducing core inlet flow to that induced by natural circulation; this reduces core power to about 30%.

It is interesting to note that although the main turbine stop valve closes in 0.1 s, the GE study results show that the resulting reactor power excursion* is much less severe than the excursion that occurs when the MSIVs are shut with a closing time of 4 s. The reason is that very

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*The result of pressure increase and void collapse in the core.

significant damping of pressure pulses is provided by the long run of very large piping between the reactor vessel and the main turbines.*

A steady-state balance of flows is shown for the turbine trip-ATWS accident sequence in Fig. 2.6, in which the central assumption is that the core power would be 30% under natural circulation conditions (recirculation pumps tripped and normal reactor vessel water level maintained by the feedwater control system). Makeup water to the primary system to replace the mass lost by steam relief into the pressure suppression pool is provided by a combination of vacuum drag into the main condenser hotwell and CRD hydraulic system injection.

It should be noted that the initial core thermal power reduction to 30% would not be maintained. The feedwater heaters are fed by steam extracted downstream of the turbine stop valve and therefore feedwater heating would be lost after stop valve closure. This would increase the core thermal power but would not affect the power flow from the reactor vessel. (See the discussion in Appendix B.)

As in the case of the IORV-initiated ATWS discussed in Sect. 2.3, the turbine trip-ATWS is less severe than the MSIV-closure initiated ATWS because most of the steam generated within the reactor vessel is passed to the main condensers instead of to the pressure suppression pool. At Browns Ferry, the turbine bypass valves can pass up to 25% of rated steam flow and the feedwater turbines take another 0.5%. The mass flow balance based on these assumptions is shown in Fig. 2.6 (lb/h and percent of full-power flows). There is, however, a related problem discussed in the GE study: unstable pressure fluctuations between the reactor vessel and the main turbine bypass valve control system are expected to develop; these pressure fluctuations would result in large swings of core void collapse and power increase.

The turbine trip-initiated ATWS accident sequence is not further addressed in this report. The reason for this is that it is believed to have less severe consequences than those of the MSIV-closure initiated ATWS for plants such as Browns Ferry. It should also be noted that severe core damage cannot occur unless the core is uncovered and this would convert the turbine trip-ATWS into an MSIV closure-ATWS because low reactor vessel water level causes MSIV closure.

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*Draft report review comment by Lowell Claasen of GE: "Codes that have been modified to include pressure wave effects tend to give results with a higher neutron flux peak on turbine trips than for MSIV closures. Because this power surge is of extremely short duration, however, the heat flux peaks for turbine trips remain lower than for MSIV closures."

References for Chapter 2

2.1 Staff Report, *Anticipated Transients Without Scram for Light Water Reactors*, NUREG-0460, Vol. 1, p. 29 (April 1978).

2.2 General Electric Company, *Assessment of BWR Mitigation of ATWS, Volume II (NUREG-0460 Alternate No. 3)*, NEDO-24222 (February 1981).

Table 2.1. Summary description of middle-of life control rod pattern illustrated in Fig. 2.1

Number of rods	Notch position	Inches withdrawn
140	48	144
8	42	126
8	24	72
4	20	60
4	04	12
21	00	0

Table 2.2. Proposed systems for ATWS mitigation in BWRs

1. Recirculation pump trip
2. Alternate rod insertion
3. Automatic two-pump standby liquid control system
4. Improved standby liquid control system injection points
5. Automatic feedwater pump runback

Table 2.3. Results of a GE study (Ref. 2.2) of the progression of an IORV-initiated ATWS at a BWR 4 MK I containment plant

Event	Time	
	(s)	(Min)
IORV	0	
PSP temperature reaches 95°F: Alarm sounds, operator initiates suppression pool cooling	120	2
PSP temperature reaches 110°F: Manual scram (fails). Timed SLC logic initiated	450	7.5
SLC system automatically starts[a]	570	9.5
Sodium pentaborate reaches core	600	10
Power less than relief valve capacity (6.5%); pressure decreases more rapidly so turbine control valves completely shut		24
MSIVs shut when pressure reaches 800 psig.[b] FW pumps lost		28
Low water level trip (470 in.) of recirculation pumps; HPCI/RCIC start		33
Peak containment temperature and pressure		95

[a]Automatic S[illegible] system not available at Browns Ferry.

[b]825 psig at Browns Ferry.

Table 2.4. Results of a GE study (Ref. 2.2) of the progression of an MSIV-closure initiated ATWS at a BWR 4 MK I containment plant

Event	Time (s)	Time (Min)
MSIVs start to close	0	
MSIVS fully closed, SRVs lift, maximum neutron flux (527%)	4	
RPT, timed SLC logic triggered, maximum heat flux (143%)	5	
RPV pressure (vessel bottom) peaks at 1296 psig	9	
SRVs start to close and pressure stabilizes at relief valve setpoint	20	
Feedwater flow reaches zero (FW runback[a])	23	
HPCI/RCIC actuated when level reaches level 2 (470 in.)	43	
HPCI/RCIC injection starts	63	1
ATWS timer complete,[a] SLCS starts	125	2
Sodium pentaborate solution enters reactor vessel	180	3
Water level reaches minimum (389 in.) and begins to rise	240	4
PSP cooling begins	11	
Hot shutdown achieved		17
Containment temperature and pressure peak		28

[a]FW runback and automatic SLCS system do not exist at Browns Ferry.

Table 2.5. Results of GE study (Ref. 2.2) of the progression of a turbine trip-initiated ATWS at a BWR 4 MK I containment plant

Event	Time (s)
Turbine trips	0
Turbine stop valve shut	0.1
Neutron flux reaches maximum (392%)	0.9
SRVs open	1.5
RPT, timed SLC logic triggered[a]	2.0
Maximum pressure (1193 psig) at vessel bottom	2.5
Maximum heat flux (133%)	2.7
SRVs start to close	9.0
Feedwater runback to zero[a]	45.0

[a]Browns Ferry does not have automatic SLC system or feedwater runback.

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				X	X	X	X	X	X	X				
			X	X	0	X	04	X	0	X	X			
		X	X	X	X	42	X	42	X	X	X	X		
	X	X	0	X	24	X	0	X	24	X	0	X	X	
X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
X	0	X	24	X	0	X	20	X	0	X	24	X	0	X
X	X	42	X	X	X	X	X	X	X	X	X	42	X	X
X	04	X	0	X	20	X	0	X	20	X	0	X	04	X
X	X	42	X	X	X	X	X	X	X	X	X	42	X	X
X	0	X	24	X	0	X	20	X	0	X	24	X	0	X
X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
	X	X	0	X	24	X	0	X	24	X	0	X	X	
		X	X	X	X	42	X	42	X	X	X	X		
			X	X	0	X	04	X	0	X	X			
				X	X	X	X	X	X	X				

Fig. 2.1. Typical middle-of-life control rod pattern for Browns Ferry showing rod notch positions. Each notch position corresponds to 3 inches (0.076 m) of travel. Fully withdrawn rods (notch position 48) are represented by "x".

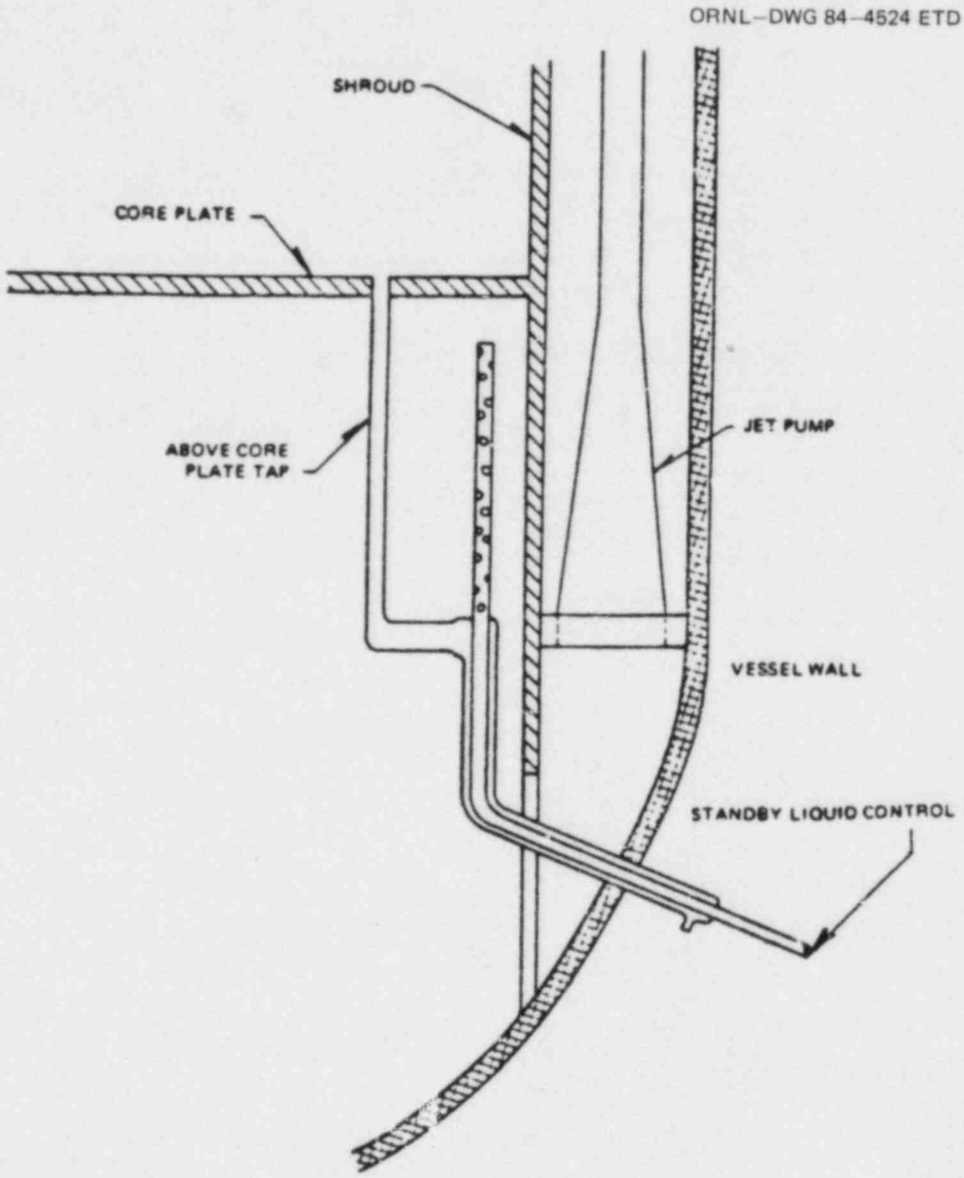

Fig. 2.2. Location of standby liquid control system injection sparger within the BWR 4 reactor vessel.

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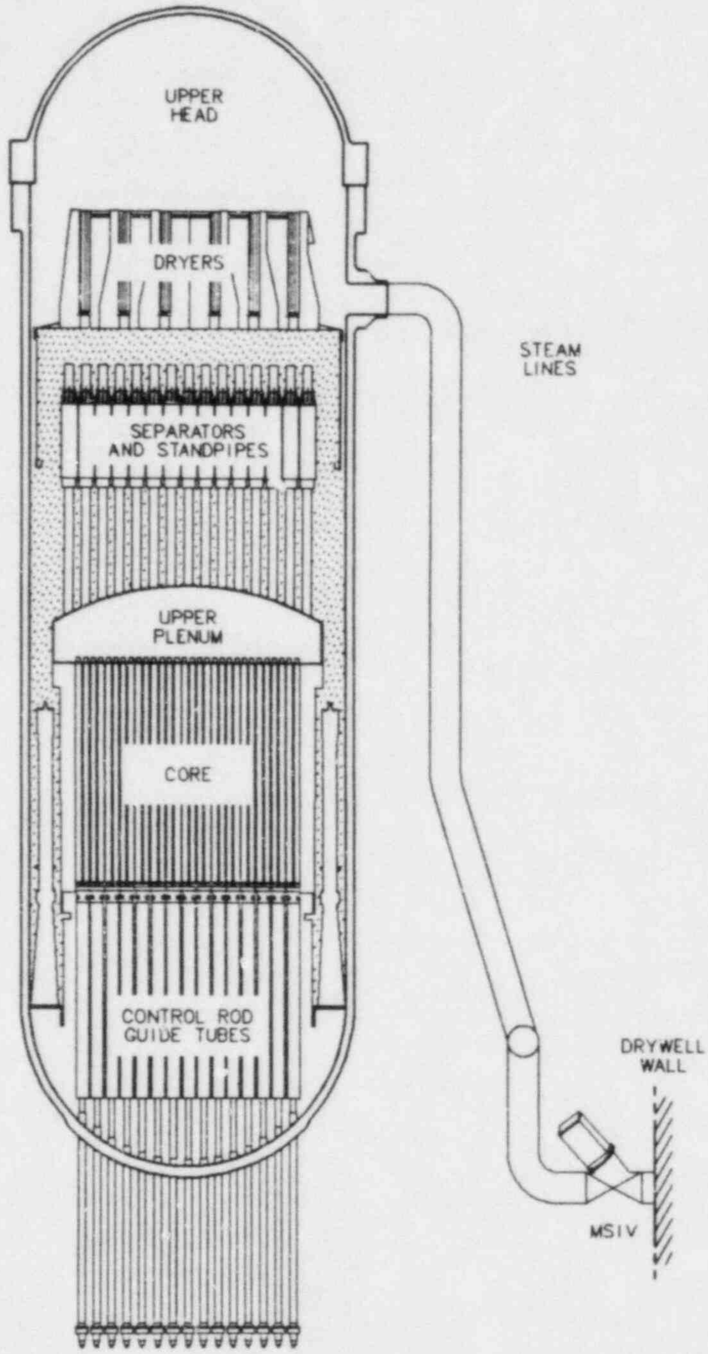

Fig. 2.3. BWR 4 reactor vessel internals.

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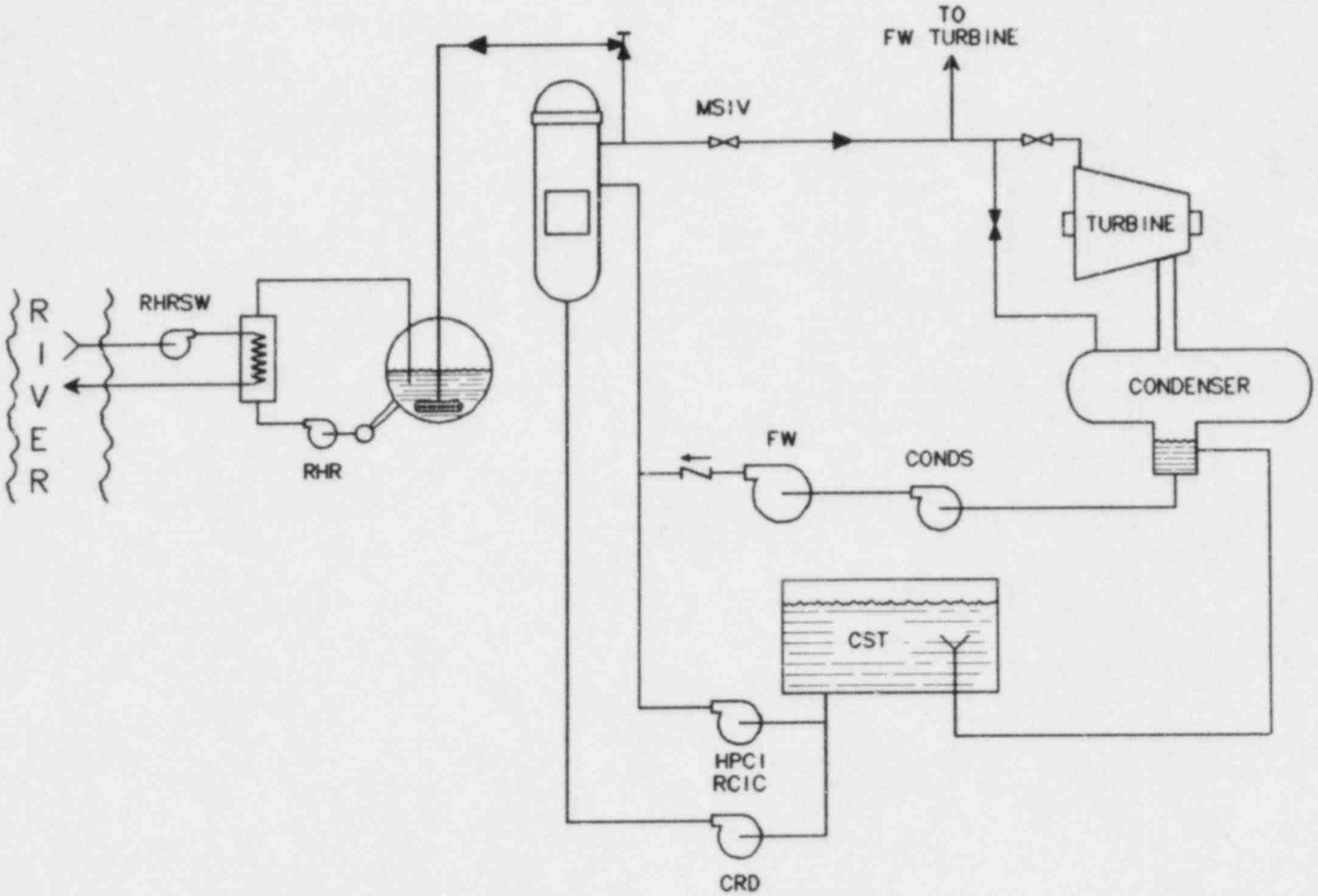

Fig. 2.4. Plant operation after failure of manual scram in the IORV — initiated ATWS sequence.

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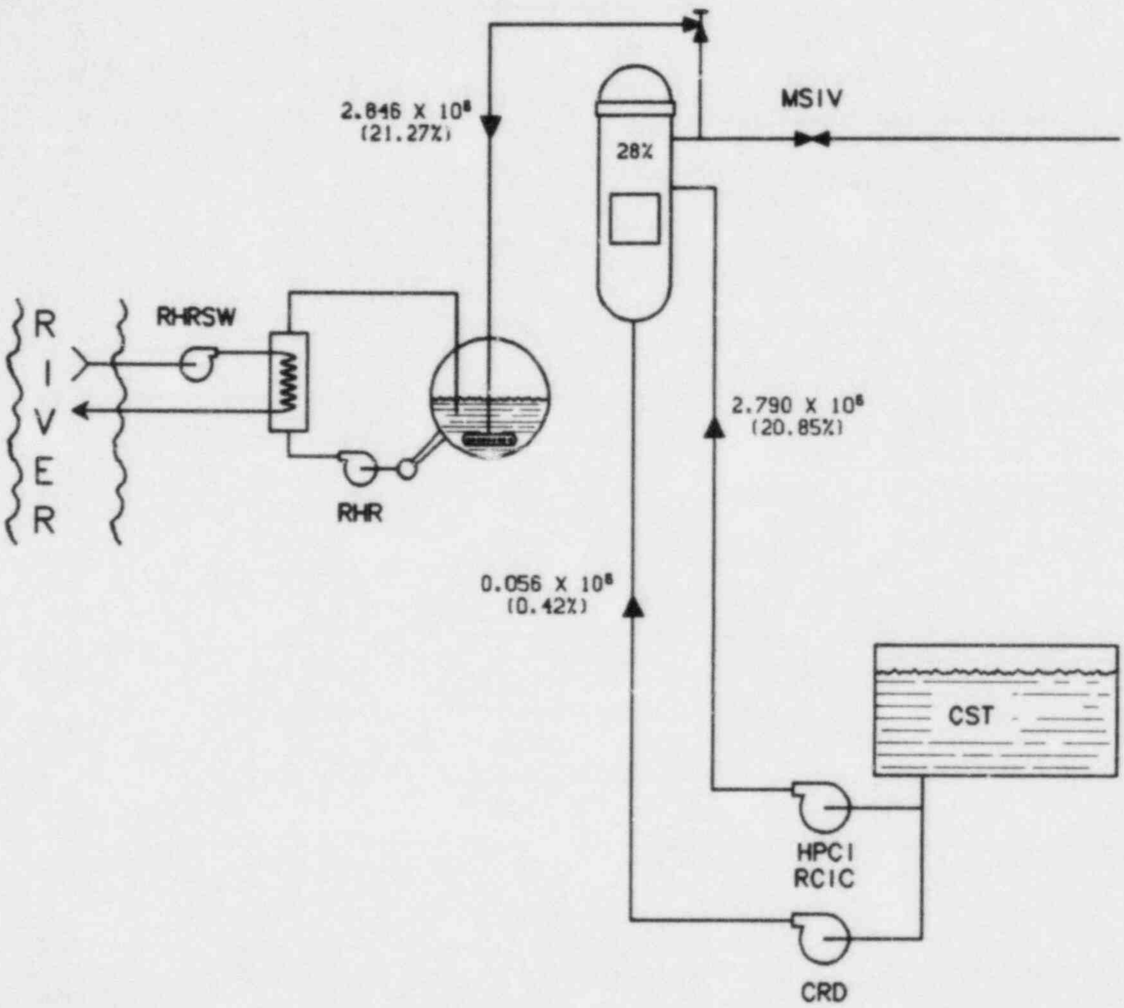

Fig. 2.5. Plant operation after failure of scram in the MSIV-closure initiated ATWS accident sequence.

ORNL-DWG 83-13727

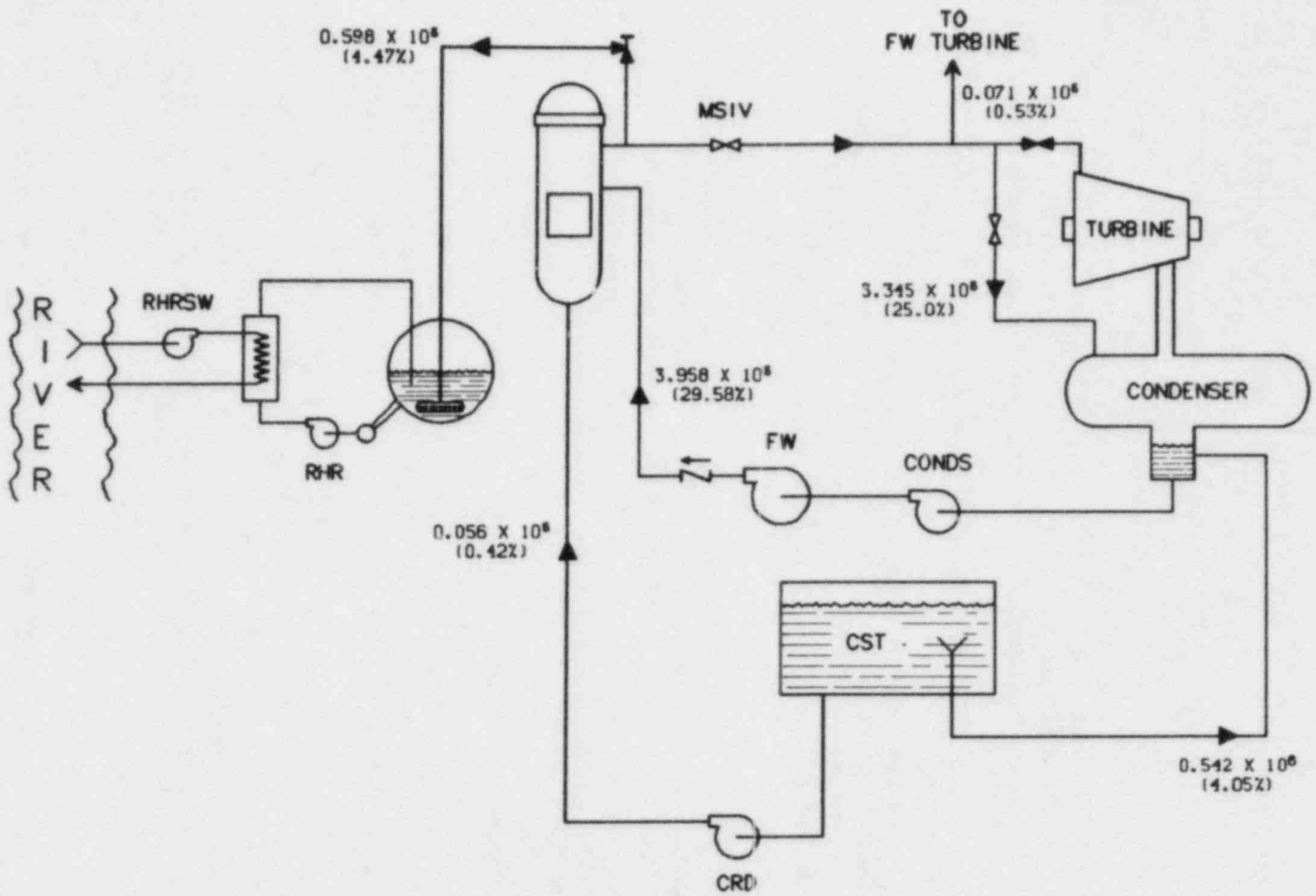

Fig. 2.6. Plant operation after failure of scram in the turbine trip initiated ATWS accident sequence.

3. MSIV-CLOSURE INITIATED ATWS WITHOUT OPERATOR ACTION

3.1 Introduction

This chapter presents the results of BWR-LACP calculations of the response of primary system and containment following an MSIV-closure initiated ATWS. After an anticipated transient such as closure of all MSIVs, the normal action of the protection system would be to cause the insertion of all 185 control rods into the reactor core, reducing the core power to decay heat levels.* For the calculations reported here, the assumption is made that none of the 185 control rods move into the core. The calculation period starts 50 s after the MSIVs begin to close and ends with the overpressure failure of the drywell about 37 min later.

Initial values (at the 50 s point) for the BWR-LACP calculation were taken from the BWR Owners Group results[3.1] discussed in Section 2.3 of this report. The BWR-LACP code is not programmed to simulate all the phenomena (e.g. vessel hydraulics with the recirculation pumps running) in effect before and immediately after the MSIV closure; thus, it is necessary to begin the BWR-LACP calculation at some time after the MSIV closure. In order to do this properly, the conditions calculated by another transient analysis code must be utilized as input to BWR-LACP for the initial values of plant parameters such as downcomer water level, reactor vessel pressure, and suppression pool temperature. The BWR Owners Group results in NEDO-24222 (Ref. 3.1), provide the desired information, calculated by the General Electric Company using proprietary transient analysis methods, for the first 50 s following MSIV closure from full power without reactor scram.

By the end of the 50 s BWR Owners Group calculation, the reactor power has readjusted from the initial 100% power level to 28% of rated power in response to the automatic trip of the reactor coolant recirculation pumps which occurs five seconds after the MSIVs begin to close. The reactor vessel is at full pressure [about 1100 psia (7.58 MPa)] and the downcomer water level is at 500 in.† (12.7 m) and decreasing.

The results presented in the following three sections are arranged around important events. The most significant of these is the loss of the HPCI system, which occurs as a result of the automatic shift of the HPCI pump suction away from the large supply of cool water in the CST (initially 362,000 gal.) to the heated water of the pressure suppression pool. The failure of the HPCI system hastens the eventual failure of

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*Scram would be demanded by four signals. In the order of receipt these are MSIV position less than 90% of full open, high neutron flux, high reactor vessel pressure, and low reactor vessel water level.

†Normal downcomer water level is 560 in. (14.23 m) above vessel zero.

primary containment and leads to severe power spikes that might cause fuel damage even prior to containment failure. The detrimental effects of the HPCI pump suction shift on long-term non-LOCA accidents have been discussed in previous SASA reports.*

Two variations of the no-operator-action sequence are discussed in Sect. 3.5: the sequence without the HPCI pump suction shift and the sequence that would result if the MSIV closure were initiated by a loss of off-site power.

3.2 Events Before Loss of HPCI (First 14.8 min.)

BWR-LACP results for a variety of important system variables during the entire accident sequence are shown on Figs. 3.1-3.7. Table 3.1 provides a timetable of significant events.

At the beginning of the calculation at time 50 s, the thermal power generation in the reactor core (Fig. 3.1) is 28% (i.e. 924 Mwt). Water level in the reactor vessel downcomer annulus (Fig. 3.2) is at 500 in.† (12.70 m) above vessel zero and is decreasing rapidly. The HPCI and RCIC systems are not yet actuated (Fig. 3.3) but the CRDHS (which runs continuously unless tripped by the operators) is injecting about 106 gpm (0.007 m^3/s) from the CST into the reactor vessel. The reactor vessel is fully pressurized, cycling between about 1100 psig and 1000 psig (7.69 and 7.00 MPa) in response to the automatic opening and closing of the SRVs (Fig. 3.4).

When the reactor vessel water level reaches 476.5 in. (12.10 m), the HPCI and RCIC systems actuate automatically and are soon injecting at full capacity — 600 gpm (0.038 m^3/s) for RCIC and 5000 gpm (0.315 m^3/s) for HPCI. The water level increases slightly and the core thermal power changes correspondingly until the total vessel injection (HPCI, RCIC, and CRDHS) is equivalent to the production rate of steam in the reactor core. After reaching this quasi-equilibrium state, the vessel water level fluctuates about a mean value of 476 in. (12.09 m) in response to the fluctuating vessel pressure.

Since the MSIVs are closed, all of the steam produced in the reactor vessel that is not used for HPCI or RCIC turbine operation is discharged through the SRVs to be condensed in the 951,000 gal (3600 m^3) of water held in the pressure suppression pool. Distribution of the steam into the pool is accomplished by T-quenchers, which are 10-in. (0.25-m) diameter horizontal perforated pipes located 10 ft (3.05 m) below the surface of the 16-ft (4.88-m) deep pool, one T-quencher at the outlet of each SRV. There are over a thousand small steam release holes in the surface of each T-quencher, sized and arranged to promote stable condensation of the escaping steam.

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*See, for example, Sect. 9.3 of Ref. 3.2.

†As discussed in 3.1, the BWR-LACP calculation begins 50 s after the MSIV closure, during which time downcomer water level has decreased from the normal 560 in. (14.23 m) indication.

During the first 15 min of the accident sequence, the pressure suppression pool temperature (Fig. 3.5) is increasing from 90 to 190 F (305 to 361 K) and the condensation effectiveness is 100%. The water level of the pool increases by more than 1 ft (0.305 m) during this period due to the added mass of water from condensed steam and also because of the slight expansion of the water as it is heated. Drywell temperature and pressure (Figs. 3.6 and 3.7) do not increase appreciably during this first part of the accident since 100% of the steam is condensed in the suppression pool and because the drywell coolers continue to run throughout the period. Drywell temperature actually decreases during the first 20 min because the trip of the recirculation pumps removes part of the heat load on the drywell coolers.

When the indicated pressure suppression pool water level reaches +7 in. (an increase of 11 inches over the initial -4 in. indication*), the HPCI system pump suction is automatically shifted away from the CST and to the suppression pool. The pool temperature at the time of the suction shift is 152°F (340 K). The HPCI system can, at least temporarily, accommodate the pumping of water at this temperature, so initially, the HPCI system would keep running and pump the heated suppression pool water at a rate of 5000 gpm (0.315 m^3/s).

As time passes, the increasing suppression pool temperature challenges the ability of the HPCI system to keep pumping. The HPCI turbine lube oil is cooled by the water being pumped. Hotter, less viscous oil can impair the bearings, the turbine governors, and the gear reducer. Detailed discussion of HPCI capability was submitted by the TVA in Amendment 67 to the Browns Ferry FSAR (pages 14.1-14.5). This discussion concludes that the HPCI can, for limited periods, pump water at 162°F (345 K) without failing, but that oil temperatures in excess of 200°F (366 K) are to be avoided. Allowing for a heat exchanger ΔT of 10°F (6 K), this upper limit translates to a maximum pumped water temperature of 190°F (361 K). Therefore, the calculations discussed in this section are done under the assumption that the HPCI fails when the pumped water (i.e., the suppression pool after the suction shift) temperature exceeds 190°F (361 K).

As shown on Fig. 3.5, the HPCI pump suction shifts at time 8.3 min and the suppression pool temperature reaches 190 F (361 K) at 14.8 min; these events cause failure of the HPCI system and end the initial phase of the accident by reducing the total vessel water injection flow from 5700 gpm (0.36 m^3/s) to only 700 gpm (0.044 m^3/s).

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*Instrument zero is 4 in. (0.1 m) below the midplane of the 31 ft. (9.45 m) diameter torus; thus, an indication of 4 in. would mean that the torus is half full of water.

3.3 Events from Loss of HPCI to ADS Actuation (14.8 min. to 18 min.)

After HPCI system failure, the total vessel injection (from RCIC and CRDHS) is about 700 gpm (0.044 m^3/s) — insufficient to replace the water inventory loss with the core critical and generating 28% thermal power. (The condensate booster pumps have been running since before the accident and are not automatically tripped as a result of the accident; however, they cannot inject water into the reactor vessel because it is still fully pressurized.) The downcomer water level decreases rapidly, and is below 413.5 in. (10.50 m) within 1.3 min. As water level decreases, the natural circulation of water within the reactor vessel decreases, reducing flow into the core and introducing additional negative void reactivity sufficient to reduce the core power to about 10%. Even at this lower power level, the vessel water inventory cannot be maintained by the RCIC and CRDHS alone, so water level continues to decrease.

Upon receipt of the low water level signal at 413.5 in. (10.5 m) indicated downcomer water level, the LPCI and Core Spray pumps start but do not immediately inject, since the vessel is still pressurized. The ADS timer also begins with the low water level signal, since the other requirements for ADS are met at this time: drywell pressure $\geq$2.45 psig (0.118 MPa), confirmatory vessel low level $\leq$546 in. (13.87 m), and sensed pressure at the LPCI or Core Spray pump discharge. The vessel water level continues to decrease, reaching the top of active fuel before ADS actuation. After the timer completes its 120 s cycle, the ADS actuates, opening six SRVs.

3.4 Events After ADS Actuation (18 min. to 37 min.)

The ADS actuation immediately opens six SRVs* and initiates a rapid depressurization of the reactor vessel (Fig. 3.4). Much of the inventory of hot water in the reactor vessel flashes and passes through the six open SRVs to be discharged in the suppression pool. The rapid loss of vessel water inventory completely uncovers the core within one minute (Fig. 3.2).

With the core uncovered, criticality cannot be sustained and the core thermal power subsides to the decay heat level. Heat-up of the fuel is relatively slow at decay heat levels, so there is no immediate fuel damage.

When vessel pressure decreases to below 418 psia (2.882 MPa) at 19.6 min, the condensate booster pumps (CBPs), in series with the

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*Immediately prior to ADS actuation there is one open SRV. If this open SRV were a member of the group of six SRVs assigned to the ADS, the ADS actuation would immediately open only five SRVs, but this would bring the total number of open SRVs to six.

condensate pumps, begin pumping water from the main condenser hotwell to the reactor vessel. Figure 3.8 shows the flow path from hotwell through the inactive feedwater heaters and turbine-driven main feedwater pumps (tripped by lack of steam since the MSIV closure) and into the reactor vessel. [For the first 19.6 min of the accident sequence, the vessel pressure is above the combined shut-off head of the condensate and condensate booster pumps; the pumps are protected from overheating by automatic flow control valve 2-29 which maintains a minimum recirculation flow (about 25% of full flow) from the booster pump discharge back to the condenser hotwell.]

The LPCI and Core Spray pumps begin injection (Fig. 3.3) within 10 s of the initiation of CBP flow, as reactor vessel pressure decreases to below their shut-off heads. The combined flow from the CBPs and the two low pressure ECCS systems peaks at about 67000 gpm (4.23 m^3/s).* This great flow recovers the core in about 20 s.

The recovery of reactor vessel water level provides enough moderator for the core to again sustain criticality. As the initial point of re-criticality is exceeded, the neutron power level in the core is several orders of magnitude below the power range, but increasing rapidly. Continued increase in water level sets the stage for a power excursion by building excess positive reactivity. The excursion is triggered when the core thermal power increases to about 5%, producing more steam than the six open SRVs can pass at the low vessel pressure of 133 psia (0.92 MPa) in effect at this instant. The resulting pressure increase collapses steam voids in the core, creating additional positive reactivity. Pressure and core power spiral upward together, the increase in one stimulating the increase of the other. The cycle of increasing power and pressure is broken when pressure reaches the relief valve setpoints and all 13 SRVs open, limiting vessel pressure to the neighborhood of 1100 psia (7.584 MPa). Core thermal power increases to 178% of the rated 3300 Mwt before the increasing moderator temperature generates sufficient voids to reverse the power increase.

Whenever the reactor vessel pressure is above 418 psia (2.88 MPa), there is no injection by the low-pressure systems. Without the massive injection that caused the power/pressure excursion, the reactor attempts to approach a stable equilibrium. The vessel for a time remains pressurized, discharging steam produced by the high but decaying reactor power. The combined RCIC and CRDHS injection of about 700 gpm (0.044 m^3/s) is insufficient to prevent a steady decrease in vessel water level. As water level decreases, the core power decreases; when the steaming rate is less than 36% (about 2 min after the beginning of the excursion) the six open SRVs are discharging more steam than is being produced, so vessel pressure begins to decrease.

When the reactor vessel pressure has decreased to below about 418 psia (2.88 MPa) the still-running CBPs and LPECCS pumps are again able to inject. This is the beginning of a nearly identical cycle consisting

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*The reactor vessel pressure does not drop low enough to permit design capacity injection by the low-pressure systems which would be about 82,500 GPM (5.20 m^3/s) as indicated in Table 3.2.

of vessel depressurization followed by a deluge of water injected by the low-pressure systems, and the resultant power excursion and repressurization of the reactor vessel. As shown on Figs. 3.1 through 3.4, this basic cycle is repeated four times before the overpressure failure of the drywell at 37 min. The first cycle is most severe, with a peak core thermal power of 178%. These BWR-LACP calculations make the assumption that these thermal power peaks do not cause any significant disruption of the core geometry.

With all MSIVs shut during the entire accident, all of the energy of the steam discharged by the SRVs must be absorbed in the primary containment. As discussed in Appendix A, the BWR-LACP calculations discussed in this report assume that 100% of the SRV discharge will be condensed if the temperature of the suppression pool water in the vicinity of the T-quencher devices is at least 10°F (5.6 K) below saturation (i.e., at least 10 F of subcooling), that none of the discharge is condensed if there is no subcooling, and that the percent condensed varies linearly between 100% and 0% as the subcooling decreases from 10°F to 0°F.

As shown on Fig. 3.5, the bulk pressure suppression pool temperature increases monotonically throughout the accident sequence. Without operator action, the RHR system pool coolers are not operating; however, their cooling would be insufficient to prevent the rapid heatup of the suppression pool even if they were operated.

During the first 21 min of the accident sequence, the bulk pressure suppression pool temperature [initially 90 F (305 K)] increases from 122°F (68 K) of subcooling to 10°F (6 K) of subcooling. During this period, 100% of the SRV discharge is condensed. As shown on Fig. 3.6, drywell pressure increases by about three psi (0.007 MPa) during this period because there is some steaming by evaporation from the surface of the suppression pool. The drywell atmosphere temperature is maintained at or below its initial temperature of 145 F (336 K) throughout most of the 21 min by operation of the drywell coolers.

After 21 min, the suppression pool does not have the 10°F (6 K) of subcooling required for 100% condensation of the SRV discharge. A fraction (between 10 and 20%) of the SRV discharge is able to bubble up through the $\geq$10 ft (3.05 m) of slightly subcooled water above the T-quencher, and break through the surface into the wetwell atmosphere. This steam easily and quickly reaches the drywell atmosphere via the 12 two-ft (0.61-m) diameter vacuum breakers, which open a direct flow path from the wetwell atmosphere to the drywell atmosphere whenever the wetwell pressure exceeds the drywell pressure by more than 0.5 psi (0.003 MPa). The direct bubble-through of steam causes a sharp increase in drywell pressure and temperature (Figs. 3.6 and 3.7) beginning at 21 min. By 37 min, the drywell pressure reaches the assumed 132 psia (0.910 MPa) failure pressure* of the drywell.

About 1.5 min before the drywell failure, the drywell pressure exceeds 110 psia (0.76 MPa) and the six ADS valves go shut. (The drywell

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*The assumed static overpressurization failure point for the drywell is taken from the information provided in Ref. 3.3.

control air pressure, normally at 115 psia (0.79 MPa) must be at least 5 psi above the drywell pressure in order to continue to hold the SRVs open). After the ADS valves close, the reactor vessel pressure immediately increases until automatic SRV actuations limit vessel pressure to the 1100 psia (7.59 MPa) range. This failure of the ADS has little effect on the overall accident sequence because it occurs after the drywell overpressure failure has been made inevitable by the cessation of steam condensation in the pressure suppression pool during the fourth vessel flooding cycle. This conclusion would be true even if the ADS valves were assumed to be closed when the drywell pressure reached 100 psia (0.69 MPa), some 10 psi (0.069 MPa) lower than the base case. As the drywell pressure was reduced after drywell failure, the ADS valves would reopen (Drywell control air pressure must be at least 25 psid (0.17 MPa) above drywell pressure in order to be able to open closed SRVs).

The calculation ends with drywell failure. BWR-LACP is not programmed to calculate events after the drywell failure, which include the possibility of loss of reactor vessel injection and severe fuel damage.

3.5 Variations of the No-Operator-Action Accident Sequence

If the MSIV-closure initiated ATWS accident sequence were compounded by a loss of off-site power (LOSP), the resulting sequence of events would be very similar to that discussed in Sections 3.2—3.4, but somewhat less severe. The reason for the difference in severity is that the condensate and condensate booster pumps are tripped upon LOSP, and therefore would not be available to contribute to the excessive reactor vessel flooding that causes the power peaks shown on Fig. 3.1.

Since the large capacity RHR and Core Spray pumps are powered by the diesel generators after LOSP, reactor vessel flooding would occur after ADS actuation, but at a slower rate. Instead of thermal power peaks attaining levels between 150% and 180% of the rated 3300 Mwt, the power peaks would be between 90% and 130%. With generally lower reactor power, the pressure suppression pool temperature would not increase as rapidly and the drywell would not pressurize as rapidly. Calculations show that the overpressure failure of the drywell would occur after 41 min instead of after 37 min.

A second variation of the MSIV-closure initiated ATWS accident sequence would occur if there were a failure of the HPCI system logic that governs the HPCI pump suction shift from the condensate storage tank to the pressure suppression pool. The resulting sequence of events differs greatly from the sequence discussed in Sections 3.2—3.4.

Without the automatic shift of the HPCI pump suction to the heated pressure suppression pool, the HPCI system would not fail but would continue to pump at full flow throughout the calculational period. Therefore, the reactor vessel water level would remain above 413.5 in. (10.50 m), and there would be no initiation of the ADS timer and no injection by the CBP, LPCI, or Core Spray systems. The reactor vessel would remain at pressure, with reactor power in the neighborhood of 28%. Without depressurization and the subsequent deluge of injection from the low-pressure high-capacity pumping systems, there would be no power spikes. Since the rate of steam release to the pressure suppression pool would on the average be lower, so also would the overpressure failure of the drywell be delayed from 37 min (or 41 min for the LOSP initiated sequence) to 51 min after the inception of the accident sequence.

It should be noted that the no-operator-action sequence without HPCI suction shift could only occur as a result of a failure of the HPCI system logic or the suction valve motor-operators.

3.6 Summary and Conclusions for Chapter 3

The sequence of events leading to overpressurization failure of the primary containment in an MSIV-closure initiated ATWS accident sequence in which no action is taken by the operators has been developed and discussed in this chapter. Containment failure has been shown to occur about 37 min. after the inception of the sequence. Actions of the installed systems provided for automatic LOCA protection cause repeated cycles of reactor vessel depressurization, injection of large amounts of relatively cold water, core power excursion, and reactor vessel repressurization during the period before containment failure.

Since it is inconceivable that the plant operators would take no action of any kind (appropriate or inappropriate) when confronted with an MSIV-closure initiated ATWS, it is obvious that the purpose of this chapter is not to provide indication of the timing and sequence of events for an actual case. Rather, the purpose of this study of the no-operator-action sequence of events is to provide information concerning what the specific goals of operator actions should be; in other words, what undesirable features of the no-operator-action sequence of events should the operators strive to prevent and what desirable event sequence features should be substituted by operator action? This information is invaluable to the analysis of the corresponding event sequences <u>with</u> operator action that are presented in Chapters 4 and 5.

The progression of the no-operator-action accident sequence is determined by automatic responses of the HPCI, feedwater, RHR, Core Spray, and ADS systems. Perhaps the most important of these responses is the early failure of the HPCI system, caused by high lube oil temperature. As a part of the overall plan for protection of the plant from a large-break LOCA, the suction of the HPCI pump is automatically shifted from the CST to the pressure suppression pool upon increased pool level. Since the HPCI system lube oil is cooled by the water being pumped and the pressure suppression pool is rapidly heated during the ATWS sequence, the HPCI system would be lost early in the sequence. Without the injection provided by the HPCI system, reactor vessel water level decreases rapidly and this leads to actuation of the ADS.

The Automatic Depressurization System (ADS) is provided for protection of the plant from a small-break LOCA in which insufficient makeup is provided by the high-pressure injection systems while the reactor vessel pressure remains above the shutoff head of the large-capacity low-pressure injection systems. In the no-operator-action ATWS accident sequence, these conditions are duplicated after failure of the HPCI system; the RCIC and CRD hydraulic systems continue to inject*, but their combined flow is insufficient to maintain reactor vessel water level.† When the water level has decreased to near the top of the core, the ADS system automatically opens six SRVs, depressurizing the reactor vessel and permitting vessel flooding by the large-capacity, low-pressure injection systems.

The low-pressure ECCS systems (Core Spray and LPCI mode of RHR) are designed to provide large quantities of water as necessary to ensure that the reactor core would remain cooled in the event of a large-break LOCA. Since the water released from the reactor vessel in a large-break LOCA accident sequence would fall onto the drywell floor and then drain into the pressure suppression pool, the low-pressure ECCS system pumps take suction on the pressure suppression pool. With the containment back-pressure provided by evaporation and subsequent steaming from the surface of the pool under ATWS conditions, sufficient net positive suction head (NPSH) would be maintained to permit ECCS pump operation as the pressure suppression pool temperature increases.

In addition to the low-pressure ECCS systems, injection into the depressurized reactor vessel would also involve the feedwater system. The condensate pumps and condensate booster pumps are driven by electric motors and therefore would remain running after the feedwater pumps, driven by steam turbines, become inoperative by means of the MSIV closure at the inception of the accident sequence. With the reactor vessel pressurized, the condensate and condensate booster pumps are protected from overheating by minimum flow lines that lead back to their suction source (Fig. 3.8); when the reactor vessel is depressurized, these pumps can deliver flow through the idle feedpumps into the reactor vessel.

Table 3.2 indicates the large potential for reactor vessel flooding when the vessel is rapidly depressurized. The table indicates the design capacity and corresponding design differential pressure across the pumps for each system. Since these systems incorporate electric motor-driven constant speed pumps, the actual rate of injection during reactor vessel depressurization would vary as a function of vessel pressure. The reactor vessel pressure at which injection would begin for each of the low-pressure system is also shown in Table 3.2. It can be seen that the condensate booster pumps would begin injection first as the reactor vessel pressure decreases, followed in order by the Core Spray system and the LPCI mode of the RHR system. It should be appreciated that, at design capacities, these systems have the ability to completely fill the reactor vessel in less than 2 min of operation.

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*The RCIC system subsequently fails by automatic trip on high containment pressure at 40 psia (0.27 MPa) at 26 min.

†It should be noted that high drywell pressure, as a confirmation that a LOCA has occurred, is required by ADS logic as a prerequisite for system operation. In an ATWS accident sequence, the necessary high drywell pressure signal would be provided by evaporative steaming from the pressure suppression pool (see Fig. 3.6).

In the no-operator-action ATWS accident sequence, the rapid reactor vessel depressurization occasioned by operation of the ADS permits the injection of enormous quantities of relatively cold water, sufficient to recover the core with relatively voidless moderator within seconds. Even though the ADS valves remain open, the resulting power spike causes an increase in reactor vessel pressure that temporarily prevents further injection by the low-pressure injection systems. The reactor vessel pressure quickly reaches the relief valve setpoint, and additional SRVs open as necessary to maintain the pressure in this vicinity. This restores the situation to that at the beginning of the cycle with reactor vessel pressure at the relief valve setpoint and, because the HPCI system is not operating, a decreasing reactor vessel water level. Thus the cycle repeats.

The steam leaving the reactor vessel during the accident sequence is discharged into the pressure suppression pool via the T-quencher devices attached to the terminus of each relief valve tailpipe. At first, all discharged steam is condensed in the relatively cool pressure suppression pool but as the pool temperature increases, the local temperatures around the discharging T-quenchers no longer permit complete steam condensation; after this, primary containment pressurization is rapid and the failure pressure is reached at 37 min after inception of the accident.

What actions might the operators take to forestall the primary containment failure that would otherwise occur at time 37 min. or, indeed, to prevent it entirely? To accomplish this, it is clearly necessary to reduce the rate of steam discharge into the pressure suppression pool and to provide pool cooling. This indicates the desirability of reducing reactor power and preventing the pressure spikes and low-pressure injection cycles so characteristic of the no-operator-action case. These considerations provide the bases for the material presented in the two follow on-chapters, in which the accident sequence with operator action is discussed.

References for Chapter 3

3.1 General Electric Company, *Assessment of BWR Mitigation of ATWS, Volume II (NUREG-0460 Alternate No. 3)*, NEDO-24222 (February 1981).

3.2 S. A. Hodge et al, "Loss of DHR Sequences at Browns Ferry Unit One — Accident Sequence Analysis," NUREG/CR-2973, ORNL/TM-8532 (May 1983).

3.3 L. G. Greimann et al., "Reliability Analysis of Steel Containment Strength," NUREG/CR-2442 (June 1982).

Table 3.1. ATWS with no operator action
[No LOSP, i.e., with condensate booster pumps (CBPs)]

Event	Time (min)	Comment
MSIV closure initiated	0	No scram
RPT	0.1	At reactor vessel pressure 1135 psia (7.83 MPa)
HPCI and RCIC start	1	At reactor vessel level 476.5 in. (12.1 m)
HPCI suction shift	8.3	At +7 in. indicated PSP level
HPCI fails	14.8	At 190°F (361 K) PSP temperature
Start LPECCS pumps and ADS timer	16.0	At reactor vessel level 413.5 in. (10.5 m)
First core uncovery	16.7	At 360 in. (9.14 m) [totally uncovered at 216 in. (5.44 m)]
ADS actuation	18.0	Two minutes after timer actuation
LPECCS and CBP injection begins	19.6	CBP at 418 psia (2.88 MPa); Core spray at 357 psia (2.46 MPa); LPCI at 346 psia (2.39 MPa)
First core recovery	19.9	At 360 in. (9.14 m)
LPECCS and CBP injection stops as reactor vessel pressure increases	20.4	LPCI 346 psia (2.39 MPa); Core spray 357 psia (2.46 MPa); CBP 418 psia (2.88 MPa)
Vessel pressure at relief valve setpoint	20.7	At 1120 psia (7.72 MPa)
First core power peak	20.7	Thermal power = 178%
Drywell coolers fail on over-temperature	22.4	At 200°F (367 K) drywell atmosphere
Second core uncovery	23.1	
LPECCS and CBP injection begins	24.4	
Second core recovery	24.7	
LPECCS and CBP injection stops	25.2	
Vessel pressure at relief valve setpoint	25.4	
RCIC turbine trip on high turbine exhaust pressure	26.0	
Second core power peak	27.7	Thermal power = 140%
Third core uncovery	27.6	

Table 3.1 (continued)

Event	Time (min)	Comment
LPECCS and CBP injection begins	29.0	
Third core recovery	29.4	
LPECCS and CBP injection stops	29.8	
Third core power peak	30.0	Thermal power 156%
Vessel pressure at relief valve setpoint	30.1	
Fourth core uncovery	32.1	
LPECCS and CBP injection begins	33.6	
Fourth core recovery	34.0	
Fourth core power peak	34.7	Thermal power = 147%
Vessel pressure at relief valve setpoint	34.7	
Drywell fails	36.8	Overpressure at 132 psia (.91 MPa)

Table 3.2. Injection characteristics of the low-pressure, high-capacity injection systems[a]

System	Design capacity [gpm (m^3/s)]	Design differential pressure [psi (MPa)]	Reactor vessel pressure at which injection begins [psia (MPa)]
Condensate booster pumps (3 pumps)	30,000 (1.893)	364[b] (2.510)[b]	418 (2.882)
Core spray system (4 pumps)	12,500 (0.789)	267 (1.841)	357 (2.461)
LPCI mode of RHR system (4 pumps)	40,000 (2.524)	250 (1.724)	346 (2.386)

[a]Systems described are those actually installed at Browns Ferry Unit 1.

[b]This is the differential pressure across both the condensate pumps and the condensate booster pumps.

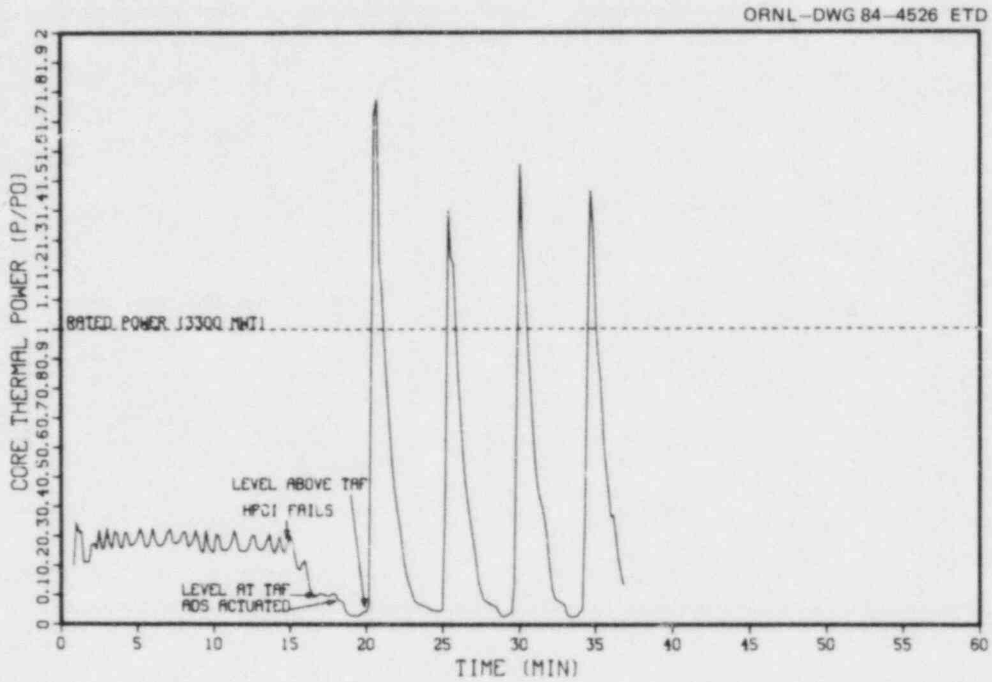

Fig. 3.1. Fraction of rated core thermal power for the MSIV closure-initiated ATWS accident sequence without operator action.

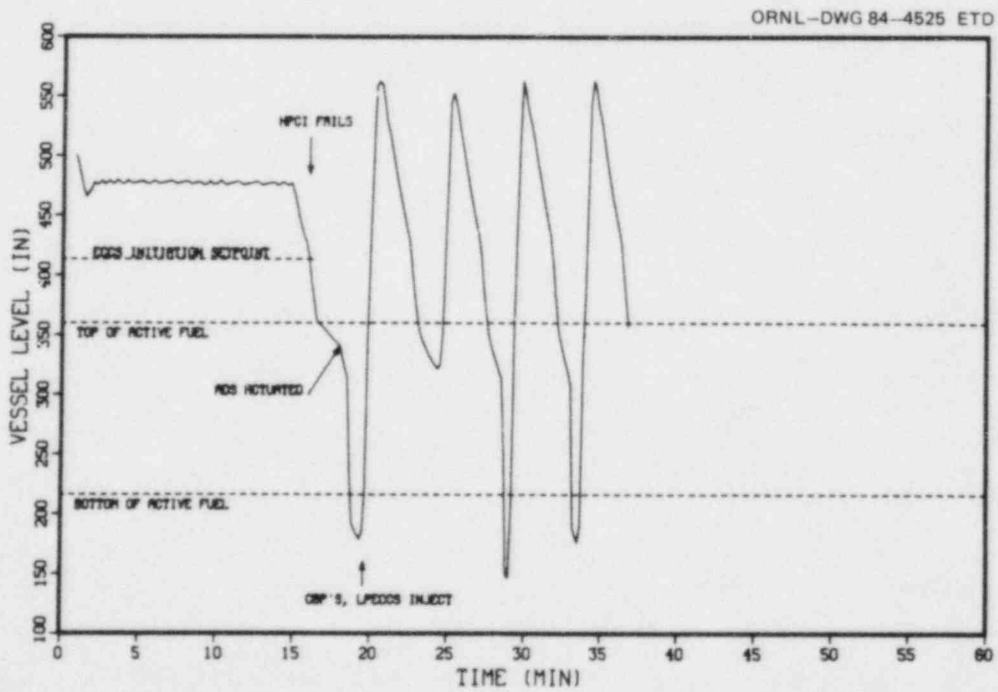

Fig. 3.2. Reactor vessel water level for the MSIV closure-initiated ATWS accident sequence without operator action.

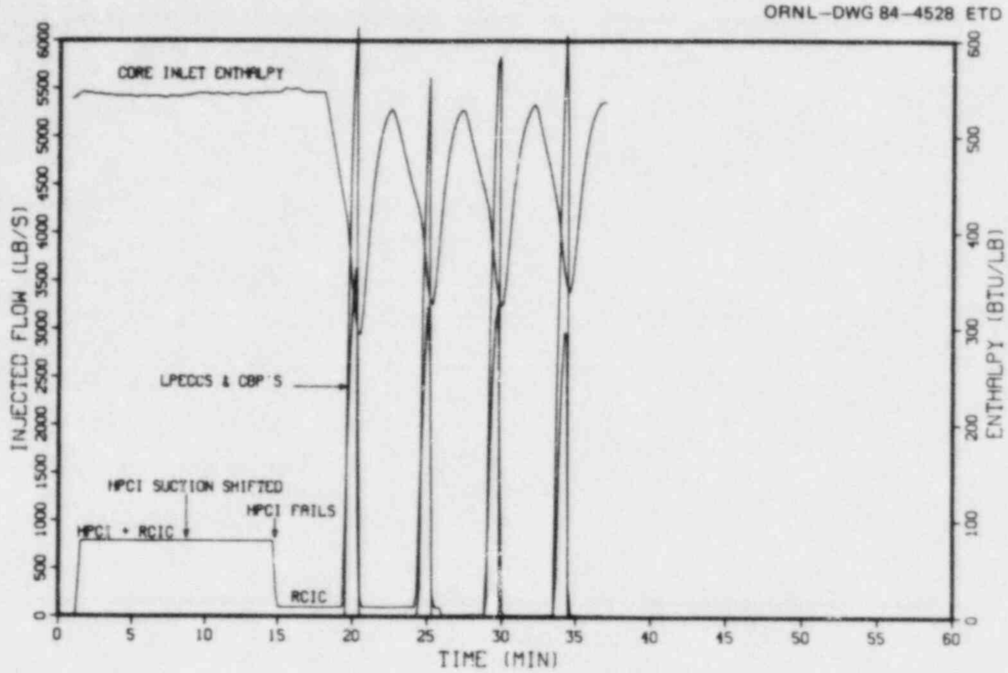

Fig. 3.3. Core inlet enthalpy and rate of injected flow for the MSIV closure-initiated ATWS accident sequence without operator action.

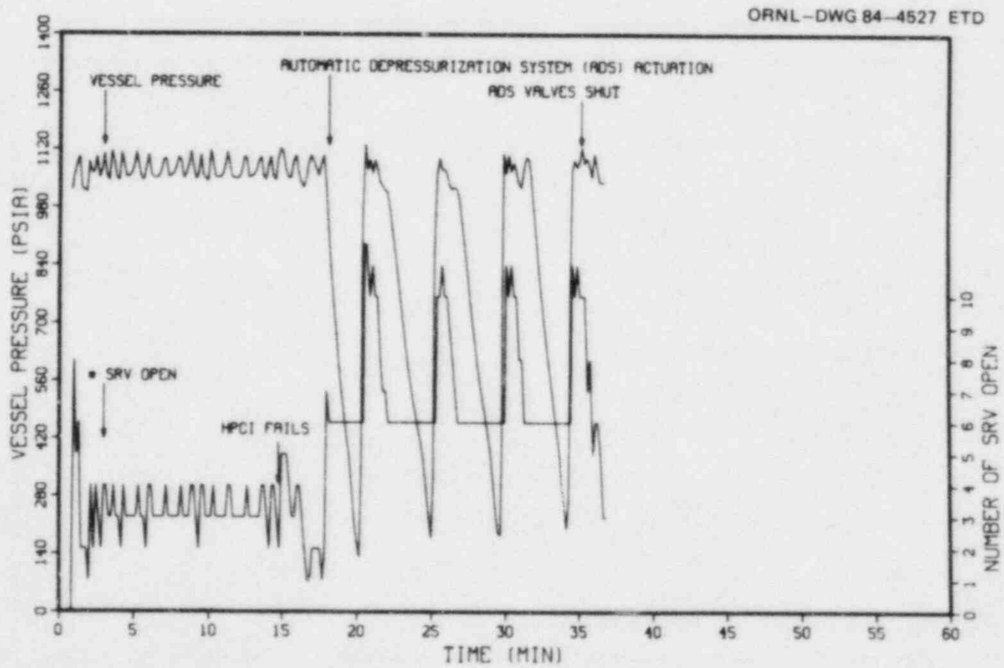

Fig. 3.4. Reactor vessel pressure and SRV operating history for the MSIV closure-initiated ATWS accident sequence without operator action. Suppression pool level zero is four inches below the horizontal centerline of the torus.

ORNL-DWG 84-4530 ETD

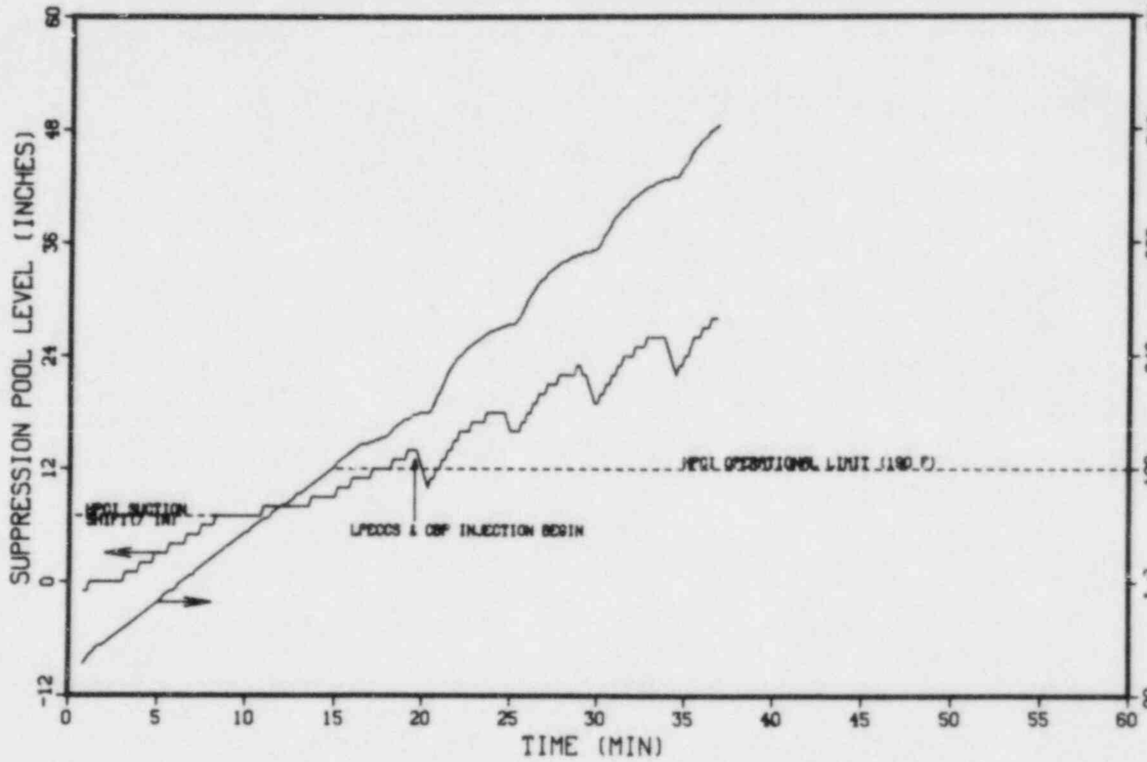

Fig. 3.5. Pressure suppression pool water level and temperature for the MSIV closure-initiated ATWS accident sequence without operator action.

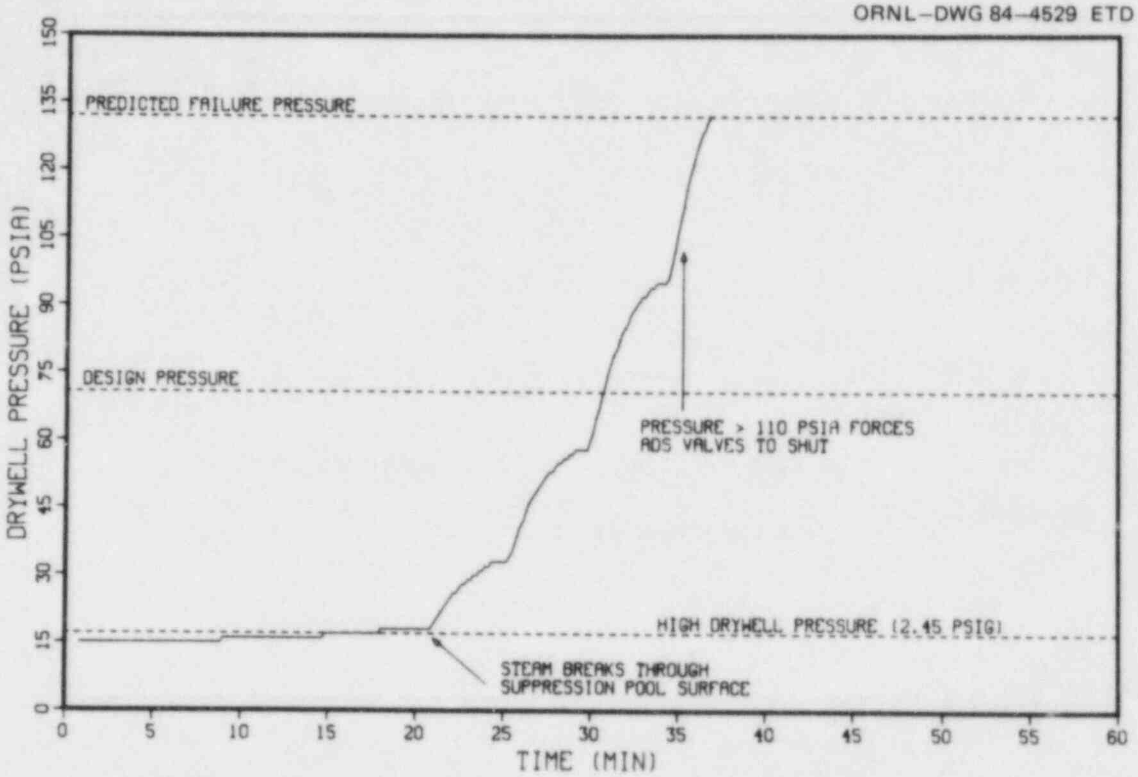

Fig. 3.6. Drywell pressure for the MSIV closure-initiated ATWS accident sequence without operator action.

ORNL-DWG 84-4531 ETD

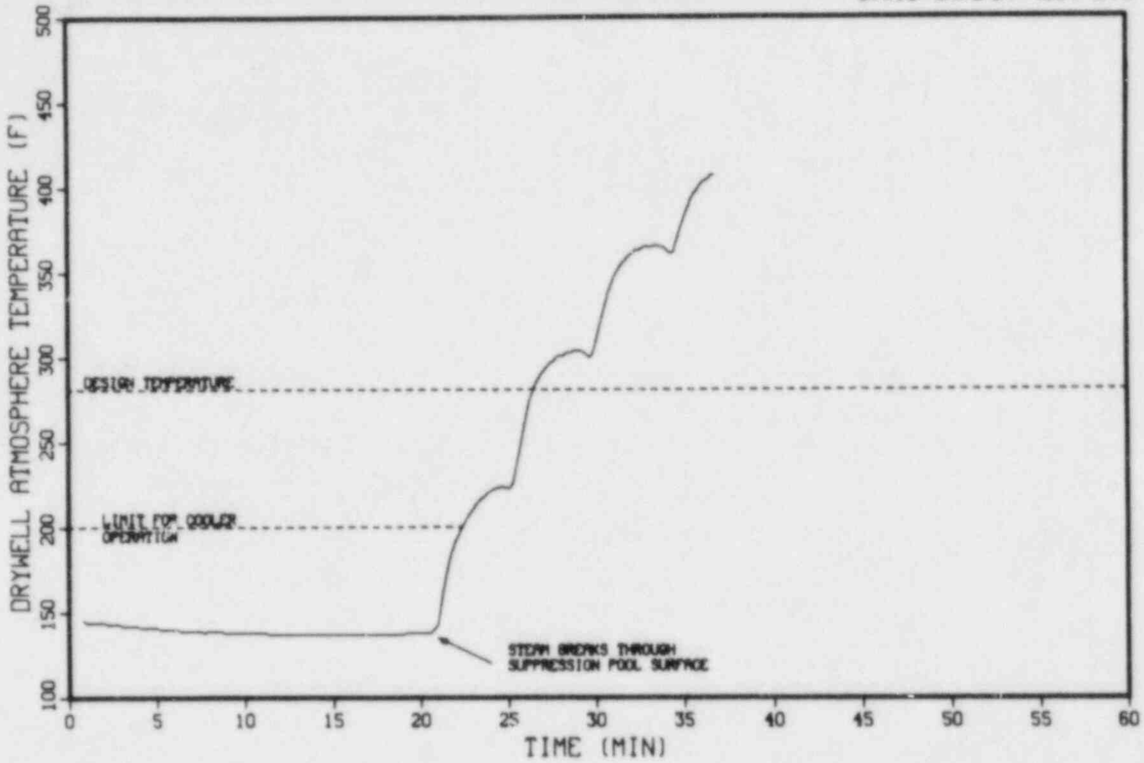

Fig. 3.7. Drywell atmosphere temperature for the MSIV closure-initiated ATWS accident sequence without operator action.

ORNL-DWG 83-13725

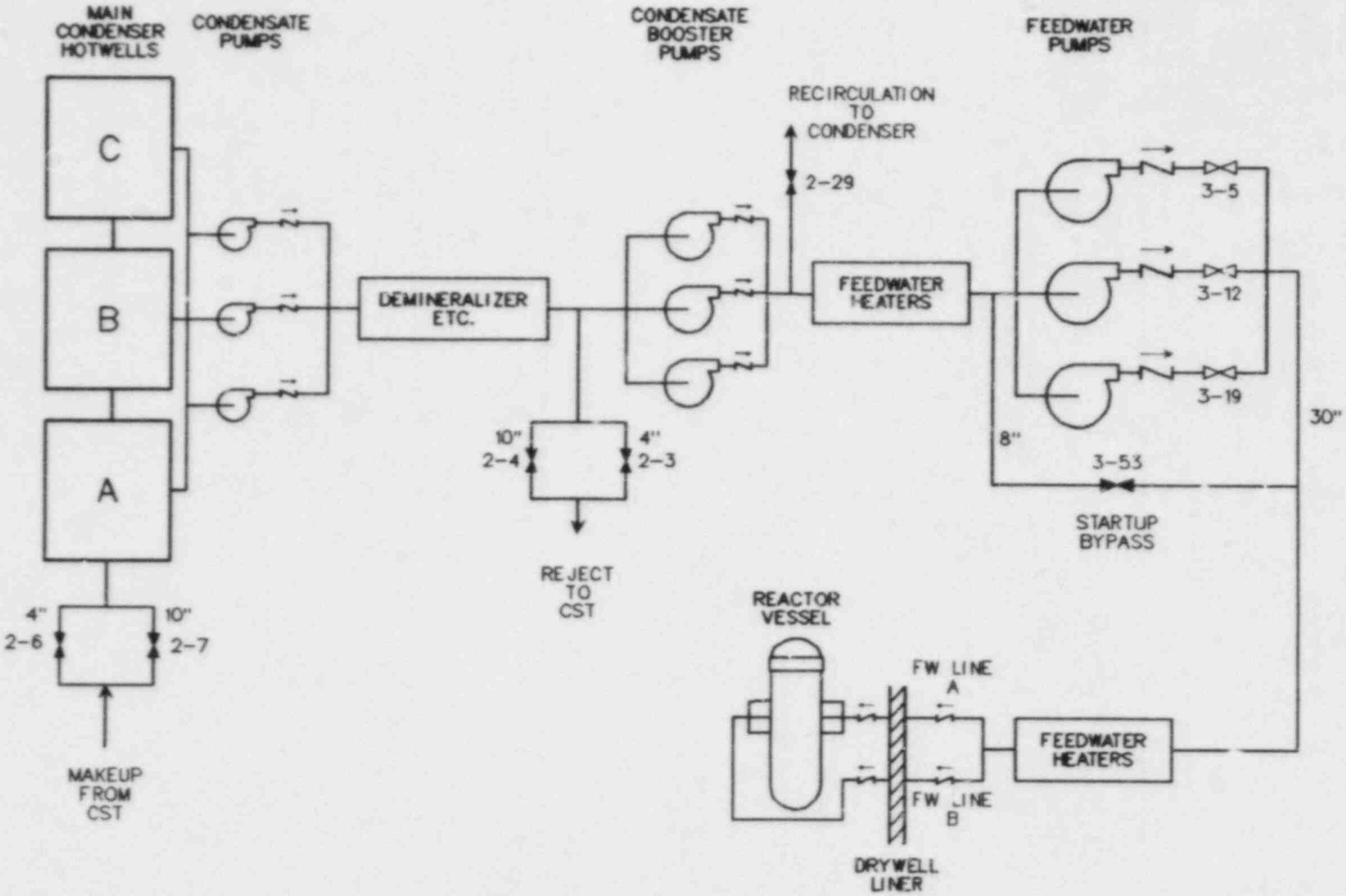

Fig. 3.8. Schematic of Browns Ferry condensate and feedwater systems.

4. MSIV-CLOSURE INITIATED ATWS WITH OPERATOR ACTION

The progression of MSIV-closure initiated ATWS accident sequences in which operator actions play a dominant role in determining the sequence of events is the subject of this chapter and of the following Chap. 5. In this chapter, the event sequences are established for several cases in which the plant operators carry out their provided written emergency instructions exactly. Some of the cases analyzed involve consideration of equipment malfunction such as stuck-open relief valves, inoperability of pressure suppression pool cooling, and failure of sodium pentaborate injection or manual rod insertion. In Chap. 5, recommendations are made concerning special procedures for mitigation of the ATWS accident sequence and for avoidance of the difficulties that are demonstrated in the sequences presented in Chap. 4.

The emergency procedures considered in this study are taken from the BWR Owners Group Emergency Procedures Guidelines. Although these procedures have not yet been implemented at Browns Ferry, the TVA has indicated that it intends to do so in the near future. The procedures are, of course, being modified as necessary to fit the specific Browns Ferry design and setpoints. Every effort has been made, after consultation with TVA engineering personnel, to incorporate the Browns Ferry-specific modifications into the calculations used in this study.

4.1 Basic Considerations for Operator Action

The control room operators would recognize the initiation of an ATWS by the existance of a combination of scram signals, continued indication of reactor power on the average power range monitors, and continued indication that multiple control rods remained in their fully withdrawn positions. (Control rod positions are prominently displayed upon a large core mockup on the front panel of the control room.) Before beginning the actual analyses of sequences with operator action, it is well to review the basic phenomenology and the plant equipment control logic that would determine the efficacy of the operator actions. This important information can be divided into four areas based upon the four goals of operator action. These are: reactivity control, reactor vessel water level control, reactor vessel pressure control, and pressure suppression pool temperature control. Each of these is discussed in turn in the following subsections.

4.1.1 Reactivity control

Given a case in which the reactor does not scram automatically following an MSIV closure event, operator action to assert reactivity control by mechanically inserting neutron absorbing poison into the core can be attempted in three ways. These are: (1) to provide a manual scram, (2) to manually insert (drive in) the withdrawn rods, or (3) to inject a liquid neutron-absorbing solution into the reactor vessel by manual initiation of the standby liquid control system (SLCS). Successful outcome of the first method would be most desirable because a manual scram would immediately terminate the ATWS accident sequence and return the reactor to a normal shutdown configuration.

Manual scram and manual insertion of control rods both involve operation of the control rod drive hydraulic system (CRDHS). This system and its modes of operation have been described in detail in a previous report.[4.1] The brief discussion provided here is focused on the considerations involved in attempted manual recovery from an ATWS.

The CRDHS is shown schematically in Fig. 4.1. A scram is accomplished by opening the scram inlet and outlet valves for each of the 185 CRD mechanism assemblies. Each open scram inlet valve permits discharge of the associated scram accumulator into the below-piston volume of the associated CRD mechanism assembly. Each open scram discharge valve provides a pathway for flow from the above-piston volume into the scram discharge volume, which is common to all of the 185 mechanism assemblies.* Thus, with pressurized water below the piston and a vented volume above the piston, each control rod is driven upward into the core when the scram inlet and outlet valves are opened.

The scram inlet and outlet valves are air-operated globe valves, held closed by control air pressure during normal operation and snapped open by internal springs when the air pressure is removed. A schematic of the control air supply to the air-operators of these valves is provided in Fig. 4.2. As shown, the control air pressure is transmitted through the solenoid-operated backup scram valves and scram pilot valves.

There are two solenoid-operated scram pilot valves associated with each scram inlet and scram outlet valve pair, each energized from a separate reactor protection system (RPS) bus (A or B) to remain in the position shown in Fig. 4.2 during normal operation. When a scram signal is received, both scram pilot valve solenoids are deenergized by the RPS and both scram pilot valves reposition so that the air operators of the scram inlet and the scram outlet valves are vented to atmosphere, permitting these valves to be opened by their internal springs.

The backup scram valves are not intended to function as an alternate means of providing rapid scram of all control rods but do provide assurance that air pressure would eventually be removed from the air operators of the scram inlet and outlet valves as protection from a common cause failure of the scram pilot valves. During normal reactor operation, the backup scram valve solenoids are deenergized and the valves are aligned as shown in Fig. 4.2. Both RPS channels A and B must trip to energize any or all of the backup scram valve solenoids but when this occurs, the backup scram valves realign so as to vent the control air lines leading to the scram pilot valves. Although all of the backup scram valves actuate whenever both RPS channels trip, the operation of any one of these valves would be sufficient to vent the air from the

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*The scram discharge volume is comprised of an east bank and a west bank of interconnected six inch headers that drain into a common scram discharge instrument volume. See Figs. E.6 and E.7 of Ref. 4.1.

supply line and accomplish a scram. However, any scram accomplished solely through action of the backup scram valves would require from 15 to 20 s because of the large volume of air that would have to be vented through the small valve ports.

It is clear that the first goal of the operator, when attempting to manually force a scram under ATWS conditions, must be to vent the air from above the air operators of the scram inlet and scram outlet valves. To this end, the plant emergency operating instructions direct the operator to press the manual scram buttons on the control room panels (one for each RPS channel) since perhaps the ATWS is due to the failure of the automatic scram signal to trip both RPS channels. If the manual scram buttons also do not produce a successful scram, then procedures call for an auxiliary operator to be dispatched to the auxiliary instrument room where a mockup panel of the reactor core provides individual toggle switches for each control rod to permit testing of the scram function. Thus the reactor might be scrammed from the auxiliary instrument room, one rod at a time.

Conversations with Browns Ferry control room operators reveal that they are well aquainted with the need to vent the air from the scram pilot valve operators under ATWS conditions. The operators indicate that if all of the previously mentioned steps failed, they would consider using the control room switch that shuts off the control air supply to the reactor building and venting the downstream piping of the scram protection system with a hacksaw.

It is of course possible that the failure-to-scram would occur even though the air had been vented from the scram pilot valve air-operators. A "water-lock" on the CRD mechanism assembly drive pistons would occur if the scram discharge volume were full at the inception of the scram so that the water volumes above the CRD mechanism assembly drive pistons could not be vented. That this is possible is proven because this was the cause of the June, 1980 partial failure-to-scram at Browns Ferry Unit 3.*[4.2]

The scram discharge volume (SDV) is vented and drained during normal reactor operation. When a scram occurs, the SDV vent and drain valves are automatically shut by action of the scram dump valves shown in Fig. 4.2 (see the discussion in Ref. 4.1). The purpose of this is to contain the onrush of water from above the CRD mechanism assembly drive pistons within the scram discharge volume and thereby build up a backpressure equal to reactor vessel pressure. Otherwise, leakage past the CRD mechanism assembly seals would provide a continual source of water into the SDV drains after the reactor has scrammed. When the scram condition has cleared and the scram logic is reset by the operator, the scram outlet (and inlet) valves are automatically closed and the SDV is again isolated from the reactor vessel, vented, and drained.

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*It should be noted that extensive piping modifications have been implemented at Browns Ferry to ensure that the particular cause of this incident, the "water-lock" in the scram discharge volume, will not happen again.

Existing emergency operating instructions direct the control room operator to attempt scram reset when confronted by an ATWS situation. The purpose of the scram reset is to open the SDV vent and drain valves in an attempt to drain the SDV so that the above-piston volumes of the CRD mechanism assemblies can indeed be vented on the next attempt at manual scram. The difficulty with this is that scram reset will not function unless the condition calling for the scram has cleared. For example, if the original scram signal were generated by high drywell pressure and an ATWS situation ensued, the scram signal could not be reset and the SDV could not be vented and drained until the drywell pressure was restored to a level below the scram signal setpoint. Thus, for a bona-fide scram signal, there is only one chance at a successful scram until the condition that caused the scram signal has cleared.*

Manual insertion (drive-in) of control rods might succeed where all attempts at scram have failed. Control rod insertion is always performed one-rod-at-a-time and, as shown on Fig. 4.1 (imagine both "insert" valves open), the control rod is moved inward without recourse to the SDV because the water displaced from the above-piston volume is dissipated into the exhaust header and from there back into the cooling header and fed into the below-piston volumes of the 184 mechanism assemblies of the control rods not being moved. Thus control rod insertion can succeed where scram has failed because of malfunction of the scram system.

The disadvantage of a reactor scram achieved by manual rod insertion lies in the time required for its achievement. BWR control rod placements for criticality and power operation vary between one-half of the control rods fully withdrawn to all of the control rods fully withdrawn. Thus between 92 and 185 rods would have to be driven in given an ATWS situation in which manual rod insertion (MRI) was the only recourse. Maximum rod speed is about 3 in./s and one fully withdrawn rod (144 in.) would require about 48 s for complete insertion. Thus at the end of core life with all rods withdrawn, about 2 1/2 h would be required until all control rods were completely inserted into the core.

Of course, it would not be necessary to manually insert all control rods in order to achieve hot shutdown. Depending on the total number of rods initially withdrawn and the particular order of insertion selected by the operators, simulator studies indicate that hot shutdown can be achieved by the manual insertion of as few as 25 control rods. This requires that the fully withdrawn high-worth rods near the center of the core be selected for initial insertion and could be accomplished in about 20 min.

Although manual rod insertion is a poor substitute for scram, it offers an effective mitigating effect in ATWS situations because continued criticality requires that the moderator temperature and void

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*It should be noted that in the case of the June, 1980 partial failure-to-scram at Browns Ferry Unit 3, a manual scram attempt was involved and therefore the scram signal could be reset as often as necessary for repeated scram attempts.

fraction in the core be reduced to offset the negative reactivity introduced by the rod insertion and ultimately, the same effect as a scram is achieved.

The final means for the operator to insert negative reactivity by mechanical methods is provided by the SLCS. This system (Fig. 4.3) employs positive displacement pumps and is designed to permit the injection of a sodium pentaborate solution into the reactor vessel at a rate of 56 gpm (0.0035 m^3/s) via the single sparger shown in Fig. 2.2. As discussed in Sect. 2.1, complete dispersal of the injected poison upward into the core region is not expected to occur unless there is a core inlet flow sufficient to induce turbulent mixing in the reactor vessel lower plenum. For this reason, the BWR Owners Group Emergency Procedure Guidelines provide for a large core inlet flow to be restored by use of the ECCS systems after sufficient poison for shutdown has been injected. This is effected by directing the operator to raise the previously depressed water reactor vessel water level back up to the normal operating range.

The operation of the SLCS pumps and the associated explosive valves is accomplished from the control room by means of a keylock switch located on the front panel. The switch has three positions, "start pump A," "off," and "start pump B." When the operator turns the switch to the "start pump A" position, pump A starts and both explosive valves fire to open the injection path to the reactor vessel. A nearby control panel instrument permits the operator to observe a decreasing level in the standby liquid control tank at the pump suction and sensed flow downstream of the explosive valves illuminates an indicating light. If the "A" pump fails to start, the operator can turn the keylock switch to the "start pump B" position. It should be noted that both pumps cannot be operated simultaneously.*

At an injection rate of 56 gpm (0.0035 m^3/s), it would take about 81 min to pump the total volume of 4550 gals (17.22 m^3) of sodium pentaborate solution from the storage tank into the reactor vessel. However, the reactor can be brought to hot shutdown more quickly than this since the amount of poison contained in just 21.3% of the tank volume is sufficient for this purpose. Specifically, after 17.27 min of injection, 212 lbs (96.2 kg) of sodium pentaborate would have entered the reactor vessel; if the reactor vessel is subsequently flooded back to its normal water level, containing 14,785 ft^3 (418.7 m^3) of solution, the sodium pentaborate concentration (assumed to be uniform) would be 320 ppm and this is sufficient for hot shutdown. It is expected that the Browns Ferry procedures currently in preparation will call for an injection time of 25 min before reactor vessel refill to provide allowance for lower-than-design injection rates and imperfect mixing.

Recent changes to the Browns Ferry emergency operating instructions have made the initiation of the standby liquid control system mandatory

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*The operator training manual for Browns Ferry explains that this is to provide more time for mixing and thereby reduce the possibility of reactivity "chugging" in the core.

under either of the following conditions:

1. Five or more adjacent control rods not inserted below 06 position* and either reactor water level cannot be maintained or suppression pool water temperature limit of 110°F is reached.
2. Thirty or more rods not inserted below 06 position and either reactor water level cannot be maintained or suppression pool water temperature limit of 110°F is reached.

The Shift Engineer or Assistant Shift Engineer is responsible for the decision to initiate the SLCS, but the written procedure permits the unit operator to take this action if the Shift Engineer and Assistant Shift Engineer are not available.

4.1.2 Reactor vessel level control

As discussed in Sect. 2.3, the high-pressure injection systems perform the role of feedwater pumps during an ATWS accident sequence initiated by MSIV closure and the combined rate of injection of the HPCI, RCIC, and CRDHS pumps determines both the reactor vessel water level and the core thermal power. The relation between core power, downcomer water level, and rate of injection is complex because, with the recirculation pumps tripped, the core inlet flow depends on the amount of natural circulation within the reactor vessel and this is a function of the downcomer water level and the power. (See the discussion in Appendix B.)

The results of calculations performed with the BWR-LACP code to investigate the effect of downcomer water level upon core thermal power and core inlet flow under ATWS conditions are shown in Figs. 4.4 and 4.5. It is emphasized that the calculations represent steady state conditions. For example, the highest downcomer water level used for the calculations was 561 in. (14.25 m) above vessel zero, which corresponds to the water level during normal reactor operation. The results shown on Fig. 4.4 indicate that if the high-pressure injection systems could supply enough water to maintain the downcomer water level at this height under ATWS conditions, then the corresponding steady-state core thermal power at normal reactor pressure would be 113%. That the power level would be higher under ATWS conditions with the recirculation pumps tripped than under normal operating conditions at the same water level is because the high-pressure injection systems inject relatively cold water [about 90°F (305 K)] from the CST whereas under normal conditions, feedwater is heated to about 377°F (465 K) before entering the reactor vessel.

The results shown in Fig. 4.5 indicate that the core inlet flow induced by natural circulation decreases as the downcomer water level is lowered and this is the cause of the steady decrease in power level shown on Fig. 4.4 as the water level is lowered from 561 to 500 in. (14.25 to 12.70 m). There is a discontinuity in the power curves as the

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*This is equivalent to 18 in. (0.46 m) of rod withdrawal. Total rod travel is 144 in. (3.66 m).

downcomer water level is lowered below 500 in. (12.70 m); this is caused by the uncovering of the feedwater spargers.

Most of the injection delivered by the high-pressure systems is provided by the HPCI and RCIC systems, which inject into the reactor vessel via the feedwater lines. The location of the feedwater spargers within the reactor vessel is shown in Fig. 4.6. As long as the downcomer water level is above the feedwater spargers, then the relatively cold injected flow is mixed with the other water in the downcomer, maintaining a relatively low temperature at the core inlet. When the downcomer water level is below the feedwater spargers, however, the injected flow is sprayed into a steam atmosphere by the nozzles in the feedwater spargers. This, in effect, provides feedwater heating and the temperature of the flow at the core inlet increases significantly. This effect produces the marked decrease in steady-state power level under ATWS conditions as the downcomer water level is lowered below 500 in. (12.70 m) as shown on Fig. 4.4.

The large calculated effect of uncovering the feedwater spargers depends upon the assumption that the HPCI and/or RCIC flow leaving the spargers is in the form of a spray with the associated large surface area that promotes efficient heat transfer with the surrounding steam. It should be noted, however, that considerations such as these are only important when one attempts to calculate steady-state reactor power as a function of reactor vessel water level. As discussed in Appendix B, the calculation of core thermal power as a function of the rate of injected flow is simple and straightforward.

The BWR Owners Group Emergency Procedures Guidelines take advantage of the effect of downcomer water level upon reactor power under ATWS conditions by instructing the operator to reduce vessel injection as necessary to lower the downcomer water level to the top of the core. As shown on Fig. 4.5, all natural circulation of water from the core region to the downcomer is stopped when the downcomer level is this low, so the core inlet flow consists only of the injected flow from the high-pressure systems plus the steam condensed within the reactor vessel. In this phase of operation, the steaming rate from the core significantly exceeds the steam flow from the reactor vessel because of the large rate of steam condensation in the vicinity of the feedwater spargers.

As shown in Fig. 4.4, the core thermal power is about 9% with the downcomer water level lowered to the top of the core and with the reactor vessel fully pressurized. The corresponding core inlet flow (Fig. 4.5) is less than 2% of that at normal full power operation. This certainly would not be enough flow to sweep the sodium pentaborate injected by the SLCS into the core. Accordingly, the BWR Owners Group Emergency Procedure Guidelines specify that the operator should restore the reactor vessel water level to the normal operating level after the amount of sodium pentaborate required for hot shutdown has been injected. This involves a period of rapid injection and restores natural circulation at decay heat levels, thus promoting the entry of the liquid poison into the core and its subsequent mixing.

It is important to consider the reactor vessel water level instrumentation available for the operator's use when he or she is attempting to maintain the water level at the top of the core. The two ranges of available instrumentation are illustrated in Fig. 4.7. The Emergency

Systems instruments are calibrated for normal operating temperatures and pressures and the range extends down to 373 in. (9.47 m) above vessel zero or 13 in. (0.33 m) above the top of the core. The "Post Accident Flooding Range" extends almost to the core midplane, but is calibrated for LOCA conditions, i.e., atmospheric pressure.

It seems that the operator would desire to maintain level indication on the more accurate Emergency Systems range and therefore would actually control downcomer water level at about 380 in. (9.65 m), or slightly above the top of the core. Table 4.1 indicates the magnitude of level indication differences between the two available instruments. The indicated level on the Post Accident Flooding instruments is too low when the reactor vessel is pressurized. With an actual level of 380 in. (9.65 m), the Emergency Systems indicated level would be 380 in. and the Post Accident Flooding indicated level would be 337 in.

One final consideration concerning reactor vessel level control under ATWS conditions remains to be discussed. It is expected that the HPCI system would be lost in an ATWS accident sequence that involved excessive pressure suppression pool temperatures unless the operator takes extraordinary action to prevent the shift of the HPCI pump suction to the pressure suppression pool by racking out the breakers to the valve motor operators for the suction valves from the pool. With the HPCI system failed, the capacity of the remaining high-pressure injection systems (RCIC and CRDHS) is insufficient to maintain the reactor vessel downcomer water level at the top of the core. Accordingly, if the water level is to be maintained at the top of the core, the operator must at least partially depressurize the reactor vessel and use a low-pressure injection system.

It seems that the easiest and safest course for the operator would be to turn off two condensate pumps and two condensate booster pumps and use the remaining condensate pump-condensate booster pump combination for reactor vessel injection. As indicated on Fig. 3.8, startup bypass valve 3-53 provides a bypass path around the idle feedpumps. Thus the operator can shut the feedpump discharge valves 3-5, 3-12, and 3-19 and provide a controlled injection into the reactor vessel by throttling valve 3-53. As indicated on Table 3.2, injection by this means is possible whenever reactor vessel pressure is below 418 psia (2.88 MPa).

A second way to provide controlled reactor vessel injection using a low-pressure system would be to use one loop of the core spray system. As an example for the loop containing pumps A and C as shown in Fig. 4.8, valve 75-25 is a throttle valve which can be operated from the control room when the reactor vessel pressure is less than 465 psia (3.20 MPa). As indicated in Table 3.2, the core spray pumps can begin injection into the reactor vessel when the vessel pressure falls below 357 psia (2.46 MPa). At higher reactor vessel pressures, the running core spray pumps would be protected by minimum flow lines (not shown on Fig. 4.8) which open to permit flow from the pump discharge to the pressure suppression pool when the total loop flow is less than 600 gpm (0.038 m^3/s).

The BWR Owners Group Emergency Procedures Guidelines recommend use of the Core Spray system for reactor vessel level control under ATWS conditions only if the level cannot be maintained by the high-pressure injection systems, the condensate and feedwater systems, or the LPCI mode of the RHR system. This is because of the unknown phenomenology associated with the spraying of large amounts of water onto the top of a partially uncovered core under ATWS conditions.

The third way to provide reactor vessel water level control with a low-pressure injection system would be to use a portion of the RHR system. This method is more complicated than either of the two methods previously discussed, but can be explained with reference to Fig. 4.9, which shows one loop of the RHR system. Under ATWS conditions, this system would be expected to be employed in the pressure suppression pool cooling mode, with the flow from the outlet of the heat exchangers returning to the pressure suppression pool through valves 74-71 and 74-73 shown on Fig. 4.9. It is evident that reactor vessel injection can occur simultaneously if valves 74-66 and 74-67, associated with the LPCI mode of RHR system operation, are opened.

LPCI outboard injection valve 74-66 and LPCI inboard injection valve 74-67 cannot both be opened from the control room unless the reactor vessel pressure is less than 465 psia (3.20 MPa) and, as indicated on Table 3.2, the shutoff head of the RHR pumps is such that vessel injection cannot occur until reactor vessel pressure falls below 346 psia (2.39 MPa). If the LPCI mode of the RHR system is automatically initiated,* then throttle valve 74-66 is interlocked to full open for 5 min. This would be expected to occur in an ATWS accident sequence if the reactor pressure falls low enough to permit injection by the RHR system because the other prerequisite for automatic initiation, a high drywell pressure signal, would be generated by evaporation from the heated pressure suppression pool earlier in the sequence. With the LPCI injection valves full open, reactor vessel flooding could only be prevented by turning off the RHR pumps during the 5 min period until valve 74-66 can be throttled.

4.1.3 Reactor vessel pressure control

Without operator action, the reactor vessel pressure would be determined by automatic SRV operation. Each SRV has a capacity equivalent to about 6.5% of full reactor power. Therefore, for example, if the reactor were generating 29% of full steam flow in an ATWS accident situation with the MSIVs closed, four SRVs would remain open passing 26% of full steam flow to the pressure suppression pool and a fifth SRV would cycle, being open about half of the time, with the reactor vessel pressure alternately rising and falling over its abbreviated blowdown range.

It is important to recognize that this presents a very unusual situation to the control room operator if he attempts to establish manual pressure control. The operator has no indication as to which of the SRVs are open as a result of reactor vessel pressure exceeding their setpoints for automatic actuation. If the operator acts to open an SRV

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*Automatic initiation occurs for (1) reactor vessel low level at 414 in. (10.52 m), or (2) drywell pressure high at 2.5 psig (0.119 MPa) and low reactor vessel pressure at 465 psia (3.20 MPa).

that is already open, nothing will happen. If the operator happens to select a shut SRV and opens it, the reactor vessel pressure will decrease slightly and one of the previously open SRVs will close; the net result is that the same number of SRVs are open and the reactor vessel pressure is about the same. Using the example of the previous paragraph, the operator's actions would not have any significant effect on reactor vessel pressure until he or she had manually opened five SRVs. This would be very confusing to operating personnel accustomed to rapid response to manual pressure control.

Furthermore, continuing the example, once the fifth SRV is manually opened, the reactor vessel pressure would suddenly begin to decrease very rapidly. This is because decreasing pressure increases the voiding in the core region, inserting negative reactivity and reducing core power. This reduces the reactor steam generation to significantly less than the capacity of the five SRVs being manually held open, which causes an increased rate of pressure decrease, further reducing core power and so forth. If the operator is not quick to act, the reactor vessel will depressurize to the point where the low pressure injection systems can flood the core, causing power and pressure spikes similar to those seen in the no-operator-action case discussed in Chap. 3.

The operator can prevent reactor vessel flooding by the low pressure systems by the simple expedient of turning the condensate booster pumps off and by turning the core spray and RHR pumps off immediately after these low-pressure ECCS systems are automatically actuated.* However, it is important to recognize that a power and pressure spike can still occur if the reactor vessel is sufficiently depressurized. The reason for this can be understood by consideration of the information presented in Table 4.2. As indicated, the change in vapor specific volume per unit change in pressure at 100 psia is 92.5 times that at 1050 psia. It follows directly that a given increase in pressure will have a much greater effect in reducing the amount of voiding in the core when the reactor vessel is at low pressure. Thus if the operator manually opens enough valves to depressurize the reactor vessel under ATWS conditions and then closes the valves when the reactor vessel is at low pressure, a power and pressure spike will be initiated by the small pressure increase that occurs at the time the valves are closed. The initial pressure increase collapses voids in the core, inserting positive reactivity and increasing reactor power. This increases the steam generation which in turn further increases the reactor pressure, and so forth.

Power spikes are undesirable because they challenge the integrity of the fuel or cladding and they would confuse the operator. Pressure spikes can be contained without threatening reactor vessel integrity by action of the SRVs and by the effect of the negative reactivity introduced by increasing power as additional voids are created in the core, which turns the power while the vessel pressure remains near the relief valve setpoint. Nevertheless, pressure spikes under ATWS conditions

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*The core spray and RHR system pumps cannot be prevented from automatically starting when the ECCS initiation signal is first received. After they have started, they can be turned off and will remain off.

would pose a serious challenge to the integrity of the primary system. This is because, although the injection valves separating the low-pressure piping of the low-pressure ECCS systems from the reactor vessel are interlocked to prevent opening until the reactor vessel pressure has been lowered to safe levels, there is no provision for automatic reclosure of these valves if the reactor vessel pressure subsequently increases. Although the installed check valves (Figs. 4.8 and 4.9) should protect the low-pressure ECCS piping from sudden pressure spikes in the reactor vessel, the potential for a LOCA outside of containment would obviously be increased with the injection valves open under the conditions of an ATWS accident sequence that involved reactor vessel depressurization and subsequent pressure spikes.

It is unfortunate that manual pressure control is so difficult and so likely to result in power and pressure spikes under ATWS conditions because, as shown on Fig. 4.4, for the same downcomer water levels, the steady-state reactor power is lower at lower reactor vessel pressures. The reduction in power as the pressure is lowered is primarily due to the increased voiding in the core at low pressures and the effect is greatest at high downcomer water levels. With a downcomer water level of 380 in (9.65 m), just 20 in. (0.51 m) above the top of the core, the steady-state power with the reactor at pressure would be about 9%. If the reactor pressure could be held at 250 psia (1.72 MPa), the thermal power would be about 5% and if the reactor pressure were 100 psia (0.69 MPa), the thermal power (including decay heat) would be only about 3 1/2%. Although the differential reduction in steady-state power obtained by lowering reactor vessel pressure from 1020 to 100 psia (7.03 to 0.69 MPa) is only 5 1/2%, the effect on the progression of the accident sequence would be very significant, because the pressure suppression pool cooling system can remove the equivalent of 3 1/2% power from the pool*, but could not prevent a continucus pool temperature increase if the reactor remains at 9% power.

The BWR Owners Group Emergency Procedures Guidelines would lead the operator to attempt manual reactor vessel depressurization under ATWS conditions if the "Heat Capacity Temperature Limit," based on the temperature of the pressure suppression pool is exceeded. The curve defining this limit for the Browns Ferry plant is shown in Fig. 4.10; combinations of pressure suppression pool temperature and reactor vessel pressure that would be represented by plotted points within the shaded area are prohibited. These limits require that reactor vessel depressurization begin when suppression pool temperature exceeds 160°F (344 K) and that reactor vessel pressure must be less than 115 psia (0.79 MPa) whenever suppression pool temperature exceeds 200°F (366 K).

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*With the pressure suppression pool at elevated temperature, the heat removal capacity of the RHR system heat exchangers is increased. A "rule of thumb" is 0.283 MW_t per °F temperature difference per heat exchanger. For a service water temperature of 80°F and four heat exchangers in operation, the heat removal rate would reach 3 1/2 % power (115 MW_t) when the pressure suppression pool temperature reached 182°F.

None of these limitations are based specifically upon ATWS considerations, but were chosen to ensure smooth condensation of the steam released by the SRV T-quencher devices without the imposition of significant loads on the containment. The basis of the 200°F limit is documented in the NRC report NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments." This is a conservative limit because it takes into account only known experimental data and does not recognize that containment back pressure increases the boiling point of the water in the suppression pool. For almost any transient in which the suppression pool temperature reached 200°F (366 K), there would be significant pressurization of the primary containment above atmospheric pressure. Nevertheless, since nothing in the written procedures proscribes the applicability of the heat capacity temperature limit curve under ATWS conditions, and because the pressure suppression pool temperature rapidly increases, it must be expected that the operators, following the Emergency Procedure Guidelines, would attempt manual reactor vessel depressurization.

Before proceeding to the general subject of pressure suppression pool cooling, it is interesting to note from Fig. 4.5 that core inlet flow is actually higher at a reactor vessel pressure of 100 psia (0.69 MPa) then it is at higher pressures although from Fig. 4.4, the core thermal power is lower. Since core thermal power increases with core inlet flow under ATWS conditions, all other considerations remaining equal, it is instructive to consider the cause behind this observation.

The core inlet flow is higher at very low pressures because the height of the two-phase mixture within the core shroud and steam separator assembly necessary to balance the weight of the water in the downcomer region is much higher, so high in fact that liquid carryover from the inner region to the downcomer region is restored. Yet the counteracting effect of increased voids in the core at very low pressure is predominant and the reactor power is lower.

4.1.4 Pressure suppression pool temperature control

Pressure suppression pool cooling would be urgently needed should an ATWS accident sequence actually occur, since the pool would be receiving steam via the SRVs at levels far exceeding the design basis for the pool cooling system. It seems direct and simple to help in this regard by procedures that require the operator to institute pressure suppression pool cooling whenever the pool temperature exceeds a certain limit. This is done, but certain interlocks and RHR system logic designed to enhance the probability of plant recovery from LOCA would dramatically interfere.

If the operator simply places the RHR system into its pressure suppression pool cooling mode early in the ATWS accident sequence, the system would automatically realign into the LPCI mode when the operator, following the Emergency Procedure Guidelines, lowered the water level to the top of the core. The operator would be expected to again take the system into the pressure suppression pool cooling mode. While the operator attempts to maintain the water level at the top of the core, simulator exercises and the results presented later in this chapter show that the sensed water level would fluctuate. If the fluctuating reactor vessel water level dropped as low as 2/3 core height, the RHR system would again automatically realign from pressure suppression pool cooling into the LPCI mode.

Established procedures do not now call for this, but the operator could circumvent the need to continually restore pressure suppression pool cooling, by moving control room switches into the "containment spray select" and "2/3 core coverage bypass" positions upon first understanding that an ATWS was in progress and while initially aligning the RHR system into its pressure suppression pool cooling mode. These actions would ensure that the RHR system would remain in its pressure suppression pool cooling mode but would have no effect on the LPCI system injection valves to the reactor vessel, which would open and remain open if reactor vessel pressure dropped to 465 psia (3.21 MPa). The situation of pressure suppression pool cooling flow with a large portion diverted into the reactor vessel would occur if the vessel pressure dropped below 346 psia (2.39 MPa) since the throttle valve for injection to the reactor vessel, once opened, is interlocked open for 5 min.

4.2 Operators Follow the Emergency Procedure Guidelines

This section and Sect. 4.3 report the results of BWR-LACP calculations of MSIV-closure initiated ATWS transients with operator action per Revision 3 of the General Electric BWR Owners Group Emergency Procedure Guidelines (EPGs) (Ref. 4.3). Just as for the calculations reported in Chap. 3, these calculations were initialized 50 s after the beginning of the MSIV closure ATWS accident. The assumption is made that none of the initially withdrawn control rods enter the reactor core as a result of the initial or subsequent scram attempts.

4.2.1 Systems function as designed

Figures 4.11—4.15 show important system variables for this accident sequence. Table 4.3 summarizes significant events and operator actions. Operator actions are to initiate SLC system injection of sodium pentaborate solution, to manually insert the control rods, and to initiate the pool cooling mode of the RHR system. These operator actions significantly mitigate this accident. After 35 min the reactor is shut down to decay heat power; the peak suppression pool temperature attained during the accident sequence is only 157°F (343 K).

At the beginning of the calculation, the thermal power generation in the reactor core (Fig. 4.11) is in the neighborhood of 25% [i.e., 823 MW(t)]. The CRDHS (which runs continuously unless tripped by the operators) is injecting about 106 gpm (6.7 l/s) from the CST into the reactor vessel. (The CRDHS runs continuously throughout all the cases examined in this chapter.) The reactor vessel is fully pressurized, cycling between about 1100 psig and 1000 psig (7.69 and 7 MPa) in response to the automatic opening and closing of the SRVs (Fig. 4.14). Water level in the reactor vessel downcomer annulus (Fig. 4.12) is at 500 in. (12.7 m) above vessel zero, but is decreasing rapidly.

When the water level reaches 476.5 in. (12.1 m), the HPCI and RCIC systems actuate automatically and are soon injecting at full capacity—600 gpm (37.8 l/s) for RCIC and 5000 gpm (315 l/s) for HPCI. The water level stops decreasing, then increases slightly, until the total vessel injection is equivalent to the steam production from the reactor vessel. After reaching this quasi-equilibrium, the vessel water level fluctuates about a mean value of 476 in. (12.1 m) in response to the fluctuating vessel pressure. The EPG level control guideline requires no immediate operator action to adjust water level at this time.

The power control guideline of the EPGs requires that operators attempt to bring about an alternative scram by one of the means discussed in subsection 4.1.1 of this report. If successful, this would quickly shut down the reactor and end the accident sequence. The operators would surely attempt alternative scram before beginning either the manual rod insertion of control rods or SLC injection of sodium pentaborate solution; however, all the calculations of this chapter assume that the alternative scram does not occur.

The manual insertion of control rods begins at 3 min. This assumed time is based on observation by ORNL investigators of operator response during simulated ATWS accidents at the TVA Browns Ferry training simulator. There is no immediate effect on reactor power because only one rod can be inserted at a time [at a speed of 3 in./s (7.62 cm/s)] and each control rod is assumed to be worth only about 0.001 $\Delta K/K$ (see Appendix A for details on the modeling of manual rod insertion).

With reactor power between 20 and 30%, the operators would be aware of the impending need to initiate the SLC system injection of sodium pentaborate solution. The EPG power control guideline requires initiation of the SLCS if the suppression pool temperature exceeds 110°F (317 K) and the reactor is not shutdown. The bulk pool temperature (Fig. 4.15) exceeds this threshold after only 2 min, but, based on observation of operator response to ATWS at the TVA Browns Ferry training simulator, it is assumed that the operators would probably spend several more minutes trying to obtain an alternative scram of the control rods. This calculation assumes that the SLC system is initiated after 5 min, beginning the injection of sodium pentaborate solution into the reactor vessel.

If boron injection is required, the EPG power control guideline requires that the operators follow Contingency #7, "Level/Power Control," and reduce the reactor vessel water level to near the top of the active fuel (TAF). The operators, in accordance with Contingency #7, trip the HPCI and RCIC systems at 7 min.* The water level in the reactor vessel downcomer annulus (Fig. 4.12) decreases rapidly and soon is below the minimum indication of the Emergency Systems Water Level Indication (see Fig. 4.7), but about 4 in. (10.2 cm) above the TAF. The HPCI system is restarted, initially at about 40% of capacity [2000 gpm (126 l/s)], to

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*The intent of EPG Contingency #7 could be achieved by smoothly reducing the HPCI and/or RCIC flow over a period of one or two minutes, and this might be preferable as it would avoid reliability problems that might accompany intermittent HPCI/RCIC turbine operation.

rapidly bring the level back on-scale. After coming back on-scale, the level continues to increase. The operator cuts the HPCI system flow back to about 20% of capacity, but level continues to increase until the operator again trips the HPCI system. Afterwards, the HPCI system is restarted whenever required to keep water level near the TAF, but above the minimum indication of the Emergency Systems Level Indication.

The BWR-LACP simulation of operator control of vessel water level using the HPCI system assumes that the operator will check vessel water level once per minute and adjust the HPCI flow between 20 and 40% of full capacity in accordance with the following rules (see also Appendix A.3.2):

1. If level is more than 5 in. (12.7 cm) from the setpoint, decrease or increase (as appropriate) the flow by 5% of the full HPCI capacity [i.e., by 5% of 5000 gpm (315 l/s)].
2. If level is more than 8 in. (20 cm) above the setpoint, decrease flow by 10%.
3. If level is more than 20 in. (51 cm) above the setpoint, decrease the flow to zero by tripping the HPCI turbine.
4. If the level is below the minimum range of the Emergency Systems Level Indication, increase flow by 10%.

The setpoint for vessel level control after the EPG Contingency No. 7 water level reduction maneuver is 380 in. (9.65 m), as determined by the range of the Emergency Systems Level Indication instrument. The minimum indication of this instrument is equivalent to 373 in. (9.47 m) above vessel zero.

The vessel water level reduction maneuver, the effect of manual rod insertion, and the small amount of sodium pentaborate mixed into the reactor coolant during the period of abundant natural circulation before the reactor vessel water level is lowered reduce the core power to below 5% of the rated 3300 MW thermal output of the reactor core by time 8 min.* The reactor power continues to decrease very slowly in response to the continued slow, but steady, manual insertion of control rods. The on-going injection of boron has little effect on core power during the period of about 20 min. after the reactor vessel level is lowered because most of the heavy sodium pentaborate solution collects in the bottom of the reactor vessel lower plenum. With downcomer water level near the TAF, there is little or no net recirculation of coolant from inside the core shroud, back to the downcomer annulus (via the standpipes and steam separators), and through the lower plenum to promote turbulent mixing.

Operator attempts to control reactor vessel pressure are not really necessary in this accident. The SRVs would by automatic actuation maintain vessel pressure between about 1100 and 1000 psig (7.7 and 7 MPa). However, the EPG pressure control guideline requires that, if any SRV is "cycling," the operator should manually open SRVs until pressure drops

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*As indicated on Fig. 4.14, a temporary pressure reduction caused by operator delay in closing manually-opened SRVs accompanies the level reduction. This also has an effect in reducing power.

to 935 psig (6.55 MPa). The SRVs are cycling during the first several minutes, so the operator begins manual SRV manipulations after 1 min.

The details of the BWR-LACP simulation of operator SRV control is discussed in Appendix A. The simulation allows the operator to check once per minute the vessel pressure and to open or close one SRV, or to leave the SRV status unchanged, as required in the attempt to maintain the vessel at pressure and to avoid automatic SRV actuations. The vessel pressure response plotted on Fig. 4.14 shows that the vessel pressure varies widely, and that the operator actions are not successful in preventing automatic SRV actuations. The vessel pressure fluctuations cause reactor power fluctuations, including one spike to 46% at 7 min, triggered when the operator closes a previously manually opened SRV to prevent an excessive decrease in vessel pressure.

The suppression pool temperature (Fig. 4.15) increases very rapidly at first, but the rate of increase slows markedly after the reactor power level is reduced by the water level reduction maneuver. Prior to initiating pool cooling, the operators must actuate the "Containment Spray Select" switch to prevent the automatic realignment of the RHR system from the pool cooling mode into the LPCI mode. The operators initiate pool cooling at 10 min, utilizing both loops of the RHR system (4 coolers, total). By 17 min, the coolers are removing as much heat (about 69 MW) as the SRV discharge is adding. The peak suppression pool temperature of 157°F (343 K) is reached at 17 min.

The containment response is mild in this case because the peak suppression pool temperature is relatively low and because the drywell coolers continue to run. The drywell temperature (not shown) remains at or below the 145°F (336 K) initial value. By the end of 60 min, the drywell pressure (not shown) has increased by about 1 psi (6.9 kPa), but is still below the 2.45 psig (118 kPa) threshold for ADS initiation.

This accident is effectively terminated after 30 min, when the operators initiate the HPCI system at full flow to raise reactor vessel water level and induce sufficient natural circulation to promote mixing of the boron solution which had previously settled into the bottom of the lower plenum. HPCI flow is discontinued after the vessel water level reaches 500 in. (12.7 m), but the level continues to increase slowly because of continued CRDHS injection [at 106 gpm (6.68 l/s)] and because of heating and swelling of the large volume of water added during the period of HPCI system injection.

4.2.2 Effect of stuck-open relief valves

Conditions for the accident sequences discussed in this subsection are identical to those assumed for subsection 4.2.1, except that one, or two, SRVs are assumed to stick open 3 min after the beginning of the MSIV closure. Since the operators take action to initiate the SLC system, manual rod insertion, and suppression pool cooling and, in addition, are able to prevent the unintended flooding of the reactor vessel by the low pressure high capacity injection systems (e.g. Core Spray), the outcome of this compounded accident is mild and very similar to the case without stuck open relief valves (subsection 4.2.1).

The effect of the SORVs on the system response variables of reactor power, vessel water level, and suppression pool water level and temperature is minor, so plots of these variables are not shown; specific differences are noted below. However, after the reactor vessel steam generation falls below the capacity of the SORVs, the SORVs cause the depressurization of the reactor vessel. The depressurization starts after 8 min, when the core power has been reduced from about 28% to less than 6.5% and there is no longer sufficient core steam production to continuously hold open even one SRV at full pressure. Figures 4.16 and 4.17 show vessel pressure for the cases with one and two SRVs stuck open.

The decreasing reactor vessel pressure in the SORV cases presents the hazard of large amounts of water injection from the large-capacity low pressure injection systems. As shown in Sect. 3.4 for the no-operator-action case, such vessel flooding would lead to very undesireable power and pressure excursions. The calculations of this section assume that the operators take action, as required, to prevent undesired injection.

The condensate booster pumps run continuously during normal operation and would continue to do so after initiation of this accident. They are not able to pump into the reactor vessel until vessel pressure decreases to below about 418 psia (2.88 MPa). The operators can trip these pumps at any time to prevent undesired injection. The Core Spray and RHR pumps automatically start on low vessel water level after the operator initiates the level reduction maneuver to reduce the core thermal power. The operator cannot prevent these pumps from automatically starting on low vessel level, but can turn them off at any time after they start. In the case of the RHR pumps, it is desirable, when possible, to shut the reactor vessel injection valves instead, so that the pumps can continue to run with the RHR system aligned to the pool cooling mode.

In the case with one stuck open SRV, the reactor vessel pressure (Fig. 4.16) begins to be affected after 8 min. (Before this time, the reactor core is generating enough steam to hold open more than one SRV.) By 23 min, the pressure has stabilized at 330 psia (2.28 MPa), but a full flow HPCI actuation between 30 and 35 min (initiated by the operators to raise vessel water level and promote mixing of the boron solution) causes the pressure to further decrease to 156 psia (1.08 MPa); pressure finally stabilizes at 215 psia (1.48 MPa).

The operator prevents unwanted injection from the hotwell by tripping the condensate and condensate booster pumps at any time prior to 17.5 min when the reactor vessel pressure becomes low enough to permit the CBP injection. The operator prevents Core Spray injection by tripping all four pumps anytime between 8 min [when the pumps start on vessel water level < 413.5 in. (10.5 m)] and 21 min (when vessel pressure is below the Core Spray pump shutoff head). To prevent unwanted RHR pump injection, the operator does not trip the RHR pumps, but instead, shuts the injection valves (numbers 74-66 and 74-67 on Fig. 4.9). This allows the RHR system to provide uninterrupted pressure suppression pool cooling. The outboard LPCI injection valve is automatically opened and interlocked open for 5 min after the reactor vessel pressure goes below 465 psia (3.21 MPa), but vessel pressure is high enough during this period (from 13.6 to 18.6 min) to prevent any of the flow from the running RHR pumps from being diverted from pressure suppression pool cooling and entering the reactor vessel.

The peak suppression pool temperature for the case with one SORV is 160°F (344 K), as compared to the 157°F (343 K) peak for the case with no SORVs. The difference is small because the additional energy input to the pool due to the partial depressurization of the reactor vessel is offset by the slightly lower reactor power at lower reactor vessel pressures. The effect of pressure on equilibrium reactor power is illustrated on Fig. 4.4.

For the case with two stuck open SRVs, vessel pressure (Fig. 4.17) begins decreasing after 8 min, continues to decrease until it reaches 174 psia (1.2 MPa) after about 25 min, and then is reduced further to below 100 psia (0.69 MPa) when the operators initiate the HPCI system at full flow after 30 min to raise the reactor vessel water level and promote mixing of the sodium pentaborate solution. The HPCI turbine steam supply is automatically isolated when vessel pressure decreases to below 115 psia (0.79 MPa) at 32 min; however, the 2 min of full flow before the isolation raises vessel water level enough to induce natural circulation in the vessel. The reactor vessel refill is continued at a slower rate with the RCIC system, whose operation is not compromised by vessel pressure in the neighborhood of 100 psia (0.69 MPa).

In the case with two stuck open SORVs, operator action to prevent vessel flooding by the high capacity low pressure injection systems must be accomplished more promptly because the depressurization of the reactor proceeds more swiftly than for the single SORV case. The condensate booster pumps must be tripped before 11 min, and the Core Spray pumps sometime between 8 min (i.e., after they start) and 12.5 min. The RHR pumps must also be tripped, causing a brief interruption of pool cooling. The outboard LPCI injection valve 74-66 (see Fig. 4.9) automatically opens at 11 min and is interlocked open for 5 min. If the RHR system is in the pressure suppression pool cooling mode and the LPCI injection valves are open, there will be injection into the reactor vessel if vessel pressure is below 300 psia (2.07 MPa). Vessel pressure is below this threshold after 13.7 min; therefore, the RHR pumps must be tripped until the 5 min interlock clears, and the LPCI outboard injection valve can be manually closed.

The peak suppression pool temperature for the case with two SORVs is 168°F (349 K), compared to 160°F (344 K) for the one SORV case and 157°F (343 K) for the no SORV case.

4.2.3 Sequence of events without pressure suppression pool cooling

This accident sequence is the same as the sequence discussed in subsection 4.2.1, except that it is assumed that the operators are not able to initiate suppression pool cooling. There is essentially no difference in the accident sequence or required operator actions and the reactor is brought to hot shutdown at time 35 min, as before. At the end of 60 min, the pressure suppression pool temperature (Fig. 4.18) is 167°F (348 K) and increasing slowly. Since the reactor is discharging only decay-heat-produced steam to the uncooled suppression pool at this time, it would require an additional period of about 24 h to build up enough pressure to threaten primary containment integrity (Ref. 3.2). Therefore, initiation of pool cooling anytime before the 25 h point would terminate the accident.

4.2.4 Emergency action levels and timing

The timing of the declaration of emergency action levels for the cases in which the backup shutdown systems do function as designed is specified on Table 4.4. The criteria for determination of emergency action levels are taken from the TVA Implementing Procedures Document applicable to the Browns Ferry nuclear plant.[4.4]

In the event of an ATWS accident, the operators would declare the unit to be on Alert status within minutes of the failure to scram. The Alert would, if not upgraded to a higher emergency status, remain in effect at least until a sufficient number of control rods could be inserted to enable the unit to reach a secure cold shutdown. Downgrading of the Alert to Unusual Event, or back to normal status, would be appropriate after a determination that no other conditions exist that would, by themselves, require the declaration of an emergency status. For example, minor fuel damage or primary coolant system crud burst might release enough radioactivity during the period while the reactor was being brought under control to require an Alert or Unusual Event status to be maintained for a more extended period.

The concomitant failure of pressure suppression pool cooling would require that the Alert status be continued. For the sequences discussed in Sect. 4.2, manual rod insertion and sodium pentaborate injection are effective so that the reactor is shutdown and generating only decay heat after 35 min. The ATWS accident thus would transform into a Loss of Decay Heat Removal (DHR) accident, which has been extensively studied in previous SASA investigations at ORNL.[3.2,4.5] Without suppression pool cooling (and with the MSIVs closed and the reactor on decay heat), the suppression pool temperature and, consequently, the primary containment pressure would slowly but continually increase. After about 20 h, the drywell pressure would exceed 50 psig (0.45 MPa), requiring the operators to declare the highest emergency action level, General Emergency.

Specific emergency actions necessary to protect the public health and safety after the declaration of the General Emergency would be very dependant upon the specifics of the accident sequence.* Given the large amount of time available for corrective action, it is unlikely that the accident would progress this far, but if the suppression pool cooling could not be recovered, the drywell pressure would reach the 117 psig

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*Emergency actions would also depend on other considerations not discussed in this report, such as the reactor site characteristics and even the weather conditions in effect at the time.

(0.91 MPa) static failure pressure about 25 h after the inception of the accident sequence.

As discussed in Ref. 3.2, the progression of the accident after drywell failure cannot be predicted with certainty. A large quantity of energy would be stored in the drywell prior to failure. A catastrophic drywell failure, releasing the stored energy to the reactor building in the form of steam in a short span of time, might cause a failure of the reactor vessel water injection function, leading to severe fuel damage and the release of fission products beginning about 3 h after the failure. A sufficiently catastrophic drywell failure involving movement of the drywell liner might even cause a breach in the reactor coolant system pressure boundary (LOCA) as well as failure of the reactor vessel water injection capability, leading to severe fuel damage starting only about 0.5 h after the drywell failure.

On the other hand, catastrophic drywell failure can be prevented by manual action to vent the containment, at least one vessel water injection system might remain unimpaired even if catastrophic failure did occur, or a backup source might exist that could be utilized to provide continued cooling of the fuel after the drywell failure. In these more likely cases, there would be no severe fuel damage and any release of radioactivity to the environment would be comparatively minor.

4.3 Cases in Which Backup Shutdown Systems do not Function

4.3.1 The case without manual rod insertion

Conditions for this sequence are the same as those for the sequence discussed in subsection 4.2.1 (systems function as designed), with the exception that there is no manual control rod insertion. All other systems and operator response are essentially the same, including operator action to initiate the SLC system injection of sodium pentaborate solution 5 min after the beginning of the accident. The outcome of this sequence is very similar; the details of the discussion in subsection 4.2.1 apply, except as pointed out below.

During the period between 10 and 30 min, the reactor power (Fig. 4.19) averages about two percent higher than for the case with both manual rod insertion and SLC injection (Fig. 4.11). After 30 min, the HPCI system injection is increased to full flow [5000 gpm (315 l/s)] to raise the vessel water level and effect the mixing of the sodium pentaborate solution. By 35 min, the core is subcritical and generating only decay heat. The maximum suppression pool temperature (not shown) is 164°F (347 K), occuring at 30 min. This is only 7°F (3.9 K) higher than the peak pool temperature for the case with both manual control rod insertion and SLC injection (Fig. 4.15).

4.3.2 The case without SLC system operation

For this sequence, all systems except the SLC system operate as designed. The results show that the operators can effectively shut down the reactor using manual control rod insertion, without the benefit of sodium pentaborate injection.

The reactor power (Fig. 4.20) is similar to, but noticeably higher than the power for the case with both manual rod insertion and SLC actuation. Although it takes about 62 min, by manual rod insertion alone, to add enough negative reactivity to reach a complete hot shutdown with no voiding in the core (see Appendices A.1.2 and A.1.3), the core is, by 35 min, operating at power levels close to decay heat. Reactor vessel level and injection flow are shown in Figs. 4.21 and 4.22, respectively.

The steaming rate during this sequence heats the pressure suppression pool until its EPG heat capacity temperature limit (Fig. 4.10) is exceeded. Therefore, in accordance with the EPG requirements (see subsection 4.1.3), the operators open three or more SRVs at 23 min and allow them to remain open thereafter. The reactor vessel pressure (Fig. 4.23) decreases rapidly, and by 26 min is below the 450 psig (3.21 MPa) setpoint for automatic opening of the Core Spray and LPCI reactor vessel injection valves.

Without operator action, the vessel pressure would soon be low enough to allow large quantities of cold water to be pumped into the reactor vessel, possibly causing very undesirable power spikes. For this sequence, it is assumed that the operators follow the EPG instructions to terminate and prevent all injection (except from the CRDHS and the SLCS, if running) prior to an emergency depressurization. The operators do this by tripping the Core Spray and RHR system pumps immediately after they automatically start and by either tripping the condensate and condensate booster pumps or by closing the main feedwater pump discharge valves.

During the depressurization, a large fraction of the reactor vessel water inventory is vaporized. The core is totally uncovered at 25 min. The operator restarts injection (Fig. 4.22) at 26 min with a flow of 1800 gpm (113 l/s) pumped from the main condenser hotwell by the series combination of one condensate pump and one condensate booster pump via the startup bypass control valve* (see Fig. 3.8). The operator might alternatively have reestablished vessel injection by restarting the HPCI system, but this flow would have lasted only until the isolation of the HPCI steam supply some 4 min later, on low vessel pressure [at 100 psig (0.79 MPa)].

The reactor vessel water level (Fig. 4.21) recovers to above the top of active fuel after 36 min, but the operator continues injection until there is positive indication on the Emergency Systems Level Indication before cutting back and then stopping the CBP flow at 40 min.

The brief period of core uncovery (11 min for the top part of the core and 3 min for the bottom part) would result in some heatup but no significant fuel damage. Even during the 3 min period of total uncovery, the fuel is partially cooled by a flow of steam flashed from the lower plenum because of the ongoing depressurization. During the refill stage, the CBP injection is resumed at 26 min; this flow fills the

*The BWR-LACP simulation of operator level control by condensate/condensate booster pump injection is described in section 4.3.3.

bottom part of the core and boils, providing steam cooling of the upper part.

Due to the depressurization and the higher average core power, the pressure suppression pool temperature (Fig. 4.24) increases more than in the previous cases. For this calculation, there is assumed to be continuous suppression pool cooling after 10 min, and the calculated peak temperature is 180°F (356 K). This prediction is non-conservative by about 4°F (2 K) because the pool cooling would actually not be in operation for a period of about 10 min, starting at 24 min. As discussed above, it is necessary to trip all the RHR pumps for at least 5 min before or during the early part of the depressurization to avoid the unwanted vessel injection that would otherwise occur after the automatic opening of the LPCI injection valves (since they are interlocked open for 5 min). In addition, the 2/3 core coverage interlock would actuate at 24 min, unless previously disabled by operator actuation of the key-locked override switch. The 2/3 core coverage interlock causes the RHR system to realign from the pressure suppression pool cooling mode into the LPCI mode. At 32 min, the level indication on the post accident monitoring range exceeds 2/3 core coverage, allowing the interlock to clear, and the operators to reestablish pool cooling if they had not previously done so by use of the key-locked override.

The drywell pressure (not shown) exceeds 2.45 psig (118 kPa) at 23 min, starting the 2 min ADS timer. Since the EPGs require the operator to prevent automatic depressurization, the calculation for this case assumes that the operator resets the timer every 2 min, or as required, to prevent ADS. However, an ADS actuation would make little difference to the outcome of this sequence since the operators initiate a manual emergency depressurization using three SRVs at about the same time.

The reactor is critical at very low pressures for a period of several minutes after reactor vessel depressurization in this accident sequence. It should be recognized that power excursions due to pressure increases are avoided during this period because the manually opened SRVs are left open, and because of the significant negative reactivity from the manual rod insertion. The negative reactivity contributed by the manual insertion of control rods enables the operators to effectively shut down the reactor without benefit of sodium pentaborate injection by the SLCS.

4.3.3 The case with neither SLC system injection nor manual rod insertion

For this case, it is assumed that the operators are unable either to start the SLC system injection or to manually drive control rods into the core. Figs. 4.25–4.30 show the results of the BWR-LACP calculations, and Table 4.5 gives the sequence of events. Even though the operators cannot insert poison into the core, they follow EPG instructions to reduce the core power level by lowering the vessel water level, and they initiate suppression pool cooling. These actions delay, but would not prevent the eventual overheating of the suppression pool to the point of overpressure failure of the drywell.

The first minutes are very similar to the previous cases: the HPCI system is running at full capacity, reactor power (Fig. 4.25) is varying about a mean value of approximately 28%, and the reactor vessel is fully pressurized with the SRVs cycling in response to both automatic and manual actuations. Operator attempts to control the SRVs to prevent automatic SRV actuation are fruitless. After the EPG-mandated water level reduction maneuver, the core power level (in response to increased core coolant voiding) decreases to below 10%, and vessel pressure (Fig. 4.28) plunges to about 700 psia (4.83 MPa) before the operators shut all but one of the manually opened SRVs. Several minutes later a power spike repressurizes the vessel, causing additional automatic SRV actuations.

Since the core is not being poisoned, the core power is higher than in previous cases. The suppression pool heat capacity temperature limit is exceeded after only 18.7 min. Following the EPG instructions, the operators open (a minimum of) three SRVs at this time and leave the control switch for each open SRV in the open position for the remainder of the accident. This brings to five the number of open SRVs, since previous operator manipulations resulted in two manually-open SRVs at the time depressurization was initiated. Prior to beginning the depressurization, the operators terminate HPCI flow (per EPG instructions) and prevent uncontrolled flooding of the vessel by tripping the low pressure injection systems before the decreasing reactor vessel pressure reaches the shutoff head of the pumps. The CRDHS runs continuously throughout the accident, injecting between 100 and 180 gpm (6.3 and 11.3 l/s) depending on reactor vessel pressure.

The depressurization causes the core to be totally uncovered (Fig. 4.26), so the core thermal power output falls to the decay heat level. For the same reasons discussed for the core uncovery in Section 4.3.2, this relatively brief uncovery does not result in fuel damage. The operators re-establish injection (Fig. 4.27), not with the HPCI system, but by using a series combination of one condensate pump and one condensate booster pump. The resulting flow from the main condenser hotwell to the reactor vessel is controlled by manipulation of the startup bypass valve (see Fig. 3.8), with the main feedwater pump discharge valves closed. The BWR-LACP code simulates operator level control of condensate booster pump flow in accordance with the following rules:

1. If the Emergency Systems level indication is off-scale low, the injection rate is set at 1800 gpm (113 l/s).
2. If the level indication is on-scale but below the desired level for manual control near the TAF [380 in (9.65 m) above vessel zero], the injection flow is set at 900 gpm (57 l/s).
3. If the level indication is above the desired level, injection flow is set at 600 gpm (38 l/s).
4. If the level indication is more than 20 in. (51 cm) above the desired level, injection flow is set to zero.
5. The operator checks the vessel water level once per minute and adjusts injection flow, as required by the preceeding four rules.

Conversations with TVA engineers led to the assumption that operators would use the Emergency Systems level indicator for control rather than the Post Accident Flooding range indicator; however, with the reactor vessel depressurized, the Post-Accident Flooding range instrument would actually provide more accurate level indication.

After 5 min of injection at 1800 gpm (104 l/s), the reactor vessel water level has been increased to within the range of the Post Accident Flooding range level indication but level is still below the TAF. The injection flow is allowed to continue until water level is also within the range of the Emergency Systems level indication, and well above the TAF.

As vessel water level increases to above the top of active fuel, the conditions for criticality are met, and then exceeded. There is no immediate apparent response because the neutron flux is several orders of magnitude below the power range. At 33 min, the core thermal power begins to increase above the decay heat level. Higher core power means more steam production, so the vessel pressure also starts to increase. The vessel pressure is sensitive to increased steam production because all five open SRVs close at 27 min due to insufficient [<20 psid (138 kPa)] reactor vessel-to-drywell pressure difference. The increasing vessel pressure compresses voids in the core, adding positive reactivity and accelerating the rate of increase in both pressure and power. All five of the previously closed SRVs reopen when the vessel-to-drywell pressure difference again exceeds 50 psid (345 kPa).

The cycle of increasing core power and vessel pressure is not broken until the vessel has repressurized to 1120 psia (7.72 MPa), automatically opening four additional SRVs. A maximum core thermal power output of 81% is reached before sufficient voids are generated in the core to reverse the excursion.

As soon as core power decreases back below about 30%, the five manually opened SRVs begin depressurizing the reactor vessel. Vessel water level decreases rapidly, and by 36.5 min the core is again entirely uncovered. This requires operator action to re-establish vessel water injection, and after the core is recovered there is another power/pressure spike very similar to the first one.

The power/pressure spikes will be repeated indefinately, about every 13 min, unless poison is added to the core, or unless the method of vessel water level or pressure control is changed. Considering that the core generates only decay heat between power spikes which extend to 60 or 80%, the time-averaged power after 30 min is about 8.3%. The dissipation of this thermal power in the suppression pool power requires more cooling capacity than the suppression pool cooling system can provide.

At the end of 2 h the suppression pool temperature is at 232°F (384 K) and is slowly increasing. If this accident were allowed to continue in the same mode for another 10 h, the pool temperature would be at about 345°F (447 K) and the steam pressure within primary containment would be sufficient to cause the overpressure failure of the drywell. At the end of 2 h, the drywell pressure is 28 psia.

In order to see if the core power spikes can be eliminated or reduced by adjustment of the injection logic, this same case was repeated with a modified strategy for operator control of vessel water level when injecting with the condensate booster pumps. It is impossible to judge whether the modified strategy or the one considered in the first part of this section would be more likely to be employed in the unlikely event of an ATWS since the training of operators in the use of the EPGs is still at an early stage. The purpose of the following exercise is solely to demonstrate the beneficial effect of increased care in the control of injection flow, particularly during the refill stage of an ATWS transient in which the downcomer water level has dropped to below the top of active fuel.

The rules for the modified strategy are:

1. The set point for manual level control is 350 in. (8.89 m) as determined from the Post Accident Flooding range indication [instead of the 380 in. (9.65 m) Emergency Systems indication setpoint used for the calculation discussed above]. The effect of this is that when the indicated water level is at the setpoint, the actual level will be below the top of the active fuel.

2. If the level is more than 6 in. (15.2 cm) below the setpoint, flow is set at 1800 gpm (113 l/s).

3. If the level is below, but within 6 in. (15.2 cm) of the set point, injection is 900 gpm (57 l/s).

4. If the level is above the setpoint, injection flow is 600 gpm (38 l/s).

5. If the level is more than 6 in. (15.2 cm) above the setpoint, the startup bypass valve is completely closed to zero the condensate booster pump injection.

The differences between this modified level control strategy and the one listed previously are that the operator is directed to control vessel water level at a setpoint which is below the top of the active fuel, instead of above, and to shut off the injection flow sooner when the desired vessel level is exceeded.

The calculated results show that this modified level control strategy eliminates almost all of the spikes in core thermal power (Fig. 4.31). The one thermal power spike that occurs after the transition to condensate booster pump injection is a result of the recovery from the emergency depressurization which had previously totally uncovered the core. After this one power spike, the operator is able to maintain vessel water level (Fig. 4.32) very close to the TAF by initiating 1 min bursts of condensate booster pump injection (Fig. 4.33) at 600 gpm (38 l/s) about once every 3 min. The nearly complete core coverage thus obtained is adequate to protect the core, and the core thermal power remains very close to the decay heat level. With all four suppression pool coolers running, the peak suppression pool temperature is 189°F (361 K), occurring 36 min into the accident. Therefore, this modified vessel level control strategy eliminates the possibility of static overpressurization failure of primary containment.

4.3.3.1 Effect of stuck-open relief valves. As demonstrated above, when there is neither manual rod insertion nor SLC injection, the EPGs require an emergency depressurization of the reactor vessel, beginning at 18.7 min (Table 4.4). Compounding these failures with one or two stuck-open SRVs has very little effect on the overall sequence since the reactor vessel becomes depressurized even without the stuck-open SRVs.

In the case with only one stuck open SRV, the reactor vessel does not depressurize sooner. Before 18.7 min, reactor thermal power is high enough to hold one or more SRVs open at full vessel pressure. After 18.7 min, the operators open three additional SRVs and depressurize the reactor vessel. They leave the hand switch for each of the manually opened valves in the "on" position, and in effect — stuck open.

In the case with two stuck-open SRVs, the reactor vessel begins depressurizing after 9 min and reaches a pressure of about 300 psia (2.07 MPa) before the operators hasten the depressurization by opening three additional SRVs when the suppression pool heat capacity temperature limit is exceeded. For the two SORV case, operators have to act to trip the low pressure, high capacity injection systems (e.g., Core Spray) about 5 min sooner than they would for the case without any SORVs. As discussed previously these pumps start automatically and, if not prevented, can flood the depressurized reactor vessel, causing severe power and pressure excursions.

4.3.3.2 The sequence of events without pressure suppression pool cooling. This section discusses the effect of compounding the failures of manual rod insertion and SLC injection with a failure of suppression pool cooling. The sequence of events is essentially the same as that for the case with pool cooling (Ref. Table 4.4, Figs. 4.25—4.30) with the important exception that the suppression pool temperature increases much more rapidly. As the pool temperature increases, its vapor pressure increases. Evaporative steaming from the surface of the pool as well as direct bubble — through of part of the SRV discharge would pressurize the wetwell. This steam discharge easily and quickly reaches the drywell atmosphere via the 12 two-ft (0.61-m) diameter vacuum breakers, which open a flow path to the drywell atmosphere when wetwell pressure exceeds the drywell pressure by more than 0.5 psi (3 kPa). By 150 min, the drywell pressure reaches the assumed 117 psig (0.910 MPa) failure pressure* of the drywell.

The calculation ends with drywell failure. BWR-LACP is not programmed to calculate events after the drywell failure, which include the possibility of severe fuel damage.

4.3.4 Emergency action levels and timing

The timing of the declaration of emergency action levels for the cases in which backup shutdown systems fail is specified on Table 4.6. The criteria for determination of emergency action levels are taken from the TVA Implimenting Procedures Document applicable to the Browns Ferry nuclear plant (Ref. 4.4).

In the event that either the SLCS injection or manual rod insertion is available, the reactor can be shutdown, so there is no need for an emergency status higher than Alert unless the accident is compounded with another serious failure such as loss of suppression pool cooling. The emergency response action levels for the case of failure of suppression pool cooling after shutdown from an MSIV closure ATWS incident are discussed in Sect. 4.2.4.

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*The assumed static overpressurization failure point for the drywell is taken from the information provided in Ref. 3.3.

If neither of the backup means of shutdown are available, the calculations of Sect. 4.3.3 show that the time averaged reactor power would exceed the cooling capacity of the suppression pool coolers. The suppression pool would be overheated, and primary containment pressure would steadily increase. The Alert status would be upgraded to General Emergency after about 6 h when drywell pressure would have exceeded 50 psig (0.45 MPa). The overpressure failure pressure of the drywell would be exceeded another 6 h later, or 12 h from the inception of the accident sequence.

If the failure of both backup means of shutdown were compounded with failure of the suppression pool cooling, then the suppression pool would be heated rapidly, and the Alert would be upgraded to General Emergency after 111 min. The drywell overpressure failure pressure would be exceeded only 150 min after the beginning of the accident.

General considerations for emergency response for accidents in which the drywell failure occurs before any severe fuel damage are discussed in Sect. 4.2.4. A detailed study of fission product release and transport following MSIV-closure initiated ATWS sequences that result in severe fuel damage is planned to be conducted at ORNL. The results of this study, to be published in a companion report, will provide a quantitative basis for planning of the optimum emergency actions for such a highly improbable eventuality.

References for Chapter 4

4.1 S. A. Hodge et al., *SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis*, NUREG/CR-2672, Vol. 1, ORNL/TM-8119/V1 (November 1982), Appendix E.

4.2 *Analysis of Incomplete Control Rod Insertion at Browns Ferry 3*, INPO/3, NSAC/20 (December 1980).

4.3 Prepublication Draft, *Emergency Procedure Guidelines, BWR 1 through 6, Revision 3* (December 1982).

4.4 *Browns Ferry Nuclear Plant, Radiological Emergency Procedures*, Implementing Procedures Document, July 6, 1983.

4.5 R. P. Wichner et al., *Noble Gas, Iodine, and Cesium Transport Following a postulated Loss of Decay Heat Removal Accident at Browns Ferry Unit One*, NUREG/CR-3617, ORNL/TM-9028 (to be published).

Table 4.1. Typical differences in indicated level between the Emergency Systems Indication and the Post Accident Flooding Indication

Pressure (psia)	1000		15	
Actual level, in.	560	380	560	380
Emergency Systems Indication, in.	560	380	588[a]	373[b]
Post Accident Flooding Indication, in.	473	337	560	380

[a]Pointer pegged at upper end of scale.

[b]Pointer pegged at lower end of scale.

Table 4.2. Relative change in specific volume of vapor per unit change in pressure at various pressures between 15.0 and 1050 psia

Pressure (psia)	Relative change in vapor specific volume per unit change in pressure
15.0	3634.4
100.0	92.5
200.0	24.7
300.0	11.0
400.0	6.4
500.0	4.2
600.0	2.9
700.0	2.2
800.0	1.7
900.0	1.3
1000.0	1.1
1050.0	1.0

Table 4.3. MSIV closure ATWS with SLC and MRI initiation

Time (min)	Event	Comment
0	MSIV closure initiated	No scram
0.1	Recirculation pumps tripped	At reactor vessel pressure 1135 psia
1	HPCI and RCIC start	At reactor vessel level of 476.5 in. (12.1 m)
1	Operator begins SRV manipulations	To prevent auto SRV actuation
1.5	Suppression pool temperature exceeds 110 F (317 F)	EPG criterion for SLC initiation
1—25	Wide reactor vessel pressure swings	Due to operator SRV manipulations
3	Operator begins manual rod insertion	One rod at a time, at rod speed of 3 in./s (7.62 cm/s)
5	Operator initiates SLC	
7	Operator trips HPCI, RCIC	Initiation of EPG level/power control
8	Core spray and RHR pumps auto-start	Reactor vessel water level <413.5 in. (10.5 m)
8.6	HPCI suction shift	Indicated suppression pool water level > +7 in.
9	Vessel Emergency Systems (ES) level indication off-scale low	Operator preferred level indication
9	Operator restarts HPCI	At 40% of capacity
10	Operator initiates suppression pool cooling	All 4 RHR coolers
11	Vessel ES level indication back on scale	
13	Operator trips HPCI	Vessel water level too high — 40 in. (1.02 m) above TAF[a]
13—21	Steadily declining vessel water level	
17	Peak suppression pool temperature reached	At 157 F (343 K)
21	Operator restarts HPCI	At 20% of capacity
24	Operator trips HPCI	Vessel water level 40 in. (1.02 m) above TAF
30	SLC injection sufficient for hot shutdown	Total 265 lbs (120 kg) boron required
30	Operator restarts HPCI	At 100% (to promote boron mixing)
35	Operator trips HPCI	At 500 in. (12.7 in.) vessel level [or 140 in. (3.56 m) above TAF]
35—end	Reactor core on decay heat	
35—60	CRDHS injection continues	At 110 gpm (0.007 m^3/s)

[a]Top of active fuel (TAF) is 360 in. (9.14 m) above vessel zero in the BWR-LACP simulation.

Table 4.4 Timing of Emergency Action Levels for MSIV closure ATWS accidents in which backup shutdown systems function (cases of Section 4.2)

Time	Action Level	Criterion
	(a) With functioning pressure suppression pool cooling	
5 min	Alert	Failure of scram system
3 h	None[a]	Completion of manual insertion of all control rods
	(b) With failure of pressure suppression pool cooling	
5 min	Alert	Failure of scram
10 min	Alert	Loss of shutdown cooling
20 h	General Emergency	Drywell pressure >50 psig (446 kPa)

[a]Downgrading of action level status would require the absence of any other condition (e.g. high radiation levels) requiring a specific emergency classification.

Table 4.5. Sequence of events for case without manual rod insertion or SLC injection, but with pool cooling

Time (min)	Event	Comment
0	MSIVs begin to close	Anticipated transient
0.1	No reactor scram	
0.1	Recirculation pumps trip	
1.5	HPCI and RCIC start	Automatic actuation, total injections 5600 gpm (353 l/s)
2	Operator control of vessel pressure begins	To prevent SRV cycling on automatic actuation
7	Operator trips HPCI and RCIC	Per EPG level/power control guideline
8	Core spray and RHR pumps start	At vessel water level <413.5 in. (12.5 m) - reactor vessel pressure too high for injection
8.4	Vessel water level below TAF	Operator restarts HPCI at 1800 gpm (113 l/s)
8.5	Reactor power below 10%	
9	Vessel pressure dropping	Operator shuts all but one SRV
10	Operators initiate suppression pool cooling with all four coolers	"Containment Spray Select" switch actuated
14.8	Vessel water level above TAF	Not back on scale of emergency systems indication
16.8	Power spike	Core thermal power to 35%
16.8	Automatic SRV actuations	
17	Operators decrease HPCI flow	Vessel water level too high
18.7	Operators begin emergency depressurization of reactor vessel	Suppression pool in violation of EPG heat capacity temperature limit
18.7	Operators trip HPCI and RCIC turbines and the core spray, condensate, condensate booster, and RHR pumps	Interrupts suppression pool cooling
19.5	Drywell pressure exceeds 2.45 psig (118 kPa)	
19.6	Core completely uncovered	Subcritical and producing only decay heat
20.1	Vessel pressure below 450 psig (3.21 MPa)	Core spray and LPCI valves open (LPCI valves interlocked open for 5 min)
20.6	Operators resume vessel injection	Using condensate booster pumps, flow controlled by startup bypass valve
27	Operators restart suppression pool cooling	After overriding 2/3 core coverage interlock
27.8	All SRVs shut	Vessel-to-drywell pressure difference <20 psi
31.8	Vessel water level recovered to >TAF	Level not back on scale of emergency systems indication
33.3	Operators discontinue injection flow	Emergency systems indication on scale but increasing too fast

Table 4.5 (continued)

Time (min)	Event	Comment
33.8	SRVs reopen	Vessel-to-drywell pressure difference >50 psi
34.6	Vessel power and pressure spike	Maximum core thermal power = 81%
34.8	Automatic SRV actuations	At 1105 psig (7.72 MPa)
36.5	Vessel pressure below 450 psig (3.1 MPa)	Depressurizing with five open SRVs
40—end	Additional power/pressure spikes	Occurring about every 13 min
120	Suppression pool temperature at 232°F (384 K)	Still increasing
720	Suppression pool temperature at 345°F (447 K)	Drywell overpressure failure imminent

Table 4.6 Timing of Emergency Action Levels for MSIV closure ATWS accidents in which backup shutdown systems fail (cases of Section 4.3)

Time	Action Level	Criterion
	(a) Cases with manual rod insertion and with pool cooling	
5 min	Alert	Failure of scram system
3 h	None[a]	Completion of manual insertion of all rods
	(b) Cases with SLC injection and with pool cooling, but no manual rod injection	
5 min	Alert	Failure of scram system
End	Alert	Control rods still not inserted
	(c) Cases with neither SLC injection, nor manual rod insertion	
5 min	Alert	Failure of scram system
6 h	General Emergency	Drywell pressure >50 psig (446 kPa)
	(d) Cases with neither SLC injection, nor manual rod insertion and without suppression pool cooling	
5 min	Alert	Failure of scram system
111 min	General Emergency	Drywell pressure >50 psig (446 kPa)

[a]Downgrading of emergency action level would require the absence of any other condition (e.g. high radiation levels) requiring a specific emergency classification.

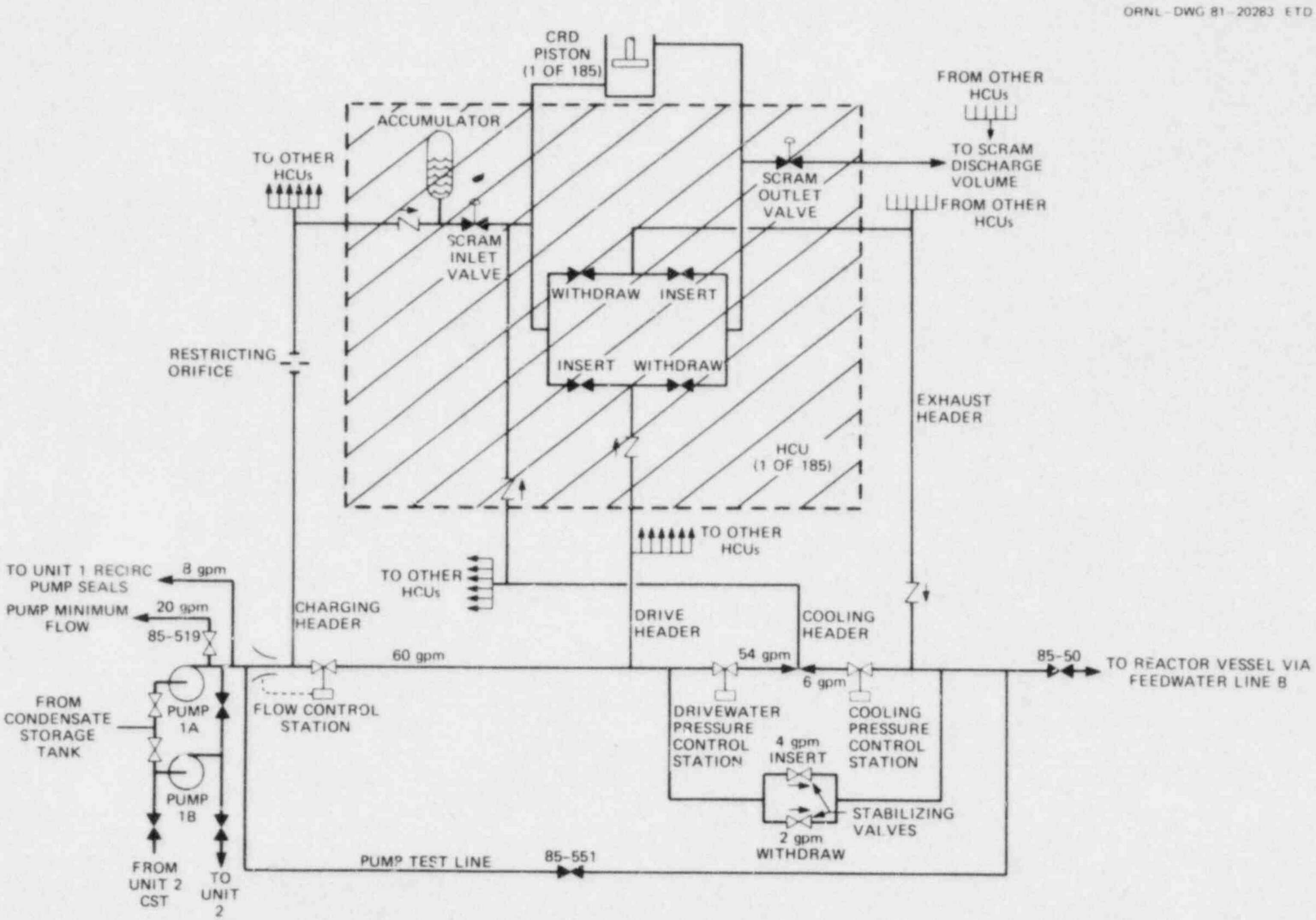

Fig. 4.1. Schematic diagram of the control rod drive hydraulic system.

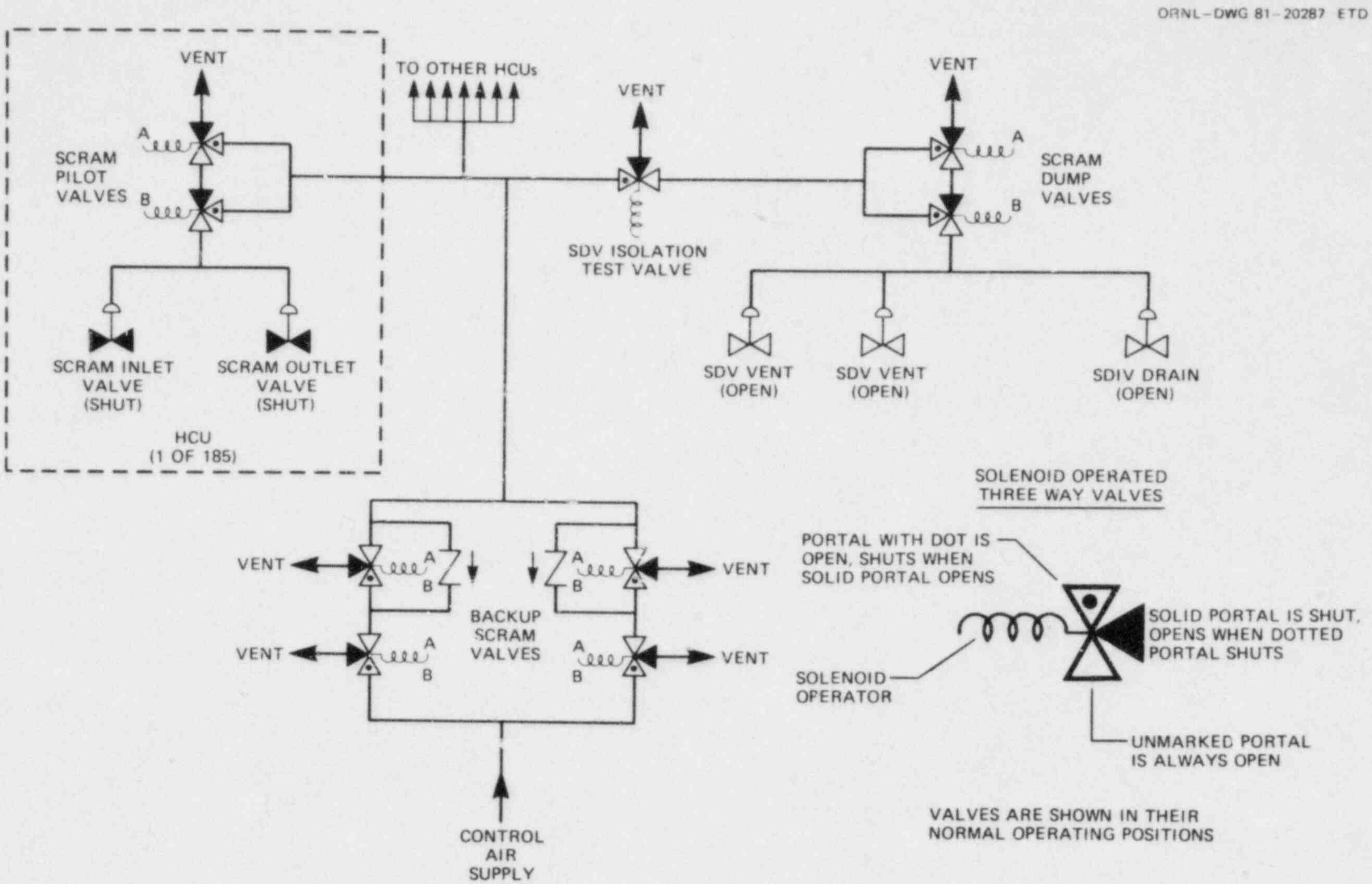

Fig. 4.2. Air operator network for the scram inlet and outlet valves and the scram dump valves.

ORNL–DWG 84–4532 ETD

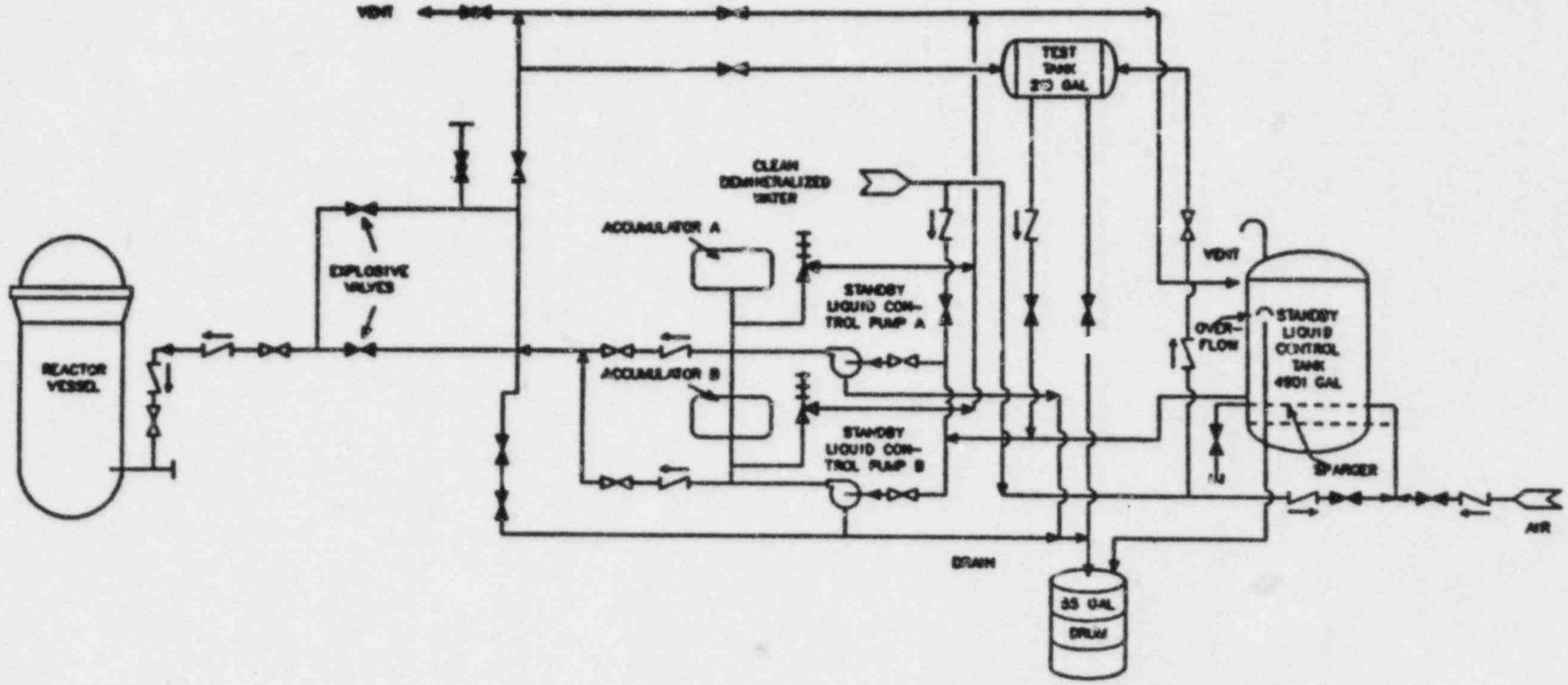

Fig. 4.3. Schematic diagram of Browns Ferry standby liquid control system.

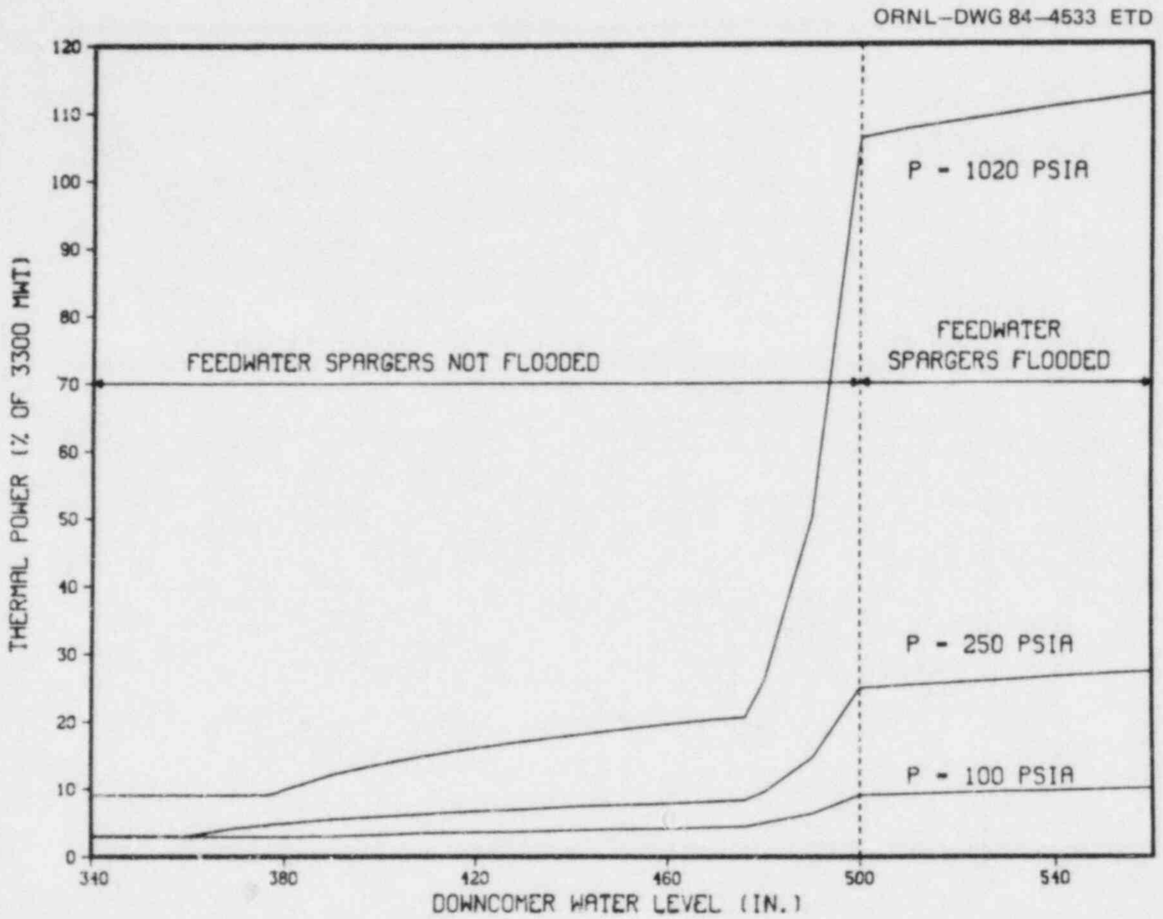

Fig. 4.4. Core thermal power as a function of water level in the reactor vessel downcomer for steady state ATWS conditions at three different pressures. The core is considered to be unpoisoned.

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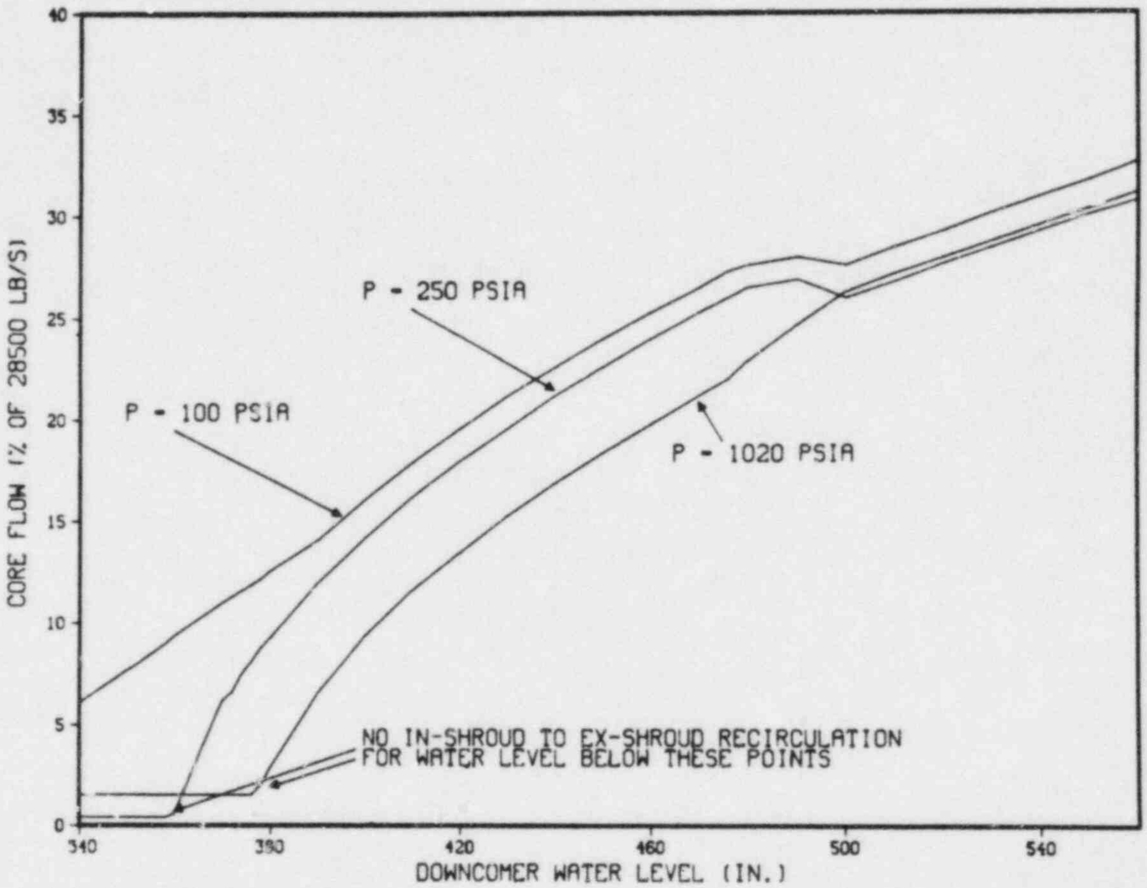

Fig. 4.5. Core inlet flow as a function of water level in the reactor vessel downcomer for steady state ATWS conditions at three different pressures. The core is considered to be unpoisoned.

ORNL-DWG 84-7792

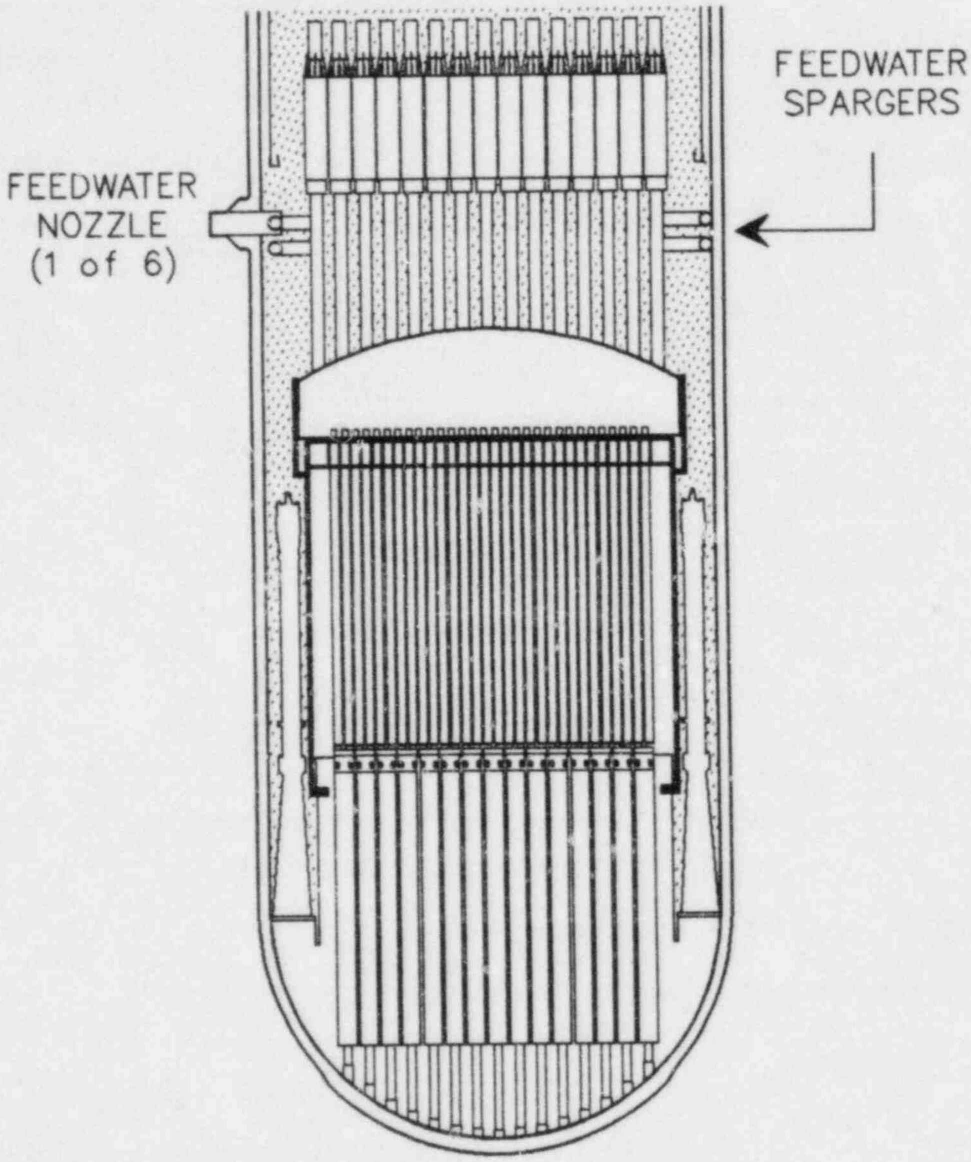

Fig. 4.6. Location of the feedwater spargers within the reactor vessel.

ORNL-DWG 84-7793

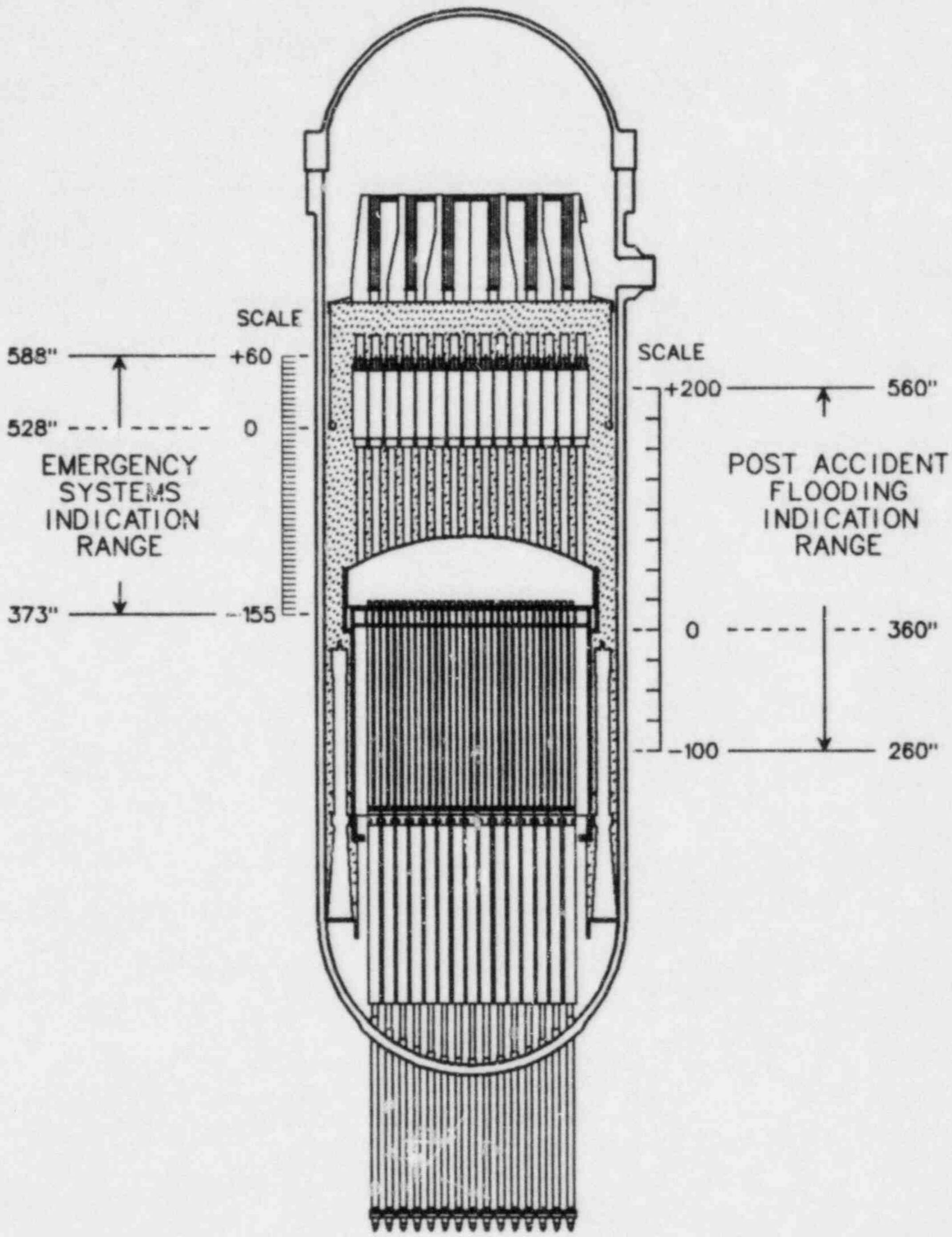

Fig. 4.7. Level instrumentation available for monitoring reactor vessel downcomer water levels near the top of the core. Non-scale dimensions are height in inches above the inner bottom of the reactor vessel.

ORNL-DWG 83-13719

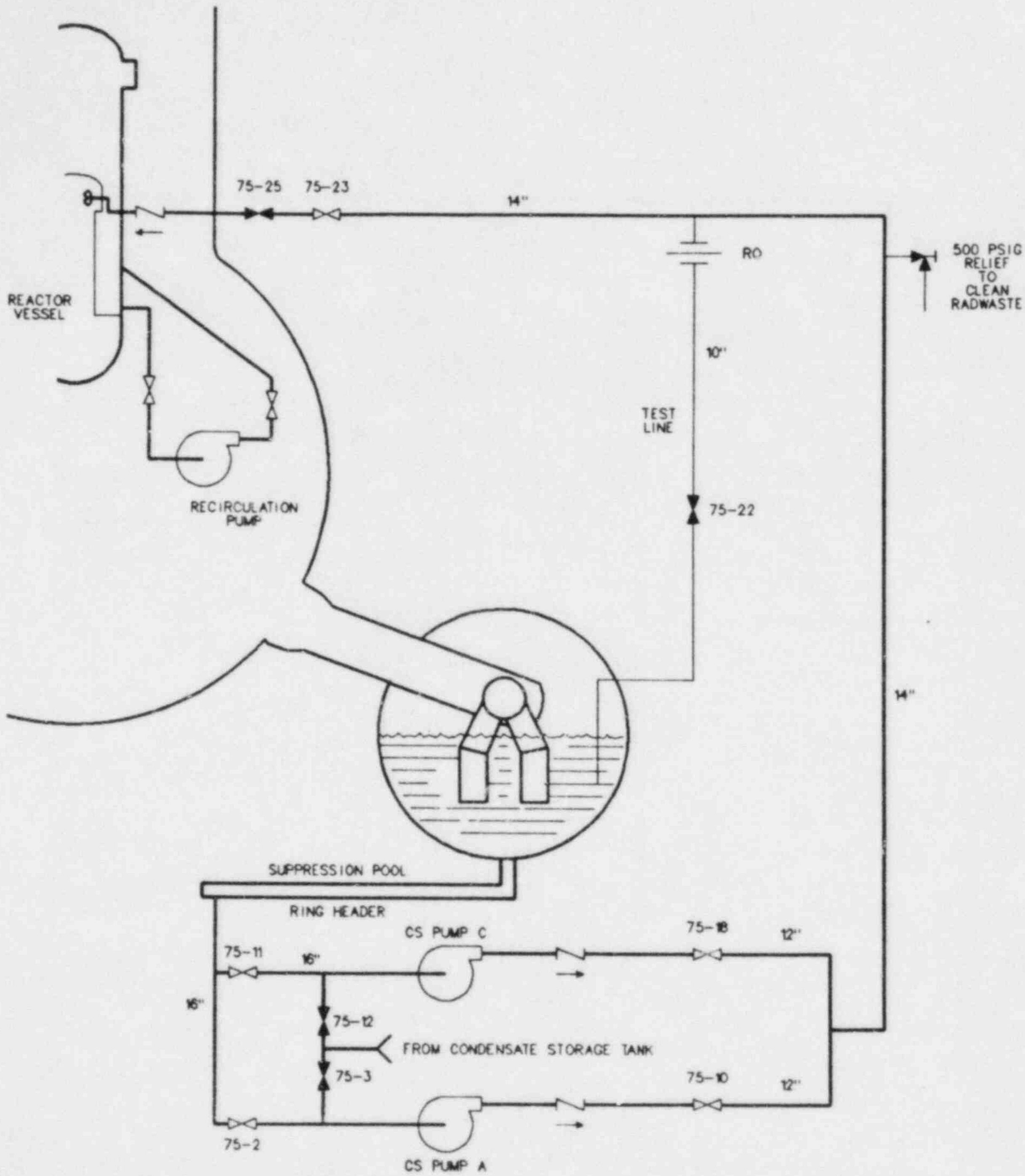

Fig. 4.8. Schematic diagram of one loop of the Core Spray system.

ORNL-DWG 82-19310

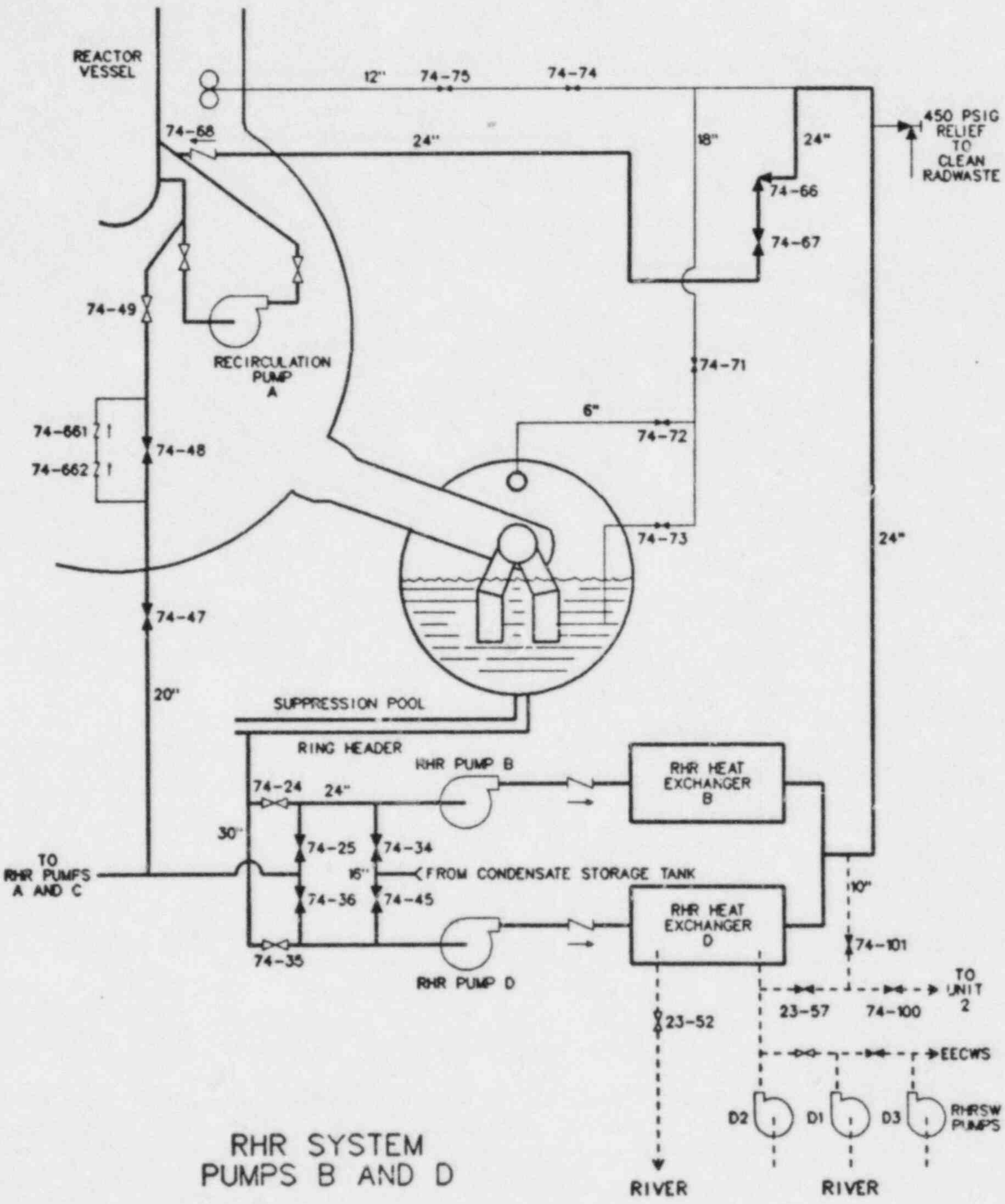

Fig. 4.9. Schematic diagram of one loop of the Residual Heat Removal system.

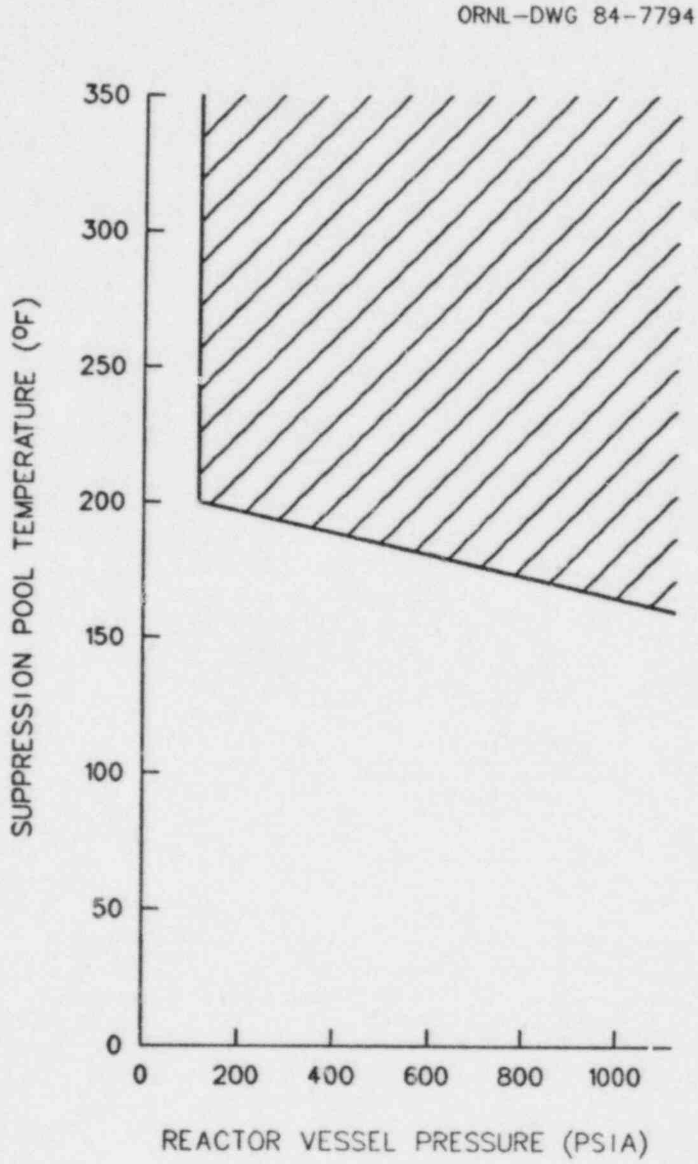

Fig. 4.10. Heat capacity temperature limit for the Browns Ferry Nuclear Plant. Reactor vessel pressure — suppression pool temperature combinations that lie within the shaded area are prohibited.

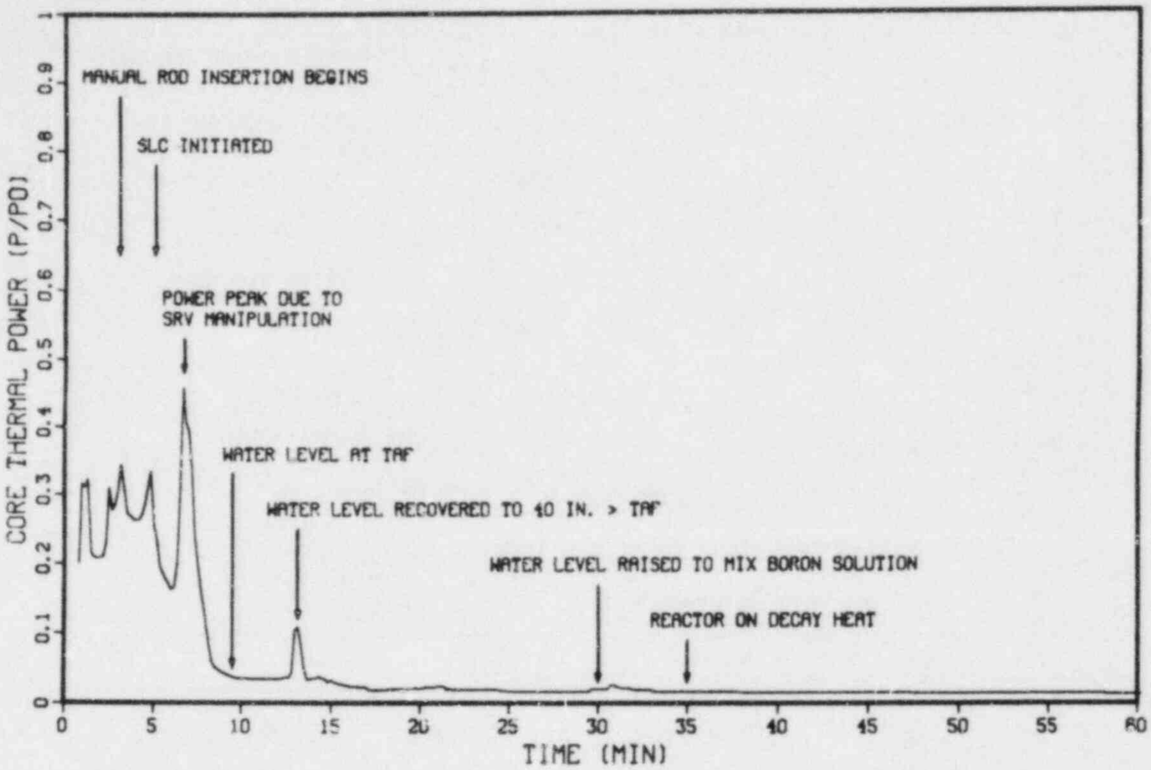

Fig. 4.11. EPG operator action sequence — core thermal power.

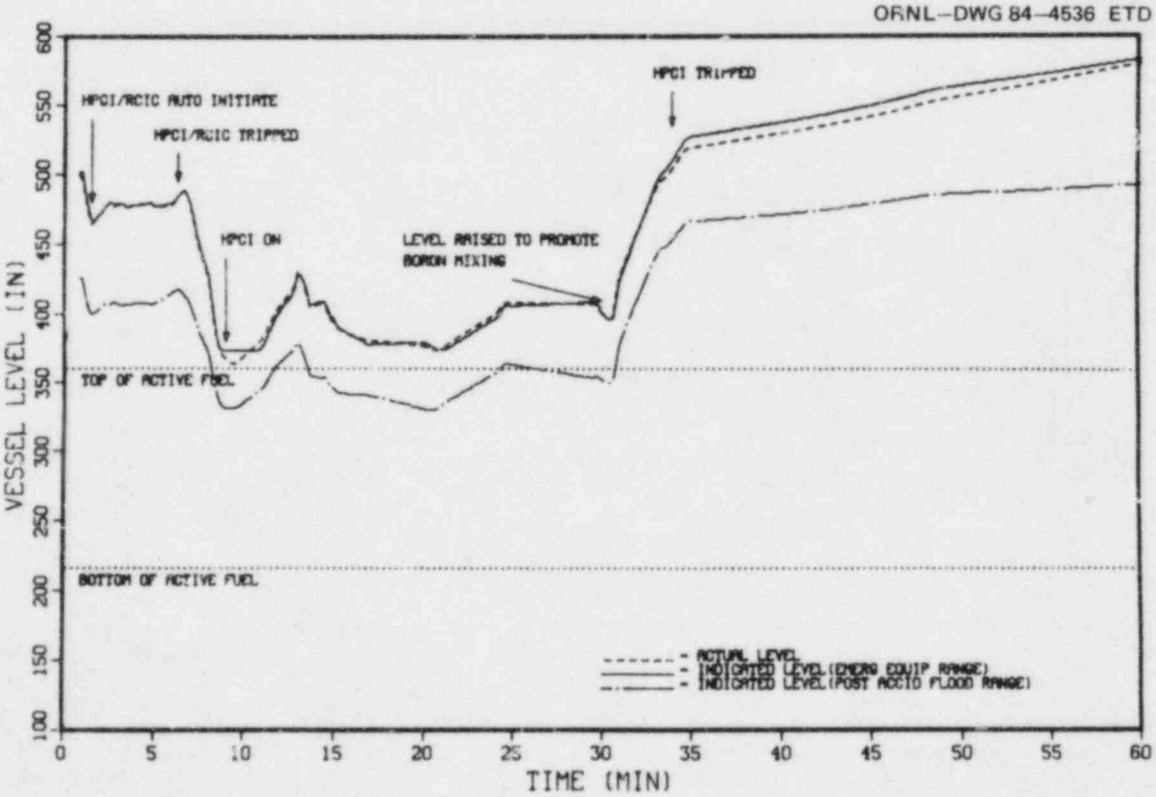

Fig. 4.12. EPG operator action sequence — vessel water level.

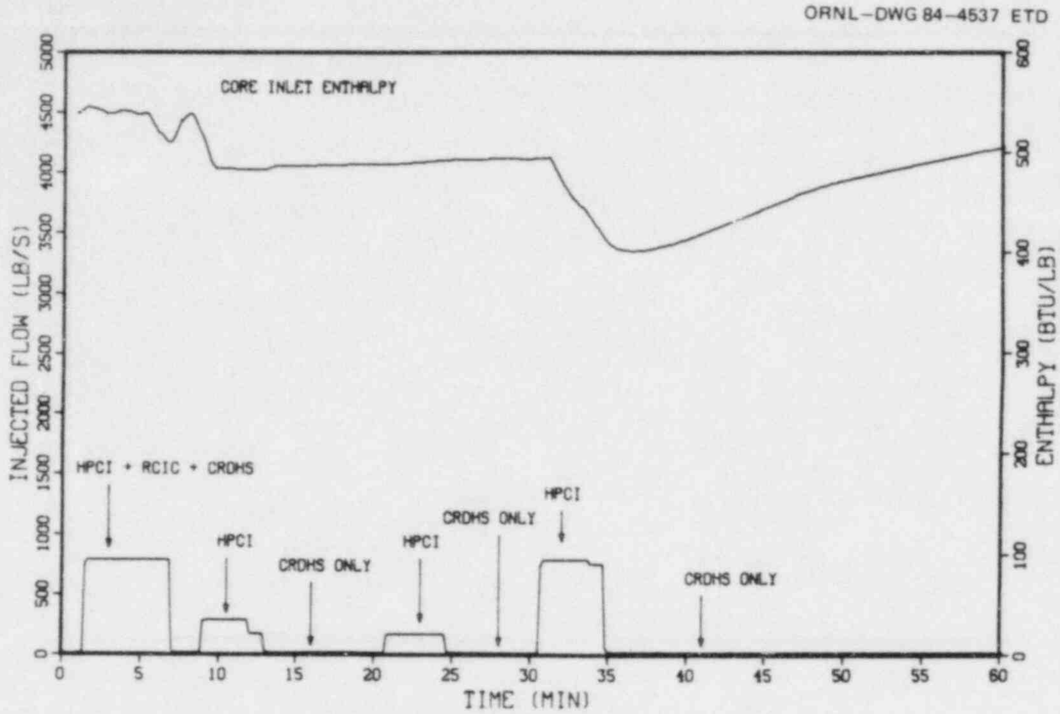

Fig. 4.13. EPG operator action sequence — injected flow.

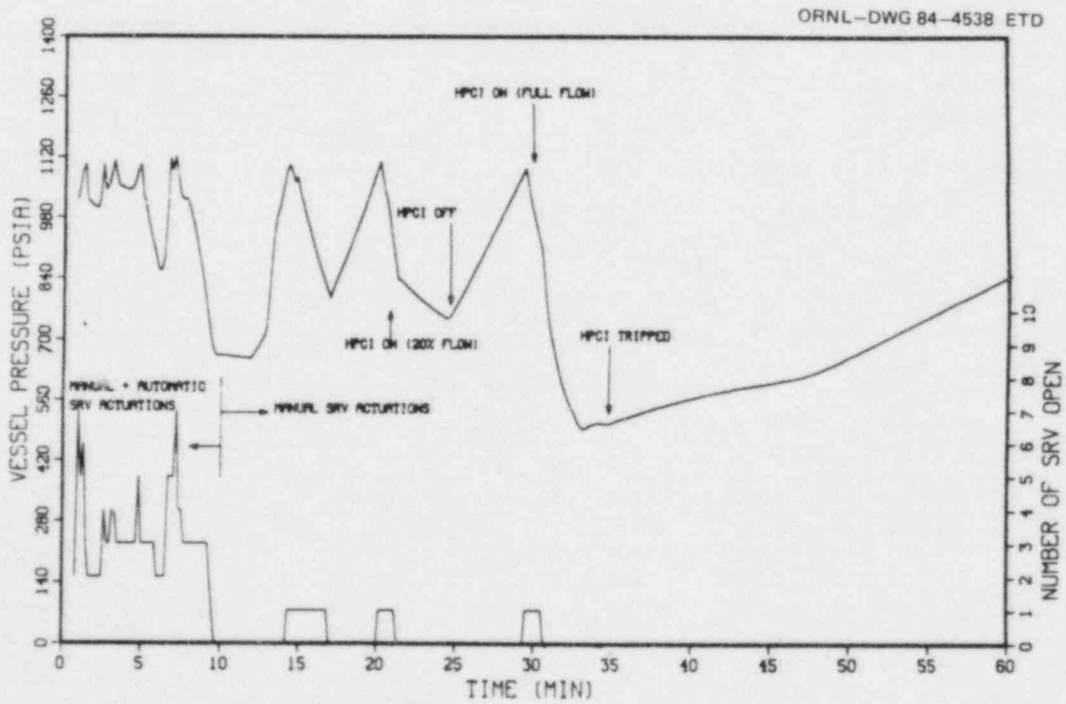

Fig. 4.14. EPG operator action sequence — vessel pressure.

ORNL-DWG 84-4539 ETD

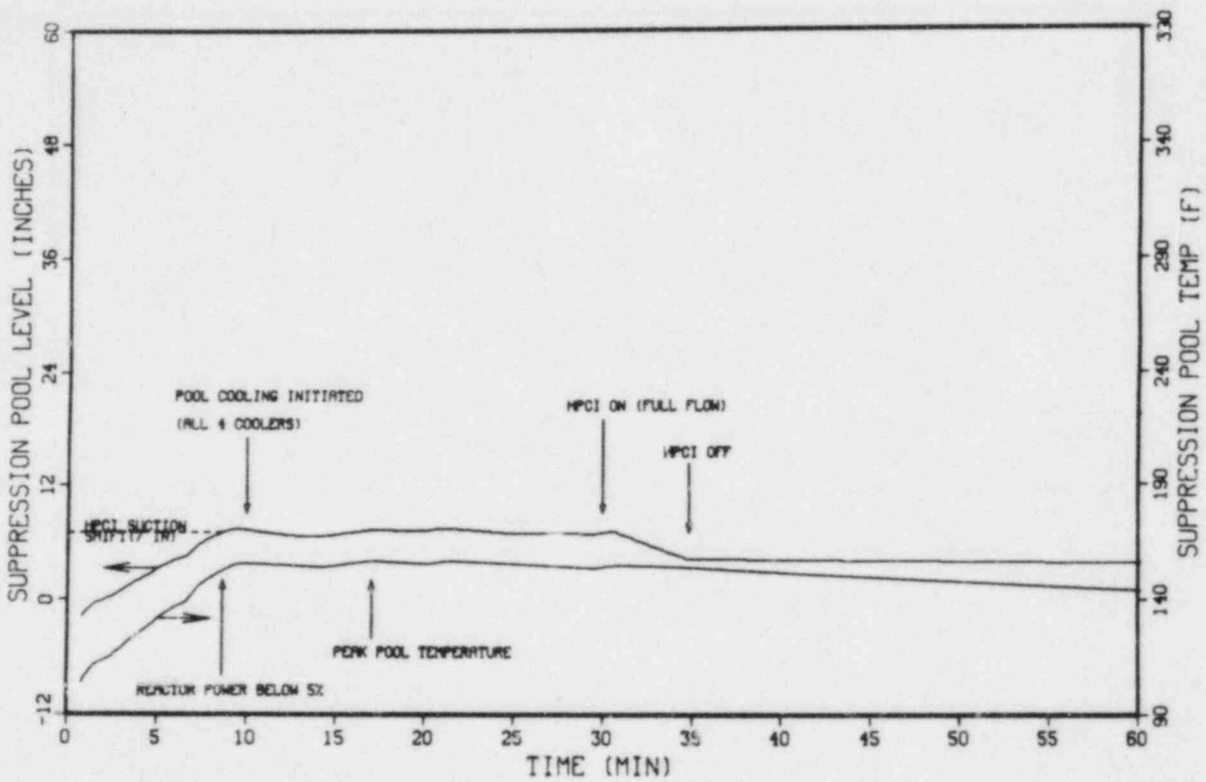

Fig. 4.15. EPG operator action sequence — suppression pool temperature and water level.

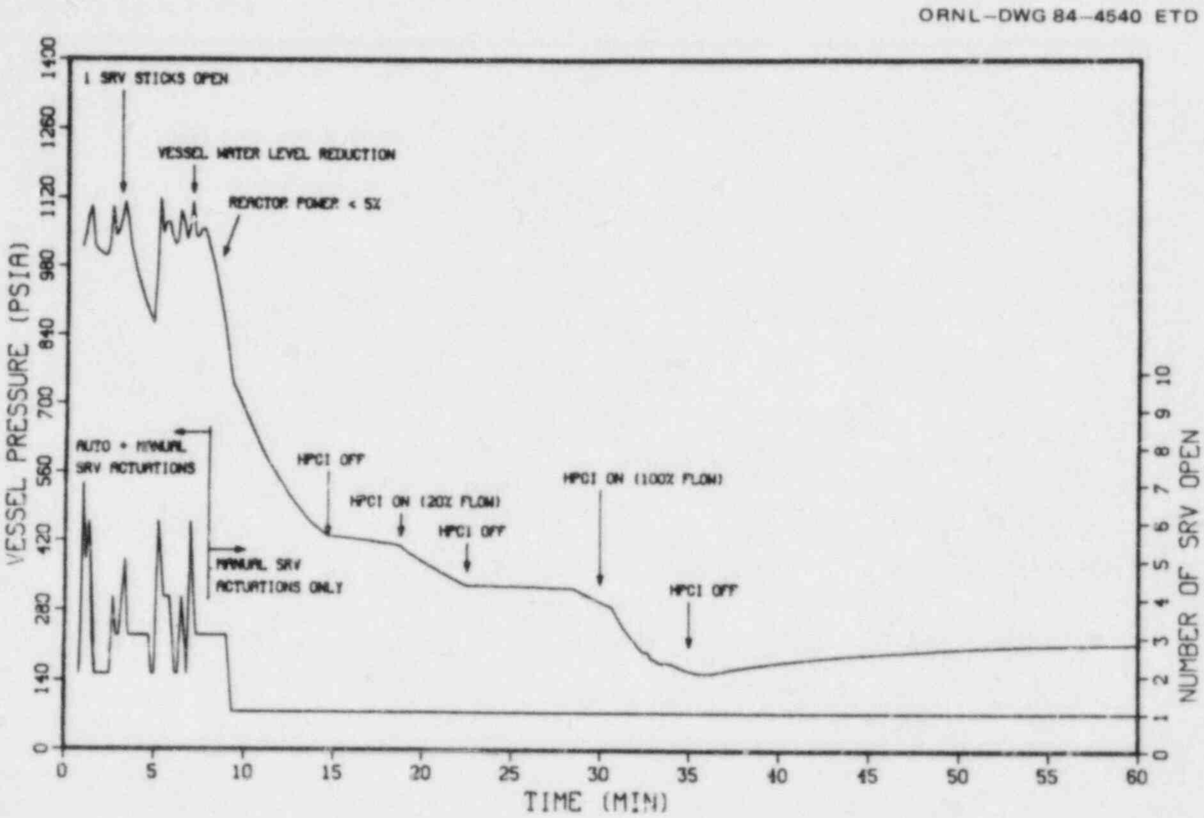

Fig. 4.16. EPG operator action sequence with one stuck open SRV — vessel pressure.

ORNL-DWG 84-4541 ETD

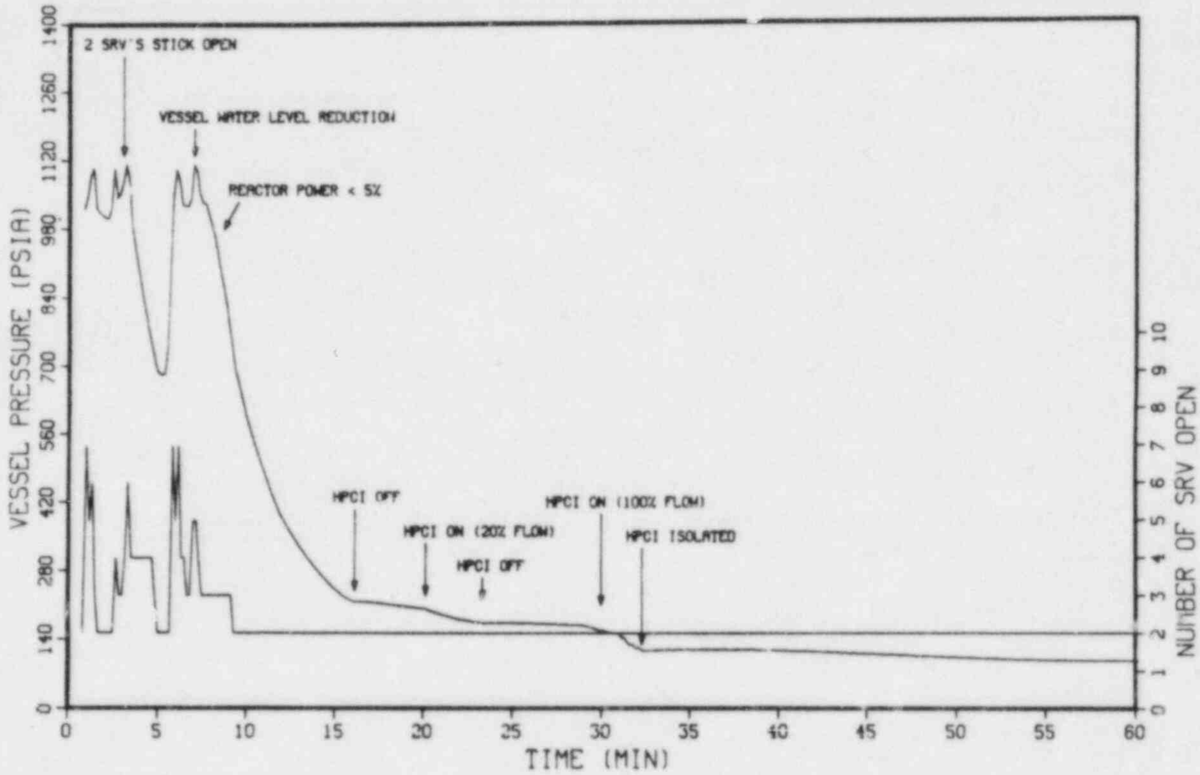

Fig. 4.17. EPG operator action sequence with two stuck open SRVs — vessel pressure.

ORNL-DWG 84-4542 ETD

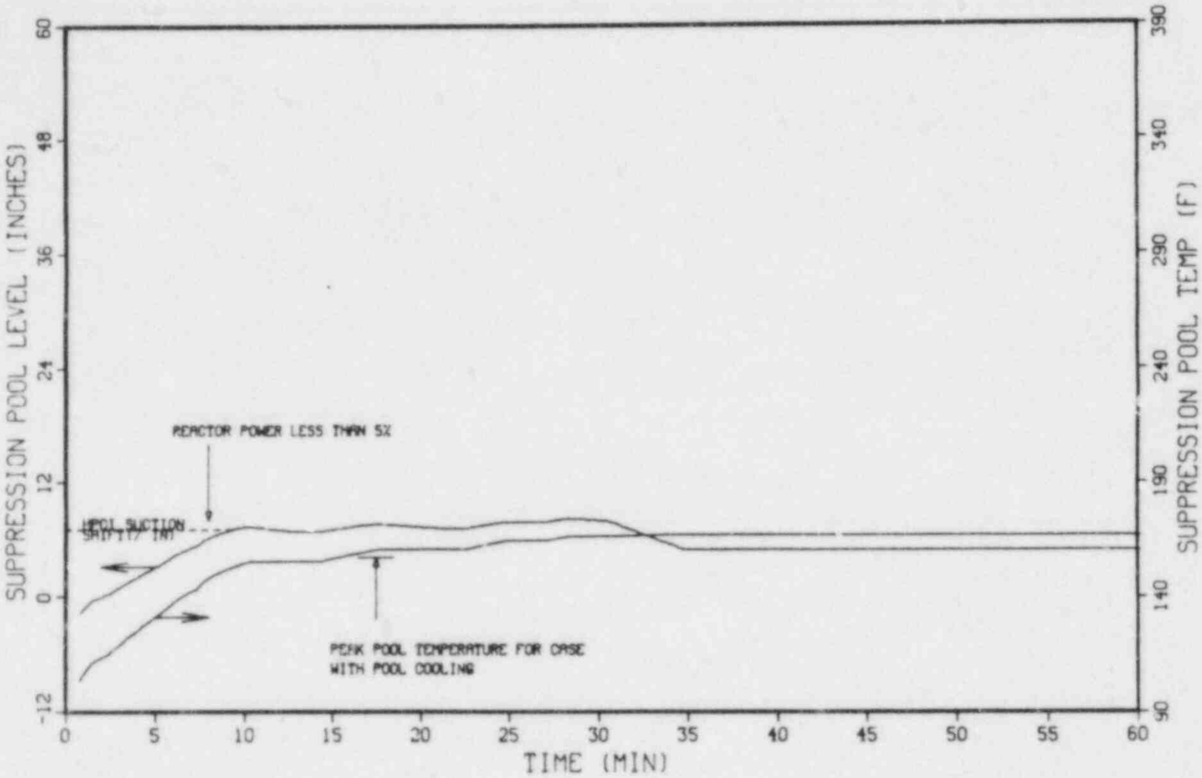

Fig. 4.18. EPG operator action sequence with failure of suppression pool cooling — suppression pool temperature and water level.

ORNL-DWG 84-4543 ETD

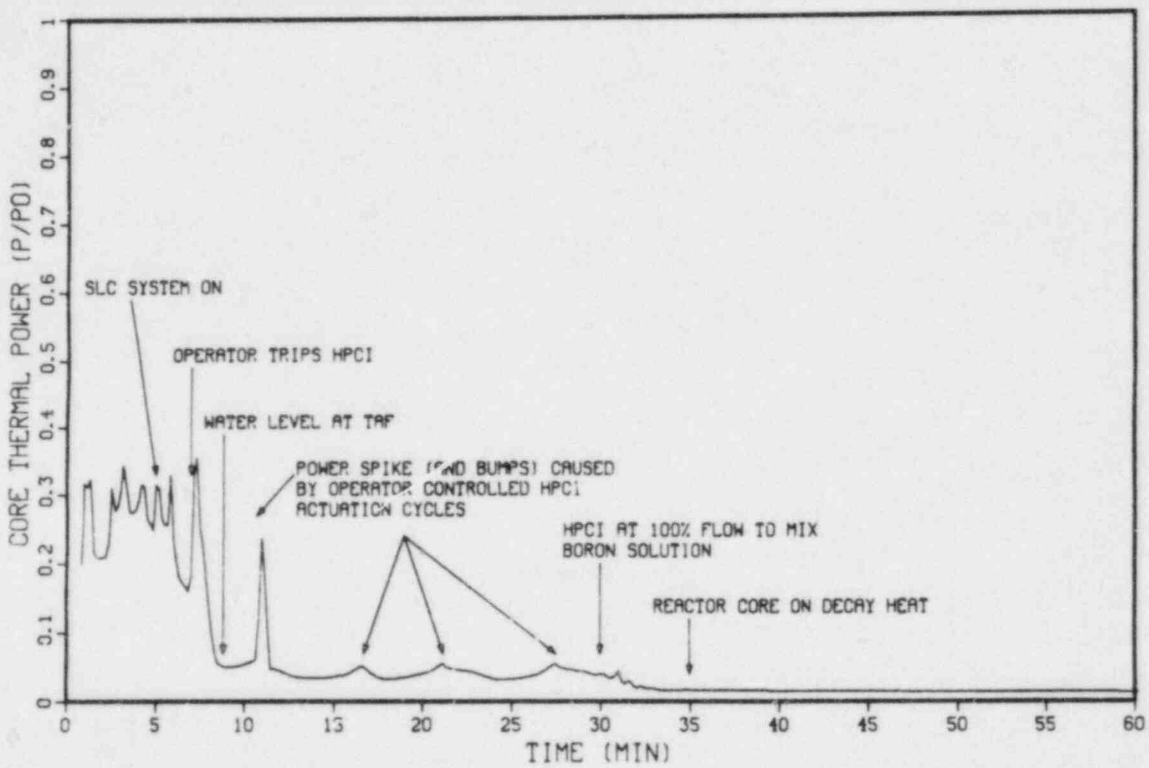

Fig. 4.19. EPG operator action sequence with failure of manual rod insertion — core thermal power.

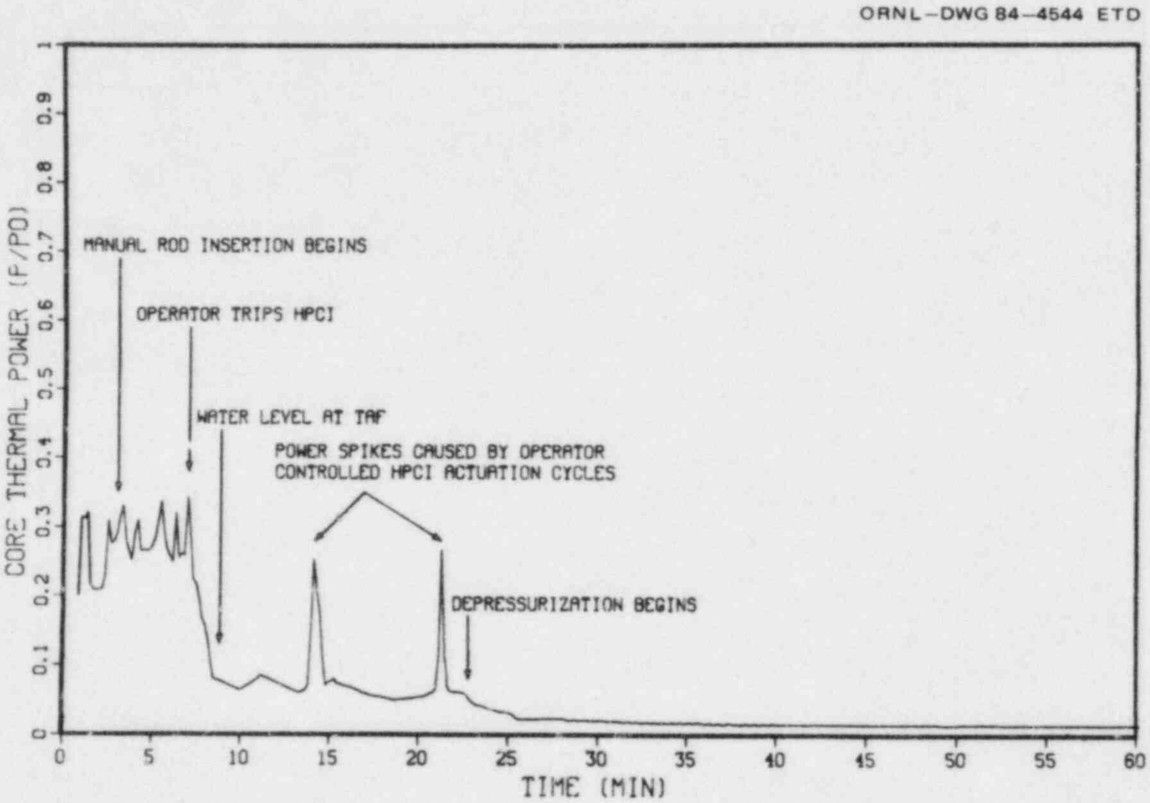

Fig. 4.20. EPG operator action sequence with failure of SLC system — core thermal power.

ORNL-DWG 84-4545 ETD

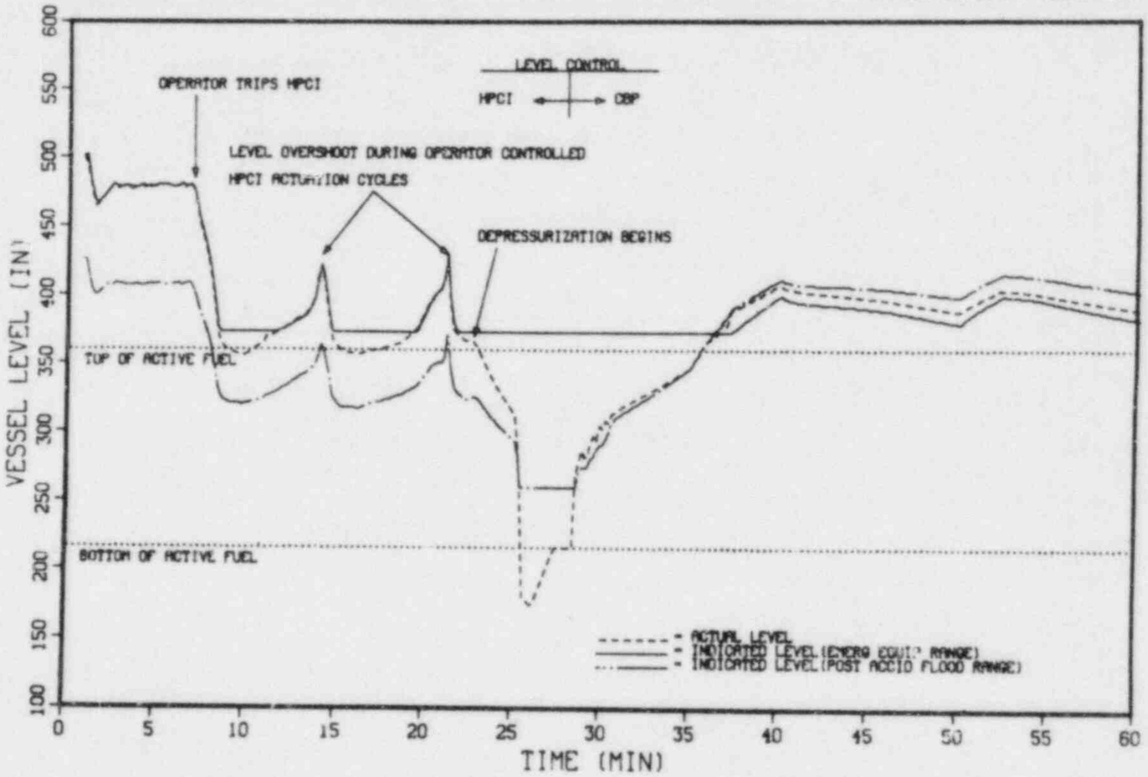

Fig. 4.21. EPG operator action sequence with failure of SLC system — vessel water level.

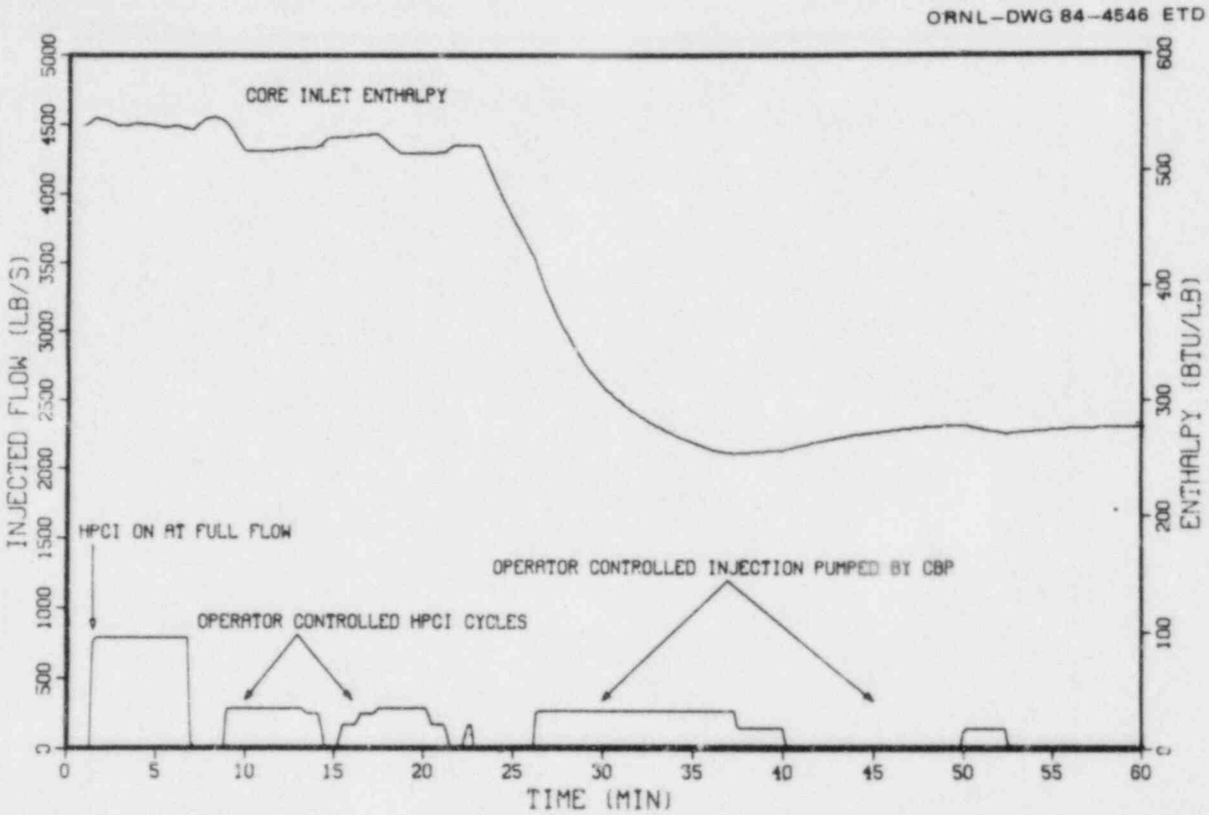

Fig. 4.22. EPG operator action sequence with failure of SLC system — injected flow.

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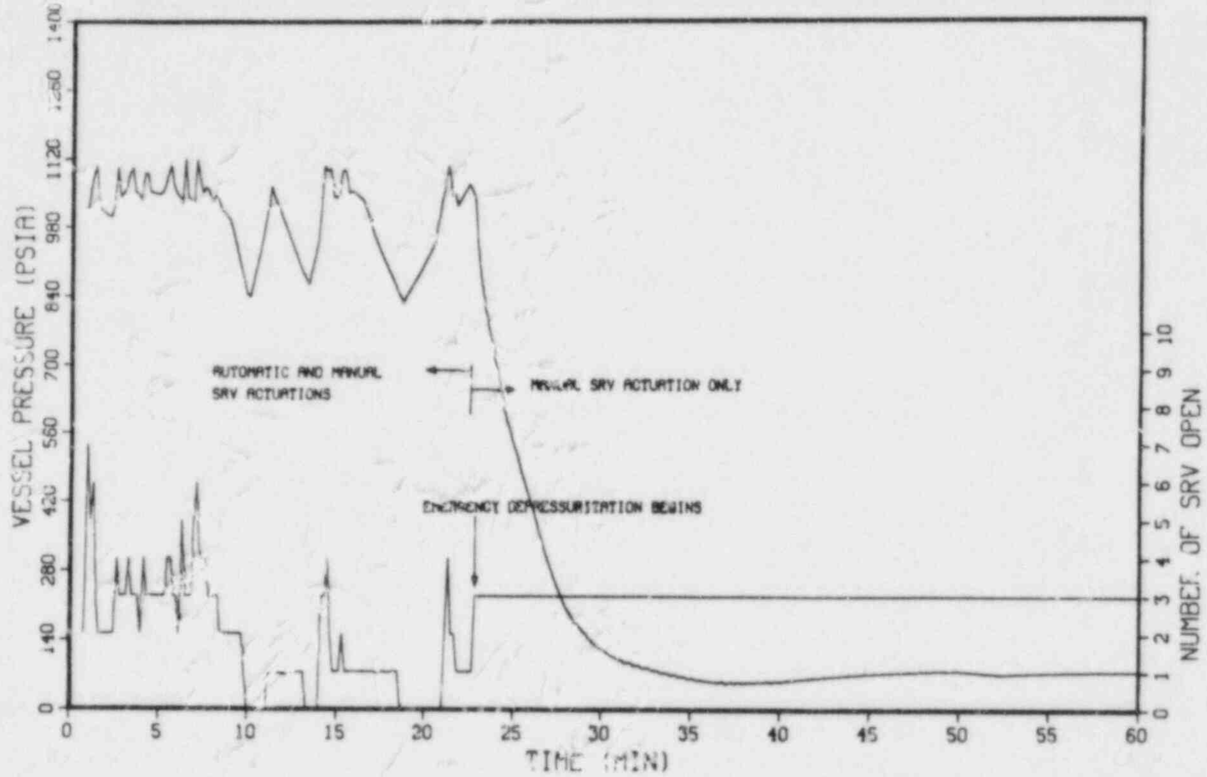

Fig. 4.23. EPG operator action sequence with failure of SLC system — vessel pressure.

ORNL-DWG 84-4548 ETD

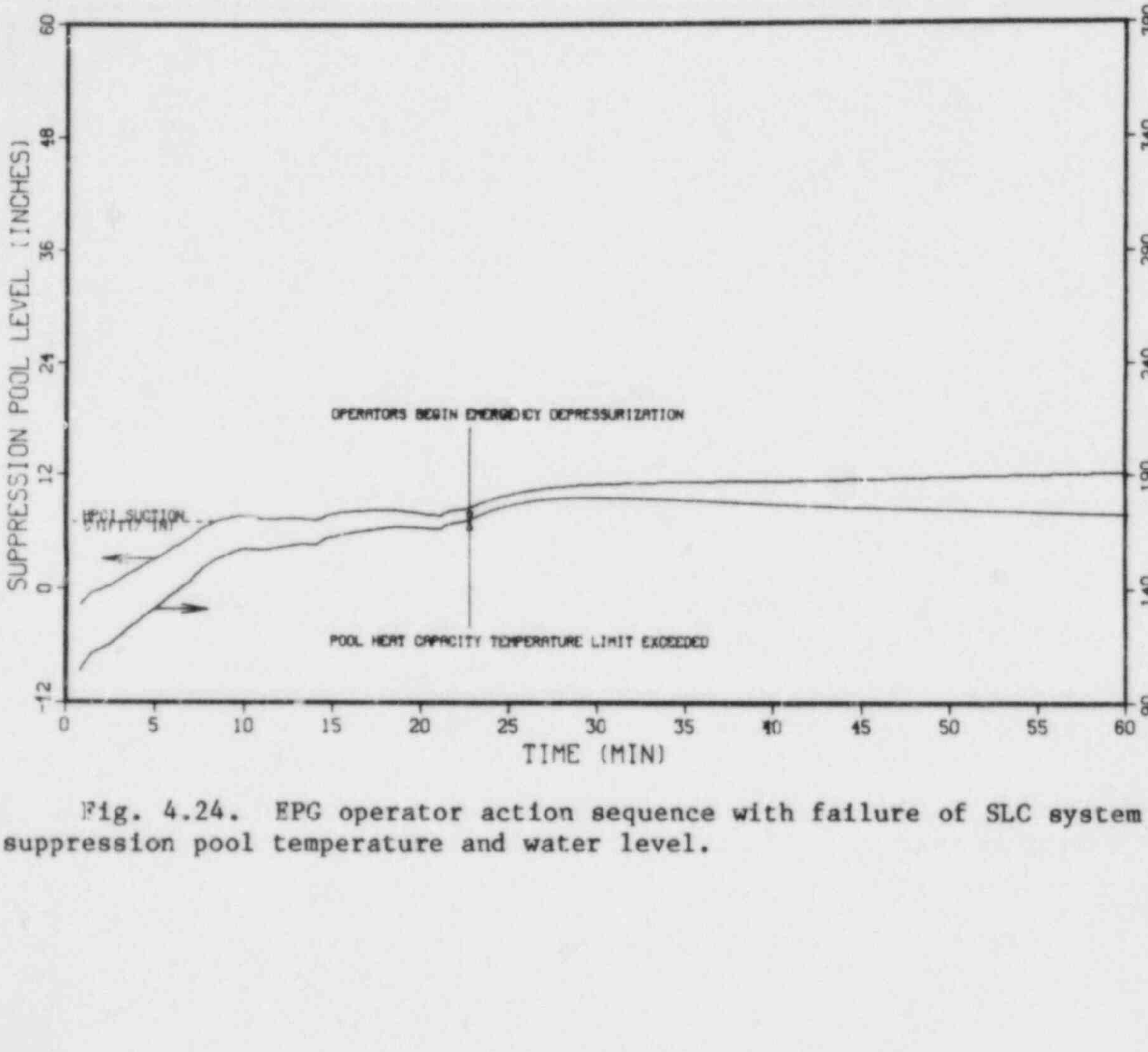

Fig. 4.24. EPG operator action sequence with failure of SLC system — suppression pool temperature and water level.

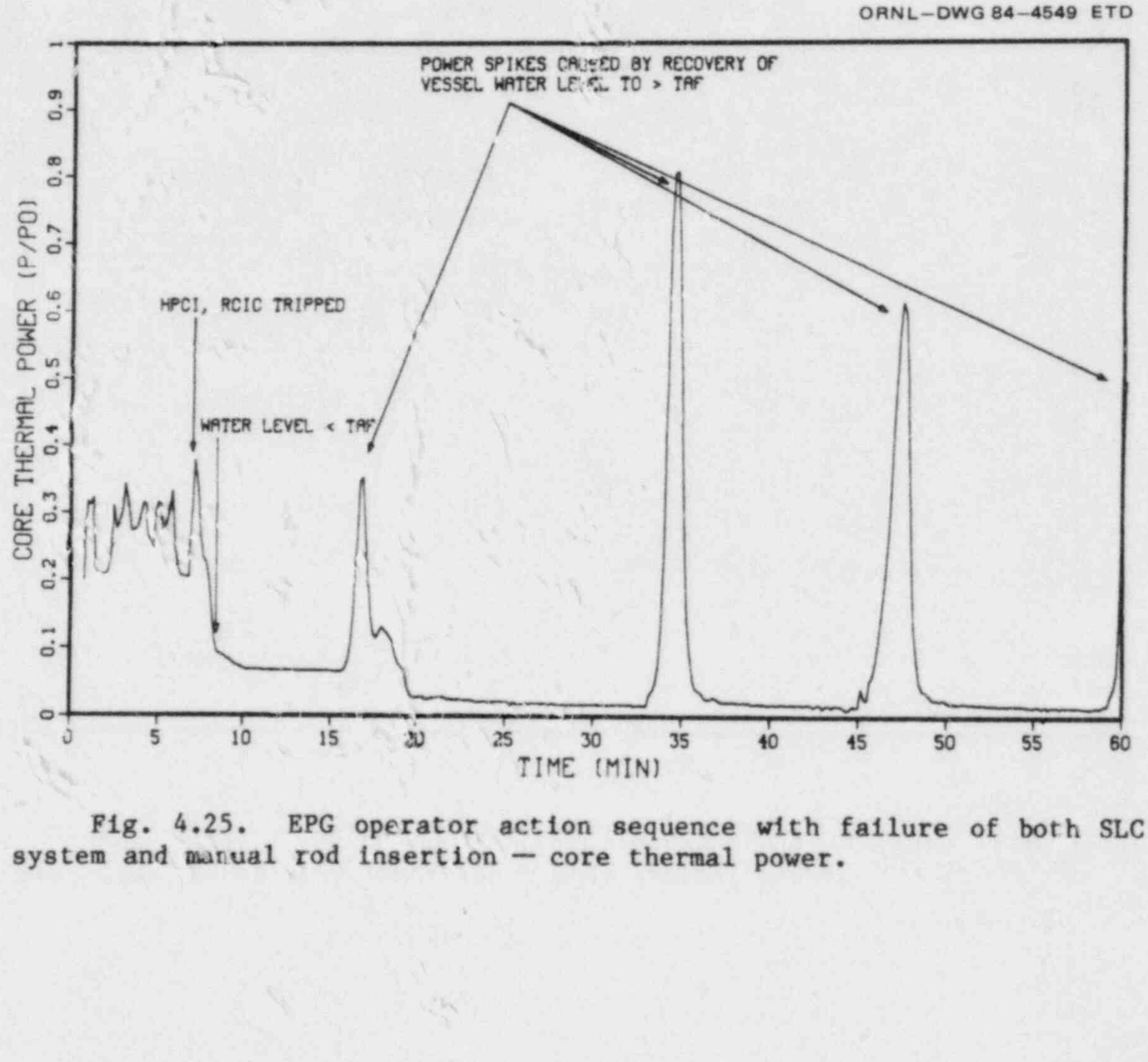

Fig. 4.25. EPG operator action sequence with failure of both SLC system and manual rod insertion — core thermal power.

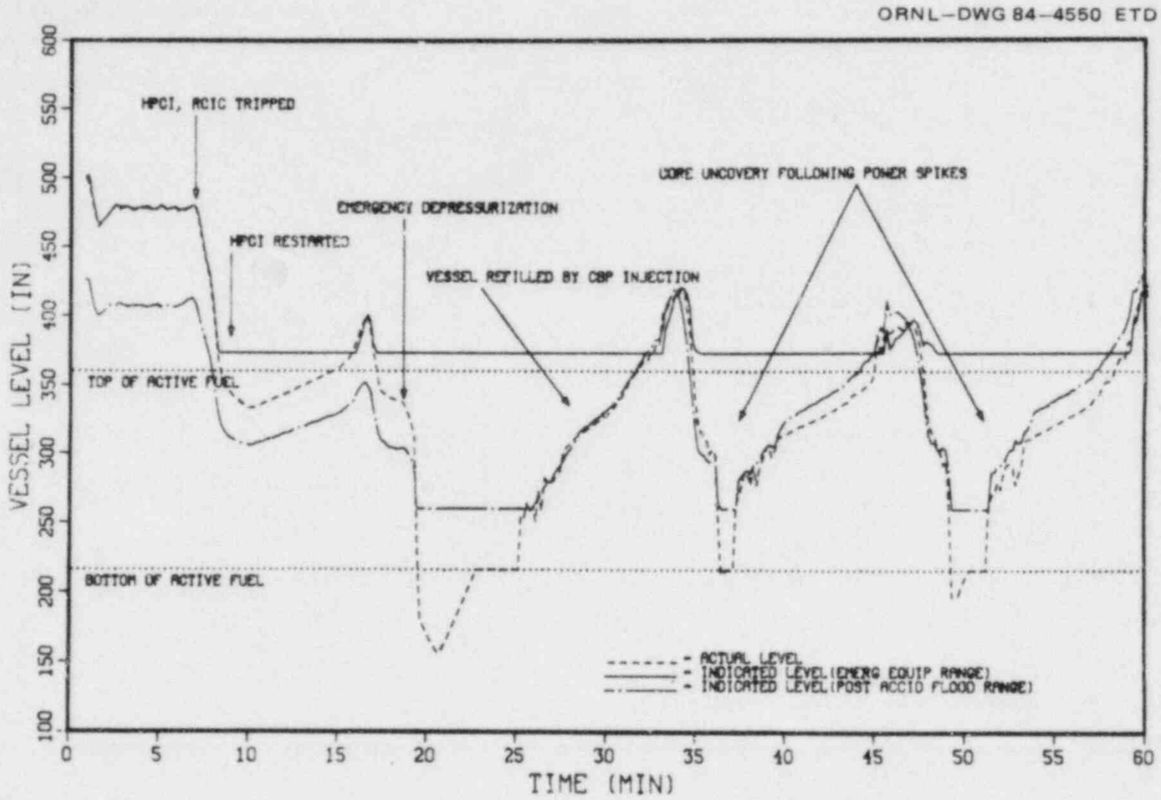

Fig. 4.26. EPG operator action sequence with failure of both SLC system and manual rod insertion — vessel water level.

ORNL-DWG 84-4551 ETD

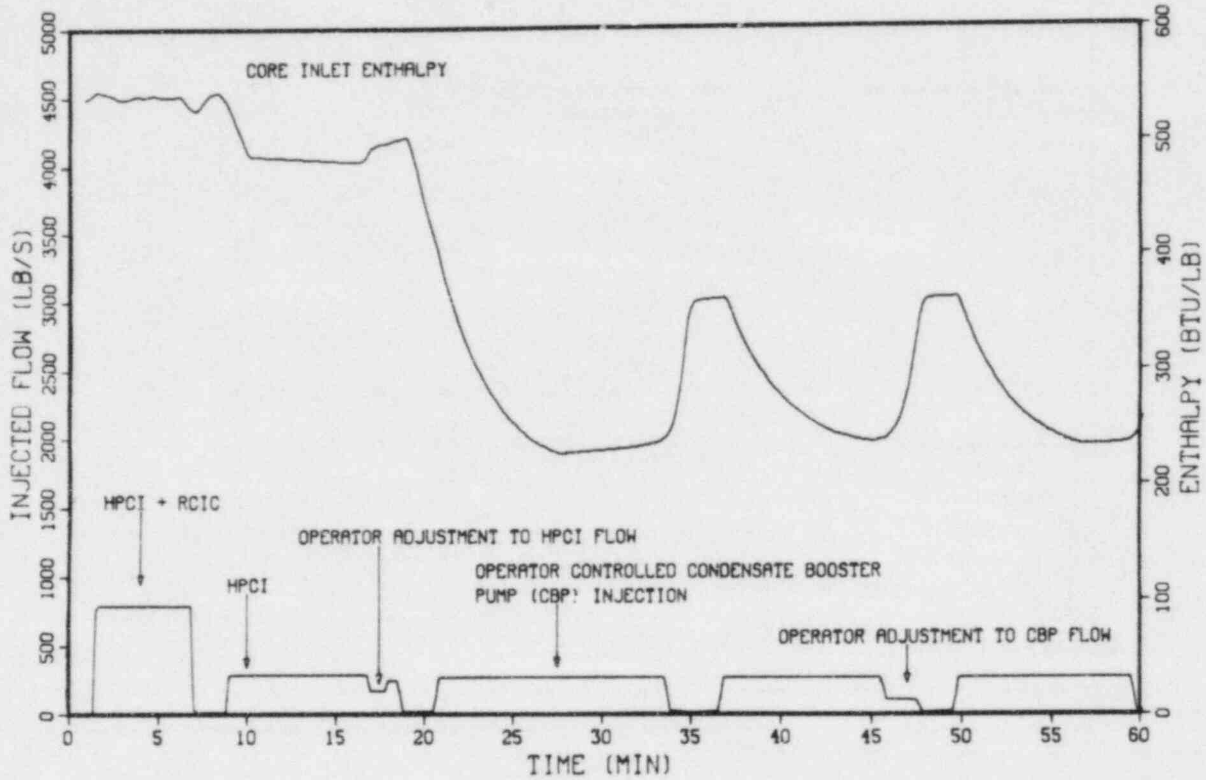

Fig. 4.27. EPG operator action sequence with failure of both SLC system and manual rod insertion — injected flow.

ORNL-DWG 84-4552 ETD

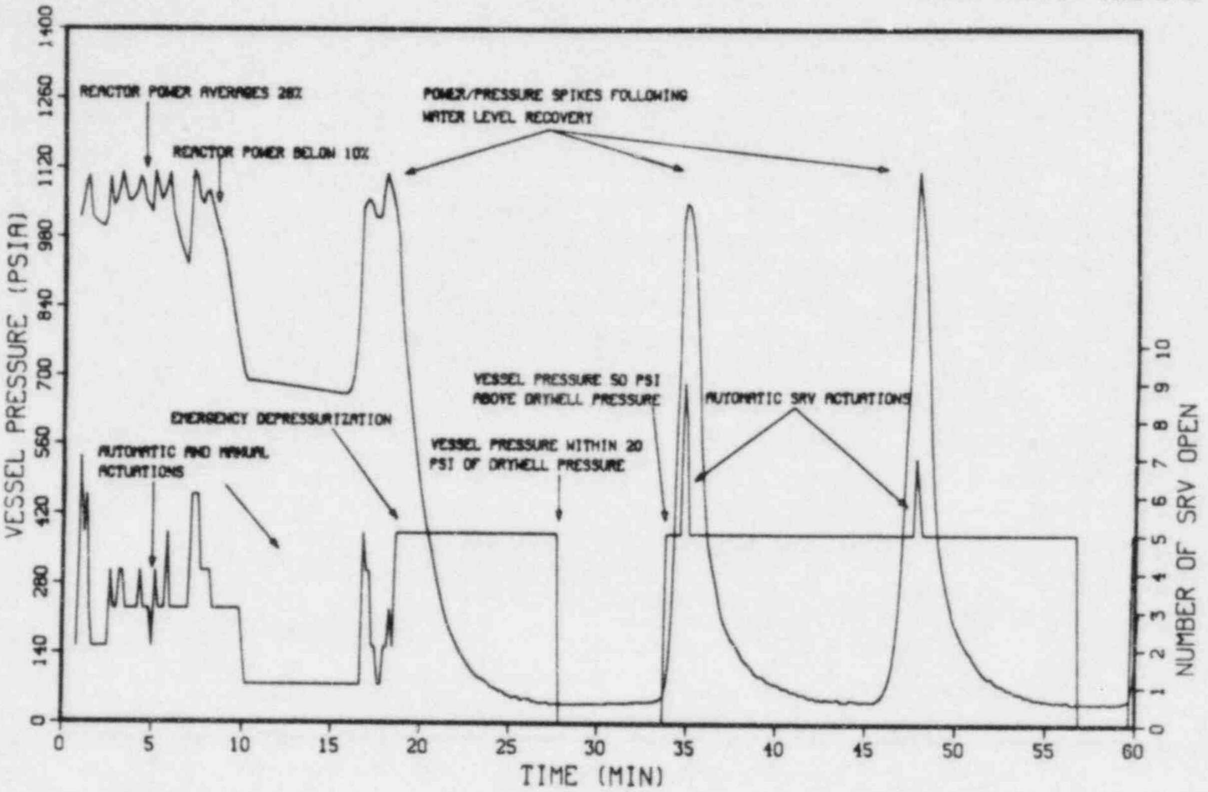

Fig. 4.28. EPG operator action sequence with failure of both SLC system and manual rod insertion — vessel pressure.

ORNL-DWG 84-4553 ETD

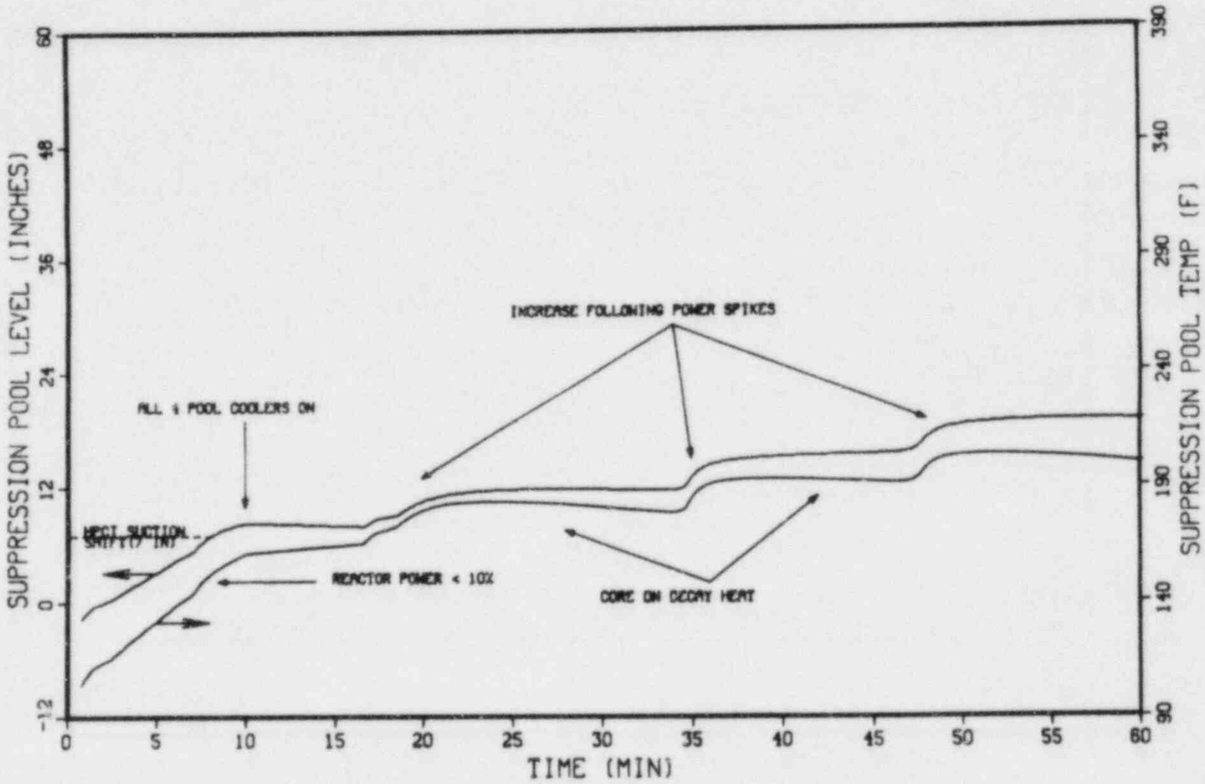

Fig. 4.29. EPG operator action sequence with failure of both SLC system and manual rod insertion — suppression pool temperature and water level.

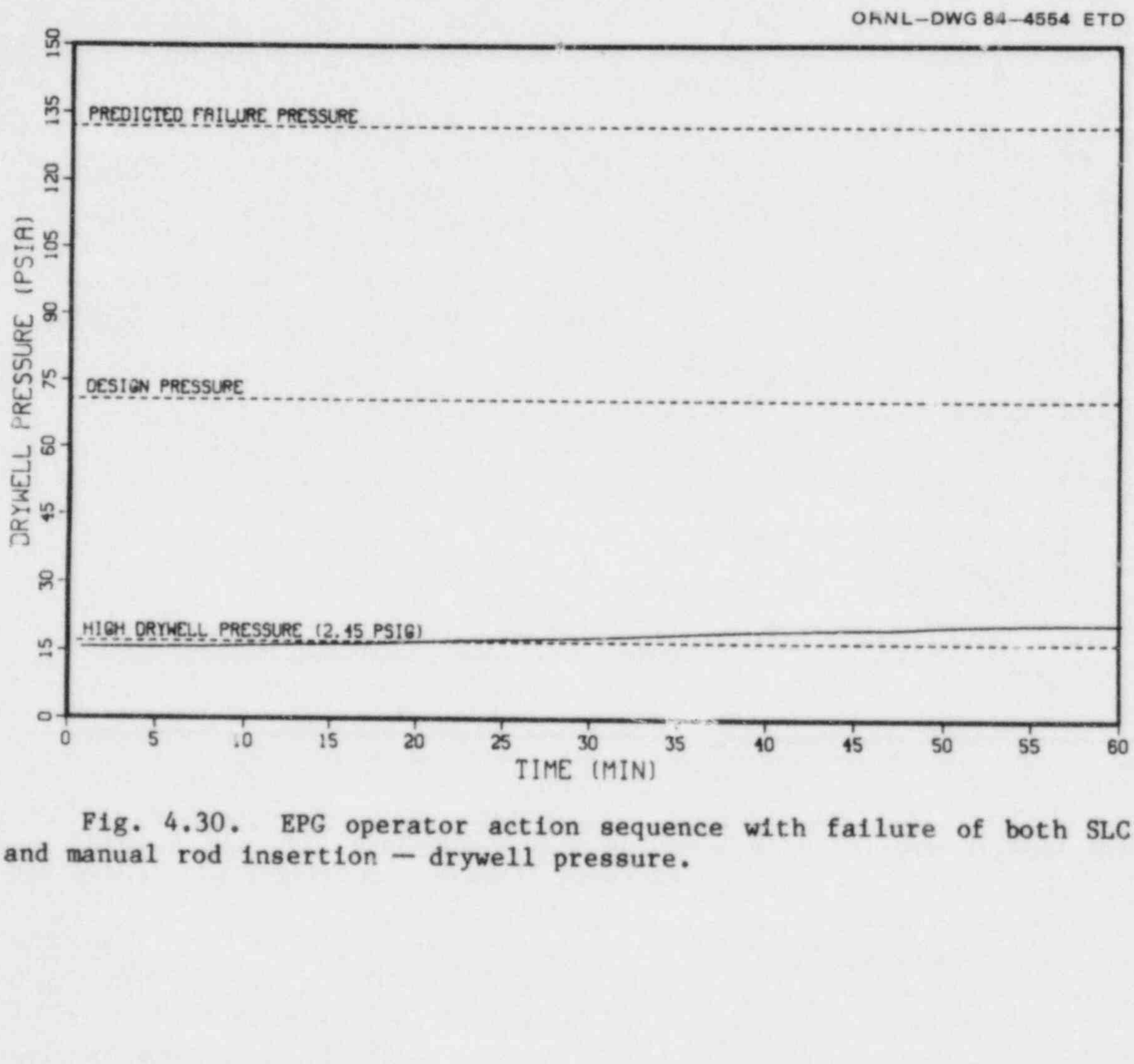

Fig. 4.30. EPG operator action sequence with failure of both SLC and manual rod insertion — drywell pressure.

ORNL-DWG 84-4555 ETD

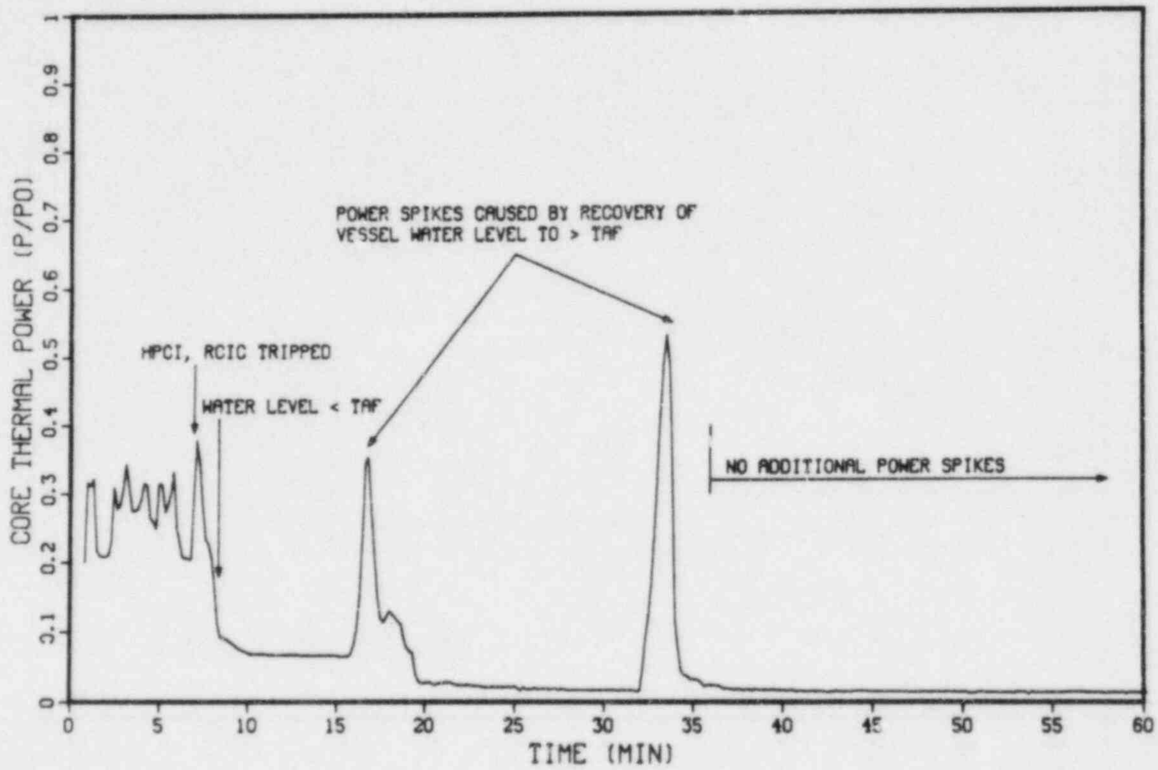

Fig. 4.31. Modified EPG level control sequence with failure of both SLC system and manual rod insertion — core thermal power.

ORNL-DWG 84-4556 ETD

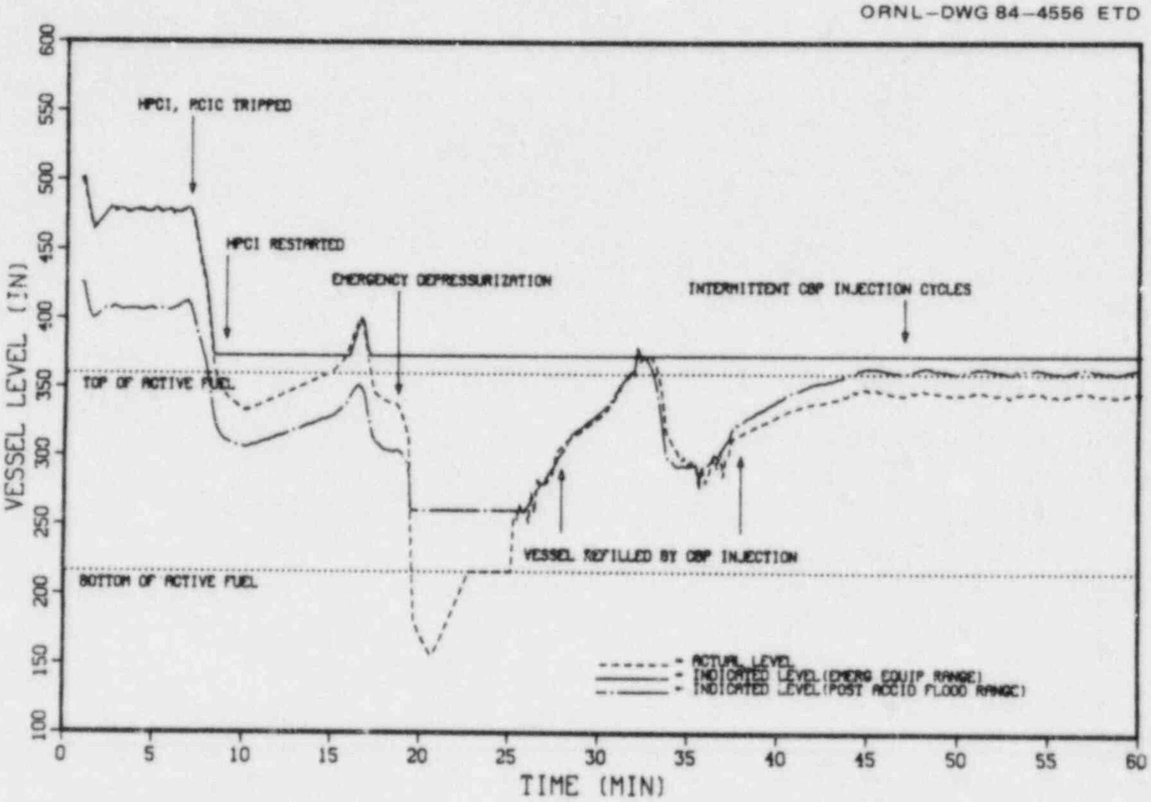

Fig. 4.32. Modified EPG level control sequence with failure of both SLC system and manual rod insertion — vessel water level.

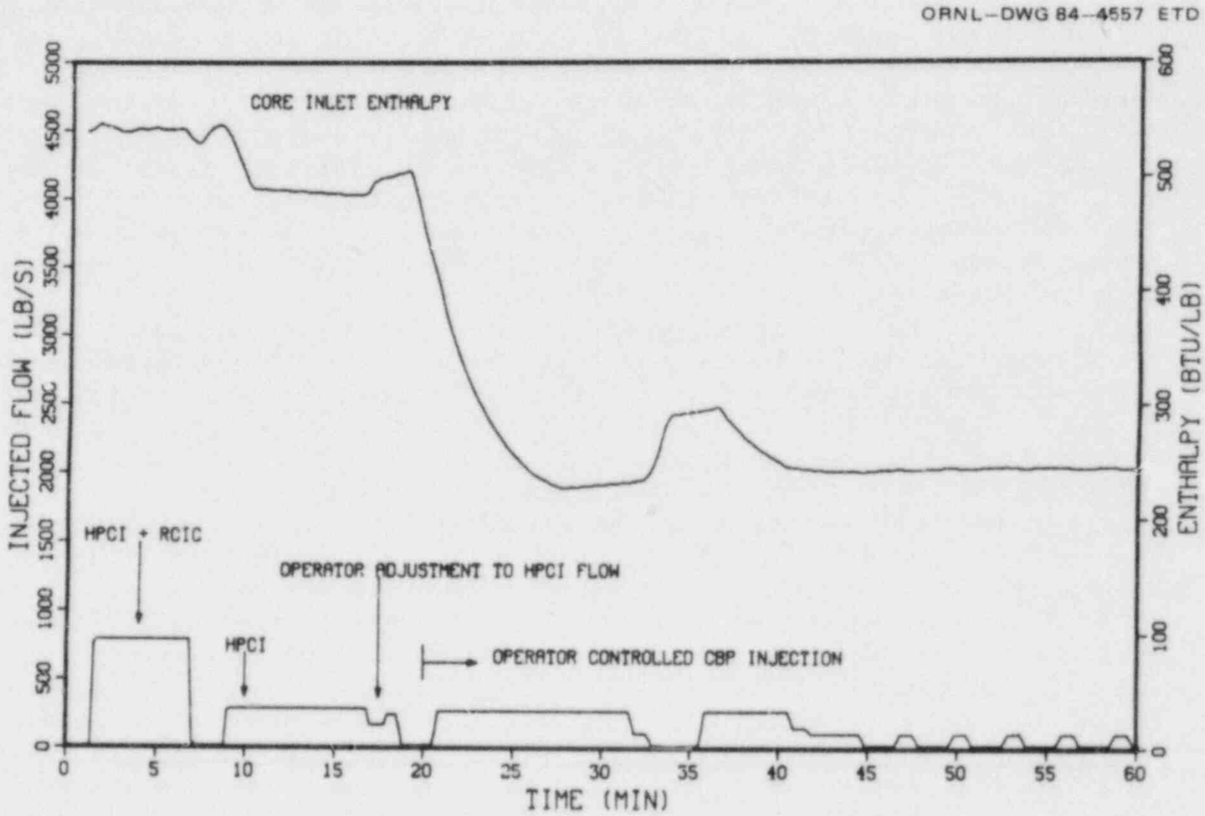

Fig. 4.33. Modified EPG level control sequence with failure of both SLC system and manual rod insertion — injected flow.

5. INSIGHTS AND RECOMMENDATIONS CONCERNING OPERATOR ACTIONS FOR THE MSIV CLOSURE — ATWS

The sequence of events for the case of an MSIV-closure initiated ATWS with no operator action was discussed in Chapt. 3. Without operator action, there is no manual rod insertion, injection of sodium pentaborate solution, or pressure suppression pool cooling. There is also no operator action to lower reactor vessel water level, but the HPCI system fails on high lube oil temperature so the water level eventually falls to below the top of the core anyway. There is no operator action to prevent ADS actuation, automatically initiated by the combination of low reactor vessel water level and high drywell pressure; the reactor vessel depressurizes and the large-capacity, low-pressure injection systems reflood the core, causing a power and pressure excursion even though the ADS valves remain open. With the reactor vessel again pressurized, the low-pressure systems cannot inject, vessel water level falls, and the depressurization — vessel reflood — power excursion — vessel repressurization cycle repeats. Containment failure is predicted to occur after just 37 min.

Chapter 4 is in effect a study of the efficacy of the operator actions mandated by the BWR Owners Group Emergency Procedures Guidelines (EPGs) in removing the many undesirable characteristics of the sequence of events described in Chap. 3. No attempt is made to adjust for the probabilities that the operator might not do exactly as the procedures prescribe; it is assumed that the procedures are followed exactly. The basic strategy of the EPGs can be described as a three-step process: (1) begin injection of sodium pentaborate, (2) lower the reactor vessel water level to the top of the core, reducing reactor power and the rate of pressure suppression pool heatup, and (3) when enough sodium pentaborate has been injected to induce hot shutdown if mixed evenly within the reactor vessel at normal operating level, restore water level to its normal operating range. During the period when step (2) is in effect, the water level is too low to support natural circulation and the core inlet flow is too small to sweep the injected sodium pentaborate into the core. Initiation of step (3) produces a large core inlet flow to reestablish reactor vessel water level and once this is done, natural circulation is reestablished. This sweeps the previously injected sodium pentaborate up into the core and produces hot shutdown.

The results discussed in Chap. 4 clearly show that the procedures specified by the EPGs are effective if properly carried out and that the Severe Accident situation described in Chap. 3 can be and should be avoided if the operators take the specified actions and all equipment functions as designed. Nevertheless, we have identified some difficulties with the procedures that we believe might confuse the operators and therefore have the potential to convert what should be a stable situation into an unstable one because of well-intentioned but counter-productive operator action. We have some suggestions to offer in this regard, based both upon our observations of ATWS runs made at the TVA Browns Ferry control room simulator as part of this study and upon our calculations. These suggestions form the bases for this chapter.

In general, we recommend that the ATWS accident procedures be separated from the overall Emergency Procedures Guidelines. The occurrence of an ATWS would produce such dramatic effect that it is inconceivable that its unique signature would escape the attention of the operators. Yet the operator actions required to mitigate an ATWS are in many cases diametric to the operator actions required for the set of accidents that might occur with the reactor shutdown and limited to decay heat power. Thus the present inclusion of the ATWS strategy among the plans for operator action to cope with other accidents have produced a set of written instructions that are unnecessarily complicated and invite confusion. The separation of the two would produce a much clearer set of instructions to be followed in the event of ATWS, and in all other cases as well.

We also make the general recommendation that, in the ATWS procedures, the operator be given guidance as to the amount of reactor vessel injection that would be required to maintain the vessel water level at the top of the core. The procedures should stress that the required injection would increase if the ATWS were compounded by leakage from the reactor vessel and would decrease as the core is poisoned by SLC injection or manual rod insertion. However, without guidance, the operator would have no idea where to begin.

In Sect. 5.1, we offer two recommendations concerning revisions to the operator actions required by the BWR Owners Group Emergency Procedures Guidelines and we give the reasons for our recommendations. In Sects. 5.2 and 5.3, we revisit the appropriate operator-action sequences of Chap. 4 and demonstrate the effect of our recommendations.

5.1 Recommendations Concerning Operator Actions

First, it is recommended that the operator not attempt manual control of reactor vessel pressure under ATWS conditions. Given the present design, the operator would not know which SRVs were already open when he began his attempts to control relief valve operation. With several relief valves automatically open, operator action to open an already-open valve would result in no change except for a control panel light indicating that the valve solenoid was energized. For a previously closed valve, the operator action would open the valve, but after only a slight decrease in reactor vessel pressure, a previously-open valve would shut and reactor vessel pressure would remain about the same.

If the operators were persistent, continuing to go to manual open on relief valve after relief valve until a recognizable effect was achieved, they would suddenly be confronted with a rapid drop in reactor vessel pressure, inviting core flooding by the low-pressure injection systems and the concomitant power and pressure spikes. The Boiling Water Reactor is very sensitive to the void coefficient of reactivity and the response of reactor power to pressure changes is greatly magnified at low pressures.

Second, if the sodium pentaborate solution cannot be injected, the operators should trip the HPCI turbine at the time this situation is recognized. Reactor vessel injection would continue via the RCIC system and the CRD hydraulic system. Reactor vessel water level would drop below the top of the core, but a RELAP5 calculation[5.1] has shown that the velocity of the steam rising past the uncovered portion of the core would preclude significant core heatup. The operator could monitor water level on the Post Accident Flooding range, but should be cognizant that the instrument reading is several inches lower than the actual downcomer water level when the reactor vessel is pressurized.

These recommended actions are intended to permit the operator to maintain control of the situation and to concentrate his or her efforts upon alternate means of reactor scram, manual rod insertion, ensuring sodium pentaborate injection, and the initiation and maintenance of pressure suppression pool cooling. Power and pressure excursions are avoided. For the case without SLC injection, the downcomer water level stabilizes at a point below the top of the core. Therefore, the reactor power is less and consequently, the rate of pressure suppression pool heatup is minimized. These results are demonstrated in the following sections.

5.2 The Accident Progression with Successful SLC System Operation but Without Other Operator Actions

The purpose of this section is to briefly discuss the results of BWR-LACP calculations made to demonstrate the efficacy of the first recommendation offered in Sect. 5.1. Accordingly, it is assumed that the operators do not attempt manual control of reactor vessel pressure. In all cases, the SLC system is assumed to be initiated by the operators 5 min after MSIV closure and to inject sodium pentaborate solution at the rate of 56 GPM (0.004 m^3/s). The rate of dispersal of the poison into the core depends on the rate of inlet flow to the core, as discussed in Appendix A, Sect. A.1.4.

If the operator initiates the SLC system but does nothing else, the BWR-LACP results show that the HPCI booster pump suction shift from the CST to the pressure suppression pool would occur at 8.8 min and the HPCI system would be lost* at 16.3 min. Since the HPCI system injects at full automatic flow [5000 gpm (0.316 m^3/s)] during its period of operation, there is sufficient core inlet flow during this period so that the injected sodium pentaborate solution is well-mixed within the reactor vessel. Therefore a slow but steady decrease of core thermal power that begins with SLC system initiation (when the power is 27%) would continue until the time of HPCI failure when the power would have been reduced to about 22%.

After HPCI system failure, the core thermal power would decrease from 22% to less than 2% within 2 min. This is a direct result of the reduction of core inlet flow and the concomitant increase of core inlet enthalpy. It is important to note that the reactor vessel water level

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*Because of pressure suppression pool temperature of 190°F (361 K).

would only decrease slightly during this 2-min period.* After this temporary decrease, the vessel water level is predicted to steadily increase as a result of the continued injection of the RCIC, the CRD hydraulic, and the SLC systems.

At time 50 min, the reactor vessel water level would reach the high-level trip setpoint of the RCIC system. The core thermal power would be at decay heat levels. Pressure suppression pool temperature would be 198°F (365 K), increasing very slowly due to the lifting of one SRV about every 2.7 min.†

The reactor vessel water level is predicted to continue to increase even after trip of the RCIC system. Enough poison has been injected for the power to be limited to decay heat while injection continues via the SLC and the CRD hydraulic system.

Calculations were terminated at time 60 min. The predicted pressure suppression pool water temperature at this time is still about 198°F (365 K).

To briefly recap this accident sequence, the operator does <u>nothing</u> except initiate the SLC system. Core thermal power is slowly reduced. The HPCI system is lost, causing a sharp reduction in core thermal power. Since the generated steam flow is less than the continuing injection by the remaining high-pressure systems, the reactor vessel water level continues to increase. There is no relief valve discharge over a long period of time because the sensible heat requirements of the injected flow exceed the core thermal power. At the 60-min point, the reactor is fully shut down and the pressure suppression pool temperature is 198°F (365 K), increasing very slowly. Throughout the accident sequence, the reactor vessel water level is maintained at least 10 ft (3.05 m) above the top of the active fuel.

The effect of just one additional operator action — to institute pressure suppression pool cooling at the 30-min point — was investigated. Maximum suppression pool temperature would be 197°F (365 K) at 30 min. By time 60 min, the pressure suppression pool temperature would be reduced to 178°F (354 K).

Since analysis of the accident sequence of events described above clearly shows that the assumed sudden failure of the HPCI system at a pressure suppression pool temperature of 190°F (361 K) is a significant event, the calculation was repeated with the assumption that the HPCI system is immune to failure by lube oil overheating. Because the inserted poison would act to keep the core thermal power below that otherwise demanded by the continued high rate of injected flow, the reactor vessel water level would steadily increase. The calculation shows that the vessel water level would reach the common high level trip setpoint of the HPCI and RCIC systems at about time 21 min. Core thermal power

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*From 506 to 482 inches (12.85 to 12.24 m) above vessel zero. At the low point, this is still some 10 ft (3.05 m) above the top of the core.

†An automatic sequence of actuations as necessary to maintain reactor vessel pressure in the range 1055-1105 psig (7.38-7.72 MPa).

would be 20.0% at this point and pressure suppression pool temperature would be 206°F (370 K).

After the simultaneous trip of the HPCI and RCIC systems, core thermal power is predicted to decrease from 20% to less than 2% within about 2 min. Water level would decrease rapidly at first, then more slowly after the power decrease. Pressure suppression pool temperature would continue to increase slowly due to the periodic lifting of one SRV as necessary to maintain reactor vessel pressure. Calculations were terminated at 60 min. The predicted pool temperature at this time is 223°F (379 K) and the reactor vessel water level is 480 in. (12.19 m) above vessel zero.

Again, the case with one additional action of the operator to institute pressure suppression pool cooling at the 30-min point has been considered. Peak suppression pool temperature would be 211°F at the 30-min point.

The calculated accident scenarios discussed in this section indicate that if the SLC system is initiated by the operators within 5 min, then the MSIV-closure ATWS can be terminated successfully even if the operators take no additional action. However, pressure suppression pool temperatures in excess of 195°F (364 K) would occur during the first hour. Pressure suppression pool temperatures this high could of course be avoided if the operators took the additional steps necessary to implement pressure suppression pool cooling and to reduce the core power by decreasing the rate of HPCI system flow so that the reactor vessel water level was lowered to the top of the core during the period of sodium pentaborate solution injection. (Cases including consideration of the effect of reduced reactor vessel water level are discussed in Chap. 4).

5.3 The Accident Progression with Neither SLC System Operation nor Manual Rod Insertion

This section presents the results of BWR-LACP runs made to demonstrate the efficacy of the mitigative strategy recommended in Sect. 5.1 for the case without poison injection. The most severe example of MSIV closure-initiated ATWS is considered: failure of both SLCS boron injection and manual rod insertion. The effect of the failure of pressure suppression pool cooling is also analyzed. The results presented in this chapter can be compared to corresponding cases in Chap. 4, for which strict operator compliance with the EPGs was assumed.

5.3.1 The sequence of events

The sequence of events is summarized by Table 5.1. Important system variables are plotted on Figs. 5.1-5.5.

The first 5 min of this accident are essentially the same as the no-operator-action case (Chap. 3). Reactor power (Fig. 5.1) averages 28% while the HPCI and RCIC systems run at full capacity and the vessel water level (Fig. 5.2) averages 475 in. (12.1 m). The total injection flow (Fig. 5.3) during this time is 5706 gpm (360 l/s) including the 106 gpm (6.7 l/s) injection by the CRDHS, which runs continuously throughout the accident. The reactor vessel pressure (Fig. 5.4) cycles between 1120 and 1020 psia (7.72 and 7.03 MPa) in response to the automatically opening and closing SRVs. During the first 5 min the operators would be attempting without success to obtain an alternative scram, to begin manual rod insertion, or to begin SLCS injection of sodium pentaborate solution.

After 5 min, the operators trip the HPCI system turbine (per the second recommendation of Sect. 5.1), thereby reducing total injection flow from 5706 to 706 gpm (360 to 44.5 l/s). [The RCIC system is allowed to keep running at its full capacity of 600 gpm (37.8 l/s)]. The vessel water level decreases rapidly until the water level is near 312 in. (7.92 m), corresponding to 2/3 core coverage. The core power decreases in response to the increased core voiding as the water level decreases. As water level passes through the TAF the core power is about 10%. After 9.5 min the vessel level settles at 2/3 core coverage, and the core power settles at about 4% (including decay heat). This core power response is preferable to that of the equivalent case in Chap. 4 (see Sect. 4.3.3 and Fig. 4.25) on two counts: time averaged power is much lower, and there are no core power excursions.

With the actual vessel downcomer water level at 312 in. (7.92 m) above vessel zero, the Emergency Systems range level indication would be off-scale low. This would cause the operators some concern since the Emergency Systems range is the preferred indication, especially since it is calibrated for a hot, full pressure reactor vessel. The Post Accident Flooding range indication range would be on-scale and could be used to determine vessel water level; however, the procedures would have to inform the operator of the magnitude of error expected when a cold-calibrated level instrument is used to read the level of hot reactor coolant (see Sect. 4.1.2 and Table 4.1). For example, an actual level of 312 in. (7.92 m) above vessel zero of fully pressurized coolant at or near saturation would indicate as 76 in. below the top of the active fuel on the Post Accident flooding range, or 284 in. (7.21 m) above vessel zero. This is an error of 28 in. (0.71 m).

It should not be surprising that the vessel water level settles near 312 in. (7.92 m); this is the level of the 20 jet pump suction inlets. When water level in the downcomer annulus is well above the jet pump inlets, water from the downcomer passes freely through the jet pumps on its way to the core and the collapsed water level in the core is approximately equal to the water level in the downcomer annulus. As the downcomer water level approaches the elevation of the jet pump inlets, water from the downcomer annulus begins to see a significant flow resistance as it flows from the downcomer to the core (via the lower plenum). If water level decreases to below the jet pump inlets, no flow can pass from the downcomer; the water level in the core and in the downcomer annulus become essentially uncoupled.*

*There would be some leakage from the downcomer through jet pump diffuser seals, etc., but this flow would be insufficient to equalize the core and downcomer collapsed water levels.

With the downcomer water level (Fig. 5.2) near 2/3 core height, injection flow (Fig. 5.3) at 706 gpm (44.5 l/s) and core power (Fig. 5.1) at 3.9%, the BWR-LACP results predict that the bottom 2/3 of the active fuel, covered by a 2 phase boiling mixture of water and steam, would be critical and generating most of the core power; the top 1/3 would be steam blanketed. Of the total 3.9% core thermal power, 3.55% would be generated in the bottom 2/3 of the core, whereas the top 1/3 of the core would be generating only decay heat, about 0.35% power.

The BWR-LACP code does not estimate fuel temperatures or steam conditions for uncovered fuel. Results of an off-line hand calculation show that steam would exit the core at about 675°F (631 K) and that maximum fuel temperature would be in the neighborhood of 850°F (728 K), well below the threshold for fuel damage by oxidation of the zirconium cladding. Results of a RELAP5 calculation[5.1] for an almost identical accident sequence predict that the fuel would remain fully covered by the boiling mixture, with no steam-cooled region and no heating of the fuel (which would remain very close to the saturation temperature of the steam/water mixture in the core). Therefore, the BWR-LACP prediction may in this respect be conservative.

Throughout this accident, the operator, per the first recommendation of Sect. 5.1, makes no attempt to manually open SRVs. As a result, vessel pressure (Fig. 5.4) is controlled over a narrower range than in the equivalent case in Chap. 4 (Sect. 4.3.3, Fig. 4.28). During the first 5 min, before HPCI is tripped, the core steam production is high enough to require between three and four open SRVs. After HPCI is tripped, the core produces only enough steam to intermittently open one SRV. A single SRV would probably repeatedly cycle throughout the remainder of the accident.

The Browns Ferry SRVs are grouped in two banks of four and one bank of five SRVs with the SRVs in each group having the same nominal setpoint; nevertheless, the actual opening pressure for a given valve may (by the ASME code) differ by as much as 1% from the nominal setting for its group. Unless pressure increases very rapidly, the single SRV with the lowest actual setting opens, and reduces the pressure before it reaches the actual setpoint of any other SRV in the same nominally set bank.

Since pressure suppression pool cooling is initiated after 10 min, and because core power is only about 4%, the suppression pool temperature increases very slowly. After 43 min, the pool is at 165°F (347 K); at this point, the EPG suppression pool heat capacity temperature limit is exceeded and (see Fig. 4.10) an emergency depressurization of the reactor vessel is required. In accordance with the recommendation of Sect. 5.1 that pressure control not be attempted, it is assumed that the operators avoid the hazards of this undesirable depressurization. The suppression pool temperature continues to increase, and would after about 6 h be close to the maximum of 206°F (370 K) achieved during this accident. Subsequently, the pool cooling is able to remove heat from the pressure suppression pool as fast as it is added.

As a result of the increasing pressure suppression pool temperature and evaporation from the pool surface, the primary containment pressure increases. After 52 min, the drywell pressure exceeds 2.45 psig (118 kPa). This completes the set of conditions required* for initiation of the ADS timer, and after an additional 2 min, the ADS would automatically open six SRVs to rapidly depressurize the reactor vessel. However, the operators avoid the ADS actuation by resetting the timer before the expiration of the 2-min period, and approximately every 2 min thereafter until the end of the accident sequence when reactor vessel water level is restored and the ADS timer is deactivated.

If the defining system failures for this accident are assumed to be compounded by failure of the pressure suppression pool cooling function, the thermohydraulic conditions in the reactor vessel would be the same but primary containment conditions would be greatly different. The suppression pool temperature and pressure would increase more rapidly, and without bound. After 4.1 h, the suppression pool temperature would be about 345°F (447 K) and the drywell would be pressurized to its predicted[3.3] 132 psia (910 kPa) failure pressure.

5.3.2 Emergency action levels and timing

The timing of the declaration of emergency action levels is given by Table 5.2. The criteria for determination of emergency action levels are taken from the TVA Implementing Procedures Document applicable to the Browns Ferry nuclear plant.[4.4]

By following the vessel pressure and level control recommendations of Sect. 5.1, the operators are better able to control the course of the accident. For the case with suppression pool cooling, the highest emergency action level achieved during the accident sequence is Alert. The results discussed in Chap. 4 (Sect. 4.3.3 and Table 4.6) show that if the operators follow the EPGs for the same case, the emergency action level would have to be upgraded from Alert to General Emergency after 6 h, and that even with pool cooling there would be an eventual overpressure failure of the drywell.

For the case without suppression pool cooling, the Alert is upgraded to General Emergency after 187 min. This is 76 min later than predicted in Chap. 4 for the analogous case in which the operators follow the EPGs.

5.4 The Effect of Stuck Open Relief Valves

This section examines the consequences of compounding the defining system failures of the case discussed in Sect. 5.3 by including a stuck open SRV. This is done because the reliance upon automatic SRV

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*Required conditions also include reactor vessel water level <413.5 in. (10.5 m), and either RHR pump or Core Spray pump discharge pressure.

operation recommended in Sect. 5.1 would cause repeated cycling of one SRV and this would increase the likelihood of an SORV. The sequence of events is outlined in Table 5.3, and selected system variables are plotted on Figs. 5.6–5.9. The overall accident progression is similar to the case without a stuck open SRV discussed in Sect. 5.3. Notable differences are discussed below.

Although the SRV sticks in the open position at 3 min, the reactor vessel does not begin depressurizing until 9.7 min, when the core steam production is no longer sufficient to hold one or more SRVs open continuously at full pressure. Vessel pressure decreases until reaching a minimum pressure of 272 psia (1.88 MPa) at 22.5 min. A significant fraction of the inventory of hot water in the reactor vessel is vaporized during the depressurization. As a result, the vessel water level decreases to below the jet pump inlets (i.e. 2/3 core height). For a period of about 10 min, the 600 gpm (37.8 l/s) RCIC injection is refilling the downcomer annulus but the flow does not reach the core. During this period, the core is uncovered, subcritical, and generating only decay heat. This period of uncovery of active fuel is not long enough to lead to serious overheating of the fuel. Information provided in Ref. 5.2 shows that the core can be uncovered for periods of 10 min without severe fuel damage if the CRD hydraulic system is operating. The injection provided by the CRD hydraulic system is boiled in the lower core and provides steam cooling for the uncovered upper portion of the core.

As the rate of depressurization slows, the 600 gpm (37.8 l/s) RCIC injection plus the 166 gpm (10.5 l/s) CRDHS injection (which is higher at lower vessel pressures) is able to exceed the rate of inventory loss due to vaporization. The downcomer water level increases to above the jet pump inlets, and this re-establishes flow of the RCIC injection from the downcomer to the core. As the core refills, criticality is restored and total core power increases to about 4%. Increased core steam production, venting to the pressure suppression pool through the single stuck open SRV, partially restores reactor vessel pressure; after 31 min, pressure is stable at about 520 psia (3.59 MPa).

The operators are assumed here to take action as necessary to prevent undesirable and possibly dangerous flooding of the reactor vessel by the low pressure, high capacity injection systems. To accomplish this, the condensate and condensate booster pumps must be tripped at any time between 0 and 15.7 min and the Core Spray pumps must be tripped anytime between 6.2 min (when they auto-start on low vessel level) and 15.7 min. The RHR pumps are not tripped since it is desirable to keep the RHR system running in the pool cooling mode. Vessel flooding by the RHR pumps is prevented by closing the LPCI injection valves after expiration of the 5 min period during which they are interlocked in the full open position. When pumping at full flow in the pool cooling mode, the RHR pumps cannot inject into the reactor vessel through the open LPCI valves unless vessel pressure decreases to below about 300 psia (2.07 MPa), which it does not.

References for Chapter 5

5.1 R. C. Gottula, results presented at SASA Program Review Meeting, Silver Spring, MD, Jan. 10-11, 1984.

5.2 R. M. Harrington and L. J. Ott, *The Effect of Small-Capacity High-Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One*, ORNL/TM-8635, Nureg/CR-3179, Sept. 1983.

Table 5.1 Sequence of events for case with operator trip of HPCI, and with failure of both SLCS and manual rod insertion

Time (min)	Event	Comment
0	MSIV closure initiated	No scram
0—end	SRVs cycling on automatic initiation	No manual SRV actuations
0.1	Recirculation pumps tripped	At reactor vessel pressure 1135 psia (7.83 MPa)
0—end	CRDHS injection continues	At 108 gpm (6.8 l/s)
1	HPCI and RCIC automatically start	Vessel water level <476.5 in. (12.1 m)
1—end	RCIC runs at full capacity	600 gpm (37.8 l/s)
1.5	Suppression pool temperature exceeds 110°F (317 K)	EPG criterion for operator initiation of SLCS injection
3	Operator attempts to manually insert rods	No rod motion
5	Operator attempts to start SLCS	Pumps inoperative, don't start
5	Operator trips HPCI	To reduce core power and to prevent HPCI failure
6.2	Core Spray and RHR pumps start	On vessel level <413.5 in. (10.5 m)
6.8	Vessel water level below TAF	Emergency Systems range level indication off-scale low
9.5	Vessel water level at 2/3 core height	Post Accident Flooding range level indication 1/2 core height
9.5—end	Vessel water level stable at 2/3 core height	Upper 1/3 of core steam cooled
10	Operators initiate suppression pool cooling with all four coolers	Containment Spray Select and 2/3 Core Coverage Override hand switches actuated
43	Suppression pool heat capacity temperature limit exceeded	Operators do not depressurize
50	ADS 2-min timer starts automatically	Drywell pressure >2.45 psig (118 kPa) + vessel water level <413.5 in. (10.5 m) + RHR pump discharge pressure sensed
52—end	Operator must reset the ADS timer every 2 min to avoid ADS actuation	
60	Suppression pool temperature at 168°F (349 K)	Slowly increasing
360	Suppression pool approaching maximum temperature	206°F (370 K) maximum bulk temperature

Table 5.2. Timing of Emergency Action Levels for case with operator trip of HPCI and failure of both SLCS and manual rod insertion

Time (min)	Action level	Criterion
	(a) Case with suppression pool cooling	
5	Alert	Failure of scram system
5–end	Alert	Reactor still not shut down
	(b) Case without suppression pool cooling	
5	Alert	Failure of scram system
10	Alert	Loss of shutdown cooling
187	General Emergency	Drywell pressure >50 psig (446 kPa)

Table 5.3 Sequence of events for case with operator trip of HPCI, failure of both SLCS and manual rod insertion, and one stuck open relief valve

Time (min)	Event	Comment
0	MSIV closure initiated	No scram
0.1	Recirculation pumps tripped	At reactor vessel pressure 1135 psia (7.83 MPa)
0—9.2	SRVs cycling on automatic initiation	No manual SRV actuations
0—end	CRDHS injection continues	Between 108 and 166 gpm (6.8 and 10.5 l/s)
1	HPCI and RCIC automatically start	Vessel water level <476.5 in. (12.1 m)
1—end	RCIC runs at full capacity	600 gpm (37.8 l/s)
1.5	Suppression pool temperature exceeds 110°F (317 K)	EPG criterion for operator initiation of SLCS injection
3	SRV sticks in open position	Failure to close after automatic actuation
3	Operator attempts to begin manual control rod insertion	No rod motion
5	Operator attempts to start SLCS	Pumps inoperative, don't start
5	Operator trips HPCI	To reduce core power and protect HPCI turbine
6.2	Core Spray and RHR pumps automatically start	On vessel level <413.5 in. (10.5 m)
6.8	Vessel water level at TAF	Emergency Systems range level indication off-scale low; Post Accident Flooding range indicates vessel level at 323 in. (8.2 m)
6.9	Reactor power <10%	
9.7	Reactor vessel starts depressurizing	Reactor power <5%
9.8	Vessel water level below 2/3 core height	Circulation from downcomer annulus to lower plenum and core stops
9.9	Reactor core subcritical	Power decreaseing to decay heat
10	Operators initiate pressure suppression pool cooling with all 4 pool coolers	Containment Spray Select and 2/3 Core Coverage Override handswitches actuated

Table 5.3 (continued)

Time (min)	Event	Comment
12	Active fuel region of core uncovered	
14.3	Minimum downcomer water level of 278 in. (7.06 m) reached	About 2.8 ft (.86 m) below the jet pump inlets
15.7	LPCI and CS injection valves open	At vessel pressure <450 psig (3.1 MPa); LPCI valves interlocked open for 5 min
15.8	Core Spray pumps, condensate, and condensate booster pumps tripped	To prevent vessel flooding
20.7	LPCI injection valves closed	After expiration of 5 min interlock, but before any injection
22.5	Vessel water level above 2/3 core height	Circulation from downcomer annulus to lower plenum and core reestablished
22.5	Active fuel region of core 2/3 covered	Core critical again
22.5	Minimum vessel pressure of 272 psia (1.88 MPa) reached	
24—end	Vessel water level stable at 2/3 core coverage	
31—end	Vessel pressure stable at 520 psia (3.59 MPa)	
32	ADS 2-min timer starts automatically	Drywell pressure >2.45 psig (118 kPa) + vessel level <413.5 in (10.5 m) + RHR pump discharge pressure
34—end	Operator must reset the ADS timer every 2 min to avoid ADS actuation	
60	Suppression pool temperature at 172°F (351 K)	Slowly increasing
360	Suppression pool approaching maximum temperature	206°F (370 K)

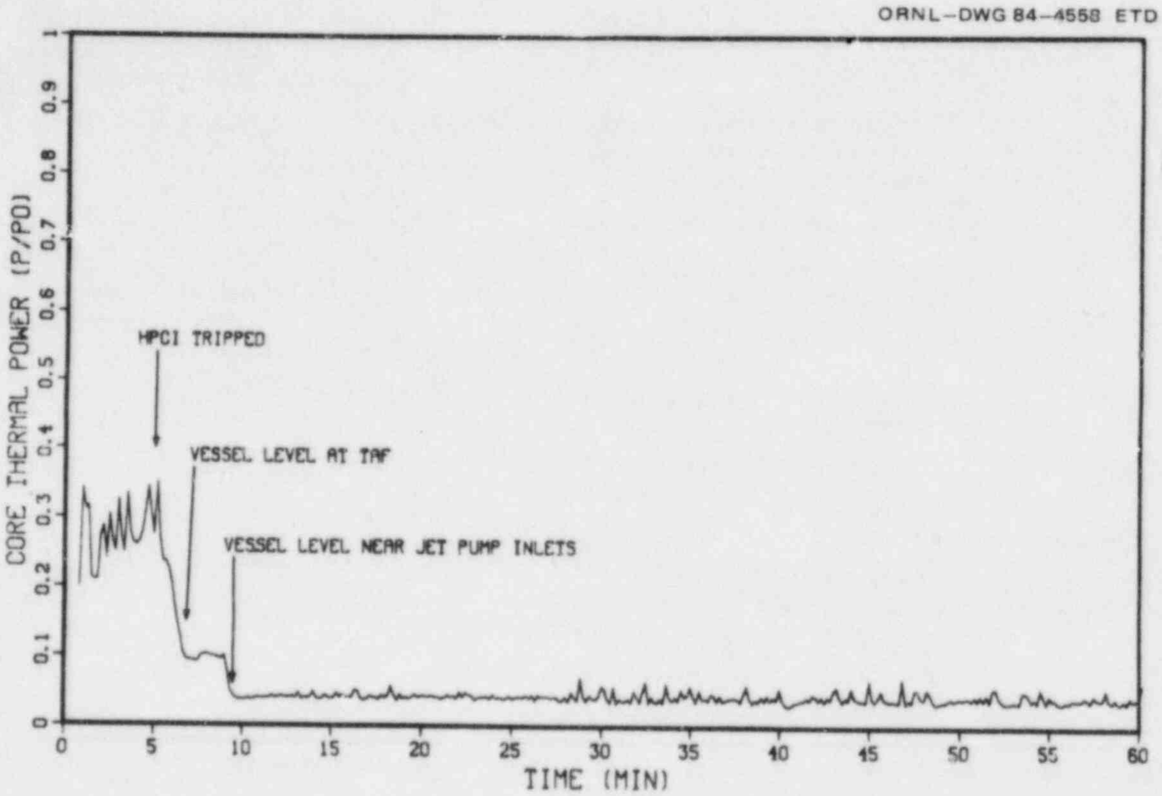

Fig. 5.1. Recommended level and pressure control sequence with failure of both SLC system and manual rod insertion — core thermal power.

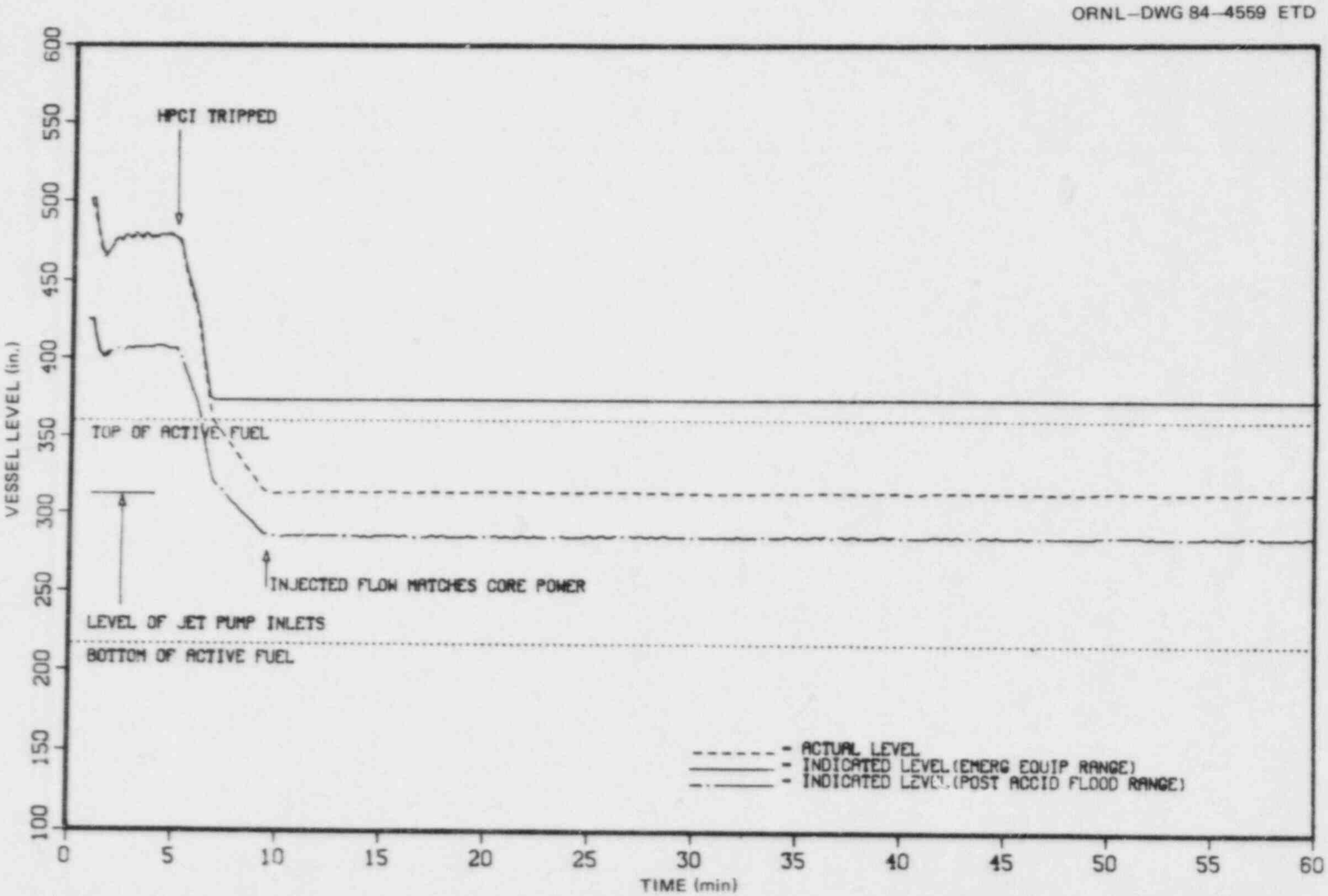

Fig. 5.2. Recommended level and pressure control sequence with failure of both SLC system and manual rod insertion — vessel water level.

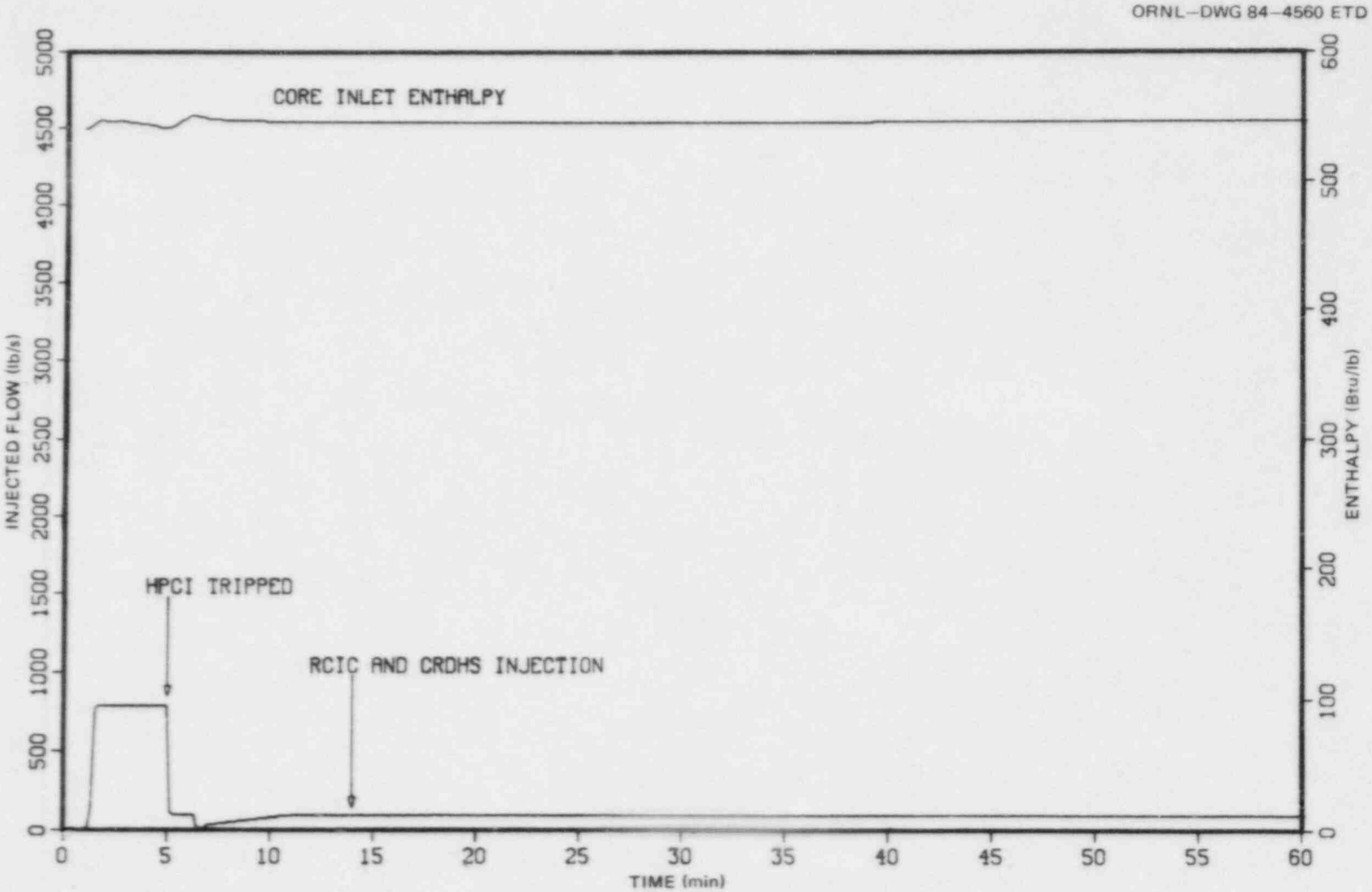

Fig. 5.3. Recommended level and pressure control sequence with failure of both SLC system and manual rod insertion — injected flow.

ORNL-DWG 84-4561 ETD

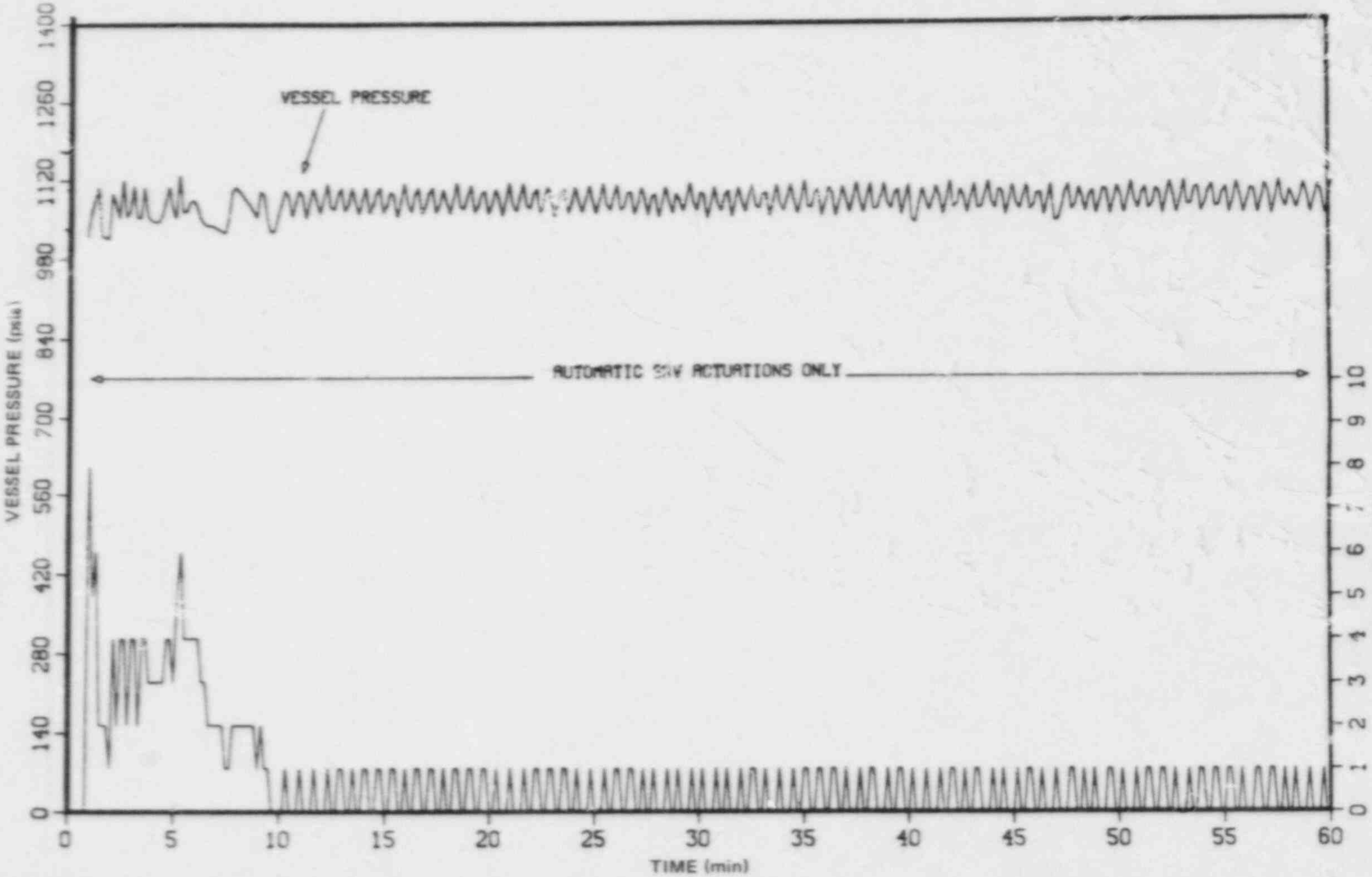

Fig. 5.4. Recommended level and pressure control sequence with failure of both SLC system and manual rod insertion — vessel pressure.

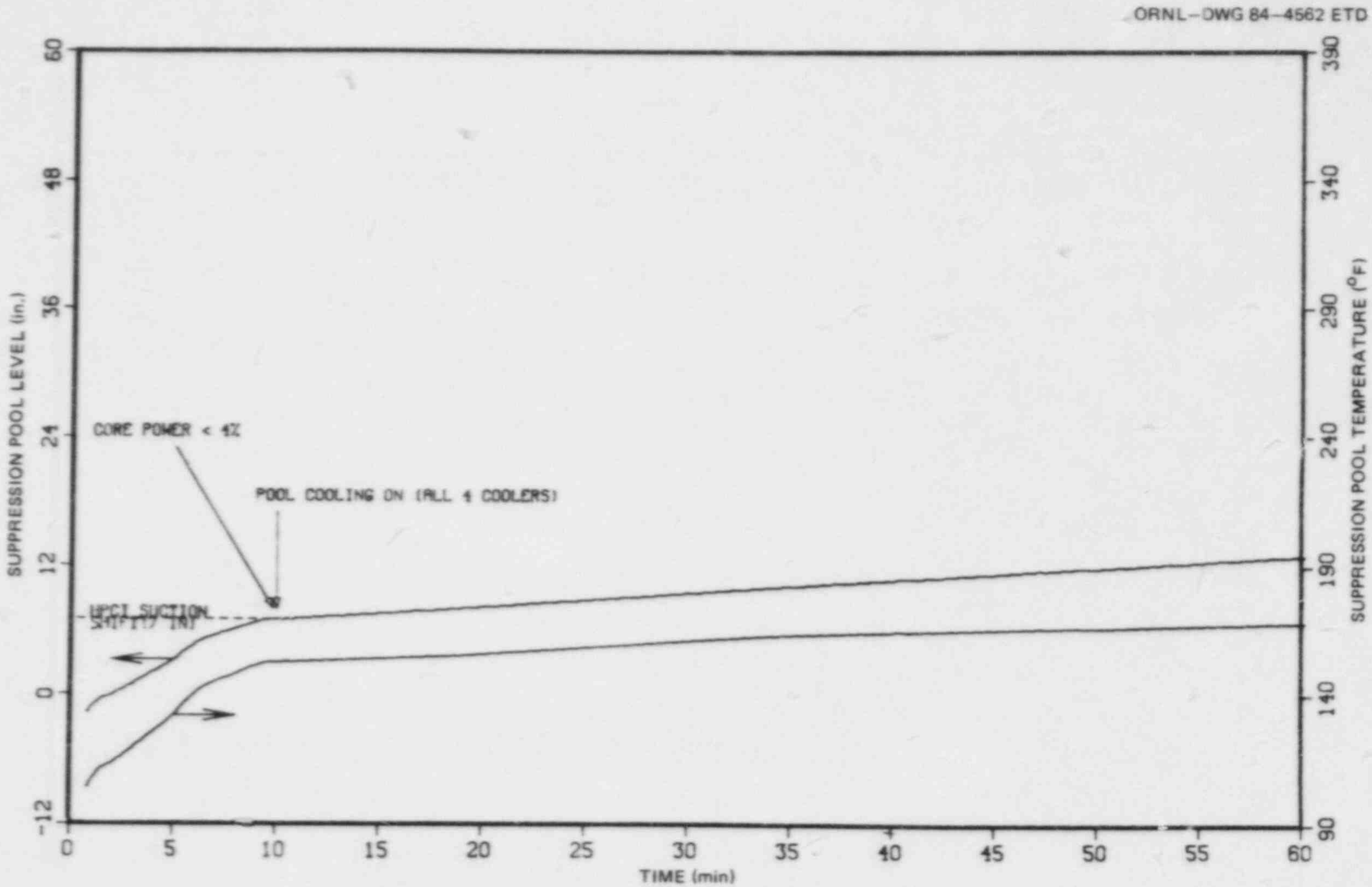

Fig. 5.5. Recommended level and pressure control sequence with failure of both SLC system and manual rod insertion — suppression pool temperature and water level.

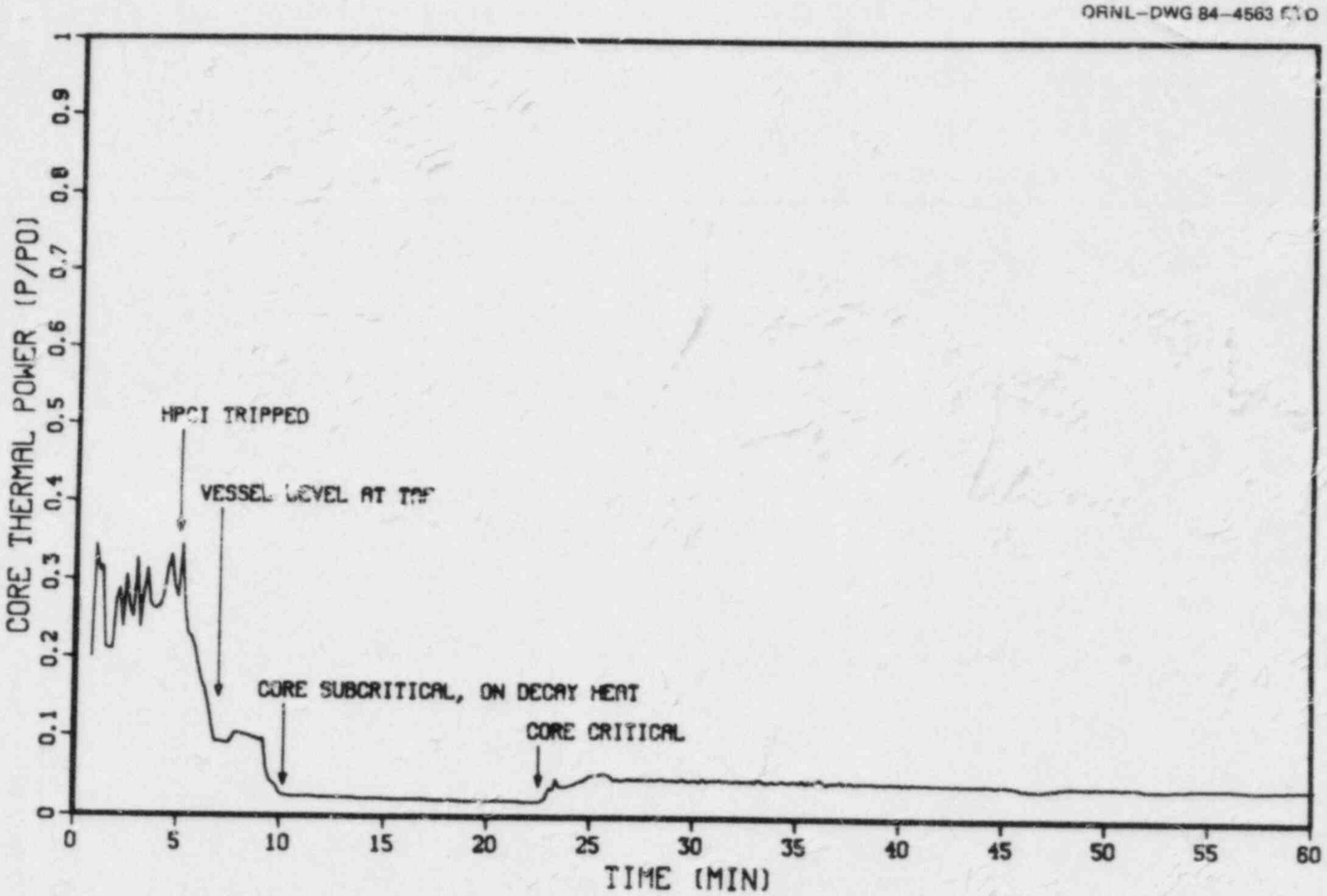

Fig. 5.6. Recommended level and pressure control sequence with failure of both SLC system and manual rod insertion and with one stuck open SRV — core thermal power.

ORNL-DWG 84-4564 ETD

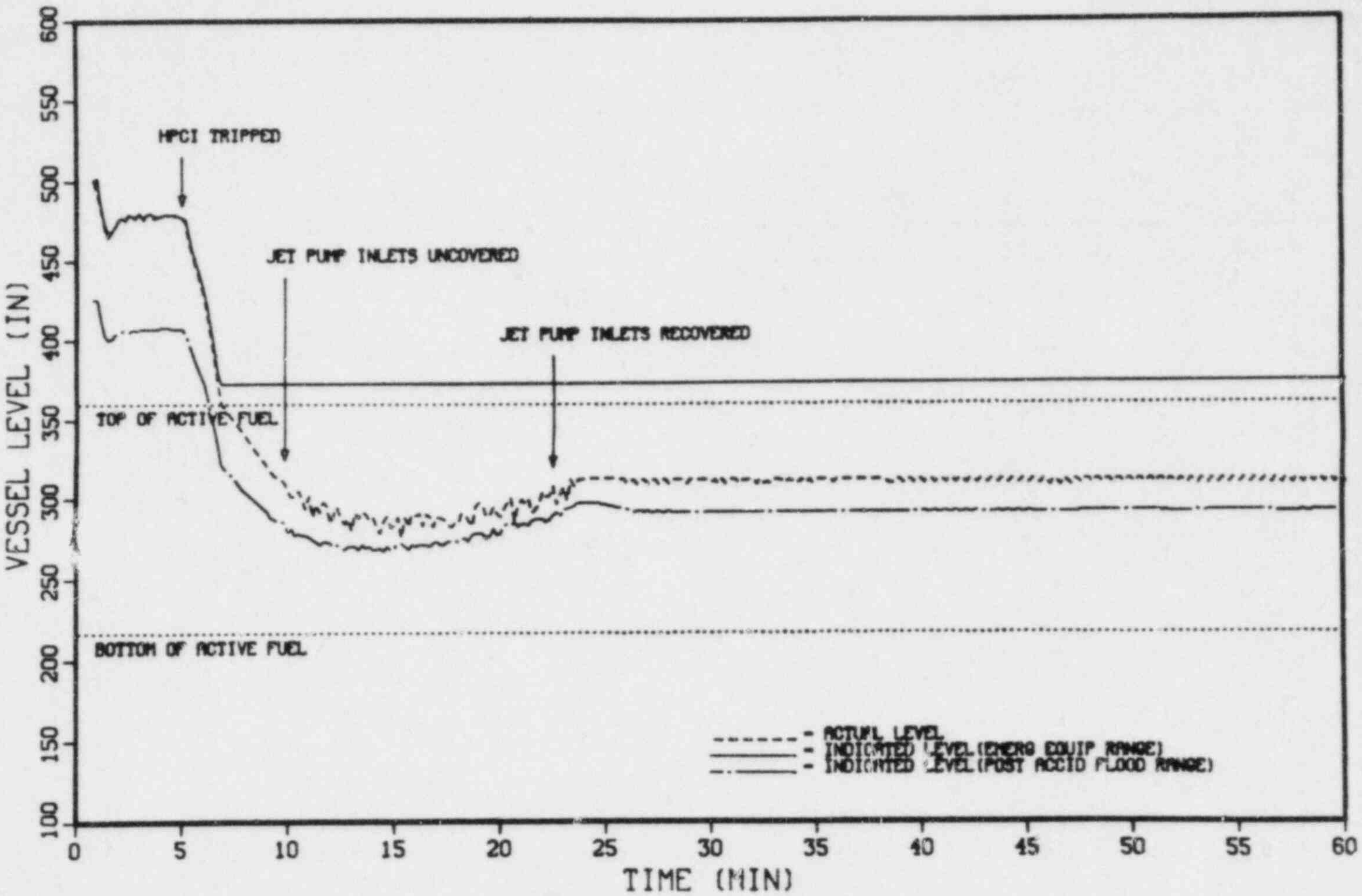

Fig. 5.7. Recommended level and pressure control sequence with failure of both SLC system and manual rod insertion and with one stuck open SRV — vessel water level.

ORNL–DWG 84–4565 ETD

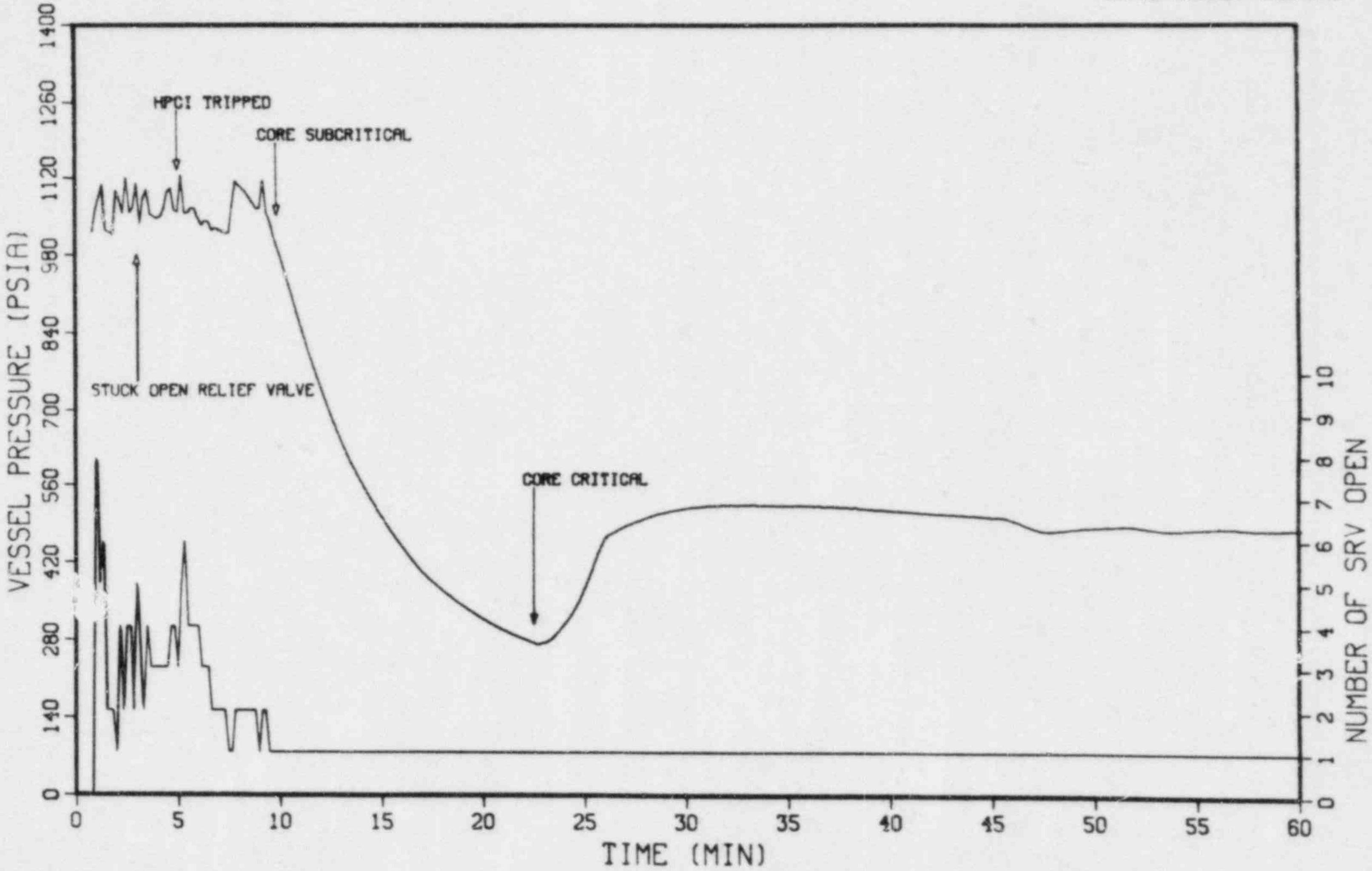

Fig. 5.8. Recommended level and pressure control sequence with failure of both SLC system and manual rod insertion and with one stuck open SRV — vessel pressure.

6. DISCUSSION OF UNCERTAINTIES

This plant-specific study of an MSIV-closure initiated ATWS is the fifth accident study based on Browns Ferry Unit 1 that has been conducted by the Severe Accident Sequence Analysis (SASA) program at Oak Ridge National Laboratory (ORNL). Both authors of this report also participated in the four previous studies so an appreciable amount of experience in severe accident analyses for a BWR of this design has been applied in this work. Nevertheless, this is unquestionably the most complex and difficult of the ORNL SASA program studies conducted to date. In spite of every effort by the authors to reduce the uncertainties associated with the results presented in this report, many remain, and some are significant. It is the purpose of this chapter to provide a discussion of the significant known uncertainties.

6.1 Uncertainties in the Calculational Model

The calculations discussed in this report were performed by R. M. Harrington using the BWR-LACP code which he developed at ORNL for use in the SASA program studies. The code incorporates reactor vessel, primary containment, and secondary containment models and in its present form is specific to Browns Ferry Unit 1. BWR-LACP was also used in the four previous ORNL studies, being expanded and improved in each case as necessary to meet the particular needs of each new study. The progressive stages in the development of the code are discussed in the reports[6.1–6.4] that document the results of the previous studies; additions made to the code for the ATWS accident sequence calculations are described in Appendix A of this report.

BWR-LACP results for a Station Blackout accident sequence have been compared to results calculated for the same sequence by the SASA team at INEL using RELAP4 Mod 7 (Ref. 6.5). BWR-LACP results for a small-break LOCA with condensate booster pump injection have been compared with results calculated at INEL for the same sequence by RELAP5 Mod 1 (Ref. 6.6). As part of the preparation for this study, and as discussed in Appendix A, an available INEL RELAP5 Mod 1.6 ATWS run was repeated at ORNL using BWR-LACP and the results were compared. Agreement has been qualitatively good in all cases.

Considering the relative simplicity of the primary system representation within the BWR-LACP code, the good agreement of its results with those of RELAP might be surprising. However, it should be recognized that primary system calculations for the portion of a severe accident sequence before core uncovery are much simpler for a BWR than for a PWR. In all cases, the MSIVs would be shut during a BWR severe accident sequence, the reactor vessel is isolated, the recirculation pumps are tripped, and the core inlet flow is a function only of the amount of makeup water injection and the effect of natural recirculation circuits within the reactor vessel. Therefore, sophisticated primary system analyses codes such as RELAP5, RETRAN, RAMONA, or TRAC are usually not necessary for BWR severe accident calculations; fundamental modeling of the processes within the reactor vessel in a properly benchmarked relatively simple code such as BWR-LACP is sufficient.

On the other hand, the interaction between the reactor vessel and its very small Mark I primary containment is very important to calculation of the progression of events for a severe accident sequence at a BWR plant of the Browns Ferry design. The BWR-LACP code is especially suited in this regard because it combines primary system and primary containment analytical capability.

Simply stated, the BWR-LACP code is a straight forward application of basic thermal hydraulic, heat transfer, and reactor kinetics theory which in its present form is specific to Unit 1 of the Browns Ferry plant. The code is not intended to be competitive with the more sophisticated and general primary system codes but rather is designed for the different purpose of rapid and inexpensive scoping analyses of the overall accident sequence in the primary system, primary containment, and secondary containment of Unit 1 at Browns Ferry. It has always been the policy of the SASA program at ORNL that important original findings obtained by use of BWR-LACP should be verified by subsequent application of the more sophisticated codes, and the requested verification of such BWR-LACP results has been forthcoming in the past.[6.5,6.6] The expansion of the BWR-LACP code to permit the calculation of reactor power as a function of reactor vessel makeup water injection rate and temperature, and reactor vessel pressure under ATWS conditions strengthens the need for continuation of this policy. Current overall SASA program planning includes the issuance of reports concerning Browns Ferry ATWS calculations by INEL using RELAP5/CONTEMPT and by BNL using RAMONA; the results presented in Chaps. 3, 4, and 5 of this report have early been made available to these laboratories and it is expected that the more sophisticated codes will provide the necessary reliable verification of the general accuracy of the sequence of events and the timing predicted by BWR-LACP.

The known modeling deficiencies in the BWR-LACP code are not believed to introduce significant inaccuracies in the predicted progression of the ATWS accident sequence. The known deficiencies are:

1. The calculated reactor decay heat power level is representative of infinite operation at 100% power and does not reflect the effect of the brief periods of reactor operation at elevated powers that would occur after recirculation pump trip. Reactor fission product decay power is calculated as if a reactor scram had occurred at the inception of the accident sequence.
2. Heat transfer from the uncovered portions of the fuel rods to the surrounding steam is not modeled during the brief periods of partial core uncovery that occur during the portion of the accident sequence analyzed by use of BWR-LACP.
3. During ATWS accident sequence runs performed at the TVA Browns Ferry Control Room Simulator in support of this study, it was observed that the calculated flows injected to the reactor vessel by the HPCI and RCIC systems fluctuated significantly with the rapid cycling of reactor vessel pressure that would occur during ATWS accident sequences in which the operator attempted to control reactor vessel pressure. This effect is due to the sophistication of the simulator modeling of the time delays inherent in the governor

control of the steam supply valves for the HPCI and RCIC systems. This modeling level is not replicated in the BWR-LACP code, in which the injection rate for the high pressure turbine-driven ECCS systems is assumed to be constant and as set by the operator and is not affected by reactor vessel pressure. This simplification has a neglible effect in the calculated results.

4. The calculation of reactor power does not include the effect of the relatively slowly changing xenon reactivity. The buildup of xenon after a power decrease can, over a long period, help to shutdown the reactor. Since most of the transients discussed in this report would have run their course in one or two hours, the buildup of xenon would not provide a significant fraction of the reactivity required to reach hot shutdown.
5. The model of the reactor vessel water level sensors assumes that the sensor reference legs move instantaneously to their equilibrium values: The Post Accident Monitoring range reference leg is always at drywell temperature, and the Emergency Systems range reference leg is always 40% of the way between drywell temperature and the reactor vessel saturation temperature (see Appendix A.5). This assumption introduces a slight inaccuracy during the most rapidly moving parts of the transients, but does not affect the final values reached. This is true because the reference legs will ultimately reach their equilibrium temperature.

In addition to the modeling considerations discussed above, uncertainties exist in the input parameters supplied to the BWR-LACP code for the study of the-MSIV closure initiated ATWS accident sequence. These include:

1. One very important assumption of the BWR-LACP ATWS model involves the in-vessel heating of injected HPCI or RCIC flow. As illustrated by the graph on Fig. 4.4, in-vessel feedwater heating causes a dramatic decrease in reactor thermal power when the vessel water level is reduced sufficiently to uncover the feedwater spargers.

   When the downcomer annulus water level is below the level of the feedwater spargers, the HPCI/RCIC injected flow is heated by direct contact condensatiaon of steam while falling toward the water surface beneath the spargers. The BWR-LACP input assumes[6.7,6.8] that a fall through 2 ft (0.61 m) of steam environment is sufficient to heat the injected water to saturation. With only saturated water entering the core, there is more in-core voiding and hence a lower power level, as shown on Fig. 4.4.

   Recent preliminary work at Brookhaven National Laboratory with the RAMONA code[6.9] has indicated that the amount of in-vessel heating of injected flow might be much less than assumed for BWR-LACP [even if the flow falls through as much as 12 ft (3.66 m) of invessel steam environment]. Consequently, the RAMONA code predicts much higher core power than does BWR-LACP when the reactor vessel water level is low. If the BNL results are sustained by ongoing peer review within the SASA program, this will have a overwhelming influence upon the planning for operator actions to mitigate ATWS transients. The reactor vessel water level reduction recommended in the EPGs would be much less effective in reducing the core power. Since the steady state core thermal power is determined by the injection rate (see Appendix B), the procedure of tripping the HPCI turbine recommended in Chap. 5 or some other means of ensuring reduction of the total injected flow would be necessary for mitigation of the MSIV-closure initiated ATWS.

2. The primary system events during the very brief period (50 s) after the MSIVs begin to close in which the effects of recirculation pump trip and feedwater turbine coastdown are dominant in determining the conditions within the reactor vessel cannot be modeled by the BWR-LACP code. Instead, the BWR-LACP calculations are initiated at time 50 s into the ATWS accident sequence using initial values taken from the results of the recent GE study discussed in Sect. 2.3.
3. It is assumed in this study that the only coolant loss from the reactor vessel is through the SRVs to the T-quenchers in the pressure suppression pool or via the steam supply valves to the RCIC or HPCI turbines. In fact, there would also be a slight leakage from the various components of the primary system into the drywell (less than 25 gpm) and a slight leakage through the shut MSIVs into the main condensers. The amount of leakage is uncertain and has been neglected in this study.
4. Leakage from the primary containment has been modeled as equivalent to that measured during actual containment integrated leak rate tests, which were conducted at 40 psia (0.274 MPa), as adjusted for differing containment pressures. This is only a realistic approximation to the actual leakage rates that might occur in a future accident sequence.
5. The HPCI system lubricating oil (gears, shafts, control system, etc.) employs a cooler for which the cooling water supply is the water being pumped by the system. In the ATWS accident sequences, the pressure suppression pool level rises quickly because a large amount of steam is condensed. This causes an automatic and irreversible shift of the HPCI pump suction to the overheated pool; HPCI failure by overheated lube oil will occur.

   In this study, HPCI system failure is assumed to occur at the time when bulk-average pressure suppression pool temperature reaches 190°F (361 K). This is 50°F (28 K) higher than the turbine manufacturer's recommended maximum for lube oil cooler inlet water temperature and of course the <u>oil</u> temperature at this time would be significantly higher. Nevertheless, the authors of this report cannot produce evidence showing that HPCI system failure would occur at this temperature. The reader should recognize, however, that pressure suppression pool water temperature would rise very rapidly in the MSIV-closure initiated ATWS sequence and therefore an increase in the assumed pressure suppression pool temperature at which HPCI system failure occurs would produce a delay in system failure of only a few minutes.
6. It has been assumed that the drywell coolers would fail when the drywell atmosphere temperature reaches 200°F (366 K). This is far beyond the design bases of the drywell coolers but it is of course uncertain at what temperature these coolers would actually fail. This assumption has little effect upon the time at which a high drywell pressure signal would be sensed as a result of evaporative

steaming from the overheated pressure suppression pool because in the ATWS sequences, the high drywell pressure signal occurs before the drywell temperature exceeds 200°F (367 K).

7. An important uncertainty in any ATWS analysis, regardless of the computer code employed, is the accuracy of the predicted core power. The core power determines the injection flow requirements and the rate of the suppression pool heatup. The BWR Owners Group Emergency Procedures Guidelines recommend that the operators reduce the reactor vessel water level to near the top of active fuel (TAF) to reduce the core power. If the core power after this maneuver, for example, were 12% instead of 9% (as BWR-LACP predicts for full system pressure), the suppression pool would heat up about 33% faster.

   No experiments have been performed to check the results of numerical predictions of core power level with water level at the TAF (and control rods at their full power withdrawn positions). One INEL estimate put the maximum uncertainty of RELAP-calculated estimates under these conditions at 100% (Ref. 6.10). The General Electric Company, in work performed for the BWR owners group, specified a maximum uncertainty band of 50% (Ref. 6.11). The uncertainty in the BWR-LACP core power calculations can reasonably be expected to be of the same order as those of the INEL or GE predictions. Sophisticated thermohydraulic/neutronic calculations are planned or underway at INEL, BNL, and GE; it is hoped that the results will reduce the current uncertainties in the estimates of core power under ATWS conditions.

6.2 Uncertainties with Regard to Operator Actions

MSIV-closure initiated ATWS sequences with operator action have been discussed in Chaps. 4 and 5 of this report. The written procedures that would guide the operators in the unlikely event that one of these accident sequences should actually occur are currently in the process of revision by the TVA. The revised procedures will be based upon the BWR Owners Group Emergency Procedure Guidelines,[6.12] with plant-specific data for Browns Ferry substituted in the appropriate places for the general example data provided in the guidelines. Every effort has been made by the authors of this study to consult with the TVA engineering staff as necessary to obtain the Browns Ferry plant specific data. As usual, TVA cooperation has been excellent and all available information has been obtained. Nevertheless, several uncertainties remain. These include:

1. The very important and somewhat controversial question of whether or not the operators will be instructed by the developing plant specific procedures to attempt to control reactor vessel pressure under ATWS conditions remains to be resolved. The Emergency Procedure Guidelines provide a general requirement for reactor vessel depressurization whenever suppression pool temperature exceeds

160°F (344 K).* This requirement is not based upon ATWS considerations but rather is based upon the desirability of assuring smooth condensation of SRV T-quencher discharge in the suppression pool by remaining within the parameters envelope of existing experimental investigation. Calculations performed attendant to this study show that once begun, the depressurization must be complete [i.e., to below 115 psia (0.793 MPa)] because the increased steam release to the pressure suppression pool during reactor vessel depressurization increases the pool heatup rate, and according to the graphical requirement for operator action (Fig. 4.10), the increased pool temperature requires further depressurization.

It has been assumed for the calculations presented in Chap. 4 of this study that the operators would act under ATWS accident conditions to depressurize the reactor vessel in accordance with the requirements of Fig. 4.10. However, the reader should note that the results of this study have indicated that it is extremely risky to operate a critical boiling water reactor at low pressures under ATWS conditions because of the potential for a rapid upward spiral of reactor power and reactor vessel pressure, caused by the positive coefficient of reactivity for void collapse and the very large void collapse with small pressure increases at low pressure (see Table 4.2). Indeed, reactor power and reactor vessel pressure spikes are predicted by BWR-LACP and reported in the results presented in Chap. 4. It is possible, however, that the final TVA emergency operating instructions provided for the use of the Browns Ferry operators will instruct the operators to maintain the reactor vessel pressure near its normal operating value under ATWS conditions.

2. The results presented in Chaps. 4 and 5 have been calculated under the assumption that the operator would not use the core spray system under ATWS conditions as long as other low-pressure injection systems are available. This is in accordance with the instructions provided in the Emergency Procedure Guidelines which are based on the fact that the effect upon core power and reactivity of a top-down spray into the individual fuel channels of a partially uncovered BWR core under ATWS conditions cannot be calculated by any existing code. The assumption that the effect of the core spray can be neglected is reasonable in the ATWS sequence because the low-pressure injection into the reactor vessel would be dominated by the condensate booster pumps, which have a much larger capacity than the core spray pumps and are capable of injecting at a higher reactor vessel pressure (see Table 3.2).

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*See Fig. 4.10 and the discussion in Sect. 4.1.3.

References for Chapter 6

6.1 R. M. Harrington et al., *Station Blackout at Browns Ferry Unit One — Accident Sequence Analyses*, NUREG/CR-2182, ORNL/NUREG/TM-455/V1 (November 1981), Chap. 4.

6.2 R. M. Harrington et al., *SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis*, NUREG/CR-2672, ORNL/TM-8119/V1 (November 1982), Appendix A.

6.3 R. M. Harrington et al., *Loss of DHR Sequences at Browns Ferry Unit One — Accident Sequence Analyses*, NUREG/CR-2973, ORNL/TM-8532 (May 1983), Appendix B.

6.4 R. M. Harrington and L. J. Ott, *The Effect of Small-Capacity, High-Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One*, NUREG/CR-3179, ORNL/TM-8635 (August 1983), Appendix A.

6.5 R. M. Harrington, *Comparison of Station Blackout Calculations on BWR-LACP, TVA Browns Ferry Simulator, and RELAP IV Mod 7*, letter report to NRC SASA Program Technical Monitor (August 1981).

6.6 W. C. Jouse and R. R. Schultz, *A RELAP5 Analysis of a Break in the Scram Discharge Volume at the Browns Ferry Unit One Plant*, EGG-NTAP-5993 (August 1982). Also published as Appendix G to Ref. 6.2.

6.7 L. Classen, personal communication, August 19, 1983.

6.8 A. J. Rogers and J. E. Torbeck, *Depressurization Performance of the General Electric Boiling Water Reactor High Pressure Coolant Injection System*, APED-5447 (Class I), May 1969.

6.9 P. Saha et al., "Review of RAMONA-3B Calculations for Browns Ferry ATWS Study under NRC SASA Program," presented at SASA program inter-lab information exchange meeting at Brookhaven National Laboratory, April 11, 1984.

6.10 Interoffice Correspondence, W. C. Jouse to R. C. Gottula, *BWR SASA TAF Calculation — WCJ-2-83*, EG&G Idaho, September 8, 1983.

6.11 L. Chu, *Power Suppression and Boron Remixing Mechanism for General Electric Boiling Water Reactor Emergency Procedure Guidelines*, NEDC-22166 (August 1983).

6.12 General Electric Company prepublication draft, *Emergency Procedure Guidelines, Revision 3, BWR 1 through 6* (December 1982).

7. IMPLICATIONS OF RESULTS

The purpose of this chapter is to provide a discussion of the state of readiness at the Browns Ferry Nuclear Plant to cope with an ATWS accident sequence initiated by an MSIV closure event. As studied here, this accident sequence involves a complete failure of all control rods to move inward from their normal positions for 100% power operation in response to the scram signal generated by MSIV closure or as a result of subsequent scram signals. Total failure of rod movement constitutes the most severe ATWS case, but is also the most improbable of the possible scram system failures. Thus the results of this study are intended to provide an upper bounding estimate of the consequences of these very unlikely events. The available control room instrumentation, the state of operator training, the written emergency procedures, and the overall system design at Browns Ferry Unit 1 are discussed in Sects. 7.1 through 7.3 from the standpoint of their adequacy in the actual event of an MSIV-closure initiated ATWS accident sequence. Information concerning the computer calculations employed in this study is summarized in Sect. 7.4.

7.1 Control Room Instruments

There is no specific alarm or other indication that would signal the initiation of an ATWS event to the plant operators. On the other hand, there is ample indication accompanied by both audio and visual alarms within the control room to signal when a scram condition has been satisfied and a scram signal has been generated. Since many abnormal transients result in multiple scram signals before they are brought under control, one control room display indicates all scram signals in effect by solidly backlighted transparent lettered panels except that the panel representing the first scram signal received is highlighted by flashing backlights. To determine the success of the scram, the operator, in accordance with established written procedures, must scan the instrument readouts concerning control rod position and reactor power. This information is prominently displayed.

All control room and other plant instrumentation that would be available after a normal reactor scram would also be available for operator use during an ATWS accident sequence even if a loss of offsite power were also involved. The primary system parameters displayed in the control room that would be of particular interest include reactor power from the average power range monitors (APRMs),* the reactor vessel

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*With a loss of offsite power, the RPS buses that power the APRMs would be lost until the RPS motor-generator sets were locally restarted on the diesel-generators. The SRMs and IRMs are battery-powered, however, and the IRMs can indicate reactor powers as high as 40%. The SRMs and IRMs are inserted into the core by operator action following a scram.

water level from the two indicating systems that have ranges extending over the portion of the reactor vessel near the top of the core, the reactor vessel pressure, and the rates of injection into the reactor vessel from the feedwater system, the ECCS systems, and the CRD hydraulic system. The control room indication ranges for each of these parameters is provided in Table 7.1.

As discussed in Chaps. 3 and 4, the reactor thermal power can significantly exceed normal operating levels during power excursions initiated by core flooding or by small pressure increases during critical operation at low pressure under ATWS conditions. As shown in Table 7.1, the upper limit of control room indication is 125%. (It should also be noted that the APRMs measure the power level suggested by the level of nuclear activity, rather than the core thermal power, which lags the neutron flux.) Thus, the range of available power indication would not permit the control room operator to see the peaks of the power spikes. However, since the power spikes are of brief duration and the operator would be apprised of an abnormally high power level, this instrument limitation is not expected to have any effect on the sequence of events.

The two available reactor vessel water level indication systems that would permit the control room operator to monitor water levels near the top of the core were not designed for service under ATWS conditions and therefore are not ideal for this purpose. As indicated in Table 7.1, the Emergency Systems instrument is calibrated for normal operating pressure but the lower end of its indicating range is 13 in. (0.33 m) above the top of the core. On the other hand, the indicating range of the Post Accident Flooding instrument extends down to 1/3 core height, but this instrument is calibrated for atmospheric pressure and, because its lower tap is into the surface of a jet pump discharge cone, it would not be expected to provide accurate reactor vessel water level indication unless the reactor vessel were depressurized and the flow through the jet pumps was zero or very low.

As discussed in Sect. 4.1.2, the BWR Owners Group Emergency Procedures Guidelines direct the operator to take action to lower the reactor vessel water level to the top of the core when confronted with an ATWS situation. The purpose of this action is to reduce reactor power but, as indicated in Fig. 4.2, the major effect is achieved when the water level is lowered below the feedwater spargers. Therefore, the reactor vessel water level could be maintained significantly above the top of the core while still achieving the main purpose of the lowering. Specifically, the operator could maintain the water level near the bottom of the Emergency Systems indication range, thereby maintaining reliable indication while sacrificing almost nothing in power reduction.

Since all thirteen of the reactor vessel relief valves would be open if the vessel pressure exceeded 1125 psig (7.86 MPa), and because the increased voids attendant to high reactor vessel power would insert a large amount of negative reactivity, thereby turning power and preventing further pressure increase, and because it would not seem possible to alert an operator more than by an indicated vessel pressure of 1500 psig (10.40 MPa), the upper limit of indicated reactor vessel pressure given in Table 7.1 seems adequate.

The makeup flows from the feedwater and from the ECCS systems that would be injected into the reactor vessel under ATWS conditions lie within the ranges of available control room instrumentation as documented in Table 7.1. However, it should be recognized that when the RHR system is aligned for pressure suppression pool cooling and the LPCI mode injection valves are opened for simultaneous reactor vessel injection, then the rate of reactor vessel injection can only be ascertained by subtracting the pressure suppression pool cooling flow from the total RHR system flow.

The CRD hydraulic system injects 60 gpm (0.004 m^3/s) of cooling water flow past the 185 control rod drive mechanism assemblies during normal reactor operation. In an ATWS situation, if the failure-to-scram prevents the opening of the scram inlet valves, this flow would not be increased. Otherwise, the opening of the scram inlet valves permits a large flow to bypass the CRDHS flow control station and the actual flow into the reactor vessel would be in excess of the upper limit of the available indication. The fact that the CRD hydraulic system injects much more water than is recognized by the operator under accident situations has been discussed in previous ORNL SASA program reports and is not a new result of this study.

It was also reported in previous studies that the actual position of the SRVs is not displayed in the control room. When the control room operator acts to manually open an SRV, a control panel light informs him that the solenoid operator for that valve is energized, nothing more. As discussed in Sect. 4.1.3, attempted reactor vessel pressure control by manual SRV actuations would be very confusing to the operator. Acoustic monitors have been installed to indicate the presence of flow through the SRVs, but this indication is consigned to read-out in secondary panels, outside of the control room. It is recommended that consideration be given to moving this indication so that the control room operator would be able to ascertain how many and which relief valves are actually open at any time during a BWR accident sequence.

The primary containment parameters measured by the available instruments and displayed in the control room include the temperature of the drywell atmosphere, the temperature and level of the water in the pressure suppression pool, the temperature of the wetwell atmosphere, and the overall pressure in the primary containment. The range of indication and the associated alarms for each of these parameters are provided in Table 7.2.

As discussed in Chap. 3, the best-estimate failure pressure for the Browns Ferry MK I containment is 132 psia (0.91 MPa). Therefore, if the operators failed to take corrective action in an ATWS accident sequence so that the failure pressure was approached, the drywell and wetwell pressure instruments would be off-scale high. The pressure suppression pool water level instruments would also be off-scale high as the pool continued to swell in response to heating and the absorption of the SRV discharge. On the other hand, the existing drywell and pressure suppression pool temperature indication would remain onscale throughout the period of the accident sequence before containment failure.

Since the wetwell atmosphere would be virtually 100% steam as the primary containment pressure approached failure levels, the pressure could be inferred during the period after the pressure instruments became off-scale high from the indicated pressure suppression pool temperature and the saturation tables.

7.2 System Design

A design consideration first identified in the SASA study of Station Blackout at Browns Ferry[7.1] also has direct application to the ATWS accident sequence. There is provision for an automatic shift of the high pressure coolant injection (HPCI) booster pump suction from the condensate storage tank to the pressure suppression pool on high sensed suppression pool level. The change in HPCI pump suction lineup is accompanied by the opening of two DC-motor-operated valves in the line from the suppression chamber header (Fig. 7.1) followed by the closing of the DC-motor-operated valve in the suction line from the condensate storage tank. (A check valve in the line from the suppression pool prevents backflow from the condensate storage tank into the pool during the changeover.) Once accomplished, the shift is irreversible; the operator cannot switch the pump suction back to the condensate storage tank. Because the HPCI turbine lubricating and control oil is cooled by the water being pumped* and the pressure suppression pool temperature is elevated in many accident sequences, this automatic shift can cause failure of the HPCI system by overheating of the lubricating oil.

The automatic shift of the HPCI booster pump suction will occur when the pressure suppression pool level increases to an indicated level of +7 in. Since the normal pool level is maintained between -2 and -6 in.,[7.2] this increase implies the addition of between 68,000 and 98,000 gals (257 and 371 m^3) of water to the pool.† For the MSIV-closure initiated ATWS accident sequence, this would occur about 10 min after the inception of the accident, when the suppression pool temperature had increased to about 160°F (344 K). The pool temperature would continue to increase rapidly after the shift. Since the HPCI system lube oil cooler is designed for a maximum inlet water temperature of 140°F (333 K), the oil would be overheated, leading to probable system failure within a few minutes following the shift.‡

The water pumped from the condensate storage tank into the reactor vessel, converted to steam within the reactor vessel, transferred from the reactor vessel as steam via the SRVs to the pressure suppression

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*As shown on Fig. 7.1, a portion of the booster pump discharge is diverted through the gland seal condenser and the lube oil cooler and returned to the pump suction.

†Some of the level increase would be caused by the increase in specific volume of the water mass as it is heated.

‡For this study, it has been assumed that system failure would occur when the suppression pool temperature reached 190°F (361 K). The oil temperature at this time would, of course, be considerably higher.

pool, and condensed within the pressure suppression pool would increase the pool volume to the equivalent of an indicated level of +7 in. long before the condensate storage tank was emptied. Since the condensate storage tank volume is maintained at about 362,000 gals. (1370 m^3) during normal operation,[7.2] an ample amount of relatively cool water would remain available in the condensate storage tank at the time the HPCI booster pump suction was shifted.

The threat to the HPCI system identified here is not unique to ATWS accident sequences. It also exists in all other BWR accident sequences such as Station Blackout and Loss of Decay Heat Removal* in which the pressure suppression pool would be overheated. High pressure suppression pool temperature would be caused by the pool heating attendant to the condensation of steam in the pool, which would also be the reason for the increased pressure suppression pool level that would cause the self-destructive shift of the HPCI booster pump suction to the overheated pool.

It should be noted that separate provision is made for an automatic shift of the HPCI booster pump suction if the normal condensate storage tank water source becomes exhausted. Thus it appears that the provision for the automatic high suppression pool water level shift must have been straight-forwardly based on a concern for the effect of high water level in the wetwell although, since there is a clearance of some 16 ft (4.88 m) from the pool surface to the top of the torus under normal operating conditions, it seems incongruous that an increase in indicated level of 13 in. (maximum) should require the pump suction shift from the standpoint of preserving torus structural integrity. Also, the wetwell airspace-to-drywell vacuum breakers would continue to function at pressure suppression pool water levels much above the setpoint for pump suction shift.

All efforts to determine the basis for the HPCI system booster pump suction shift upon high sensed pressure suppression pool level have been unsuccessful. There is no corresponding shift for the reactor core isolation cooling (RCIC) system, whose operation can also lead to higher-than-normal water levels in the torus. A survey of plant drawings does not reveal why an indicated water level of +7 in. in the wetwell should be of concern. Discussions with TVA engineering staff and GE vendor personnel do not produce the reason.†

It is recommended that action be taken to remove the threat of HPCI system loss caused by automatic actuation of safety system logic and the resultant loss of lubricating oil cooling during severe accident sequences. This might be done either (1) by replacing the existing oil by an oil qualified for high temperatures,[7.3] (2) by revising the existing logic so that the operator, recognizing the automatic suction shift and

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*This is the TW sequence in WASH-1400 parlance.

†The best guess seems to be that the HPCI booster pump suction shift was intended to ensure that enough volume would remain in the torus airspace to permit collection of the non-condensible gases from the drywell in the event of a large-break LOCA.

realizing that the pressure suppression pool is overheated, could restore the pump suction to the condensate storage tank, or (3) by removing altogether the automatic pump suction shift upon high sensed pressure suppression pool level. Since this deficiency in plant protective logic has come up again and again in the BWR SASA studies, the authors of this report strongly recommend that some kind of preventative action be taken.

A second consideration in regard to plant design involves the inability of the control room operators to know which SRVs are actually open at any particular time during an accident sequence. If the control room operators act to manually open an SRV, they are rewarded with a control panel light indicating that the solenoid operator for that valve has been energized. It is emphasized that the actual valve position is not indicated in the control room. For example, should the operator act to manually open a valve that, by happenstance, was already open because the setpoint for its automatic actuation had been exceeded, he would be rewarded with a light, but nothing would change. Acoustic monitors that are effective in detecting actual discharge through the SRVs have been installed, but the readout is on the back-panels, out-of-sight of the control room operators.

7.3 Operator Preparedness

The TVA Browns Ferry control room simulator does have the capability to model the portion of an ATWS severe accident sequence that would occur before drywell failure or fuel damage. For the purpose of determining the general reliability of the sequence of events as predicted by the simulator computer, the no-operator-action MSIV-closure initiated ATWS accident sequence of events has been calculated using both BWR-LACP and the simulator.

The simulator results were taken from special equipment that recorded the control room instrument readings as they would be seen by the operator. Thus, whenever a control room instrument was pegged high or low during the accident sequence, the recorded data remained at the upper or lower end of the range of the instrument until the magnitude of the parameter being measured came back into the measurement range. Also, since the simulator does not model failure of the HPCI system on high lube oil temperature, it was necessary for the simulator control console operator to impose an artificial failure. Thus, for the purpose of facilitating a comparison of the simulator results with the BWR-LACP results, a constant has been added to all simulator event timing so that the time of HPCI failure matches that calculated by BWR-LACP. It should also be noted that the accident is initiated at time 4.5 min on the simulator scale and therefore normal power operation is represented by the plotted simulator results before this time.

The simulator results for core thermal power are shown on Fig. 7.2 and may be compared with the BWR-LACP results shown on Fig. 3.1. The simulator computer software does not include models to recognize the effect of lowered reactor vessel water level in reducing core power. Thus the simulator results do not indicate a reduction in reactor power when the HPCI fails (at time 16 min) nor is the power reduced to decay heat levels when the core is completely uncovered during the periods between the power peaks. The magnitude of the plotted simulator power peaks is limited to 1.25 because, as indicated on Table 7.1, the indicating range of the APRMs is 0-125%.

The simulator results for reactor vessel downcomer water level are shown in Fig. 7.3; these results can be compared with the BWR-LACP results shown on Fig. 3.2. It is interesting to note that the water level during the period between recirculation pump trip and HPCI system failure is predicted to be about 440 inches (11.2 m) by the simulator and about 475 inches (12.1 m) by BWR-LACP. Since the water levels are relatively stable during this period, all of the injected water is being boiled to steam. Comparison of Figs. 7.2 and 3.1 shows that the calculated power levels are about the same during this period.

After HPCI system failure, the water level falls as shown on Fig. 7.3, leading to initiation of the large-capacity, low-pressure, ECCS systems and the ADS timer, followed two minutes later by automatic opening of the six SRVs controlled by the ADS system. When the reactor vessel pressure has decreased to below the shutoff heat of the low-pressure ECCS pumps, vessel injection floods the core, causing a power excursion. The simulator (erroneously) models the ADS valves as closing each time the reactor vessel water level is restored. The reactor vessel water level shown on Fig. 7.3 does not go below 260 in. (6.6 m) because, as listed on Table 7.1, this is the bottom of the instrument indicating range.

The simulator results for the rate of injected flow are shown on Fig. 7.4 and may be compared with the BWR-LACP results shown on Fig. 3.3. Feedwater flow is lost after MSIV closure and only the CRD hydraulic system provides injection until HPCI and RCIC system injection is automatically initiated upon a low reactor vessel water level signal. The simulator models for the rates of HPCI and RCIC system injection are more sophisticated than those in BWR-LACP, taking into account the turbine governor control systems and the effect of varying reactor vessel pressure.

After failure of the HPCI system, reactor vessel injection is supplied only by the high-pressure RCIC and CRD hydraulic systems except for the brief periods when the reactor vessel is depressurized sufficiently to permit injection by the low-pressure systems, including the condensate booster pumps.

The simulator results for the reactor vessel pressure are shown on Fig. 7.5; these can be compared with the BWR-LACP results shown on Fig. 3.4. The reactor vessel pressure increases briefly after MSIV closure, but recirculation pump trip reduces core power and subsequently, reactor vessel pressure remains at the relief valve setpoints as some relief valves remain open and other relief valves cycle. Large decreases in reactor vessel pressure occur when the ADS system is actuated upon decreasing reactor vessel water level. These pressure drops permit the low-pressure injection systems to flood the core, thereby producing a power excursion and also resetting the ADS logic and closing the ADS valves as the water level rises.* Reactor vessel pressure is rapidly restored and the pressure is controlled by automatic actuation of the SRVs at their relief setpoints during the periods between ADS actuations.

The simulator results for the temperature of the pressure suppression pool are shown on Fig. 7.6. Comparison with Fig. 3.5 shows that the simulator modeling produces a much higher pool heatup rate. As listed in Table 7.2, the upper limit for indication of suppression pool temperature is 400°F (478 K) and this is the reason for the plateau shown on Fig. 7.6. The simulator greatly overpredicts suppression pool temperature.

Figure 7.7 shows the drywell pressure history during the accident sequence as predicted by the simulator. Comparison with Fig. 3.6 reveals that the simulator predicts much lower containment pressures. Taken with the information in the previous paragraph, it must be concluded that the simulator does not model evaporative steaming from the surface of the heated pressure suppression pool.

What must be judged here is the efficacy of the TVA control room simulator as an operator training device, capable of instilling the knowledge needed by the operators to cope with an actual ATWS event. It was not designed for this exercise. The simulator does not model the effect of low reactor vessel water level on reactor power. Also, the simulator does not model evaporative steaming from the surface of the pressure suppression pool. These modeling defects, from the standpoint of ATWS application, directly cause the predicted pool water temperature to be much too high and the predicted primary containment pressure to be much too low. Also, other simulator models do not reflect the difference between the downcomer water levels that would be predicted by the Emergency Systems instruments and the Post Accident Flooding instruments so the operator under training is unrealistically exposed to a situation in which all reactor vessel water level instruments indicate the same water level under accident conditions. The simulator predicts erroneously that the open ADS valves would shut each time the reactor vessel water level is restored; this has a minor effect on the magnitude (too high) and the duration (too low) of power spikes. Nevertheless, the general sequence of events predicted by the simulator is sufficiently accurate so that the simulator can be useful for operator training to deal with ATWS events. Obviously, improvement of the simulator models is desirable.

The concept of symptom-oriented procedures for operator action in response to emergency conditions has been implemented by the BWR Owners Group Emergency Procedures Guidelines. It is a conclusion of the authors of this study that the ATWS accident sequence is easily identifiable by the operators and should have a separate procedure. The general concept of symptom-oriented procedures is workable because almost all accident sequences demand the same operator actions, i.e., keep the core covered. Yet in the ATWS accident sequence, the operator

*This is an error in the simulator logic. Once open, the ADS valves would not close upon increasing reactor vessel water level.

must reduce the reactor vessel water level to the top of the core. In all other accident sequences, the main effort should be to increase or maintain injection; in the ATWS accident sequence the operator must reduce the injection flows and control the downcomer water level near the top of the core.

In other accident sequences, the reactor is scrammed, core power is at decay heat levels, and the operator can easily control reactor vessel pressure by manipulation of one SRV. For the ATWS accident sequence, the operator attempting to control pressure by manual SRV actuation would be confused by the fact that reactor vessel pressure would be unaffected by his efforts until he had acted to manually open several SRVs, but then would suddenly decrease when he opened one more. For these reasons, the ATWS accident sequence seems to be the odd-man-out; that is, procedures for its mitigation are unique and therefore cannot be simply fitted into the general envelope of procedures for mitigation of other BWR accident sequences. It should also be noted that delegation of the ATWS corrective actions to a separate procedure would greatly simplify the remaining symptom-oriented guidelines.

7.4 Summary of Computer Calculations used in this Study

It is the purpose of this section to briefly summarize the calculational methodology used in this study.

The results of General Electric company calculations were used for the first 50 s of each accident sequence analysis (see Chap. 2).

The BWR-LACP code was initiated at the 50 s point for each analysis. Results of the BWR-LACP calculations are presented in Chaps. 3, 4, and 5.

Identical sequence calculations were performed using BWR-LACP and RELAP5 through a cooperative effort between INEL and ORNL. The comparison is discussed in Appendix A. The results are similar except that the timing of the events predicted by RELAP has been expanded because the calculated power in RELAP5 is lower. Since the BWR-LACP calculated power is within the estimated error band (±10% power) of the RELAP power, no attempt has been made to adjust the BWR-LACP power calculations.*

Identical ATWS scenarios were calculated using BWR-LACP and the Browns Ferry simulator computer through a cooperative arrangement between the ORNL SASA program and the TVA. The results of the comparison are discussed in Sect. 7.3. None of the simulator results has been used for any purpose other than for the discussion in Sect. 7.3.

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*Subsequent to these calculations, an error was found in RELAP5 that tended to make the calculated power too low. This error has been corrected but the decision was made not to delay the issuance of this report to permit a new comparison of results.

The Browns Ferry simulator is not and was never intended to be an engineering analysis tool. Nevertheless, information obtained during three visits to personally witness the simulated control room response to various ATWS accident sequences has convinced the authors of this report that the ATWS simulation is reasonably accurate. However, the realism could be significantly improved by correction of the known deficiencies in the simulator models (discussed in Sect. 7.3).

References for Chapter 7

7.1 S. A. Hodge et al., *Station Blackout at Browns Ferry Unit One — Accident Sequence Analysis*, NUREG/CR-2182, ORNL/TM-455/V1 (November 1981).

7.2 Browns Ferry System Operating Instruction No. 64, *Primary Containment Unit I, II, or III*.

7.3 This simple and non-controversial solution was first suggested by Ed Kozinsky of the General Physics Corporation.

Table 7.1. Control room indication ranges for primary system parameters important to analysis and control of an ATWS accident sequence

Parameter	Indication range
Percent of rated thermal power (3293 MW_t)	0—125%
Reactor Vessel Water Level	
Emergency systems, inches above vessel zero[a]	373—588
Post accident flooding, inches above vessel zero[b]	260—560
Reactor Vessel Pressure, psig	0—1500
Feedwater Flow	
Total feedwater flow (recorder), lb/hr	0—16 x 10^6
Feed flow line a, lb/hr	0—8 x 10^6
Feed flow line b, lb/hr	0—8 x 10^6
ECCS Injection Flow	
HPCI system flow gpm	0—6000
RCIC system flow, gpm	0—700
Core spray flow, gpm[c]	0—10000
RHR system total flow (recorder)[c], gpm	0—40000
RHR containment spray/cooling flow[c], gpm	0—20000
CRD Hydraulic System Flow, gpm	0—100

[a]Calibrated for normal operating pressure.

[b]Calibrated for atmospheric pressure.

[c]The system has two independent loops. There is one indicator for each loop.

Table 7.2. Control room indications and alarms of primary containment variables important to analysis and control of an ATWS accident sequence

Variable	Range or alarm setpoint
Drywell pressure	
Indication, psia	0–80
Alarms, psia	16.30
	16.35
	16.45
	16.70
Drywell atmosphere temperature	
Indication, °F	0–400
Alarms, °F	145
Wetwell Pressure	
Indication, psia	0–80
Alarms, psia	16.7
Pressure suppression pool temperature	
Indication, °F	0–400
Alarm, °F	95
Pressure suppression pool level[a]	
Indication, in.	-25 to + 25
High level alarm, in.	+6
Low level alarm, in.	-6

[a]Instrument zero is 15.2 feet above the bottom of the wetwell torus. Zero water level means that the torus is approximately half-filled with water.

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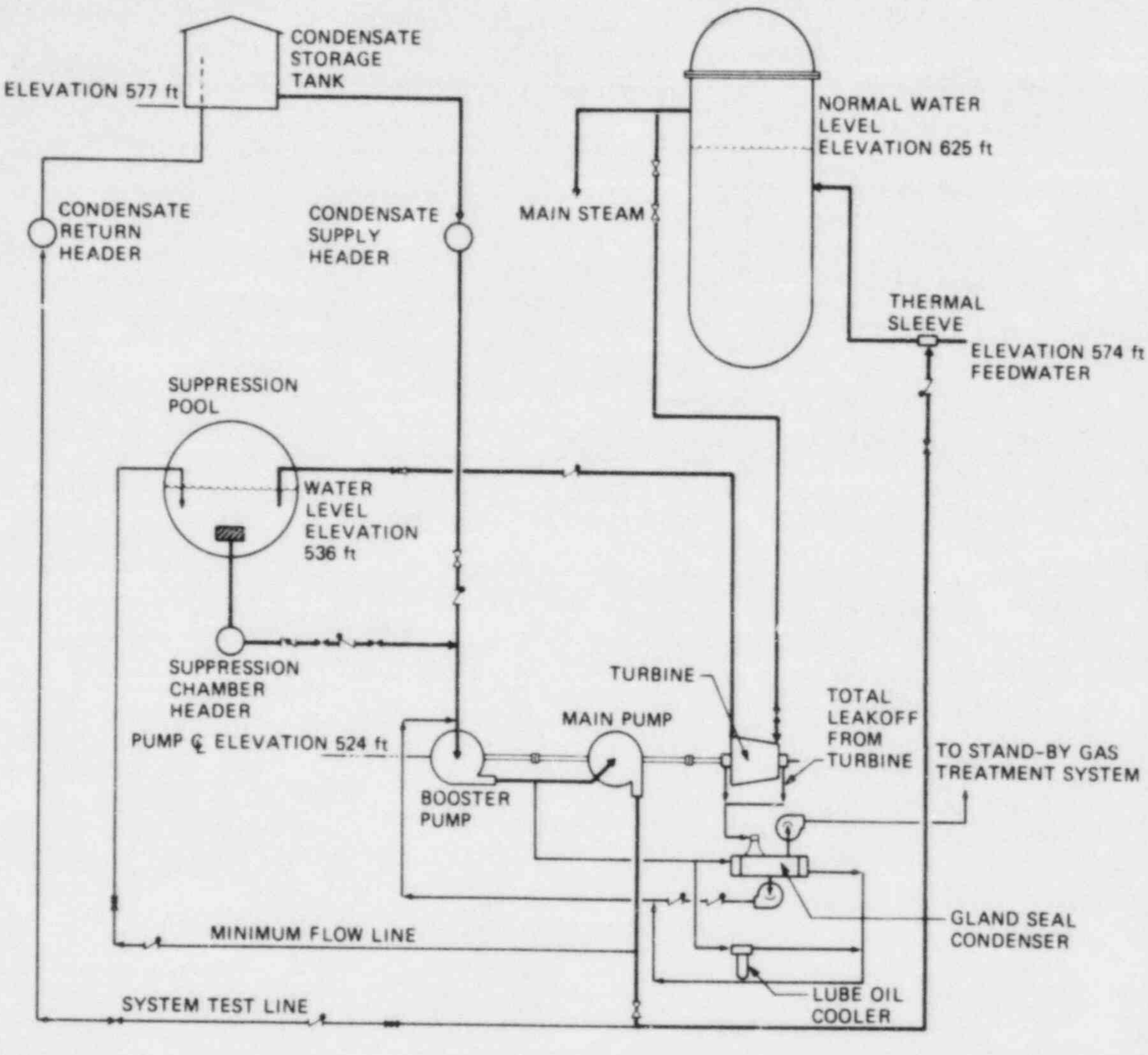

Fig. 7.1. Schematic drawing of the high pressure coolant injection (HPCI) system.

ORNL-DWG 84-7800

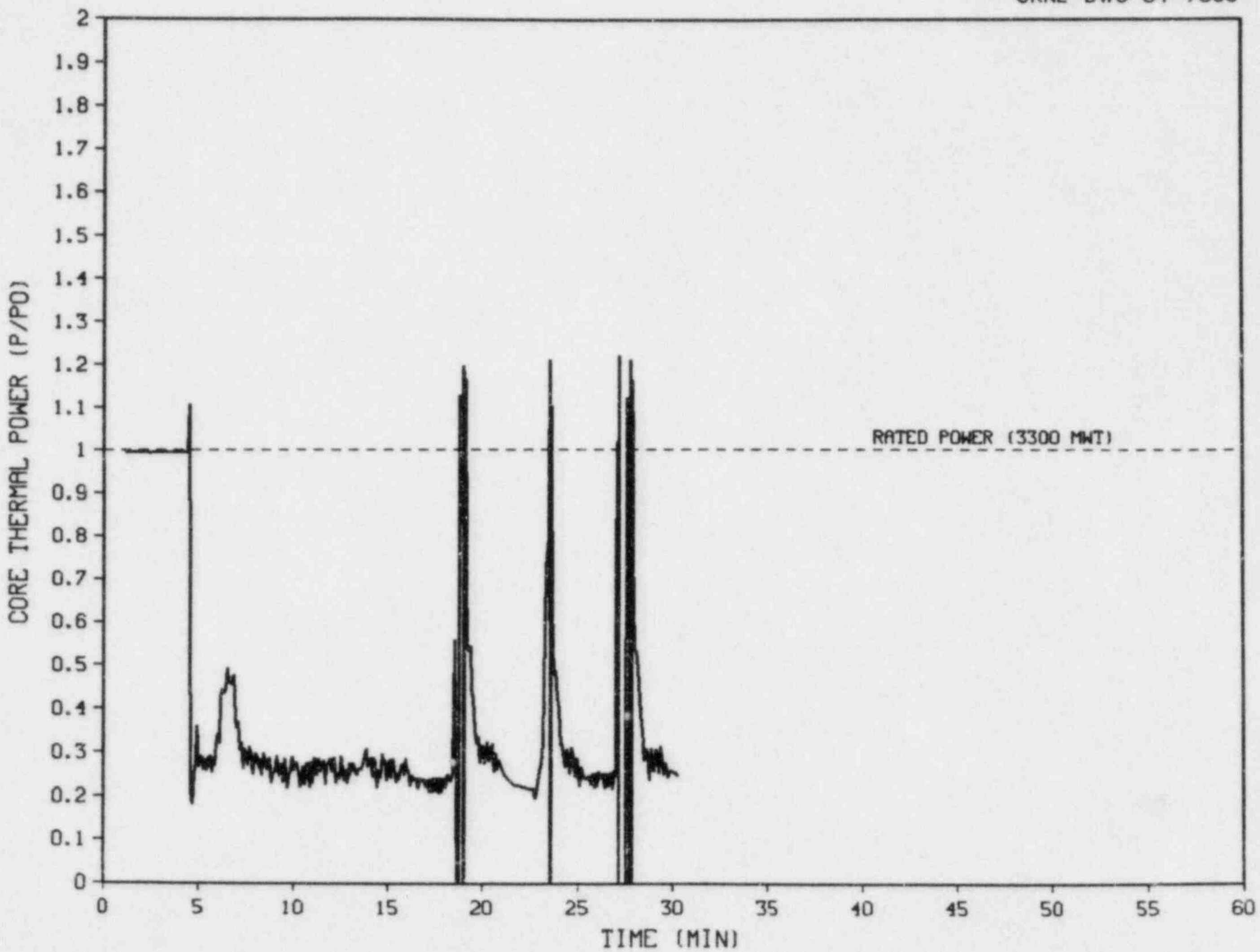

Fig. 7.2. TVA Browns Ferry control room simulator results for core thermal power during the no-operator-action MSIV closure-initiated ATWS accident sequence.

ORNL-DWG 84-7799

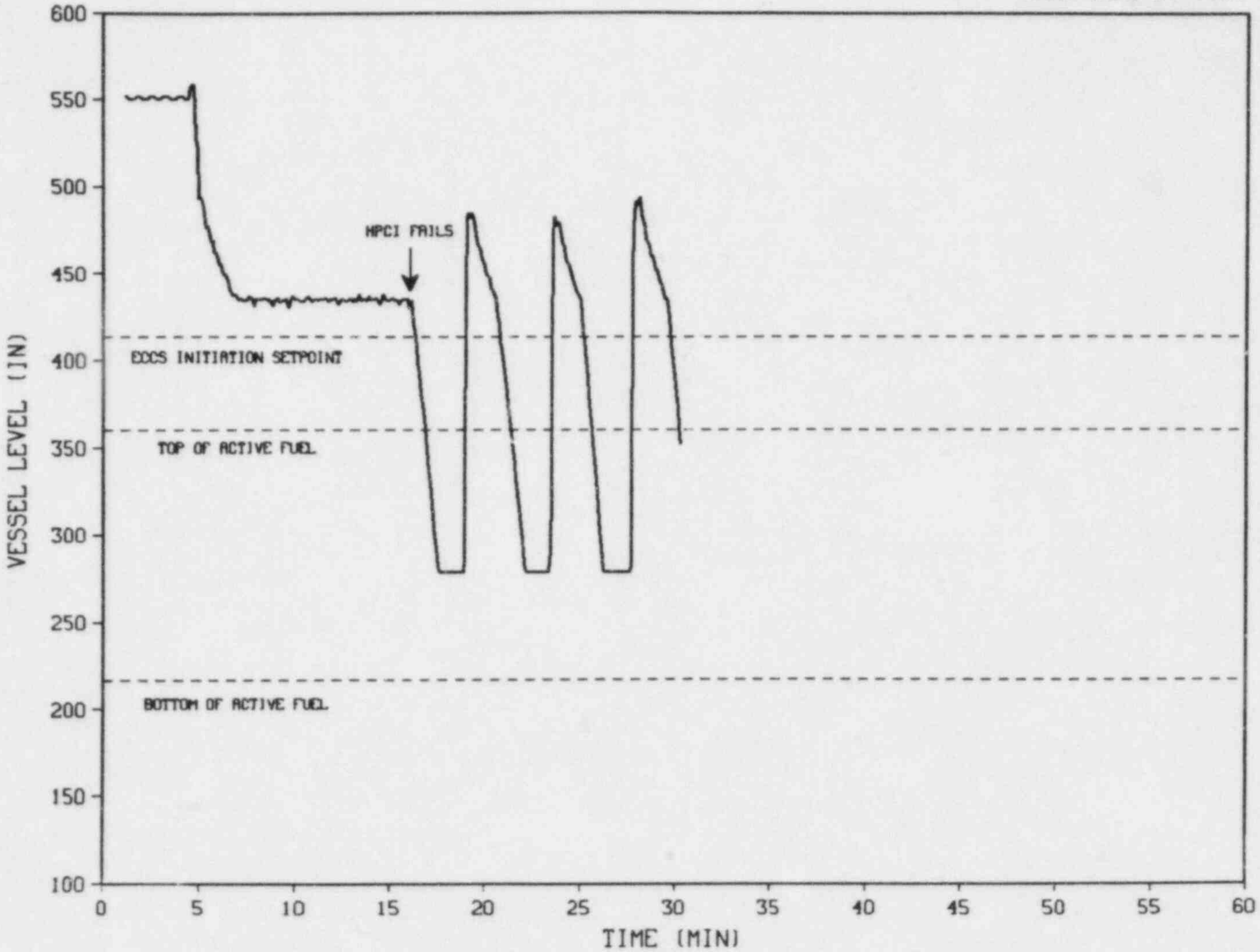

Fig. 7.3. TVA Browns Ferry control room simulator results for reactor vessel downcomer water level during the no-operator-action MSIV closure-initiated ATWS accident sequence.

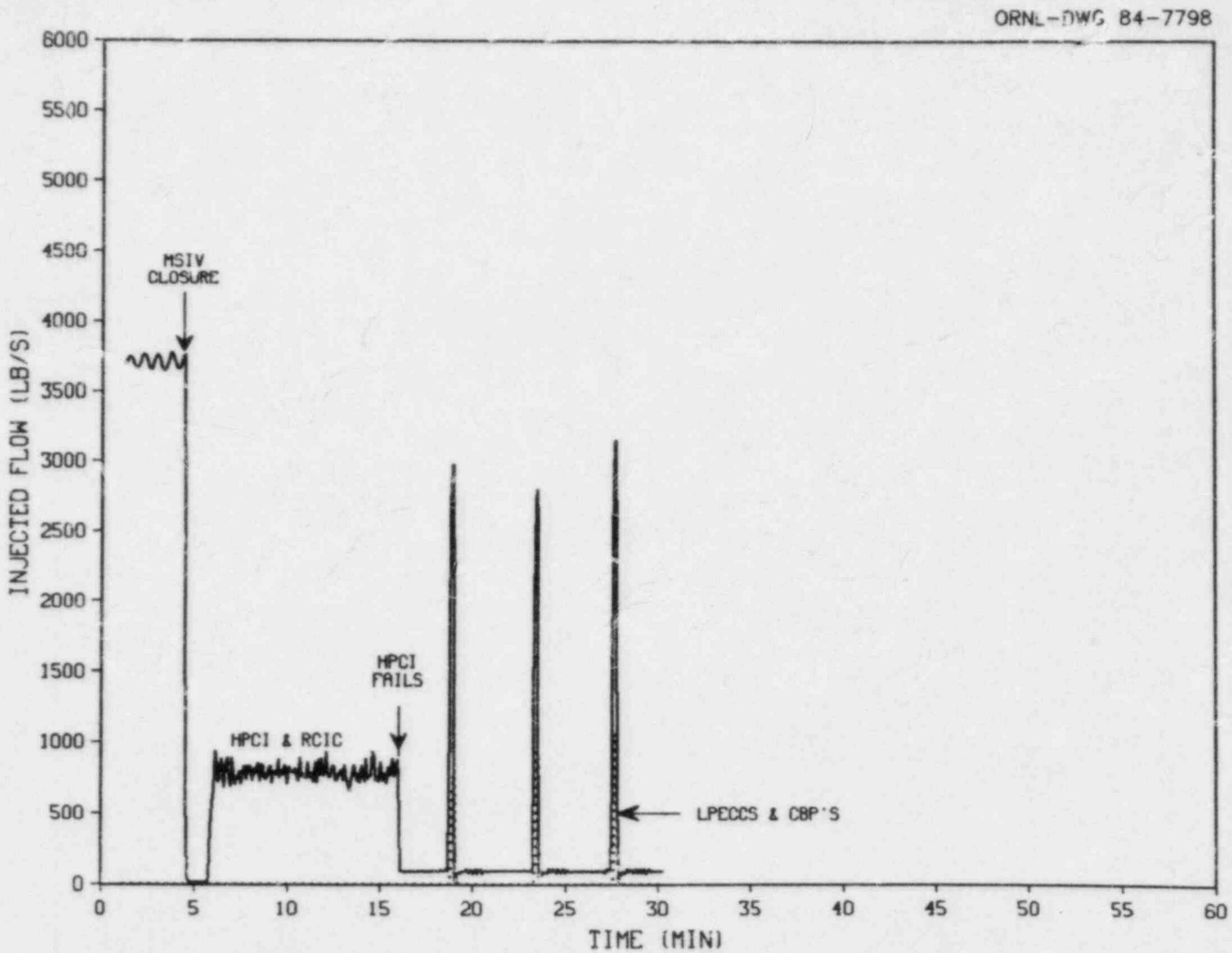

Fig. 7.4. TVA Browns Ferry control room simulator results for the rate of injected flow during the no-operator-action MSIV closure-initiated ATWS accident sequence.

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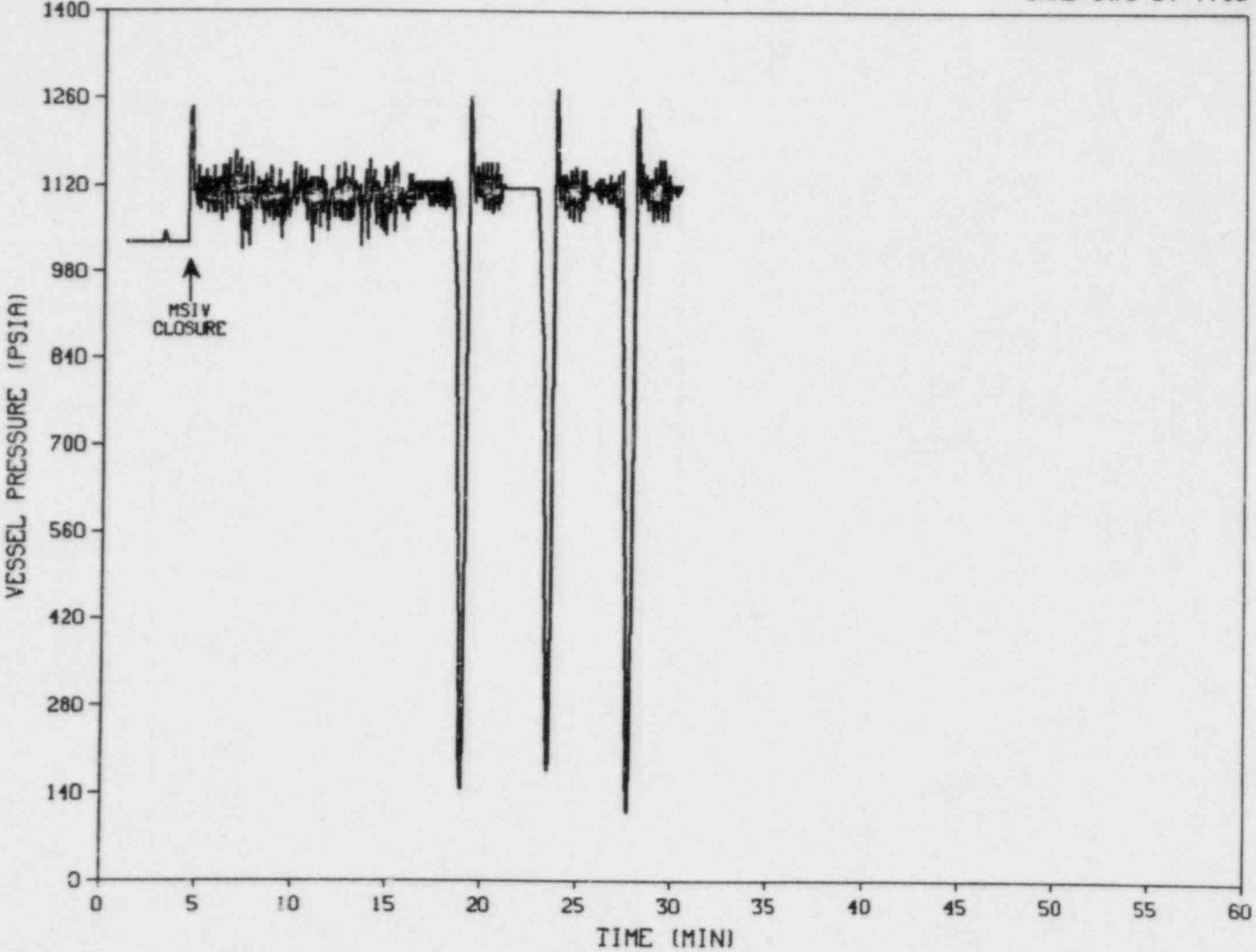

Fig. 7.5. TVA Browns Ferry control room simulator results for the reactor vessel pressure during the no-operator-action MSIV closure-initiated ATWS accident sequence.

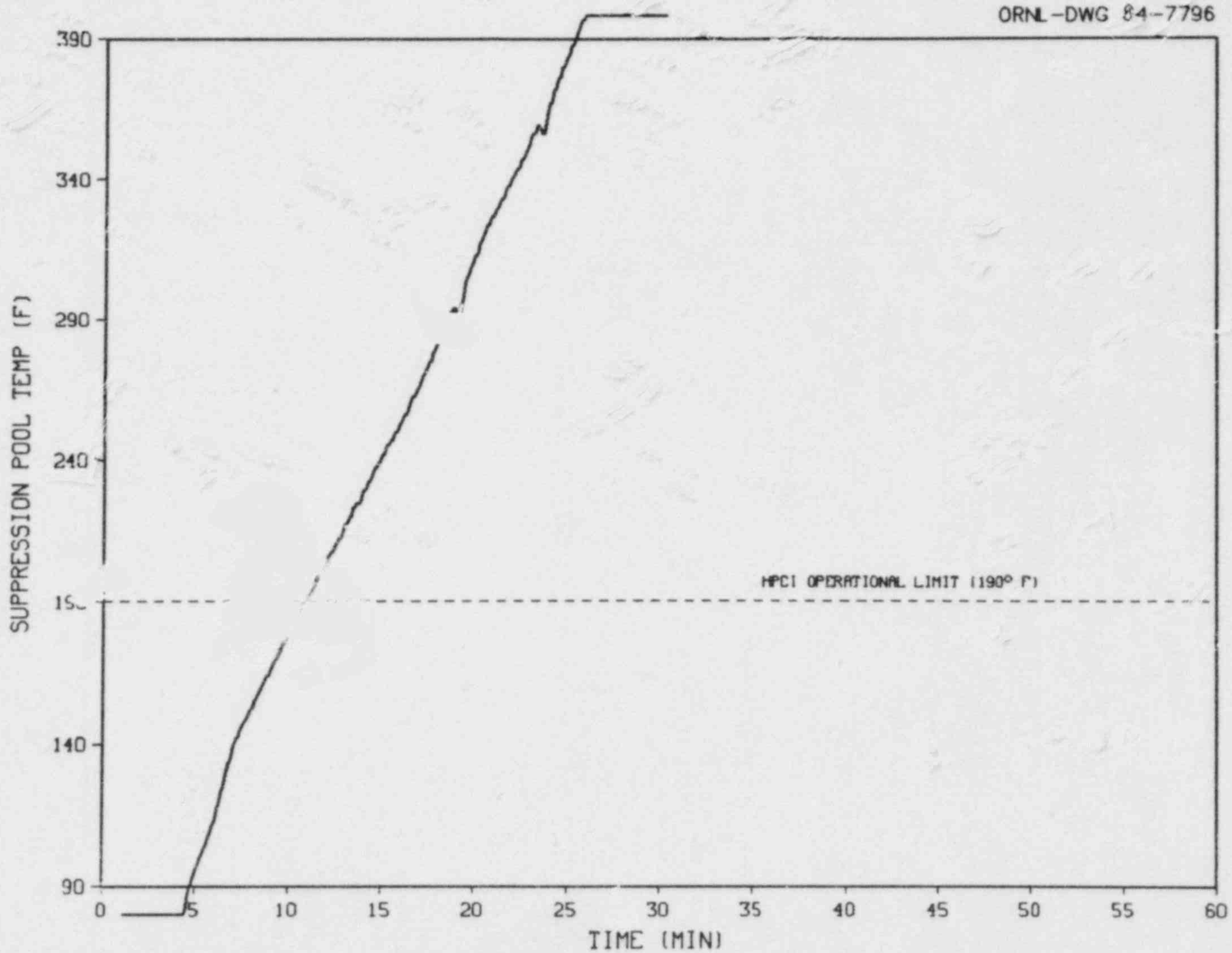

Fig. 7.6. TVA Browns Ferry control room simulator results for the pressure suppression pool temperature during the no-operator-action MSIV closure-initiated ATWS accident sequence.

ORNL-DWG 84-7797

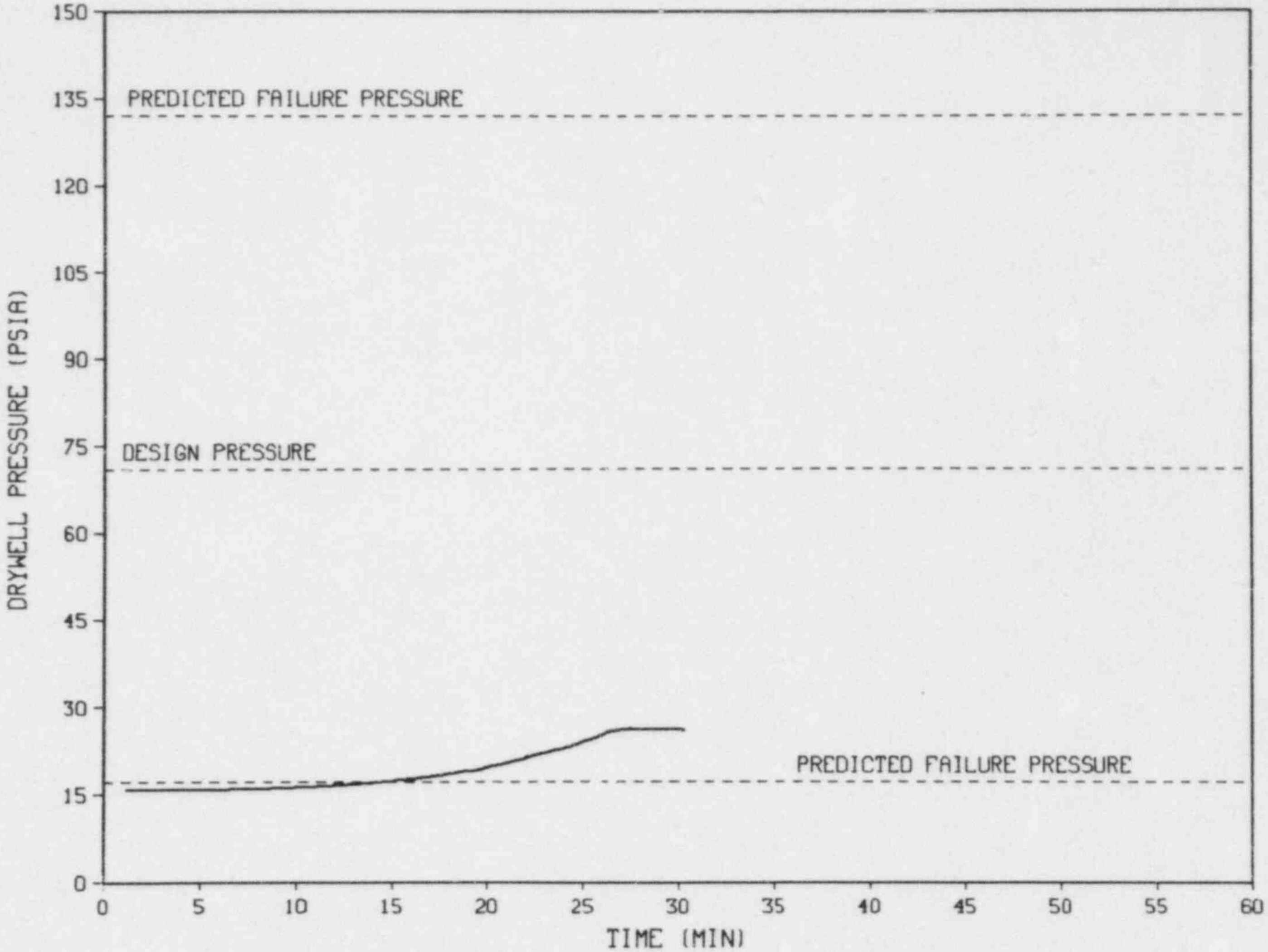

Fig. 7.7. TVA Browns Ferry control room simulator results for the primary containment pressure during the no-operator-action MSIV closure-initiated ATWS accident sequence.

APPENDIX A: MODIFICATIONS TO THE BWR-LACP CODE FOR THIS STUDY

This appendix provides details of modifications to the BWR-LACP code made specifically for this study. Some of this new coding is a straightforward translation of the expected behavior of system components, such as SRVs and injection systems, into mathematical rules. The most important of the modifications — the new routines that calculate core voiding and fission power — are simplified solutions of a set of very complex neutronic and thermohydraulic problems.

The models used in BWR-LACP to calculate core voiding and fission power are considerably simplified in comparison to the detailed, first principles models used in codes such as RAMONA, RELAP, or TRAC. To assess what differences might exist between BWR-LACP and the more complex codes, a comparison is made in this appendix of the RELAP5 results (provided by the SASA team at INEL) and BWR-LACP results for the same test transient.

The results, of course, show some differences between the two codes, but the qualitative similarities prove that BWR-LACP is an adequate scoping tool even for a complex accident such as ATWS. System variables show the same trends and, most importantly, both codes predict a severe power/pressure spike occurring at the end of the reactor vessel depressurization. This confirms one of the major recommendations of this report: that the reactor vessel not be depressurized during an ATWS accident. In general, it is the desire of the authors that the major recommendations of this report be confirmed by investigations using the more sophisticated codes, i.e., TRAC, RELAP, or RAMONA, as applicable.

A.1 Calculation of Reactor Power

In an ATWS accident the reactor power is the sum of decay heat power plus fission power. The fission power is a transient function of the reactivity of the core; decay heat power is a function of the elapsed time since reactor shutdown. Whenever the negative reactivity insertion brings the core subcritical, the total power in BWR-LACP is set equal to the decay heat power as soon as the calculated fission power is negligible.

The decay heat function is calculated in accordance with the ANS 5.1-1979 standard decay heat curve. This calculation for decay heat is exactly correct only for the case of a full scram; however, it is a reasonable approximation for most of the cases examined in Chaps. 4 and 5 because reactor power is below 10% after about 7 min.

$$P_{dk} = f(t, P_o)$$

where,

P_{dk} = decay heat power (fraction of full power)
t = elapsed time since the scram or accident initiation
P_o = initial reactor power (=100% for all cases).

The prompt-jump approximation to the 6 delay group point kinetics equations is solved for fission power. These equations can be found in any nuclear analysis textbook and are not discussed here. The code input for delayed neutron properties is listed on Table A.1. Four sources of reactivity are considered: fuel temperature change (via doppler coefficient), coolant void fraction change (via void coefficient), control rod insertion, and boration of reactor coolant. Each of these sources of reactivity is discussed in the following subsections.

A.1.1 Fuel temperature and reactivity feedback

A single average fuel temperature is calculated by solving the following equation

$$\frac{dT_f}{dt} = 108\ P_t - .1624\ (T_f - T_{sat})$$

where,

T_f = volumetric average fuel temperature (F)
P_t = fission plus decay heat power (fraction of full power)
T_{sat} = saturation temperature (F), of the coolant in the core.

The numerical coefficients in the above equation take into account the fuel heat capacity and the average fuel-to-coolant heat transfer coefficient.* The coefficient of P_t is the thermal equivalent of full power, divided by the fuel heat capacity. The coefficient of $(T_f - T_{sat})$ is the effective heat transfer coefficient between the volumetric average fuel temperature and the coolant average temperature, divided by the fuel heat capacity.

The net reactivity due to a change in average fuel temperature is a function of the doppler coefficient which is corrected for change in coolant void fraction:

$$\Delta\rho_d = (T_f - 1210)\ \alpha\ (D_o + D_1 V)$$

where

$\Delta\rho_d$ = the change in total doppler reactivity ($\Delta k/k$)
T_f = average fuel temperature (F)

1210 = average fuel temperature (F) at full power
α = doppler coefficient
= $-1.58(10)^{-5}$ ($\Delta k/k/F$)

D_o = doppler correction factor with 0% core average void
= .83

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*See Section 3 of Browns Ferry FSAR for fuel weights, steady state volumetric average temperatures, and average heat flux. A value of 0.08 Btu/lb F was used for UO_2 specific heat (Nuclear Engineering Handbook, H. Etherington, Editor).

D_1 = rate of change of doppler coefficient with core average void ($\Delta\alpha/\%$)
= $4.4(10)^{-3}$
V = core coolant average % void (=38% at full power).

Numerical values given above for the doppler coefficient, including the effect of coolant void fraction, are from Amendment 21 to the Browns Ferry FSAR, Figs. 3.6-5 and 3.6-6. The doppler coefficient, above, includes a weighting factor of 1.33, as recommended by NEDO-20964. This 1.33 factor accounts for the greater temperature changes in the more important parts of the fuel.

If the reactor is brought from full power to hot shutdown at 1000 psia, the fuel, on average, cools by about 660°F since the fuel temperature is very close to the coolant saturation temperature at hot shutdown. By the above formula, a negative reactivity of 0.00865 Δk/k would have to be added to compensate for the increased reactivity of the cooler fuel.

A.1.2 Void reactivity

The calculation of void reactivity is based on the average void fraction in the average channel. As explained in A.2, the void fraction is calculated at 1 ft axial intervals up the average channel. The calculation of average void fraction weights the void in each 1 ft section with the square of the normalized axial power distribution over that section. Table A.2 gives the axial power distribution used for the weighting. The use of flux squared weighting accounts for the greater reactivity of a given void when it is in a higher worth axial location.

The equation for void reactivity change accounts for the change in void reactivity coefficient with void fraction (void coefficient increases as void increases):

$$\Delta\rho_V = C_o(V - V_{100}) + C_1(V^2 - V_{100}^2)/2$$

where

$\Delta\rho_V$ = the change in total void reactivity (Δk/k)
V = average void fraction (%)
V_{100} = average void fraction at 100% power (%)
= 38%
C_o = void coefficient with no voids present (Δk/k/%)
= $-5.3(10)^{-4}$
C_1 = rate of change of void coefficient with void ($\Delta k/k/(\%)^2$)
= $-.1138(10)^{-4}$.

As the reactor is brought from full power to hot shutdown, the core average void changes from 38% to 0%. By the above formula, a negative reactivity of 0.0283 Δk/k would have to be added to compensate for the increased reactivity of the core without any voids. By adding this void reactivity change to the doppler reactivity change (see the bottom of subsection A.1.1), one can estimate that a total negative reactivity of

0.0369 would bring the reactor from full power to hot shutdown. This estimate does not consider the relatively slowly changing xenon reactivity which would, during the first ~8 h after accident initiation, help to shut down the reactor. In a period of only one or two hours, the buildup of xenon would not provide a substantial fraction of the reactivity required to reach hot shutdown. Therefore, in the relatively short-term ATWS transients examined in this report, either the control rods or coolant boration must supply a negative reactivity of at least 0.0369 $\Delta k/k$ to bring the reactor to hot shutdown.

A.1.3. Control rod reactivity

The reactivity due to manual control rod insertion in an ATWS accident would be a function not only of the physics and configuration of the reactor core, but also would depend on the reactor operators. Exercises conducted at the TVA Browns Ferry simulator showed that the proclivity of operators to perform all the manipulations necessary to maintain continuous control rod insertion during an ATWS would depend heavily on characteristics of the individual operator. Since constant attention is required to maintain continuous control rod insertion it is assumed here that an operator could easily be diverted from the manual rod insertion task 50% of the time. Therefore, the reactivity insertion rate is based on an effective average control rod speed of 1.5 in./s instead of the nominal rod speed of 3.0 in./s. The assumption of faster sustained control rod insertion can not be assumed at present because the training of operators to the EPG procedures for ATWS is still in an early stage.

$$t_i = 144 \text{ in.}/1.5 \text{ in./s} = 96 \text{ s}$$

where,

t_i = time consumed for each rod inserted (s)
144 in. = distance traveled by rod for full core insertion.

Page 3.6-11 of the Browns Ferry FSAR states that a control rod worth 10^{-3} $\Delta k/k$ would be very weak. Using this to represent average rod worth, the rate of reactivity addition during periods of manual rod insertion would be

$$\dot{\rho} = -10^{-3}(\Delta k/k)/t_i \cong -10^{-5} \ \Delta k/k/s$$

where,

$\dot{\rho}$ = average rate of reactivity insertion after the initiation of manual rod insertion.

This is the value used for the manual rod insertion calculations reported in Chap. 4.

A.1.4 Boron concentration and reactivity

The boron concentration in the reactor coolant depends on the rate at which the sodium pentaborate solution is pumped into the reactor vessel, the total volume of coolant in the reactor vessel and the mixing of the boron solution within the reactor coolant. Volume 4 of the Browns Ferry Hot License Training Manual states that there is 990 lb of boron in a volume of 4550 gal in the SLCS storage tank, and that, upon SLCS actuation, this solution is pumped into the reactor vessel at a rate of 50 gpm. Therefore, the rate of injection of boron into the reactor vessel is:

$$W_{binj} = \frac{990 \text{ lb B}}{4550 \text{ gal}} \; \frac{50 \text{ gal}}{\text{min}} \; \frac{1 \text{ min}}{60 \text{ s}} = 0.181 \text{ lb B/s} \; .$$

If the boron mixes perfectly within the reactor vessel, the boron concentration after SLC initiation is

$$C_b = t_i(0.181 \text{ lb B/s})/V_t$$

where,

C_b = boron concentration (lb B/ft^3)
t_i = elapsed time since SLCS initiation
V_t = total volume of water within the reactor vessel.

According to TVA operations analysis engineers, a boron fraction in the coolant of 320 ppm would bring the reactor from full power to hot shutdown. Using a coolant volume of 14785 ft^3 at the normal reactor water level of 561 in., the mass of boron within the reactor vessel would be:

$$M_b = 14785 \text{ ft}^3 \; \frac{45.4 \text{ lb } H_2O}{\text{ft}^3} \; \frac{320 \text{ lb B}}{10^6 \text{ lb } H_2O} = 215 \text{ lb B.}$$

Therefore, hot shutdown could be reached after only 19.8 min of SLC injection at 50 gpm.

When the Browns Ferry specific EPGs are written, they will probably reflect a slightly more conservative hot shutdown mass of 265 lbs B, based on a boron fraction of 395 ppm boron in reactor coolant required to reach hot shutdown with a margin of 0.02 $\Delta k/k$. The corresponding SLCS injection time would be 24.4 min.

For the calculations of Chapt. 4 with boron injection it was necessary to calculate the boron reactivity at each instant during the transient. The method used for this is based on the TVA estimate of the hot shutdown ppm boration requirement and the boron mixing information presented in the GE BWR owners group report "Power Suppression and Boron Remixing Mechanism for General Electric Boiling Water Reactor Emergency Procedures," DAC 261, NEDC-22166, August 1983 (prepared by L. Chu).

Boron concentration is calculated for two subvolumes within the reactor vessel: (1) the volume of coolant at the bottom of the vessel lower plenum, and (2) all other coolant within the vessel. As explained in NEDC-22166, if the core inlet flow is less than 5% of its full power value, 100% of the injected boron solution sinks into the bottom of the lower plenum (i.e., the initial mixing efficiency is 0%). At 25% flow the initial mixing efficiency climbs to 75% and it is 100% at full flow. The residence time of the heavier boron solution in the lower plenum is also dependent on the reactor coolant flow. If primary coolant flow is 4% or less, the residence time is infinite but when primary coolant flow is above about 15%, the residence time is only about 22 s. In the BWR-LACP model, the mass of boron in each of the two control volumes is calculated using the following set of equations

$$d(M_{blp})/dt = (1 - E_{im})\, W_{binj} - M_{blp}/T_{rm}$$

$$d(M_{bg})/dt = E_{im} W_{binj} + M_{blp}/T_{rm}$$

where,

M_{blp} = mass of boron stratified in the bottom of the lower plenum (lb)
M_{bg} = mass of boron in general circulation, in the balance of the coolant (lb)
E_{im} = initial mixing efficiency (lb B mix/lb B injected)
T_{rm} = residence time(s) of stratified boron in the bottom of the lower plenum.

The concentration of boron in general circulation, also assumed to be the boron concentration of the coolant in the core, is

$$C_{bg} = M_{bg}/V_t$$

where,

C_{bg} = boron concentration (lb/ft^3) in reactor coolant
V_t = total coolant volume (ft^3) in the vessel.

The net boron reactivity is then

$$\rho_b = \Delta k_{hsd}\, C_{bg}/C_{bhsd}$$

where,

ρ_b = total boron reactivity
Δk_{hsd} = total reactivity that must be supplied to reach hot shutdown
= -0.0369 $\Delta k/k$ (per Sect. A.1.2)
C_{bhsd} = boron concentration corresponding to the TVA estimate of 320 ppm B required to reach hot shutdown
= 0.0145 lb B/ft^3.

A.2. Calculation of Core Void Fraction

BWR-LACP calculates the void fraction profile at 1 ft intervals over the length of an average fuel assembly channel. Each time the void fraction routine is called it is given the core thermal power, the vessel pressure, and downcomer water level and enthalpy. The core inlet flow must also be known to allow calculation of the core void profile. The void fraction routine calculates the core inlet flow by means of an iterative procedure that assumes steady-state thermohydraulic conditions over each time step.

At the beginning of the iteration, a primary coolant flow is assumed, and the core void profile of the average channel is calculated at 1 ft intervals from the inlet to the outlet. Since the core is 12 ft long, this amounts to 13 node points. The average channel is a representative fuel assembly (one of a total of 764) that is assumed to generate (1/764)-th of the total core thermal power and to receive the same fraction of the total core inlet flow. The axial power distribution assumed for the average channel is specified by Table A.2.

The conservation of energy is applied across each 1 ft axial segment to calculate the steam generation rate. If the bulk coolant temperature is below saturation, a void fraction of zero is assigned. After coolant reaches saturation, the void fraction is calculated from the steam and water flows by the drift flux equations:

$$V = J_g/(C_o J + V_{gj})$$
$$J_g = XG/\rho_g$$
$$J = G[X/\rho_g + (1 - X)/\rho_f]$$

where,

G = mass flux
V = void fraction
J_g = steam mass velocity
C_o = concentration parameter = 1.0
J = total mass velocity
V_{gj} = drift velocity = 1.0 ft/s
X = flow quality (steam flow/total flow)
ρ_g = saturated vapor density
ρ_f = saturated fluid density.

The values used for the C_o and V_{gj} parameters were taken from the report "BWR Low Flow Bundle Uncovery Test and Analysis," NUREG/CR-2232, EPRI NP-1781, GEAP-24964, by D. S. Seeley and R. Muralidharan (April 1962).

After the core void profile is calculated, the unrecoverable pressure drops around the primary coolant loop are calculated. These unrecoverable losses include friction and/or form losses in the average channel unheated and heated portions, core outlet plenum, standpipes, steam separators, and jet pumps. The equations used to calculate these losses, and typical coefficients for each loss term, were taken from the EPRI report, "NATBWR; A Steady-State Model for Natural Circulation in Boiling Water Reactors," EPRI NP-2856-CCM, by J. M. Healzer and D. Abdollahian, S. Levi, Inc. (February 1983).

The only major difference between the natural circulation calculations in NATBWR and BWR-LACP is that the natural circulation through the core bypass path (mainly the interstitial region between fuel assemblies into which the control rods insert) is neglected in BWR-LACP. At full power conditions, about 10% of the core inlet flow bypasses the active fuel, flows up through the bypass paths, and rejoins the main flow in the core outlet plenum. Under natural circulation conditions, the direction of bypass flow can reverse, with coolant from the core outlet plenum flowing downward through the bypass paths to join with the majority of the core flow into the active fuel. The core bypass flow path was left out of BWR-LACP because it was felt that its relatively high flow resistance would limit the bypass flow to a small fraction of the total natural circulation flow. If this circulation path were included in BWR-LACP, the additional core flow under conditions of low vessel water level (i.e., downcomer water level near the top of the active fuel) would decrease the in-core coolant voiding and therefore lead to the prediction of higher core power (but certainly within the existing uncertainty bands quoted by leading investigators in References 6.8 and 6.9). The effect would be negligible for a normal vessel water level (i.e., 10 to 15 ft above the top of active fuel) because the change in core flow would be small compared to the already substantial natural circulation.

Elevation pressure drops (gains) around the reactor vessel primary coolant natural circulation flow path are also calculated after the void fractions are calculated. At the end of each iteration, the net elevation head (elevation pressure increases minus drops) around the loop is compared to the total unrecoverable losses around the loop. The value of flow for use in the next iteration is determined by the formula:

$$W_{new} = W_{old} \sqrt{\Delta P_{te} / \Delta P_{tu\ell}}$$

where

- W_{new} = total natural circulation flow to be used on the next iteration
- W_{old} = current iteration value of flow
- ΔP_{te} = net elevation pressure gain around the loop in the direction of natural circulation
- $\Delta P_{tu\ell}$ = total unrecoverable pressure losses around the natural circulation loop

If the new flow iteration is within 0.5% of the current flow iteration, further iteration is unnecessary and control is returned to the main program.

A.3 Reactor Vessel Injection Systems

A.3.1 Core spray, LPCI, and condensate booster pumps

In the no operator action case presented in Chap. 3 there are three systems that provide high capacity, low pressure injection. The two low pressure ECCS systems, Core Spray and LPCI, actuate automatically and pump from the pressure suppression pool into the reactor vessel. The condensate pumps, in series with the condensate booster pumps run continuously during normal operation and continue to run after a reactor scram, pumping from the main condenser hotwell, through the idle main feedwater pumps, into the reactor vessel. Using TVA-supplied pump head vs. capacity curves and schematic piping diagrams, equations for injected flow as a function of reactor vessel pressure were developed at ORNL for each of these injection systems:

$$B_{lpci} = 41266\sqrt{1 - (P_v - P_p)/331}$$

$$B_{cs} = 3879\sqrt{1 - (P_v - P_p)/342}$$

where,

B_{lpci} = bulk flow (gpm) injected by all four LPCI pumps
B_{cs} = bulk flow (gpm) injected by all four Core Spray pumps
P_v = reactor vessel pressure (psia)
P_p = pressure suppression pool pressure (psia).

The condensate/condensate booster pump injection flow as a function of reactor vessel pressure is given by Table A.3. The following conditions apply: three condensate and three condensate booster pumps are running, and the main condenser hotwell is assumed to be at atmospheric pressure.

A.3.2 HPCI system

The HPCI provides some injection in all the ATWS transients presented in this report. The following assumptions are made concerning characteristics of the HPCI system: (1) the HPCI turbine automatic flow control system adjusts HPCI flow to any operator-set flow demand between 20% and 100% of the 5000 gpm full capacity; (2) the automatic flow controller cannot respond to operator flow demands below 20% because of the minimum HPCI turbine speed limitation; and; (3) the HPCI system will fail due to overheating of the HPCI turbine lube oil if water hotter than 190°F is pumped.

The assumption of HPCI failure at 190°F pumped water temperature is based on the discussion on pages Q14.1-4 and 5 of Amendment 67 to the Browns Ferry FSAR. This information was submitted in support of the limited-duration pumping of suppression pool water at 162°F during certain design basis accidents. The HPCI turbine lube oil is used in the HPCI turbine bearings, in the turbine governors, and in the gear reducer. Since the oil is cooled by the pumped water it will always be hotter than the pumped water. The FSAR discussion states that oil temperature in excess of 200°F is "to be avoided." Allowing for an oil cooler ΔT of 10°F, the corresponding limiting water temperature would be 190°F, and HPCI turbine failure is assumed to occur if the water exceeds this temperature.

In the cases examined in Chap. 4, the operators manipulate the HPCI injection flow to control vessel water level after the EPG-directed water level reduction maneuver. Although each operator would approach the task of level control in a slightly different way, the basic process would be the same in every case: the operator would periodically check the indication of water level and HPCI flow, and would either increase, decrease, or not change HPCI flow depending on the proximity of the indicated to the desired water level. BWR-LACP simulates operator control of HPCI flow in accordance with the following assumptions:

1. The operator would check vessel water level once per minute and may make up to one adjustment in HPCI flow per minute.
2. The operator would attempt to maintain the vessel water level above the minimum indication of the Emergency Systems range level instrument; the preferred vessel water level (setpoint) would be 380 in., which is 7 in. above the 373 in. bottom of the Emergency Systems range.
3. In the interest of preventing excessive reactor power* in an ATWS accident, the operators would not inject at a rate exceeding 2000 gpm.
4. If level is more than 5 in. from the setpoint, the operator would increase or decrease (as appropriate) the flow by 5% of the full HPCI capacity (i.e., 5% of 5000 gpm).
5. If level is more than 8 in. above the setpoint, the operator would decrease flow by 10%.
6. The operator would zero the flow if level is more than 20 in. above the setpoint.
7. If level is below the minimum range of the Emergency Systems instrument, the operator would increase flow by 10%.

A.3.3 Operator controlled Condensate/Condensate Booster pump injection

For all the cases in Chap. 4 that result in emergency depressurization, it is assumed the operators would provide needed reactor vessel injection by using one condensate and one condensate booster pump, in series, to pump from the main condenser hotwell to the reactor vessel. They would close the feedwater pump discharge isolation valves (to prevent vessel flooding) and bypass the feedwater pumps by opening the

*Without the addition of poison to the core, flow injected into the reactor vessel is the major determinant of reactor power. This fact is a basic premise of the EPG procedures for ATWS (see Appendix B).

startup bypass valve (see Fig. 3.8). The startup bypass valve is installed in an eight inch pipe; its position is indicated in the control room. Main feedwater flow is also indicated in the control room. The startup bypass valve provides a means to supply the moderate to low flow required in an ATWS transient by using the motive power of the very high capacity condensate and condensate booster pumps.

Operator control of injected flow using the startup bypass is simulated in BWR-LACP in accordance within the following assumptions:

1. The operator checks vessel water level once per minute and may adjust injected flow once per minute.
2. If the Emergency Systems level indication is off-scale low, the injection rate is set at 1800 gpm (113 l/s).
3. If the level indication is on-scale of the Emergency Systems range instrument, but below the desired level for manual control near the TAF [380 in. (9.65 m) above vessel zero], the injection flow is set at 900 gpm (57 l/s).
4. If the level indication is above the desired level, injection flow is set at 600 gpm (38 l/s).
5. If the level indication is more than 20 in. (51 cm) above the desired level, injection flow is set to zero.

A.4 Safety Relief Valves (SRVs)

Each of the 13 SRVs automatically opens when vessel pressure exceeds the opening setpoint and closes when pressure decreases to about 5% below the opening setpoint. The first bank of four SRVs is set at 1120 psia, the second bank of four is set at 1130 psia, and the third bank has five SRVs set to open at 1140 psia. According to the ASME code, the valves must open within 1% of the nominal opening setpoint. Conversation with TVA operations analysis engineers reveals that the closing pressures range between 6% and 11% below the nominal opening pressures. The opening and closing pressures used for the BWR-LACP simulation are given by Table A.4. These actual setpoints were derived by spreading the nominal setpoints over the ranges discussed above.

Each of the 13 SRVs can be opened or closed by operator manipulation of hand switches in the control room. The EPGs direct the operators to actively attempt to control vessel pressure by manual SRV control. It was desired to simulate operator control of SRVs as realistically as possible in order to avoid an excessively choppy vessel pressure behavior. The simulation of operator control of vessel pressure includes operator recognition of the absolute vessel pressure as well as its rate of change and general upward or downward trend. The BWR-LACP simulation is based on the following assumptions:

1. The operator checks vessel pressure once per minute, and may make up to one SRV manipulation per minute.
2. The upper and lower bounds for desired vessel pressure are 1050 and 900 psia, respectively. After emergency depressurization these bounds would be shifted downward to 300 and 0 psia.

3. If vessel pressure is outside of the desired range and is 60 psi further from the desired range than one minute ago, one SRV is opened or closed, as appropriate.
4. If vessel pressure is outside of the desired range and has either increased or decreased by more than 120 psi over the previous three minutes, one SRV is opened or closed as appropriate.

A.5. Vessel Level Indication

There are two vessel water level instruments mentioned in this report: the Emergency Systems range indicator and the Post-Accident Flooding range indicator. Their ranges in relation to the reactor vessel and internals are shown on Fig. 4.7. Both these instruments measure the collapsed water level in the downcomer annulus of the reactor vessel.

The Emergency Systems indication covers the range from above normal water level down to about only 1 ft above the top of active fuel. This indication is calibrated to read correctly when the reactor coolant is hot and at full pressure. The Yarway system of reference leg compensation minimizes the error when the reactor coolant is cooled to below operating temperature. The variable leg is outside the reactor vessel and is clamped physically and thermally to the reference leg. Steam from the reactor vessel condenses in the constant head condensing chamber and circulates back to the reactor vessel through the variable leg, transferring enough heat to maintain the reference leg temperature about 50% of the way between the drywell air temperature and the reactor coolant temperature.

The Post-Accident Flooding range indicator covers the range from 100 in. below the TAF to 200 in. above the TAF. The indication is calibrated to read correctly when the reactor vessel is depressurized and reactor coolant is at about 212°F. The variable leg is inside the reactor vessel (it is the vessel downcomer annulus), and the reference leg is not heated. The reference leg will therefore remain close to the temperature of the drywell atmosphere.

Either of the level indication systems under consideration here consists of a ΔP sensing element, an electronic circuit to transform the ΔP signal to a level signal, and the indicating meter. The ΔP element measures the difference between the pressure at the bottom of the reference leg and the pressure at the bottom of the variable leg. The reference leg is (or should be) always water-filled*; the amount of water and/or steam depends on the actual water level inside the vessel downcomer.

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*During rapid reactor vessel depressurization the heated reference leg of the Yarway instrument can flash, causing a temporary full-to-the-top level indication. This effect is not simulated in BWR-LACP.

The potential for error addressed here is in the circuitry that transforms the pressure difference into a water level. This circuitry is designed to always give the same level indication for the same measured pressure difference. Suppose that the vessel water level stays the same, but that the density of the water either in the reference leg or in the variable leg changes; the measured pressure difference would change and thus the indicated water level would change when, in fact, there was no change in actual water level.

The following equations are used in BWR-LACP to compute the effect on indicated level of reference leg or variable leg conditions that differ from calibration condition:

$$L_{ind} = L_{max} - (\Delta P - \Delta P_t^*)(\Delta L_i)/(\Delta P_b^* - \Delta P_t^*)$$

where,

L_{ind} = indicated height of water in the downcomer annulus,
L_{max} = height of the upper end of the indication range,
ΔP = measured pressure difference,
ΔP_t^* = pressure difference that would be measured at calibration conditions if the vessel water level were at the top end of the indication range,
ΔL_i = length of the indication range, and
ΔP_b^* = pressure difference that would be measured at calibration conditions if the reactor vessel were at the bottom end of the indication range.

The measured ΔP is a function of the actual vessel water level and the reference leg and variable leg water densities:

$$\Delta P = (\Delta L)\rho_r - (\Delta L_\ell)\rho_\ell - (\Delta L - \Delta L_\ell)\rho_g$$

where,

ΔL = distance between the upper and lower ΔP taps,
ρ_r = water density of the reference leg,
ΔL = reactor vessel downcomer liquid level above the lower ΔP tap,
ρ_ℓ = density of variable leg water (i.e., reactor coolant in the downcomer annulus
ρ_g = density of the reactor vessel steam

The BWR-LACP calculation makes the assumption that ρ_ℓ is equal to the density of saturated fluid evaluated at reactor vessel pressure. The steam density is set equal to the density of dry saturated vapor at vessel pressure. The reference leg density is evaluated at reference leg temperature and vessel pressure:

$$\rho_r = \rho_r(T_r, P_v)$$

where

$T_r = 0.4\ T_{sat} + 0.6\ T_{dw}$ for the Emergency System range,
$T_r = T_{dw}$ for the Post-Accident Monitoring range
P_v = reactor vessel pressure,
T_{sat} = saturation temperature at P_v, and
T_{dw} = drywell atmosphere temperature.

The remaining terms in the expression for L_{ind} are given by the following

$$\Delta P_t^* = \Delta L_i(\rho_r^* - \rho_\ell^*)$$

$$\Delta P_b^* = \Delta L_i(\rho_r^* - \rho_g^*)$$

For the Emergency Systems level indication, the reference leg calibration density, ρ_r^*, is evaluated at 290°F and ρ_ℓ^* and ρ_g^* are evaluated at 1015 psia saturation condition.

For the Post-Accident Monitoring level indication, the reference leg calibration density, ρ_r^*, is evaluated at 135°F, and ρ_ℓ^* and ρ_g^* are evaluated at a 14.7 psia saturation condition.

A.6 Comparison of RELAP and BWR-LACP Results

This section provides a comparison of RELAP and BWR-LACP results for a hypothetical MSIV-closure initiated ATWS accident with operator recovery actions to control reactor vessel pressure and water level, but without SLCS sodium pentaborate injection or manual rod insertion. The RELAP5/MOD1.6 run was performed at INEL and sent to ORNL by W. C. Jouse of EG&G, Idaho, Inc., by letter dated November 14, 1983 ["Need to Identify and Assess Computational Uncertainties Associated with Plant Transient Simulations for Severe Accident Sequence Analysis (SASA) Program — (WCJ-3-83)"].

In attempting to replicate the RELAP results with BWR-LACP an effort was made at ORNL to use the same rules for the simulation of the operator actions to control vessel pressure and water level that were used at INEL for the RELAP work. Input for both codes assumes that there is an automatic HPCI suction shift (on high suppression pool water level) and subsequent failure of the HPCI injection when suppression pool temperature reaches 180°F (slightly lower than the 190°F failure criterion used in the body of the report). No attempt was made to see that code inputs not related to operator or automatic control actions are the same. For example, the BWR-LACP code may have slightly different doppler or void reactivity coefficients.

The BWR-LACP simulation of operator control of the SRVs (as modified for this comparison) is based on the following rules:

1. The desired vessel pressure (setpoint) for operator control is the lower of 950 psia or the EPG limit on vessel pressure based on the suppression pool heat capacity temperature limit.*
2. The operator checks vessel pressure continuously.
3. If pressure is above the setpoint, one SRV is opened.
4. If pressure is more than 50 psi below the setpoint, one SRV is closed.
5. No more than one SRV opening or closing is allowed in any one 20 s period.

The BWR-LACP simul tion of operator control of vessel water level using the HPCI system (as modified for this comparison) is based on the following rules:

1. The desired setpoint for operator control is a level equivalent to the top of the active fuel.
2. The amount of flow demanded by the operator is equal to the difference between the actual level and the setpoint, multiplied by a gain of 500 gpm/ft.
3. There is a 20 s lag (time constant) between the formation of the flow demanded by the operator and the realization of this flow via the HPCI system (i.e. to simulate delay in the operator response).
4. Demanded flow may not go below 1200 gpm.
5. The operators prevent the automatic initiation of injection by LPCI and Core Spray systems.

The RELAP and BWR-LACP results for this transient are plotted on Figs. A.1—A.5, and the sequence of events is summarized on Table A.5. The major events predicted by each code are essentially the same, but BWR-LACP predicts that the events happen much sooner. The reason for this quicker response is that BWR-LACP predicts a higher reactor power throughout the transient. Since reactor power is higher, the pressure suppression pool (PSP) heats faster and consequently the vessel pressure setpoint is reduced faster by the PSP heat capacity temperature limit. The effect is amplified because the faster depressurization heats the pool faster, thereby reducing the vessel pressure setpoint even more rapidly.

The length of the time scale for the plots of the BWR-LACP results has been stretched relative to the length of the RELAP time scale, on the basis of the time required for the vessel pressure setpoint to reach 255 psia. This was done to emphasize the basic similarity of the trends in system variables. The BWR-LACP code was not changed to decrease the predicted reactor power closer to the RELAP-predicted reactor power. This would have extended the time required for the sequence to unfold, bringing the BWR-LACP event timing into closer agreement with the RELAP

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*The reactor vessel pressure vs. pressure suppression pool temperature curve used here can be found in Section SP/T of Rev. 3 to the GE BWR Owners Group EPGs. This curve is different from the curve TVA is intending to use for the Browns Ferry specific version of the EPGs (Fig. 4.10, this report).

timing. There does not seem to be a compelling justification for such a move because, at present, the differences between RELAP and BWR-LACP reactor power predictions are not greater than the uncertainty inherent in either method. INEL has estimated that the maximum uncertainty on the RELAP prediction of core power level under ATWS conditions with water level at or near the top of the active fuel is 100%[6.8] and the General Electric Company has specified a maximum uncertainty band of ±50%.[6.9]

At a recent SASA program interlab information exchange meeting, preliminary results were presented that indicate that the RAMONA code, being run at BNL, predicts higher power levels than either BWR-LACP or RELAP [presentation by P. Saha and G. Slovik, Department of Nuclear Energy, Brookhaven National Laboratory, Upton, New York (April 11, 1984)]. Since RAMONA employs a more sophisticated calculation of the core neutronics, it seems inappropriate at the present time to manipulate BWR-LACP code input to reduce the predicted core power levels in an attempt to force agreement with the RELAP results.

A.7 Condensation of SRV T-quencher Discharge

Steam discharged by a T-quencher into the pressure suppression pool (PSP) is condensed by an induced flow of subcooled water from the vicinity of the T-quencher. This induced flow mixes with and exchanges heat with the steam as it flows into and up through the surrounding water. The minimum induced flow of subcooled water necessary for complete condensation is

$$W_{mif} = W_{srv} (h_s - h_f)/(h_f - h_{local})$$

where

W_{mif} = minimum induced flow of subcooled water necessary for 100% condensation
W_{srv} = flow of SRV steam being discharged by the T-quencher
h_s = enthalpy of the steam being discharged
h_f = enthalpy of saturated fluid evaluated at wetwell pressure
h_{local} = enthalpy of the subcooled water surrounding the T-quencher.

From this equation we see that complete condensation would not be possible if the PSP were saturated because the induced flow of water feeding the quenching process would have to be infinite. Without applicable experimental data, it is very difficult to predict exactly how much subcooling is required for complete condensation. D. C. Cook concluded from a survey of available experimental data that a minimum of about 10°F of subcooling is required for complete condensation [D. H. Cook, doctoral dissertation, "Pressure Suppression Pool Thermal Mixing," NUREG/CR-3471, ORNL/TM-8906 (to be published)]. Based on Cook's conclusions, the following condensation relationship was built into BWR-LACP:

1. Condensation = 100% ....If $T_{subcooling} > 10°F$
2. Condensation = 0% ....If $T_{subcooling} < 0°F$
3. Condensation = $T_{subcooling}/10$ ....If $0 < T_{subcooling} < 10°F$

where,

$T_{subcooling} = T_{sat} - T_{local}$

T_{sat} = saturated fluid temperature evaluated at the total pressure in the vicinity of the T-quencher.

T_{local} = temperature of the water surrounding the T-quencher.

It is important to note that the suppression pool does not cease all condensation when the pool temperature reaches the point of less than 10°F subcooling. When steam bubbles, uncondensed, through to the surface of the PSP there is an increase in the pressure of the wetwell atmosphere over the pool. This additional pressure increases the subcooling of the PSP water and allows the condensation process to continue. As the SRV discharge continues, most of the thermal energy of the discharge is absorbed by condensation in the pool. Only a portion of the SRV discharge escapes condensation as necessary to maintain a subcooling somewhere between 0 and 10°F.

A.8 Pressure Suppression Pool Temperature Distribution

The primary assumption of the BWR-LACP model of the suppression pool is that the temperature of water throughout the suppression pool is uniform. During an accident involving extended SRV discharge, this assumption leads to the result that the very large water mass of the whole pool is available as a heat sink for the thermal energy discharged by the SRVs.

It is logical to question how there could be sufficient circulation around the approximately 350 ft (107 m) circumference of the pool to justify the assumption of a well-mixed pool. Without such circulation, only water in the vicinity of a discharging T-quencher could act as a heat sink; incomplete condensation of SRV discharge would begin much sooner, and primary containment pressure would build up faster. D. H. Cook has studied this question extensively and has developed a two dimensional multi-node computer model that calculates the transient variation of pool temperature with depth (distance from the bottom of the pool) and with angular displacement around the torus [D. H. Cook, doctoral dissertation, "Pressure Suppression Pool Thermal Mixing," NUREG/CR-3471, ORNL/TM-8906 (to be published)]. The code allows a wide variety of combinations of discharging T-quenchers, and allows an arbitrary mass discharge vs. time for each T-quencher.

Cook's code has been run in conjunction with the BWR-LACP code (in replacement of the BWR-LACP uniform pool temperature model) for selected

ATWS transients. In general, the more detailed pool calculation shows that the water temperature above a discharging T-quencher is higher than the bulk pool temperature and that this temperature difference sets up powerful density currents that mix the contents of the whole pool. The hot, less dense water rising above a discharging T-quencher flows upward to the surface and spreads across the top of the pool, while a subsurface current of relatively cool water flows in the opposite direction along the bottom of the pool toward the discharging T-quencher. As a result, the whole pool is able to function as a heat sink and the rate of pressure buildup is not significantly faster than would be obtained with the uniform pool temperature model.*

For example, the Chapter 5 case plotted on Figs. 5.1-5.5 was run with Cook's pool model, and with the following assumptions: (1) the long period of intermittent actuation of a single SRV is through the same SRV, discharging to the same T-quencher, and (2) there is no pool cooling. The first assumption maximizes local temperature buildup. The second assumption also maximizes local temperature buildup by preventing the significant (~40000 gpm) pool circulation that goes along with pool cooling.

After the first hour, the volumetric average pool temperature was 189°F (361 K), the maximum single-node pool temperature (occurring above the discharging T-quencher) was 202°F (368 K), and the average bottom temperature was 177°F (357 K). This result shows that water near the discharging T-quencher is 13°F higher than the volumetric average pool temperature. However, this doesn't adversely affect the condensation of the T-quencher discharge because the T-quenchers are submerged 10 ft (3.05 m) below the surface of the pool and the water that feeds the condensation process is cooler than the water at the surface of the pool.

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*It should be recognized that this conclusion is based upon an ATWS accident sequence, in which the rate of discharge into the pool is relatively large.

Table A.1 Neutron kinetics data[a]

Delayed Neutron Group	Fraction	Decay Constant (s^{-1})
1	$0.207(10)^{-3}$	0.0127
2	$0.1163(10)^{-2}$	0.0317
3	$0.1027(10)^{-2}$	0.115
4	$0.222(10)^{-2}$	0.311
5	$0.699(10)^{-3}$	1.4
6	$0.142(10)^{-3}$	3.87

[a]From "RAMONA Analysis of the Peach Bottom-2 Turbine Trip Transients," by Scandpower, Inc., EPRI Report No. NP-1869, June 1981

Table A.2 Assumed[a] full power steady state axial power distribution

Distance from bottom of Active Fuel (ft)	Relative power
0	0.61
1	1.04
2	1.16
3	1.19
4	1.16
5	1.11
6	1.09
7	1.07
8	1.05
9	1.03
10	0.92
11	0.72
12	0.33

[a]Applicable to end-of-cycle, equilibrium xenon full power operation.

Table A.3 Condensate/condensate booster pump injected flow as a function of reactor vessel pressure

Vessel Pressure (psia)	Injected Flow (lb/s)
418	0
404	743
366	1486
303	2229
217	2973
106	3716
42	4087
0	4292

Table A.4 Setpoints for automatic SRV actuation

Valve	Bank	Nominal Opening Pressure (psia)	Actual Opening Pressure (psia)	Nomonal Closing Pressure (psia)	Actual Closing Pressure (psia)
1	1	1120	1115	1064	1052
2	1	1120	1118	1064	1030
3	1	1120	1120	1064	1042
4	1	1120	1125	1064	1014
5	2	1130	1126	1073	1023
6	2	1130	1130	1073	1062
7	2	1130	1131	1073	1042
8	2	1130	1135	1073	1051
9	3	1140	1138	1083	1072
10	3	1140	1140	1083	1032
11	3	1140	1141	1083	1060
12	3	1140	1145	1083	1053
13	3	1140	1147	1083	1015

Table A.5 Sequence of events for RELAP/BWR-LACP comparison transient

RELAP Time (s)	BWR-LACP Time (s)	Event
0	NA	Beginning of MSIV closure. RELAP calculation begins
3.75	NA	Peak reactor power (275%)
3.68	NA	Recirculation pumps tripped, reactor power decreasing rapidly
14	NA	Peak vessel pressure of 1312 psia, all 13 SRVs open (automatic actuation)
NA	50	BWR-LACP calculation begins
68	60	HPCI,RCIC Systems on at full flow (5600 gpm total injection)
50–150	50–150	RELAP power averages 22.5%, BWR-LACP power averages 30%.
150	150	Operators begin vessel level, pressure control. RCIC tripped, HPCI flow reduced to 1200 gpm.
175	190	Last automatic SRV actuation until power/pressure spike at end of depressurization
325	230	PSP heat capacity temperature curve begins to reduce vessel pressure setpoint
357	280	Vessel water level reaches: TAF RELAP power level = 7%, BWR-LACP power level = 9%
992	992	RELAP power level below 3%. BWR-LACP power level below 5%
1850	1280	PSP heat capacity temperature limit on vessel pressure reaches 255 psia, stops decreasing (@ PSP temp $\cong$ 160°F)
2000	1330	Operator begins closing SRVs to attempt to control vessel pressure at 255 psia
2400	1420	Reactor power spike (RELAP to 120%, BWR-LACP to 68%) accompanied by repressurization of the reactor vessel and automatic SRV actuations
2480	1680	Failure of HPCI system after PSP temperature exceeds 180°F (total injection flow reduced to the ~ 200 gpm from the CRD hydraulic system)
2500	1740	Vessel water level below the TAF and decreasing
2900	1570	Vessel pressure below 250 psia and decreasing

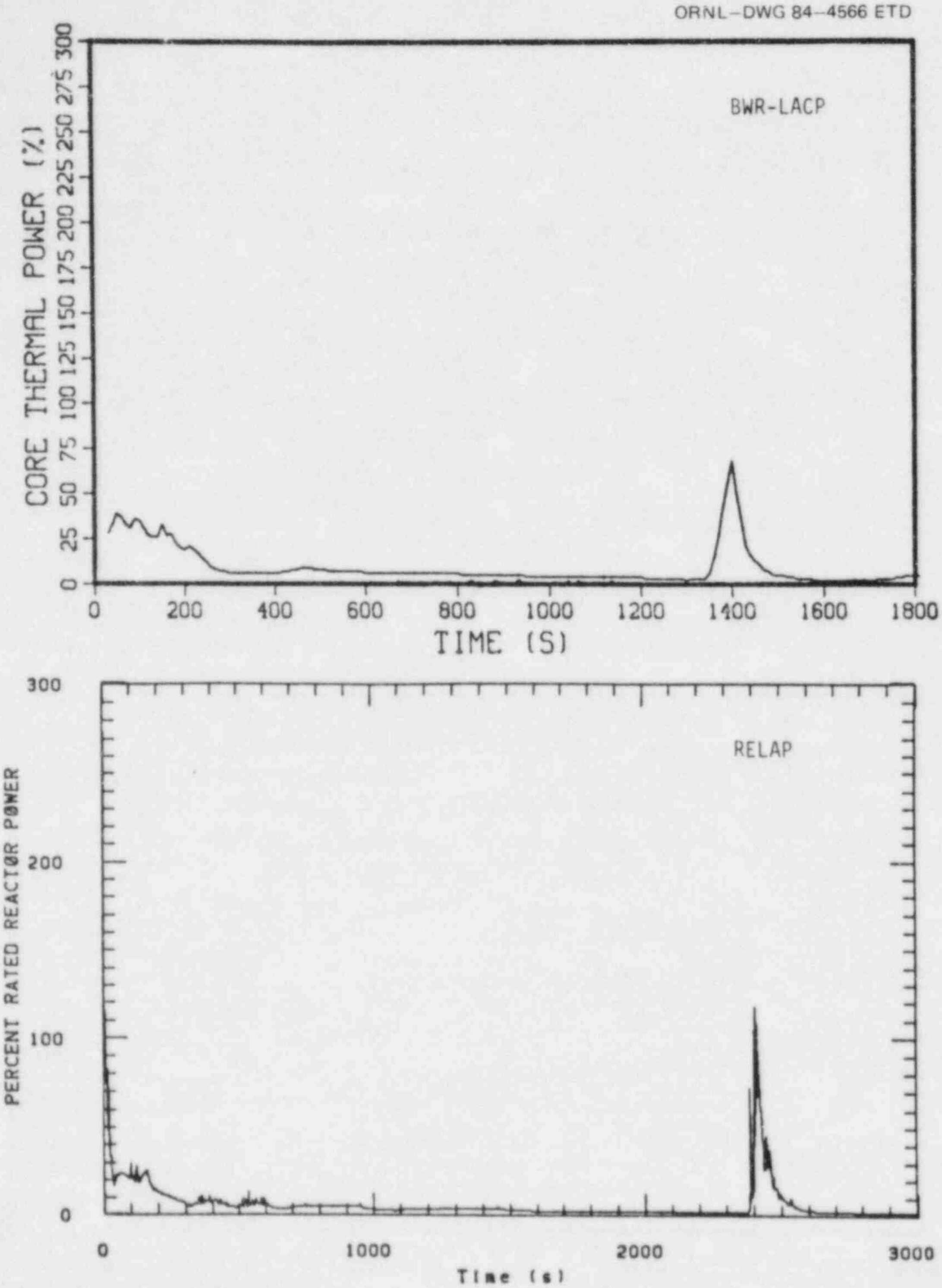

Fig. A.1. RELAP 5/MOD 1.6 vs BWR-LACP comparison for hypothetical ATWS recovery — reactor power.

ORNL-DWG 84-4567 ETD

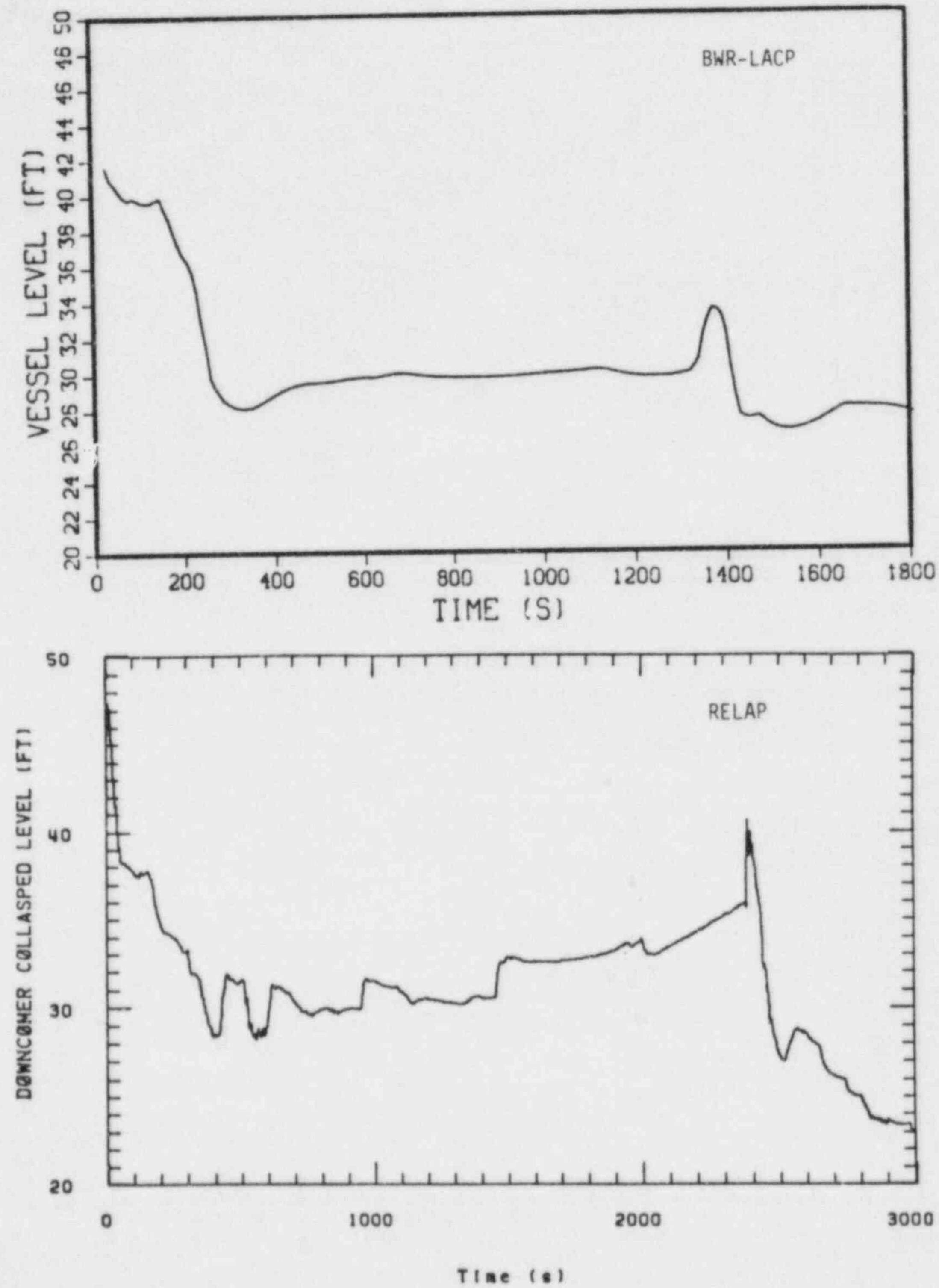

Fig. A.2. RELAP 5/MOD 1.6 vs BWR-LACP comparison for hypothetical ATWS recovery — reactor vessel water level.

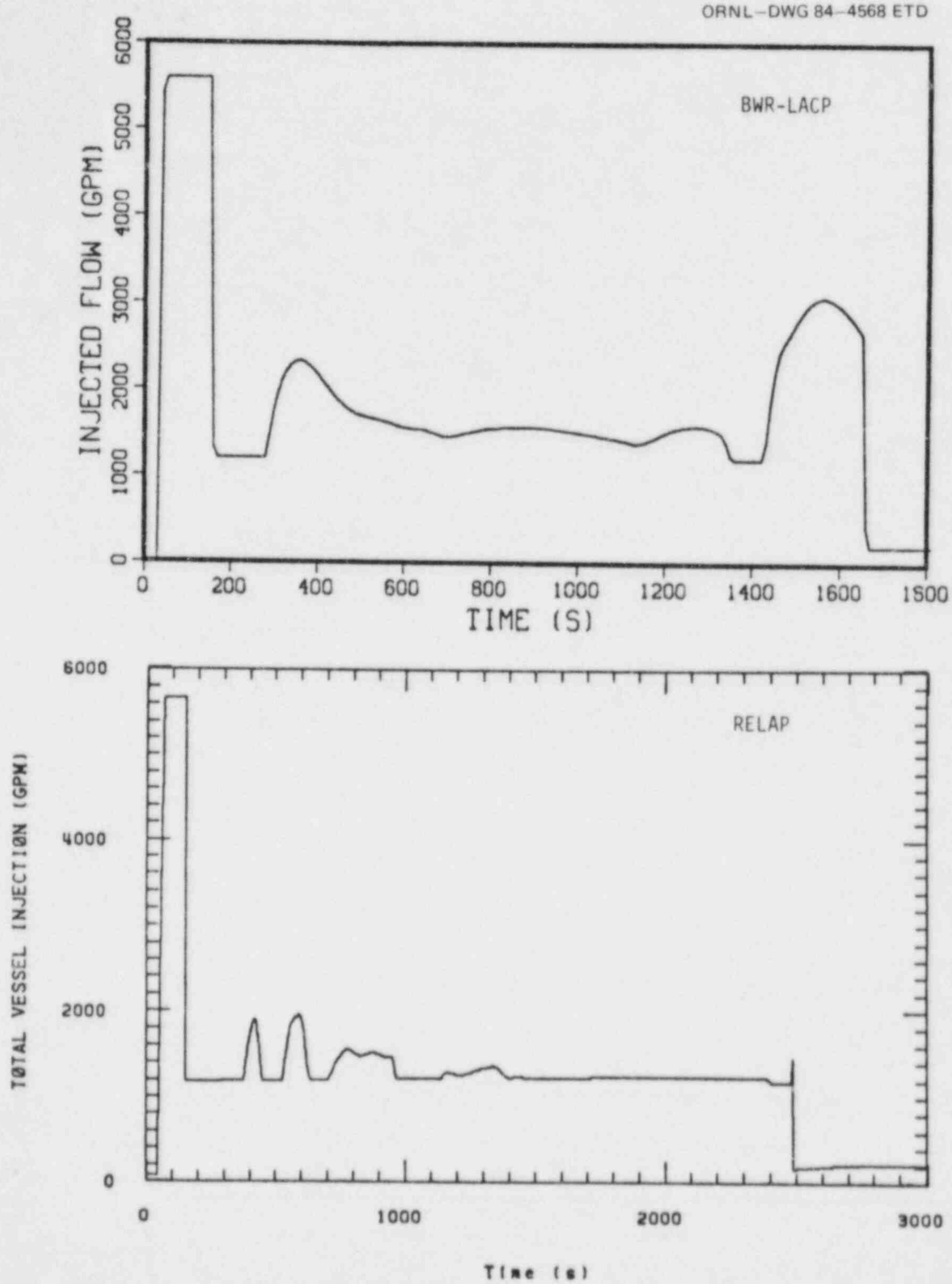

Fig. A.3. RELAP 5/MOD 1.6 vs BWR-LACP comparison for hypothetical ATWS recovery — injected flow.

ORNL-DWG 84-4569 ETD

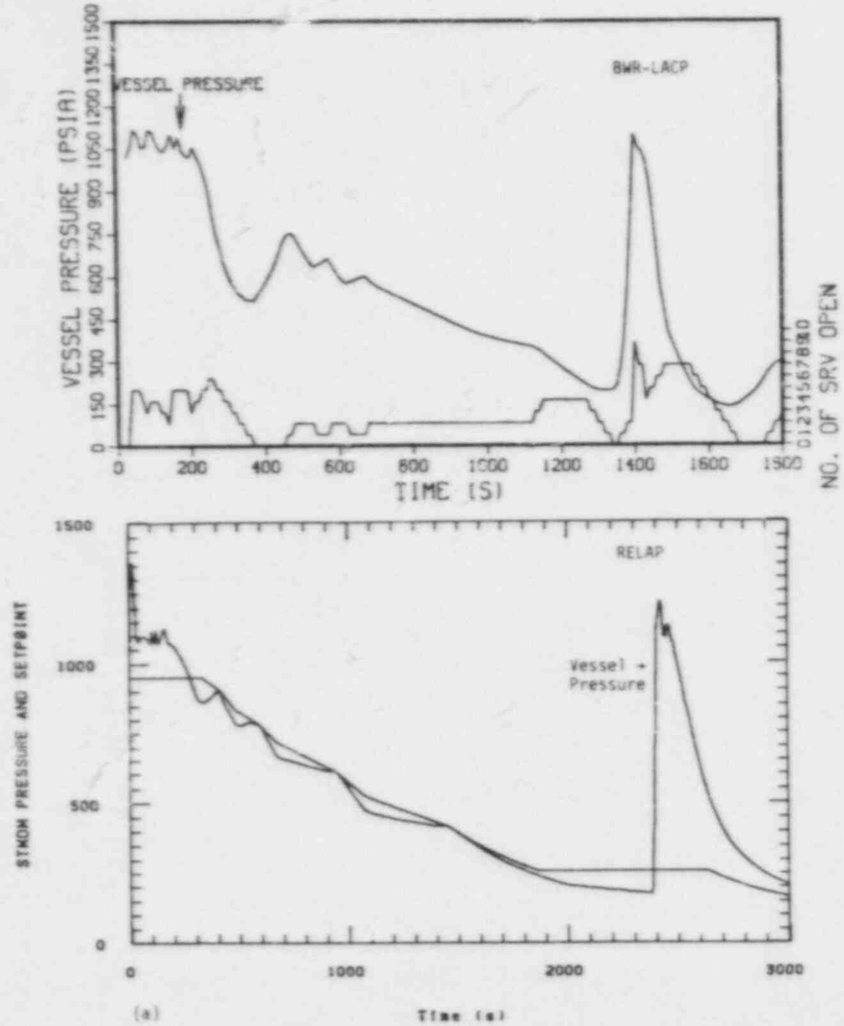

Fig. A.4(a). RELAP 5/MOD 1.6 vs BWR-LACP comparison for hypothetical ATWS recovery — reactor vessel pressure.

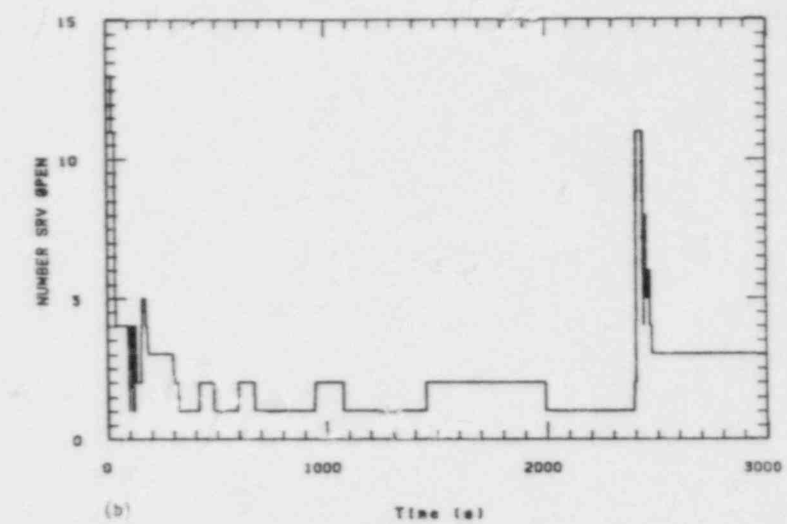

Fig. A.4(b). RELAP 5/MOD 1.6 results for SRV actuations. BWR-LACP results are shown on Fig. A.4(a).

ORNL-DWG 84-4570 ETD

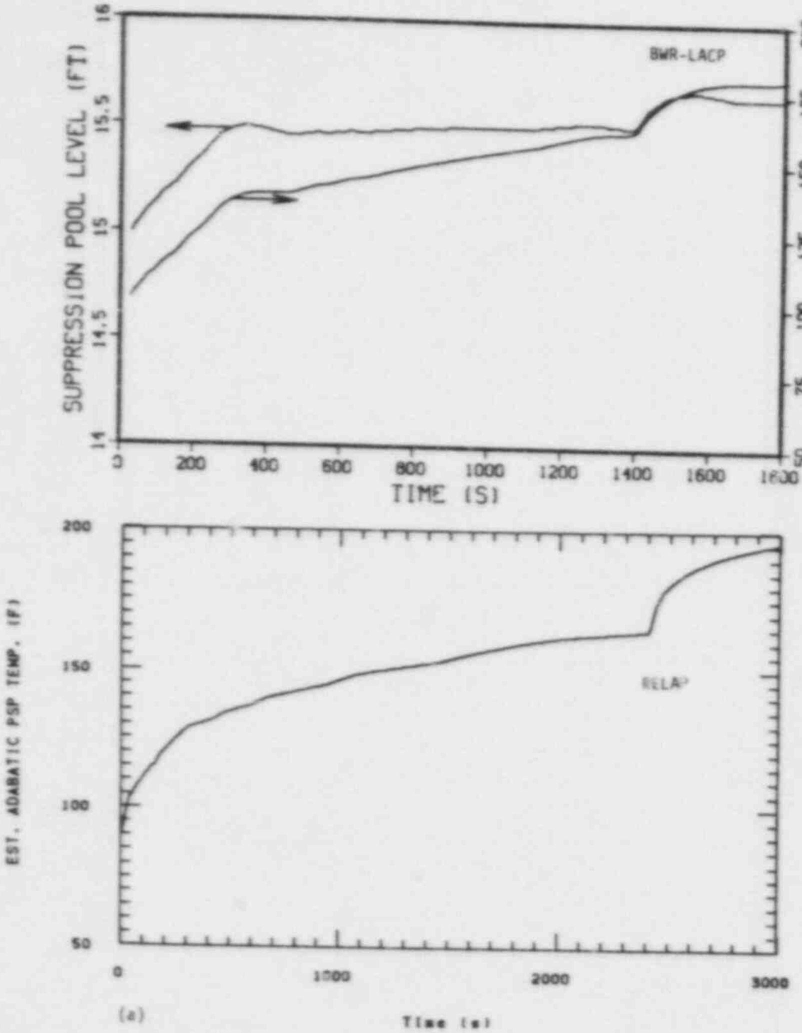

Fig. A.5(a). RELAP 5/MOD 1.6 vs BWR-LACP comparison for hypothetical ATWS recovery — pressure suppression pool temperature.

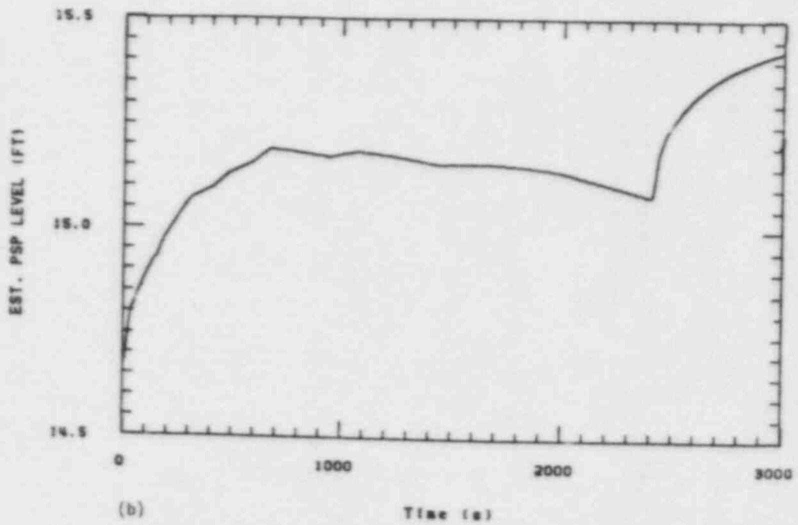

Fig. A.5(b). RELAP 5 results for pressure suppression pool level. BWR-LACP results are shown on Fig. A.5(a).

APPENDIX B: ATWS CALCULATIONS FOR THE STEADY STATE

B.1 Introduction

An ATWS accident sequence would be initiated by an anticipated transient demanding reactor scram for which the negative reactivity insertion that would be provided by inward control rod movement does not occur. If the MSIVs are shut, all steam exiting the reactor vessel is discharged into the pressure suppression pool, and the pool temperature increases rapidly. To avoid primary containment failure and the consequences, the operators must act to manually introduce enough negative reactivity to temporarily reduce reactor power until enough liquid neutron poison has been injected to provide permanent reactor shutdown.

The purpose of this appendix is to discuss the calculational sophistication required to determine reactor power under the conditions of an MSIV-closure initiated ATWS. The operators can manually reduce the reactor power by taking control of the high-pressure injection systems and decreasing the injection rate. It is shown in Sect. B.2 that if the injection rate to the reactor vessel is specified, then the steady state power can be determined by a simple hand calculation. On the other hand, if operator control of the reactor vessel water level is specified, then the calculation of steady state power is much more complicated, as explained in Sect. B.3. The conclusions of this appendix are summarized in Sect. B.4.

B.2 The Case with Specified Injection Rate

Unless the operators take action to depressurize the reactor vessel, makeup flow under the conditions of an MSIV-closure initiated ATWS could only be provided by the HPCI, RCIC, and CRD hydraulic systems. The HPCI and RCIC systems inject into the reactor vessel through the feedwater lines whereas the relatively small CRD hydraulic system flow enters the reactor vessel through the control rod guide tubes.

At least one SRV would remain continuously open as long as the steam release from the reactor vessel constituted more than 6.5% of the steam flow at normal full-power operation. The definition of terms for the power calculations is shown in Fig. B.1, where:

$\dot{M}_W$ = injection mass flow, lb/h
h_W = specific enthalpy of injection flow, Btu/lb
$\dot{Q}$ = core thermal power, Btu/h
$\dot{M}_s$ = steam flow through SRVs, lb/h
h_s^s = specific enthalpy of steam, Btu/lb

At steady state, the reactor vessel water level would be constant, $\dot{M}_s$ is equal to $\dot{M}_W$, and

$$Q = \dot{M}_W \, (h_s - h_w) \text{ Btu/h} \tag{B.1}$$

A simple but accurate "rule of thumb" for the Browns Ferry Unit 1 reactor can be developed by assuming that the reactor vessel pressure is at the setpoint of the lowest-set bank of SRVs (1120 psia) and that the injection temperature (from the condensate storage tank) is 90°F. Then

$$h_s = 1187.3 \text{ Btu/lb} \tag{B.2}$$

$$h_w = 58.1 \text{ Btu/lb} \tag{B.3}$$

and

$$Q = 1129.2 \, \dot{M}_W \text{ Btu/h} \ . \tag{B.4}$$

Equation (B.3) can be cast into a more useful form by use of the following relations:

$$1 \text{ Btu/h} = 2.931 \times 10^{-7} \text{ MW}_t \tag{B.5}$$

$$100\% \text{ power} = 3293 \text{ MW}_t \tag{B.6}$$

$$1 \text{ GPM} = 499.3 \text{ lb/h (at 90°F)} \ . \tag{B.7}$$

Then

$$P_R = 5.02 \times 10^{-3} \, F_W \ \% \tag{B.8}$$

where

P_R = reactor thermal power as percent of full power operation,
F_W = injection rate, GPM.

As an example of the use of Eqn. (B.8), the combined injection rate of the HPCI and RCIC systems after automatic initiation is 5600 GPM. An additional injection of about 100 GPM would be provided by the CRD hydraulic system. From Eqn. (B.8), the steady state reactor thermal power would be 28.6%.

Although the reactor thermal power is 28.6% with the water makeup provided by automatic actuation of the high pressure injection systems, the percentage of full power steam flow delivered to the pressure suppression pool would be somewhat less. To verify this, a simple expression for the steam flow from the reactor vessel as a percent of normal full power can easily be developed.

At steady state, the mass flow from the reactor vessel is equal to the mass injection rate $\dot{M}_W$. Steam flow at 100% power is 13.381×10^6 lb/h. If we assume that the enthalpy of the steam leaving the reactor vessel under ATWS conditions is the same as the enthalpy of the exiting steam during full-power operation, then the ATWS power expressed as a percentage of full power is

$$P_p = 100 \times \frac{\dot{M}_W}{13.381 \times 10^6} \% \qquad (B.9)$$

Equation (B.9) can be converted into a more useful form by use of Eqn (B.7). Then

$$P_p = 3.73 \times 10^{-3} F_W \% \qquad (B.10)$$

where

P_p = power delivered to the pressure suppression pool as a percent of the power exiting the reactor vessel during full power operation.

F_W = Injection rate, GPM.

Continuing the previous example, Eqn. (B.10) predicts that with a combined HPCI, RCIC, and CRD hydraulic system injection of 5700 GPM, the power delivered to the pressure suppression pool is 21.3% of the power exiting the reactor vessel under normal full-power operating conditions. Actually, the percentage would be slightly less because the steam enthalpy at 1120 psia [Eqn. (B.2)] is slightly less than the enthalpy at full power which is 1191.6 Btu/lb at 1020 psia.

Comparison of Eqn. (B.10) with Eqn. (B.8) reveals that the percent of full power delivered to the pressure suppression pool under MSIV-closure initiated ATWS conditions is about three-fourths of the percent of reactor thermal power. This will always be true because of the additional sensible heat required to increase the temperature of the incoming makeup water to saturation. Under normal operating conditions, feedwater enters the reactor vessel at a temperature of 377°F* whereas under MSIV-closure initiated ATWS conditions, the makeup water enters the reactor vessel at a temperature of about 90°F.

*The rated thermal power of 3293 MW(t) is based on this.

B.3 The Case with Known Reactor Vessel Water Level

The BWR Owners' Group Emergency Procedure Guidelines (EPGs) do not direct the operator to maintain a specified rate of reactor vessel injection under ATWS conditions but rather require the operator to maintain an indicated reactor vessel water level (at the level of the top of the active fuel in the core). Thus the analytical problem is greatly expanded from the simple exercise described in Sect. B.2 to a complex challenge in which the injection rate necessary to maintain the specified water level in the reactor vessel must be calculated. This can only be done by first calculating the reactor thermal power from detailed considerations of the conditions within the reactor vessel. Once the percent (P_R) of full power is known, Eqn. (B.8) can be recast in the form

$$F_W = 199.20 \; P_R \text{ GPM} \tag{B.11}$$

and solved for the required injection rate.

B.4 Conclusions

1. Given an ATWS situation in which the reactor core is capable of unrestricted power operation, the steady state power depends only on the injection rate [Eqn. (B.8)].

2. Under ATWS conditions, the core thermal power expressed as a percent of the normal full power [Eqn. (B.8)] will always be greater than the power exiting the reactor vessel expressed as a percent of the power exiting the reactor vessel during normal full power operation [Eqn. B.10)]. This is because of the requirement for additional power expenditure within the reactor vessel to heat the makeup flow taken from the condensate storage tank.

3. Since it is known that, with all four RHR system heat exchangers in operation, about four percent power can be removed from the pressure suppression pool while keeping the pool temperature at about 200°F, it is reasonable to ask why the instructions to the operator do not merely require him or her to maintain injection at a rate of about 1100 GPM, which, from Eqn. (B.10), would result in the injection of about four percent power into the pool.

4. The answer is that the resultant reactor vessel water level is not known if the operator is simply instructed to maintain a certain injection rate. For example, an injection rate of 1100 GPM might well result in an ATWS situation in which a substantial portion of the upper core is uncovered while significant power generation continues in the lower core.

5. The BWR Owners' Group EPGs simply direct the operators to maintain the indicated reactor vessel water level at the top of the active fuel. This, of course, is to ensure that core uncovery does not occur while still maintaining the reactor vessel injection rate as low as possible.

6. The seemingly simple shift of the operator control parameter from the injection rate to the indicated reactor vessel water level greatly complicates the calculation of the steady state power. This is because the actual water level would differ from the indicated level and because the core thermal power must now be calculated from detailed consideration of the conditions within the reactor vessel.

ORNL-DWG 84-9497

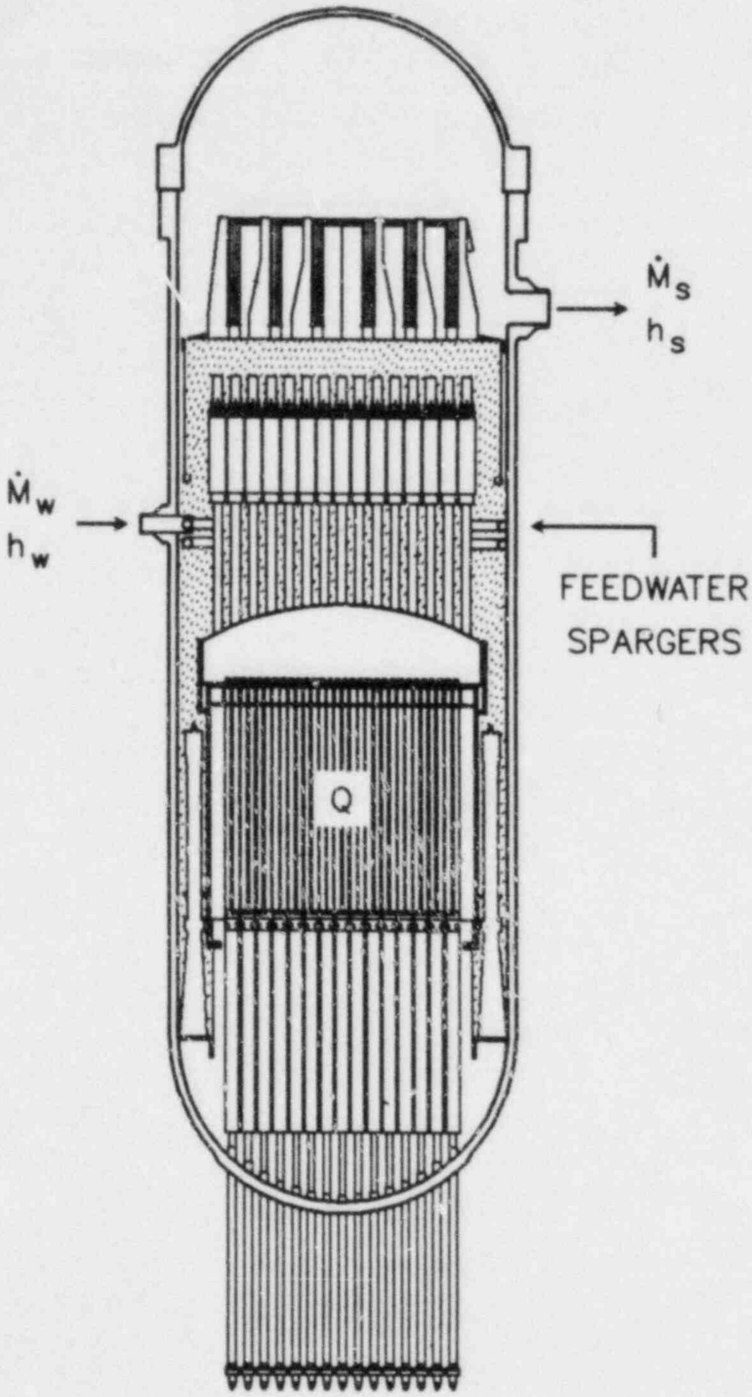

Fig. B.1. Identification of terms used in the calculation of steady state power.

APPENDIX C

Engineering Physics and Mathematics Division

PRELIMINARY HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS

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PRELIMINARY HUMAN FACTORS REVIEW FOR SEVERE ACCIDENT SEQUENCE ANALYSIS

ABSTRACT

Human factors considerations associated with operator performance are assessed for the Anticipated Transient Without Scram (ATWS) at Browns Ferry Nuclear Power Plant Unit 1 (BF1). Although human factors problem identification is moderated by the current transition to symptom-based EPGs, issues addressed include human engineering deficiencies in control room design, and human reliability of critical operator actions. Analyses are somewhat cursory due to multiple objectives of the study, but they do demonstrate the utility of human factors research methods. Critical operator actions identified in the EPGs as related to ATWS are qualitatively assessed in terms of expected performance and constraints to success. A detailed task analysis was completed for several of these actions, and a quantitative human reliability analysis was performed. Human factors research needs for ATWS are identified and reflect broader recommendations supporting further involvement with SASA studies.

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1. INTRODUCTION

The purpose of the human factors review for the Severe Accident Sequence Analysis (SASA) program is to support SASA analysts by systematically identifying and assessing salient human factors issues in the BWR Anticipated Transient Without Scram (ATWS). Through a plant-specific analysis of the Browns Ferry Unit 1 (BF1) ATWS,[1] this study serves as a demonstration of contributions from human factors research to SASA efforts. Human factors issues addressed in this review include operator reliability in performing safety-related actions, human engineering analysis of control room design, and types of procedures. Operator training for severe accidents and computer-based operator aids were also recognized as potentially important factors shaping operator performance.

Preliminary assessments of human factors issues are reported in this appendix to support the SASA evaluation of the BWR ATWS. The analysis includes a description of critical operator actions affecting the ATWS sequence and how these actions may be modified by human factors problems. Identification of issues in operator performance, and development of a system/task data base using the BF1 control room simulator, were conducted by an integrated team from the ORNL SASA project and the ORNL Reliability and Human Factors Group. More comprehensive documentation of human factors analyses will be reported in a separate technical document upon completion of the review.

The selection of human factors issues studied was streamlined to accommodate objectives and constraints of the program. Multiple objectives required: first, review of operator actions from initiation of the transient up to core damage (front end), and, second, assessment of actions during accident management involving mitigation of core damage (back end). The back end of the accident is to receive major emphasis in the human factors study. This appendix discusses the approach, analyses, findings and recommendations for the human factors review of the front end phase. Several cross-references are included to sections of the ORNL ATWS report. Analysis of the front end required extensive coordination of time and level of effort with SASA analysts. Considering that emergency procedures for BF1 were being changed to symptom-based procedures and that these procedures are still being reviewed for possible modification, the front end analysis was constrained to preliminary evaluations using best available information.

2. HUMAN FACTORS ISSUES IN OPERATOR PERFORMANCE

Human factors research in nuclear power plant operations addresses an array of issues related to operator performance. During familiarization with BF1 ATWS sequences juxtaposing automatic system responses with operator actions, two human factors issues were identified and are discussed in this section. These issues include emergency procedures and a human engineering analysis of control room design.

2.1 BF1 Emergency Procedures

At the time of this study, the emergency procedures used at BF1 were undergoing a transition from event-based Emergency Operating Instructions (EOIs) to symptom-based Emergency Procedure Guidelines (EPGs) developed by the BWR Owners Group. Event-based procedures require operators to first diagnose the type of transient before taking corrective actions. With symptom-based EPGs diagnostic efforts are minimized such that operators selectively detect and attend to critical safety functions that are off-normal. The Tennessee Valley Authority (TVA) is currently assessing the compatibility of the technical contents of the EPGs with BF1 system design and safety analysis.

The development of symptom-based procedures was an attempt to reduce the cognitive workload of control room operators in diagnosing the type of transient. Through use of the EPGs during a transient it is intended that operators verify the adequacy of critical safety functions. One advantage of event-based procedures, however, is that operators may immediately relate causes and consequences of off-normal conditions and subsequently directly act to mitigate accident progression.

SASA analysts have made the recommendation in Sect. 5.1 of the ORNL ATWS report that the emergency procedures for ATWS be separated from the EPGs. The human factors analysis assisted in defining some of the problems operators may experience with the current structure of the EPGs. One of these problems is that certain operator actions called for in response to ATWS are substantially different from actions appropriate to other accidents. Some of these actions are also contrary to operational practices on which operators are trained. One example related to ATWS is the instruction in the EPGs to lower and maintain vessel level at the top of the fuel in order to reduce power. Under other accident conditions, low vessel level would be an off-normal condition and the EPGs would instruct operators to restore vessel level to within more acceptable bounds.

From a human factors standpoint, the structure of the EPGs presents some difficulties for operators in relation to ATWS. However, the solution proposed by SASA analysts to separate those instructions relevant solely to ATWS may or may not be entirely satisfactory. Operator performance during a transient would be based on several factors including training and operator aids, such as the Safety Parameter Display System (SPDS), in addition to procedures. These factors and others should be considered across a range of accidents to optimally guide operator response before targeting the restructuring of procedures to address problems related to one specific accident sequence.

Several operator actions identified in the EPGs as critical to the progression of ATWS are examined in some detail and results of these analyses are presented later in this appendix. The timing of this study vis-a-vis ongoing adaptation of the EPGs for BF1 precluded an extensive assessment of the EPGs using NRC human factors guidelines for evaluation of procedures.

2.2 Human Engineering Analysis of Control Room Design

A human engineering analysis of control room design concerns the functional layout of controls and displays comprising the man-machine interface. On the one hand, this study did not intend to undertake a comprehensive human engineering assessment of the BF1 control room using NRC guidelines. On the other hand, several human engineering issues were identified during simulator exercises. These exercises provided input to both the human factors analysis and the SASA analysis. Simulator exercises were conducted and videotaped to provide a record of operator actions during runs of different ATWS sequences. Exercises were held on two occasions using two BWR SRO-instructors as operators. On both occasions an instructor was furnished by TVA and the second operator was from the ORNL human factors project team. The following discussion is based on instructors' comments and analysts' observations resulting from these exercises. The three human engineering issues related to ATWS included reactor level control, reactor pressure control, and manual control rod insertion.

2.2.1 Reactor level control

During an ATWS, operators monitor reactor vessel level and manually adjust coolant injection systems based on displayed level information. The problem is that, depending on their type, level indicators may be inaccurate or have insufficient range. Operators basing their actions on these displays may erroneously misjudge actual level. An additional problem is that some level indications, which do have sufficient range, are located on panels located away from the controls for coolant injection systems. Another operator must interrupt his work to read and communicate level information from these particular displays.

There are four vessel level monitoring systems with ten total indicators in the BF1 control room. Types and function include, first, narrow range GEMACs which cover the range from 528 to 588 inches (BF 0 to +60 inches). There are three of these sensor systems in the control room and one of any two sensor outputs is fed to a permanent recorder. The narrow range sensors are used for normal operation in both manual and auto control modes.

Second, wide range YARWAYS cover the range from 373 to 588 inches (BF +60 to -155 inches) and are used in off-normal conditions. There are two of these systems and they are not fed to a recorder.

Third, post-accident flooding range/shroud level range sensors cover the range from 260 to 560 inches (BF -100 to +200 inches). There are two of these systems and these sensors are used mainly in conjunction with the emergency core cooling systems. There is a recorder indication in the range of 360 to 460 inches (BF 0 to +100 inches). The post-accident flooding range and the shutdown flooding range systems are "cold" calibrated for use when the reactor is in or near cold shutdown temperatures. This predicates some type of variable normalization or correction factor which the operators must apply when attempting to monitor reactor level with the reactor at power.

Fourth, shutdown flooding range indication has one sensor and it covers the range from 528 to 928 inches (BF 0 to +400 inches). This instrument monitors level when the total vessel is required to be flooded.

One of the design problems is the lack of reliable information on reactor level. The wide and narrow range monitors are calibrated "hot" against various operating temperatures and therefore give reliable level information during an ATWS. However, none of the monitors allow level monitoring at or slightly below the top of the active fuel. The wide range "Bottoms-Out" at 373 inches which corresponds to 13 inches above the active fuel. During the ATWS, the operators are forced to use the post-accident flooding range system. Since this system is cold calibrated, however, level information will be unreliable and will constrain operator performance in maintaining water level close to the top of the active fuel in accordance with the EPGs.

A second design problem is related to level monitoring. Operators are trained to use the narrow range and then shift to the wide range monitors in off-normal conditions. In the ATWS, the lead operator (Operator #1) would be attempting to control the reactivity of the unit by manually inserting control rods and injecting boron via the SLCS. The second operator (Operator #2) would likely use the narrow/wide range indications as long as they supply needed level information, which during ATWS should be a very short period in duration. Both of these systems are physically displayed within the control room at distances from approximately 20 to 35 feet from the controls for the SRVs and coolant injection systems. The specific difficulty is that Operator #2 who controls coolant injection systems has to heavily rely upon Operator #1 for reading and communicating the level values from the wide range monitors. This interrupts the work of Operator #1 and adds to his already apparently high workload. This increase in workload also raises the possibility of display reading and communication errors.

2.2.2 Reactor pressure control

The operator may be hindered during an ATWS in attempting pressure control by, among other concerns, not knowing if the SRV being manually opened is already automatically activated. This is because no auto SRV position indication is located adjacent to manual SRV controls.

The BF1 unit has thirteen safety relief valves distributed among four main steam lines exiting the pressure vessel. These valves have two functions, to protect against overpressure transients, and to depressurize the reactor when required during off-normal conditions. Any of the valves can be opened manually with switch action by the operators and will be automatically opened by steam pressure once their set points are exceeded. The valve set points range from 1105 to 1125 psig.

Six of the SRVs are dedicated to the automatic depressurization system (ADS). This system initiates on high drywell pressure and low vessel water level. The ADS autotimer has a two minute cycle. If the low level signal does not clear, or the operator does not recycle the timer prior to the end of the two minutes, all six valves open. Once ADS activates the six SRVs, the SRVs will not close until reactor pressure drops to about 20 psi above drywell pressure or the operator manually resets the ADS timer.

The design problem is an absence of any individual indication of auto SRV activation adjacent to the SRV controls. Experienced operators may hypothesize that SRVs are automatically cycling based on pressure, flow, and other monitors. There are acoustic monitors for the SRVs, but these are displayed at the rear of one of the back panels. The only front panel indication for the operators is the switch handle mode and a small light adjacent to each switch. This light tells the operator only that the valve solenoid has been energized, not that the valve has actually opened. In summary, the operator is not provided timely information about valve position unless he takes several seconds to walk to the back panel to observe the acoustic monitors.

The potential error from this design problem is that the operator may open a valve which is already in the blowdown mode from overpressure. This action of trying to open an SRV, then, would not add to a further decrease in pressure. An additional problem which complicates the ATWS sequence is that he may attempt to close a valve which has actually stuck open. The operator would then need to examine the acoustic monitors, along with other relevant instrumentation, to diagnose this failure.

2.2.3 Manual control rod insertion

Two human engineering problems related to manual control rod insertion were identified. First, the switch to insert rods is a multifunction deadman lever with which errors of commission may occur. Second, positioning errors may result while turning the rod sequence selector switch until the desired rod select pushbutton is illuminated.

Failure of control rods to insert automatically during ATWS should be followed by operator attempts to manually scram the rods according to the EPGs. The multifunction deadman switch constrains operator mobility and may contribute to error. Once the operators have diagnosed the ATWS and have also experienced manual scram failure, the EPGs instruct them to manually insert the control rods one at a time. The process takes about one minute per rod. The procedure requires switching to manual insertion effectively bypassing the rod sequencing and rod blocks. The operator then reads from the rod pattern charts to select and insert high worth control rods.

A design problem identified is that the switch which inserts the rods is a multifunction spring-loaded deadman lever which also withdraws rods. The operator has to continually activate and overpressure the spring to move a rod. He is limited to the reach of his arms and cannot change position more than a few feet in either direction of the switch. The operators on the simulator were observed making commission errors in selecting the incorrect mode of the control switch. They did in every case recover and place the switch in the correct mode within one second.

The second problem concerns potential errors in positioning the rod sequence selector switch to enable the desired rod select pushbutton. During the ATWS it is desirable to insert high worth control rods in the center of the core to achieve the quickest reduction in reactor power. To insert the high worth control rods requires the operator to deviate from the pre-programmed rod sequence. The Rod Worth Minimizer (RWM) can be easily bypassed with a keylock switch in the control room. However, the Rod Sequence Control System (RSCS) cannot be bypassed in the control room. The control room operator must communicate with an auxiliary operator in the instrument room to bypass rod groups as necessary, delaying control rod insertion. The operator must also manipulate two control room switches for RSCS to insert control rods because the Reactor Manual Control System (RMCS) imposes RSCS rod blocks when the emergency insert is used.

The RSCS switches must be positioned to permit selection and movement of the desired control rod. A problem is the need to position the rod sequence selector switch when changing from one rod group to another which increases the time delay for rod selection and insertion. The operator manipulates the rod sequence selector switch until the desired rod select pushbutton is illuminated. The rod select pushbuttons are small and lighted from the back. This switch positioning problem is further complicated by the distant location of the switch which makes it difficult for the operator to read the rod select pushbuttons while manipulating the switch. This may lead to a number of errors in positioning the rod sequence selector switch until the desired rod pushbutton is selected.

3. ASSESSMENT OF OPERATOR ACTIONS DURING ATWS

The purpose of this section is to discuss the approach and results of the human reliability assessment of operator actions during ATWS. The section begins with the identification of critical operator actions for review, followed by a qualitative analysis of these actions. In addition, a quantitative human reliability analysis (HRA) was completed for several of these actions. Rather than assess operator actions throughout the ATWS the overall analysis was limited to only those operator actions in the EPGs judged to be most critical to the sequence of ATWS. Primary emphasis concerning the human factors assessment was on operator actions contained in the EPGs, although input to the HRA included data collected through a task analysis of operator actions following the EOIs. It was assumed that these latter actions called for by both the EPGs and EOIs would be performed by operators in a closely similar manner. This similarity is held to support the assumption that results of the HRA, while based on the EOIs, may be relevant to the EPGs.

3.1 Identification of Critical Operator Actions

The identification and selection of critical operator actions was coordinated with SASA analysts based on an evaluation of key branching points in the ATWS sequence. Inputs to the selection process included: (1) examination of the EPGs, (2) consideration of operator actions included in computer codes used for systems analysis, (3) review of an Operator Action Event Tree (OAET) developed for ATWS, which identifies major branches in the sequence of key operator actions necessary to mitigate the accident,[2] and (4) critical review of operator actions observed during simulator exercises of ATWS. The six operator actions selected for analysis included:

(1) Selection and manual insertion of individual control rods given failure to scram (refer to Section 4.1.1 of the ORNL ATWS report).

(2) Verification of conditions for use of the Standby Liquid Control (SLC) system and initiation of poison injection into the vessel (refer to Section 4.1.1 of the ORNL ATWS report).

(3) Initiation of pressure suppression pool (PSP) cooling through residual heat removal (RHR) system (refer to Section 4.1.4 of the ORNL ATWS report).

(4) Operator control preventing overpressure of the vessel by manually opening SRVs before 1105 psig is reached for auto actuation (refer to Section 4.1.3 of the ORNL ATWS report).

(5) Operator control of coolant injection systems to lower and maintain reactor vessel water level at the top of active fuel (refer to Section 4.1.2 of the ORNL ATWS report).

(6) Depressurization of the reactor vessel in accordance with the PSP heat capacity temperature curve (refer to Section 4.1.3 of the ORNL ATWS report).

3.2 Qualitative Review

At the time of this writing TVA was continuing to modify the EPGs in accordance with BFl plant design. This imposes some constraints to the assessment of operator actions. A preliminary Operating Sequence Overview, which is an NRC task analysis technique,[3] was developed from review of the EPGs and is shown in Fig. 1. The identification of major operator actions is similar to those identified in the ATWS OAET reported in Reference 2.

Plant: BFNP

Operator Function/Subfunction: Supervise and Control Plant Operations/ Mitigate the Consequences of an Accident

NSSS/Type: GE/BWR

Operating Sequence ID: 7

C.R. Type: Multiple

Operating Sequence: Anticipated Transient Without Scram, Following MSIV Closure

Initial Conditions: Plant operating at 100% power and all systems in normal line-up.

Sequence Initiator: MSIV Closure

Progress of Action: The crew acknowledges the closure of the MSIVs, and recignizes that the reactor did not scram. All attempts to manually scram the reactor fail. Control rods are manually inserted using reactor manual control system. The reactor recirculation pumps trip automatically on high reactor pressure. Level rapidly decreases due to coolant loss through the safety/relief valves, and HPCI and RCIC automatically initiate on low level. The operators verify that conditions require initiation of standby liquid control and begin injection. Concurrently, coolant injection is manually throttled so that level is lowered and maintained at the top of active fuel to reduce power. Manual control rod insertion continues using RMCS.

The residual heat removal system is placed in the suppression pool cooling mode. Suppression pool temperature is monitored to maintain the torus heat capacity temperature limit. Reactor pressure is limited by automatic/manual opening of safety/relief valves, and if SRVs are cycling or the RPV must be depressurized SRVs are manually opened until pressure drops.

Following injection of boron by SLC according to technical specifications, water level is raised using coolant injection systems to circulate poison through the core.

The Shift Supervisor declares an alert, and notifies appropriate on-site personnel.

Final Conditions: The plant is in hot shutdown with torus cooling in operation. Reactor level is being maintained using RCIC

Major Systems: Reactor Recirculation, Reactor Manual Control, Main Steam, Residual Heat Removal, RHR Service Water, Nuclear Instrumentation, HPCI, RCIC, SLC, Rod Worth Minimizer, Rod Sequence Control System, Primary Containment Isolation System, Water Level Instrumentation.

Fig. 1. Operating sequence overview with EPG-based operator actions.

Operator actions to insert control rods are critical to shutting the reactor down in the event of failure of automatic systems to scram the reactor. A considerable amount of time would be required to manually insert all withdrawn control rods. However, through expeditious selection of high worth rods and inserting these first the operator can reduce power at a moderate rate. The cognitive and physical requirements of this task are likely to require the full attention of one operator. Once the power level is considerably reduced, operator workload may permit handling other tasks in the immediate area of the console. The operator is tied to the switch for inserting the selected control rod, as it is a deadman lever. The two BWR SRO-instructors used in the simulator exercises reported an apparently accelerated learning curve in selecting higher worth rods over practice runs. The instructors also reported some concern about introducing uneven flux in certain areas of the core when a reasonable rod pattern was not maintained.

Checking conditions and initiating SLC injection are critical tasks insofar as poison injection satisfies the functional requirement of inserting negative reactivity to shut the reactor down. Poison injection in a BWR is also controversial with regards to lost plant availability during lengthy cleanup. In general, the execution and timing of this task are subject to question. The procedures relieve the operator of some of the burden in this decision-making process. When either of the conditions listed in Section 4.1.1 of the ORNL ATWS report exist, the operator is required to initiate SLC. This action may be taken by the operator in the absence of the Shift Engineer. Even with the procedural requirement, however, the operators may try other alternatives for manually inserting control rods before initiating SLC injection. The uncertainty associated with this task should be incorporated as part of the HRA.

Initiation of PSP cooling is important for protecting primary containment integrity in the absence of the main condenser following MSIV closure. Reliability issues concern initiation of PSP cooling using both RHR loops, and the timing of operator actions in relation to PSP temperature and rate of temperature increase. The timing of this task is especially critical when the operator must concurrently perform other important tasks. For example, control of reactor pressure and water level may delay initiation and completion of PSP cooling. In addition, some delay results from the required continual operation of the suppression pool test line valve. When the deadman control switch for this valve is released, valve motion stops. The operator must return to the control switch to continue and complete valve motion if he is drawn away to perform other essential tasks.

Actuation of SRVs to prevent vessel overpressure necessitates monitoring of pressure displays. Operators may perform this task either before pressure reaches 1105 psig or after pressure reaches automatic SRV operation levels (1105 to 1125 psig). When the operator does not manually open an SRV until 1105 psig or higher is reached in the vessel, he may unknowingly be attempting to open an SRV already open automatically and thereby add nothing to pressure control (refer to Section 4.1.3 of the ORNL ATWS report).

Based on the EPGs, the operator should lower and maintain the reactor vessel water level at the Top of Active Fuel (TAF) while sodium pentaborate solution is being injected. Upon injection of a predetermined amount of poison, the operator is to restore the water level to its normal operating range, thereby mixing poison throughout the core and bringing the reactor subcritical. As a preliminary test of procedures, these steps were included during the simulator exercises. The instructors involved in these exercises reported an apparent increase in success across successive trials in maintaining level at TAF during poison injection. However, several considerations limit confidence in inferences drawn from such preliminary observations. Among these considerations are possible limitations within the computer software supporting the BF1 simulator as reported by TVA, and the validity of results based on only two SRO-instructors using draft procedures.

Some deficiencies became apparent during the simulator experiments related to reactor water level instrumentation effecting operator performance in maintaining level at TAF. The operator controlling reactor water level using RCIC and HPCI would tend to frequently monitor the level instruments displayed with the HPCI/RCIC controls. This operator would also tend to call on the reactor operator for level readings from the emergency range YARWAYS. Deficiencies with the level instruments in close proximity to the HPCI and RCIC systems are that they are uncompensated and calibrated to read accurately only when the reactor is depressurized and the recirculation pumps are tripped. During an ATWS these instruments may read as much as 43 inches lower than actual reactor water level (refer to Chapter 4 of the ORNL ATWS report). Some of these level instruments also provide insufficient level indication since the wide range level instrument's bottom end is 13 inches above TAF. There is the possibility of operator error in converting the reading from wide range instruments to the post accident flooding range instrument reading, since each instrument range has a different reference zero. This type of error was identified in the analysis of the TMI accident, and recommendations have been made in the past to correct this problem.

The location of the emergency range instruments presents some difficulty to the operators. The operator controlling the reactor water level using HPCI and RCIC must depend on the reactor operator to call out readings from the YARWAYS because of the distance between the indicators and the controls for these systems. The indicators are located on the reactor panel to provide level indication when operating the feedwater system with reactor level below the normal range. These indicators should be retained in their present location and could be supplemented with additional instrumentation visible from a distance.

An additional difficulty with level control concerns use of high pressure injection systems. SASA calculations show some ATWS cases in which the pressure suppression pool (PSP) level increases to the limit for HPCI suction shift from the condensate storage tank to the PSP. Subsequently, the HPCI pump fails from high lube oil temperature unless the operator manually trips the pump. An anecdotal observation from the simulator exercises was an operator error of commission involving manually shifting suction of RCIC to the PSP following automatic HPCI suction shift, leading to failure of both systems.

The last operator action of concern in following the EPGs involves the situation in which the PSP temperature has increased to a point on the PSP heat capacity temperature curve that vessel depressurization is prescribed. Human engineering deficiencies in SRV automatic position indication have been previously described. In general, manual depressurization is a difficult task when the vessel is at high pressure. Especially important is the ability of operators to execute this procedure while anticipating reactor response to low pressure coolant injection. Observations of simulator exercises involving initiation of ADS showed injection control to be a severely difficult and apparently unmanageable task for operators in terms of uncontrolled cycling of low pressure injection followed by pressure and power spikes. Avoidance of power and pressure spikes should be practiced through simulator training involving operation of low pressure injection systems. A set of recommended operator actions for controlling low pressure injection following vessel depressurization is described in Section 4.1.2. of the ORNL ATWS report. The EPGs may need to better structure a series of steps for increasing operator reliability in controlling low pressure injection systems to avoid power and pressure oscillations.

An ancillary issue is related to controlling PSP temperature using the RHR system. The simulator experiments revealed difficulties in the operation of PSP cooling when reactor water level is lowered in accordance with the EPGs. Two valve interlocks will cause an isolation of the PSP cooling flow path unless the operator takes action to prevent the automatic valve closure. The first isolation occurs at the reactor level where the LPCI initiation occurs (476.5 inches). The second isolation occurs at two-thirds core coverage (312 inches). These isolations of PSP cooling are intended to prevent diversion of LPCI for containment cooling during a LOCA. However, during an ATWS reactor water level is to be controlled at or near the TAF. The isolation of PSP cooling would likely divert the operators' attention away from controlling coolant injection. Training and procedures should emphasize the need to bypass the two-thirds core coverage interlock and place the containment spray valve select switch in the SELECT position prior to reducing water level to the top of the core.

3.3 Quantitative HRA

Presentation of the HRA is divided into three sections. First, a task analysis of critical operator actions during ATWS is reported. Second, the steps in conducting the analysis using the Technique for Human Error Rate Prediction or THERP[4] are summarized, along with a listing of the quantitative human reliability estimates. The use of THERP was primarily relevant to estimating operator reliability during particular tasks selected for analysis on the basis of their importance to ATWS. Third, results of the analysis using the Operator Performance Simulation (OPPS) computer model[5] are described. The use of OPPS to supplement the THERP analysis provided a time-reliability estimate across all operator actions during ATWS.

3.3.1 Task Analysis

An input requirement to THERP is a task analysis providing systematic descriptions of operator actions. The task analysis of critical operator actions used in this review followed the standard NRC task analysis format[3] which describes tasks at three levels of detail. At a high level is the Operating Sequence Overview identifying the general progression of actions by plant systems and operators. The ATWS Overview incorporating the EPGs was previously shown in Fig. 1. At a middle level of detail is the Task Sequence Chart identifying the normative ordering of tasks, the purpose of operator actions, cues that initiate the task, technical specifications of procedures, and plant systems involved in the task. The most specific level of detail is the Task Data Form (TDF) listing all discrete human actions comprising the task. A sample TDF is shown in Fig. 2 for initiation of PSP cooling and illustrates types of information collected. TDFs were completed for the four tasks selected for HRA.

Inputs to the task analysis were:

(a) BF1 procedures including EOIs, EPGs and general operating instructions.

(b) Videotapes of BWR SRO instructors conducting ATWS exercises on the BF1 control room simulator.

(c) Computer records of operators' switch manipulations during the simulator exercises collected through the Performance Measurement System.[6]

(d) Expert judgment of operator actions using a task analysis panel of an SRO instructor, an SRO-SS from Oak Ridge National Laboratory, and a human factors specialist.

The task analysis resulted in a normative description of actions transcending idiosyncratic performance characteristics of the SRO instructors on the simulator.

3.3.2 Human reliability estimates using THERP

THERP is a recognized and accepted technique for assessing operator reliability in nuclear power plant operations. It has undergone considerable development by Swain and his associates at Sandia National Laboratory.[7] THERP is a technique in which operator behaviors comprising a task are identified through a task analysis. These actions are assigned nominal human error probabilities (HEPs) which are modified by performance shaping factors (PSFs), and the final success probability is then calculated. The task analysis of operator actions must be at a level compatible with HEP data bases. HEPs reported in the THERP human error data base (Chapter 20 of Ref. 4) have been subjected to some criticism dealing with their adaptation from a non-nuclear power plant operator source. However, the final version of this data base has reportedly been supplemented with HEPs from relevant sources, and other human error data bases are also available such as those developed through simulator experiments.[5,8] An additional issue in the use of THERP is the matching of task analysis data with descriptions of operator actions listed in the human error data base.[9] Depending upon the task being assessed by

TASK DATA FORM (DESCRIPTIVE)

Page No. 1 of 3

PLANT IDENTIFICATION

Plant Name: Browns Ferry
Unit Number: 1
NSSS Vendor: General Electric
A-E: Utility
TG Vendor: General Electric
CR Type: Multiple
OL Date:

TASK IDENTIFICATION

Operating Sequence: Anticipated transient without scram
Operating Sequence ID: 7
Operator Function: Supervise and control plant operation
Operator Sub-function: Mitigate consequences of an accident
Comments:
CUE: Procedure

Task Statement: Initiate suppression pool cooling
Task Purpose: To limit suppression pool temperature
INPO Task Code:
Task Sequence No.: 20
Task Duration: 2 minutes 41 seconds
Procedures: GoI-100-1 Section VII emergency shutdown with MSIV closure
Data Collected at: Simulator

JOBCAT	JLOC	Behavior		Object of Action						MEANS	Communication Link		
		TIME	VERB	COMPONENT	PARAMETER	STATE	OTHER OBJECT	PLANT SYSTEM	INPO EQUIV		RJC	RLOC	CONTENT
R02	12	12:43 12:44	Positions	Pump	Power	On		RHR		Discrete Control			
R02	12	12:43 12:44	Positions	Pump	Power	On		RHR		Discrete Control			
R02	12	12:43 12:44	Observes	Pump	Power	On		RHR		Indicator Light			
R02	12	12:43 12:44	Observes	Pump	Power	On		RHR		Indicator Light			
R02	12	12:47 12:51	Positions	Valve	Position	Open		RHR		Discrete Control			
R02	12	12:49 14:38	Positions	Valve	Position	Open		RHR		Discrete Control			
R02	12	12:54 12:54	Positions	Pump	Power	On		RHRSW		Discrete Control			

TX-5197

Fig. 2. Sample Task Data Form.

THERP, the reliability between analysts in selection of HEPs for operator actions may need to be reviewed to ensure the accuracy of the analysis.

Nominal HEPs were taken from the THERP human error data base reported in Chapter 20 of Ref. 4. Assignment of HEPs was coordinated between authors to verify reasonableness of their selection for matching task analysis data.

One PSF assumed to bear on operator performance during ATWS was stress. The effect of stress on performance was assumed to weigh more significantly on the initial cognitive determination of whether to perform the task given the abnormal condition of the plant. That is, stress was held to more likely distract the operator from executing the task but once the task is undertaken operator competence overrides adverse effects from stress. Attributing stress effects to decision-making seems a better reflection of the complex and confusing stimuli with which operators are attempting to filter, but once a course of action is selected the relative affects of stress are reduced. This description parallels the distinction made in the THERP Handbook between dynamic decision-making tasks and step-by-step tasks. That is, HEPs are more heavily modified by stress for dynamic tasks.

HEPs were further modified from effects of dependence defined as the extent success on one action effects success on the subsequent action. Dependence was assessed using guidelines reported in Ref. 4.

Modified HEPs comprising complete success paths were used to calculate final task success probabilities. Only actions for which errors would contribute to system failure were included in the calculations. A sample THERP event tree for SLC injection is shown in Fig. 3 with HEPs adjusted according to the preceding discussion. Estimated failure probabilities are reported in Table 1 for the four tasks assessed by THERP. Uncertainty bounds (UCBs) are also reported reflecting best case (lower UCB) and worst case (upper UCB) performance. In most cases UCBs were calculated to show effects from stress on initiating execution of procedures under off-normal plant conditions.

Prevention of vessel overpressure by manual operation of SRVs has an estimated nominal HEP of 2.72E-02. This is interpreted as a probability that about three percent of the time when an operator should execute this operation he would fail to operate SRVs. The task extends over a considerable period of time starting shortly after initiation of this ATWS event when the MSIVs close and vessel pressure increases.

Manual insertion of control rods has an estimated nominal HEP of 1.82E-01, and requires careful interpretation. This HEP was calculated on the basis of selection of approximately twelve control rods inserted in such a pattern that power was reduced to less than one percent on the simulator computer and in combination with poison injection. The selection, insertion and position change verification of a single control rod has an estimated HEP of 9.48E-03 adjusted for dependence. Performance of the entire task, however, includes operation of the master group select switch used when the operator shifts from one group of control rods to another according to the pattern being developed for insertion of rods. Interpretation of the final task HEP must consider that there were 85 task elements included in the task. It is important to note that although dependence was factored in with failure probabilities, the

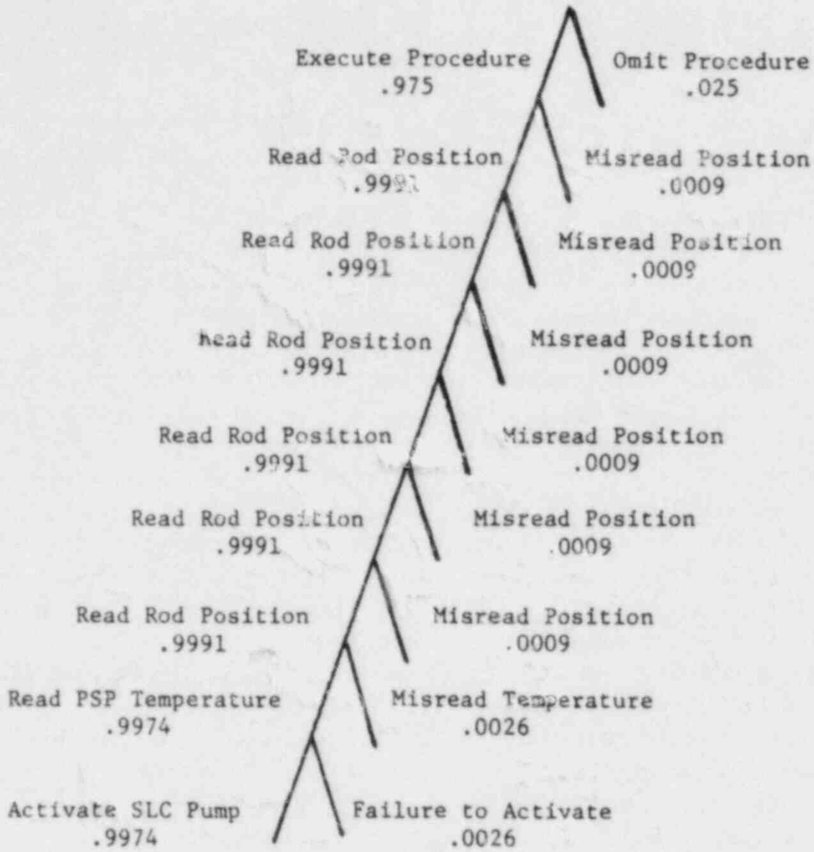

Fig. 3. HRA event tree for operator actions involving SLC injection.

Table 1. Human failure probabilities for selected tasks during ATWS

Task description	Nominal HEP	Uncertainty bounds Upper	Uncertainty bounds Lower
Manually operate SRVs before 1105 psig reactor pressure is reached	2.72E-02	2.61E-01	1.74E-02
Manual control rod insertion	1.82E-01	3.71E-01	1.63E-01
Initiate suppression pool cooling	1.27E-01	3.28E-01	3.92E-02
Verification of conditions for and initiation of SLC injection	3.69E-02	2.59E-01	1.47E-02

overriding significance of this task to mitigating the ATWS by bringing the reactor subcritical supports an assumption that most errors would be eventually, if not immediately, recovered by the reactor operator.

Operator initiation of PSP cooling has an estimated nominal HEP of 1.27E-01. A major contributor to operator error is whether the operator recognizes the increase of PSP temperature, including acknowledgment of the PSP high temperature annunciator within the first ten minutes of its initiation. THERP uses a time reliability distribution for assigning HEPs in situations involving failure to diagnose events. Within the first ten minutes of problem initiation the HEP is 0.1 which was used in calculating the nominal HEP, and from ten to twenty minutes the HEP for failure diagnosis is 0.01. This indicates that the operator is more likely to recognize the heatup of the PSP as more time passes. The upper UCB is based on a diagnosis failure during the first ten minutes and worst case high stress, whereas the lower UCB assumes less probable diagnosis failure and nominal high stress.

Use of SLC during ATWS has an estimated nominal HEP of 3.69E-02, an upper UCB of 2.59E-01, and a lower UCB of 1.47E-02. The complexities of this task include the considerable difficulty operators would have in deciding to execute the task and the high level of stress accompanying the decision. Based on these considerations it may be more appropriate to take the worst case scenario and use the upper UCB as a more conservative estimate.

3.3.3 OPPS time reliability curve

Supplementary assessment of operator actions throughout the ATWS was provided through use of the Operator Performance Simulation (OPPS) computer model. The OPPS model, developed in the Safety-Related Operator Actions (SROA) program,[5] simulates operator responses to transient conditions in a nuclear power plant. Results are in the form of a time reliability distribution. A major advantage of OPPS, as with other simulation models,[10] is assessing systematic variations in input and process conditions for subsequent effects on output variables. Computer models incorporate features pertinent to task performance and may include task, operator, time, and organization variables. The OPPS model was programmed using the SAINT simulation language and assumes that operator performance is guided by procedures. During an OPPS iteration, the simulated control room crew is timed for completion of branches through pre-alarm detection, event diagnosis, selection of procedures, execution of operator actions following procedure steps, execution of actions outside the control room, and assessment of recovery from errors of omission and commission.

Results of the OPPS analysis includes a time reliability distribution shown in Fig. 4. Curves are plotted by relative and cumulative frequencies based on 1000 iterations of simulated task performance. Performance time for completion of all required operator actions averaged 2005 seconds (33.42 minutes) with a minimum of 1382 seconds (23.03 minutes) and a maximum of 2629 seconds (43.82 minutes). The number of errors of omission averaged 3.68.

HISTOGRAM OF THE AVERAGE FIR STA STATISTIC FOR TASK 41 (STOP)

OBSV FREQ	RELA FREQ	CUML FREQ	UPPER CELL LIMIT
0	0.0	0.0	0.1200E+04
0	0.0	0.0	0.1260E+04
0	0.0	0.0	0.1320E+04
0	0.0	0.0	0.1380E+04
3	0.003	0.003	0.1440E+04
0	0.0	0.003	0.1500E+04
6	0.006	0.009	0.1560E+04
8	0.008	0.017	0.1620E+04
29	0.029	0.046	0.1680E+04
33	0.033	0.079	0.1740E+04
66	0.066	0.145	0.1800E+04
96	0.096	0.241	0.1860E+04
102	0.102	0.343	0.1920E+04
130	0.130	0.473	0.1980E+04
130	0.130	0.603	0.2040E+04
93	0.093	0.696	0.2100E+04
93	0.093	0.789	0.2160E+04
67	0.067	0.856	0.2220E+04
51	0.051	0.907	0.2280E+04
33	0.033	0.940	0.2340E+04
22	0.022	0.962	0.2400E+04
17	0.017	0.979	0.2460E+04
16	0.016	0.995	0.2520E+04
3	0.003	0.998	0.2580E+04
2	0.002	1.000	0.2640E+04
0	0.0	1.000	0.2700E+04
0	0.0	1.000	0.2760E+04
0	0.0	1.000	0.2820E+04
0	0.0	1.000	0.2880E+04
0	0.0	1.000	0.2940E+04
0	0.0	1.000	0.3000E+04
0	0.0	1.000	INF

1000			

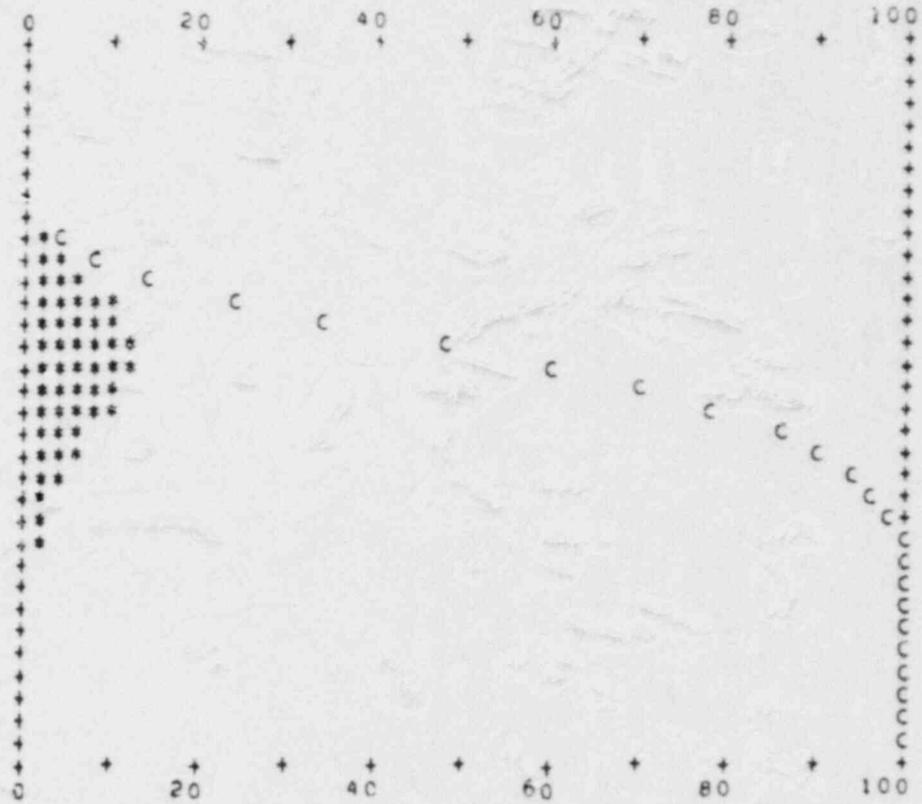

Fig. 4. OPPS time reliability distribution with relative and cumulative frequencies.

Inputs and assumptions to this OPPS analysis were that 105 control room switch manipulations are necessary (based on the task analysis) to mitigate ATWS, that no actions were required of auxiliary operators outside the control room, and that equipment delay time was embedded in the procedures. Regarding diagnosis of ATWS, branches selected were that annunciators indicate specific conditions rather than general alarms for identifying ATWS, that five indications are sufficient to diagnose the type of disturbance, and that operator diagnosis is terminated at the symptom level rather than extending to the root cause of rod failure to insert. Additional branches concerning planning and procedures were selected to reflect that procedures are written, are indexed, are memorized to determine immediate operator actions, and that the ATWS scenario is used in training.

While the OPPS model calculates an average simulated performance time of 33.42 minutes, not all safety-related actions must be completed within that time interval to ensure plant safety. Operators may complete more critical actions immediately following the transient and, upon verifying improvements in plant conditions, take additional time to complete remaining actions. In summary, the OPPS model provides an estimate of time reliability for assessing operator performance. The interpretation of its output is circumspect to input assumptions and limitations inherent to model design.

4. ASSESSMENT AND RECOMMENDATIONS

The work accomplished to date in this human factors review of ATWS at BF1 provides preliminary conclusions concerning operator performance and reliability, and serves as a demonstration of potential contributions to other SASA investigations. The review has assisted in the evaluation of ATWS by assessing effects of safety related actions and identifying human factors issues shaping operator performance.

Initial findings concern operator reliability in performing critical tasks. Effects of human engineering deficiencies in control room design and certain instructions contained in the symptom-based EPGs are also assessed. Tasks for which operator performance appears susceptible to certain types of error include:

(1) Selection of high worth control rods and manually inserting them requires considerable time and number of actions.

(2) Verification of conditions and initiation of SLCS injection presumes a complex decision which operators may defer for some period of time until after other means of achieving reactor shutdown are attempted.

(3) Initiation of PSP cooling is important in the context of the timing of the recognition of PSP temperature increase.

(4) Lowering and maintaining reactor vessel water level at TAF may be constrained by inadequate level indication.

(5) Following vessel depressurization, controlling low pressure injection systems is important to prevent oscillating pressure and power spikes.

The EPGs include a step for initiation of PSP cooling. The event-based EOIs do not include such a step. In using the EPGs, then, operator reliability in executing this task should be higher since relevant instructions specifically guide these particular actions.

Analysis of operator training for ATWS was limited in this review to informal interviews with TVA BWR instructors. In general, operators are trained for ATWS through a combination of classroom instruction and simulator exercises. This human factors assessment of issues in operator reliability, however, underlines many of the considerations included in a front-end training analysis related to severe accidents. Training for severe accidents should be based on probabilistic risk analysis (PRA) and SASA analysis and would be optimized through a structured approach using the Systems Approach to Training concept. Performance requirements would be identified using PRA and SASA studies leading to an identification of learning objectives to be addressed in classroom instruction and simulator practice.

There are three recommendations for control room modifications emerging from the human engineering analysis. The first recommendation concerns position indication of the SRVs corresponding to their automatic actuation. Operators are blind to their position unless they check acoustic monitors on a back panel (refer to Section 4.1.3 of the ORNL ATWS report). A status lamp would be sufficient to supply the necessary data to guide manual SRV actuation. The second recommendation concerns vessel level indication associated with HPCI and RCIC. These displays should be upgraded to allow greater operator control in lowering and maintaining level with TAF in accordance with the EPGs. A possible solution is to install a large digital indicator referenced to TAF and which can be read at a distance. The third recommendation concerns the multifunction deadman switch for control rod insertion. An apparent solution is that, when in the emergency manual insertion mode, the switch would have a momentary block. This would permit the operator to remove his hand from the switch and have a short period of time for other tasks.

Operator performance on level control would likely be more reliable if vessel-level indications were upgraded corresponding to information needs associated with the task. The complexities of this task should be fully explained to operators through specialized ATWS training. Classroom instruction should address steps in the EPGs involving lowering vessel level which seem contrary to the heavily emphasized goal of maintaining a normal high level. Operators should have simulator practice and undergo evaluation to ensure appropriate skills for safely lowering and maintaining level. This should follow the reported intentions of TVA to upgrade computer software supporting the simulator to increase its compatibility with the EPGs.

Further work in this human factors review of operator actions for mitigating ATWS should include additional analysis of the EPGs. However, the scope of the current study precludes more detailed assessments. On the one hand, SASA analysts have made a recommendation (see Section 5.1 of the ORNL ATWS report) that a separate procedure be written for the ATWS. On the other hand, the EPGs were developed to, among other reasons, guide operator actions so as to restore off-normal safety functions rather than deal with equipment failures. It is recognized that the EPGs may require some restructuring to make them easier to follow and more directly instruct the operator to take actions that are unique to the ATWS. Operator reliability in mitigating ATWS by following the EPGs should also be interpreted in the context of how other factors (such as training, operator aids, control room design, and management practices) may influence performance.

The remainder of this study, in fact the majority of effort, is addressing operator performance for mitigation of core damage as part of accident management. A functional classification is being developed identifying functions and performance requirements associated with accident management, including protection of plant safety equipment and processes and protection of the health and safety of personnel and the public.

The SASA program benefits from human factors analysis following incorporation of the operator in overall systems analysis. Operator errors influence the timing and sequence of deteriorating off-normal system parameters. The assessment of salient human factors issues provides means for reducing the potential for such error.

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Appendix D

ACRONYMS AND SYMBOLS

ADS	Automatic Depressurization System
ANS	American Nuclear Society
ANSI	American National Standards Institute
APRM	Average Power Range Monitor
ATWS	Anticipated Transient Without Scram
BAF	Bottom of Active Fuel
BCL	Battelle Columbus Laboratories
BNL	Brookhaven National Laboratory
BFNP	Browns Ferry Nuclear Plant
BWR	Boiling Water Reactor
CBP	Condensate Booster Pump
CFR	Code of Federal Regulations
CILRT	Containment Integrated Leak Rate Test
CP	Condensate Pump
CRD	Control Rod Drive
CRDHS	Control Rod Drive Hydraulic System
CS	Core Spray System
CST	Condensate Storage Tank
DF	Decontamination Factor
DHR	Decay Heat Removal
DW	Drywell
ECCS	Emergency Core Cooling System
EECW	Emergency Equipment Cooling Water
EPA	Electrical Penetration Assembly
EPG	Emergency Procedure Guideline
EOI	Emergency Operating Instruction
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
FW	Feedwater
GE	General Electric Company
GPM	Gallons per Minute
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
ID	Internal Diameter
INEL	Idaho National Engineering Laboratory
IORV	Inadvertently Open Relief Valve
IREP	Interim Reliability Evaluation Program
kPA	Kilopascal
LACP	Loss of AC Power
LDHR	Loss of Decay Heat Removal
LPCI	Low Pressure Coolant Injection Mode of the RHR System
LPECCS	Low Pressure Emergency Core Cooling Systems
LOCA	Loss of Coolant Accident
LOCA/OC	Loss of Coolant Accident Outside Containment
LOSP	Loss of Offsite Power
MARCH	Meltdown Accident Response Characteristics
MPa	Megapascal

MRI	Manual Rod Insertion
MSIV	Main Steam Isolation Valve
MWd/te	Megawatt Day per Tonne
MW(e)	Megawatt electrical
MW(t)	Megawatt thermal
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OI	Operating Instruction
ORNL	Oak Ridge National Laboratory
Pa	Pascal
PCV	Pressure Control Valve
PCIS	Primary Containment and Reactor Vessel Isolation Control System
PCS	Power Conversion System
PSID	Pounds Per Square Inch Differential
PRA	Probabilistic Risk Assessment
PSP	Pressure Suppression Pool
PV	Pressure Vessel
PWR	Pressurized Water Reactor
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling System
RES	Office of Nuclear Regulatory Research
RHR	Residual Heat Removal System
RHRSW	Residual Heat Removal Service Water
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup System
SASA	Severe Accident Sequence Analysis
SBGTS	Standby Gas Treatment System
SGT	Standby Gas Treatment System
SBLOCA	Small Break Loss of Coolant Accident
SDV	Scram Discharge Volume
SI	International System of Units (Systeme International)
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SNL	Sandia National Laboratories
SORV	Stuck Open Safety Relief Valve
SRV	Safety Relief Valve
TAF	Top of Active Fuel
TIP	Traveling Incore Probe
TQUV	Transient event initiation by reactor scram and failure of normal feedwater system to orovide core make-up water, accompanied by failure of HPCI and RCIC, and by failure of low pressure ECCS
TVA	Tennessee Valley Authority
VWI	Vessel Water Injection
WW	Wetwell
Zr	Zirconium

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This study describes the predicted response of Unit One at the Browns Ferry Nuclear Plant to a postulated complete failure to scram following a transient occurrence that has caused closure of all Main Steam Isolation Valves (MSIVs). This hypothetical event constitutes the most severe example of the type of accident classified as Anticipated Transient Without Scram (ATWS). Without the automatic control rod insertion provided by scram, the void coefficient of reactivity and the mechanisms by which voids are formed in the moderator/coolant play a dominant role in the progression of the accident. Actions taken by the operator greatly influence the quantity of voids in the coolant and the effect is analyzed in this report. The progression of the accident sequence under existing and under recommended procedures is discussed. For the extremely unlikely cases in which equipment failure and wrongful operator actions might lead to severe core damage, the sequence of emergency action levels and the associated timing of events are presented.

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