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Loss of DHR Sequences at Browns Ferry Unit One—Accident Sequence Analysis

D. H. Cook
S. R. Greene
R. M. Harrington
S. A. Hodge

Prepared for the U.S. Nuclear Regulatory Commission
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D. H. Cook R. M. Harrington
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SUMMARY

This study describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to an extended post-shutdown loss of decay heat removal (DHR) capability. The postulated loss of DHR involves the prolonged loss of the power conversion system (PCS) and both the pressure suppression pool cooling and the reactor vessel shutdown cooling operational modes of the residual heat removal (RHR) system. With the decay heat removal capabilities of the PCS and the RHR system unavailable, the reactor decay heat energy would be concentrated in the pressure suppression pool.

The loss of DHR accident sequences have been selected for the Severe Accident Sequence Analysis (SASA) study presented in this report because they constitute six of the eight dominant accident sequences leading to core melt which have been identified for Browns Ferry Unit One by the NRC's Interim Reliability Evaluation Program (IREP). The IREP study is a probabilistic risk assessment (PRA) whose function 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, sequences. The SASA approach, on the other hand, is to examine a particular category of accident sequences in far greater depth than would be possible in a PRA study.

The purpose of the SASA studies is to pre-determine the probable course of the identified dominant severe accidents so as to establish the timing and the sequence of events for use in the unlikely case that one of these accidents might actually occur. The SASA studies also produce recommendations concerning the implementation of better system design and better emergency operating instructions and operator training. In the interest of efficiency, it is desirable that the SASA effort be directed toward the dominant accident sequences identified by the IREP or other PRA studies as in the case of the Loss of DHR accident sequences at Browns Ferry Unit One.

The basic initiating events for a Loss of DHR sequence include a reactor scram, closure of the main steam isolation valves (MSIVs) so that the main condenser cannot function as a heat sink, and subsequently, failure of the RHR system to provide either suppression pool cooling or reactor vessel shutdown cooling. The steam produced by decay heat is relieved from the reactor vessel by the safety/relief valves (SRVs) and is condensed in the pressure suppression pool. The suppression pool temperature increases monotonically and the resulting increase of pressure in the primary containment ultimately threatens containment integrity.

Reactor vessel water level can be maintained during the early stages of a loss of DHR accident sequence by operation of either the high pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC) pumps. The control rod drive (CRD) hydraulic system pump injects water into the reactor vessel at a rate of 0.0038 m^3/s (60 gpm) if the scram is reset and 0.011 m^3/s (170 gpm) when a scram signal is in effect. All three pumps take suction on the condensate storage tank, and operating procedures provide that there would be an initial supply of water in the tank sufficient to last well beyond the time of containment failure in a loss of DHR accident sequence.

The BWR-LACP code developed at Oak Ridge National Laboratory for BWR analysis has been used for the analysis of the sequence of events before core uncovery. The assumed containment failure pressure has been taken from a recent study conducted at the Ames Laboratory which predicts failure of the Browns Ferry steel containment by static overpressurization at 0.910 MPa (117 psig) and that the failure would occur at the juncture of the cylindrical and spherical geometries in the drywell.

The rate of pressure increase in the primary containment during a loss of DHR sequence depends to some extent on the nature of the initiating event. If the scram is caused by a transient event and at least one pump and basic piping loop of the RHR system is available for circulation and mixing of the suppression pool water, then the suppression pool can be treated as a well-mixed volume of water undergoing a uniform pool heatup. An example fitting this case would be a loss of offsite power combined with a failure of the RHR service water (RHRSW) system; the RHR system would remain available for circulation of the suppression pool water but there would be no cooling flow to the secondary side of the RHR heat exchangers. The discharge of each RHR loop enters the pool through an elbow which is aligned so that the effluent flows axially in the torus to promote mixing, and experiments have shown that the operation of one RHR pump will effectively eliminate thermal stratification in the pressure suppression pool.

For the case of a loss of DHR accident sequence with RHR pump operation and uniform heatup of the pressure suppression pool, the containment pressure reaches the assumed static overpressurization failure point of 0.910 MPa (117 psig) after 35 h. In the interim, events at several important milestones determine the temporal plant response.

The drywell pressure is 0.108 MPa (1.1 psig) at the inception of the accident. After 1 h of suppression pool heatup with cooling unavailable, the drywell pressure reaches 0.115 MPa (2 psig). This is a scram setpoint and also causes the diesel generators, the standby gas treatment system, the high pressure coolant injection (HPCI) system, and the RHRSW pumps assigned to the emergency equipment cooling water (EECW) system to start.* Also, the valves included in groups two, six, and eight of the primary containment isolation system (PCIS) are automatically shut to isolate the drywell and torus.

Even though all control rods would have been inserted at the inception of the accident, the scram signal generated by high drywell pressure at the 1 h point is particularly important to the course of the loss of DHR accident sequence. This is because the control rod drive (CRD) hydraulic system injection into the reactor vessel increases from 0.004 m^3/s (60 gpm) to 0.011 m^3/s (170 gpm) when the scram inlet valves are opened pursuant to a scram signal. Since the drywell pressure remains above 0.115 MPa (2 psig) throughout the loss of DHR sequence after 1 h, the operator cannot reset the scram signal during this period and the injection to the vessel would remain at the higher rate. The CRD hydraulic

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*It should be noted that all of these events with the exception of HPCI system actuation would occur at the inception of the accident sequence if the initiating event were a loss of offsite power.

system pump takes suction on the condensate storage tank and thus the flow does not depend on the status of the pressure suppression pool.*

The operator would control reactor vessel level with the RCIC system during the initial stages of the loss of DHR accident sequence so the larger capacity HPCI system would not be needed and the HPCI turbine would be manually tripped shortly after its automatic initiation on high drywell pressure.

The emergency operating instructions require the operator to begin reactor vessel depressurization when the pressure suppression pool temperature reaches 49°C (120°F) and this also occurs at the 1 h point. The depressurization proceeds at a rate corresponding to a 55.5°C/h (100°F/h) cooldown of the reactor vessel and is completed at the 3.5 h point. Thereafter, the operator maintains reactor vessel pressure at about 0.689 MPa (85 psig) which is sufficient to run the RCIC turbine when necessary.†

After the 4 h point, the reactor decay heat has decreased sufficiently so that all required reactor vessel makeup injection is supplied by the CRD hydraulic system pump and all other vessel injection is terminated. The reactor vessel water level increases slowly over the next several hours until at the 8.6 h point, the operator must begin to throttle the CRD hydraulic pump discharge to prevent overfill of the reactor vessel.

Although injection by the RCIC system pump is not required after the 4 h point, this system would remain available for a significant period of time thereafter until it was isolated on high temperature [366.5 K (200°F)] in the torus room at about the 13 h point. The RCIC turbine high exhaust pressure trip setpoint of 0.276 MPa (25 psig) in the wetwell would be reached soon thereafter, at about the 14 h point. The low-pressure ECC systems (RHR and core spray) would remain available thereafter for injection to the reactor vessel from the condensate storage tank as long as the reactor vessel remains depressurized.‡

The primary containment design pressure of 0.487 MPa (56 psig) would be exceeded at the 21.5 h point. At the 24 h point, the pressure in the drywell would exceed 0.550 MPa (65 psig) and the SRVs could no longer be remote-manually operated as necessary to keep the reactor vessel depressurized. The reactor vessel would therefore repressurize, reaching the

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*If offsite power is not available, the spare CRD hydraulic pump can be operated with power from a diesel generator.

†The pressure suppression pool temperature exceeds the maximum design lube oil cooler inlet temperature [60°C (140°F)] for the RCIC (and HPCI) system at the 1.6 h point. Since the lube oil is cooled by the water being pumped, RCIC pump suction should be kept in its normal alignment, i.e., to the condensate storage tank. It should be noted that operation of the HPCI system becomes questionable after the 2-1/2 h point, when the indicated suppression pool level exceeds +7 in. and the suction of the HPCI booster pump is automatically (and irreversibly) shifted to the heated pressure suppression pool.

‡Operator action would be required to realign the suction of these systems from the pressure suppression pool to the condensate storage tank.

setpoint [7.72 MPa (1105 psig)] for automatic actuation of the lowest-set SRV at about the 28 h point.*

The pressure in the primary containment would reach the assumed failure pressure of 0.910 MPa (117 psig) at the 35 h point. The reactor vessel would have been pressurized during the seven hour period immediately preceeding containment failure with the pressure controlled by automatic actuation of the lowest-set SRV and the water level maintained by operation of the CRD hydraulic system pump. At the time of containment failure, the temperature of the pressure suppression pool is 446 K (343°F) and the temperature in the drywell atmosphere is 500 K (440°F).†

The sequence of events after containment failure is uncertain. The physical integrity of the primary system might be lost because of a violent displacement of the drywell during blowdown. The capability for sufficient reactor vessel injection to keep the core covered might be lost because of the harsh environmental conditions in the reactor building combined with an inability to depressurize the reactor vessel so that the low-pressure injection systems located outside of the reactor building could be used. Thus the possibilities range from a large-break LOCA with loss of injection to continued adequate core cooling and consequently, no severe accident. For this study, it has been assumed that the integrity of the primary system is maintained but all reactor vessel injection capability is lost. This is the approach adopted by the Reactor Safety Study (WASH-1400) and subsequent PRAs.

The MARCH code has been used for the analysis of the depressurization of the primary containment and the subsequent events. The MARCH computations were initiated just before the primary containment pressure reached the failure level, with initial conditions provided by the results of the BWR-LACP code at the 34 h point. Based on recent analytical work at the AMES laboratory, the primary containment is assumed to fail in the drywell, at the juncture of the cylindrical and spherical portions of the liner with a failure area of 0.929 m^2 (10 ft^2).

The MARCH results predict primary containment failure at the 35 1/4 h point and all water injection to the reactor vessel is assumed to cease at this time. As previously discussed, the primary system is assumed to maintain its integrity during and after the primary containment blowdown, and a pressurized boiloff of the water in the reactor vessel at the time of containment failure follows. Because of the large inventory of water in the reactor vessel that must be boiled away through the relief valves and the low level of decay heat this long after shutdown, core uncovery does not occur until about 2 1/2 h after the loss of injection. The onset

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*No coolant is lost from the reactor vessel during the repressurization and the level swell caused by heating of the water would cause the operator to keep the CRD hydraulic pump off during most of the repressurization. (The mass of water in the vessel remains constant but the density decreases.)

†The drywell coolers are lost early in the accident sequence as a result of automatic load shedding when the core spray actuation signal of a combination high drywell pressure-low reactor vessel pressure occurs. Drywell heating is accelerated during the latter part of the accident sequence when the reactor vessel has repressurized.

of fuel melting occurs about 1 h later, or 38-3/4 h after the inception of the accident.

The results of this study illustrate the characteristically slow nature of the loss of DHR accident sequence and the very long time available for the operator to take corrective action.

One purpose of this work has been to determine if additional information and calculations might affect the conclusion of the IREP study that Loss of DHR accident sequences constitute a major portion of the total risk of core melt at Browns Ferry Unit 1. This assignment is a natural and intended function of the SASA program, since this task involves a detailed consideration of a specific set of accident sequences.

The PRA done under the auspices of the IREP program identifies the Loss of DHR sequences as dominant as a result of an attempted consideration of all possible accident sequences at Browns Ferry Unit 1. With such a broad scope of study, available RHR system cross-ties between units were neglected and several other simplifying assumptions were necessarily made. These include:

1. Reactor vessel injection by the CRD hydraulic system was neglected,
2. The containment was assumed to remain at atmospheric pressure during the heatup of the pressure suppression pool,
3. The ample source of cool water (not affected by pressure suppression pool heatup) available to the ECCS systems from the condensate storage tank was neglected,*
4. It was assumed that the RHR system function totally fails if the minimum flow bypass valves provided for pump protection do not close, and
5. The analysis does not include consideration of the use of the standby coolant supply system, which can be used if necessary in a loss of DHR accident sequence to periodically inject river water into the reactor vessel directly or into the drywell or wetwell spray headers as a means to reduce the pressure in the primary containment and thereby avoid containment failure. Since removal of water from the pressure suppression pool can be accomplished in several ways, especially if the wetwell is pressurized, river water spray would be an effective long-term heat removal mechanism to substitute for the normal decay heat removal functions.

With the simplifying assumptions employed in the IREP study, all reactor vessel water injection capability is lost when the pressure suppression pool water temperature reaches 355 K (180°F), about 5 h after the inception of the loss of DHR accident sequence, and core uncovery occurs shortly thereafter. However, the sequence of events determined by the more detailed analysis presented in this report shows that the reactor vessel water injection capability can be maintained at least until the containment fails by overpressurization, more than 24 h after the inception of the accident sequence. This allows much more time for corrective action by the operators. Thus the IREP study treatment of assumptions (1)

*The IREP study did not recognize that the RHR system and the core spray system pumps can take suction on the condensate storage tank, or that the condensate storage tank normally holds enough water to maintain the core covered beyond the point of containment failure by over pressurization.

through (5) above might have caused the loss of DHR accident sequences to unrealistically appear to constitute the majority of the dominant core melt sequences.

Accordingly, it is recommended that the order of dominant sequences established by the IREP study be reconsidered because it is probable that this will lead to a significant reduction in the core melt frequency assigned to the loss of DHR sequences. For example, a probability should be assigned as to whether or not the CRD hydraulic system is available during the accident sequence rather than assuming that it is not available, which is tantamount to assigning a 100% failure probability to this important system.

LOSS OF DHR SEQUENCES AT BROWNS FERRY UNIT ONE –
ACCIDENT SEQUENCE ANALYSES

D. H. Cook R. M. Harrington
S. R. Greene S. A. Hodge

ABSTRACT

This study describes the predicted response of Unit One at the Browns Ferry Nuclear Plant to a postulated loss of decay heat removal (DHR) capability following scram from full power with the power conversion system unavailable. In accident sequences without DHR capability, the residual heat removal (RHR) system functions of pressure suppression pool cooling and reactor vessel shutdown cooling are unavailable. Consequently, all decay heat energy is stored in the pressure suppression pool with a concomitant increase in pool temperature and primary containment pressure. With the assumption that DHR capability is not regained during the lengthy course of this accident sequence, the containment ultimately fails by overpressurization. Although unlikely, this catastrophic failure might lead to loss of the ability to inject cooling water into the reactor vessel, causing subsequent core uncovery and meltdown. The timing of these events and the effective mitigating actions that might be taken by the operator are discussed in this report.

1. INTRODUCTION

This is the third report in a series of accident studies concerning the BWR 4 – MK I containment plant design.* These studies have been conducted 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. These studies have been done under the auspices of the Severe Accident Sequence Analysis (SASA) program, 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 pre-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 case that one of these accidents might actually occur. These studies also produce 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.

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*Previous reports concern Station Blackout (NUREG/CR-2182) and Scram Discharge Volume Break (NUREG/CR-2672).

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 contruction. Unit 1 began commercial operation in August 1974, 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 the common refueling floor.

This report presents an analysis of the sequence of events during a prolonged loss of decay heat removal (DHR) capability following a scram at Unit 1 of the Browns Ferry Nuclear Plant. This accident category was selected for analysis because it is included in six of the eight dominant accident sequences identified for Browns Ferry Unit 1 by the Interim Reliability Evaluation Program (IREP).[1.1] The postulated loss of DHR involves the loss of the power conversion system* and both the pressure suppression pool cooling and the reactor vessel shutdown cooling modes of the residual heat removal (RHR) system. With the RHR decay heat removal capability unavailable, the reactor decay heat energy would be concentrated in the pressure suppression pool. The pressure suppression pool response depends to some extent on the manner in which the decay heat energy is introduced; Chap. 2 provides a discussion of the general classification of initiating events.

Loss of DHR accident sequences have been previously considered in Probabilistic Risk Assessment (PRA) studies such as the Reactor Safety Study (WASH-1400). These studies have treated the pressure suppression pool as a well-mixed volume of water. There is some justification for this approach, since operation of the RHR system pumps (even without the heat exchanger function to remove heat from the flow) would provide good pressure suppression pool mixing during the general pool heatup. The response of Browns Ferry Unit 1 after a scram with loss of DHR function and uniform pool heatup is presented in Chap. 3 of this report.

Given that the normal modes of decay heat transfer to the plant cooling water systems are not available, there is still the opportunity for the operator to use ingenious methods to remove decay heat from the overall plant. Methods for mitigation and normal recovery from the loss of DHR function are discussed in Chap. 4.

If not even one pump and basic piping loop of the RHR system is available to induce suppression pool mixing, then the effect of thermal stratification in the pool water will cause containment failure by overpressurization earlier than would be predicted using the assumption of uniform pool heatup. The results of analyses of containment response without the assumption of a well-mixed pool are discussed in Chap. 5.

Two of the IREP-identified Browns Ferry dominant sequences involving loss of DHR capability include a stuck-open relief valve in the initiating

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*Loss of the Power Conversion System means that decay heat cannot be removed via the main condensers.

event.* With a stuck-open relief valve, all of the decay heat energy from the reactor vessel is transmitted into the suppression pool at one location, and the reactor vessel is depressurized at the time when the containment fails. Accident sequences with a stuck-open relief valve are discussed in Chap. 6.

As shown by the detailed SASA program analysis provided in this work, such a long time is required for pressure suppression pool heatup to the point where containment failure would occur by overpressurization and there is so much opportunity for equipment repair and so many mitigating actions available to the operating staff that it is doubtful that loss-of-DHR accident sequences should be eligible for inclusion in the category of "dominant sequences" leading to core uncovery and melting at BWR MK I containment plants. Nevertheless, this study includes consideration of the possible Severe Accident phases of a Loss of DHR accident. A Severe Accident by definition is an accident that in the absence of effective corrective action by the operating staff proceeds through core uncovery, core meltdown, and the release of fission products from the fuel. The events in the Severe Accident phases of a prolonged Loss-of-DHR event have been analyzed by application of the MARCH code and are described in Chapters 7 and 8.

The implications of the results of this study are discussed in Chap. 9. The discussion includes an evaluation of the available instrumentation, the level of operator training, the existing emergency procedures, and the overall system design from the standpoint of requirements for mitigation of this accident. The final portion of Chapter 9 provides a discussion of the need for reconsideration of the IREP study findings in light of the results of this work.

The conclusions of this study and a brief discussion of the uncertainties involved are discussed in Chap. 10.

A simple schematic diagram of the reactor vessel injection systems considered in this study is provided in Fig. 1.1. With the exception of the control rod drive (CRD) hydraulic system, all of these injection systems can taken suction on either the condensate storage tank or the pressure suppression pool and have injection capabilities much larger than that required to replace the water boiled to steam and lost from the vessel through the SRVs after a scram. The CRD hydraulic pump injects condensate storage tank water into the reactor vessel at a rate of 0.0038 m^3/s (60 gpm) under normal operating conditions. This flow increases when a scram is in effect,[1,2] to about 0.0070 m^3/s (100 gpm) while the reactor vessel is pressurized and 0.011 m^3/s (170 gpm) when the vessel is depressurized.

An understanding of the RHR system is important to the consideration of the general category of loss of DHR accidents and the necessary information concerning this important system is provided in Appendix A.

Appendix B contains a description of the additions to the computer program BWR-LACP made for this study; this is the code developed by R. M. Harrington at ORNL to model operator actions and the associated primary

*These are (1) anticipated transient with loss of the power conversion system and (2) loss of offsite power.

system and reactor building response during the period prior to core uncovery in accident sequences at Browns Ferry.

Appendix C provides a discussion of the computer code developed by D. H. Cook at ORNL as a dissertation project to provide a realistic model of suppression pool heatup in a BWR Mark I containment system, with consideration of thermal stratification and localized pool heating.

The MARCH code input for the Severe Accident phases of this study is provided in Appendix D.

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

The primary sources of plant-specific information used in the preparation of this report were 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 was also applied in this effort.

The setpoints for automatic equipment response used in this study are the currently established safety limits. In many cases these differ slightly from the actual setpoints used for instrument adjustment at the BFNP because the instrument adjustment setpoints are established so as to provide margin for known instrument error.

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

References for Section 1

1.1 S. E. Mays et al., "Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant," NUREG/CR-2802, EGG-2199, July 1982.

1.2 S. A. Hodge et al., "SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis," NUREG/CR-2672, Volume 1, ORNL/TM-8119/V1 (November 1982), Sect. E.3.

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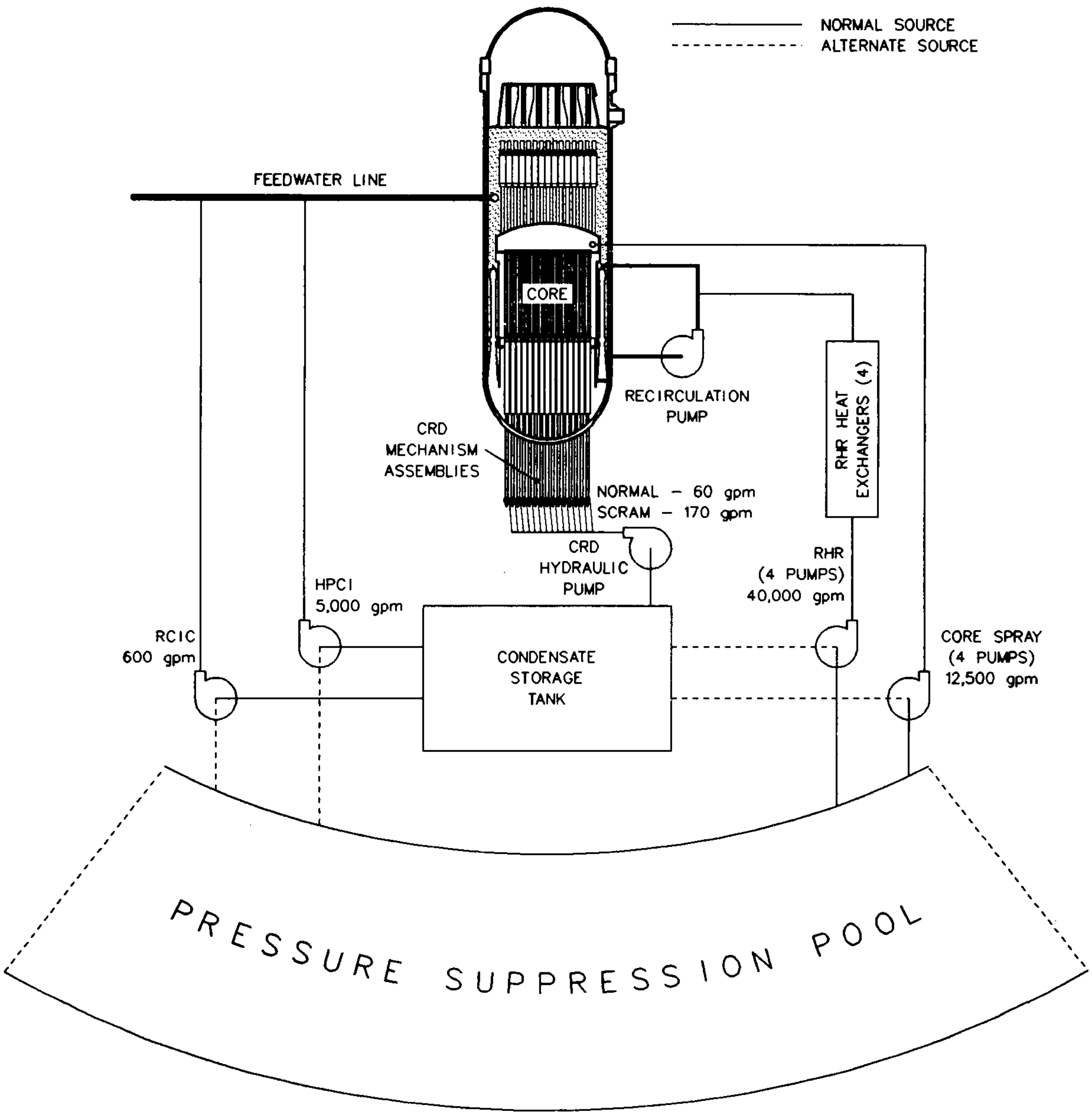

Fig. 1.1. Simplified diagram of reactor vessel injection systems.

2. INITIATING EVENTS

Loss of the decay heat removal (DHR) function means that the residual heat removal (RHR) functions of pressure suppression pool cooling and reactor vessel shutdown cooling are unavailable during an accident sequence in which the power conversion system (PCS) is also not available. Thus decay heat cannot be removed to the RHR service water system via the heat exchangers in the RHR system nor to the main circulating water system via the main condensers. Under these circumstances, all decay energy is passed from the reactor vessel through the safety/relief valves (SRVs) to the pressure suppression pool.

Since a large amount of energy is absorbed in the pressure suppression pool over a long period of time, the pressure suppression pool temperature steadily increases. As the temperature of the pool upper layer increases above 373 K (212°F), the pressure of the primary containment drywell-wetwell combination begins to increase significantly and, unless successful operator action is taken to restore the DHR function or to vent the containment, the pressure will ultimately reach the failure pressure of the drywell.*

The pressurization rate of the primary containment during a Loss of DHR accident sequence depends on the nature of the initiating event. Therefore it is convenient to the purposes of this study to group all initiating events into four classes according to their effect on suppression pool heatup. Each of these classes is discussed in the following sections.

2.1 Transients with Uniform Pool Heatup

The accident initiators in this category lead to Loss of DHR accident sequences in which the pressure suppression pool can be treated as a well-mixed volume of water undergoing a uniform pool heatup. This requires that at least one loop of the RHR system be operable for circulation and mixing of the pressure suppression pool water, even though the heat removal function of the loop is not available.†

Loss of DHR accident sequences with the assumption of uniform pool heatup are discussed in Chap. 3.

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*As discussed in Ref. 2.1, static overpressure [0.910 MPa (117 psig)] is expected to cause failure of the primary containment in the drywell at the cylinder-sphere interface.

†For example, the failure might be in the RHR service water system, leaving the RHR system fully available for suppression pool circulation. Experimental results discussed in Ref. 2.2 show that the mixing induced by operation of one RHR loop will effectively eliminate thermal stratification in the pressure suppression pool. This mixing is enhanced by a piping elbow within the torus that directs the RHR pump discharge in a direction parallel to the torus axis, i.e., around the circumference of the pool. This elbow will be installed within the Unit 1 torus during the refueling outage to begin in March, 1983. It has been assumed to be in place for the calculations performed in this study.

2.2 Transients with Pool Thermal Stratification

If in addition to the loss of DHR function, no RHR pump and basic piping loop is available for suppression pool circulation and mixing, then significant thermal stratification will occur in the pool water during the heatup. [See the discussion in Appendix C]. Since the temperature of the upper layer of the suppression pool water will be significantly higher than the pool bulk average temperature, the containment pressurization rate will be higher and the drywell failure pressure will be reached earlier.

It should be noted that operator action to manually operate the reactor vessel relief valves, alternating among the 13 valves as required to distribute the relief valve discharge evenly around the circumference of the pressure suppression pool, would also be effective in providing a more uniform pool heatup. However, manual relief valve actuation is only possible when the available control air pressure is 0.172 MPa (25 psi) or more higher than the pressure in the drywell. Since the average drywell control air pressure is 0.722 MPa (90 psig), manual relief valve actuation will not be possible after the drywell pressure has reached 0.550 MPa (65 psig). This will occur in every Loss of DHR accident sequence that proceeds to containment failure, which is expected to occur at 0.910 MPa (117 psig).[2.1] When the relief valves can no longer be manually operated, the reactor vessel will repressurize to the setpoint [7.722 MPa (1105 psig)] for automatic actuation of the lowest-set relief valve and this relief valve will repeatedly actuate thereafter.

Loss of DHR accident sequences analyzed with consideration of the effect of thermal stratification in the pressure suppression pool are discussed in Chap. 5.

2.3 Stuck-open Relief Valve

The third broad category of Loss of DHR accident sequences involves transients with a stuck-open relief valve. These cases are represented in two of the eight dominant sequences identified for Browns Ferry Unit 1 by the IREP study[2.3] and therefore analyses of the follow-on accident sequences have been included in this study.

The Loss of DHR accident sequences with a stuck-open relief valve differ from those discussed in Chaps. 2.1 and 2.2 because the reactor vessel remains depressurized to a pressure about 0.396 MPa (50 psi) above containment pressure throughout the latter part of the accident sequence. This alters the characteristics of the energy addition to the suppression pool, i.e., there is a relatively slow continuous discharge from the T-quencher (located at the tailpipe terminus of the stuck-open valve) instead of the intermittent bursts of high steam flow which occur in the other cases during the periods when the reactor vessel is pressurized.

Loss of DHR sequences with a stuck-open relief valve are discussed in Chap. 6.

2.4 LOCA

The fourth broad category of Loss of DHR accident sequences comprises those sequences associated with a loss of coolant accident (LOCA) in the drywell. These accident sequences were not included among the dominant sequences identified by the IREP study and therefore are not considered in detail here.

Nevertheless, it should be noted that with an intermediate or large LOCA in the drywell, the decay heat energy enters the pressure suppression pool through the 96 downcomers rather than through the relief valve T-quencher discharge devices.* Since the T-quenchers are located about 3.20 m (10.5 ft) beneath the surface of the pool and the downcomers discharge just 1.04 m (3.4 ft) below the surface of the pool, it is probable that thermal stratification would be more severe in the case of a loss of DHR function following a LOCA in the drywell.[2.4]

References for Section 2

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

2.2 B. J. Patterson, "MARK I Containment Program Monticello T-quencher Thermal Mixing Test Final Report Task Number 7.5.2," NEDO-24542 Class I, August 1979.

2.3 S. E. Mays, et al., "Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant," NUREG/CR-2802, EGG-2199, July 1982.

2.4 K. W. Wong and H. S. Yao, "MARK I Containment Program Downcomer Reduced Submergence — Functional Assessment Report Task Number 6.6," NEDO-21885 Class I, June 1978.

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*For a small LOCA in the drywell, the entry to the pressure suppression pool would be divided between the T-quenchers and the downcomers.

3. TRANSIENTS WITH LOSS OF DHR: CALCULATIONS ASSUMING UNIFORM POOL TEMPERATURE

3.1 Introduction

The defining system failures for the Loss of DHR accident sequence occur after an initiating incident and successful reactor scram, whereby the Main Steam Isolation Valves (MSIVs) close, the condenser cannot function as a heat sink, and the RHR system is unable to provide either suppression pool cooling or shutdown cooling. The steam produced by decay heat is relieved from the reactor vessel by the SRVs and is condensed in the suppression pool. The temperature of the uncooled suppression pool increases monotonically, leading to escape of steam from the pool surface and therefore to a pressure buildup which eventually causes high pressure failure of the drywell.

The vessel water injection function is not initially impaired, and it is assumed that the operators would act to maintain reactor vessel water level near the normal 14.25 m (561 in.) above vessel zero. Manual control of the SRVs is also initially unimpaired, and the operators would control reactor vessel pressure according to the emergency operating instructions which require initiation of a 56°C/h (100°F/h) depressurization before suppression pool temperature exceeds 49°C (120°F). If the RHR system can be operated to circulate the suppression pool water,* it is assumed that the operators would do so.

The calculations reported in this section were performed with the ORNL-developed BWR-LACP code. Appendix B gives detailed input assumptions and discusses sequence-specific modifications which were necessary to adequately model these sequences. The most significant modeling assumption for the results reported in this section is that the temperature of the suppression pool is uniform throughout the pool. It is known that hotter pool water tends to rise to the top of the pool and that the water in the vicinity of a discharging SRV T-quencher is hotter than bulk pool temperature.[3.1] However, with at least one RHR pump operating (without heat removal) to circulate the pool water, both of these effects would be minimized. Modifications are planned for the next (March 1983) refueling outage to equip the Unit 1 pool cooling discharge lines with elbow and fittings which will discharge horizontally along the circumference of the torus to promote circulation and mixing of the whole pool. In addition, the Browns Ferry emergency operating instruction for main steam isolation valve (MSIV) closure requires that operators alternate their selection of relief valves in order to minimize local temperature buildup in the vicinity of a discharging T-quencher.

If the suppression pool is not mixed by the operation of at least one RHR pump, the net effect of locally higher temperatures would be a more rapid buildup of primary containment pressure than reported in this section. The results reported in Chaps. 5 and 6 were calculated using a

*Albeit without the RHR heat exchanger function to provide pool cooling.

special suppression pool model which can be used to calculate the temperature as a function of location in the pool when the water is not well-mixed.

3.2 Summary and Conclusions

Following accident initiation, the operators would maintain reactor vessel water level by control of the Reactor Core Isolation Cooling (RCIC) system. After about 4 h, the 0.011 m^3/s (170 gpm) injection provided by the Control Rod Drive (CRD) hydraulic system pump is sufficient to maintain vessel level, without the aid of the higher capacity RCIC system. Dependence of the vessel injection function upon the status of the pressure suppression pool is avoided because the 1370 m^3 (362,000 gal) supply of cooling water stored in the Unit 1 Condensate Storage Tank (CST) is sufficient to last throughout the sequence.

As the suppression pool temperature increases, steam escaping from the surface of the pool increases the primary containment pressure until, 35 h after accident initiation, the 0.91 MPa (117 psig) failure pressure[3.2] of the drywell is exceeded. The calculations reported in this section are terminated after 35 h — no attempt has been made to model events after drywell failure with the BWR-LACP code.

If the flow area of the drywell rupture (the weak point in the drywell is at the intersection of the spherical and cylindrical segments[3.2]) were small, then the subsequent energy release would be spread over a long period of time. This would minimize the disruptive effect of drywell failure on safety systems in the reactor building and drywell; it is possible that the vessel water injection could be maintained and that there would be no core damage at any time during the sequence.

If the flow area of the drywell rupture were large then a great amount of energy would be released over a short period of time with potentially catastrophic effects on safety systems in the reactor building and drywell. The subject of drywell failure modes is discussed in detail in Chap. 7. The MARCH calculations reported in Chap. 8 were performed with the assumption that vessel water injection fails after drywell failure, leading to core uncovery and severe core damage.

3.3 Detailed Results

Figures 3.1-3.13 show the BWR-LACP results for reactor coolant system and primary containment variables throughout the entire 35 hours before primary containment failure. Table 3.1 summarizes major events during the first 35 hours.

3.3.1 Reactor vessel water level

Figure 3.1 shows water level in the downcomer region of the reactor vessel.* Figure 3.2 shows total vessel injection flow rate and the total amount of water injected is shown in Fig. 3.3.

Throughout the Loss of DHR accident sequence the preferred source of water for injection into the reactor vessel would be the Unit 1 Condensate Storage Tank (CST). The normal water volume of the CST is 1370 m^3 (362,000 gal) and this amount of water is assumed to be present at the beginning of the accident. It is possible, but not likely, that there could be significantly less volume at the beginning of the accident. The main condenser hotwell draws its make-up from a standpipe within the CST and flow through the standpipe could conceivably reduce the supply of condensate to 511 m^3 (135,000 gal). For this to occur would require a breach of the condensate system because the Browns Ferry operating procedures require replenishment of the CST (from the 1420 m^3 (375,000 gal) demineralized water storage tank) upon receipt of a CST low level alarm (which corresponds to a volume of 1301 m^3 (344,000 gal). If CST level cannot be rapidly restored following a low-low level alarm at 579 m^3 (153,000 gal), the procedure requires an orderly shutdown of the unit.

The suppression pool might be used instead of the CST as a source of water for vessel injection by the ECCS and RCIC systems; however, the HPCI and RCIC turbines depend on the pumped water for cooling of their lube oil. The recommended maximum water temperature for long term operation is 60°C (140°F) (see Browns Ferry FSAR, Amendment 24, Section Q14.1-4), and this temperature is exceeded in the pressure suppression pool about 2.0 h into the loss of DHR sequence.

The calculation represented in Figs. 3.1 through 3.13 was initialized 30 s after reactor scram, with the MSIVs closed and with reactor vessel water level at 12.7 m (500 in.) above vessel zero. Section 10.2 discusses the uncertainty in this assumed value of initial vessel level. One cycle of High Pressure Coolant Injection (HPCI) system actuation, initiated by the operators before level decreased to the setpoint for automatic initiation, is required to bring level back to the normal range; after this and for the next four hours, the 0.038 m^3/s (600 gpm) RCIC system is more than adequate to maintain normal vessel level.

The control rod drive (CRD) hydraulic system pumps water into the reactor vessel throughout the sequence. Following reset of the initiating scram (i.e. if the scram condition has cleared), the CRD hydraulic system injection drops to 0.0038 m^3/s (60 gpm). After 1 h,† the drywell pressure exceeds the 0.115 MPa (2 psig) high drywell pressure scram setpoint, causing the 185 CRD scram inlet and outlet valves to open. This second scram does not affect the already fully inserted control rods, but with the scram inlet valves open, the CRD hydraulic system injection flow to the reactor vessel increases to 0.011 m^3/s (170 gpm).

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*This is the level which the control room instruments are designed to indicate. To read the full range of level variation shown on Fig. 3.1, operators would have to consult the wide range instruments.

†This assumes continuous operability of the drywell coolers (see Sect. 3.3.4).

After about 4 h, the CRD hydraulic system is providing all the vessel water injection and the RCIC system is no longer needed. Several hours later (8.6 h after event initiation) the full amount of CRD vessel injection is more than enough and the operators have to take action to prevent excessively high vessel water level. This could be accomplished either by intermittent CRD pump operation, or by throttling the CRD hydraulic pump discharge. The Unit 1 "A" CRD hydraulic pump can only be throttled by local-manual control of the discharge valve. The "B" pump (which serves as a spare pump for both Units 1 and 2) can be throttled from the control room by remote-manual control of its discharge valve; therefore, the operators would most likely switch to the "B" pump in order to maintain continuous control of injection flow from the main control room.

Late in the sequence, when remote-manual control of the SRVs is lost (see Sect. 3.3.2) the reactor vessel undergoes repressurization and for a period of 4 h, while vessel pressure is building to the automatic SRV actuation pressure, no steam is lost from the vessel. During this period the approximately constant mass of water in the vessel undergoes a thermal expansion of about 22%, swelling to well above the normal range (some water would overflow into the main steam lines) even though all vessel injection is cut-off during most of the repressurization.

3.3.2 Reactor vessel pressure

Figures 3.4 and 3.5 show reactor vessel pressure and total steam flow to the suppression pool.

Since the MSIVs are closed, steam produced in the reactor vessel is relieved to the suppression pool through the SRVs and through the RCIC and HPCI turbine exhaust during the periods when these turbines are operating. The lowest-set SRVs would actuate automatically at 7.72 MPa (1105 psig) and reclose after vessel pressure has been reduced by about 5%. However, the Browns Ferry emergency operating instructions require the operator to minimize automatic SRV actuations by remote-manual operation of a single SRV at a pressure slightly lower than the setpoint for automatic actuation so as to reduce vessel pressure by about 20% instead of 5%. This not only minimizes the total number of valve actuations but also allows the operators to alternate their selection of SRVs around the suppression pool such that local pool heatup in the neighborhood of a discharging T-quencher is minimized.

When pool temperature reaches 49°C (120°F), the emergency operating instructions require that the operators initiate a 55°C/h (100°F/h) depressurization of the reactor vessel, with the final target pressure below 1.48 MPa (200 psig). This depressurization rate is achieved at first by intermittent, and then by continuous operation of a single SRV. The final pressure attained is about 0.69 MPa (85 psig) — well above the isolation pressure of the RCIC turbine steam supply line.* The history of the reactor vessel pressure during the accident sequence is shown in Fig. 3.4.

After depressurization, with vessel pressure in the neighborhood of 0.69 MPa (85 psig), the steam production rate is nearly in balance with

*The RCIC system automatically isolates if the reactor vessel pressure drops below 0.448 MPa (50 psig).

the capacity of a single SRV, so that, with one SRV remaining open, vessel pressure does not decrease further. Although not apparent on Fig. 3.4, the operators would occasionally close the open SRV and simultaneously open another SRV to direct the discharge to another part of the pool.

In order to enable remote-manual SRV operation, the pressure of the drywell control air must exceed the drywell pressure by at least 0.17 MPa (25 psid). The Group II isolation on high drywell pressure (Groups VI and VIII isolations are also triggered by the high drywell pressure signal) which occurs after about 1 h would isolate the drywell control air suction, thereby compromising long-term remote-manual operability of the SRVs (and also operation of the drywell coolers, whose discharge dampers require control air to remain open). This situation could, however, be remedied because the operators are required by emergency operating instructions to valve-in the station control air, which is maintained at a pressure* very close to that of the drywell control air. Station control air compressors A and D can be powered by the diesels in the event of loss of offsite power, but they would have to be restarted locally after the load-shedding which would occur after about 2 h due to combined low reactor vessel pressure and high drywell pressure.

After about 24 h, the drywell pressure exceeds 0.55 MPa (65 psig), and there is no longer the pressure differential required for remote-manual SRV actuation. The open SRV therefore closes, and cannot be opened in response to operator action. After the reactor vessel repressurizes to 7.72 MPa (1105 psig), the lowest-set SRV would begin automatic actuation as shown in Fig. 3.5 (this mode of SRV operation does not require control air).

It would be desirable to maintain a depressurized reactor vessel throughout this sequence in order to minimize heat losses to the drywell atmosphere. The design temperature of many components in the drywell (including the SRV remote-manual actuation solenoids) is 138°C (281°F). Additionally, when vessel pressure is low, both high and low pressure injection systems would be able to function if necessary.

3.3.3 Suppression pool temperature and water level

Suppression pool temperature (Fig. 3.6) and water level (Fig. 3.7) increase steadily throughout the Loss of DHR sequence except during periods when there is neither SRV discharge nor any HPCI or RCIC turbine exhaust into the pool. The T-quencher underwater steam discharge device has replaced the ramshead design at Browns Ferry for SRV steam discharge. The T-quenchers can produce smooth condensation at pool temperatures up to near-saturation without the instability phenomenon known to occur with the ramshead device.

The calculations reported in this section assume that complete condensation will occur if the pool is at least 1.1°C (2°F) subcooled. Uncertainties associated with this assumption are discussed in Sect. 10.2.

The rate of increase of pool temperature (Fig. 3.6) is greatest during the first several hours of the sequence because the production of decay heat is greater during this period and also because of the reactor

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*Average station control air pressure is about 0.724 MPa (90 psig).

vessel depressurization, which begins after 1 h [when pool temperature reaches 49°C (120°F)]. When the reactor vessel has been depressurized about 3 h after sequence initiation, the pool temperature has increased to 71°C (160°F).

When pool temperature reaches 100°C (212°F) one might expect that condensation of the SRV discharge or turbine exhaust steam would cease. This does not happen because by this time the total wetwell pressure has increased from atmospheric pressure to 0.18 MPa (11 psig) and the corresponding saturation temperature at the surface of the suppression pool is 117°C (242°F) instead of 100°C (212°F). The pool heating during the Loss of DHR sequence is slow enough such that the evaporation of water vapor from the pool surface can contribute to the total (nitrogen plus water vapor) pressure over the pool. In this manner, the pool remains slightly subcooled during the heatup and continues to condense 100% of the SRV steam discharge.

At the time when the drywell failure pressure is exceeded (35 h after sequence initiation), the suppression pool water temperature has increased to 173°C (343°F).

Suppression pool water level (Fig. 3.7) increases not only because warmer water is less dense, but also because of the additional mass of condensed steam in the pool. At the end of the sequence (35 h), the water level has increased by 1.37 m (4.5 ft). Plant instrumentation would indicate a level lower by about 10% as this measurement is not temperature compensated. The suppression pool-to-drywell vacuum breakers would at this point be partially submerged [i.e., water level about 0.15 m (6 in.) above the bottom of the 0.46 m (18 in.) valves], but should still be able to function to keep drywell pressure from being significantly below wetwell pressure. The calculations reported in this section assume that the wetwell-to-drywell vacuum breakers are unimpaired; thus, whenever wetwell pressure exceeds drywell pressure by more than 3.45 kPa (0.5 psid), the vacuum breakers will open and equalize the pressures. The vacuum breakers open numerous times during the sequence because there is a significant amount of net mass transfer (water vapor plus nitrogen) from suppression pool atmosphere to drywell atmosphere.

3.3.4 Primary containment atmosphere pressure and temperature

The major driving force which affects primary containment pressure (Fig. 3.8) is the vapor pressure of the suppression pool water, which increases from 4.8 kPa (0.7 psia) to 848 kPa (123 psia) as the pool is heated from 32°C (90°F) to 173°C (343°F) during the 35 h Loss of DHR sequence before containment failure. The drywell pressure and suppression pool pressure (not shown) remain very close because of the action of the 12 wetwell to drywell vacuum breakers, which prevent the wetwell pressure from exceeding drywell pressure by more than 3.45 kPa (0.5 psid).

The temperature of the wetwell atmosphere (not shown) is held very close to suppression pool water temperature (Fig. 3.6) throughout the sequence by combined convective and evaporative heat transfer from the surface of the slowly heated pool. The drywell temperature (Fig. 3.9) is determined by competing influences: the hot surfaces of the reactor vessel and piping, the cool surfaces of heat sinks such as the 2.86 cm (1.125

in.) thick steel drywell liner, the influx of hotter nitrogen and steam from the wetwell atmosphere, and heat removal by the drywell coolers.

The design heat removal capacity of the drywell coolers is about 1.5 MW, but the actual heat removal rate depends on drywell atmosphere temperature and humidity. During a Loss of DHR sequence the coolers help to control not only drywell temperature but also primary containment pressure by condensing part of the steam which flows from the wetwell airspace to the drywell. The drywell coolers run continuously and are available after accident initiation.

Following a loss of offsite power initiator, the drywell coolers can be powered from an emergency diesel generator bus. After receipt of the core spray initiation signal,* the diesels shed nonessential loads including the drywell cooler blowers.† There is system logic which would prevent the operators from subsequently restarting the blowers from the control room. The coolers can, however, be restarted and operated by utilizing local handswitches on the 480 V shutdown boards and motor control centers.

Since the Browns Ferry emergency operating instructions do not provide explicit procedures for restart of the drywell coolers under emergency conditions, and because loss of off-site power is a potential loss of DHR initiator, two different calculations of primary containment pressure and temperature have been performed. One calculation (Figs. 3.8 and 3.9) assumes that the drywell coolers run only up to the time (2 h after event initiation) of the core spray initiation signal; the other calculation (Figs. 3.10 and 3.11) assumes that the coolers are restarted after load shed, and continue to run until the blowers fail (17 h) due to the combined deleterious effects of high drywell pressure and temperature [the assumed failure temperature is 93°C (200°F)]. The variables shown on Figs. 3.1 through 3.7 and discussed in the preceding subsections were calculated assuming loss of the drywell coolers after 2 h. The effect of the drywell coolers on the performance of these variables after 2 h is negligible; therefore, no discussion of the effect of drywell cooling was provided in the corresponding subsections (3.3.1, 3.3.2, or 3.3.3).

As shown by a comparison of Figs. 3.10 and 3.8, extended operation of the drywell coolers can delay by about 2.5 h the eventual high pressure failure of the drywell. A comparison of Figs. 3.11 and 3.9 shows that the drywell temperature is also lower for the case with extended operation of the drywell coolers and this would have the beneficial effect of maintaining the temperature-sensitive equipment in the drywell below the 138°C (281°F) long-term and the 163°C (325°F) short-term design temperatures for an additional period of about 3 h. This equipment includes the solenoid valves which are necessary for remote manual operation of the SRVs.

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*The core spray system pumps automatically start upon a combination of high drywell pressure and low reactor vessel pressure.

†The logic provides that the drywell cooler loads are shed if there is a core spray initiation signal and if diesels are running and loaded.

3.3.5 ECCS pump net positive suction head (NPSH)

Pump NPSH* is of special concern during the Loss of DHR sequence due to the need to pump the very hot suppression pool water. For example, the most direct way to recover from the Loss of DHR accident would be to regain the suppression pool cooling mode of the RHR system. The success of recovery would depend on whether the RHR system could pump without severe cavitation if the pool is heated to 100°C (212°F) or more.

In-plant testing at Browns Ferry has shown that the RHR pumps can operate down to about 65% of the manufacturers recommended minimum NPSH with the following consequences: 10% degradation of developed pumping head, acceptable pump motor vibration, but severe audible cavitation. This would not jeopardize short-term operation although impeller cavitation damage would be expected in the long-term. The in-plant tests did not include reduction of NPSH to the point at which short-term pump operation would be jeopardized by sudden and severe loss of developed head and/or severe vibration.

Fig. 3.12 shows the calculated NPSH† with one RHR pump operating to circulate suppression pool water‡ at 0.63 m^3/s (10,000 gpm) throughout the pool heatup. For the case in which the drywell coolers cease operation after 2 h (see discussion in Section 3.3.4), the NPSH is greater than 90% of the manufacturer's recommended minimum at all times. Thus no difficulty with pump operation should occur. For the case of extended drywell cooler operation (curve 2), the RHR pumps could probably not function at full flow after 14 h (840 min) since the NPSH would be below the degraded region explored by the Browns Ferry tests, and attempted operation could result in pump motor failure and/or loss of all pump developed head.

When the drywell coolers operate, more water vapor escapes from the pool surface, mixes with the wetwell atmosphere above the pool, then flows through the vaccum breakers to the drywell where much of it is condensed. This process lowers the total pressure in the primary containment and tends to wash the nitrogen out of the pool atmosphere so that after 14 h there is only steam and water left in the suppression chamber and saturation conditions exist. Nitrogen is also washed out of the wetwell atmosphere when the drywell coolers are not operating, but to a much lesser extent.

The NPSH margin for acceptable RHR pump operation can be extended by operator action, throttling the flow as necessary to reduce the RHR pump discharge from the rated flow of 0.63 m^3/s (10,000 gpm). With reduced flow, there is a slight decrease in the required NPSH at the pump inlet

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*NPSH is the static plus velocity pressure at the pump inlet, less the vapor pressure of the fluid being pumped, expressed in equivalent head of the fluid being pumped. The manufacturers minimum recommended NPSH is based upon a 3% decrease in developed head but no significant audible cavitation.

†The calculation takes into account the increased depth of water in the pool (Fig. 3.7) which increases the NPSH at the RHR pump suction by about 0.76 m (2.5 ft) at the 14 h point.

‡But without pool cooling.

according to the information supplied by the pump manufacturer.* A much more important effect is that the reduction in flow serves to increase the actual NPSH available at the pump inlet by reducing the frictional pressure losses incurred in the suction piping from the pressure suppression pool. During the Browns Ferry RHR system tests, the RHR pump continued to operate well as the NPSH was lowered to 5 m (16.4 ft) with full flow. At 80% of full flow, the RHR pumps were observed to perform well at an NPSH of 4.33 m (14.2 ft).

Calculations have been performed to determine the lowest NPSH which might be encountered during the long-term Loss of DHR accident sequence it RHR flow were reduced to 80% of normal. The lowest calculated NPSH is 4.94 m (16.2 ft) which is higher than the region at which the pumps were demonstrated to be operable during the Browns Ferry tests. Therefore, if the RHR pump discharge were throttled, the RHR pumps could be operated to provide pool circulation throughout the Loss of DHR sequence even in the case of extended drywell cooler operation.

3.3.6 Reactor building environmental considerations

This section provides an evaluation of the effect of the hot, uninsulated torus on the temperature of the reactor building atmosphere. Another concern might be the heating effect over long periods of operation of the ECCS pumps, which are located in the corner rooms of the reactor building basement; however, excessive building air temperature from these sources is prevented by the ECCS room coolers, the ventilation flow maintained by the Standby Gas Treatment (SGT) system,† and the thick concrete walls acting as heat sinks.

The Browns Ferry pressure suppression chamber is located in a room (torus room) which occupies the central portion of the reactor building basement. The torus room is 11.6 m (38 ft) from floor to ceiling and is essentially closed except for four open 1.83 m (6 ft) height manways leading to the reactor building corner rooms, and several relatively small openings in the ceiling (such as the annular space between pipes which extend into the torus room through the drywell personnel access room).

As the suppression pool temperature increases, the surface of the torus begins to lose heat by radiation to the thick concrete walls and by natural convection to the torus room air. Hotter air would tend to rise, and stratify at the top of the torus room. There would be a net circulation of air into the torus room from the basement corner rooms that would exit into the ventilation ductwork and also into the drywell personnel access room. This circulation of air from the torus room would be relatively small (see Browns Ferry FSAR, Section 5.2.6.3) and not capable of transporting a large amount of heat to any of the major floor areas of the reactor building.

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*The required NPSH is 7.93 m (26 ft) at full flow and 7.62 m (25 ft) at 80% of full flow.

†The SGT system is automatically actuated when the drywell pressure reaches 0.115 MPa (2 psig).

The temperature of the torus room air and concrete have been calculated to estimate the rate of heat loss from the torus. The results indicate that the most significant heat loss is by radiant heat transfer to the concrete walls, ceiling, and floor. The torus room air temperature remains approximately mid-way between concrete and torus surface temperatures. Figure 3.13 shows the average torus surface temperature, the torus room air temperature, and the surface temperature of the concrete surroundings. Torus room air temperature exceeds the 93°C (200°F) isolation setpoint of the HPCI and RCIC turbine steam lines after 13 h.* By this time, sufficient reactor vessel injection is being provided solely by the CRD hydraulic pumps, so this isolation would not be a serious problem. At the end of the 35 h calculation, when the drywell is predicted to fail by overpressurization, the torus surface is at about 166°C (330°F), the torus room air is 147°C (297°F) and the surface of the concrete has been heated to 132°C (269°F).

References for Section 3

3.1 B. J. Patterson, "Mark I Containment Program — Monticello T-Quencher Thermal Mixing Test Final Report," GE NEDO-24542 Class I, August 1979.

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

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*The HPCI system can be isolated by temperature sensors located in both the torus room and in the HPCI room. The RCIC system can also be isolated by temperature sensors in the torus room. These sensors are positioned to detect steam leaks from the HPCI and RCIC turbine steam lines.

Table 3.1. Timetable of events for unmitigated loss of DHR with uniform pool heatup

Time (h)	Event
0	Initiating reactor trip followed by MSIV closure and failure of both pool cooling and shutdown cooling modes of the RHR system.
1	High drywell pressure scram at 0.115 MPa (2 psig). Diesel generators and SGTS automatically initiated. Drywell control air compressors isolated. Operators valve station control air into drywell control air header.
1	Pool temperature exceeds 49°C (120°F) — operators begin controlled depressurization of reactor vessel.
2	Core spray initiation signal [reactor vessel pressure <3.21 MPa (465 psia) and drywell pressure >0.115 MPa (2 psig)] causes load shedding if loss of offsite power is still in effect. Operators must use local control stations to restore diesel power to station control air compressors (A and D) and drywell coolers.
2	Suppression pool temperature exceeds the 60°C (140°F) recommended maximum temperature for cooling of RCIC and HPCI lube oil.
4	CRD hydraulic system provides sufficient reactor vessel injection — no RCIC system operation after this time.
8.6	Operators must begin to throttle CRD hydraulic system pump to avoid overfilling the reactor vessel.
13	HPCI and RCIC system steam supply line isolation caused by high [93°C (200°F)] torus room temperature.
14	RCIC turbine high exhaust pressure trip at containment pressure >0.28 MPa (25 psig).
21.5	Drywell design pressure [0.49 MPa (56 psig)] exceeded.
23.5	SRVs become inoperative in remote-manual mode because drywell pressure exceeds 0.55 MPa (65 psig).
35	Drywell fails when internal pressure exceeds 0.91 MPa (117 psig). Suppression pool temperature has increased to 173°C (343°F).

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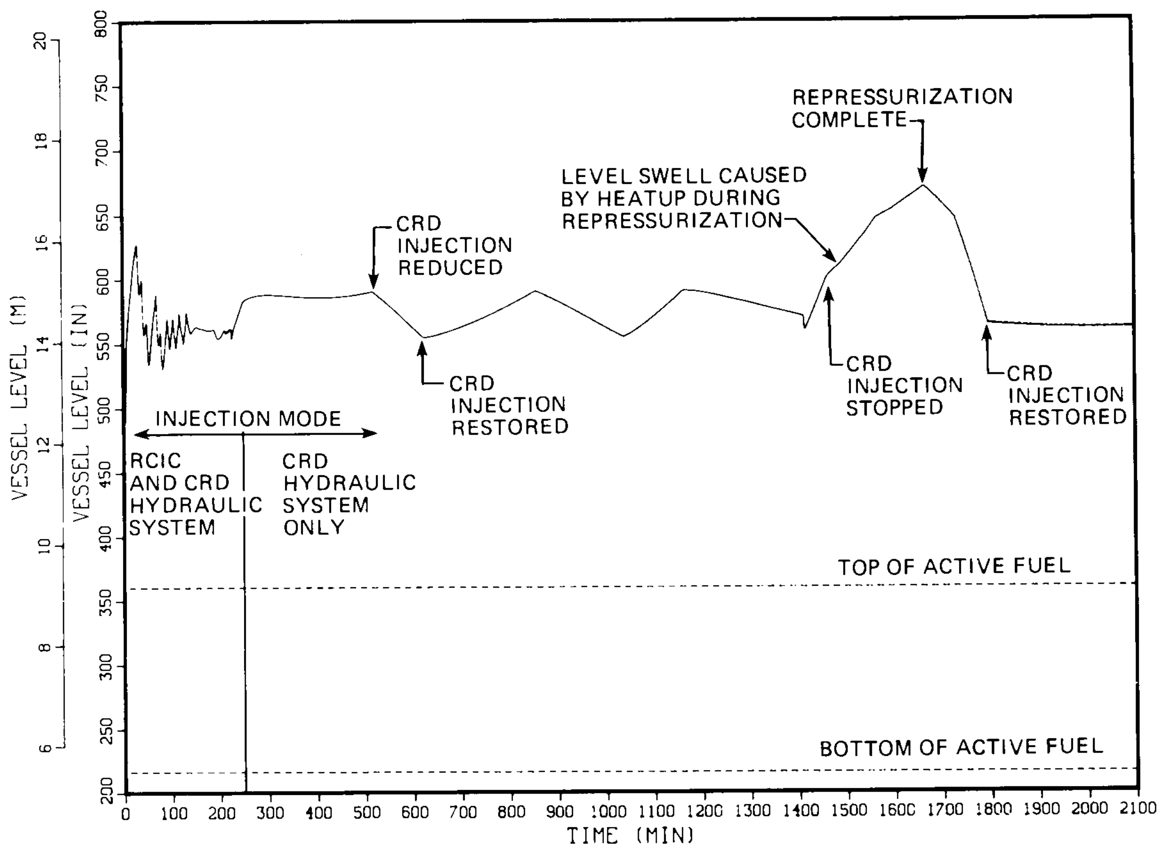

Fig. 3.1. Unmitigated Loss of DHR – reactor vessel water level.

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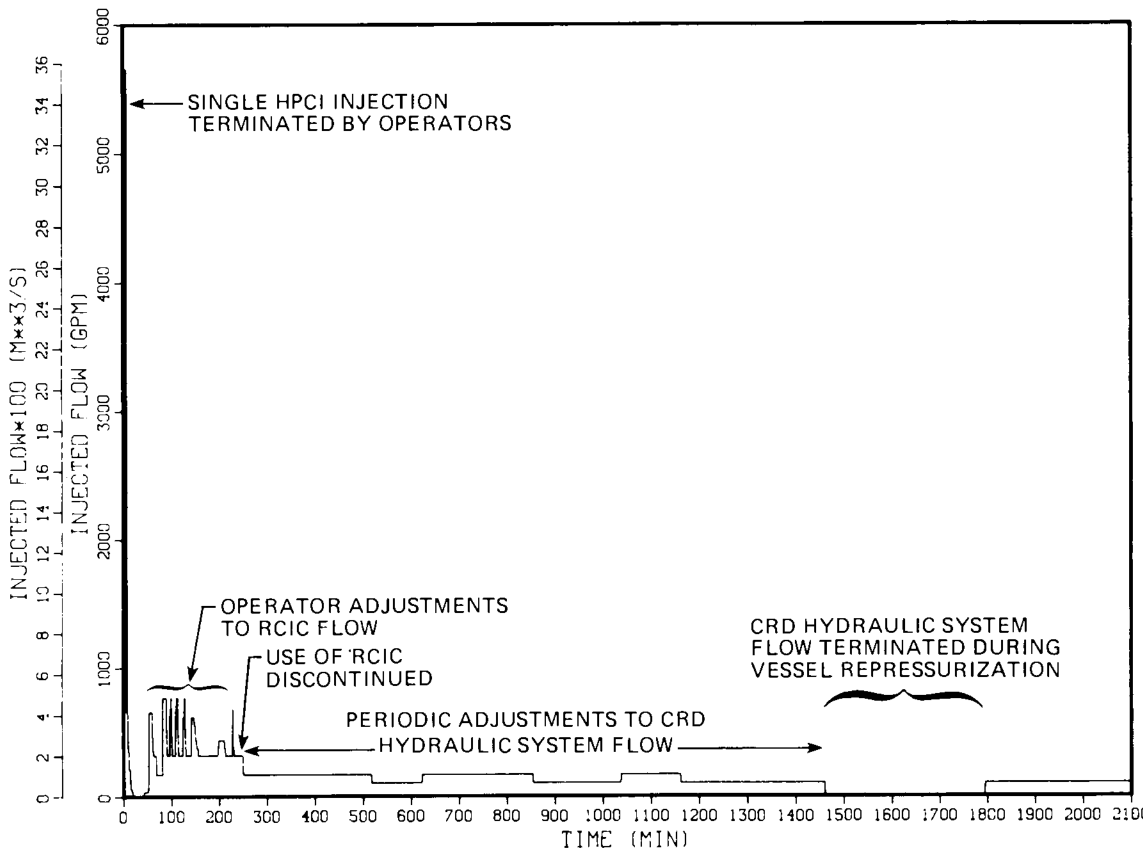

Fig. 3.2. Unmitigated Loss of DHR — total rate of injection flow.

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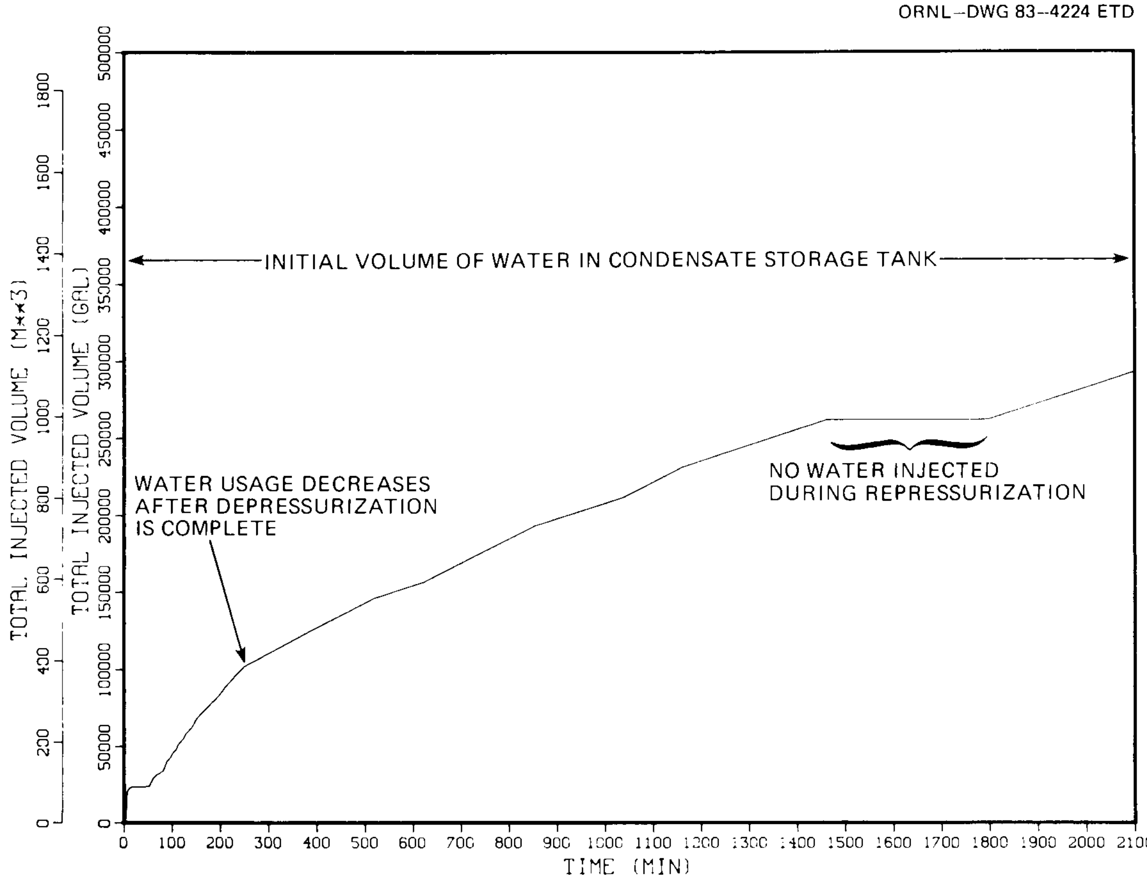

Fig. 3.3. Unmitigated Loss of DHR – total volume injected into reactor vessel.

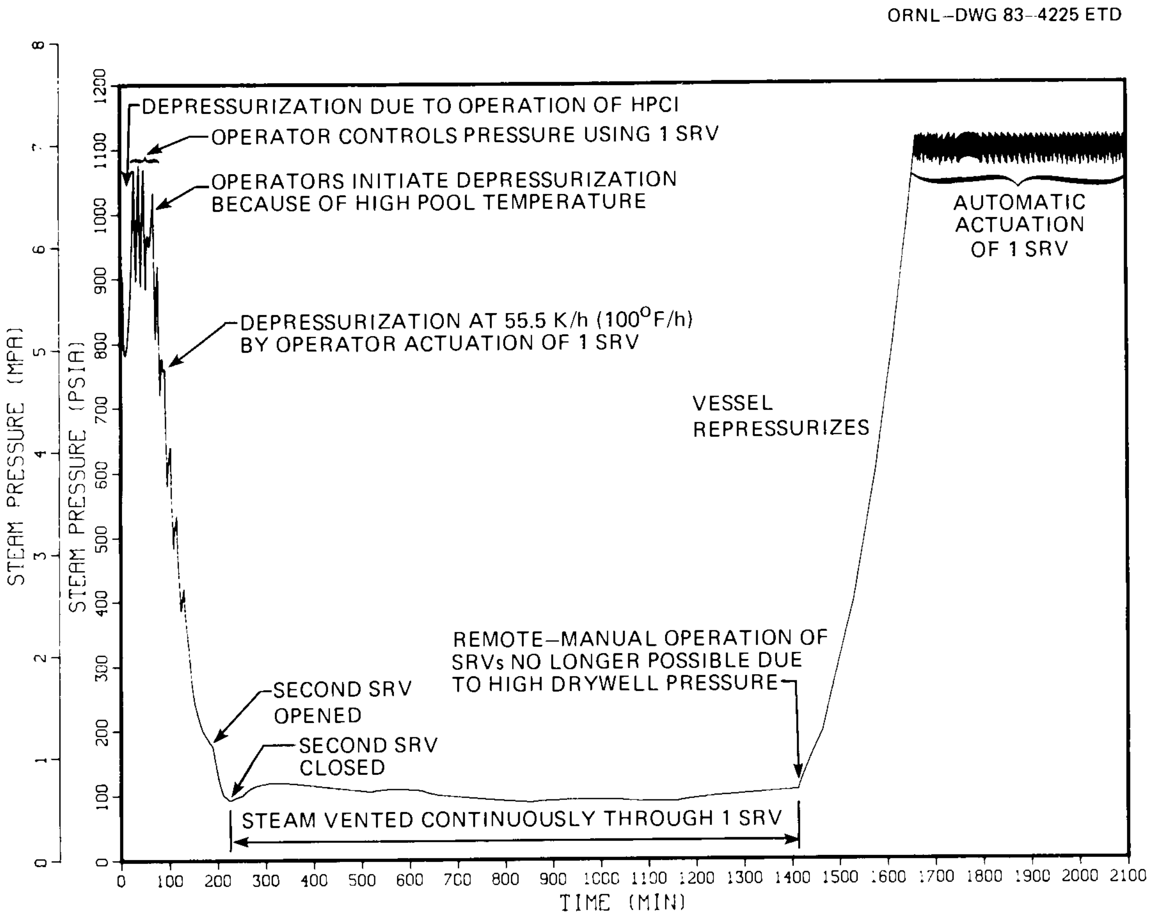

Fig. 3.4. Unmitigated Loss of DHR – reactor vessel steam pressure.

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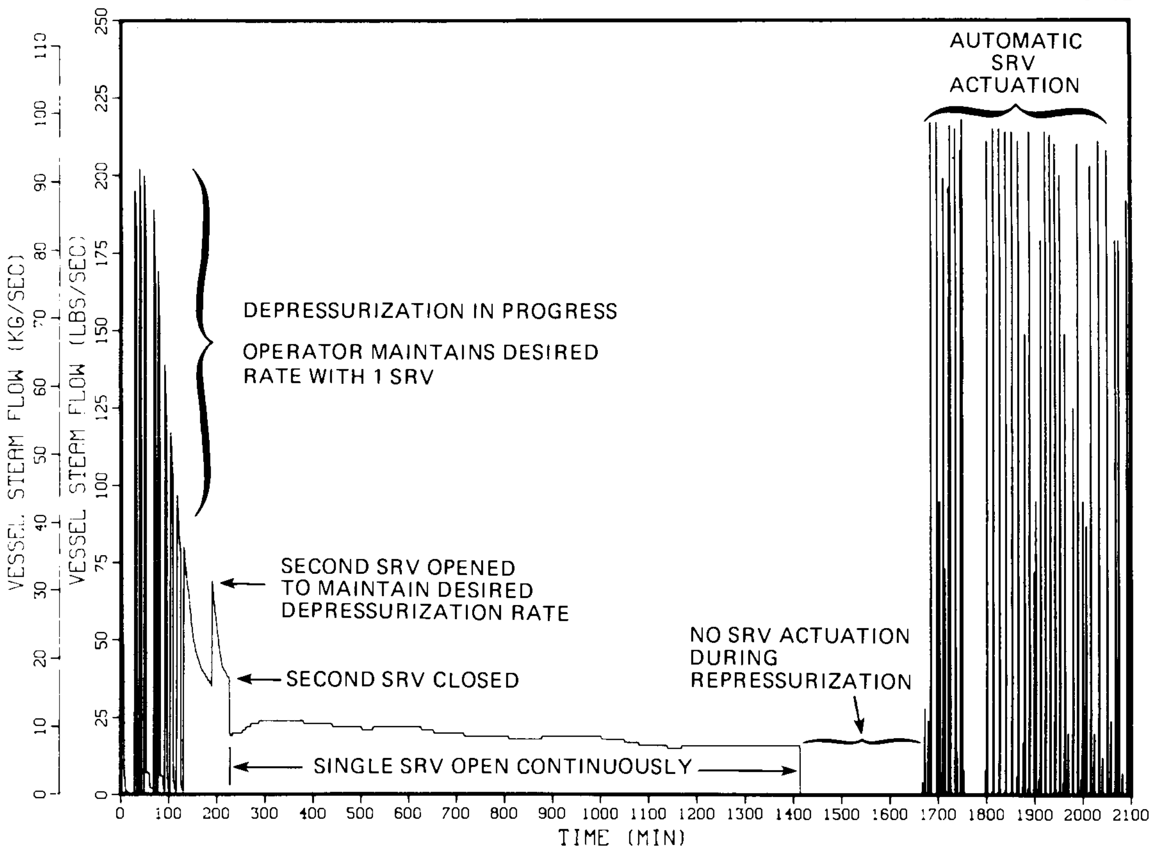

Fig. 3.5. Unmitigated Loss of DHR – total steam flow from reactor vessel.

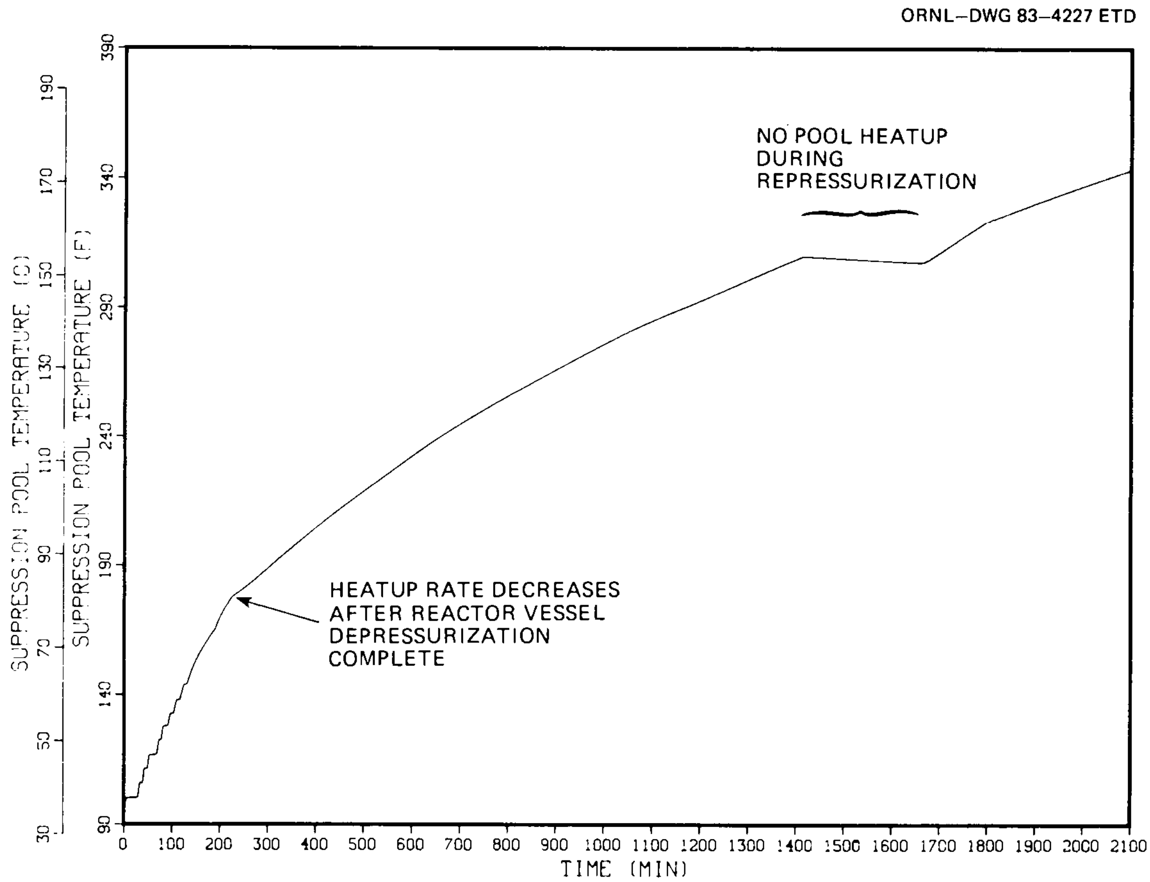

Fig. 3.6. Unmitigated Loss of DHR – suppression pool temperature.

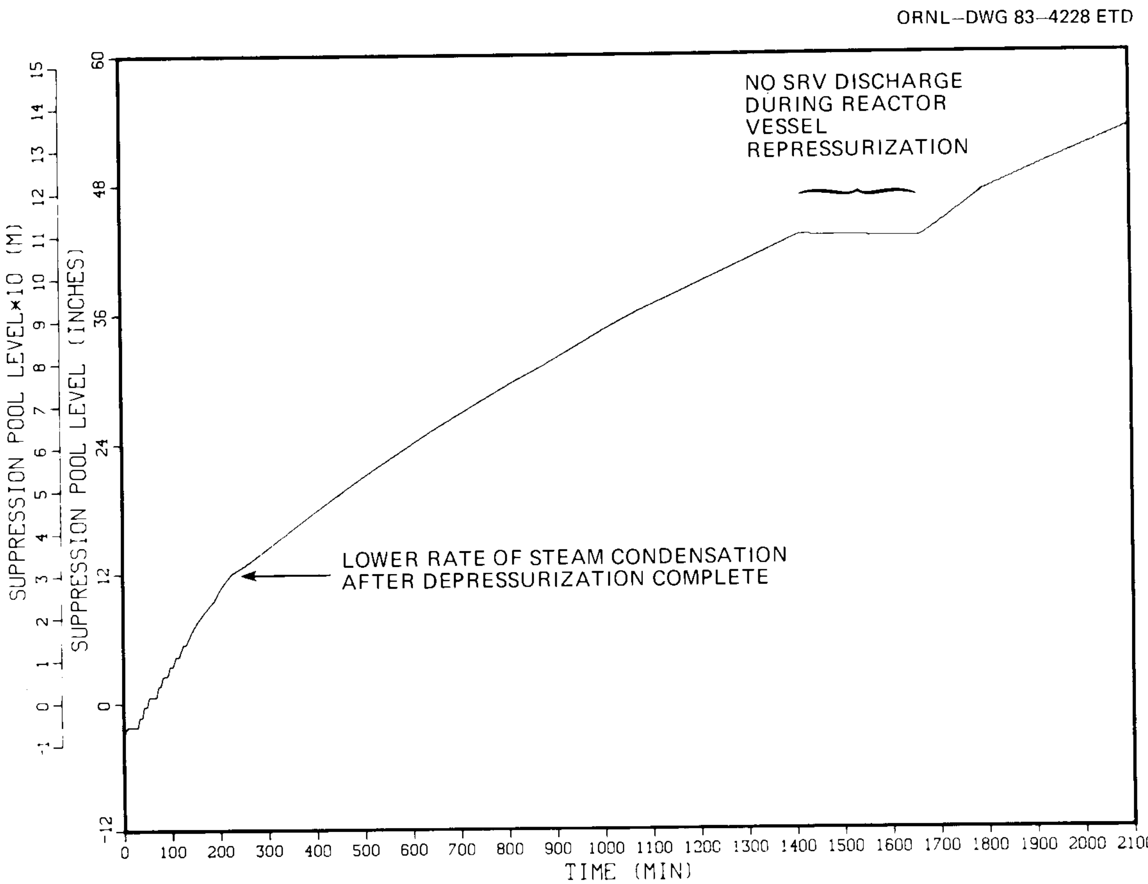

Fig. 3.7. Unmitigated Loss of DHR – suppression pool water level.

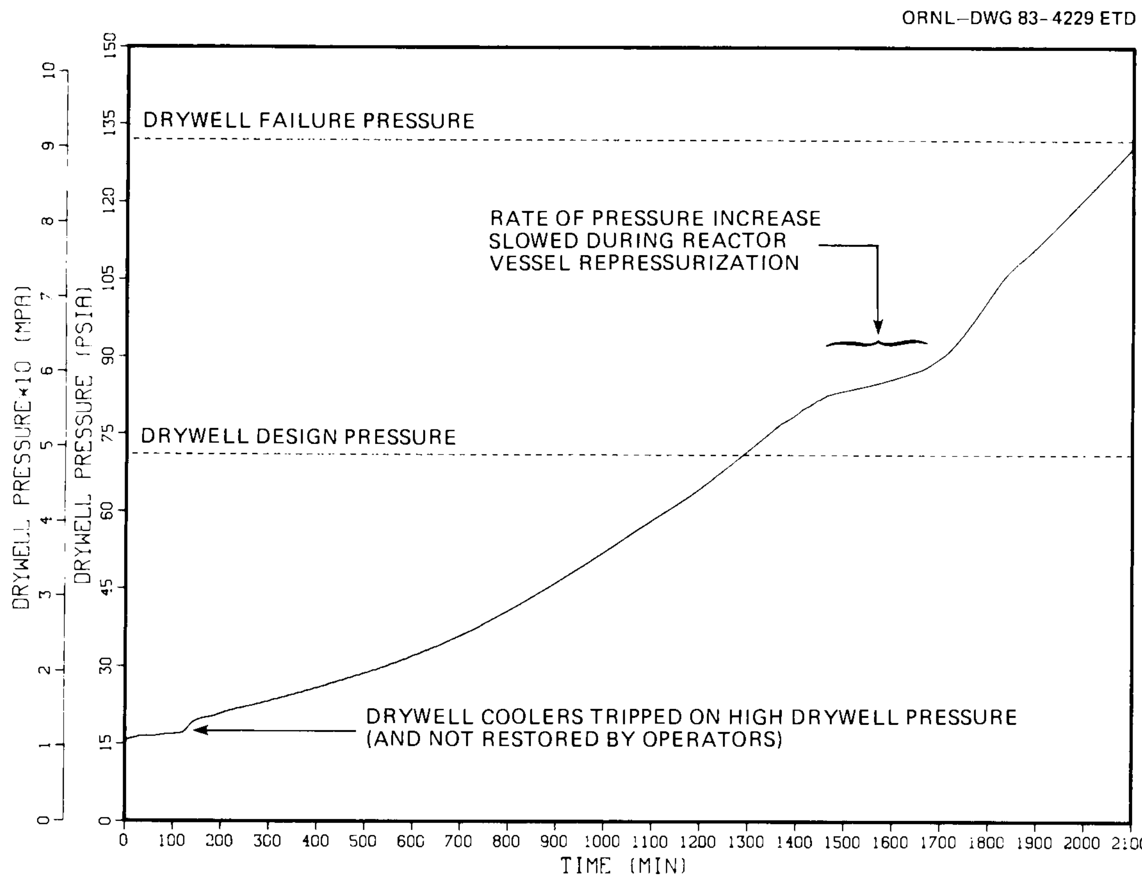

Fig. 3.8. Unmitigated Loss of DHR - drywell pressure (with early trip of drywell coolers).

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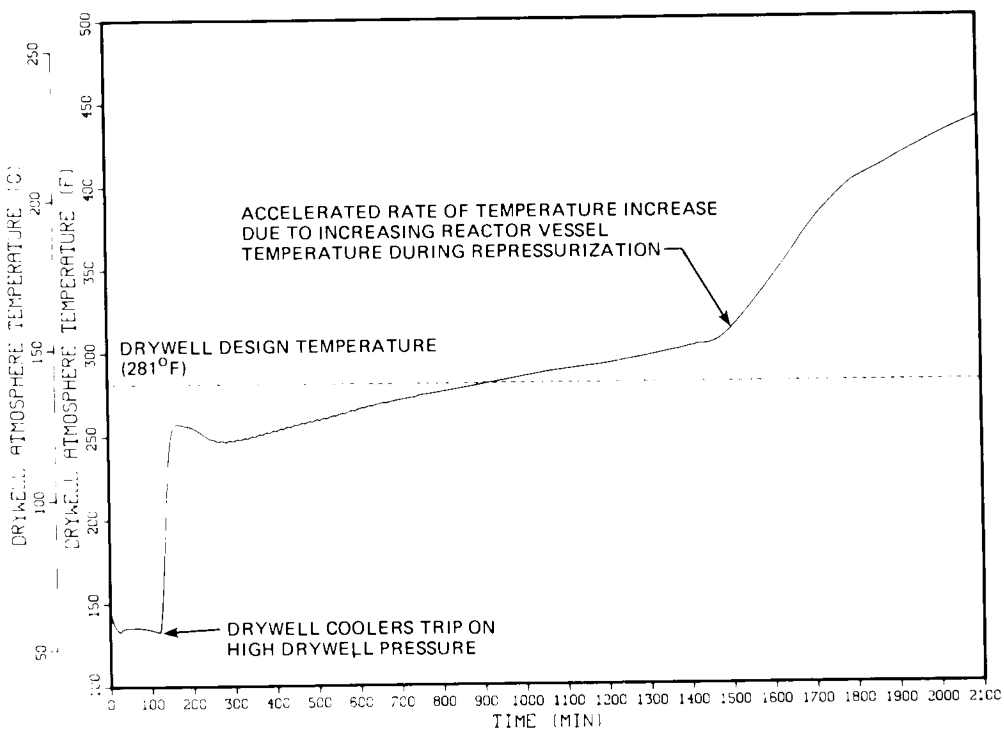

Fig. 3.9. Unmitigated Loss of DHR - drywell temperature (with early trip of drywell coolers).

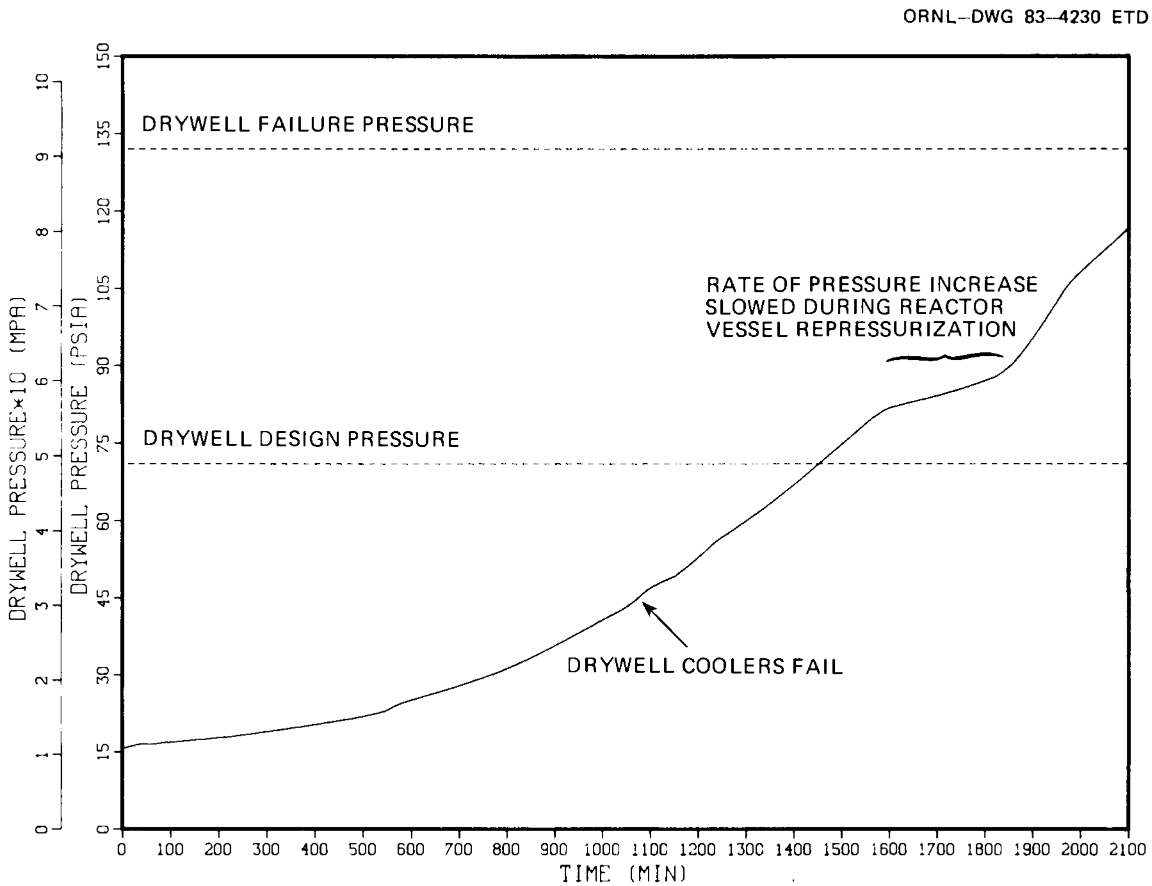

Fig. 3.10. Unmitigated Loss of DHR – drywell pressure (drywell coolers operated until failure).

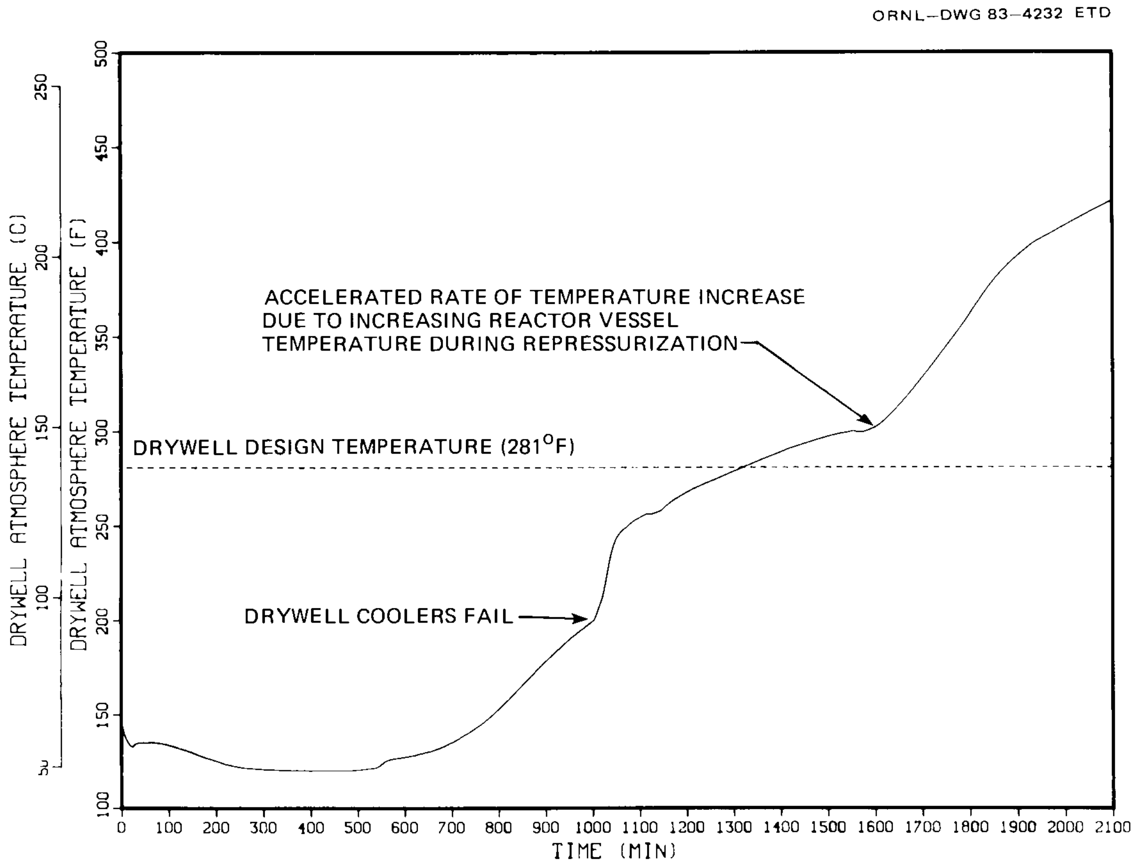

Fig. 3.11. Unmitigated Loss of DHR - drywell temperature (drywell coolers operated until failure).

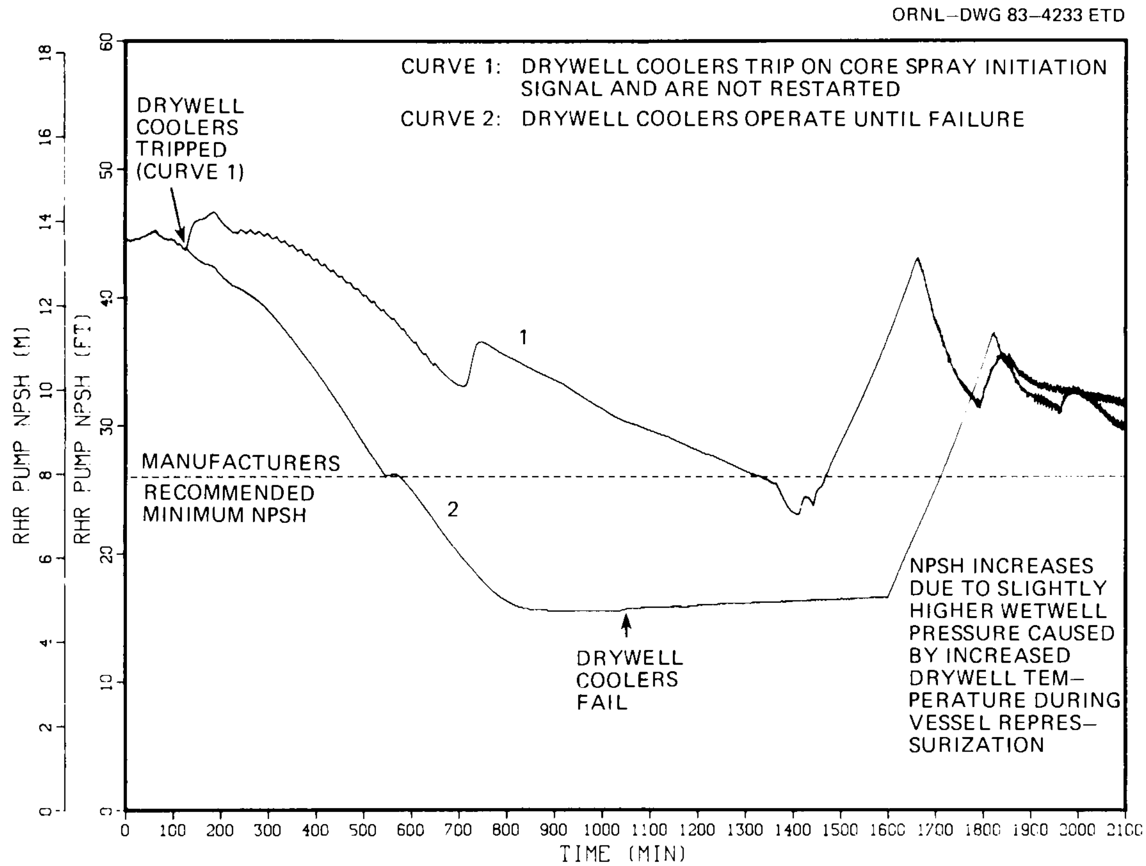

Fig. 3.12. Unmitigated Loss of DHR - RHR pump net positive suction head.

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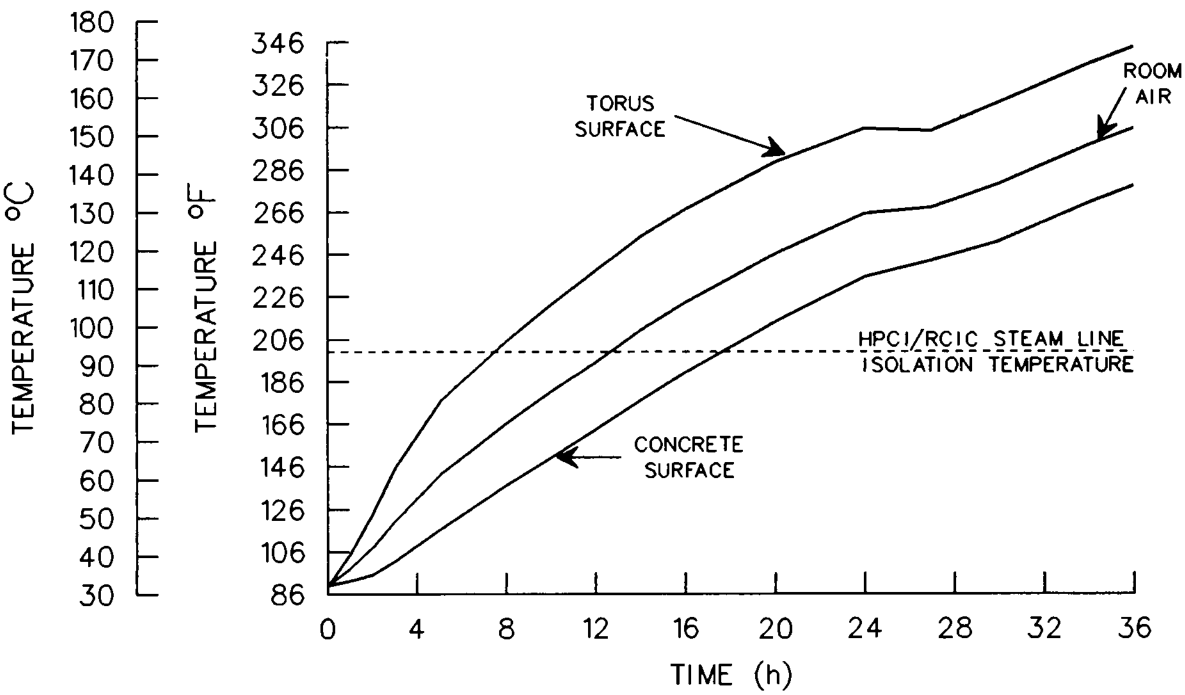

Fig. 3.13. Average torus room temperature during the Loss of DHR accident sequence.

4. NORMAL RECOVERY FROM AND MITIGATION OF LOSS OF DHR FUNCTION

4.1 Summary

Recovery of either the main condenser or the RHR system at any time in the period of more than 24 h before containment failure would preclude further increase in suppression pool temperature and therefore also prevent the eventual primary containment failure. Both the main condensate pumps and the RHR system can be powered from the diesel generators in the event of loss of off-site power.

Any one of the four Unit 1 RHR heat exchangers is capable of preventing the suppression pool from reaching excessive temperatures. In addition (see Appendix A.5.5), two of the four Unit 2 RHR pumps and heat exchangers can be aligned to cool the Unit 1 pool, and there are procedures and training to instruct operators in the use of this option.

Suppression pool temperature and primary containment pressure continue to increase in an unmitigated loss of DHR accident. When drywell pressure reaches half of design pressure [0.241 MPa (20 psig)] the operators would initiate primary containment sprays (see Appendix A.3). This action is required by both the Browns Ferry emergency operating instructions and the Emergency Procedure Guidelines.[4.1]

The spraying of hot suppression pool water* would result in a temporary pressure decrease (the magnitude of which would depend on circumstances prevailing at the time of initiation), but would not be capable of preventing the eventual containment overpressure failure. The maximum pressure decrease due to use of containment sprays would occur if the pool were uncirculated (as discussed in Chap. 5). In this case the thermal stratification effect would result in hotter water at the surface of the pool and cooler water at the bottom. The RHR pump suction is near the bottom of the pool. Initiation of containment sprays with a stratified suppression pool would therefore cause a significant temporary pressure decrease, but the spraying process itself would promote some pool mixing and the pool would continue to heat due to continued condensation of the SRV discharge; therefore, the containment pressure would continue to increase and the time of containment failure would be only slightly postponed.

The initiation of containment sprays (even using the hot suppression pool water) would have the significant long term beneficial effect of minimizing the temperature of the drywell atmosphere, which exceeds 204°C (400°F) in the latter stages of the Loss of DHR accident (Figs. 3.9 and 3.11).

There are two non-standard operational strategies that could prevent the eventual primary containment failure that would otherwise be the result of an unmitigated loss of DHR accident. One would be to open the drywell and/or suppression pool vent lines early in the sequence, to prevent an ultimate catastrophic containment failure. The other would be to

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*Use of the RHRSW system to spray cool river water into the primary containment is not considered here.

provide alternative pool cooling by a feed-and-bleed maneuver which would feed cool river water into the torus and reject heated water from the torus. These operational strategies have not been evaluated by either the utilities or the NRC, so it is unlikely that they would be employed under accident conditions.

4.2 Minimum Number of RHR Coolers Necessary for Pool Cooling

To determine if one RHR cooler would adequately cool the suppression pool, a calculation, very similar to those reported in Sect. 3, was performed with the same input except that one RHR heat exchanger was assumed to start after 0.5 h and run continuously thereafter with nominal RHR flow of 0.63 m^3/s (10,000 gpm) and RHR service water (RHRSW) flow of 0.28 m^3/s (4,500 gpm). The design RHR heat exchanger heat transfer coefficients (which include substantial fouling allowances on both tube and shell side) were used. The calculated peak pool temperature (Fig. 4.1) was 82°C (179°F), occurring after 10 h. In order to determine the sensitivity of this result to conservative assumptions, the calculation was repeated with RHR heat exchanger effectiveness degraded by 10%, and with the nominal ANS (1979) standard decay heat (with actinides) increased by a factor of 1.2 during the first 900 s and by a factor of 1.1 after 900 s.* These conservatisms increased the peak pool temperature from 82°C (179°F) at 10 h to 88°C (191°F) at 11 h. Either of these peak temperatures would be acceptable. The more conservative result is similar to a calculation performed by the TVA (in response to 10 CFR 50 Appendix R - Fire Protection requirements) which also showed that a single RHR heat exchanger† is adequate to keep pool temperature below 93°C (200°F).

An initial delay of an hour or two before starting suppression pool cooling would have little effect on peak pool temperature since the heat transfer in the RHR heat exchanger increases in direct proportion to pool temperature. A more substantial delay would lead to undesirably high pool temperatures and possibly to an insufficient RHR pump net positive suction head (NPSH). As discussed in Sect. 3.3.5, the NPSH of the RHR pumps can be maintained above the recommended minimum by decreasing RHR flow from full capacity to 0.5 m^3/s (8000 gpm). The RHR pumps and heat exchangers are designed for water temperatures up to 177°C (350°F) and pressure to 3.21 MPa (450 psig), so they should be able to function at any time during an unmitigated loss of DHR sequence before primary containment failure.

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*These multiplying factors were chosen to confirm with those used in an existing TVA calculation.

†In the TVA calculations, the heat exchanger was started after 1 h, but it was assumed that the associated RHR pump was also being used to provide vessel water injection. Therefore, pool cooling was interrupted periodically to allow for alignment to vessel injection followed by an assumed 10 min delay for realignment to pool cooling.

4.3 Mitigation Measures

4.3.1 Primary containment venting

This section investigates the possibility that primary containment venting could prevent the drywell over-pressure failure which eventually occurs in an unmitigated loss of DHR accident.

The primary containment ventilation and inerting systems are shown in Fig. 4.2. The Browns Ferry design includes two 5.1 cm (2 in.) lines (one from the drywell and one from the wetwell) that come together into a common line (of the same size) that is connected directly to Standby Gas Treatment (SGT) System ductwork. The flow of primary containment atmosphere into these lines is controlled by globe valves 84-19 and 84-20 as shown on Fig. 4.2. These vent lines can be used for minor pressure adjustments during normal operation, but they are also designed for high pressure use in the containment reinerting operation which might be required after a loss of coolant accident. The control valves automatically isolate when drywell pressure exceeds 0.115 MPa (2 psig), but the operators can over-ride the isolation from the main control room.

The 5.1 cm (2 in.) vent lines are obviously not large enough to hold primary containment pressure near atmospheric during an extended Loss of DHR accident sequence. In order to be able to prevent containment failure they would have to be capable of venting the total decay heat steam production at some pressure below containment failure pressure. As reported in Chap. 3, it takes about 35 h for drywell pressure to exceed 0.91 MPa (117 psig), at which time the decay heat steam production is about 7.3 kg/s (16 lb/s). The drywell and wetwell vents, combined, can pass only about 1.14 kg/s (2.5 lb/s) at this elevated pressure. To determine the maximum effect of this venting path, the Loss of DHR calculations of Chap. 3 were repeated with the assumption of continuous venting through these lines during the accident sequence. The results showed that drywell failure would be delayed by about 4 h (i.e. delayed from 35 to 39 h.)

The Browns Ferry design also includes two 46 cm (18 in.) lines (one from the drywell, and one from the wetwell) that can vent the primary containment directly to the main ventilation system ductwork. The flow through these lines is controlled by 46 cm (18 in.) butterfly valves 64-29 and 64-32 as shown on Fig. 4.2. These ventilation system lines are used during shutdown when the primary containment is being inerted with nitrogen prior to startup or when the nitrogen inerted primary containment atmosphere is being purged with air prior to personnel entry. They are not intended for use under high pressure [procedures require primary containment pressure below 0.103 MPa (0.25 psig) prior to venting]. The 46 cm (18 in.) buttefly valves isolate when drywell pressure exceeds 0.115 MPa (2 psig); the operators cannot over-ride the isolation signal.

If the operators, during the early part of a Loss of DHR accident, were to open the 46 cm (18 in.) butterfly valves before they were automatically shut and held shut on a high drywell pressure isolation signal, the primary containment pressure could be held very near atmospheric pressure and subsequent failure of the drywell by over-pressurization would be prevented. As reported in Sect. 3.3.1, the high drywell pressure signal would occur 1 h after event initiation. However, if the wetwell-to-drywell ΔP compressor were not operated following the initiating event and the 5.1 cm (2 in.) vents were opened to slow the pressure rise, then it would take about 3.8 h, instead of 1 h, for the high drywell pressure signal to occur (see the discussion of Item 8 in Sect. 10.2). Therefore, the operators would have, at most, 3.8 h in which to consider whether to initiate the option of venting through the 46 cm (18 in.) lines.

The suppression pool temperature reached in a loss of DHR sequence with a vented primary containment at atmospheric pressure would be limited to 100°C (212°F). After the pool reached saturation, no more SRV exhaust steam would be condensed in the pool; the steam would escape from the pool surface and would have to be vented from the primary containment. The amount of decay heat-generated steam being produced at the time the pool temperature reaches 100°C (212°F) is about 10 kg/s (22 lb/s), or 16.5 m^3s (35,000 cfm) at atmospheric pressure. This is the maximum steam venting rate that would occur during the accident sequence with vented containment, and is about 25% above the SGT system blower capacity, but well below the main ventilation system blower capacity. Therefore, the main ventilation system would have to be operational to avoid a steam environment in the reactor building. The 16.5 m^3/s (35,000 cfm) of steam would flow at a bulk velocity of 101 m/s (333 ft/s) in the 46 cm (18 in.) vent line. This high velocity would indicate some possibility of damage to the ductwork, and the attendant risk of the release of steam into the reactor building.

The possibility of ventilation system damage caused by releasing the decay heat-generated steam through the reactor building ventilation system must be weighed against inaction and the attendant risk of severe damage to the primary system caused by the subsequent catastrophic primary containment failure by overpressurization. The possible post containment failure phenomena are discussed in Chap. 7.

4.3.2 Alternative pool cooling

This section investigates the possibility that a feed-and-bleed suppression pool cooling method could limit pool temperature sufficiently to prevent the eventual containment failure otherwise caused by an unmitigated Loss of DHR accident. The direct addition of cool water with the removal of a corresponding amount of heated pool water* would cool the pool in a manner similar to that of the closed-cycle pool cooling mode of the RHR system. No emergency operating instruction (EOI) for such a procedure currently exists; the following brief analysis is only intended to establish the feasibility of the approach and to offer suggestions as to possible methods of implementation.

In an unmitigated loss of DHR accident, pool temperature slowly increases until containment failure. If a sufficient flow of feed-and-bleed pool cooling could be started, the trend of increasing pool temperature would be stopped or reversed. The heat removal by feed-and-bleed cooling is directly proportional to the difference between the feed temperature

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*In order for pool water level to remain constant, the bleed flow rate would have to exceed the feed flow rate because the bleed flow would include both the feed flow and the condensed SRV discharge.

[assumed to be 32°C (90°F) for the calculations presented here] and the bleed (i.e. bulk pool) temperature. As pool temperature increases, the temperature difference between the feed and the bleed temperatures becomes greater, so less cooling flow is required to remove the heat associated with the decay heat-generated steam production.

This effect is quantified in Table 4.1, which specifies the suppression pool temperatures (taken from the Sect. 3 calculations) reached with no pool cooling, along with the results of a simple heat-balance calculation of the minimum flow of feed-and-bleed cooling that would prevent further temperature increase. For example, in the first 5 h following the loss of DHR, the suppression pool temperature would (without any pool cooling) increase from 32°C (90°F) to 87°C (189°F). If a feed-and-bleed cooling flow of at least 0.13 m^3/s (2020 gpm) were begun at 5 h, there would be no further increase in bulk pool temperature*.

The RHR service water (RHRSW) pumps might be operable in a loss of DHR accident; if so, they could be used to feed river water directly into the suppression pool (see also Appendix A.5.2). Each of the four Unit 1 RHRSW pumps can pump 0.28 m^3/s (4500 gpm) against a head of 0.93 MPa (120 psig), and could therefore accomplish feeding even with primary containment pressurized to the drywell failure point of 0.91 MPa (117 psig). The RHRSW pumps can be powered from the diesel generators, so off-site power is not necessary for this operation. The river water would be fed into the pool via the 46 cm (18 in.) recirculation pump test line which is also used to discharge cooled water into the pool when the RHR system is in the normal pool cooling mode.†

The RHR drain pumps are used for routine suppression pool level adjustment. Each of the two drain pumps can pump 0.05 m^3s (800 gpm), with a developed head of 46 m (150 ft), to the main condenser hotwell or and/or to the radwaste system. These pumps might be operable in a loss of DHR sequence, and, if so, they could remove hot water from the pool for feed-and-bleed cooling.

An alternative pool cooling strategy, based on 0.1 m^3/s (1600 gpm) of RHRSW feed of river water with the RHR drain pumps used to bleed the same amount of hot pool water, would, if initiated after 7 h, prevent pool temperature from exceeding about 99°C (210°F) (see Table 4.1). If this same cooling were initiated before 7 h, the peak pool temperature would be lower.

The risks attendant to feed-and-bleed pool cooling would, in general, be in proportion to how much time was allowed to elapse before beginning the procedure. For example, after 12 h without pool cooling, the pool would have reached a temperature of 119°C (247°F) and a pressure of 0.25 MPa (21 psig). If this hot water were bled from the pool, its pressure would have to be lowered to atmospheric pressure at some point, and about 3.5% would flash to steam. This flashing might occur inside the piping

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*If the same flow of feed were initiated but without bleed flow, the pool temperature increase would be prevented but the water level of the pool would increase and become excessively high after several hours.

†As noted in Sect. A.5.2, the RHRSW pump discharge can also be directed into the containment spray headers. It is assumed here that this path is not available.

system used for the bleeding operation. If so, the steam might contact cool water residing in the piping before initiation of the bleed, resulting in violent condensation and possibly pipe rupture. The postulated rupture would then cause a flooding and steam environment hazard, with the seriousness of the hazard depending on break size and location (the reactor zone would be the worst location).

In any feed-and-bleed cooling technique, the sheer volume of bleed created would cause difficulty. However, the pool heatup is slow and there should be sufficient time during a Loss of DHR accident sequence to develop a practical strategy. The ideal receptors for the bleed would be the two 1893 m^3 (500,000 gal) pressure suppression pool water storage tanks. If the pool water cannot be transferred to these tanks, then it might be moved into the Unit 2 pressure suppression pool or some flooding might be acceptable if it could be limited to a non-critical location.

The risks of feed-and-bleed pool cooling would have to be weighed against the risk of letting the Loss of DHR accident sequence proceed to primary containment failure by overpressurization, which is discussed in Chap. 7.

Reference for Section 4

4.1 "Emergency Procedures Guidelines BWR 1 through 6," NEDO-24934, Revision 2, Prepublication draft, May, 1982.

Table 4.1. Minimum feed flow required in a feed-and-bleed cooling scheme to prevent suppression pool temperature increase during Loss of DHR accident

Time elapsed since event initiation (h)	Pool temperature[a] reached without any cooling [°C (°F)]	Required feed flow[b,c,d] [m³/s (gpm)]
2	61 (141)	0.32 (5000)
3	73 (163)	0.20 (3130)
5	87 (189)	0.13 (2020)
8	102 (216)	0.09 (1400)
12	119 (247)	0.06 (1010)
16	133 (272)	0.05 (810)

[a]Taken from the calculation reported in Sect. 3.3.3.

[b]Minimum required to prevent further increase in pool temperature. No flow until corresponding elapsed time, then continuous flow thereafter at the indicated rate.

[c]If suppression pool water level is to be maintained constant, the volumetric bleed flow would have to exceed by 10–20% the feed flow. Bleed flow would include both the condensed SRV discharge as well as the feed flow.

[d]Calculated by simple heat balance, assuming 32°C (90°F) feed, and decay heat steam production given by 1979 ANS Standard Decay Heat (with Actinides, and without 1.1 conservatism factor).

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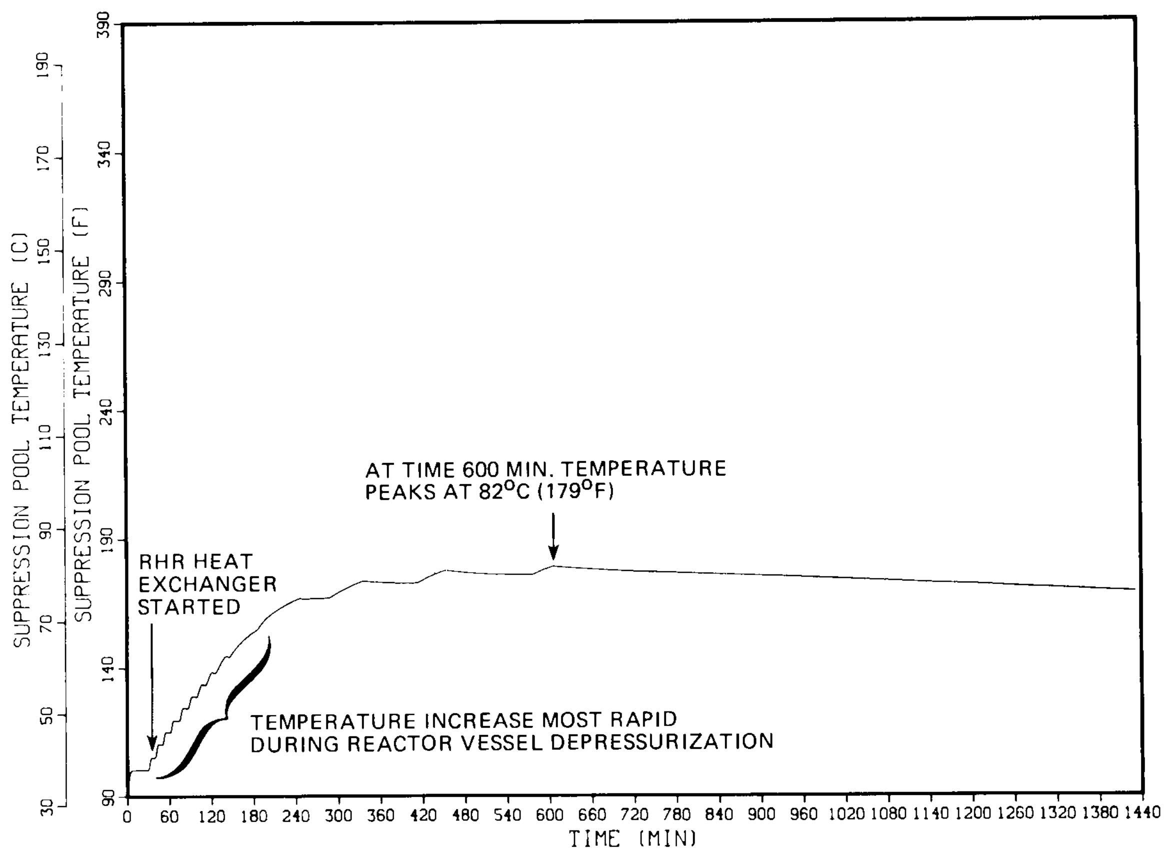

Fig. 4.1. Suppression pool temperature with one RHR heat exchanger in operation.

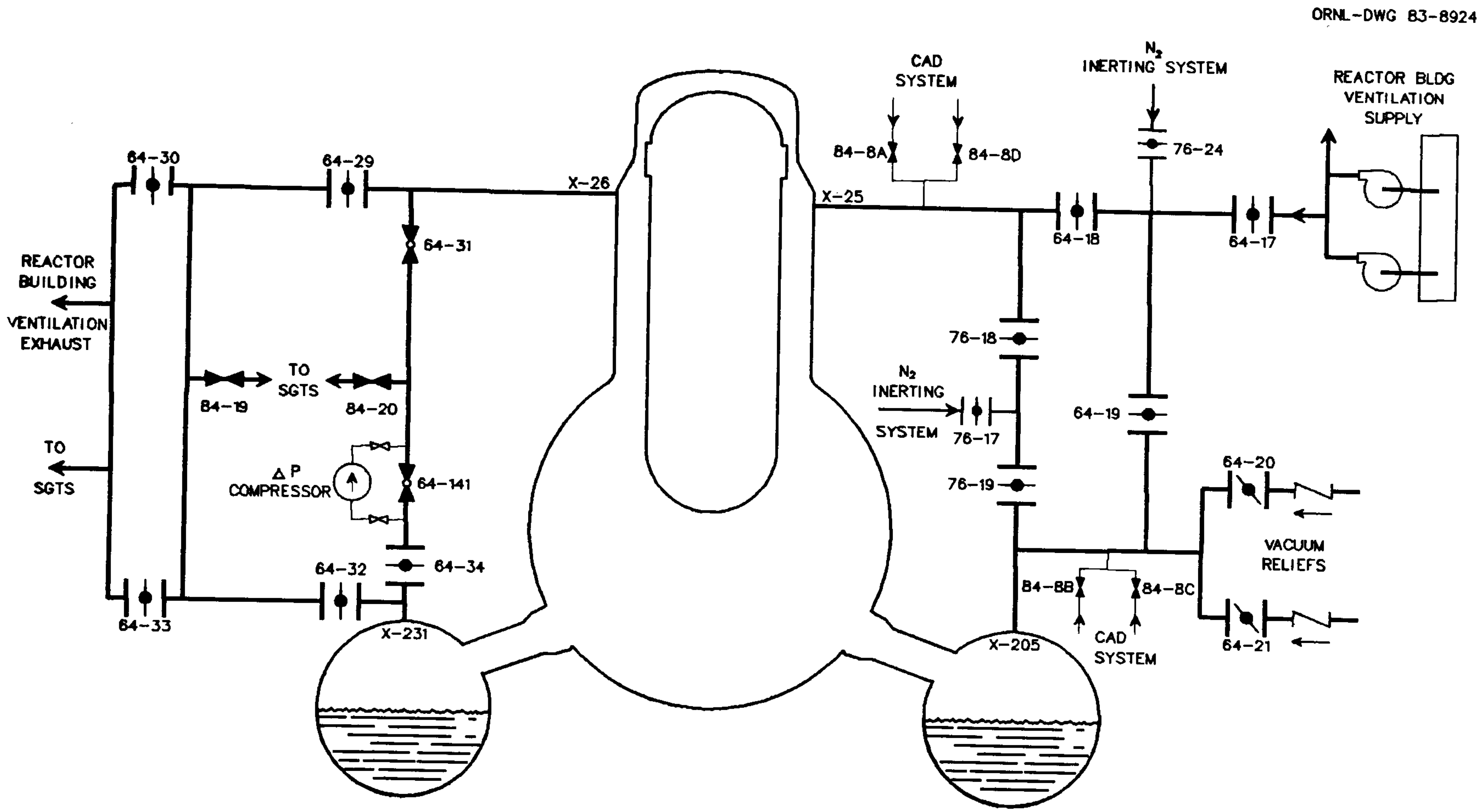

Fig. 4.2. Primary containment ventilation and inerting systems.

5. TRANSIENTS WITH LOSS OF DHR: CALCULATIONS PERFORMED FOR THE CASE OF THERMAL STRATIFICATION IN THE SUPPRESSION POOL

5.1 Introduction

The accident sequence considered in this section is exactly the same as that considered in Sect. 3, except that the assumption of uniform pool temperature is not made. The uniform pool temperature assumption is equivalent to the assumption that the RHR pumps are operational (even though the pool cooling is not) and that they are used to circulate and mix the pool.

Without forced circulation of the pool, the average water temperature in the locality of a discharging T-quencher ("local" temperature) will be higher than the average temperature of the whole pool ("bulk pool" temperature). In addition, there will be thermal stratification throughout the pool (i.e., the temperature of the water at the surface of the pool will be higher than the bulk pool temperature). The Monticello T-quencher tests[5.1] showed that local temperature exceeds bulk pool temperature by as much as 24°C (43°F) during extended SRV discharge, and that there is considerable thermal stratification throughout the pool.

During a loss of DHR accident sequence the higher surface and local temperatures will increase the rate of escape of steam from the pool to the wetwell atmosphere, and thereby decrease the time required for pressure to build to the point of primary containment failure. To study these effects, a detailed thermal-hydraulic model of the suppression pool was developed (see Appendix C for a description) and used to calculate the spatial temperature distribution of the pool during the Loss of DHR accident sequence. This model was utilized independently for the calculations reported in this chapter. The transient input to the pool model consisted of the SRV flow, the total wetwell pressure, and the evaporation rate from the pool surface throughout the sequence. This input was taken from the BWR-LACP calculations of Chap. 3.

5.2 Summary

When there is no forced circulation of the suppression pool, an unmitigated loss of DHR could lead to primary containment failure by overpressurization as early as 28 h after reactor shutdown. This is seven hours sooner than for the case with uniform suppression pool temperature.

5.3 Detailed Results

The purpose of this section is to provide an estimate of how much sooner the containment would fail in a Loss of DHR sequence if the RHR system is not circulating the pool so that the assumption of uniform pool temperature is not valid. The subsections that follow specify the input assumptions for the pool temperature distribution calculation, present the calculated temperature distribution, and evaluate the effect of the temperature distribution on the time of containment failure.

5.3.1 Input and assumptions

The thermal-hydraulic model described in Appendix C was used to calculate the temperature distribution of the pool during the first 15 h of the 35-h-long base-case* Loss of DHR sequence (described in detail in Section 3.3). A longer calculation was not attempted because of numerical difficulty associated with the very low sustained SRV discharge rate reached after 15 h. In addition, the model is not presently programmed to calculate the condensation and transport of T-quencher discharge during the operating mode reached after about 24 h, in which the reactor vessel has repressurized and a single SRV is discharging intermittently, but at a high flow rate, into a pressure suppression pool which is close to saturation. Nevertheless, the 15 h of available calculational results provide considerable insight into pool behavior without forced circulation, and allow an estimate to be made of the effect on containment failure time.

The special suppression pool model was run independently for the 15 h calculation period. The following information was input as a function of time: rate of discharge of steam from the SRVs to the pool (Fig. 3.5), total pressure in the wetwell (essentially equal to the drywell pressure shown in Fig. 3.8), and the rate of evaporative steaming from the surface of the suppression pool (not shown — is equal to about seven percent of the total decay heat steaming rate at the 15 h point). It was not necessary to specify to the pool model which of the 13 SRVs were actuating because the model assumes that all the SRV discharge occurs through the same T-quencher at a constant, fixed location in the pool throughout the calculation. This assumption introduces an element of conservatism into the results between 0 and 24 h because during the first ~24 h of the Loss of DHR sequence, the operators are able to rotate their selection of SRVs around the pool as specified by the emergency operating procedures. After 24 h, the remote-manual SRV actuation capability is lost, and the SRV discharge after this time would occur by automatic actuation of the single, lowest-set SRV in the group of four SRVs that have an individually-set actuation pressure of 7.72 MPa (1105 psig). The discharge after this time would therefore enter the pressure suppression pool from the same T-quencher.

5.3.2 Transient pool temperature distribution

The special suppression pool model divides the toroidal pool into 16 equal angular segments which correspond to the 16 bays of the wetwell

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*Chapter 3 considers two Loss of DHR sequences: one with and one without operator restart of the drywell coolers after their automatic trip on high drywell pressure early in the sequence. The sequence without operator restart of the coolers is the base case — see Section 3.3.4 for more discussion of this point.

torus. The water in each of the 16 bays is further divided into four vertical regions at initially equal depth increments. A temperature is calculated for each of the 64 regions. The bulk pool temperature is the mass-weighted average of all 64 pool temperatures. The surface temperature is the average temperature of the 16 surface nodes. Local temperature applies to the region between 1.14 m (3.75 ft) and 2.13 m (7.0 ft) in the bay of SRV T-quencher discharge [the centerline of the T-quencher is 1.52 m (5 ft) from the bottom of the normally ~4.6 m (15 ft) deep pool]. The local temperature is a weighted average of both the very hot water within the plume of condensed SRV discharge as well as cooler water flowing toward the T-quencher before it makes contact with the discharging steam.

Figure 5.1 presents the bulk pool temperature, the average surface temperature, and the average temperature of the bottom regions of the pool. The difference between surface and bulk temperatures reaches a maximum of 13°C (23°F) after 4 h, and then declines steadily, reaching 5°C (9°F) at the end of the 15 h calculation period. This behavior is expected since the rate of SRV discharge is also declining throughout the period after 4 h (i.e., the depressurization of the reactor vessel is complete, and decay heat is decreasing).

Local pool temperature is plotted in Fig. 5.2. For comparison purposes, bulk pool temperature and average surface temperature are also shown in Fig. 5.2. The difference between bulk and local temperature is 10°C (18°F) at 2.75 h, and declines steadily thereafter. This shows that natural circulation is effectively distributing the decay heat energy from the bay of discharge into the other 15 bays around the circumference of the suppression pool.

5.3.3 Effect of pool temperature distribution on primary containment pressure build-up

The excessive pressure buildup in the Loss of DHR sequence is due to the escape of steam from the suppression pool, which can occur either by evaporative steaming from the heated surface of the pool or by failure of the pool water to condense all of the SRV T-quencher discharge. In the uniform pool temperature results of Chapter 3, evaporation was the primary means for the buildup of primary containment pressure. The thermal stratification which occurs when there is no circulation of the pool would accelerate the evaporative steaming from the pool surface. However, this effect alone could hasten primary containment failure by only about two hours, since the average surface temperature (Fig. 5.1) is only 5°C (9°F) above bulk pool temperature (and decreasing) after 15 h.

Complete condensation of the T-quencher discharge can only occur if the temperature of the water surrounding the T-quencher is sufficiently below the saturation temperature. NUREG-0783 (Ref. 5.2) specifies that a minimum subcooling* of 11°C (20°F) should exist to ensure stable condensation of SRV T-quencher discharge at rates not exceeding 205 kg/m² s [42

*The subcooling is defined as the difference between local temperature (see Sect. 5.3.2) and saturation temperature at the T-quencher depth, and thus is increased by the increase in local saturation temperature provided by the ~34.5 kPa (5 psid) static overpressure of the water above the T-quencher.

lb/(ft^2 s)], and that local temperature should not exceed 93°C (200°F) for SRV T-quencher discharge rates exceeding 460 kg/m^2 s [94 lb/(ft^2 s)]. These limits reflect the current extent of experimental determination of the conditions necessary for stable condensation. If local temperature exceeds either of these limits, there will not necessarily be unstable or incomplete condensation; on the other hand, stable condensation is assured as long is these limits are not exceeded.

For the Loss of DHR accident sequence, local subcooling (Fig. 5.3) starts at over 55°C (100°F) and has decreased to about 17°C (30°F) by the end of the 15 h calculation period. Subcooling would continue to slowly decrease after 15 h, and might be slightly below the 11°C (20°F) minimum subcooling requirement after 23.5 h, when the period of sustained, continuous SRV discharge ends due to the loss of remote-manual control of the SRVs;* however, it is likely that the pool would continue to completely condense the T-quencher discharge throughout the first 23.5 h of the Loss of DHR sequence.

After 23.5 h, the drywell pressure is too high to permit remote-manual SRV actuation and flow through the SRVs ceases for several hours while the reactor vessel repressurizes. When a reactor vessel pressure equal to the opening setpoint of the lowest-set SRV is reached (at 27.7 h), T-quencher discharge resumes but at a high rate of flow and in intermittent bursts instead of the essentially continuous low-flow discharge of the period before 23.5 h. With T-quencher discharge at high flow into an uncirculated and nearly saturated suppression pool, it is likely that the local subcooling would be well below 11°C (20°F) and might be lost entirely, allowing direct bubble-through of steam into the wetwell atmosphere. Without any condensation of SRV discharge, it would take about 20 min. to pressurize the primary containment from its 0.61 MPa (74 psig) pressure at 27.7 h to the 0.91 MPa (117 psig) primary containment failure pressure. Therefore, as a worst case, the containment failure would occur after 28 h instead of after 35 h.

References for Section 5

5.1 B. J. Patterson, "Mark I Containment Program — Monticello T-Quencher Thermal Mixing Test Final Report," GE NEDO-24542 Class I, August 1979.

5.2 T. M. Su et al., "Suppression Pool Temperature Limits for BWR Containments," USNRC Report No. NUREG-0783, November 1981.

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*As discussed in Sect. 3.3.2, control air pressure must be at least 0.17 MPa (25 psi) higher than drywell pressure to permit remote – manual actuation of the SRVs.

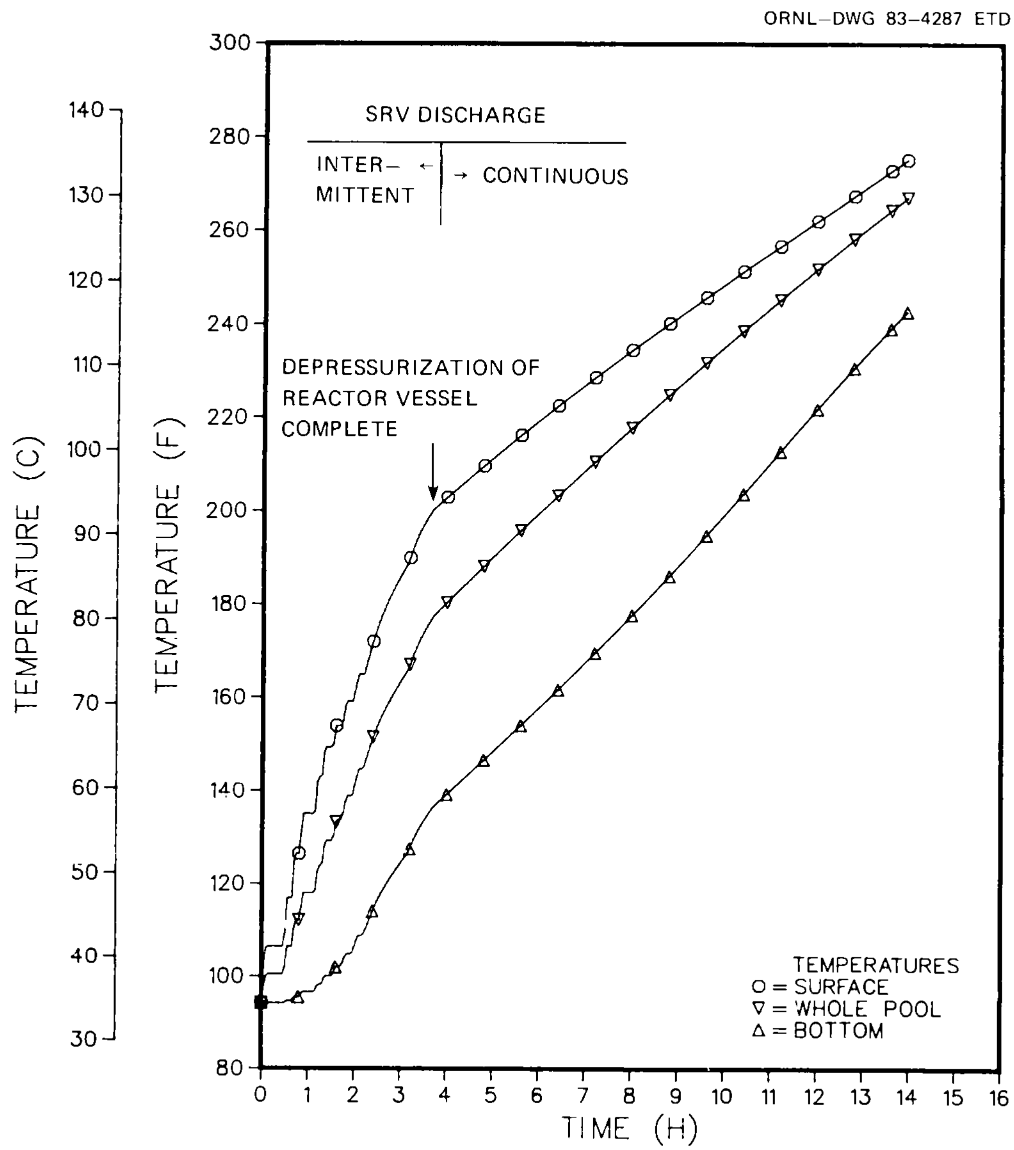

Fig. 5.1. Unmitigated Loss of DHR – suppression pool average temperatures.

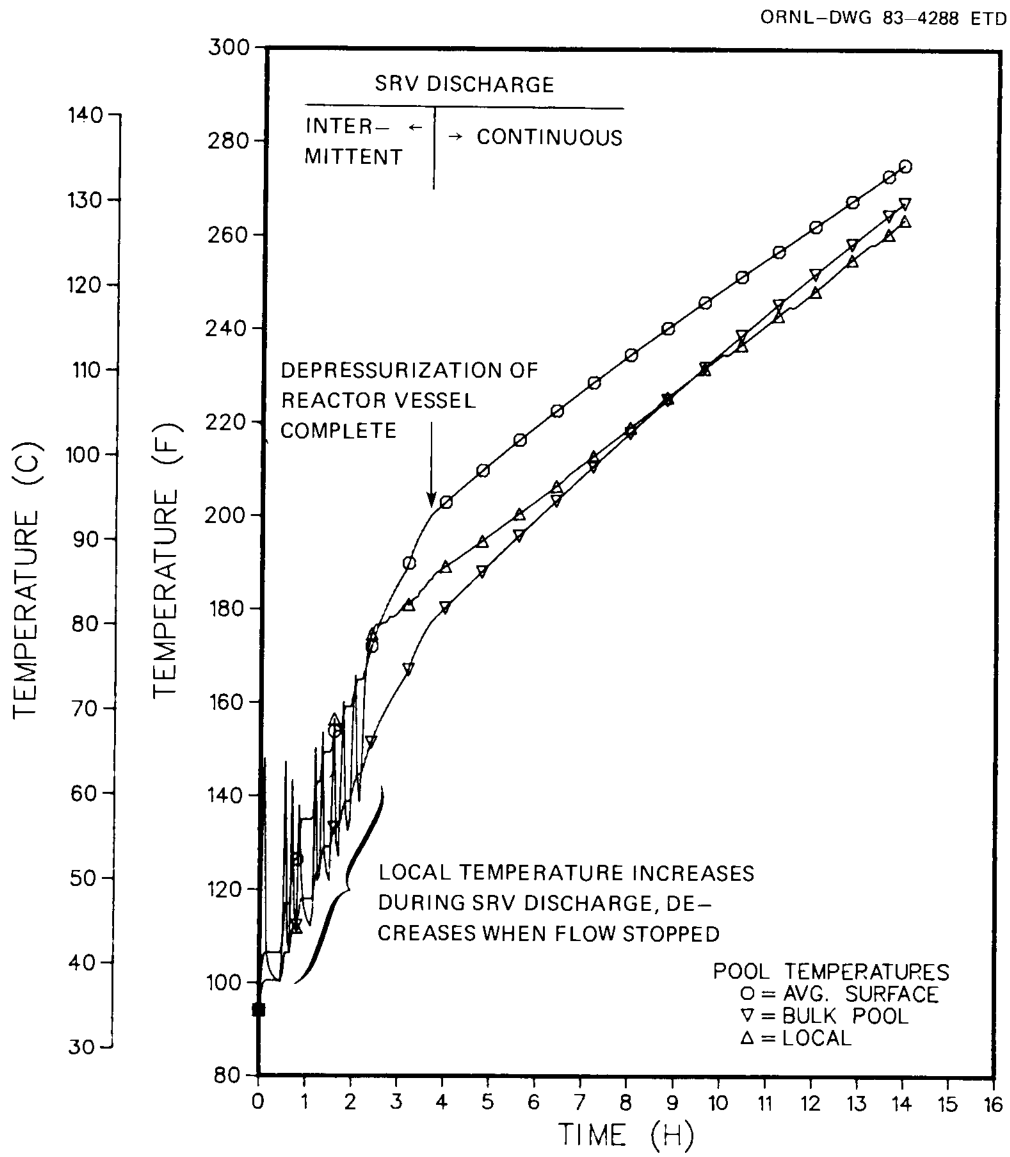

Fig. 5.2. Unmitigated Loss of DHR – suppression pool average temperatures and local temperature in the Bay of SRV discharge.

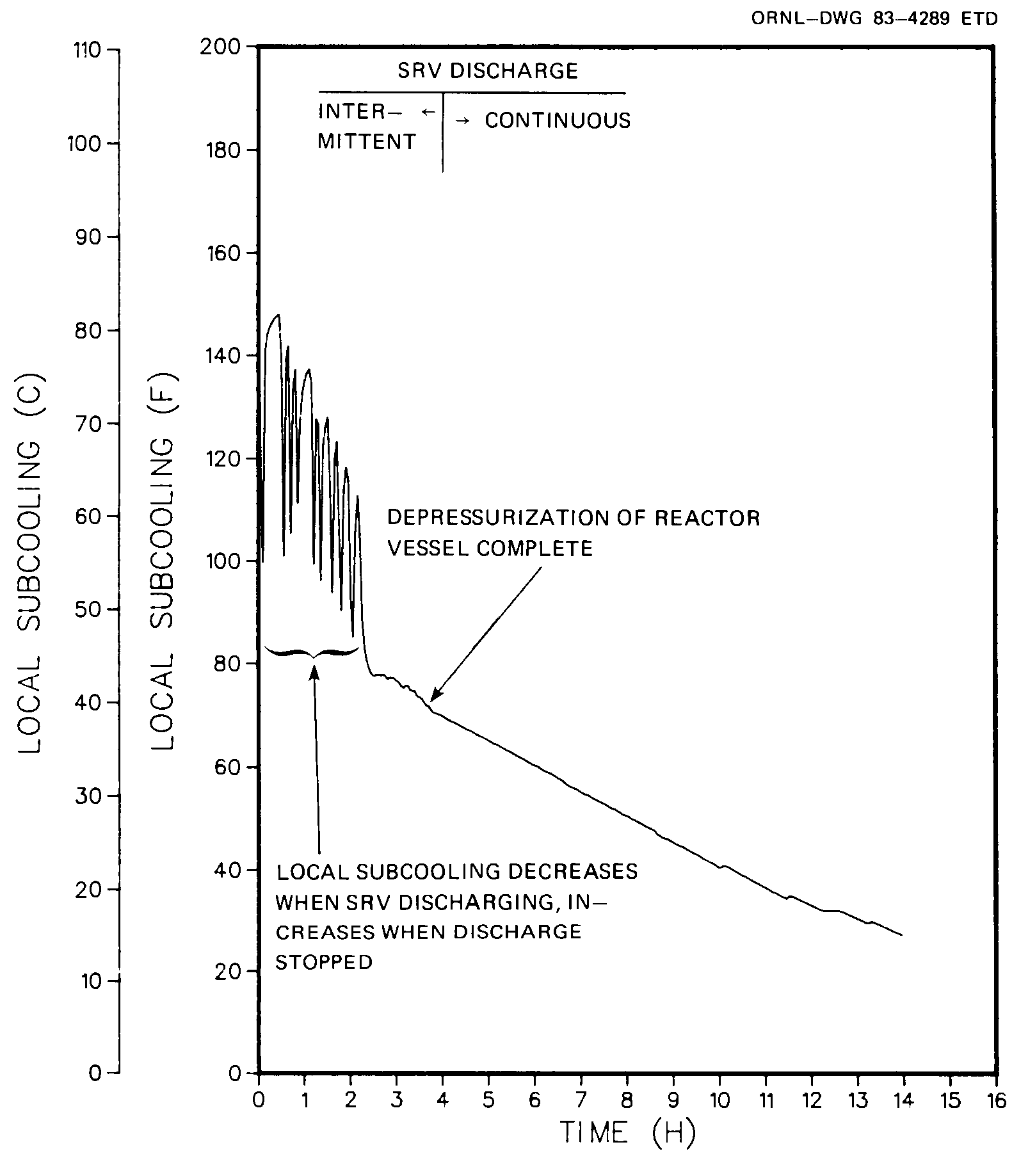

Fig. 5.3. Unmitigated Loss of DHR - Local Subcooling in the Bay of SRV discharge.

6. LOSS OF DHR WITH STUCK-OPEN RELIEF VALVE

6.1 Introduction

This chapter considers the effect of a stuck open relief valve (SORV) on the Loss of DHR accident sequence. The assumption of a SORV is in addition to the assumption of all of the other failures that must take place* in order to cause a loss of DHR (see Sect. 3.1). During the early part of the Loss of DHR accident, numerous actuations of the SRVs are required in order to control reactor vessel pressure. For the analysis of this chapter, it is assumed that a single SRV, after opening when called upon, fails to close, either automatically [when the reactor vessel pressure has decreased to below the 7.38 MPa (1055 psig) closing pressure of the lowest set group of SRVs] or in response to operator manipulation of the remote-manual SRV controls in the main control room.

The calculations for this section were performed using the BWR-LACP code. As in Chap. 3, a uniform suppression pool temperature was assumed for one set of calculations, since the RHR pumps might be available to circulate the pool water and thereby minimize thermal stratification and the existance of local temperatures in excess of the bulk pool temperature. Additional calculations were performed using the special suppression pool model (used in Chap. 5 and described in Appendix C) which is able to predict local temperature and thermal stratification in the event that there is no forced pool circulation.

6.2 Summary and Conclusions

For the case of suppression pool circulation and uniform pool temperature, an unmitigated Loss of DHR sequence with a SORV would lead to eventual primary containment failure by overpressurization at 34 h. This is close to the 35 h failure time estimated in Chap. 3 for the non-SORV case.† If a loss of vessel water injection were to result from containment failure, the reactor vessel would be at low pressure at the time of core uncovery. A pressurized boil-off was predicted (Chap. 3) for the non-SORV case.

A SORV does not have a great impact on overall system behavior during a Loss of DHR sequence. The continuously open SRV depressurizes the reactor vessel sooner and more rapidly than would the controlled depressurization [55°C/h (100°F/h)] that the operators are required by procedure to initiate when suppression pool temperature exceeds 49°C (120°F). As a

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*It is possible that the reactor shutdown following the SORV could be the Loss of DHR accident initiator.

†Primary containment failure occurs earlier for the case with an SORV because repressurization of the reactor vessel does not occur. As discussed in Chap. 3, suppression pool level and temperature remain constant during the lengthy period of reactor vessel repressurization, thereby delaying the further increase of primary containment pressure.

result, the pool heats up faster during the first several hours. However, the pool heatup is soon limited by the rate of decay heat steam production in the reactor vessel. After 3 h, the pool temperature (Fig. 6.5) has reached 73°C (163°F), and this is no higher than the case without the SORV (Fig. 3.6).

The slightly lower reactor vessel pressure does not cause the turbine-driven RCIC system to become unavailable at any time during the accident sequence. The RCIC system actuates intermittently during the first 6 h, but the vessel pressure remains well above the 0.45 MPa (50 psig) setpoint for isolation of the RCIC turbine steam supply line although the vessel pressure has decreased to below the 0.79 MPa (100 psig) setpoint for isolation of the HPCI turbine steam line at about the 6 h point. After 6 h, the CRD hydraulic system pumps provide all required vessel water injection.*

6.3 Detailed Results

Results of the BWR-LACP calculations are shown in Figs. 6.1–6.9. In most cases, these results are very similar to those for the Loss of DHR sequence with no SORV (see Chap. 3) in which the reactor vessel is depressurized by the operator early in the sequence. Therefore, the discussion below is limited to major points which are unique to the case of depressurization by a SORV. A more complete description of the physical basis for, and operator actions which can affect, the behavior of each system variable can be found in the corresponding paragraphs of Sect. 3.3.

6.3.1 Reactor vessel water level

Figure 6.1 shows water level in the downcomer region of the reactor vessel.† Figure 6.2 shows the total rate of vessel injection flow, and the total amount of water injected is shown in Fig. 6.3. The results are very similar to those shown in Figs. 3.1, 3.2, and 3.3 for the case without a SORV. The vessel level is maintained in an acceptable range throughout the 34 h period prior to containment failure. During the first 6 h, injection is provided by the CRD hydraulic system and the RCIC system in combination. There is a single cycle of HPCI operation, which the operators initiate during the first minutes in order to rapidly restore vessel level to the normal operating range. After 6 h the injection requirements are fully met by the CRD hydraulic system, alone. If the RCIC

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*With an SORV, more coolant is lost from the reactor vessel during the early portion of the accident sequence and must be replaced by the operating injection systems. Without the SORV (but with an operator-controlled depressurization), all required reactor vessel injection can be supplied by the CRD hydraulic system after 4 h, as discussed in Chap. 3.

†This is the water level indicated on the control room instruments. To read the full range of level variation shown on Fig. 6.1, operators would have to consult the wide range instruments.

system had been unavailable after 4 h, reliance solely on the CRD hydraulic system injection would not have resulted in an unacceptably low vessel level. The single RCIC actuation between 4 and 6 h (Fig. 6.2) is based on the operators desire to maintain vessel water level as close as possible to the normal value of 14.25 m (561 in.) above vessel zero.

6.3.2 Reactor vessel pressure

Figure 6.4 shows reactor vessel pressure following the SORV with loss of DHR. Pressure falls very rapidly at first, due to the combined effect of the SORV and of the HPCI system which draws steam from the vessel to run its turbine, and injects a large quantity of cold water into the vessel.

After the HPCI system is shutdown, the pressure falls less rapidly. As pressure decreases, the mass flow of steam (at sonic velocity) through the fully open SORV also decreases. The rate of depressurization becomes slower and slower, until pressure reaches a minimum of 0.69 MPa (85 psig) after about 6 h. Pressure then increases slightly, levels off, and begins to decrease very slowly until 19 h after accident initiation. By this time the pressure in the primary containment including the wetwell, which receives the SRV discharge, has increased to the point at which the downstream pressure is not low enough to maintain critical flow through the SORV. Therefore, throughout the remainder of the sequence, SRV flow capacity is impaired and the vessel pressure slowly increases to about 1.03 MPa (135 psig) at the estimated 34 h primary containment failure time.

6.3.3 Suppression pool temperature and water level

Suppression pool temperature (Fig. 6.5) and water level (Fig. 6.6) increase steadily throughout the SORV accident sequence because of the continuous discharge of steam through the fully open SRV, without any pool cooling. The results shown in Figs 6.5 and 6.6. were calculated assuming a uniform pool temperature.

The RHR pumps might not be available to circulate the pool water to minimize local temperature and thermal stratification. Accordingly, the transient pool temperature distribution throughout the pool was also calculated using the detailed thermal-hydraulic suppression pool model described in Appendix C. This model used as input the BWR-LACP results for SORV discharge flow and suppression pool pressure throughout the 34 h long sequence. The results (not shown) indicate that as expected, the temperature is higher in the bay of SRV discharge and that the pool tends to stratify, with the hotter fluid on top. However, due to the relatively low release rate of steam into the suppression pool during the early part of the accident sequence, the local temperature differences are small. At the end of 10 h, the average pool surface temperature is only about 3°C (5°F) above the whole-pool bulk temperature, and the temperature of the water in the locality of the discharging T-quencher is approximately equal to the whole-pool bulk temperature. The effect of the higher surface temperature on primary containment pressure is assessed below.

6.3.4 Primary containment pressure and temperature

Drywell pressure and atmosphere temperature are shown in Figs. 6.7 and 6.8 for the case with uniform pool temperature (as if the RHR pumps were circulating the pool water, but without pool cooling). As discussed in Sect. 3.3.4, the drywell coolers might trip after about 1 h, leading to an increase in drywell temperature (Fig. 6.8). As pool temperature increases, steam escapes from the surface of the pool, pressurizing the primary containment, eventually leading to drywell failure after 34 h.

If there were no forced pool circulation, higher surface temperatures would result from the thermal stratification effect and there would be a more rapid loss of steam to the primary containment atmosphere. With the 3°C (5°F) higher surface temperature reported in Sect. 6.3.3, the drywell failure pressure would be exceeded at 32 h instead of at 34 h. This is 4 h later than for the corresponding case without the SORV.

6.3.5 ECCS pump net positive suction head (NPSH)

Figure 6.9 shows the NPSH that would be available if a single RHR pump were operated (taking suction on the pool and discharging back to the pool) throughout the sequence. The NPSH at the pump suction is below the manufacturers recommended minimum NPSH during the final one-third of the sequence. As discussed in Sect. 3.3.5, it is possible to operate below the recommended minimum; in addition, the available NPSH at the pump suction can be increased by throttling the pump discharge to operate the pump at a lower flow. Therefore, RHR pump operation could be maintained (or begun) throughout the sequence.

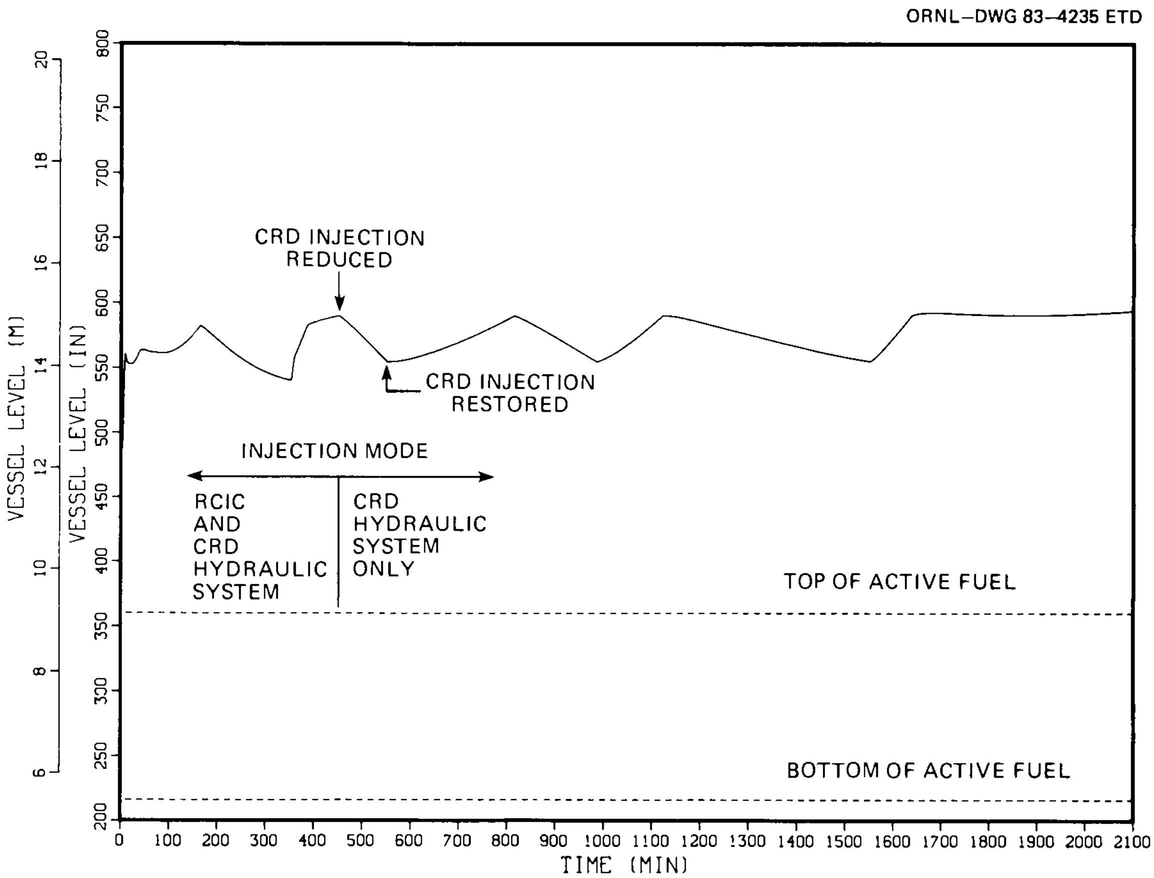

Fig. 6.1. Unmitigated Loss of DHR with SORV – reactor vessel level.

ORNL-DWG 83-4236 ETD

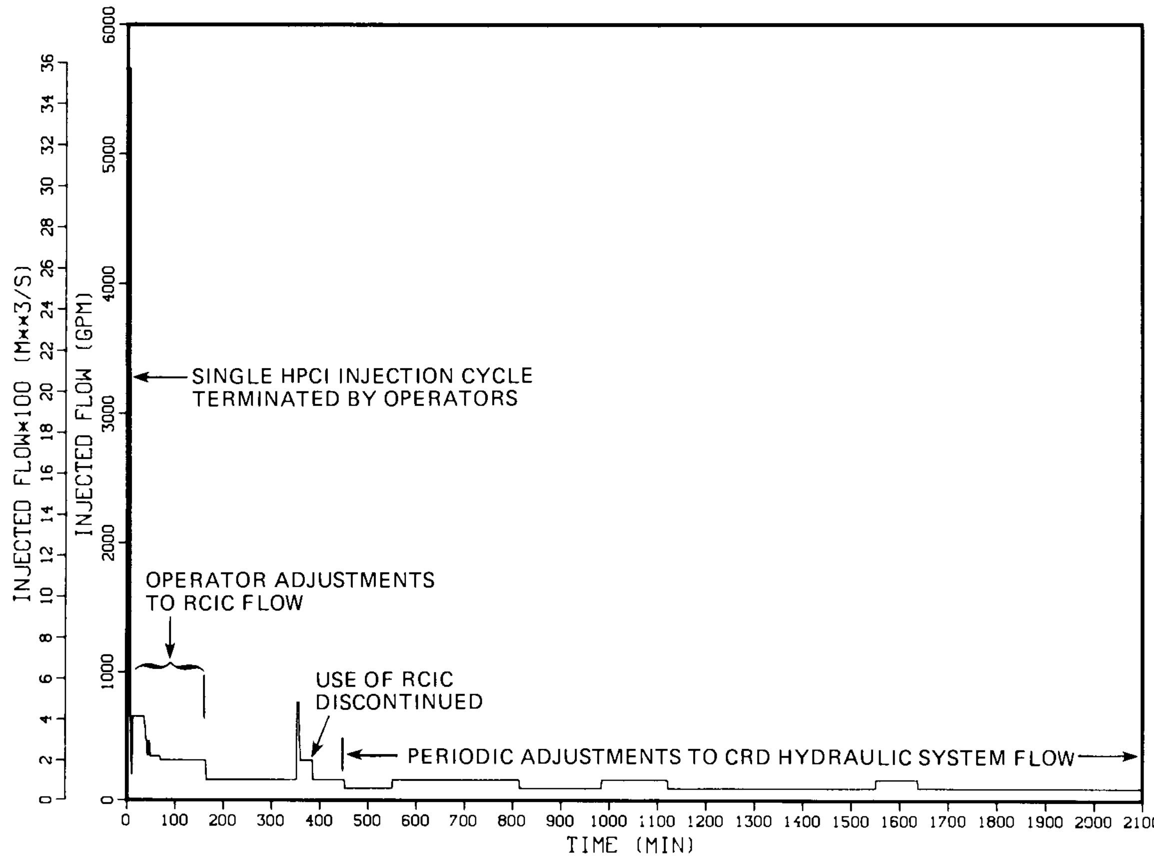

Fig. 6.2. Unmitigated Loss of DHR with SORV – total rate of injection flow.

ORNL–DWG 83–4237 ETD

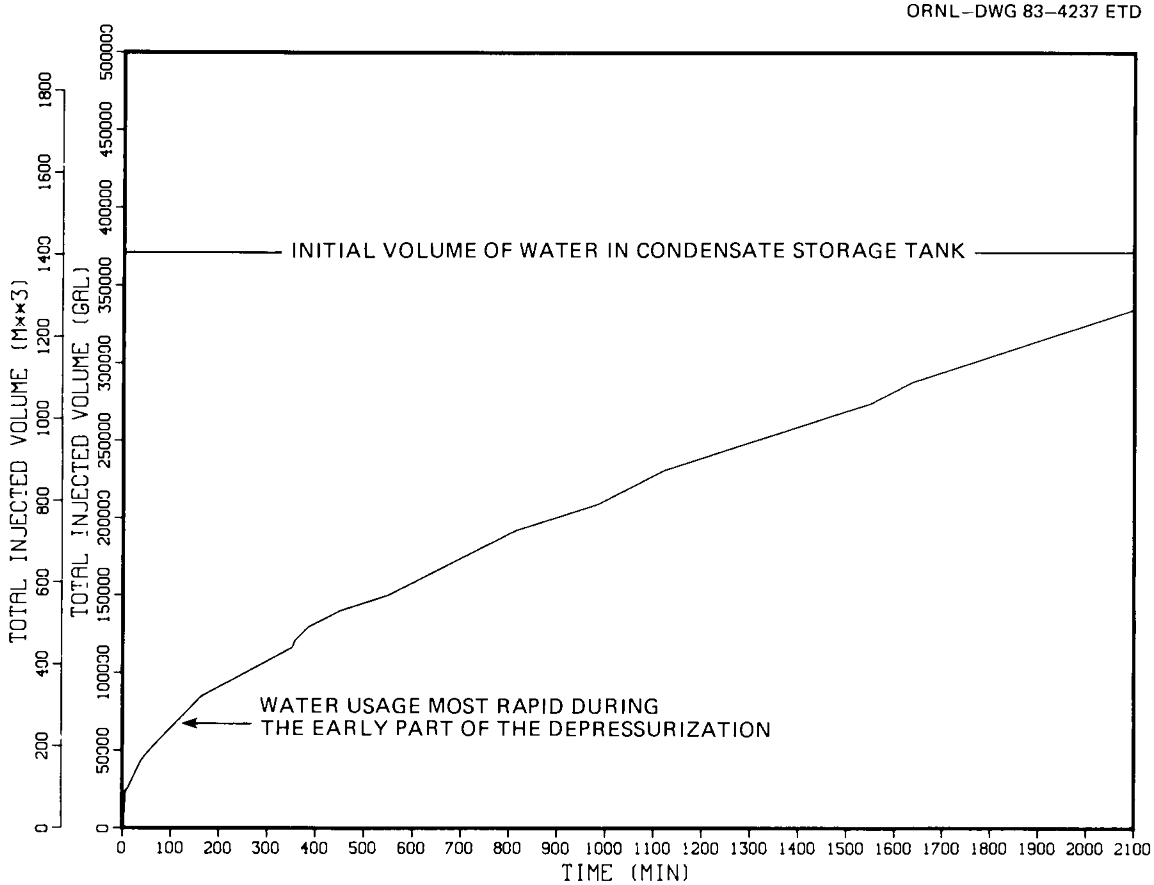

Fig. 6.3. Unmitigated Loss of DHR with SORV – total volume injected into reactor vessel.

ORNL–DWG 83–4238 ETD

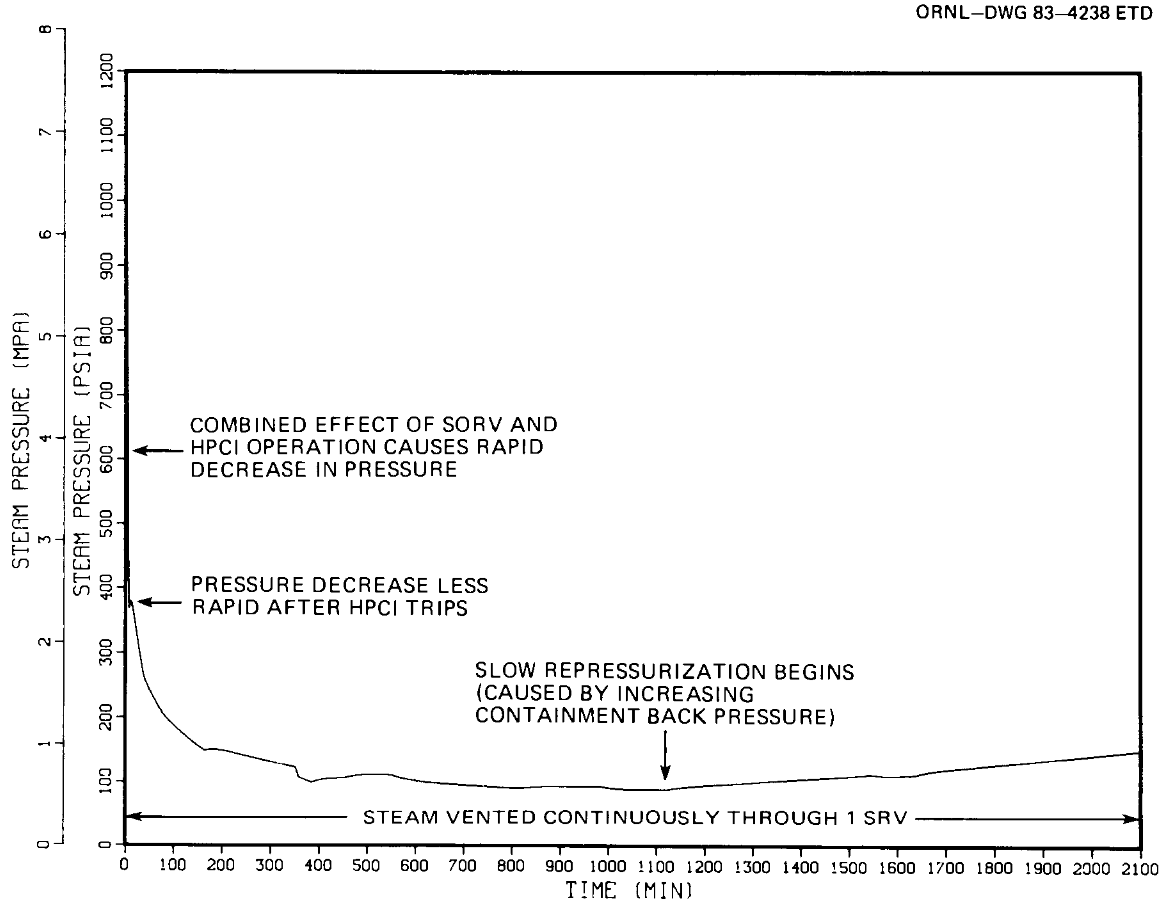

Fig. 6.4. Unmitigated Loss of DHR with SORV – reactor vessel pressure.

ORNL-DWG 83-4239 ETD

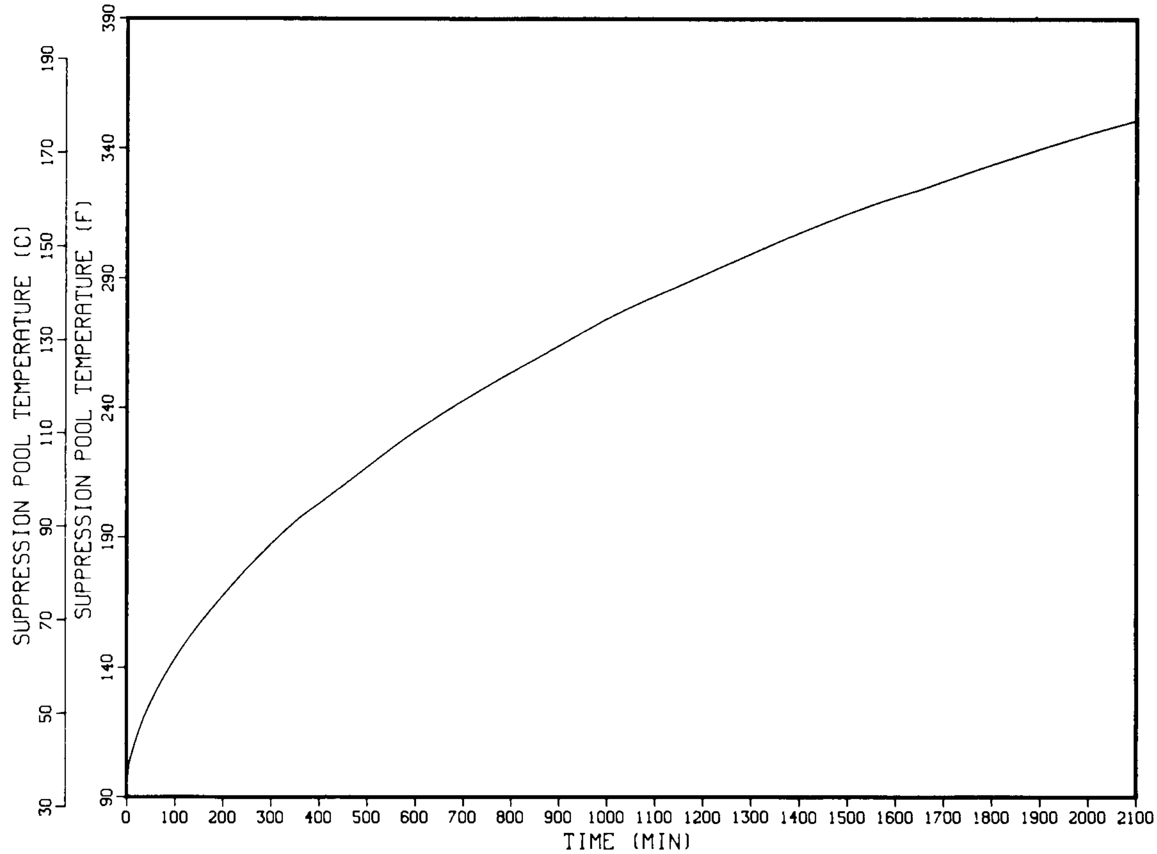

Fig. 6.5. Unmitigated Loss of DHR with SORV - suppression pool temperature.

ORNL-DWG 83-4240 ETD

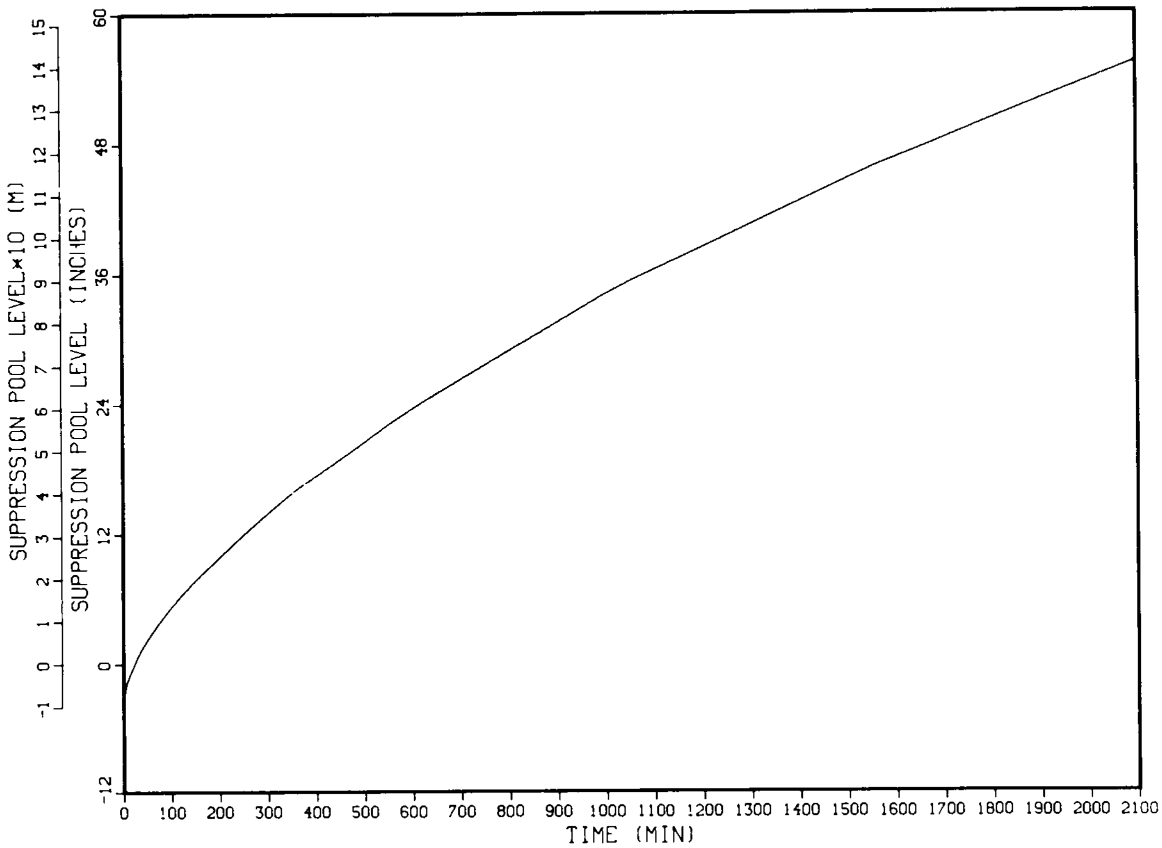

Fig. 6.6. Unmitigated Loss of DHR with SORV - suppression pool water level.

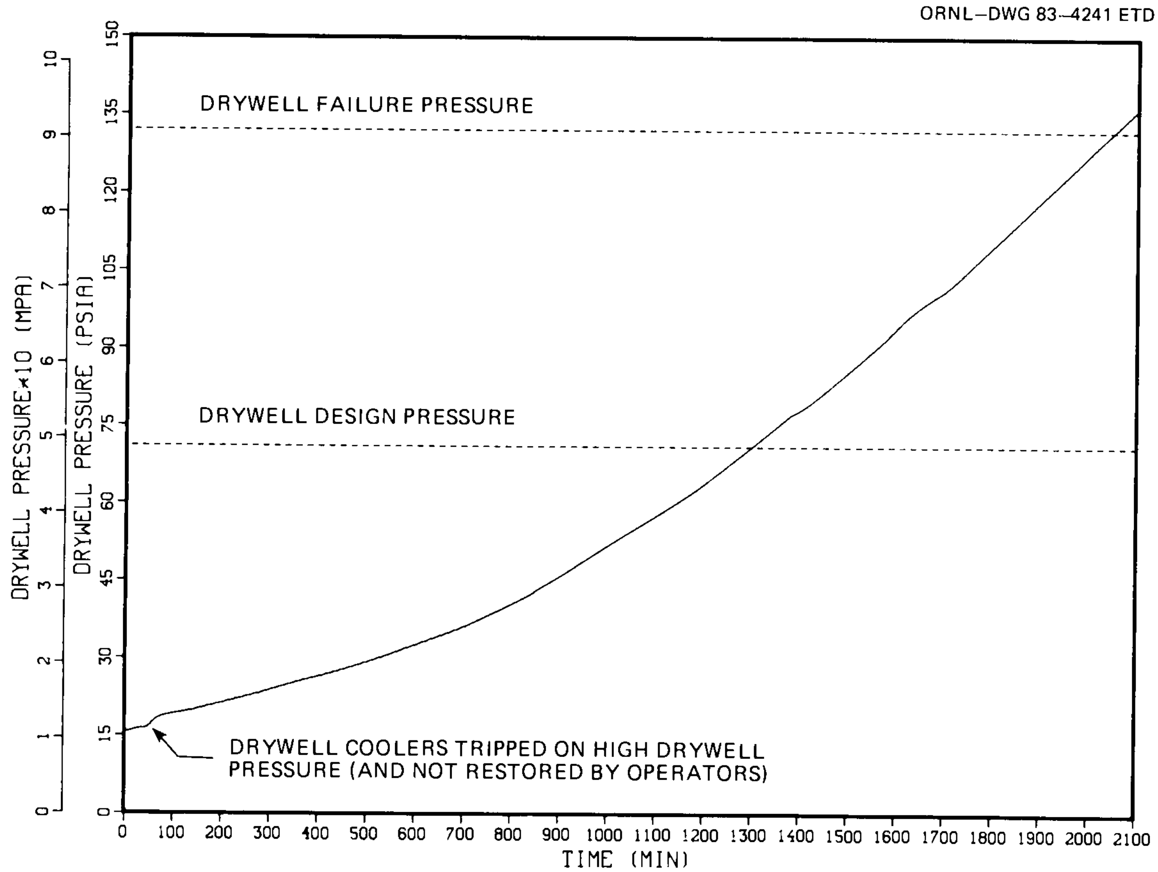

Fig. 6.7. Unmitigated Loss of DHR with SORV – drywell pressure.

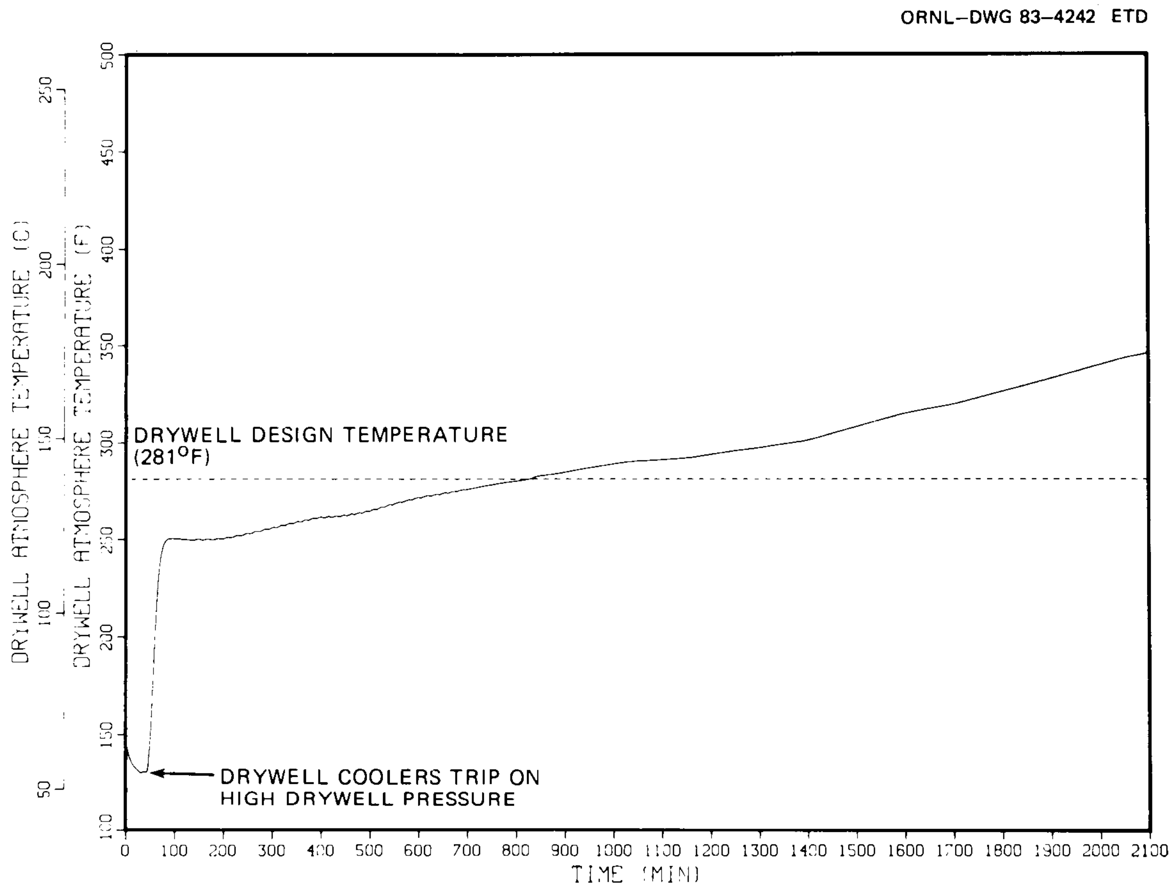

Fig. 6.8. Unmitigated Loss of DHR with SORV – drywell temperature.

ORNL–DWG 83–4243 ETD

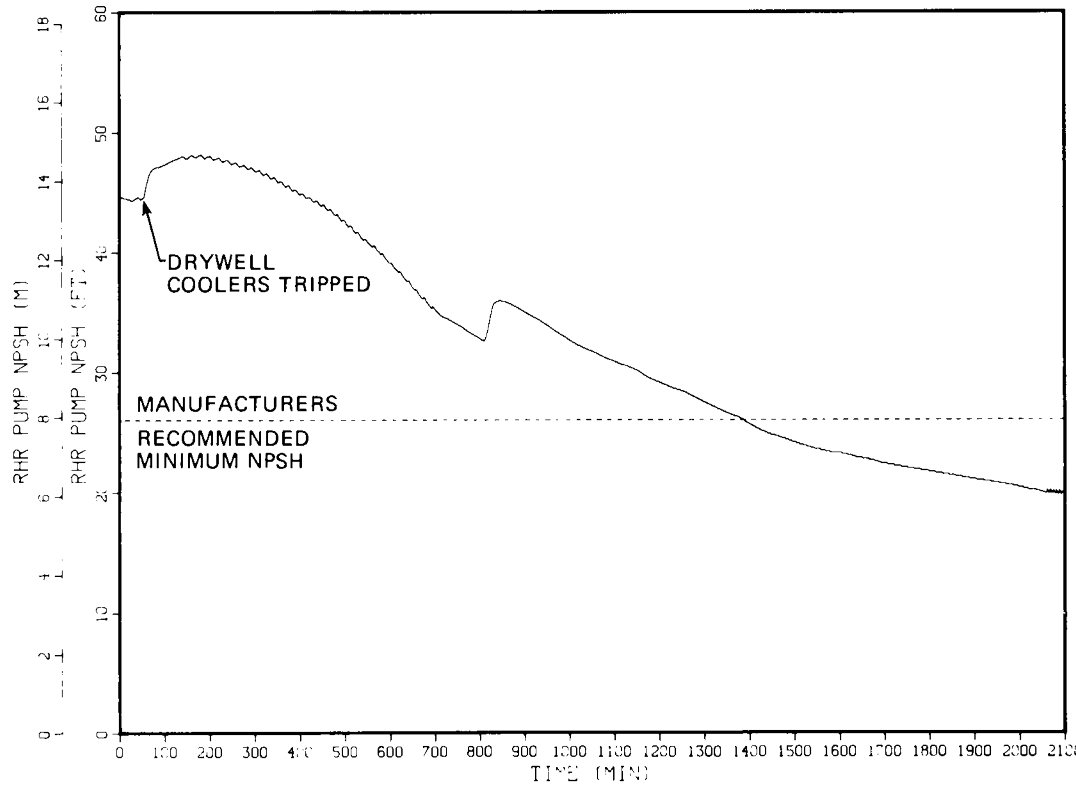

Fig. 6.9. Unmitigated Loss of DHR with SORV - RHR pump net positive suction head.

7. STATIC OVERPRESSURIZATION CONTAINMENT FAILURE MECHANISMS AND PHENOMENA

7.1 Introduction

The loss of decay heat removal accident sequences described in Chapters 3 through 6 produce a gradual heatup and pressurization of the primary containment system. If the accident is allowed to progress indefinitely, this pressurization will eventually result in loss of containment integrity, i.e. containment failure. The purpose of this chapter is to briefly review static overpressurization containment failure mechanisms and the phenomena induced in a MARK I containment system by such a failure. Sections 7.2 and 7.3 present a brief review of the Browns Ferry MARK I containment system design. Section 7.4 discusses static over-pressure containment failure mechanisms, and Sect. 7.5 describes thermodynamic and physical phenomena which might be expected to occur following containment failure. Section 7.6 is a discussion of the long term core cooling requirements and system capabilities of the Browns Ferry Unit 1 reactor during the Loss of DHR accident. An evaluation of the possible impacts of post-containment failure phenomena on reactor vessel injection system capabilities is presented in Sect. 7.7. Section 7.8 summarizes the phenomenological discussions presented in this chapter and Sect. 7.9 describes their significance in light of current Loss of DHR accident probabilistic risk assessment practices.

7.2 BFNP Primary Containment Structural Design*

The Browns Ferry Reactors employ a MARK I pressure suppression containment system which houses the reactor vessel and coolant recirculation loops. The design consists of a drywell, constructed in the shape of an inverted light bulb, a toroidal pressure suppression chamber, which normally contains approximately 3785 m^3 (one million gallons) of water, and a connecting vent system between the drywell and the pressure suppression pool (Fig. 7.1). Pertinent primary containment design parameters are given in Table 7.1.

The drywell is a steel pressure vessel with a spherical lower portion 19.8 m (65 ft) in diameter and a cylindrical upper portion 11.7 m (38 ft, 6 in.) in diameter. The overall height of the drywell is ~35 m (115 ft). The drywell is designed for an internal pressure of 0.478 MPa (56 psig) coincident with a temperature of 411.5 K (281°F), plus the dead, live, and seismic loads imposed on the shell. The thickness of the drywell wall varies from a minimum of 1.9 cm (3/4 in.) in the cylindrical section, to a maximum of 5.9 cm (2 5/16 in.) in the toriodal sphere/cylinder knuckle region.

The pressure suppression chamber is a steel pressure vessel of toroidal shape, located below and surrounding the drywell. The centerline

*The majority of the information in this section is excerpted from Ref. 7.1.

diameter of the torus is ~33.8 m (111 ft) and the cross-sectional diameter is 9.5 m (31 ft.). It contains ~3823 m^3 (135,000 cubic ft) of water at maximum pool level. The thickness of the torus wall varies between 1.9 and 2.9 cm (3/4 and 1-1/8 in.). The suppression chamber is designed to the same material and code requirements as the steel drywell vessel, and all attachments to the torus are by full penetration welds.

The drywell and suppression chamber are connected by a vent system which conducts flow from the drywell into the suppression pool and distributes this flow uniformily around the pool. Eight circular vent pipes, each 2.06 m (6.75 ft) in diameter, connect the drywell to the suppression chamber. Jet deflectors are provided in the drywell at the entrance to each vent pipe. These vents are connected to a 1.45-m (4-ft, 9-in.) diameter vent header of toroidal shape, which is contained within the airspace of the suppression chamber. Ninety-six downcomer pipes, each 0.61-m (24-in.) diameter, project downward into the suppression pool, terminating 1.22 m (4 ft) below the surface of the pool. Vacuum breakers discharge from the suppression chamber free space into the vent pipes to equalize the pressure between the drywell and suppression chamber. The suppression chamber, which is located in a separate room in the reactor building basement, is accessable only through two normally closed 1.22-m (4-ft) diameter manhole entrances with double testable seals and bolted covers.

Several types of piping and electrical penetrations, as well as personnel and equipment access hatches penetrate the primary containment. The general design of the piping penetrations incorporates a penetration sleeve which passes from the reactor building, through the shield wall concrete, and projects into the gap region between the shield wall and the drywell liner. Guard pipes and expansion bellows are incorporated where necessary to allow for movement and protection of process lines. Personnel and equipment hatches incorporate double, testable seals to ensure containment integrity.

7.3 Drywell Liner Gap Construction

The BFNP drywell is surrounded by reinforced concrete for shielding purposes, but the steel drywell liner is in direct contact with this surrounding concrete only below elevation 548.79 ft (Fig. 7.2).[7.2,7.3] Above this elevation, the gap between the drywell liner and the reactor building concrete is filled with a variety of materials. Between elevation 548.79 ft and 550.29 ft, the drywell liner is surrounded by a sand transistion zone which is designed to transmit seismic loads from the drywell liner evenly to the concrete foundation.[7.4] This sand transition zone is drained by eight, 10.2-cm (4-in.) diameter, sand filled pipes, which are spaced at equal intervals around the drywell.[7.5-7.8] These drain pipes discharge onto the suppression chamber room floor. Between elevation 550.29 ft and 566.0 ft the drywell liner is surrounded by a 6-cm (2-3/8-in.) layer of fiberglass. Above the 566.0-ft elevation, the drywell liner is surrounded by a 5.7-cm (2-1/4-in.) layer of polyester-based foam filler. This foam has a maximum service temperature of 413.7 K (285°F),[7.9] and is designed to accomodate compression due to thermal expansion of the drywell liner. Between this foam filler and the concrete is a fiber glass laminated concrete pouring form. Both the foam filler and the fiber glass form extend up to the 636.67-ft elevation.

Pressurization of the volume enclosed by the drywell liner, reactor building concrete, and drywell shield plugs is prevented by several leakage paths which vent to various regions of the reactor building (Fig. 7.3). A major gap vent path is provided by the annular spaces between embedded containment penetration sleeves and their associated penetration assemblies. This vent area is available at all sleeves for piping penetrations, the personnel air lock, and the two equipment access locks.[7.5] All embedded sleeves are a minimum of 15.2 cm (6 in.) larger in diameter than their associated penetrations.[7.5] A typical penetration sleeve configuration is shown in Fig. 7.4. The annular gap between the inside of the penetration sleeve and the outer surface of the penetration assembly extends into the drywell liner gap region, providing a direct pathway for flow from the drywell liner gap into the reactor building. Over one hundred drywell penetrations of various sizes are scattered over the face of the drywell liner, affording a combined drywell liner gap vent flow area in excess of 9.29 m^2 (100 ft^2).

As shown in Fig. 7.3, the drywell gap can also vent to the suppression chamber room via the annular gaps between the eight drywell vent pipes and their surrounding vent sleeves. The total flow area for these eight flow paths is approximately 9.29 m^2 (100 ft^2). The suppression chamber room is connected to the HPCI, RCIC, RHR, and core spray pump rooms via four (one for each room) open manways (Fig. 7.5). These ECC pump rooms are in turn connected to the remainder of the reactor building via open stairwells. The suppression chamber room also connects directly to the drywell personnel access room on the 565-ft level via the floor penetration sleeves of the RHR system shutdown supply and return lines. The drywell personnel access room connects to the surrounding reactor building atmosphere via the access room valve operator roof openings at the 580.0-ft level. These room openings provide a minimum flow area of 0.93 m^2 (10 ft^2).[7.10]

The annular gap between the drywell liner and the surrounding reactor building concrete shield is sealed where the removable drywell head joins the liner (elevation 639 ft) by drywell-to-reactor-building bellows which are designed to accomodate the differential expansion between the drywell liner and the reactor building concrete during plant heatup and cooldown. The bellows is a single-piece stainless steel structure.[7.11] In the event this bellows seal were breached, the liner gap would vent into the upper drywell head region below the drywell shield plugs. These shield plugs are constructed in a three layer, six piece, circular configuration with an 11.56 m (37-ft, 11-in.) inner diameter, and weigh between 67,100 and 90,700 kg (74 and 100 tons) each. Since the plugs are held in place only by their weight, they do not form a leakproof seal between the reactor building refueling floor and the drywell cavity.[7.12]

7.4 Static Overpressure Containment Failure Mechanisms

As previously stated, the loss of decay heat removal accident sequences described in Chapters 3 through 6 produce a slow heatup and pressurization of the primary containment system, ultimately resulting in containment failure. An early study of the maximum BWR containment capability for such accidents was conducted as part of the Reactor Safety Study (RSS).[7.12] The containment studied was that of the Peach Bottom plant, a BWR4/MARK I reactor/ containment system very similar to the Browns Ferry facility. The design pressure of both the Peach Bottom and Browns Ferry containments is 0.49 MPa (56 psig). The conclusion of the RSS study was that a best estimate failure pressure for the subject containment design is 1.21 $\pm$ 0.17 MPa (175 $\pm$ 25 psia). This pressure corresponds to a stress level in the base material midway between the yield and ultimate strength of the metal. [If the ultimate strength of the structure could be developed, the study concluded that the failure pressure would be 1.724 MPa (250 psia)]. The failure was predicted to occur in the upper half of the pressure suppression pool wall (Fig. 7.6, point A), although it was also stated that the toroidal knuckle between the drywell spherical and cylindrical sections (point B in Fig. 7.6) is a potential failure area. No estimate of the size of the failure opening was made, but the failure was assumed to be of sufficient size to rapidly depressurize the containment system.

In a recent study conducted by the Ames Laboratory at Iowa State University,[7.13] the failure pressure of the Browns Ferry containment was estimated to be 0.908 MPa (117 psig). Maximum circumferential member strain was utilized as the failure criterion. The study assumed uniform static internal pressure loading, and only shell failure modes were considered. The effects of penetrations, anchorages, etc., were ignored.* A failure of indefinite size was predicted to occur at the toroidal knuckle interface between the drywell spherical and cylindrical sections (point B, Fig. 7.6).

Neither of these two studies made any definitive statements regarding the shape, size, or propagation rate of the containment failure opening. It is probable that the failure would take the form of a circumferential ductile rip which would propagate at subsonic speeds 1/4 to 1/2 of the way around the drywell sphere/cylinder knuckle.[7.14–7.15] The ultimate size of the break would probably be between 0.003 and 0.929 m^2, i.e., greater than a few square inches but less than ten square feet.[7.15]

As previously stated, neither of the two studies mentioned above incorporated explicit treatment of containment penetrations and both studies assumed that the drywell liner is free to expand in a radial fashion. The drywell liner could yield and deform significantly prior to shell failure. The Ames researchers (Ref. 7.13) indicated, however, that the maximum radial expansion of the drywell liner into the gap region would be less than 3 cm (1.2 in.). It is uncertain whether containment failure would occur due to the mechanism described in the two reports cited above, but the drywell liner can expand only 5 cm (2 in.) before contacting the surrounding concrete. It is possible that localized strains near liner penetrations would exceed shell strain values, perhaps causing seal and gasket leakage around intact penetration assemblies prior to shell failure.

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*The study notes, however, that construction codes require the containment penetrations to be designed to more stringent standards than the liner itself.

Failures of this type might result in slow containment depressurization transients, rather than the violent containment blowdown which might be expected to occur following a gross rupture of the drywell liner shell.

7.5 Post Containment Failure Phenomena

Based on the containment failure pressures, locations, and sizes discussed in Sects. 7.3 and 7.4, the post containment failure phenomena which might occur in a typical MARK I containment system will be briefly discussed in this section. For the purposes of this discussion, it has been assumed that the drywell liner fails in the knuckle region, at a pressure of 0.908 MPa (117 psig), and that the pressure suppression chamber boundary remains intact following drywell liner failure. The size of the failure opening is assumed to be 0.929 m^2 (10 ft^2). It has also been assumed that at the time of drywell liner failure, the suppression pool is near its normal operating level and at the saturation temperature corresponding to 0.908 MPa (117 psig), i.e. 449°K (349°F).

Since it is probable that the drywell would begin leaking at some pressure below 0.908 MPa (rather than failing catastrophically at that pressure), the reader should regard the analysis and results presented in this section as a reasonable upper limit approximation to the forces involved in the drywell blowdown transient. An additional conservatism is introduced by the high mass flows employed in this analysis. These flow predictions are based on extremely simple models which do not account for the pressure losses induced by the complex flow path configurations which would be involved in an actual drywell liner rupture accident.

The immediate impact of a drywell liner failure under the conditions assumed above is a drop in drywell and suppression chamber pressure. Since the suppression pool water is originally saturated, this pressure reduction results in flashing of the water in the drywell vent downcomers and in the main body of the pressure suppression pool. The resulting steam would enter the drywell via the eight 2.06-m (6.75-ft) diameter vent pipes, and leak from the containment via the drywell liner rupture. The original nitrogen atmosphere of the containment would be swept out by the large drywell break flow expected in this scenario. As will be described in Chapter 8, MARCH calculations for this accident indicate that over 453,500 kg (1,000,000 lbm) of steam is generated and leaked from the containment during the one hour period following containment failure. Peak break flows immediately following liner rupture are predicted to exceed 570 kg/s (75,000 lbm/min) or approximately 164 m^3/s (347,000 ft^3/min) of steam.

Detailed analysis of the scenario described above is extremely difficult due to uncertainties in the break size, flow topology, and flow path configurations available to material exiting the drywell break. As previously described in Section 7.3, the gap between the drywell liner and the reactor building concrete is filled with a polyester foam material with a maximum design service temperature of 413.7 K (285°F). The drywell gas temperatures prior to liner failure would typically vary between 450 and 478 K (350 and 400°F). It is reasonable to assume that the foam filler would lose structural integrity under these conditions, allowing the drywell leakage to flow into various reactor building floors, the reactor building refueling floor, and the suppression chamber room via the flow paths described in Section 7.3. The specific flow paths involved are particularly difficult to predict since ballooning of the drywell liner could effectively close flow paths above or below the liner knuckle region. The configuration of the break opening is also uncertain. Four possible configurations are shown in Fig. 7.7.

The rapid flow of material through the break opening results in a thrust force, F, which acts on the drywell compartment. This thrust force, can be calculated as[7.16]

$$F = \dot{M}U_1 + (P_e - P_g)A_1 \qquad (7.1)$$

where,

$\dot{M}$ = break mass flow rate = 570 kg/s
U_1 = break flow velocity = Volumetric flow rate/break area (164/0.929 = 180 m/s)
P_e = break exhaust pressure
P_g = Ambient pressure in drywell gap
A_1 = Break area = 0.929 m²

Depending upon the exact geometry of the break, the exhaust pressure, P_e, can assume any value between the internal drywell pressure, P_d, and the drywell gap pressure P_g. It is probable, due to the proximity of the break to the surrounding vertical wall, that a stagnation pressure between P_d and P_g would develop at the break exit. In any event, based on the mass flows previously quoted, the thrust forces can be bounded by

$$F_L = \frac{\dot{M}U_1}{g_c} \approx \frac{(570)(164)}{(1)0.929} \approx 100{,}000 \text{ N} \qquad (22{,}500 \text{ lbf})$$

and

$$F_u = \frac{\dot{M}U_1}{g_c} + (P_d - P_g)A_1 \approx 100{,}000 + (806{,}000)(0.929)$$

$$\approx 850{,}000 \text{ N} \qquad (191{,}000 \text{ lbf})$$

where g_c is a dimensional constant in English units.
This force would be directed radially inward around one-fourth to one half of the drywell knuckle perimeter. The effects of this thrust force on the drywell structure are uncertain, but the exiting steam would impact the

adjacent reactor building wall, exerting a radial force, T,[7.17] of

$$T = \frac{\rho U_1^2 A}{g_c} = \frac{\dot{M} U_1}{g_c} = 100{,}000\ \mathrm{N} \qquad (22{,}500\ \mathrm{lbf})$$

on the concrete. This force corresponds to a normal stress, S, of

$$S = T/A_1 = 0.11\ \mathrm{MPa}\ (15.6\ \mathrm{psi}).$$

The shield wall concrete is designed for a minimum compressive strength of 20.7 MPa (3000 psi).[7.18] It is, therefore, improbable that steam jet impingement forces would result in yielding of the shield wall. Further analysis is required to determine whether sustained exposure of the shield wall to such steam jet flows could induce wall failure due to ablation or disintegration of the concrete.

The impact of internal drywell blowdown forces and environments on drywell structures and equipment is also an area of concern. During the blowdown transient the steam produced by suppression pool flashing flows up through each of the eight drywell vent pipes, impinging on the jet deflectors near the bottom of the drywell sphere. The impingement force on the jet deflectors can be estimated as

$$F = \frac{1}{8}\frac{\dot{M} U_2}{g_c} = \frac{1}{8}\frac{\dot{M}}{g_c}\ \frac{U_1 A_1}{A_2} \approx \frac{(570)(180)(0.929)}{26.6\ (8)}$$

$$\approx 448\ \mathrm{N} \qquad (100\ \mathrm{lbf})$$

where A_2 = total vent pipe flow area = 26.6 m².
It is unlikely that forces of this magnitude would result in detachment of the jet deflectors.

Degradation of drywell equipment operability due to harsh environmental conditions is probable after drywell failure. The effects of equipment exposure to long term, high velocity steam flows such as those predicted to occur during the containment depressurization phase of this accident are exceedingly difficult to quantify. As will be described in Sect. 7.6, the continued operability of the primary system SRVs following containment depressurization is a topic of particular concern. Damage to SRV control air lines during the blowdown phase of the accident could result in inability to regain remote manual operability of the valves following drywell depressurization.

7.6 Long-Term Core Cooling Requirements and System Capabilities

As a result of the relatively slow heatup of the pressure suppression pool, primary containment failure pressures are not achieved until some 30 to 40 h after reactor scram. At the time of containment failure, the reactor vessel injection requirements are substantially reduced due to the low decay heat levels involved (less than 1% of full power). Indeed, the analyses presented in Chapt. 3 indicates that at 34 h after scram a throttled CRD hydraulic pump flow of only 0.006 m^3/sec (100 gpm) is sufficient to maintain a covered core, without any assistance from other reactor vessel water injection systems.

Table 7.2 is a summary of reactor vessel water injection systems which might be available for maintenance of vessel water level during emergency conditions. The data in Table 7.2 is based on the BFNP Emergency Operating Instruction No. 41[7.19] and the BFNP FSAR.[7.20]

The HPCI, RCIC, core spray, RHR, and CRD hydraulic system pumps are located in the basement of the reactor building, in rooms which are adjacent and open to the suppression chamber room. As noted in Table 7.2, the plant auxilliary boiler system provides a secondary steam source for the HPCI and RCIC pump turbines, although operation in this manner is possible only after the installation of a piping spool piece which requires approximately four hours of labor. However, it is clear that this capability significantly increases the utility of the HPCI and RCIC systems during long term accident situations.

It should be noted that the CRD hydraulic system pump automatically injects approximately 0.0085 m^3/s (170 gpm) of water into the reactor via the CRD mechanism assemblies when a reactor scram is in effect and the reactor vessel is depressurized. No operator action is necessary to initiate this flow. Alternatively, the operator can manually realign the system to inject 0.01 m^3/s (200 gpm) into the reactor vessel via a feedwater line.[7.21]

The standby liquid control (SLC) system is a low flow, high pressure injection system which can be utilized under accident conditions to inject small amounts of demineralized water into the reactor vessel. The major SLC system components are located in the northeast corner of the reactor building on the 639.0-ft floor level.

The condensate/condensate booster pump system is a low pressure makeup system which has the capacity to pump water through the idle turbine-driven feedwater pumps into the reactor vessel at vessel pressures up to approximately 2.9 MPa (400 psig). The major components of this system are located on the lower level of the turbine building, on floor levels 551 and 557 ft.

The condensate transfer pumps (located on the 565-ft level of the turbine building) are low pressure pumps, which could only be utilized following reactor vessel depressurization.

The RHR drain pumps are located in the suppression chamber room of the reactor building, adjacent to the RHR/LPCI pumps. The RHR service water pumps are located in a reinforced concrete building at the land end of the river water intake channel. Both of these systems have large pumping capacities, but the reactor vessel would have to be depressurized prior to placing either system in operation.

In summary, it is clear that the profusion of BFNP emergency injection systems and system operating modes provides significant assurance that reactor vessel injection flow would be available during a wide spectrum of accident conditions. The use of these systems under abnormal conditions is prescribed by BFNP Emergency Operating Instruction No. 41.

7.7 Impact of Post Containment Failure Phenomena on Injection Availability

Having reviewed the BFNP containment system design, static overpressurization containment failure mechanisms and phenomena, and long term reactor core cooling requirements and capabilities, the possible impacts of containment failure phenomena on reactor vessel injection systems availability will now be examined.

Following drywell liner failure, steam will begin dumping into the refueling floor, the pressure suppression chamber room, and various other reactor building floors via the flow paths discussed in Section 7.3. As described in Section 7.3, the suppression chamber room is connected to the rooms containing the HPCI, RCIC, RHR, core spray, control rod drive hydraulic and RHR drain pumps via open manways. The rate at which these rooms fill with steam is, of course, dependent upon the drywell rupture area and the flow paths involved. It is reasonable to assume that the HPCI, RCIC, RHR, core spray, CRD, and RHR drain pump room atmospheres would eventually become filled with saturated steam [i.e., 100°C (212°F) and 100% relative humidity]. Due to the lower surface temperatures of the room walls and equipment, significant amounts of condensation would be expected. Since the room coolers for these areas are not designed to function under such conditions, it is probable that the HPCI, RCIC, core spray, RHR, CRD, and RHR drain systems would be rendered inoperable following containment failure.*

A review of Table 7.2 indicates that there are four remaining systems which could inject water into the reactor vessel after the six systems noted above were rendered inoperable. Only one of these (the SLC system) is a high pressure system, and as described in Chap. 9, the pumping capacity of this system is insufficient to maintain the reactor vessel level above the top of the core at the time containment failure occurs. If the reactor can be depressurized following containment failure, the pumping capacity of any one of the three remaining low-pressure injection systems listed in Table 7.2 would be sufficient to cool the reactor core - provided the containment failure phenomena do not result in a LOCA. This is a particularly interesting consideration, since it is difficult to envision a mechanism by which all three of these systems would be rendered inoperable <u>unless</u> the vessel feedwater, recirculation, head spray, and core spray lines were severed as a result of containment failure phenomena. Following containment depressurization, remote manual operability of a single relief valve would be sufficient to enable the operator to utilize the low pressure emergency injection systems discussed above. Since containment pressure will drop following drywell liner rupture, such

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*As discussed in Chapter 3, the HPCI and RCIC would be rendered inoperable before containment failure in this accident sequence because of high temperature in the suppression chamber room.

remote SRV actuation should be possible (see Chap. 3) if the SRV actuators and control air systems are not damaged by the disruptive forces associated with containment blowdown.

7.8 Loss of DHR Accident Containment Failure Phenomena - Summary

An assumption commonly employed in probabilistic risk assessments (PRAs) of the Loss of DHR accident is that post containment failure phenomena result in the loss of all vessel injection capability.[7.22-7.24] This is a particularly critical assumption for two reasons. First, the Loss of DHR sequences have commonly been held to be risk dominating sequences in BWRs with Mark I containments, and second, the probability of injection loss following containment failure is rarely, if ever, incorporated in the event sequence probability calculations. Under these circumstances, an examination of the validity and implications of this assumption is particulary important.

A thoughtful review of the discussions presented in Sect. 7.1 through 7.7 will reveal that there are actually six possible outcomes of the Loss of DHR accident containment failure event. These six scenarios are listed in Table 7.3. Scenarios 1 and 4 are not expected to result in core melting. Scenario 1 is similar to the large-break LOCA design basis accident except that decay heat levels are significantly lower. Scenario 4 would not result in core melting since, as previously discussed, less than 0.006 m^3/s (100 gpm) of injection is necessary to cool the core at the time of containment failure. The outcome of scenario 5 depends upon the amount of vessel injection available; this scenario is not expected to result in core melting if the reactor vessel can be depressurized and any injection system other than the SLC system is available following containment failure.

Scenarios 3 and 6 would definitely lead to core melt. Scenario 6 is the common PRA assumption discussed above. This scenario corresponds to a situation in which all of the injection systems in Table 7.2 are rendered inoperable due to harsh environmental conditions following containment rupture, or loss of high pressure injection capability due to harsh environmental conditions together with failure to regain remote control of any one SRV following drywell failure. Since it seems unlikely that the RHR service water and condensate trains would be disabled due to environmental conditions, it appears that the probability of this scenario is dominated by the product of the probability of losing the HPCI, RCIC, and CRD hydraulic systems, and the probability of failure to regain remote control of a single SRV following containment rupture.

Scenario 3 is rarely, if ever, discussed in probabilistic risk assessment studies of the loss of decay heat removal capability sequence. In this scenario, total loss of injection could occur due to both environmental conditions in the reactor building following drywell rupture, and severence of some reactor vessel injection lines during the drywell blowdown transient (i.e., a LOCA). Detailed analysis of this sequence is beyond the scope of this report, however, it is clear that this scenario would result in a more severe accident than scenario 6, since (unlike scenario 6) the fission products released prior to reactor vessel melt-through would bypass the pressure suppression pool, directly entering the containment atmosphere.

7.9 Loss of DHR Post Containment Failure Event Sequence - Implications

In summary, it is evident that there are six possible scenarios for the loss of DHR event sequence following containment failure. Two of the scenarios (1 and 4) would not lead to core melting, while two scenarios (3 and 6) would result in a severe accident. The outcome of the two remaining scenarios (2 and 5) is dependent on the amount of injection remaining available. Historically, probabilistic risk assessments of the Loss of DHR accident have ignored all except the sixth scenario described above. As discussed in Sect. 7.8, this might lead to non-conservative estimates of accident consequences since scenario 3 would involve direct release of fission products into the drywell atmosphere prior to failure of the reactor vessel bottom head (bypassing the suppression pool scrubbing capability). It appears that the probability of the traditional "loss of injection following containment failure" PRA assumption (scenario 6) is dominated by the probability of failure of high pressure injection systems due to environmental concerns coupled with failure to regain control of a single SRV.* The total failure of all reactor vessel injection systems due to post containment failure environmental conditions alone seems quite unlikely due to the physical location of the systems.

The remaining mechanism by which all reactor vesel injection capability could be lost is by severance of reactor vessel injection piping during the disruptive blowdown of the drywell. This accident sequence (scenario 3, Table 7.3), which could be induced by a violent drywell depressurization transient following containment failure, has not been considered in previous BWR probabilistic risk assessments. Due to the suppression pool bypass phenomena previously described, this scenario would lead to the most severe fission product releases of the six scenarios identified in the analysis.

Future probabilistic risk assessments should consider and assign probabilities to each of the six scenarios listed in Table 7.3 rather than assume that scenario 6 is the only valid event sequence for the Loss of DHR accident after containment failure.

Finally, it does appear that the total probability of a loss of DHR induced core melt accident could be significantly reduced if the emergency operating instructions required that the operators vent the primary containment as necessary to preclude drywell failure by over-pressurization, thus reducing the probability of damage to the reactor vessel water injection lines and the SRV control air system.

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*Possibly because of loss of the drywell control air system.

References for Chapter 7

7.1 Browns Ferry FSAR, Chap. 5.

7.2 TVA Browns Ferry Nuclear Plant Drawing #41N711-1.

7.3 TVA Browns Ferry Nuclear Plant Drawing #41N720.

7.4 Browns Ferry FSAR, p. Q12.4-1/Q12.4-2.

7.5 Browns Ferry FSAR, p. Q12.2.2.11-1.

7.6 TVA Browns Ferry Nuclear Plant Drawing #41N720.

7.7 TVA Browns Ferry Nuclear Plant Drawing #47W481-11.

7.8 TVA Browns Ferry Nuclear Plant Drawing #47W482-4.

7.9 Browns Ferry FSAR, p. Q12.5-1/Q12.5-2.

7.10 Browns Ferry FSAR, Fig. 5.2-19.

7.11 Browns Ferry Nuclear Plant Hot License Training Program Manual, Vol. 4.

7.12 Reactor Safety Study, U.S. Nuclear Regulatory Commission, WASH-1400, 1975.

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

7.14 Dick Cheverton, Private Communication, November 16, 1982.

7.15 L. G. Greimann, Private Communication, October 28, 1982.

7.16 M. Barrere, A. Jaumotte et al., Rocket Propulsion, Elsevier Publishing Co., 1960, p. 25.

7.17 Stephen Whitaker, Introduction to Fluid Mechanics, Prentice-Hall, Inc., 1968, p. 255.

7.18 Browns Ferry FSAR, Chap. 12, p. 12.2-5.

7.19 Browns Ferry Nuclear Plant Emergency Operating Instruction No. 41, November 7, 1979.

7.20 Browns Ferry FSAR, Chap. 4.

7.21 W. A. Condon et al., SBLOCA Outside Containment at Browns Ferry Unit One - Accident Sequence Analysis, NUREG/CR-2672, Vol. 1, ORNL/TM-8119/V1 (November 1982), Sect. E.3.

7.22 S. W. Hatch, P. Cybulskis, R. O. Wooton, Reactor Safety Study Methodlogy Applications Program: Grand Gulf #1 BWR Power Plant, NUREG/CR-1659/4, October 1981.

7.23 S. E. Mays, J. P. Poloski et al., Interim Reliability Evaluation Program: Analysis of the Browns Ferry Unit 1 Nuclear Plant, Main Report, July 1982.

7.24 Limerick Generating Station Probabilistic Risk Assessement, June 1982.

Table 7.1. Principal design parameters and characteristics of the BFNP primary containment

Pressure suppression chamber	
Internal design pressure, psig	56
External design pressure, psig	2
Drywell	
Internal design pressure, psig	56
External design pressure, psig	2
Drywell free volume, ft^3	159,000
Pressure suppression chamber free volume (min.), ft^3	119,000
Pressure suppression pool water volume (max.), ft^3	135,000
Submergence of vent pipe below pressure suppression pool surface (normal), ft	4
Design temperature of drywell, °F	281
Design temperature of pressure suppression chamber, °F	281

Table 7.2. Emergency reactor vessel (RPV) water injection capabilities

System	Steam source	Water source	Injection point	Flow rate		Shutoff head	
				m^3/s	gpm	MPa	psid
HPCI	RPV or Auxiliary boiler	CST PSP	Feedwater line	0.31	5000	>7.9	>1150
RCIC	RPV or Auxiliary boiler	CST PSP	Feedwater line	0.04	600	>7.9	>1150
Core spray	N/A	PSP CST	Spray header	0.79	12500	2.4	342
RHR (LPCI)	N/A	PSP	Recirc loops Head spray	2.52	40000	2.3	331
CRD	N/A	CST	Via control rod drives	0.0103	170	10.3	~1500
			Via Feedwater line[a]	0.0124	200	10.3	~1500
SLC	N/A	Demineralized H_2O	SLC sparger	0.0035	56	9.7	1400
Condensate	N/A	CST via Condenser Hotwells	Feedwater line	1.89	30000	2.9	415
Condensate transfer	N/A	CST	Core spray header Head spray Recirc loops	0.06	1000	Unknown[b]	
RHR drain	N/A	PSP CST	Recirc loops Head spray	0.10	1600	0.4	65
RHR service water (Standby coolant supply system)	N/A	River	Recirc loops Head spray	0.57	9000	1.1	162

[a]See Sect. 7.6.

[b]The rated head is 200 ft.

Table 7.3. Loss of DHR accident post containment failure scenerios

Scenerio		Probable outcome
(1) Cont. failure + LOCA + all injection	=	no melt
(2) Cont. failure + LOCA + some injection	=	?
(3) Cont. failure + LOCA + no injection	=	melt
(4) Cont. failure + no LOCA + all injection	=	no melt
(5) Cont. failure + no LOCA + some injection	=	no melt[a]
(6) Cont. failure + no LOCA + no injection[b]	=	melt

[a]Assumes some injection capability in addition to SLC.

[b]Other than SLC.

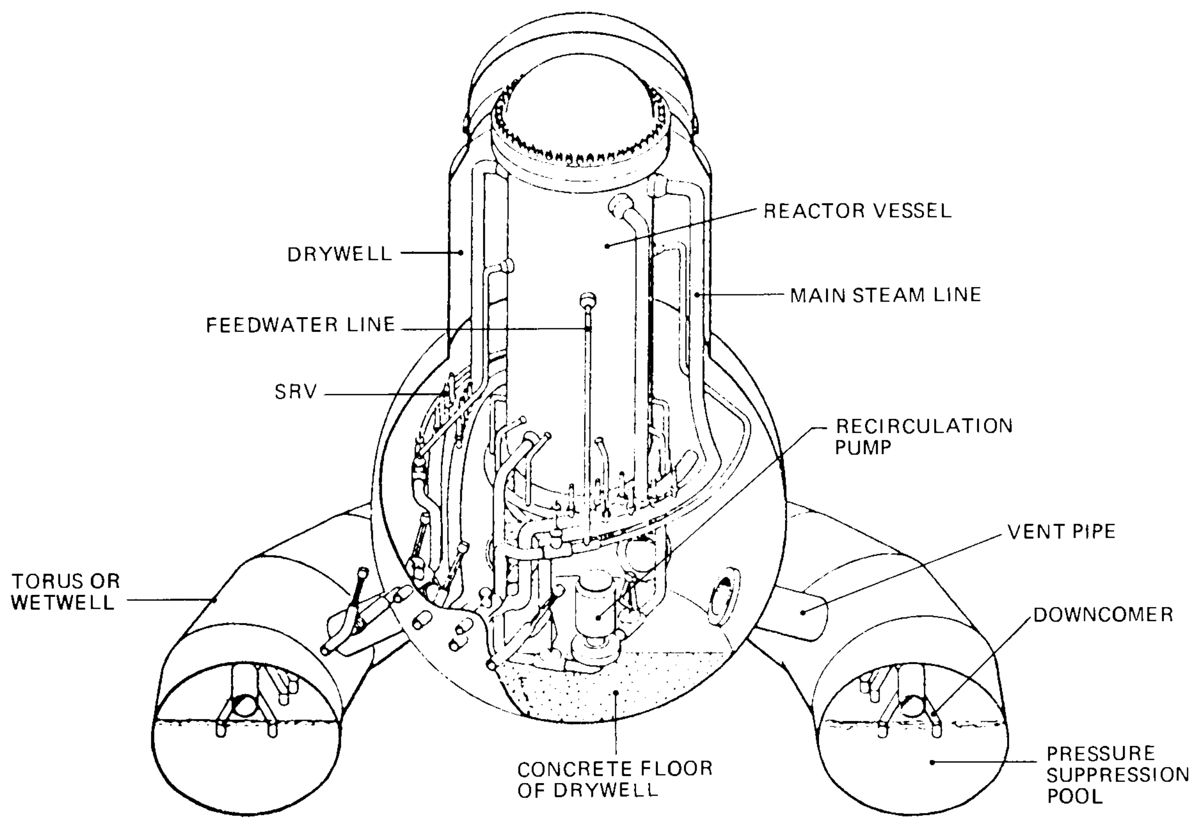

Fig. 7.1. BWR MARK I containment system.

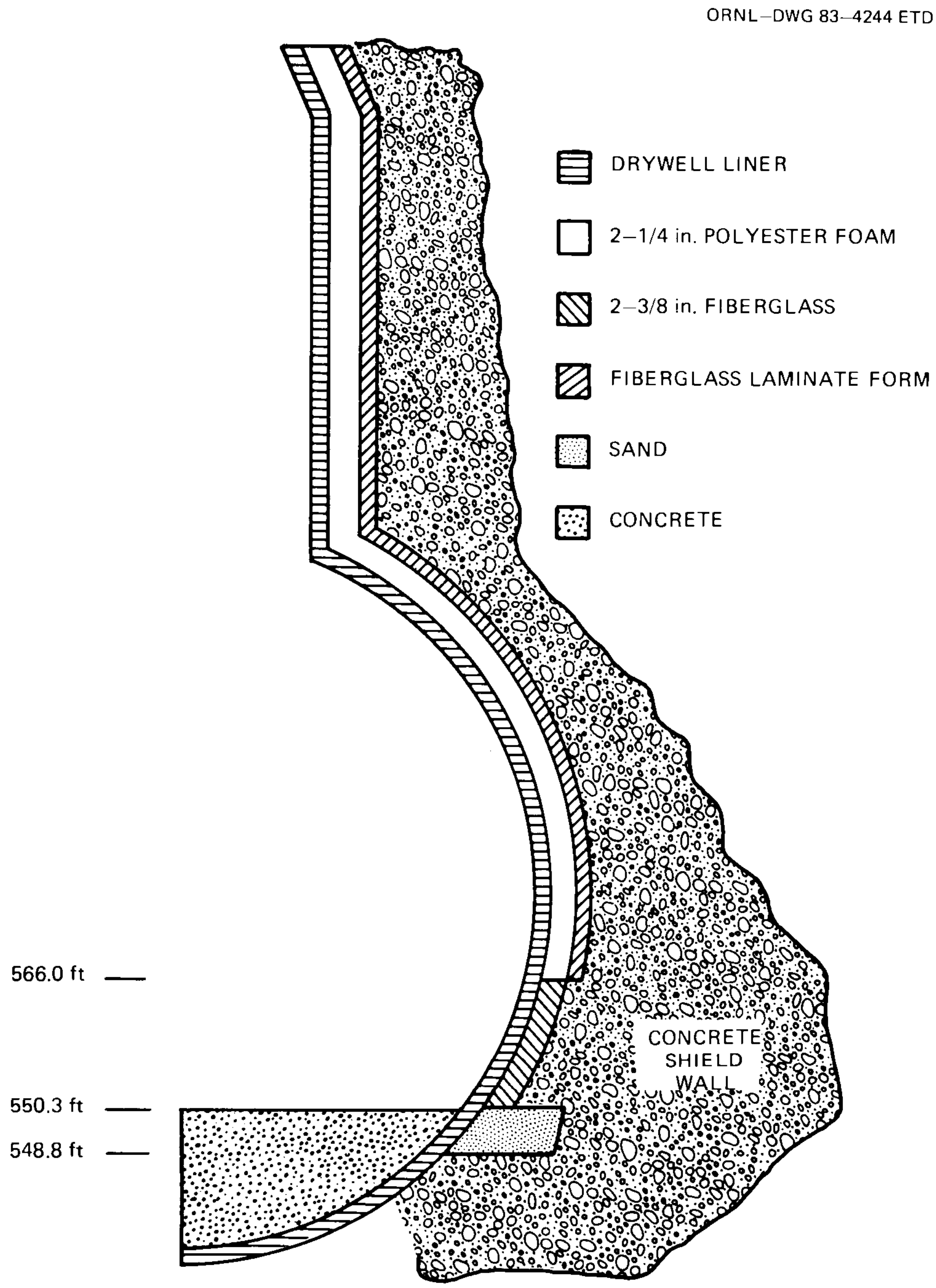

Fig. 7.2. BFNP drywell liner – concrete shield wall gap construction.

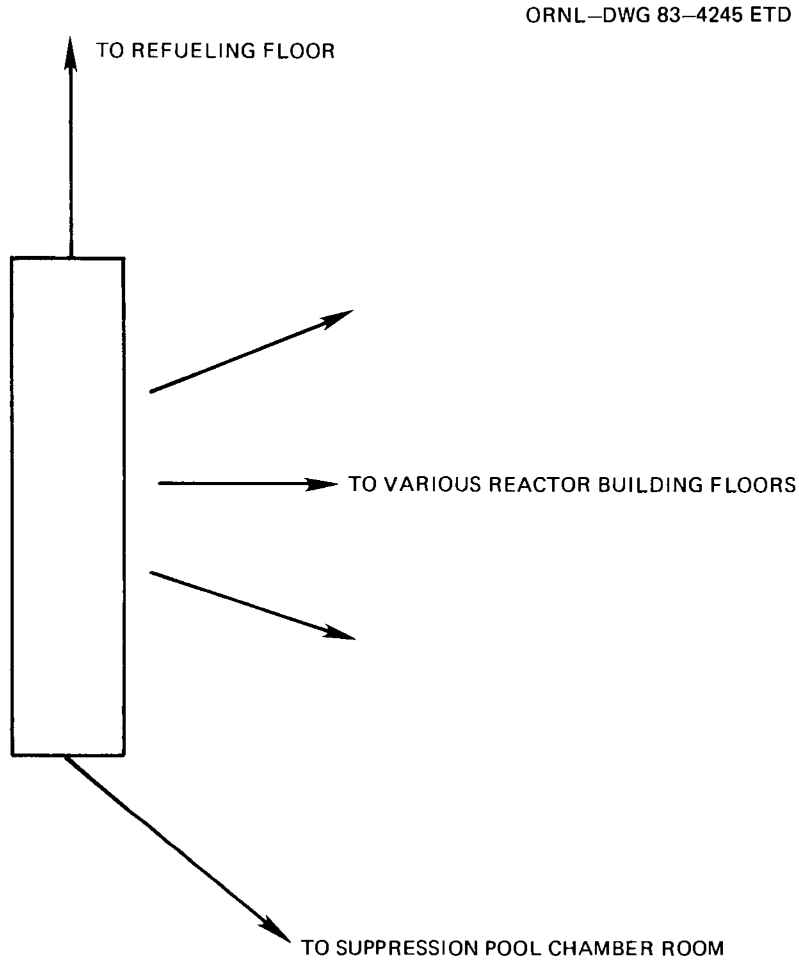

Fig. 7.3. Drywell liner – concrete shield wall gap venting paths.

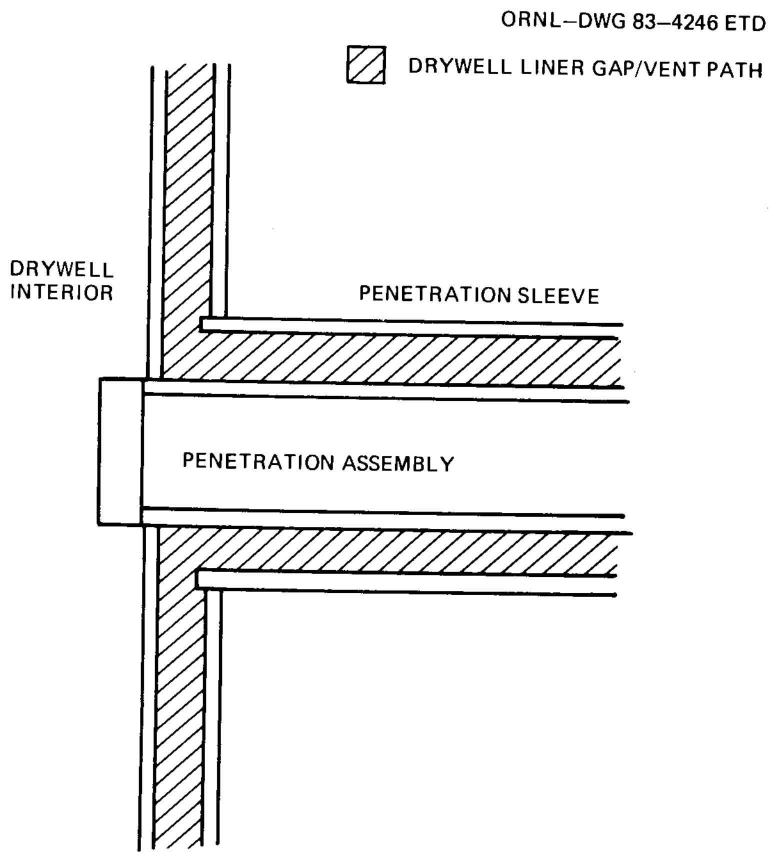

Fig. 7.4. Drywell penetration sleeve configuration.

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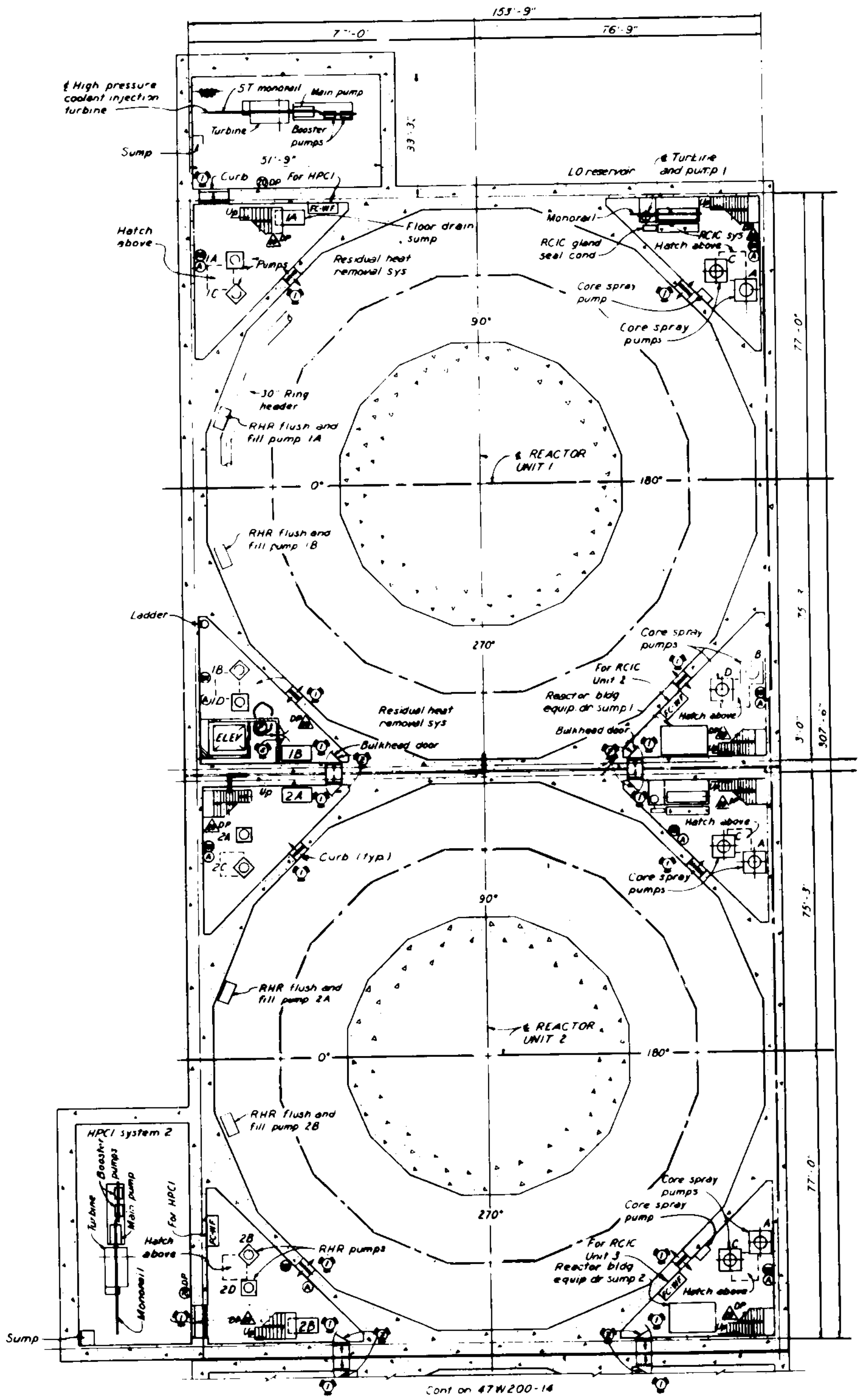

Fig. 7.5. Unit 1 and Unit 2 pressure suppression chamber rooms and ECCS pump rooms (Elevation 519 ft.).

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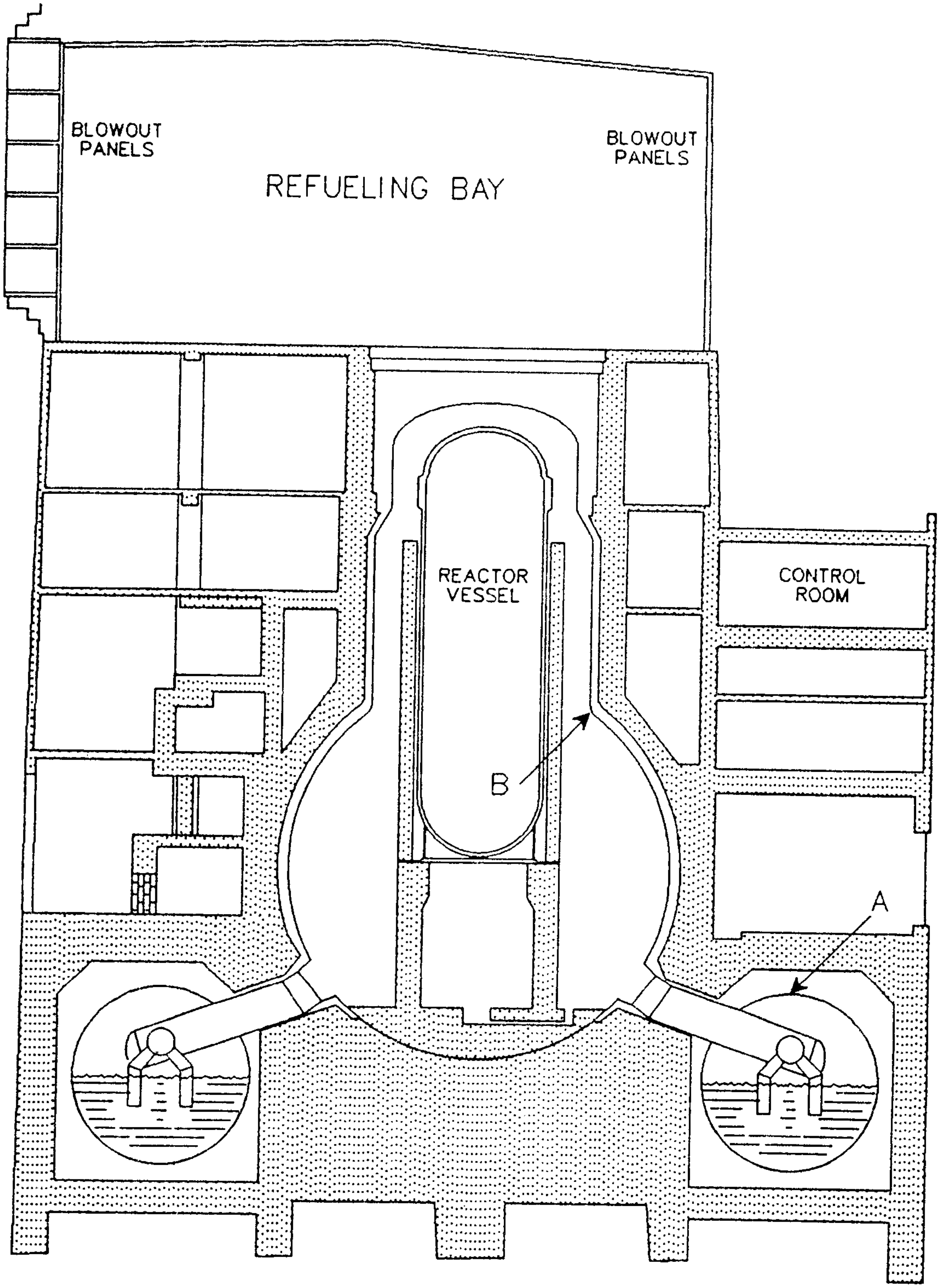

Fig. 7.6. Static over-pressurization containment failure locations.

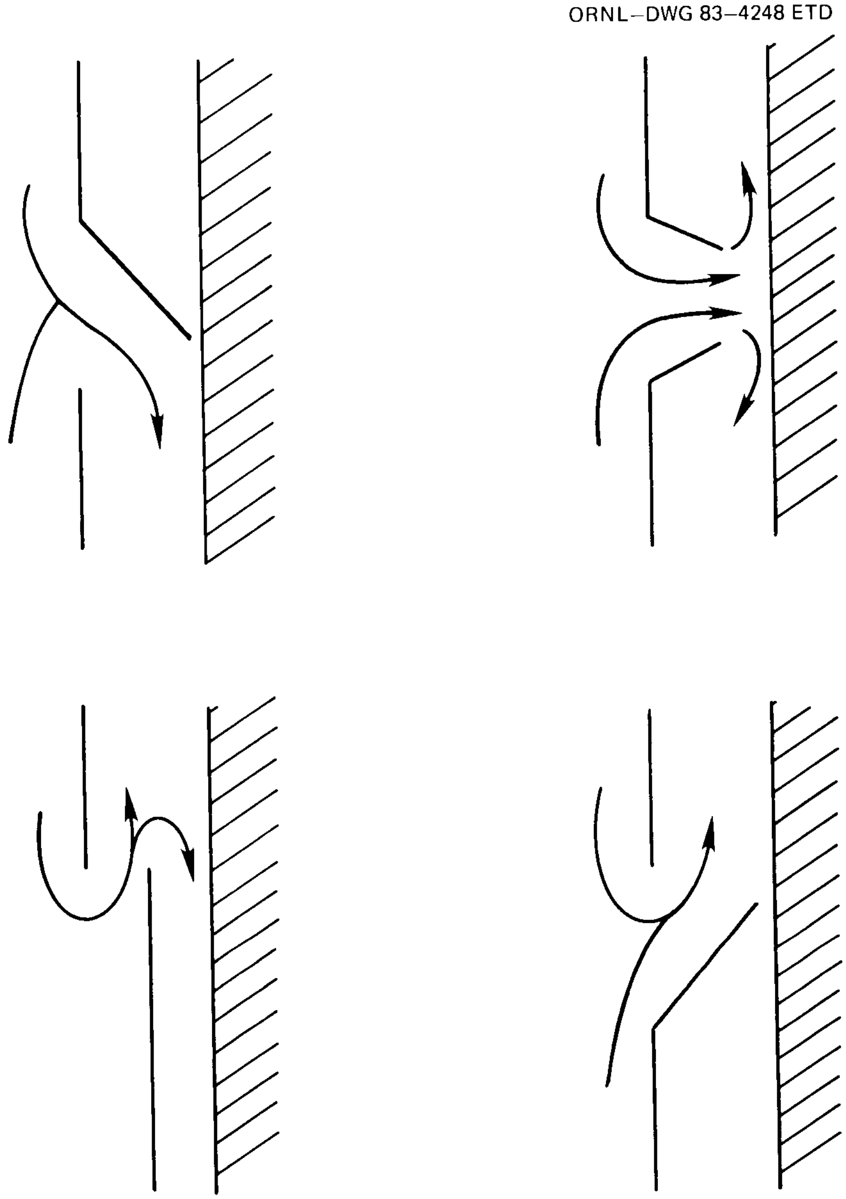

Fig. 7.7. Drywell liner rupture configurations.

8. ANALYSIS OF CONTAINMENT FAILURE AND POST CONTAINMENT FAILURE LOSS OF DHR EVENT SEQUENCE

8.1 Introduction

Events preceeding containment failure in the uniform pool heating Loss of DHR sequence were described in Chap. 3. This chapter will describe the containment failure event and post containment failure event sequence for this accident (Scenario 6, Table 7.3). This scenario, which is commonly analyzed in probabilistic risk assessments, involves a containment failure event followed by loss of all reactor vessel water injection capability. The primary containment is assumed to fail in the drywell at the juncture of the cylindrical and spherical portions of the liner with a failure area of 0.929 m^2 (10 ft^2). The primary system is assumed to maintain its integrity and a pressurized boiloff of the water in the reactor vessel follows containment failure.

All analyses described in this section are based on evaluations performed with the MARCH[8.1] code as installed and modified[8.2] at ORNL. The initial conditions for the analysis were derived from the BWR-LACP results discussed in Chap. 3. Table 8.1 is a summary of the initial conditions incorporated in the MARCH analysis described here.

8.2 Loss of DHR with Loss of Injection Following Containment Failure

Table 8.2 presents a summary of the accident event timing for this scenario as predicted by MARCH. The MARCH results for this accident are shown in Figs. 8.1 through 8.13. Containment failure is predicted to occur 35 1/4 h after scram, as the drywell pressure (Fig. 8.9) reaches 0.908 MPa (117 psig). All water flow into the reactor vessel is assumed to cease at that time. Core uncovery is not predicted to occur until almost 2-1/2 h (147 min) after loss of injection (Fig. 8.1). This relatively slow core uncovery process is due to the low decay heat levels involved and the large water inventory in the reactor vessel at the time of loss of injection.

The only mechanism for water loss from the reactor vessel prior to vessel head failure is via the reactor vessel relief valves, which are cycling frequently during this period (Figs. 8.2-8.3). The large relief valve flow variations shown in Fig. 8.3 are due to errors in MARCH's relief valve model, which represents the valves as orfices with flow rates dependent upon downstream (suppression pool) pressure. In actuality the BFNP SRVs are critical flow devices that are designed to maintain constant flows independent of containment pressure. This MARCH modeling error accounts for the SRV flow spike (Fig. 8.3) and rapid primary system depressurization (Fig. 8.2) predicted to occur after containment failure. Reactor water temperature (Fig. 8.4) is predicted to hold very near the primary system saturation temperature throughout the period prior to vessel failure.

Following core uncovery at 37.7 h (2262 min) after inception of the accident sequence, maximum fuel temperatures are predicted to rise rapidly to 1533 K (2300°F) (Fig. 8.5) and the Zircaloy fuel cladding is predicted to oxidize rapidly (Fig. 8.6). Hydrogen leakage rates into the suppression pool peak at ~0.25 kg/s (33 lbm/m) (Fig. 8.7). The energy from the Zircaloy oxidation reaction increases the fuel heatup rate (Fig. 8.5) resulting in initiation of fuel melting (Fig. 8.8) at 2321 min, almost 3 1/2 h after loss of injection. The length of time between loss of injection and inception of core melting in this accident is closely related to the low decay heat levels involved and the intensity of the Zr-H_2O reaction, since the heat generation from this reaction can easily equal or exceed the decay heat levels present 30 to 40 h after reactor shutdown. Once initiated, core melting continues at a moderate rate, until 75% of the core is molten (Fig. 8.8), at 40.2 h (2413 min) into the accident. At this time the core is allowed to slump onto the core plate, based upon a MARCH user input option.

Figures 8.9 through 8.13 are plots of the drywell pressure, leak rate, temperature, hydrogen molar fraction and liner temperature throughout the post containment phase of the sequence. Following drywell failure at 2115 min, containment pressure is predicted to drop rapidly as the containment depressurizes through the 0.929 m^2 (10 ft^2) (assumed) opening in the drywell liner. Drywell pressure drops to 0.4 MPa (55 psia) within 10 min and to atmospheric pressure within 64 min. During this period the drywell volumetric leak rate is predicted to hold rather steady at approximately 158 m^3/s (335,000 ft^3/min). This flow rate corresponds to a mass velocity of 177 m/s (558 ft/s) through the break. Following drywell failure, the temperature of the drywell atmosphere drops substantially, due to depressurization and flashing of the wetwell through the downcomers attached to the vent header. Following core uncovery at 2262 min, the temperature of the drywell atmosphere increases substantially due to gas and steam influx from the suppression pool which, in turn, is receiving the hot gas and steam from the uncovered core via the reactor vessel relief valves.

References for Chapter 8

8.1 R. O. Wooten and H. I. Avci, MARCH User's Manual, NUREG/CR-1711.

8.2 W. O. Condon, S. R. Greene, R. M. Harrington, S. A. Hodge, "SBLOCA Outside Containment at Browns Ferry Unit One-Accident Sequence Analysis," NUREG/CR-2672, November 1982.

Table 8.1. Initial conditions for LDHR sequence

Time from scram, h	34
RPV water level, in.	562
RPV pressure, psia	1072
RPV water temperature, °F	554
RPV water injection flow, gpm	102
Drywell pressure, psia	125
Drywell temperature, °F	431
Drywell relative humidity, %	26.5
Drywell liner temperature, °F	362
Assumed drywell failure size, ft^2	10

Table 8.2. Loss of DHR accident event timing

Event	Time from scram (min)	Time from LI[a] (min)
Containment failure	2115	
Loss of RPV injection	2115	0
Core uncovers	2262	147
Core melting begins	2321	206
Core slump	2413	298
RPV bottom head failure	2504	389

[a]Time from Loss of Injection.

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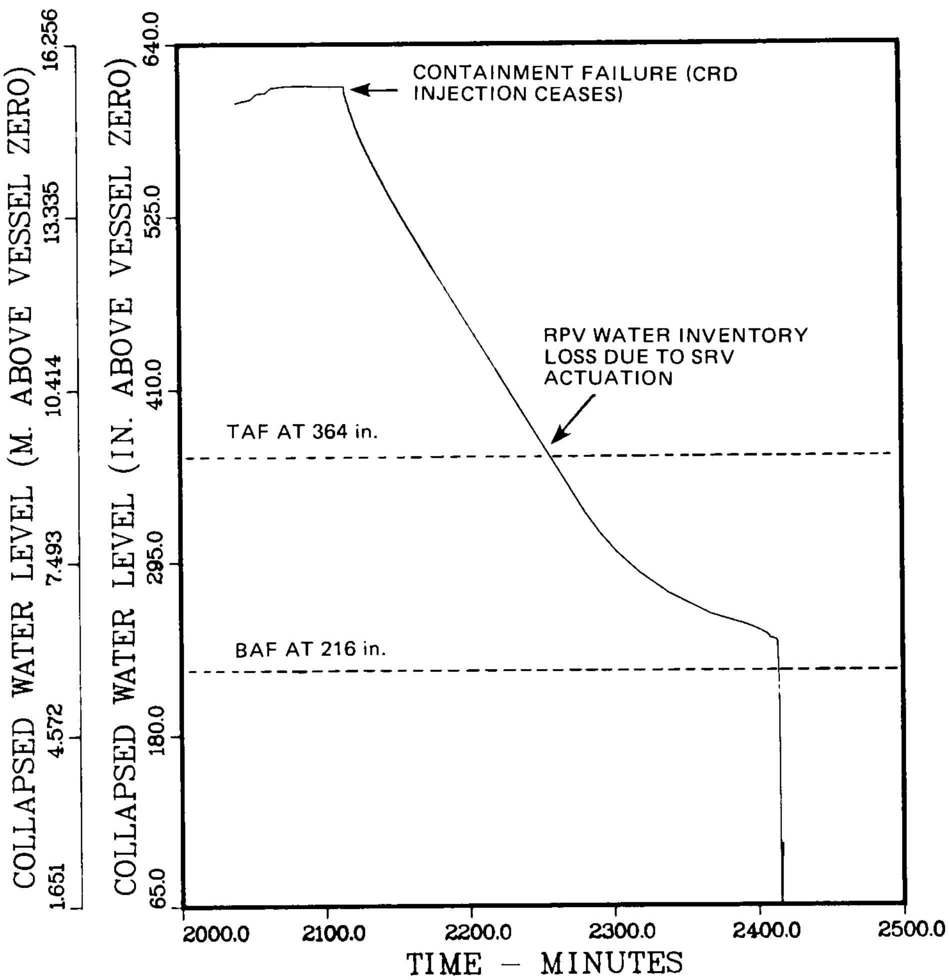

Fig. 8.1. Reactor vessel water level. (Time is counted from reactor scram).

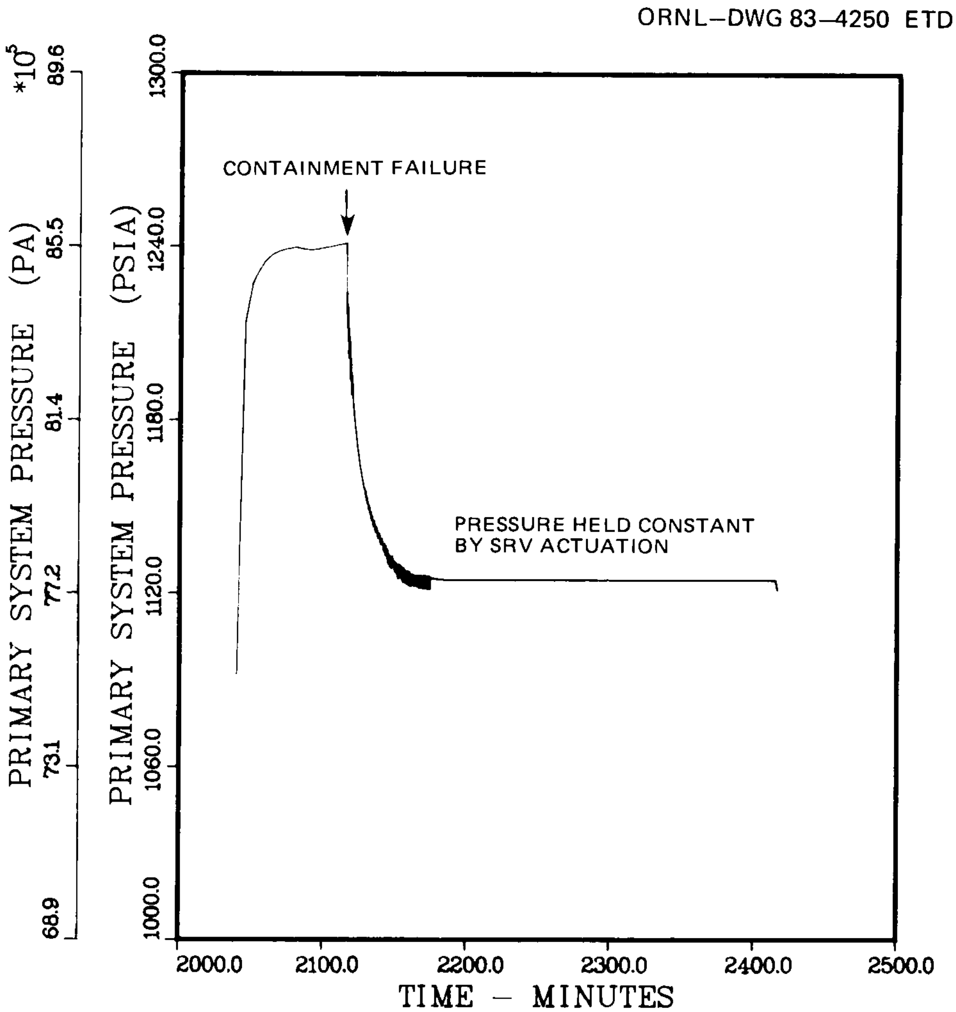

Fig. 8.2. Reactor vessel pressure.

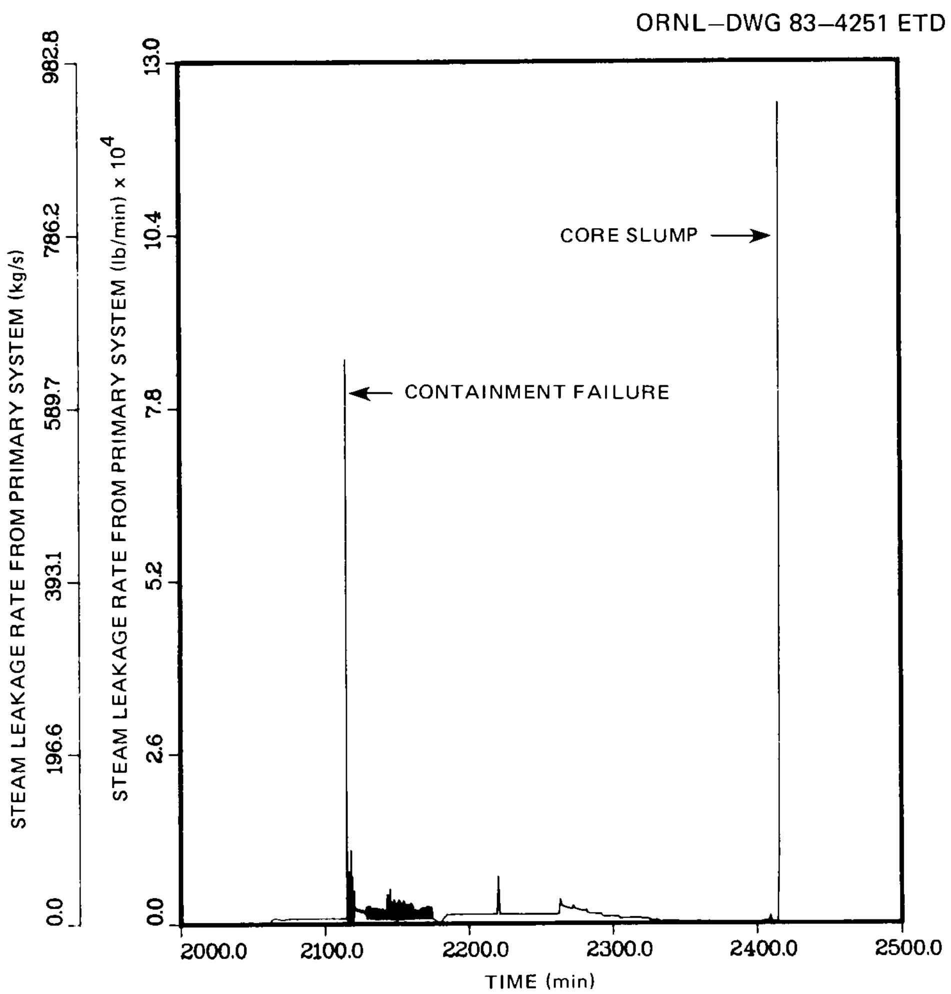

Fig. 8.3. SRV steam flow.

ORNL–DWG 83–4252 ETD

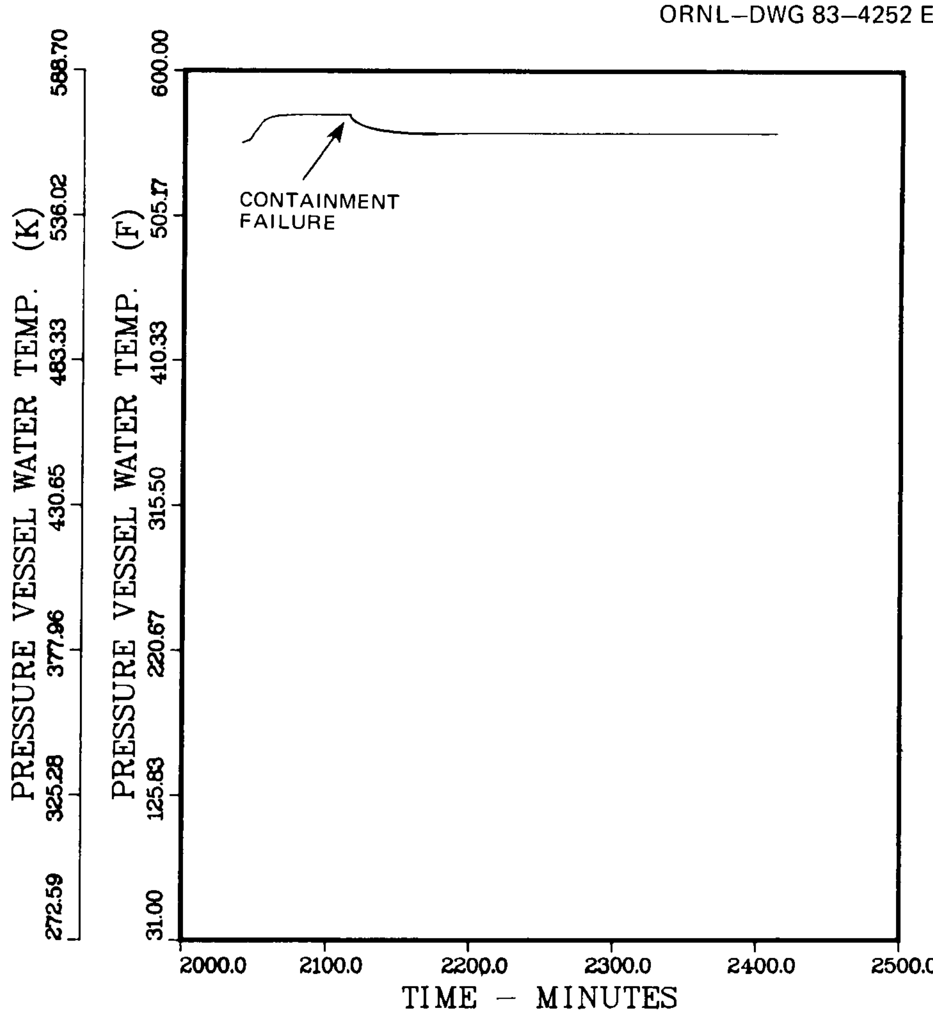

Fig. 8.4. Reactor vessel water temperature.

ORNL–DWG 83–4253 ETD

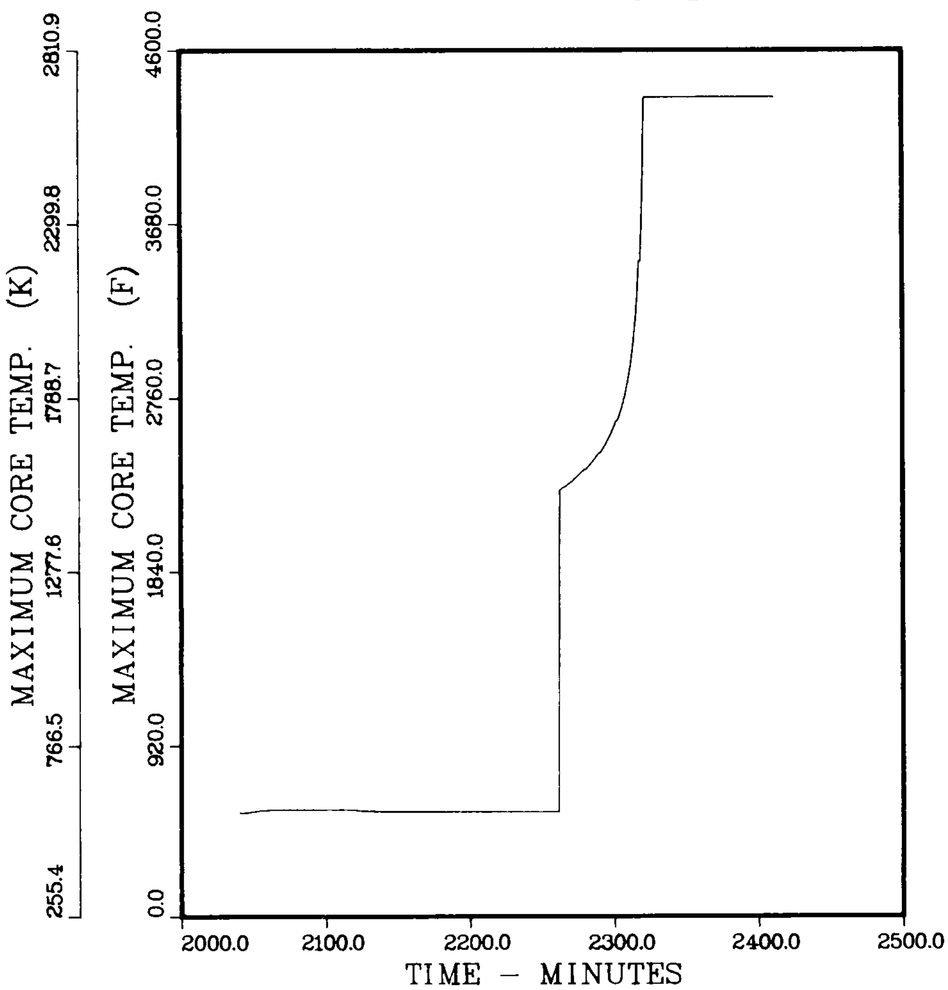

Fig. 8.5. Maximum fuel temperature.

ORNL-DWG 83-4254 ETD

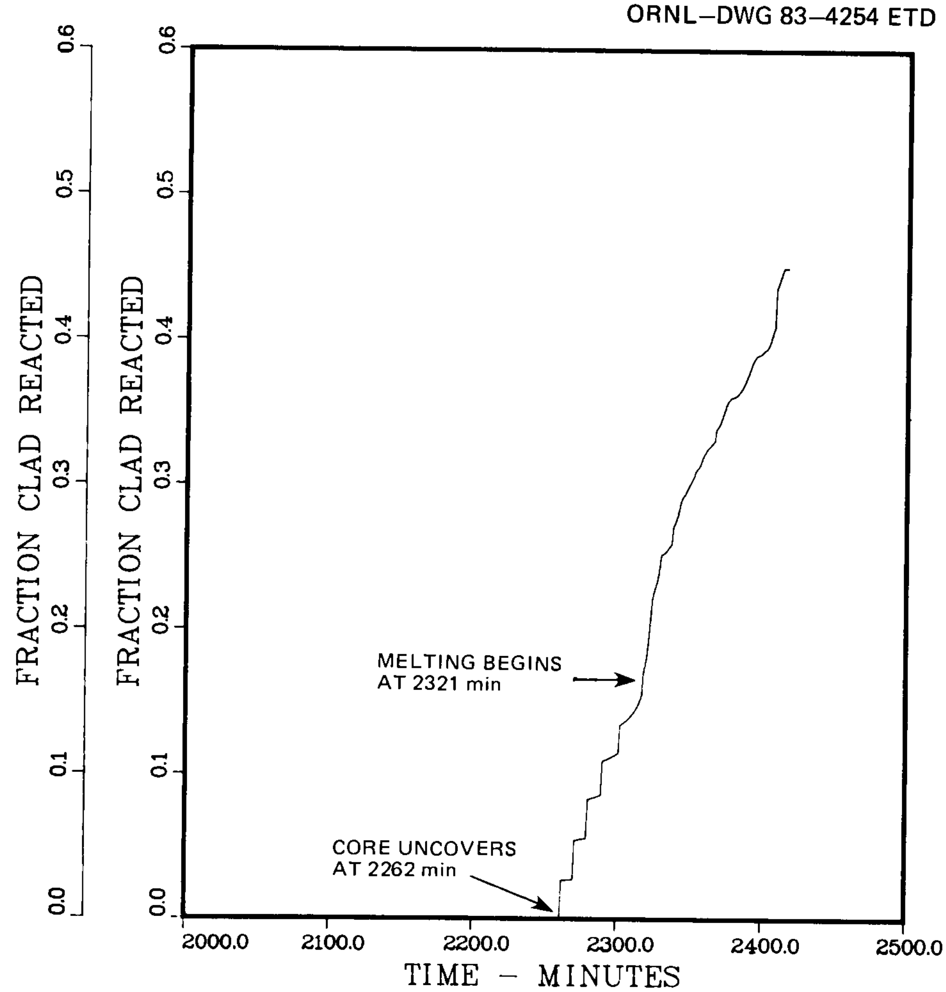

Fig. 8.6. Total fraction of fuel cladding reacted.

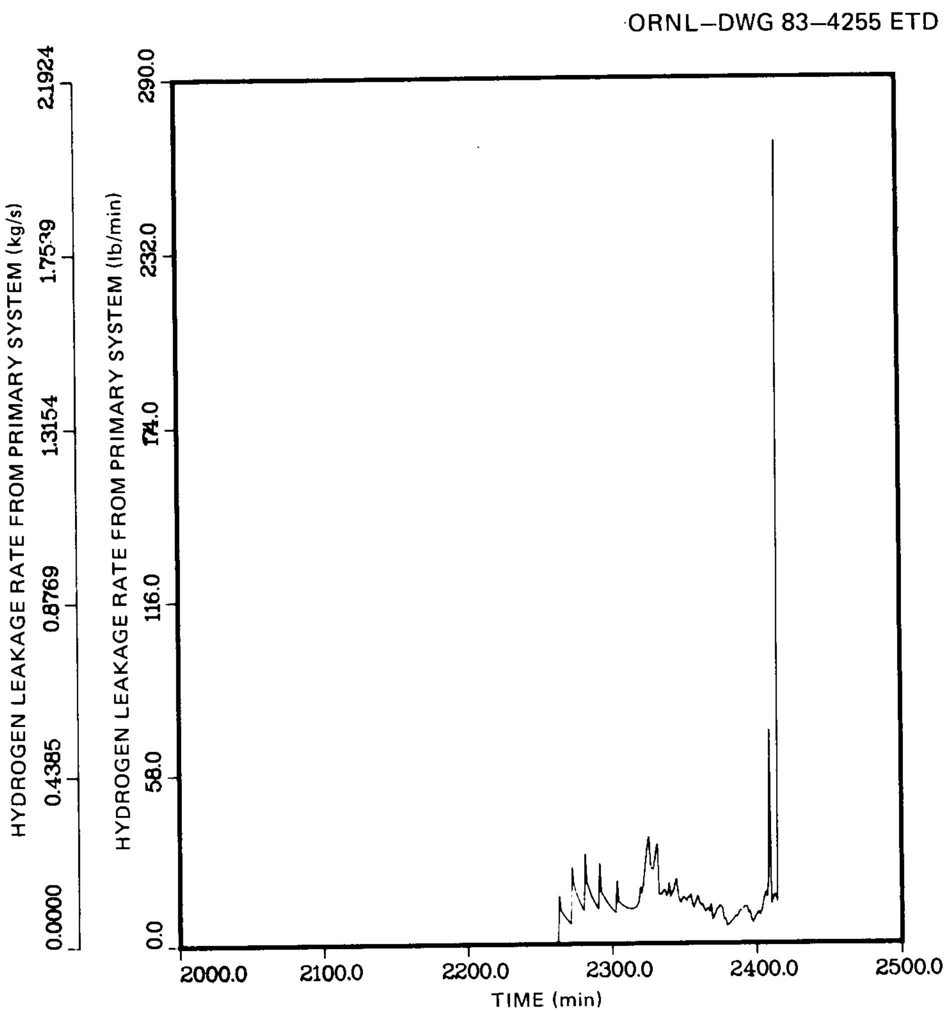

Fig. 8.7. Hydrogen leakage from the reactor vessel.

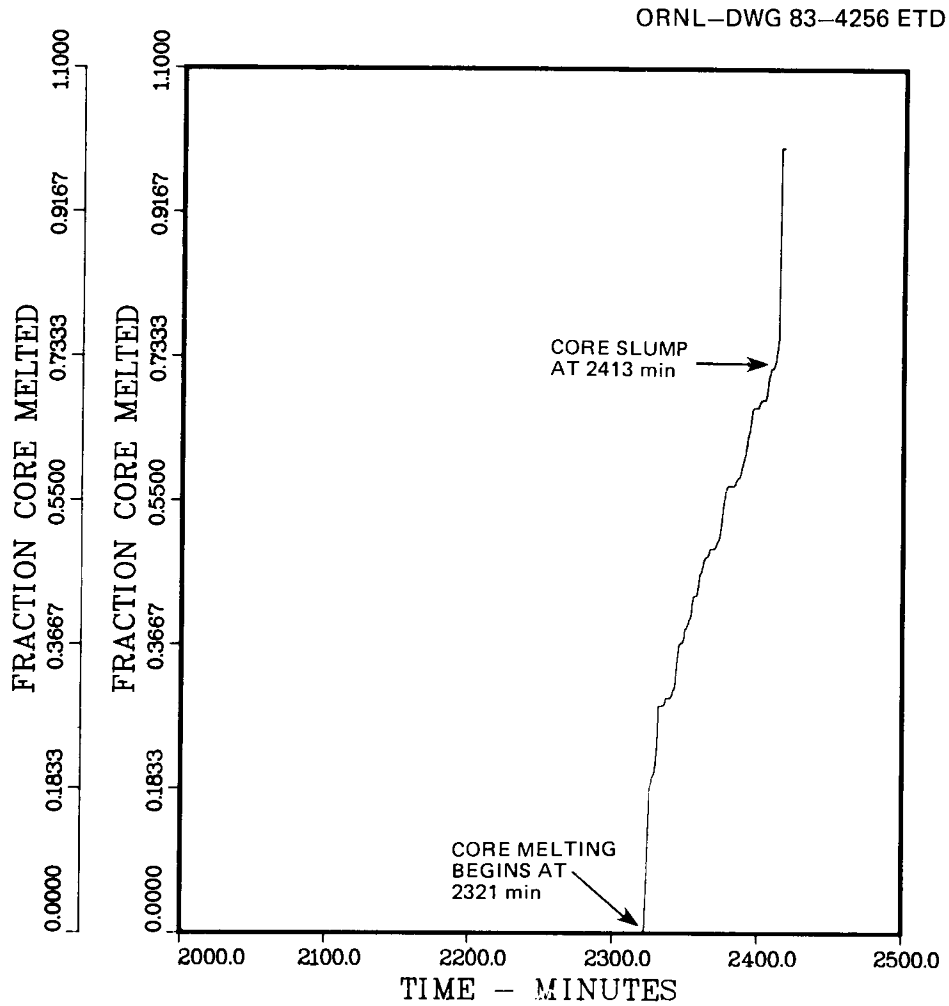

Fig. 8.8. Fraction of core melted.

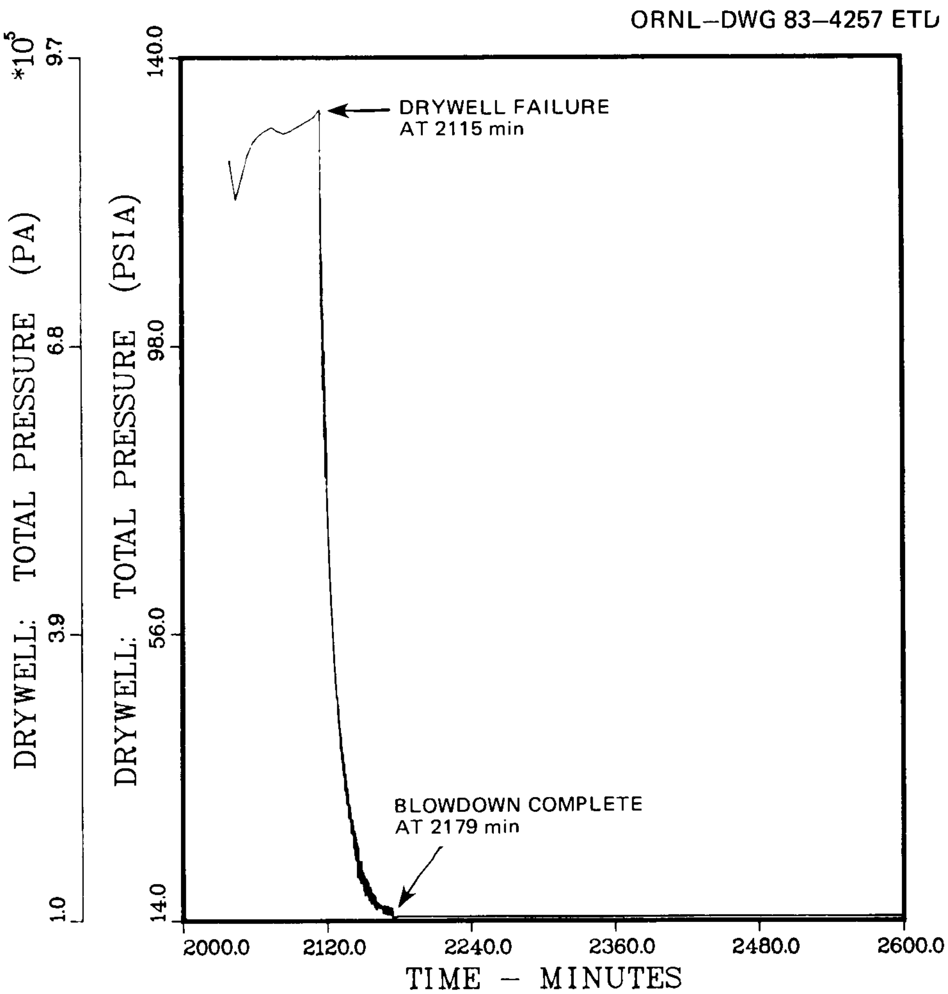

Fig. 8.9. Drywell pressure.

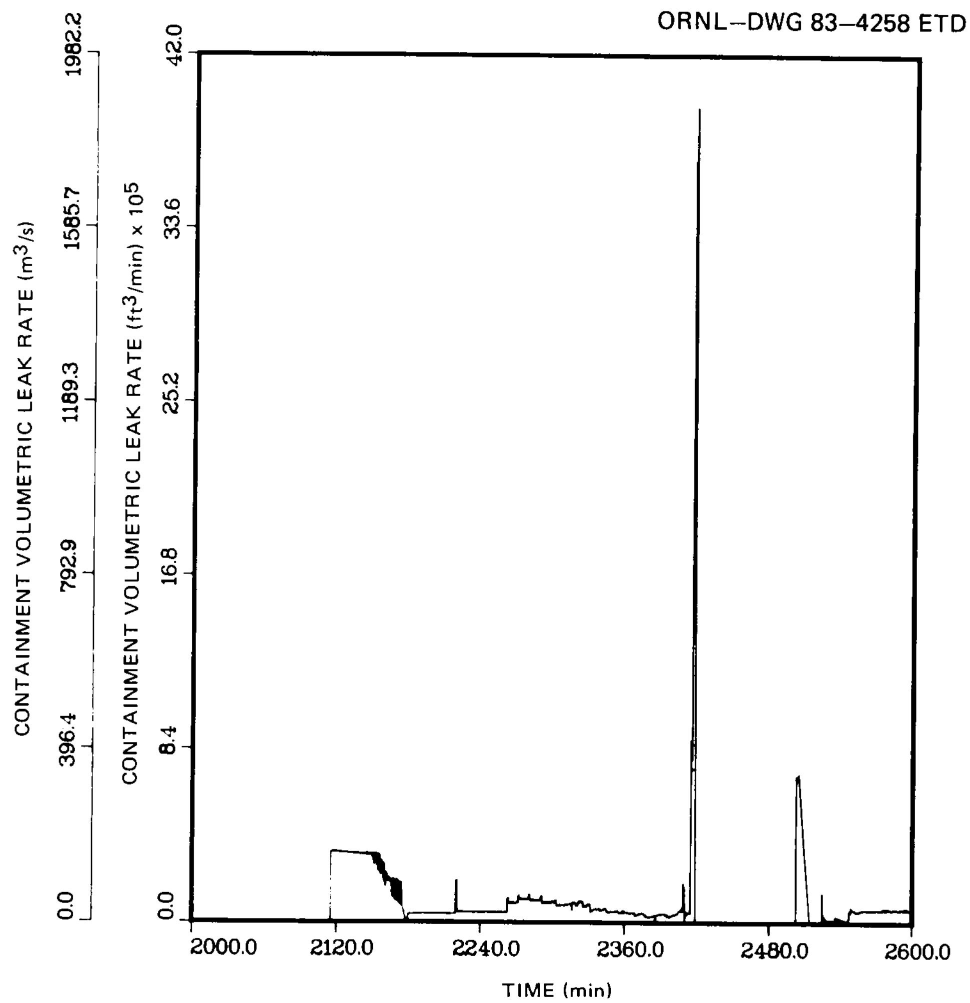

Fig. 8.10. Drywell volumetric leakage rate.

ORNL–DWG 83–4259 ETD

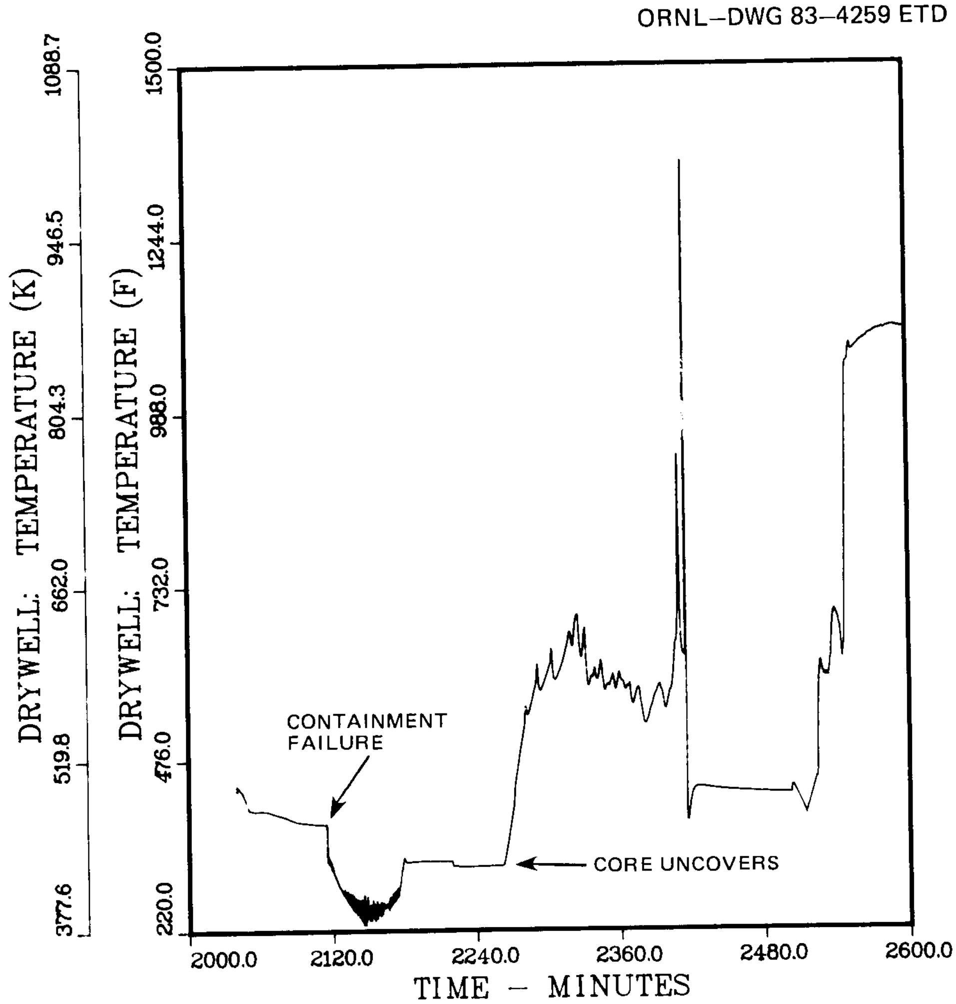

Fig. 8.11. Drywell temperature.

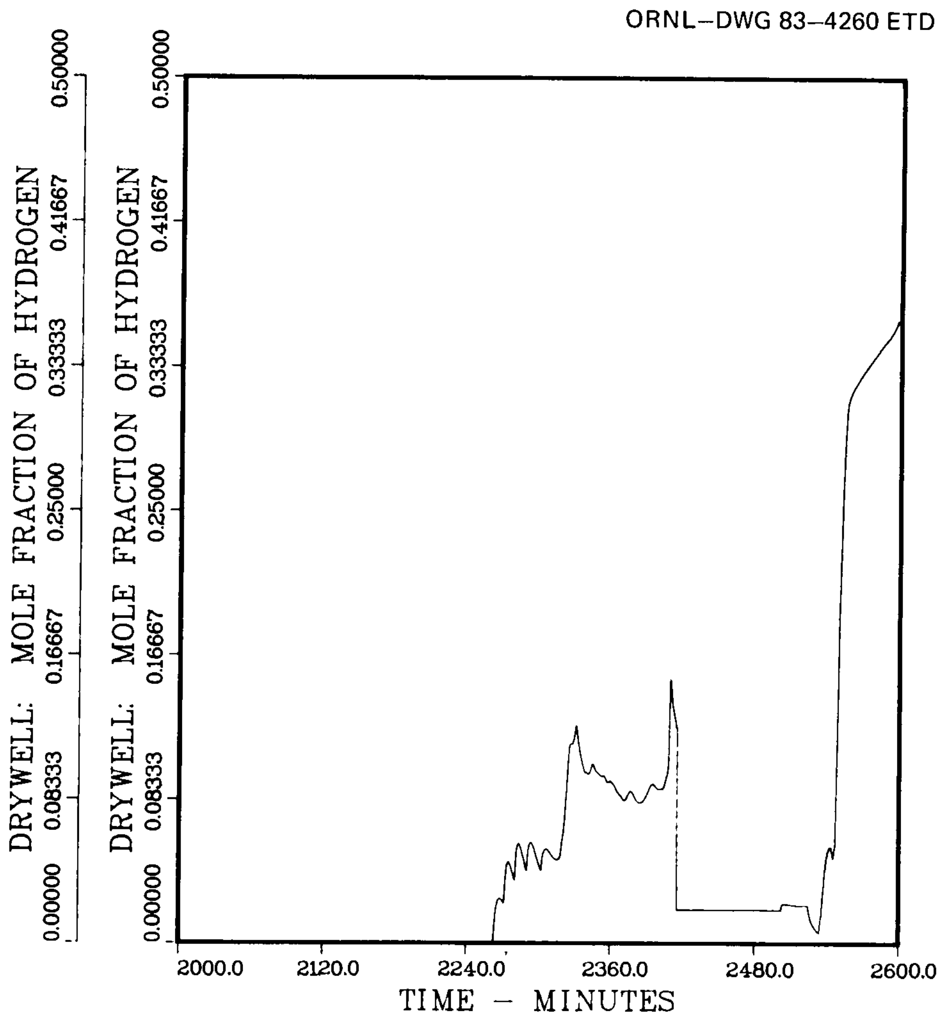

Fig. 8.12. Drywell molar hydrogen fraction.

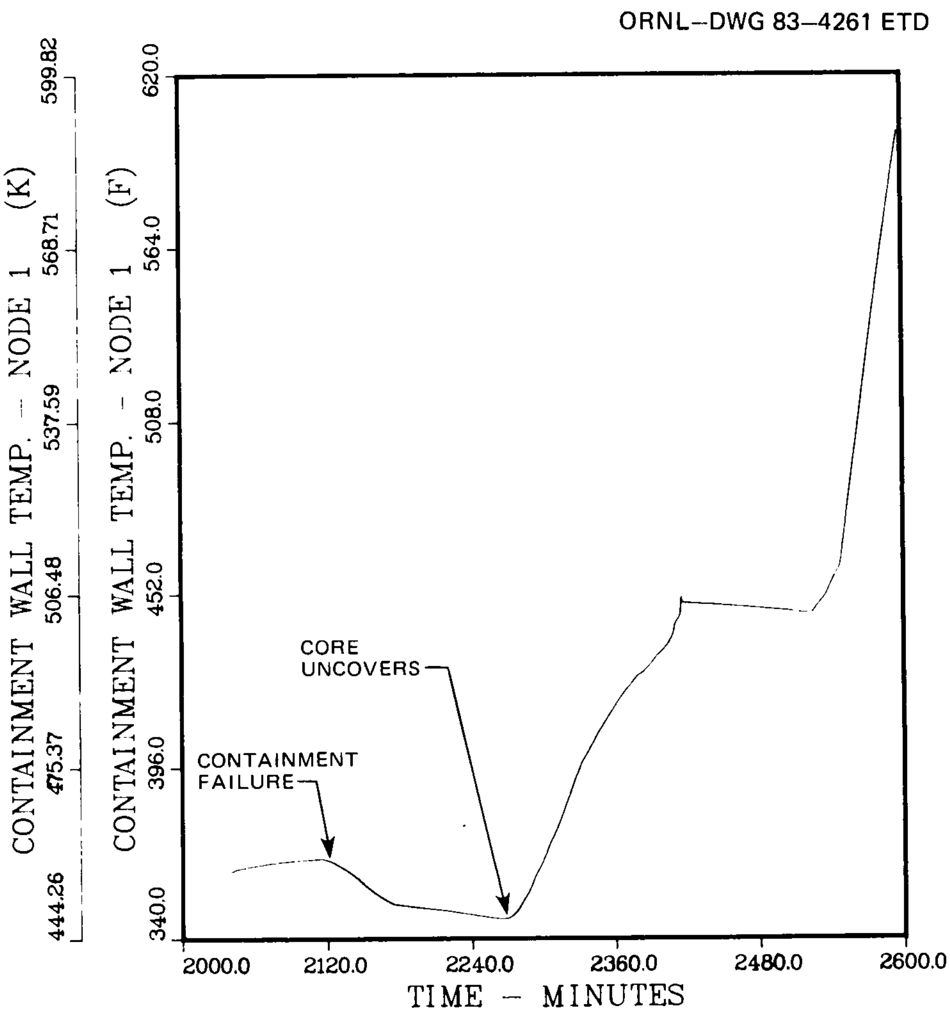

Fig. 8.13. Drywell liner temperature.

9. IMPLICATIONS OF RESULTS

The first purpose of this chapter is to provide a discussion of the present state of readiness at the Browns Ferry Nuclear Plant to cope with an accident sequence involving a long-term loss of decay heat removal (DHR). This accident involves an improbable combination of inability to use the main condenser, condensate, and feedwater systems for decay heat removal with an inability to use the residual heat removal (RHR) system in either the pressure suppression pool cooling mode or the shutdown cooling mode.* The unavailability of these RHR system operational modes might be due to complete system failure, or the failure might be confined to an inability to use the residual heat removal service water (RHRSW) system to remove heat from the heat exchangers in the RHR system. In the latter case, the RHR system could be used to circulate the pressure suppression pool water to promote mixing and avoid thermal stratification. Both cases of loss of RHR system decay heat removal function have been analyzed in this report. The available instrumentation, the level of operator training, the existing emergency operating instructions, and the overall system design at the Browns Ferry Nuclear Plant Unit 1 are discussed in Sects. 9.1 through 9.3 from the standpoint of adequacy in the event of a long-term Loss of DHR accident sequence.

The second purpose of this chapter is to provide a discussion of the impact of this detailed SASA analysis of the Loss of DHR accident sequence on the conclusions of the recently completed Interim Reliability Evaluation Program (IREP) report,[9.1] which identifies this accident sequence as a major contributor to the overall risk attendant to the operation of the Browns Ferry Nuclear Plant. This discussion is provided in Sect. 9.4.

9.1 Instrumentation

All control room and other plant instrumentation normally available after a reactor scram would be available for operator use during a long-term Loss of DHR accident sequence even if a prolonged loss of offsite power were involved. The primary containment parameters measured by the available instruments and displayed in the control room include the pressure and temperature of the drywell atmosphere, the temperature and level of the water in the pressure suppression pool, and the pressure in the wetwell. The range of indication and the associated alarms for each of these parameters are provided in Table 9.1.

As discussed in Sect. 7.4, the best-estimate pressure currently available for failure of the Browns Ferry Mark I primary containment is 0.908 MPa (117 psig). Therefore, during the latter stages of a loss of DHR accident sequence which the operators permitted to go to completion (i.e., containment failure by overpressurization), the drywell and wetwell pressure instruments would be off-scale high. The pressure suppression pool

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*These RHR system operational modes are described in Appendix A. The accident also implies inability to use the RHR system unit crosstie capability described in subsection A.5.5.

level instruments would be offscale-high after about 700 min (11.7 h) into the accident sequence as the suppression 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 almost all of the accident sequence.

Since the wetwell atmosphere would be virtually 100% steam during the latter part of the accident sequence, the pressure in the primary containment could be inferred from the pressure suppression pool temperature and the saturation tables.

9.2 Operator Preparedness

The Browns Ferry training simulator does have the capability to model the portion of a Loss of DHR accident sequence before drywell failure. However, a complete run-through of this slowly-developing accident sequence would require about 35 h of simulator time and it is doubtful that the simulator computer could continue to develop a realistic model of the plant response for such a long period of computation. For these reasons, a Loss of DHR sequence is not included in operator training although the importance of pressure suppression pool cooling and the methods for accomplishing this function are stressed.

After reactor scram or manual shutdown under accident conditions when the main condenser is not available, the pressure suppression pool serves as the heat sink for the decay heat generated by the reactor core. Thus it is important that operators understand the temperature-time response of the suppression pool when all of the reactor decay heat must be absorbed therein. If the suppression pool cooling does not function, the operator should appreciate the enormous capacity of the pool for energy storage. This has safety implications, because the operators might be reluctant to use the pool to best advantage under accident conditions if they do not understand its potential.

For example, in a Loss of DHR accident sequence the operators might be reluctant to depressurize the reactor vessel when required because of an unjustified assumption that this action might lead to a large increase in pool temperature and consequently in primary containment pressure. In fact, over the long term the only significant difference in the amount of energy stored in the pressure suppression pool with or without a reactor vessel depressurization is the difference in the sensible heat of the water mass in the reactor vessel associated with the saturation temperatures at the pressures before and after depressurization. Similarly, the existence of a stuck-open relief valve does not cause the pressure suppression pool to be heated much more rapidly; in the long term, the energy storage within the pool is limited to the decay heat generation within the core in either case.

It is recommended that methods to promote better operator understanding of the capabilities and response of the pressure suppression pool when pool cooling is not available be included in operator training.

It is also recommended that Emergency Operating Instruction (EOI) 41 be upgraded to indicate the amount of reactor vessel injection available under the various operating modes of the CRD hydraulic system and as a function of reactor vessel pressure. It is particularly important that operators recognize that the cooling water injection by the CRD hydraulic system is significantly increased when a scram is in effect. The fact that the raw cooling water system is needed for CRD pump cooling should also be taught. The portion of EOI 41 dealing with the injection capability of the standby liquid control (SLC) system should also be modified to indicate the correct injection rate under the consideration that only one SLC pump can be operated at a time.

The importance of running the RHR system in the suppression pool cooling mode even if cooling water to the heat exchangers is not available should be stressed. The action significantly reduces the thermal stratification in the pool.

Finally, it is recommended that the appropriate EOI stipulate that if very high containment pressures are encountered in a Loss of DHR accident sequence in which the core remains covered and there has been no release of fission products into the primary containment system, then the primary containment should be vented by whatever means necessary to preclude containment failure by overpressurization. In such a case, there is no need to risk the chance that violent disruption of the drywell might cause rupture of piping systems connected to the reactor vessel.

9.3 System Design

As described in Appendix A, pressure suppression pool cooling for Browns Ferry Unit 1 is provided by four RHR pumps and heat exchangers arranged in two separate loops. Each RHR pump is powered from a separate shutdown board; in the event of a loss of offsite power, each shutdown board is supplied from a different diesel generator. It is shown in Sect. 4.2 that adequate suppression pool cooling is provided if any one RHR pump and its associated heat exchanger are in operation in the suppression pool cooling mode.

Successful heat exchanger operation requires that RHR service water (RHRSW) be supplied to the tube side for cooling of the suppression pool water which is on the shell side of the heat exchangers. RHRSW is pumped from the Tennessee River to each of the RHR heat exchangers by pumps located on the intake structure at the river and dedicated for this purpose. Each unit 1 heat exchanger is served by a different RHRSW pump, with a backup pump available if required. The arrangement of diesel generator power supplies for use in the case of a loss of offsite power is shown in Table 9.2.

Should all of the RHR pumps and/or heat exchangers on Unit 1 be unavailable, a crosstie arrangement permits the use of the A or C RHR pumps and heat exchangers on Unit 2 to circulate and cool the Unit 1 pressure suppression pool water. The crosstie network is designed for a flow of 0.315 m^3/s (5000 gpm) which would permit operation of one Unit 2 heat exchanger at about 91 percent of its heat transfer capability at full flow [0.630 m^3/s (10,000 gpm)].[9.2]

The results of this study do not indicate that an improvement in the design of the RHR system or RHRSW system at Browns Ferry is required from the standpoint of readiness to cope with the initiation of a loss of DHR accident sequence involving the RHR pumps and heat exchangers of Unit 1. This conclusion is based on the results presented in Chap. 3 which show that about 21.5 h would be available before the containment pressure increased to its design value of 0.487 MPa (56 psig) and a total of 35 h would be available before the estimated drywell failure pressure of 0.910 MPa (117 psig)[9.3] is reached. This, combined with the redundancy of the systems involved, would seem to ensure a very high probability that at least one of the six* available RHR pumps and heat exchangers could be brought to bear on the Unit 1 pressure suppression pool as necessary to prevent containment failure.

A design consideration first identified in the SASA study of Station Blackout at Browns Ferry[9.4] also has direct application to the loss of DHR accident sequence. Provision is made 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. Once this shift occurs, the pump suction cannot be transferred back to the condensate storage tank. Because the HPCI turbine lubricating 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 normal pressure suppression pool level is between −2 and −6 in. indicated and the automatic HPCI pump suction shift will occur if the water level increases to +7 in. This would occur between 2 and 4 h after the inception of the accident, when the suppression pool temperature had increased to about 344 K (160°F). The pool temperature would continue to increase after the shift. Since the turbine oil cooler is designed for a maximum inlet water temperature of 333 K (140°F), the oil would become overheated leading to a possible failure of the turbine bearings.

An ample amount of relatively cool water would remain available in the condensate storage tank at the time the HPCI pump suction was automatically shifted to the pressure suppression pool. High suppression pool level at +7 in. requires the addition of between 257 and 375 m^3 (68,000 and 99,000 gal) whereas the normal condensate storage tank volume is about 1370 m^3 (362,000 gal). Thus water transferred from the condensate storage tank into the reactor vessel and from there to the suppression pool as steam via the vessel relief valves would produce a pool level of +7 in. long before the condensate storage tank was emptied.

The threat to the HPCI system identified here is not unique to Loss of DHR sequences; it would also exist in other accident sequences because high suppression pool temperature is caused by the pool heating attendant to the condensation of steam in the pool, which is also the source of the increased water level.

It should be noted that separate provision is made for an automatic shift of the HPCI pump suction if the normal source of condensate storage tank water becomes exhausted. Thus it appears that the automatic high

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*This includes the two available via the crosstie connection to Unit 2.

pool water level shift must have been straight-forwardly based on a concern for the effect of high water level in the pressure suppression pool. The basis is not given in the plant Technical Specifications and there is no corresponding high-level shift for the Reactor Core Isolation Cooling (RCIC) system, whose operation can also lead to high suppression pool level. A survey of plant suppression pool drawings does not reveal why a suppression pool level of +7 in. should be of concern.

It is recommended that the desirability of an automatic shift of the HPCI pump suction on high sensed suppression pool level without the opportunity for reversal by the operator be reconsidered.

It is a conclusion of the authors of this study that the Loss of DHR accident sequence is probably not a dominant contributor to the overall core melt probability at Browns Ferry when all available mitigation factors* are taken into consideration (see the discussion in the following Sect. 9.4). However, should subsequent PRA studies confirm that loss of DHR is a relatively high-risk threat that must be guarded against, then

(1) The control air system pressure should be increased to permit remote-manual operation of the SRVs at high containment pressures.

(2) The pressure suppression pool water level instrumentation should be temperature-compensated and the range of control room readout should be expanded, and

(3) the range of control room readout of drywell and wetwell pressure should be expanded.

9.4 Reconsideration of IREP Study Conclusions

A probabilistic risk assessment (PRA) of Browns Ferry Unit 1 was recently completed as part of the overall effort of the Nuclear Regulatory Commission Interim Reliability Evaluation Program (IREP).[9,1] One of the specific goals of the study was to identify the dominant contributors to core melt. The accident sequences so identified are listed in Table 9.3.

The discussion can be simplified by combining certain of the accident sequences. As indicated in Table 9.3, the fifth sequence is identical to the first and the eighth sequence is identical to the third except that the HPCI system is used for vessel injection instead of the RCIC system in sequences Nos. 5 and 8. Since it makes no difference to the progression of these Loss of DHR accident sequences whether the RCIC or HPCI systems are used for vessel level control during the periods when the reactor vessel is pressurized,† the eight dominant sequences identified by the IREP study can be regrouped into six sequences as listed in Table 9.4. This is done by combining sequences 1 and 5 and combining sequences 3 and 8. If all of the sequences listed in Table 9.4 that involve loss of DHR are combined,‡ we have an initial frequency of 1.8×10^{-3} and a final core melt

*Including the potential for effective operator action over a 30 h interval.

†There might be a difference if it were the HPCI system that was unavailable in either sequence, since it has a higher turbine exhaust pressure trip [1.138 MPa (150 psig)] than does the RCIC system [0.276 MPa (25 psig)].

‡These are sequences 1, 3, 4, and 6.

frequency of 1.4 x 10^{-4} events per reactor year for BWR 4 — Mark I containment plants identical to Browns Ferry Unit 1. This is a high predicted probability of core melt as a result of an inability to cool the pressure suppression pool or establish reactor vessel shutdown cooling. Indeed, the following excerpt from the executive summary to the IREP study illustrates the importance of these findings:

> "Six of the eight dominant sequences identified involve failure of the torus cooling and shutdown cooling modes of the RHR system. These sequences account for ~73% of the sum of the dominant sequence frequencies. Therefore, no significant reduction in core melt frequency can be achieved without reducing the unavailability of the RHR system or providing an alternate means of long-term decay heat removal. Thus, the RHR system is the most risk-critical system at BF1."

It is one purpose of this report to make recommendation as to whether these findings of the IREP study should be reconsidered based upon the results of this detailed SASA study of the Loss of DHR accident sequence. The SASA approach involves a much more detailed analysis than was possible in the particular PRA methodology employed in the IREP study. The findings of the SASA analysis that are believed to have a major bearing on an evaluation of the conclusions of the IREP analysis are discussed in the following sections:

9.4.1 Operation of the CRD system hydraulic pump

The operation of the CRD system hydraulic pump was neglected in the IREP study. This pump takes suction on the condensate storage tank and serves to inject 0.004 m^3/s (60 gpm) of control rod drive mechanism cooling flow into the reactor vessel under normal operating conditions. Following a scram, the vessel injection flow increases to over 0.006 m^3/s (100 gpm)* until the scram is reset, when injection flow is again reduced to 0.004 m^3/s (60 gpm).[9.5]

In the Loss of DHR accident sequence, a scram signal is continuously present after the drywell pressure reaches 0.115 MPa (2 psig). Thus vessel injection by the CRD hydraulic system would be available at an increased rate throughout most of the accident sequence.† After 4 h, this injection rate would be sufficient to maintain reactor vessel water level without the aid of any higher capacity system; in fact, the operator would have to throttle the flow or cycle the CRD hydraulic pumps on and off to prevent overfilling the vessel. The condensate storage tank can easily be refilled from a variety of sources, if necessary, but the volume of water normally maintained in this tank would be more than sufficient to maintain vessel level until the time of containment failure as shown in Fig. 3.3.

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*The flow depends on reactor vessel pressure, ranging from 113 gpm with the vessel fully pressurized to 182 gpm with the vessel fully depressurized.

†The drywell pressure reaches 0.115 MPa (2 psig) about 1 h after inception of the accident sequence. This is a scram trip setpoint.

Since the CRD hydraulic pump is capable of supplying all required reactor vessel injection after the first 4 h of a Loss of DHR accident sequence and because the pump takes suction on a water source that is not affected by the heatup of the pressure suppression pool, it seems that the effect of its existence must be considered in any analysis of a probability of core melt by way of a Loss of DHR accident sequence. Operation of the CRD hydraulic pump prevents core melt during the pool heatup phase of a Loss of DHR sequence because it is independent of the status of the pressure suppression pool.

The effect of the CRD hydraulic system injection should be factored into the PRA by assigning a probability that this injection would not be available during a Loss of DHR accident sequence. For a transient-induced sequence or a transient-induced sequence with SORV (sequences 1 and 4 of Table 9.4), this probability would be very low since no operator action is required and there is no common mode failure mechanism for the CRD hydraulic pump. Thus consideration of the CRD pump operation should significantly affect the calculated frequency of these sequences.

In the case of the Loss of DHR sequences whose initiating events include a loss of offsite power (sequences 3 and 6 of Table 9.4), the reduction in core melt frequency due to consideration of CRD hydraulic pump operation would not be as significant because only the spare CRD hydraulic pump 1B is supplied from an electrical bus (shutdown board A) powered by a diesel generator. Thus operator action would be necessary to restore CRD hydraulic pump operation following a LOSP including the starting of a raw cooling water pump for CRD pump motor cooling. Nevertheless, the very long period of time available for such action should be taken into consideration.*

9.4.2 Effect of containment backpressure

The IREP study assumes loss of injection capability to the reactor vessel as soon as the pressure suppression pool bulk temperature reaches 355 K (180°F) because of loss of net positive suction head (NPSH) to the ECCS pumps which take suction on the suppression pool. This assumption leads to early core uncovery (about 5 h) with the containment intact and is unrealistic for the following reasons:

1. Containment backpressure acts to maintain the NPSH above the minimum recommended for operation by the RHR pump manufacturer for over 9 h if the drywell coolers remain operating and for over 22 h if the drywell coolers are not restarted by the operator after tripping on a combination of low reactor vessel pressure and high drywell pressure at about 2 h after inception of the accident (see Fig. 3.12). As discussed in Sect. 3.3.5, in-plant testing at Browns Ferry has shown that the RHR pumps are capable of operation with a NPSH significantly lower than

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*For example, the RCIC system would remain available for injection from the condensate storage tank until the turbine trip setpoint of 0.276 MPa (25 psig) containment pressure is reached about 13 h into the accident.

the manufacturer's recommended minimum. A very low pumped flow [less than 0.009 m^3/s (150 gpm)] is sufficient to keep the core covered after 4 h (see Fig. 3.2).

2. The RHR pumps and the core spray pumps can be manually shifted by the operator at any time* to take suction on the condensate storage tank instead of the pressure suppression pool. This action would maintain operation of these systems even if sufficient NPSH did not exist for suction on the suppression pool. The RCIC system normally takes suction on the condensate storage tank and would remain so aligned unless shifted by the operator.†

For these reasons, it is not reasonable to assume that all injection to the reactor vessel is lost when the suppression pool temperature reaches 355 K (180°F), even if the contribution of the CRD hydraulic pump (Sect. 9.4.1) continues to be neglected. In any event, the probability that there is no containment backpressure and the probability that the operator would not shift the suction of the RHR or core spray pumps to the condensate storage tank should be included in the analysis. This inclusion would reduce the frequencies of all of the Loss of DHR accident sequences (Nos. 1, 3, 4, and 6) listed in Table 9.4.

9.4.3 Content of condensate storage tank

The IREP study assumed a volume of only 511 m^3 (135,000 gal) in the condensate storage tank at the inception of the accident; this corresponds to the volume guaranteed to the ECCS systems and the CRD hydraulic pumps by the existence of a standpipe within the tank which feeds all other systems.‡ The assumption used in the IREP study is unrealistic because the Browns Ferry operating instructions for the condensate system require the operator to keep the condensate storage tank nearly full. The median volume for the allowable operating band is 1370 m^3 (362,000 gal). As shown in Fig. 3.3, this amount of water is more than sufficient to maintain vessel level up to the time of containment failure by overpressurization. Even if the initial volume were significantly lower, the condensate storage tank can be easily refilled from several sources, including a firetruck from a nearby town if necessary during the very long Loss of DHR accident sequence. The matter is important because at Browns Ferry all ECCS systems can take suction on a condensate storage tank§ and therefore successful injection does not depend on the status of the pressure suppression pool as long as water remains in the condensate storage tank. Thus the probability that the condensate storage tank is at only 500 m^3

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*There are no interlocks on the pump suction valves.

†The HPCI pump is also normally aligned to the condensate storage tank. However, the HPCI booster pump suction is automatically and irreversibly shifted to the suppression pool when the sensed suppression pool level reaches +7 in.

‡Primarily makeup to the main condenser hotwell.

§It should be noted that the condensate storage tanks can be cross-connected.

(135,000 gal) at the inception of the accident and the probability that the tank is not refilled if necessary during the accident should be factored into the analysis.

9.4.4 RHR system minimum flow bypass valves

The RHR system is equipped with minimum flow valves which open to permit a limited pumped flow to the pressure suppression pool when the main pump discharge paths are closed. This is for protection of the pumps, and the minimum flow bypass valves are automatically interlocked to close when a flow is established in the main discharge paths. The IREP study assumed a complete failure of the RHR system function if the minimum flow bypass valves failed to close. This is unrealistic because 90% of the flow to the main RHR discharge path can be maintained with the minimum flow bypass valves open. Recognition that the RHR system can perform its function during a loss of DHR accident even with the minimum flow bypass valves failed open would reduce the frequencies of the loss of DHR sequences listed in Table 9.4 by a factor of 22.*

9.4.5 Use of the SBCS system to control primary containment pressure

The IREP study did not consider use of the standby coolant supply system (SBCS)† for control of primary containment pressure. If all else failed, this system could be used to inject river water into the drywell or wetwell spray headers as a means to reduce the pressure in the primary containment and thereby avoid containment failure. Since the shutoff head of the RHRSW pumps is about 1.22 MPa (162 psig) and the elevation differential between the river and the suppression pool is insignificant, this method of containment spray could be used at any time during the loss of DHR accident sequence.

9.4.6 Requirements for the EECW system

The emergency equipment cooling water (EECW) system provides cooling water under emergency conditions to the diesel generators and other safety systems. For the IREP study, it is assumed that three of the four EECW pumps are necessary for the performance of function unless the operator takes action to manually eliminate the less-essential loads. This assumption has a significant effect on the initial frequencies for the sequences listed in Table 9.4 that involve a loss of offsite power. In fact, the elimination of less-essential loads is automatic,‡ reducing the requirement for EECW performance to two out of four pumps. Since the IREP study

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*This according to the sensitivity study given in the IREP report.

†The SBCS is described in Sect. A.5.2 of Appendix A.

‡The service air compressor and RBCCW heat exchanger loads are automatically eliminated when water pressure in the EECW headers falls below a preset value.

did grant credit for manual action for load reduction, correction of this assumption should not significantly affect the final frequencies listed in Table 9.4.

9.4.7 Other considerations

Other difficulties with the conclusion of the IREP study that loss of DHR sequences are dominant at Browns Ferry include the study procedure that no credit is ever taken for recovery of the power conversion system (PCS) as a heat sink and that the potential for suppression pool cooling via the provided cross-connection to other units is ignored. These considerations have direct application to the loss of DHR accident sequences and should not be neglected, especially after these sequences have been tentatively identified as dominant.

The IREP study also did not consider use of the standby liquid control (SLC) system as an alternate method for high pressure water makeup to the reactor vessel. The primary purpose of this system is injection of the neutron-absorbing sodium pentoborate in the unlikely event that this became necessary due to failure of control rod insertion. However, it is possible to shift the suction of the injection pumps from the SLC tank, which contains a solution of sodium pentaborate, to a storage tank containing approximately 1,419 m^3 (375,000 gal.) of demineralized water.

There are two positive-displacement SLC pumps at each Browns Ferry unit, each capable of injecting approximately 0.004 m^3/s (56 gpm) of water into the reactor vessel. However, the pump control circuitry is provided with interlocks to ensure that only one pump is operated at a time. The procedure for using one of the SLC pumps to inject demineralized water into the reactor vessel for high pressure makeup under emergency conditions is contained in Browns Ferry Emergency Operating Instruction (EOI) No. 41.*

The IREP study omission of consideration of the use of the SLC system as an alternate method for reactor vessel injection is probably not significant. First, the system is not needed: adequate vessel injection to replace the water boiled to steam in the reactor vessel and transferred to the suppression pool through an SRV can be provided by occasional augmentation of the CRD hydraulic system flow by the RCIC system during the first 4 h after a scram, even if the operator acts to depressurize the reactor vessel during this period. Successful operation of these systems does not depend on the status of the pressure suppression pool, and operation of the CRD hydraulic system alone is sufficient to keep the core covered after 4 h following reactor scram.†

Secondly, the SLC injection capacity of 0.004 m^3/s (56 gpm) is simply not enough to maintain a constant reactor vessel water level even at the time predicted for containment failure 35 h after scram and inception of the loss of DHR accident. It might be postulated that credit should be

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*It should be noted that this EOI erroneously claims that 0.008 m^3/s (112 gpm) can be injected by this method. As noted above, only one 0.004 m^3/s (56 gpm) pump can be operated at a time.

†Assuming the reactor vessel has been depressurized, as per procedure.

taken in the IREP study for use of the SLC injection capability in conjunction with another system whose normal capacity is degraded. However, the needed injection rate is a very small fraction of the normal injection rate of all other systems.* If these systems operate at all, they should provide the necessary injection [less than 0.011 m^3/s (170 gpm) after 4 h].

9.4.8 Recommendations

There is concern that the IREP study assumptions concerning the factors discussed in Sects. 9.4.1 through 9.4.6 might have caused the loss of DHR accident sequences to unrealistically appear to constitute the majority of the dominant sequences for core melt at Browns Ferry Unit 1. Accordingly, it is recommended that the conclusions of the IREP study with regard to the Loss of DHR accident sequences be reconsidered, with the information discussed in this section and summarized below:

1. The CRD hydraulic pump is capable of supplying all needed vessel injection after the first 4 h of a Loss of DHR accident sequence. For a transient-induced accident sequence without loss of offsite power (LOSP), this requires no operator action. If LOSP is involved, then the operator must start the standby CRD pump, which is powered from a diesel-generator bus.
2. The Browns Ferry operating instructions for the condensate system require that the condensate storage tank for each unit be maintained nearly full, i.e., between 1419 and 1325 m^3 (375,000 and 350,000 gal). As shown on Fig. 3.3, this quantity of water is more than sufficient to maintain reactor vessel level up to the time of calculated containment failure by overpressurization (about 35 h). Furthermore, in the unlikely event that the condensate storage tank were at a significantly lower level at the inception of the accident, this tank can be easily refilled from several sources.
3. With the effect of containment backpressure considered in the calculations, there would be sufficient NPSH to permit RHR pump operation with suction on the pressure suppression pool at any time during the accident sequence. It might be necessary to operate at reduced flow during the latter part of the sequence, but the reduced flow would be sufficient to satisfy the requirements for reactor vessel injection or suppression pool cooling (if the RHR heat exchangers were restored to service).
4. The RHR pumps and the core spray pumps can be manually shifted by the operator at any time to take suction on the condensate storage tank instead of the pressure suppression pool.
5. The RHR pumps can perform their reactor vessel injection or pressure suppression pool cooling function even with their minimum flow bypass valves failed open.

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*In English units, the normal capacities are: HPCI (5,000 gpm), RCIC (600 gpm), each of 4 RHR-LPCI modes (10,000 gpm), each of 2 core spray modes (3,125 gpm). All of these systems can take suction on the condensate storage tank.

6. The cooling loads supplied by the EECW system are automatically reduced if necessary so that the essential cooling loads (including the diesel-generators) can be carried by any two of the four available EECW pumps.

7. The SBCS could be used to inject river water into the drywell or wetwell spray headers at any time during the accident sequence.

8. With over 30 h available for remedial action, there is a significant probability that the PCS could be recovered before containment failure.

9. Unit 1 suppression pool cooling can be accomplished by certain Unit 2 RHR pumps and heat exchangers, via cross-connecting piping.

References for Chapter 9

9.1 S. E. Mays et al., "Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant," NUREG/CR-2802, EGG-2199, July 1982.

9.2 Browns Ferry Nuclear Plant, Final Safety Analysis Report, Revision 67, Section 4.8.6.4, Tennessee Valley Authority.

9.3 L. G. Griemann et al., "Reliability Analysis of Steel Containment Strength," NUREG/CR-2442, June 1982.

9.4 D. H. Cook et al., "Station Blackout at Browns Ferry Unit One — Accident Sequence Analysis," NUREG/CR-2182, ORNL/TM-455/V2 (November 1981).

9.5 S. A. Hodge et al., "SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis," NUREG/CR-2672, ORNL/TM-8119/V1, November 1982, Sect. E.3.

Table 9.1. Control room indication and alarms of primary containment variables important to analysis and control of a Loss of DHR accident sequence

Variable	Range or setpoint
Drywell pressure	
Indication, MPa (psia)	0–0.55 (0–80)
Alarms, MPa (psig)	0.112 (1.6)
	0.113 (1.65)
	0.114 (1.75)
	0.115 (2.00)
Drywell atmosphere temperature	
Indication, K (°F)	0–477 (0–400)
Alarms, K (°F)	336 (145)
Wetwell pressure	
Indication, MPa (psia)	0–55 (0–80)
Alarms, MPa (psig)	0.115 (2)
Pressure suppression pool temperature	
Indication, K (°F)	0–477 (0–400)
Alarm, K (°F)	308 (95)
Pressure suppression pool level[a]	
Indication, m (in.)	−0.64 − +0.64 (−25 − +25)
Alarm, m (in.)	less than −0.15 (−6)
	more than +0.15 (+6)

[a]Instrument zero is 4.6 m (15.2 ft) above the bottom of the wetwell torus. A water level of zero indicates that the wetwell is ~1/2 filled with water.

Table 9.2. Arrangement of emergency diesel generator power supplies to the pumps associated with each unit 1 heat exchanger

Unit 1 heat exchanger	A	B	C	D
Diesel assigned to RHR pump	A	C	B	D
Diesel assigned to RHRSW pump				
Primary RHRSW pump	A	C	B	D
Backup RHRSW pump	A	3C	B	3D

Note: Diesels denoted A, B, C, and D are shared between Units 1 and 2. Diesels denoted 3A, 3B, 3C, and 3D are provided for Unit 3. Diesels A and 3A, B and 3B, etc., can be run in parallel.

Table 9.3. Browns Ferry Unit 1 dominant accident sequences as identified by the IREP study

Relative order of final frequency	IREP nomenclature and description	Frequency[a]	
		Initial	Final
1	$T_U R_B R_A$: Transient, PCS unavailable, loss of DHR	1.3×10^{-4}	9.7×10^{-5}
2	$T_U B$: ATWS, PCS unavailable	5.1×10^{-5}	5.1×10^{-5}
3	$T_P R_B R_A$: LOSP, loss of DHR	1.5×10^{-3}	2.8×10^{-5}
4	$TKR_B R_A$: Transient, SORV, loss of DHR	1.2×10^{-5}	9.3×10^{-6}
5	$T_U QR_B R_A$: Same as No. 1 except RCIC unavailable so HPCI used instead	5.5×10^{-6}	4.1×10^{-6}
6	$T_A BM$: ATWS, PCS available, no recirculation pump trip	3.7×10^{-6}	3.7×10^{-6}
7	$T_P KR_B R_A$: LOSP, SORV, loss of DHR	8.3×10^{-5}	1.6×10^{-6}
8	$T_P QR_B R_A$: Same as No. 3 except RCIC unavailable so HPCI used instead	6.2×10^{-5}	1.2×10^{-6}

[a]The initial frequency pertains to the probability that the sequence will be initiated. The final frequency takes into account the potential for recovery before the sequence proceeds to core melt. Units are events per reactor-year.

Table 9.4. Dominant accident sequences at Browns Ferry considering HPCI and RCIC system use to be equivalent

Relative order of final frequency	IREP nomenclature and description	Frequency[a]	
		Initial	Final
1	$T_U R_B R_A / T_U QR_B R_A$: Transient, PCS unavailable, loss of DHR	1.4×10^{-4}	1.0×10^{-4}
2	$T_U B$: ATWS, PCS unavailable	5.1×10^{-5}	5.1×10^{-5}
3	$T_P R_B R_A / T_P QR_B R_A$: LOSP, loss of DHR	1.6×10^{-3}	2.9×10^{-5}
4	$T_K R_B R_A$: Transient, SORV, loss of DHR	1.2×10^{-5}	9.3×10^{-6}
5	$T_A BM$: ATWS, PCS available, no recirculation pump trip	3.7×10^{-6}	3.7×10^{-6}
6	$T_P KR_B R_A$: LOSP, SORV, loss of DHR	8.3×10^{-5}	1.6×10^{-6}

[a]The initial frequency pertains to the probability that the sequence will be initiated. The final frequency takes into account the potential for recovery before the sequence proceeds to core melt. Units are events per reactor-year.

10. CONCLUSIONS

The major conclusions of this study are itemized in Sect. 10.1 of this chapter. A reference follows each of the conclusions, indicating the location within this report where the subject is discussed in detail. A brief discussion of the uncertainties pertinent to the analyses is provided in Sect. 10.2.

10.1 Itemized Conclusions

1. If the pressure suppression pool water is circulated and mixed by the operation of at least one RHR pump during the Loss of DHR accident sequence, then the assumption of uniform suppression pool heatup is justified for calculational purposes (Sects. 2.1 and 3.1).

2. The normal supply of water in the condensate storage tank is sufficient to maintain a normal water level in the reactor vessel throughout the accident sequence during the period before primary containment failure by overpressurization (Sects. 3.2 and 3.3, Fig. 3.3).

3. Without cooling, the suppression pool temperature would increase to 49°C (120°F) after 1 h, requiring the operator to begin a controlled manual depressurization of the reactor vessel (Sect. 3.3.3 and Fig. 3.6).

4. Normal reactor vessel water level can be maintained by continuous operation of one CRD hydraulic pump augmented by periodic operation of the RCIC pump during the first 4 h of the accident sequence. After 4 h, the reactor vessel is depressurized and the CRD pump operation alone is sufficient to maintain vessel level (Sect. 3.3.1 and Fig. 3.1).

5. No operator action is required to establish and maintain the CRD hydraulic system injection rates assumed in this study unless the Loss of DHR initiating event includes a LOSP. If a LOSP occurs, then operator action is required to restore and maintain continuous CRD pump operation (Sect. 9.4.1).

6. The continued availability of the HPCI and RCIC systems would be threatened during the accident sequence by the following automatic control actions:

Time (h)	Event
~3	HPCI pump suction shifted to overheated pressure suppression pool (Sect. 9.3).
~13	HPCI and RCIC turbine steam supply lines isolated because of high torus room temperature (Sect. 3.3.6).
~14	Containment pressure exceeds RCIC turbine exhaust high pressure trip setpoint (Table 3.1).

7. Remote-manual SRV operability would be lost after about 24 h and the reactor vessel would begin a slow process of repressurization. The reactor vessel would be fully pressurized at the time of containment failure (Sect. 3.3.2, Fig. 3.4).

8. Containment failure pressure would be reached about 35 h after the inception of the accident sequence assuming that the drywell coolers operate for only the first 2 h (Sect. 3.3.4).

9. Sufficient NPSH can be maintained to permit RHR pump operation with suction on the pressure suppression pool throughout the accident sequence (Sect. 3.3.5).

10. If the operator takes action to restore drywell cooler operation after the 2 h point, this can have a significant effect on the accident sequence timing. The containment failure would be delayed about 2.5 h and the RHR pump flow would have to be throttled to permit continued operation after about the 14 h point (Sects. 3.3.5 and 3.3.6).

11. Operation of any one RHR heat exchanger in the pressure suppression pool cooling mode would prevent the pool water from reaching excessive temperature and would therefore preclude containment failure (Sect. 4.2 and Fig. 4.1).

12. The 5.1 cm (2 in.) vent lines from the drywell and wetwell are not large enough to prevent primary containment pressure from increasing to the failure point during an extended Loss of DHR accident sequence (Sect. 4.3).

13. For a Loss of DHR accident sequence in which there is no forced circulation of the pressure suppression pool, primary containment failure by overpressurization is estimated to occur at the 28 h point. This is seven hours earlier than for the case where assumption of a uniform pool temperature is justified (Chap. 5).

14. The occurence of a SORV would not have a major effect on overall system behavior during a Loss of DHR accident sequence. If mixing and uniform suppression pool heatup is assumed, the primary containment failure is advanced just 1 h, to the 34 h point. If there is no forced suppression pool circulation then the existence of the SORV delays the containment failure by about 4 h, to the 32 h point (Chap. 6).

15. It is very unlikely that the blowdown forces associated with primary containment failure would result in degradation of the concrete shield wall surrounding the drywell (Sect. 7.5).

16. Loss of reactor vessel injection as a result of over-pressurization failure of the containment in a Loss of DHR accident sequence might be caused either by (A) loss of all of the vessel injection systems located within the reactor building as a result of the harsh environmental conditions there after containment failure coupled with an inability to depressurize the reactor vessel to permit use of the low-pressure pumps in the turbine building or by (B) loss of the vessel injection systems within the reactor building and the occurrence of piping breaks sufficient to render all remaining vessel injection systems ineffective. Case (B) has not been considered in previous PRAs (Sects. 7.7 and 7.8).

17. There are several possible consequences of a failure of the containment by overpressurization. There might or might not be a LOCA caused by the disruptive blowdown of the drywell. Sufficient reactor vessel injection capability to maintain the core covered might or might not remain. Thus it is unrealistic to assign a 100% probability to the scenario in which all vessel injection is lost but no piping breaks occur. The actual probability would be the sum of (A) the product of the probability of the loss of all reactor building injection systems because of harsh environmental conditions and the probability of failure to regain remote-manual

control of even one SRV after containment failure and (B) the product of the probability of failure of some injection systems because of harsh environmental conditions and the probability of piping breaks sufficient to render the remaining injection systems ineffective (Sects. 7.7 and 7.8).

18. Lack of consideration of Case (B) of conclusion 16 above can lead to nonconservatism in PRA analyses since the piping breaks could permit early release of the volatile fission products into an already-failed drywell (i.e., the fission product scrubbing function of the pressure suppression pool would be bypassed) (Sects. 7.7 and 7.8).

19. If all reactor vessel injection is assumed lost at the time the primary containment fails and it is assumed that there is no LOCA, then core uncovery will occur after boiloff of the volume of water above the core at the time of drywell failure. MARCH runs indicate that the core would uncover about 2 1/2 h after loss of injection and that core melting would begin about 1 h later (Chap. 8).

20. During the latter stages of a Loss of DHR accident sequence, the drywell and wetwell pressure indication and the pressure suppression pool level indication would be off-scale high. The drywell and pressure suppression pool temperature indication would remain onscale and the primary containment pressure could be inferred from the suppression pool temperature and the steam tables (Sect. 9.1).

21. The automatic shift of the HPCI pump suction from the condensate storage tank to the pressure suppression pool on high sensed suppression pool level should be removed or modified to permit the operator to return the suction to the condensate storage tank if necessary (Sect. 9.3).

22. The conclusions of the IREP study regarding the probability of core melt at Browns Ferry unit 1 as a consequence of Loss of DHR should be reconsidered, based upon the better accident sequence definition provided by the detailed analysis presented in this report (Sect. 9.4).

10.2 Uncertainties in the Analysis

The calculation of accident sequence events before containment failure was performed using the ORNL-developed BWR-LACP code which incorporates reactor vessel, primary containment, and secondary containment models specific to Browns Ferry Unit 1. The BWR-LACP code was also used in two previous SASA studies; additions made to the code for the Loss of DHR accident sequence calculations are described in Appendix B of this report. Code results for a Station Blackout accident sequence have been compared to results calculated for the same sequence by the Browns Ferry simulator and RELAP4 Mod 7.[10.1] Code results for a small-break LOCA with condensate booster pump injection have been compared with results calculated for the same sequence by RELAP5 Mod 1.[10.2] Agreement was good in all cases.

It should be noted 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. The MSIVs are shut during a severe accident sequence and the reactor vessel is isolated. In general, the recirculation pumps are tripped and core flow is solely due to natural convection circuits within the reactor vessel itself. Therefore, for other than large-break LOCA or ATWS studies, sophisticated primary system analysis codes such as RELAP, RETRAN, or TRAC are not necessary; fundamental modeling of the processes within the reactor vessel in a relatively simple code such as BWR-LACP is sufficient.

On the other hand, the interaction between the reactor vessel and its relatively small primary containment is very important to the determination of the sequence of events for a BWR severe accident. In this regard, BWR-LACP is efficient because it combines primary system, primary containment, and secondary containment analyses in one code. There is no need to convert the output of one code into the input for another with the attendant opportunity for error. BWR-LACP is specific to the Browns Ferry MK I containment system and is therefore a straightforward application of basic thermohydraulic and heat transfer theory. The uncertainty in the results presented in Chapters 3 through 6 caused by modeling inaccuracies is believed to be negligible.

Uncertainties do exist in the input parameters supplied to the BWR-LACP code for the study of the Loss of DHR accident sequence before core uncovery. These include:

1. The primary system events during the very brief period (~1 min) after scram and MSIV closure when multiple SRVs are open and the feedwater turbines are coasting down can not be modeled by BWR-LACP. Normal reactor vessel indicated water level is 561 in. above vessel zero and the BWR-LACP calculations are begun at time 30 s with a water level equivalent to an indicated 500 in. in consideration of the effect of level shrink upon MSIV closure. This assumption is based upon the accident studies presented in Chap. 14 of the Browns Ferry FSAR and upon the level indications at the Browns Ferry simulator when scram and MSIV closure are simulated. Since reactor vessel water level is subsequently controlled, the uncertainty in the brief period just after accident initiation is not believed to be important.

2. It has been 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 as a driving force to the RCIC or HPCI turbines.* In fact, there would be a slight leakage (less than 25 gpm) into the drywell, and a slight leakage through the MSIVs into the main condensers. The amount of leakage is uncertain, and has been neglected in the analysis.

3. As the pressure suppression pool is heated, evaporation from the water surface tends to increase the volume of steam in the primary containment atmosphere and consequently the pressure. Leakage from the primary containment has been modeled as equivalent to that measured during the most recent containment integrated leak rate tests [conducted at 0.274 MPa (25 psig)], adjusted for different containment pressures.†

It is entirely possible that the leakage paths from the primary containment would both enlarge and become more numerous as the internal pressure increased above design pressure and approached the failure level. For this study, it has been assumed that no new or enlarged leakage paths develop. This approach is conservative and produces the earliest catastrophic failure of the drywell. If enough additional drywell leakage paths did develop to permit the escape of sufficient steam to maintain the primary containment pressure constant at some level below failure pressure, then reactor vessel injection capability would not be threatened and a Severe Accident with core uncovery would not occur.

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*Which also exhaust to the pressure suppression pool.

†See Sect. B.2 of Ref. 10.3.

4. As discussed in Chap. 3, operation of the drywell coolers has a significant effect upon the results of this study. It is uncertain what action the operators might take to restore the drywell coolers once they have been automatically tripped as a consequence of a LOSP and a core spray initiation signal. Both the case of no restoration and the case of immediate restoration of drywell cooling have been included in the analyses.

5. The rate of CRD hydraulic system injection into the reactor vessel is not known with certainty. The system employs centrifugal pumps and with a scram in effect, the injection is primarily leakage past the graphitar seals in the CRD mechanism assemblies. The Browns Ferry FSAR estimates the injection to be 182 gpm with the reactor vessel depressurized and 113 gpm with the reactor vessel pressurized for one pump in operation with the normal system lineup for the case of a scram in effect. A maximum injection of 170 gpm was used for the analysis.

As shown on Figs. 3.2 and 3.4, the reactor vessel is depressurized during most of the early part of the accident sequence. During the first four hours before the vessel is depressurized the operator is able to control vessel level by running the RCIC system in conjunction with the CRD hydraulic system and if the actual available CRD hydraulic system injection varied from that assumed in the analysis then the operator could adjust for the difference by running the RCIC pump for longer (or shorter) periods. In the latter part of the accident sequence the vessel is repressurized, but by this time the decay heat is low enough so that the operator throttles the flow to less than 113 gpm. In summary, since the rate of CRD hydraulic system injection is an operator-controlled variable, uncertainty in the maximum available flow at various times during the sequence does not significantly affect the analysis.

6. The containment failure pressure and the failure location are uncertain. A value of 0.910 MPa (117 psig) for failure at the cylinder-sphere intersection in the drywell was assumed for this study, based on the information presented in Ref. 10.4.

7. The calculations of Chapts. 3, 5, and 6 assumed that pressure fluctuations in the pressure suppression pool during SRV discharge would be mild, and that (except as noted in Sect. 5.3.3) 100% of the discharge would be condensed providing that the pool remained as much as 1.11°C (2°F) subcooled. The assumption of mild pressure fluctuations during SRV discharge is realistic because the T-quencher was specifically designed and extensively tested to eliminate the violent condensation oscillations possible with the formerly employed rams-head (the T-quencher has many very small perforations over a long, 30-cm diam, submerged pipe, whereas the rams-head has only two 25-cm diam discharge stubs).

The uncertainty to be examined here concerns the amount of subcooling required for complete condensation of SRV discharge. The range of experimentally proven quencher performance as a function of pool local subcooling (difference between saturation temperature defined at quencher depth and pool temperature) is summarized on Fig. 9 of NUREG-0783. These data can be used to define the minimum subcooling required for 100% condensation: 11.1°C (20°F) for mass fluxes below 206 kg/s/m^2 (42 lb/s/ft^2), corresponding to SRV discharge when reactor vessel pressure is low, and 22.2°C (40°F) for mass fluxes exceeding 460 kg/s/m^2 (94 lb/s/ft^2), corresponding to SRV discharge when reactor vessel pressure is high.

In the loss of DHR sequences, a subcooling requirement can be viewed as a requirement that saturation temperature at total suppression pool pressure must exceed the pool water temperature by the specified margin. If primary containment pressure is too low there will be incomplete condensation and pressure will increase until subcooling approaches that required for complete condensation. At the primary containment failure pressure of 0.908 MPa (117 psig), the saturation temperature is 176°C (348°F). The suppression pool temperature at containment failure would be approximately equal to 176°C (348°F) less the assumed pool subcooling requirement. As summarized on Table 10.1, the effect on containment failure time of the stricter pool subcooling requirements specified in the previous paragraph would be to shorten the reported times to 28 h.

8. The containment failure mechanism assumed in this report is failure by overpressure at 0.908 MPa (117 psig). Since the drywell atmosphere temperature exceeds the 138.3°C (281°F) design temperature of the drywell before the time of overpressure failure, it is appropriate to consider in this section the possibility of containment failure due to excessive temperature. Drywell components, including electrical penetrations, must be able to withstand temperatures in excess of design temperature for limited periods. If a failure temperature of 162.8°C (325°F) is assumed for the drywell electrical penetrations, then the time to containment failure would be 26 h — about 2 h shorter than the shortest overpressure failure time considered in this report.

9. There is uncertainty regarding the sequence of events during the first few hours of the accident. The results presented in Chap. 3 were obtained with the assumption that the operator would not open the 5.1 cm (2 in.) vents from the drywell and wetwell and that the CRD hydraulic system injection would increase when a high drywell pressure scram occurred at 0.115 MPa (2 psig). The wetwell-to-drywell differential pressure compressor, which maintains the drywell pressure about 0.008 MPa (1.1 psi) higher than the wetwell pressure,* was modeled for automatic actuation.

In their review of this report, the Tennessee Valley Authority (TVA) reviewers pointed out that the setpoint for the high drywell pressure scram had been increased to 0.119 MPa (2.5 psig) and that the differential pressure compressor is operated in the manual mode so that operator action is required. They also suggested that the operator would open the 5.1 cm (2 in.) vents when the primary containment pressure reached 0.115 MPa (2 psig) in an effort to forestall the high drywell pressure scram.

To determine the effect of the uncertainties revealed by the TVA comments, an additional BWR-LACP calculation was performed in which it was assumed that:

(a) The differential pressure compressor was not operated so that the flow from the wetwell airspace to the drywell was only through the vacuum breakers, which are open only when the wetwell pressure exceeds the drywell pressure by at least 0.003 MPa (0.5 psi).

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*The purpose is to keep the downcomers in the pressure suppression pool almost totally free of water. The compressor and associated piping are shown on Fig. 4.2.

(b) The 5.1 cm (2 in.) vents were opened when the primary containment pressure reached 0.115 MPa (2 psig).
(c) The high drywell pressure scram setpoint was 0.119 MPa (2.5 psig).

The calculation indicates that the high drywell pressure scram would occur at time 3.8 h under the new assumptions as opposed to time 1 h as reported in Chap. 3. Although this delays the increase in CRD hydraulic pump injection, the RCIC system remains available for injection throughout this period so there is no effect on the ability to control reactor vessel level. The opening of the primary containment vents delays the time of primary containment failure from 35 h as reported in Chap. 3 to 40.5 h.

10. The Reactor Water Clean-up (RWCU) system operates continuously during normal reactor operation, removing impurities from the primary coolant, and also removing a small quantity of heat (about 4.5 MW if the reactor is at full pressure and temperature). Although it is possible that the RWCU system could continue to operate during the Loss of DHR sequence, the analyses of Chapters 3, 5, and 6 took no credit for heat removal by the RWCU system. When the reactor coolant system is depressurized and therefore at a lower temperature, the rate of heat removal by the RWCU system is only about 2.5 MW. However, this is 10% of the rate of heat generation by decay heat 12 h after reactor scram. It is estimated that if the RWCU system were operated continuously throughout the 35 h Loss of DHR sequence, the ultimate containment failure would be delayed by about 10% (i.e. 4 h).

The primary reason for assuming no heat removal by the RWCU system is that one can envision circumstances in which it would not be operated in a Loss of DHR accident sequence. In order for the RWCU system to remove heat from the primary coolant system, both the Reactor Building Closed Cooling Water (RBCCW) system and either the Raw Service Water (RSW) system or the Emergency Equipment Cooling Water system must be operational. The RBCCW system supplies cooling water to the RWCU non-regenerative heat exchanger via its non-essential loop, which is automatically isolated in the event of a loss of offsite power. After a successful start of the station diesels, the RBCCW non-essential loop isolation valve can be opened by the operators; however, the plant operators might decide to allow the valve to remain closed in compliance with the non-essential category to which this cooling load has been designated.

The MARCH code was used for calculation of the results presented in Chap. 8 of this report. The general limitations of this code with respect to LWR accident analysis with emphasis on PWR applications have been discussed elsewhere.[10.5] However, the specific limitations with regard to BWR analysis are even more confining. Many of the difficulties with respect to the use of MARCH for BWR accident analysis are discussed in Appendix B of Ref. 10.6. Some of these difficulties have been bypassed in the present study by initiating the MARCH analysis at a time just preceeding containment failure. However, other MARCH code limitations do have an effect on the current study. These include:

1. The core model does not represent the Zircaloy channel boxes and the control rods present in a BWR core.

2. Reactor vessel pressure control by the SRVs is not correctly represented in that a continuous release of steam from the reactor vessel is

modeled instead of the actual periodic blowdowns followed by relatively quiescent periods within the reactor vessel between relief valve lifts. Both of these limitations have a significant effect on the calculation of the metal-water reaction rate and consequently, on the core heatup after partial uncovery.

Since the BWR-LACP code has been used to establish the approximate time of containment failure for this study, the chief function of the MARCH code has been to provide a basis for examination of the possible effects of containment failure on reactor vessel injection capabilities and to establish the timing of the events after containment failure for the case in which all reactor vessel injection is assumed lost. As discussed in Chap. 8, the MARCH code predicts core uncovery 2-1/2 h after containment failure and loss of injection followed 1 h later by core melting. With the code's present limitations, these can be considered to be no more than reasonable approximations to the actual timing.

A fission product transport analysis is underway as a follow-on to the accident sequence analysis presented in this report. Ongoing modifications to the MARCH code are intended to remove the more significant limitations of this code in time for an improved version to be available for support of the fission product transport analysis.

References for Chapter 10

10.1 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.

10.2 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 NUREG/CR-2672, ORNL/TM-8119/V1).

10.3 S. A. Hodge et al., "Station Blackout at Browns Ferry Unit One — Iodine and Noble Gas Distribution and Release," NUREG/CR-2182 Vol. 2, ORNL/NUREG/TM-455/V2, August 1982.

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

10.5 J. B. Rivard et al., "Interim Technical Assessment of the MARCH Code," NUREG/CR-2285, SAND81-1672.R3, November 1981.

10.6 S. R. Greene et al., "SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis," NUREG/CR-2672, Vol. 1, ORNL/TM-8119/V1, November 1982.

Table 10.1. Effect of stricter suppression pool subcooling requirement on time to containment failure[a]

Sequence(s) discussed in section	Nominal assumed subcooling requirement °C (°F)	Reported containment failure time (h)	Containment failure time for stricter subcooling requirement (h)
3	1.11 (2)	35	28
5	1.11 (2) 22.2 (40)	28	28
6	1.11 (2)	32	28

[a]The stricter requirements are: 11.1°C (20°F) for low mass flux and 22.2°C (40°F) for high mass flux.

Appendix A. DESCRIPTION OF THE BROWNS FERRY UNIT ONE RESIDUAL HEAT REMOVAL SYSTEM

A.1 Introduction

The residual heat removal (RHR) system at Browns Ferry Unit 1 comprises four pumps and heat exchangers arranged in two basic piping loops. The major piping associated with components B and D is shown in Fig. A.1; the piping for loops A and C is similar. The RHR system valves on this figure are shown in their normal positions during reactor operation.

The three basic operating modes for the RHR system are:

1. Low pressure coolant injection (LPCI),
2. Primary containment cooling, and
3. Reactor vessel shutdown cooling.

It is the purpose of this appendix to describe the operation of the RHR system in each of its modes in sufficient detail to provide the background necessary to an understanding of the material discussed in the main body of this report. Discussions of the design and operation of this system are available in much greater detail elsewhere.[A.1,A.2]

The LPCI operational mode is an ECCS mode provided for use in the event of a design basis accident. As such, it would not be utilized in a loss of DHR accident sequence unless the accident initiator included a loss of coolant accident (LOCA). The LPCI mode of the RHR system is described in Sect. A.2.

The primary containment cooling mode of the RHR system is utilized for pressure suppression pool cooling as well as for drywell or wetwell spray. This operational mode is discussed in Sect. A.3.

The reactor vessel shutdown cooling mode is used after reactor shutdown and vessel depressurization for long-term decay heat removal. Section A.4 provides a brief description of this operating mode, which is not available by definition in a Loss of DHR accident sequence.

Other features of the RHR system include the capability to pump condensate storage tank water or to channel river water through some of the piping. These and other special system features are discussed in Sect. A.5.

A.2 LPCI Operational Mode

Referring to Fig. A.1, in the LPCI operational mode RHR pumps B and D take suction on the pressure suppression pool ring header through valves 74-24 and 74-35, respectively. Each pump's discharge is routed through the associated heat exchanger and thence into a common 0.61 m (24 in.) line leading through valves 74-66 and 74-67 and check valve 74-68 into the reactor vessel via the piping on the discharge side of recirculation pump A. The arrangement for RHR pumps A and C is similar.

The LPCI operational mode is automatically initiated if the water level in the reactor vessel drops to 9.77 m (384.5 in.)* above the bottom of the vessel or if high drywell pressure [0.115 MPa (2 psig)] exists in conjunction with a low reactor vessel pressure [less than 3.20 MPa (450 psig)]. The RHR pumps start immediately upon an automatic initiation signal, and valves 74-66 and 74-67 automatically open when reactor vessel pressure is less than 3.20 MPa (450 psig). The pumps are protected by a minimum flow line to the pressure suppression pool (not shown in Fig. A.1).

The purpose of the LPCI operational mode is to restore and maintain the reactor vessel coolant inventory after a loss of coolant accident. Water pumped into the reactor vessel would spill from the piping break into the drywell and flow from there back into the suppression pool; this establishes a closed cycle for the flow.

Cooling water for the secondary side of the RHR heat exchangers is provided by the RHR service water system. (The portion of this system that provides cooling water to the D heat exchanger is represented by the dashed lines in the lower right corner of Fig. A.1). However, service water flow to the RHR heat exchangers is not required immediately after a LOCA and there is no provision for automatic actuation of the service water flow to the RHR heat exchangers.

The valve configuration shown in Fig. A.1 supports the LPCI operational mode and is the normal lineup of the RHR system. Furthermore, all RHR system motor-operated valves will automatically realign to the configuration shown in Fig. A.1 if they should be in another alignment when a LPCI initiation signals is sensed.† As previously discussed, valves 74-66 and 74-67 will open only when the reactor vessel pressure is less than 3.20 MPa (450 psig).

With a rated flow of 0.631 m^3/s (10,000 gpm) per pump, the LPCI mode provides a means to rapidly recover the core following a design basis accident. Over the long term, however, the decay heat would have to be removed from the closed cycle. This can be accomplished by continuing to operate one loop in the LPCI mode while switching the other loop to the suppression pool cooling mode, which is described in Sect. A.3.

A.3 Primary Containment Cooling Operational Mode

In the suppression pool cooling mode, the RHR pumps take suction on the suppression pool ring header as in the LPCI mode, but valves 74-66 and 74-67 are shut and the pump discharge passes into the suppression pool through valves 74-71 and 74-73. The RHR flow through the heat exchangers is cooled by river water circulated by the RHR service water (RHRSW) pumps. As shown in the lower right corner of Fig. A.1, RHRSW pump D2 normally serves RHR heat exchanger D with return to the river through

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*This is about 0.61 m (2 ft) above the top of the active fuel in the core. Since the high drywell temperature will cause a decrease in the water density in the reference leg of the level instruments, the actual water level would be 10.34 m (407 in.).

†The one exception if the case where the RHR system is aligned for shutdown cooling. See the discussion in Sect. A.4.

valve 23-52. (To avoid unnecessary clutter, the RHRSW piping for RHR heat exchanger B is not shown.)

Manual action by the operator is necessary in order to place either loop of the RHR system in the pressure suppression pool cooling mode. If a LPCI initiation signal has occurred, an interlock prevents closing valve 74-66 until at least five minutes has elapsed. Valves 74-71 and 74-73 can not be opened unless the reactor core is at least 2/3 covered with water. The operator must also manually initiate the flow of RHR service water to the RHR heat exchangers. However, the RHR system can be operated in the suppression pool cooling alignment to circulate and mix the suppression pool water even if RHR service water is not available for cooling.

One modification of the primary containment cooling mode involves diversion of about 5% of the pressure suppression pool cooling flow through valve 74-72 to a single ring header* located in the uppermost portion of the airspace in the wetwell above the pressure suppression pool. As in the case of valves 74-71 and 74-73, valve 74-72 is interlocked closed unless the reactor core is at least 2/3 covered. In addition, valves 74-71 and 74-72 are interlocked so that both can not be simultaneously open unless a LPCI initiation signal is present. Since a LPCI initiation signal automatically realigns all motor-operated valves on the discharge side of the RHR pumps to the configuration shown in Fig. A.1, an operator desiring to spray the wetwell airspace must first gain control of the RHR system valves by placing a control room selector switch in "manual". If necessary, the requirement that the reactor core be 2/3 covered can be over-ridden by a keylock bypass switch.

A second modification of the primary containment cooling mode involves diversion of some or all of the pressure suppression pool cooling flow to a drywell spray header† via valves 74-74 and 74-75. These two valves can be opened simultaneously only if a LPCI initiation signal is present, the reactor core is at least 2/3 covered, and drywell pressure is at least 0.108 MPa (1 psig).‡ The operator gains control of the valves by placing a selector switch in the "manual" position; if necessary, the requirement that the reactor core be at least 2/3 covered can be over-ridden by a keylock bypass switch.

A.4 Reactor Vessel Shutdown Cooling Mode

During a normal reactor shutdown and cooldown, the reactor vessel is depressurized by steaming to the main condenser at a cooldown rate of less than 37.8 K/h (100°F/h) until the reactor vessel pressure is less than 1.03 MPa (135 psig). Then suppression pool suction valves 74-24 and 74-35 are shut§ and shutdown cooling suction valves 74-47 and 74-48 are opened.

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*This ring header is served by both RHR loops.

†There are two drywell spray headers, each fed from one of the RHR loops.

‡This protects against vacuum-induced internal collapse of the primary containment.

§Assuming the portion of the RHR system shown in Fig. A.1 is to be used. A lineup similar to that described in this subsection could be accomplished using pumps A and C.

The flow induced by one RHR pump is sufficient for the purposes of shutdown cooling, and the operator opens valve 74-25 if RHR pump B is selected.* RHR service water flow is established to the selected heat exchanger. The selected pump is started and valve 74-66 is throttled as necessary to maintain the desired reactor vessel cooldown rate.

If a LPCI initiation signal should occur when the RHR system is operating in the shutdown cooling mode, the operating RHR pump will trip but the suction valves will not automatically realign. Thus, it would be necessary for the operator to shut valves 74-47 and 74-48, reopen valves 74-25 and 74-36, and start the desired RHR pump or pumps if LPCI injection were desired. In this connection, it should be recalled that the RHR system would not be operated in the shutdown cooling mode unless the reactor vessel is at low pressure so that the probability of a large-break LOCA is remote.

A.5 Special RHR System Features

A.5.1 Injection of condensate storage tank water

The suppression pool suction valves 74-24 and 74-35 can be shut at any time and valves 74-34 and 74-45 can be opened to permit RHR pump suction to be taken on the condensate storage tank. This operator action would be desirable if the RHR pumps were to be operated during the latter stages of a loss of DHR accident sequence when the suppression pool temperature/pressure profile might not supply the required net positive suction head to the RHR pumps.

A.5.2 Standby coolant supply

The Unit 1 RHR loop comprising the B and D pumps and heat exchangers includes provision for the passage of river water into the reactor vessel or into the containment spray headers by virtue of a connection to the RHR service water system. With reference to Fig. A.1, it can be seen that with valves 23-57 and 74-101 open and valve 23-52 shut, river water can be injected into the RHR system downstream of the heat exchangers and flow from there into whatever RHR system discharge path is available.† The RHRSW pumps have a rated flow of 0.205 m^3/s (3250 gpm) and a shutoff head of 1.22 MPa (162 psig). Thus the standby coolant supply feature can supply reactor core coverage and cooling if the reactor vessel is depressurized or containment sprays at any time before containment failure by overpressurization.

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*or valve 74-36 if RHR pump D is selected.

†For Unit 1, the only cross-connection between the RHRSW and the RHR system is that shown on Fig. A.1 (i.e., there is no similar arrangement in the piping loop containing pumps A and C.)

A.5.3 Head spray

At the latter stage of reactor vessel shutdown cooling, it becomes desirable to completely fill the reactor vessel with water in preparation for refueling. An RHR system connection to a spray nozzle in the reactor vessel upper head is provided for this purpose (not shown in Fig. A.1). The spray acts to condense the steam in the reactor vessel upper head and thereby facilitates flooding of the reactor vessel.

A.5.4 Fuel storage pool cooling

Connections not shown on Fig. A.1 can be used to augment fuel storage pool cooling. This might be necessary if all of the fuel in the reactor vessel had to be unloaded under emergency conditions.

A.5.5 Cross-connections to unit 2

Provision has been made at the Browns Ferry Nuclear Plant for cross-connections between the individual RHR systems of each of the three units. The purpose is to maintain a long-term reactor vessel and pressure suppression pool cooling capability which does not depend on the integrity of the primary containment or the operability of the RHR system associated with a particular unit. The entire cross-connection network is illustrated in Fig. 4.8-1 of Ref. A.2.

The suctions of RHR pumps B or D in Unit 1 can be cross-connected with the suctions of RHR pumps A or C in Unit 2. The common discharge line from heat exchangers B and D in Unit 1 can be cross-connected with the common discharge from heat exchangers A and C in Unit 2 through valves 74-101 and 74-100 shown in Fig. A.1.* Thus water from the reactor vessel or pressure suppression pool of Unit 1 can be circulated through the A or C heat exchangers in Unit 2 and returned to the source in Unit 1 if the Unit 1 RHR pumps or heat exchangers are not functional. Similarly, the B or D RHR pumps on Unit 1 can be used to circulate Unit 2 pressure suppression pool or reactor vessel water through the B or D heat exchangers on Unit 1.

An arrangement similar to that described above for Units 1 and 2 exists between the B and D RHR pumps and heat exchangers of Unit 2 and the A and C RHR pumps and heat exchangers of Unit 3.

The piping in the crosstie network is sized for a minimum flow of 0.315 m^3/s (5000 gpm) whereas under normal conditions full heat exchanger primary side flow is 0.630 m^3/s (10000 gpm). With the lower flow, the heat transfer capability of a RHR heat exchanger is about 91% of the capacity at full flow.[A.2]

The cross-connection piping can also be used for net transfer of water between units by leaving the RHR pump suction valves of one unit in normal alignment while opening the heat exchanger cross-connection to the adjacent unit. In this manner, pressure suppression pool water from the first unit, which has been cooled by the first units' heat exchangers, can

*For simplicity, these cross-connections have not been shown on Fig. A.1.

be discharged into the second units' suppression pool, or used to flood the reactor vessel or spray the drywell or wetwell airspace of the second unit.

References for Appendix A

A.1 Systems Manual - Boiling Water Reactors, Inspection and Enforcement Training Center, U.S. Nuclear Regulatory Commission, Sect. 10.6.

A.2 Browns Ferry Final Safety Analysis Report, Sect. 4.8.

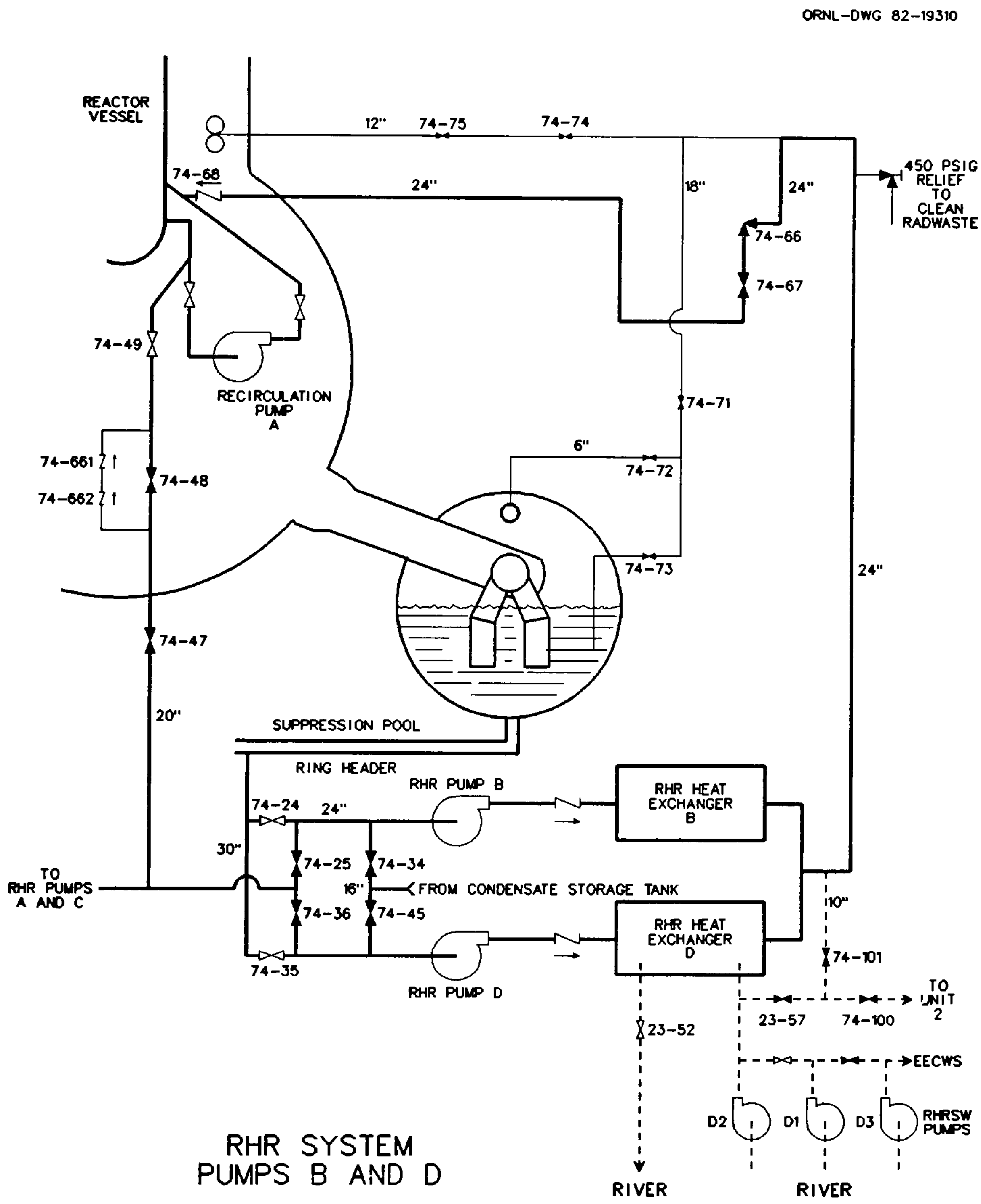

Fig. A.1. Unit 1 RHR system.

Appendix B

MODIFICATIONS TO BWR-LACP FOR THIS STUDY

The BWR-LACP code was used to perform the calculations discussed in Chaps. 3, 4, 5, and 6. Modifications to the code that were necessary for the Loss of DHR accident sequences are described in this appendix. The BWR-LACP calculations for Chaps. 5 and 6 also used a special pool model, described in Appendix C, which is able to calculate the thermal stratification effect which occurs when the pool is heated by the SRVs without RHR flow to mix the pool. The BWR-LACP code has been described in previous ORNL SASA reports.[B.1, B.2]

B.1 RHR Heat Exchangers

The RHR system accomplishes pool cooling by circulating water from the suppression pool, through the shell side of the RHR heat exchangers, and back to the pool. The RHR Service Water (RHRSW) is pumped in an open cycle from the river through the tube side and back to the river.

The rate of heat exchange is calculated by utilizing the effectiveness formulation:

$$Q = E\, C_{min}\, (T_{h,in} - T_{c,in})$$

where,

Q = rate of heat transfer from pool water to river water,
E = heat exchanger effectiveness,
C_{min} = the smaller of the tube side and the shell side mass flow times specific heat products,
$T_{h,in}$ = inlet temperature of the hotter fluid as it enters the heat exchanger, and
$T_{c,in}$ = inlet temperature of the colder fluid (i.e., the river water).

Each of the RHR heat exchangers has two tube passes and one shell pass. The formula for effectiveness of this arrangement is:[B.3]

$$E = 2/[(1+C_r) + \sqrt{1+C_r^2}\,(1+e^{-x})/(1-e^{-x})]$$

where,

C_r = C_{min} (defined above) divided by C_{max} (the larger of the shell side and tube side mass flow rate*specific heat products,
x = $UA(\sqrt{1+C_r^2})/C_{min}$,
U = overall heat transfer coefficient, based on the total heat transfer area, A.

The total heat transfer coefficient, U, is dominated by the thermal resistance of the tube metal and by the shell-side and tube-side fouling allowances;[B.4] therefore, the effectiveness is not sensitive to fluid temperatures and there is no need for an iterative solution.

B.2 Pump Net Positive Suction Head (NPSH)

In general, pump NPSH can be expressed as

$$H_{npsh} = (P_s - P_v)/W + V^2/2g$$

where,

P_s = static pressure at the centerline of the pump suction inlet pipe
P_v = vapor pressure of the fluid being pumped
W = weight density of fluid being pumped
V = velocity of fluid in the pump suction inlet
g = acceleration of gravity

For the case of the ECCS pumps taking suction on the suppression pool, the formula, above, is equivalent to the following:

$$H_{npsh} = \Delta H_c + (Z_{npl} - Z_{ps}) - H_1$$

where,

ΔH_c = a combination of terms which is the same for all pumps which take suction on the pool
= $(P_{tspa} - P_v)/W + (L_p - L_{po})/12$
P_{tspa} = total pressure of the wetwell atmosphere
L_p = measured level of the pool (in. from instrument zero)
L_{po} = normal measured level of pool (i.e., 4 in. below instrument zero)
Z_{npl} = normal elevation of the pool surface (i.e., elevation when pool level is 4 in. below instrument zero)
Z_{ps} = elevation at centerline of pump suction
H_1 = head loss between pool and pump suction inlet

For each of the pumps that can take suction on the suppression pool, the specific formula for NPSH (ft) is based on information provided by the TVA for Browns Ferry Unit 1:

$$\text{H-NPSH}_{rhr} = \Delta H_c + 14.5 - (N^2_{rhrl} + 0.9)(B_{rhr}/10{,}000.0)^2$$

$$\text{H-NPSH}_{cs} = \Delta H_c + 14.8 - 1.03(N^2_{csl} + 0.83)(B_{cs}/3125.0)^2$$

$$H\text{-}NPSH_{hpci} = \Delta H_c + 12.0 - 6.5(B_{hpci}/5{,}000)^2$$

$$H\text{-}NPSH_{rcic} = \Delta H_c + 12.0 - 6.5(B_{rcic}/600)^2$$

where

N_{rhrl} = number of RHR pumps running per loop (there are two pumps in each loop and they share a length of suction piping),
B_{rhr} = flow per RHR pump (expressed in gpm)
N_{csl} = number of core spray pumps in each loop (there are two core spray pumps in each loop and they share a length of suction piping)
B_{cs} = flow per core spray pump (gpm)
B_{hpci} = HPCI pump flow (gpm)
B_{rcic} = RCIC pump flow (gpm)

B.3 Drywell Coolers

The drywell coolers transfer heat from the drywell atmosphere to the reactor building closed cooling water (RBCCS) system. The RBCCW system pumps water through the inside of the heat exchanger tubes; the cooler's blowers draw the drywell air (i.e. nitrogen and water vapor) across the exterior of the tubes. Air-side heat transfer is enhanced by the many closely spaced parallel copper sheets which are attached to the outside of the tubes.

The calculation of heat transfer within the drywell coolers is similar to that for the RHR heat exchangers in that the effectiveness formulation is used; however, an iterative solution is employed beause the heat transfer properties of the air-side are radically altered when the water vapor fraction becomes significant (after 8 h, or more, into the loss of DHR accident sequence).

The equation for counterflow heat exchanger effectiveness is:[B.3]

$$E = (1 - e^{-x})/(1 - C_r e^{-x})$$

where

$C_r = C_{min}/C_{max}$
C_{min} = smaller of the air-side mass flow * specific heat and RBCCW-side mass flow * specific heat products
C_{max} = larger of the air and RBCCW-side mass flow * specific heat products
$x = (1 - C_r)UA/C_{min}$
UA = overall air-to-water heat transfer coefficient * effective heat transfer area

The iteration proceeds as follows:

1. A value of air-side C_a is assumed (mass flow * specific heat),
2. E is calculated, along with the resulting heat transfer rate, airside ΔT, and air-side exit temperature,
3. The air-side condensation rate is then calculated from the results of step 2; combined with the known saturation pressure of steam as a function of temperature,
4. The condensation rate is then used to calculate the total heat transfer rate (latent + sensible), and a corrected value for C_a:

$$C'_a = \frac{Q}{\Delta T}$$

5. If $C'_a \cong C_a$, the iteration is terminated. If not, then steps 1-4 are repeated until convergence is achieved.

Major assumptions of the drywell cooler model include:

1. constant volumetric flow maintained by the drywell cooler blowers,
2. constant RBCCW system inlet temperature and flow,
3. constant overall air-to-RBCCW heat transfer coefficient * effective heat transfer area.

B.4 Torus Room Temperatures

In an unmitigated loss of DHR accident the surface temperature of the uninsulated torus can exceed 149°C (300°F). The greatest heat loss (at this temperature) from the surface of the torus is by radiant heat transfer directly to the ~0.9 m (3 ft) thick concrete walls (also to the floor and ceiling) of the torus room. This heat transfer rate[B.3] as evaluated using an assumed emissivity of 0.9 for both torus and concrete surfaces:

$$Q_r = SA_t(T_t^4 - T_c^4)/[1/e_t + (A_t/A_c)(1/e_c - 1)]$$

where

Q_r = total radiant heat transfer rate
S = Stefan-Boltzman constant
A_t = surface area of torus
A_c = surface area of concrete
T_t = surface temperature of the torus
T_c = surface temperature of the concrete
e_t = surface emissivity of the torus
e_c = surface emissivity of the concrete

The rate of natural circulation convective heat transfer between the torus and the torus room air is as follows:

$$Q_c = A_t h \; (T_t - T_a)$$

$$h = 5.3(10)^{-5} \; (T_t - T_a)^{0.33}$$

where the units on h are Btu/(ft²s°F)

A similar expression is employed for calculation of heat transfer between torus room air and torus room concrete surfaces.

A differential energy equation is solved for each of the following temperatures: torus surface, torus room air, and torus room concrete. The ~0.9 m (3 ft) concrete walls are divided into five parallel, slab-geometry, regions to insure accurate computation of the temperature distribution within the concrete. The surface slab is 2.54 cm (1 in.) thick, the adjacent slab 5.08 cm (2 in.) thick, and so on, with slab thickness increasing with penetration into the concrete. A typical concrete differential energy balance is:

$$dT_i/dt = [2\ K_c/(DX_i C_{pc})][(T_{i-1} - T_i)/(DX_{i-1} + DX_i) - (T_i - T_{i+1})/(DX_i + DX_{i+1})]$$

where

T_i = temperature of the i-th concrete slab
K_c = concrete thermal conductivity
DX_i = thickness of i-th concrete slab
C_{pc} = specific heat of concrete

The expression for natural circulation of air from the reactor building basement (i.e. the corner rooms), into the torus room, and out the top of the torus room is based on a discussion in Sect. 5.2.6.3 of the Browns Ferry FSAR. In adapting the circulation rate given by the FSAR, it was assumed that the rate of natural circulation is proportional to the square root of the density difference (i.e. density of reactor building basement air outside the torus room vs. the density of the air inside the torus room) and, therefore, also proportional to the square root of the temperature difference. The basement air outside the torus room is assumed to remain close to 32°C (90°F) throughout the loss of DHR sequence.

B.5 Containment Leakage and Containment Vent Flow

Two types of primary containment leakage are considered in this report: normal leakage (leakage through penetration seals, etc.) and intentional venting (leakage through the drywell and wetwell vent lines). All the BWR-LACP calculations discussed in this report assumed that the primary containment leaks at a rate consistent with that measured at the Browns Ferry plant. Only the calculation reported in Sects. 4.3.1 and 10.2 assumed that the 5.08 cm (2 in.) vent lines were open.

As a practical matter, normal primary containment leakage is not capable of retarding containment pressurization during the loss of DHR accident sequence. The Browns Ferry Technical Specifications would allow a leakage of up to 2% of total primary containment volume per day at pressures up to the design pressure. Testing conducted to date at Browns Ferry has shown the actual leak rate to be less than a tenth of the allowable leakage. In BWR-LACP, normal leakage is modeled as a constant 0.2% per day. Since the density increases with pressurization in the loss of DHR sequence, the leakage mass flow increases in proportion to the increasing pressure.

Discharge through the 5.08 cm (2 in.) drywell and wetwell vents can have a significant effect on the timing of containment failure during an otherwise unmitigated Loss of DHR sequence (see Sect. 4.3.1). The drywell and wetwell vent lines both discharge into a common 5.08 cm (2 in.) vent line which in turn discharges to the 45.7 cm (18 in.) Standby Gas Treatment system reactor building ventilation ducts; therefore, the total flow rate is limited when sonic flow occurs in the common section of discharge pipe. The flow resistance afforded by the discharge piping, valves, and fittings must be considered, even when there is sonic flow. Page 4-13 of Ref. 6 presents two different calculations of the flow rate of dry saturated steam from 1.17 MPa (170 psia) through 9.15 m (30 ft) of 2-in. Schedule 40 pipe to atmospheric pressure: both predict a discharge rate of 1.45 kg/s (3.2 lb/s). The example in Ref. 6 includes the flow resistance of not only 9.15 m (30 ft) of piping but also a fully open globe valve and a standard 90° elbow, as well as entrance and exit losses. For this analysis it was assumed that these losses are representative of the losses in the piping actually installed at Browns Ferry Unit 1.

The BWR-LACP calculation of the 5.08 cm (2 in.) vent line flow was based on the reference condition of 1.45 kg/s (3.2 lb/s) of dry steam at 1.1 MPa (170 psia). The mass flow at other conditions was calculated by multiplying the reference flow by ratios to correct for the changed conditions:

$$W = (P/P_r)(\sqrt{M/18})\ (\sqrt{T_r/T})(W_r)$$

where,

W = flow at any pressure or temperature or nitrogen/steam composition
W_r = flow at reference conditions
P = primary containment pressure
P_r = reference primary containment pressure
M = mole-fraction-averaged molecular weight
T_r = reference primary containment temperature
T = primary containment temperature

This relationship is based on the variation of sonic velocity of ideal gases with respect to upstream pressure and temperature and with respect to molecular weight.

B.6 Evaporation from Surface of Suppression Pool

Evaporation of water vapor from the surface of the suppression pool to the wetwell atmosphere is the dominating mechanism for pressurization of the primary containment during the Loss of DHR sequences analyzed in this report. As the pool is slowly heated, evaporation causes the containment pressure to rise fast enough such that the suppression pool remains subcooled and there is no direct bubble-through of SRV discharge from the T-quenchers to the wetwell atmosphere.

The wetwell atmosphere is initially a mixture of nitrogen and a small amount of water vapor, and remains a binary mixture throughout most of the Loss of DHR sequences. If the wetwell atmosphere were pure water vapor, then it would be correct to assume that the partial pressure of water vapor in the wetwell atmosphere is identical to the saturation pressure evaluated at the temperature of the water at the surface of the pool. Since the wetwell atmosphere is a binary mixture, this assumption is not correct because the rate of transfer of water vapor from the pool is limited by diffusion and convection through the air.

The relationship used to calculate the rate of evaporation from the pool is based on the heat transfer/mass transfer analog discussed in Chap. 13 of Ref. 5:

$$W = (12.3\ A_p R_m)(P_{vs} - P_{va})h/\overline{P}_n$$

where,

W = evaporation rate (lb/s)

12.3 = constant of proportionality

A_p = surface area of pool (ft^2)

R_m = Mole-fraction-weighted (i.e. between nitrogen and water vapor) perfect gas constant

P_{vs} = vapor pressure (psia) of water, evaluated at the pool surface temperature

P_{va} = partial pressure of water vapor in wetwell atmosphere

$\overline{P}_n$ = average nitrogen pressure in the convective boundary layer (psia)

h = coefficient of natural circulation heat transfer between the surface of the pool and the pool atmosphere [Btu/(s °F ft^2)]
$= 5.85 \times 10^{-5}\ (T_p - T_a)^{0.333}$

T_p = pool surface temperature (°F)

T_a = wetwell atmosphere temperature (°F)

B.7 Flow Rate of Vessel Water Injection by the CRD Hydraulic System

The CRD hydraulic system is described in detail in Appendix E of Ref. 2. The computations presented in this report make the assumption that the vessel injection flow rate before a reactor trip* is 3.8 ℓ/s (60 gpm) and that after a reactor trip the vessel injection increases to 10.7 ℓ/s (170 gpm).

The CRD vessel injection before a reactor trip is maintained at a constant flow rate by an automatic flow control valve. Part of the total normal injection flow goes to each of the 185 CRDs. By maintaining a cold flow into the reactor vessel the normal CRD flow prevents hot reactor coolant from coming into contact with the CRD seals.

After a reactor trip, the 185 scram inlet valves (connected to the charging header which is upstream of the flow control valve, but downstream from the flow measurement orifice) open and divert flow into the charging header. This causes measured flow to increase. The flow control valve closes in an attempt to hold measured flow constant, thereby diverting all the flow into the charging header. The maximum flow through the charging header is limited by a fixed flow-restricting orifice. This is necessary because the discharge head of the CRD hydraulic pumps is 9.07 MPa (1300 psig) or more, and the pressure downstream of the scram inlet valves can vary from normal reactor pressure all the way down to atmospheric pressure. The maximum flow permitted by the flow restricting orifice is 10.7 ℓ/s (170 gpm) and this flow occurs when reactor vessel pressure is low. Since the reactor vessel is depressurized to <1 MPa (130 psig) throughout most of the loss of DHR sequence, a constant flow rate of 170 gpm was assumed for the post/scram vessel injection flow rate.

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*i.e., during normal power operation, or following a reactor trip when the scram has been reset.

References for Appendix B

B.1. S. A. Hodge et al., "Station Blackout at Browns Ferry Unit One – Accident Sequence Analysis," NUREG/CR-2182, ORNL/NUREG/TM-455/V1, November 1981.

B.2. S. A. Hodge et al., "SBLOCA Outside Containment at Browns Ferry Unit One – Accident Sequence Analysis," NUREG/CR-2672, ORNL/TM-8119/V1, November 1982.

B.3. A. J. Chapman, "Heat Transfer," Second Edition, New York, The Macmillan Company, 1967.

B.4. S. Clark, "Instruction Manual for TVA Browns Ferry Residual Heat Removal Exchangers BF-1-10-2, BF-2-10-2, and BF-3-10-2," General Electric Co., San Jose, CA, August, 1970.

B.5. Frank Kreith, "Principles of Heat Transfer," Second Edition, Scranton, PA, International Textbook Company, 1965.

B.6. Crane Co. Engineering Division, "Flow of Fluids Through Valves, Fittings, and Pipe," Technical Paper No. 410 (Fourteenth printing-1974), The Crane Company, Chicago, IL.

Appendix C

MODELING FOR LOCALIZED HEATING OF PRESSURE SUPPRESSION POOLS

C.1 Introduction

Most of the pressure suppression pool (PSP) modeling that is done treats the pool as a single, well-mixed node. This is an accurate model if the energy is added to the pool at many locations or if the mass fluxes are large enough to ensure thorough mixing. In a situation where these requirements are not met (particularly during severe accidents) the single node model is inadequate. For those situations, a model that produces more detailed information about local temperatures is needed.

The overall goal of the PSP modeling is to write a computer program that will fulfill this need, i.e., to produce local PSP temperatures as a function of time. Ideally, the code would be able to handle 3-D toroidal transport with a free surface and a non-extraneous condensation source. The only practical way to model such a problem is to make such approximations as are necessary to keep computer costs low while preserving enough physics to make the solution reasonably accurate.

The PSP model must be capable of following the pool local temperatures from an initial, well-mixed condition through time to a point when the drywell fails due to overpressure. This involves modeling SRV steam flow from those at full reactor pressure (~ 200 lbm/s) to those at the low end of the decay heat curve (~ 25 lbm/s). It also involves modeling a SRV that is either stuck open or held open as well as a SRV that operates intermittently at high mass fluxes.

With the above ranges of size and scope applicable to PSP modeling outlined, the remainder of this appendix describes the phenomena of interest, a computer model of the phenomena, and some details about plume transport analysis. This Appendix is written as a brief overview of the PSP modeling effort. A detailed report of the model and results is being prepared for publication.

C.2 Phenomenology

There are many phenomena that exist in PSP dynamics, but the present work deals only with those that apply to SRV discharge through a T-quencher. A T-quencher is shown in Fig. C.1.

The T-quencher was designed to discharge toward the walls of the torus instead of circumferentially around the torus. This produces turbulent mixing in Bay D (the discharge bay) but very little turbulent mixing with the adjacent bays. As a result, local temperatures* in the bay of discharge can become high.

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*Local temperature here is defined as in Ref. 2: the average of temperatures at points directly above and below the T-quencher. In the computer model, local temperature is defined as the temperature associated with a given node.

The magnitude of the local temperature increase will be determined by the Bay D recirculation, the whole pool recirculation, and the pool thermal stratification.

The concept of a Bay D recirculation flow is shown schematically in Fig. C.2. For high steam flow rates, the momentum of the steam jets is large enough that a turbulent hot water jet impinges on the torus wall and turns upward, accelerated by its buoyancy. When it reaches the surface, it turns downward and moves back to the T-quencher. This case is shown on the right half of Fig. C.2.

For low steam flows, the buoyancy is dominant and the hot water created by the condensation forms a plume that turns upward before reaching the torus wall, rises to the surface, and spreads out. The low steam flow case is shown on the left half of Fig. C.2.

In the high steam flow case, Bay D is well mixed. In the low steam flow case, Bay D is not well mixed. The entire range of T-quencher flows falls between these two extremes. For each steam flow through the quencher, the Bay D mixing is determined by whether the dynamics are dominated by the plume behavior or the jet behavior.

The concept of whole pool recirculation can be understood on the basis of continuity. As hot water moves upward in Bay D, it forces part of the water that is there to move circumferentially out across the surface of the pool. Continuity implies that there must be a cold water inflow to Bay D from the lower layers to make up for the outflow. A whole pool recirculation flow is thus started that consists of hot water moving up to the surface in Bay D, around and across the pool, together with colder water that is moving down and back toward the T-quencher. It is this whole pool circulation that keeps the discharge bay cool. Experimental evidence[C.1] indicates that the magnitude of this recirculation flow is quite large. Correct modeling of this phenomenon is essential to determination of local temperatures in Bay D.

The thermal stratification phenomenon is very simple to understand. Hot water is buoyant and tends to distribute evenly in temperature across the surface of the pool. The cold water tends to distribute in layers underneath the hot water. However, the thermal stratification phenomenon is very difficult to model rigorously because of the lack of basic physical understanding of the turbulent mixing processes. An attempt is made to model the thermal stratification in the model based on a very simple kinematic treatment.

C.3 Pressure Suppression Pool Model

The pressure suppression pool model is an N-layer, lumped parameter model for hot water transport in the torus. The user can input N, the number of layers. The computer program is designed to follow the PSP in time from a well-mixed initial condition up to near the local saturation temperature. Stable condensation is always assumed to occur if the local fluid is subcooled. The objective of the code is to model the thermal mixing that occurs between the discharge bay and the rest of the torus, and, in the process, to produce local temperatures as a function of time.

The code currently models one SRV discharging through a T-quencher to the pool. The SRV steam flow and enthalpy as a function of time are assumed to be known input data. SRV discharge line dynamics are not modeled. The pool is coupled to the wetwell airspace through the airspace total pressure and through the pool-to-airspace evaporation rate (both of which are also assumed to be known input data).

As shown in Fig. C.3, the pool is divided into N stratified layers vertically (N=4 is shown), and 18 nodes in the Θ direction (around the torus). There is one Θ - node for each bay of the PSP, except for the discharge bay and the bay located 180° from the T-quencher. These two bays have two Θ - nodes each.

The code uses a quasistatic approach in modeling the PSP. At the beginning of each time step the local pool temperatures are used to calculate the condensation at the T-quencher. The condensed steam and the cold water feed flow that was necessary to produce the condensation are transported upward from the T-quencher within the pressure suppression pool using a steady state plume transport analysis. The steady state plumes (four are calculated for each T-quencher) are assumed to be formed instantly, and to exist over the entire timestep. The flow conditions at the end of the plume transport (the plume entrainment and outflow) are then used as effective sources for calculating the new overall pool temperatures. Once new temperatures throughout the pool are known, a new local temperature can be used to calculate the condensation source at the next timestep. Thus, feedback from the entire pool transport enters into the local condensation calculation. Figure C.4 is a schematic showing the steps in the pool calculation.

In order to perform a lumped parameter fluid flow calculation, assumptions about the flow field must be made. In the current model, there are two different flow fields. The first flow field models PSP flow from node to node when the T-quencher is discharging. The second flow field models PSP dynamics after the T-quencher stops discharging.

The first flow field is shown in Fig. C.5. Basically, the pool is treated as a very large convection cell. The energy source in Bay D consists of the flow and temperature output of the T-quencher/plume transport calculation. Inflow to Bay D is the cold water feed flow, $\dot{M}c$, which is evenly divided between the node containing the T-quencher and the nodes below. Outflow from the discharge bay consists of the cold water feed and the quencher steam flow, $\dot{M}_{st}$. Away from the discharge bay, the pool is assumed to move down uniformly (based on continuity) to accomodate the input flow to the top layer. In the discharge bay, an internal circulation is modeled by putting the entrainment flow rates from the plume transport module back into the Bay D nodes above the quencher.

The second flow field is shown in Fig. C.6. This flow field is designed to produce thermal stratification following SRV closure. An equal and opposite mass flow between each cell and its neighbor is produced that is proportional to the square root of the density difference between the cells. This approach to numerically producing the thermal stratification is based on a simple buoyancy calculation and the Taylor instability mechanism.

A general energy balance is written for each cell that permits flow into and out of each face. Each node has an arbitrary source term that can be used to model phenomena such as evaporation from the surface and entrainment from the plume. In addition, each node has a storage term to allow the mass of the node to change over the timestep. This feature is used to model the moving pool level in the surface nodes.

At each time step, the system is represented in standard state variable formulation:

$$\frac{dT}{dt} = AT + S ,$$

where

T = vector of unknown temperatures,
A = the system matrix,
S = source vector.

This equation is then solved to determine the temperatures at the new timestep.

C.4 Plume Transport Analysis

The plume transport analysis assumes that four steady state plumes on each side of the T-quencher arms are set up in Bay D at the beginning of the time step. The four different plumes correspond to the four zones of holes on the T-quencher (there are 16 plumes in all). Each plume transports an appropriate fraction of $\dot{M}_c$ (the cold water feed to the quencher) and $\dot{M}_{st}$ (the quencher steam flow) vertically through the stratified layers above the T-quencher. Entrainment occurs from these surrounding layers and is proportional to the average velocity of the plume. Figure C.7 shows a schematic view of the plume above a discharging T-quencher, along with some of the more important variables.

The unknowns in the problem are r(z), w(z), and T(z): the plume radius, velocity, and temperature, respectively, as a function of z. Known from the previous pool calculation is the average temperature [$T_{out}(z)$] of the nodes surrounding the plume.

The 1-dimensional steady state equations for conversion of mass, momentum, and energy for the plume are:

$$\frac{d}{dz} (\rho w r^2) = 2\alpha\rho_o r w , \qquad (1)$$

$$\frac{d}{dz} (\rho w^2 r^2) = r^2 g \ (\rho_o - \rho) , \qquad (2)$$

$$\frac{d}{dz} (\rho w r^2 T) = \rho_o \alpha w r T_o , \qquad (3)$$

where

$\rho = \rho[T(z)]$, the plume density
$\rho_o = \rho_o[T_o(z)]$, the layer density
α = the entrainment coefficient
$T_o = T_{out}(z)$, the stratified layer temperatures.

The above equations are re-cast in the form:

$$\underset{\sim}{x} = \begin{Bmatrix} x_1 \\ x_2 \\ x_3 \end{Bmatrix} = \begin{Bmatrix} \rho w r^2 \\ \rho w^2 r^2 \\ \rho w r^2 T \end{Bmatrix} , \tag{4}$$

$$\frac{d\underset{\sim}{x}}{dz} = f(\underset{\sim}{x}) . \tag{5}$$

Equation (5) is solved explicitly by marching in small space steps from the initial condition near the quencher to the top of the plume. At each space step, continuity and conservation of energy are enforced by iterating on the entrained mass flow rate and plume temperature. Figure C.8 shows the plume solution method.

The initial condition for the plume is found by calculating the amount of cold water (at the quencher node temperature) necessary to bring $\dot{M}_{st}$ to equilibrium at the saturation temperature. The initial plume temperature and density are assumed to be those of saturated water at the local pressure. A guess is made for r_o, the initial plume radius, and the initial plume velocity is calculated based on continuity:

$$W_o(z) = \frac{\dot{M}_c + \dot{M}_{st}}{\rho_o \pi r_o^2} \tag{6}$$

The preliminary results are found to be relatively insensitive to r_o.

The plume transport analysis is designed to calculate the Bay D mixing that occurs when a T-quencher is discharging. The analysis focuses on the plume dynamics in Bay D instead of the jet dynamics. It is expected that the results will be more applicable to medium and low SRV steam flows* than to high SRV steam flows. However, some success at high steam flows has been found by adjusting the entrainment coefficient. Adding a calculation of the mixing within the bay of discharge due to the jet dynamics would improve the accuracy at high steam flow rates and is planned for future work.

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*"high" SRV flow occurs when the reactor vessel is fully pressurized [pressure ~7.59 MPa (1100 psia)] and is about 100 kg/s (220 lb/s). The "medium" and "low" discharge rates are experienced after depressurization of the reactor vessel has been initiated.

References for Appendix C

C.1. B. J. Patterson, "Mark I Containment Program Monticello T-Quencher Thermal Mixing Test Final Report," NEDO-24542 (August 1979).

C.2. T. M. Su, "Suppression Pool Temperature Limits for BWR Containments," NUREG-0783 (November 1981).

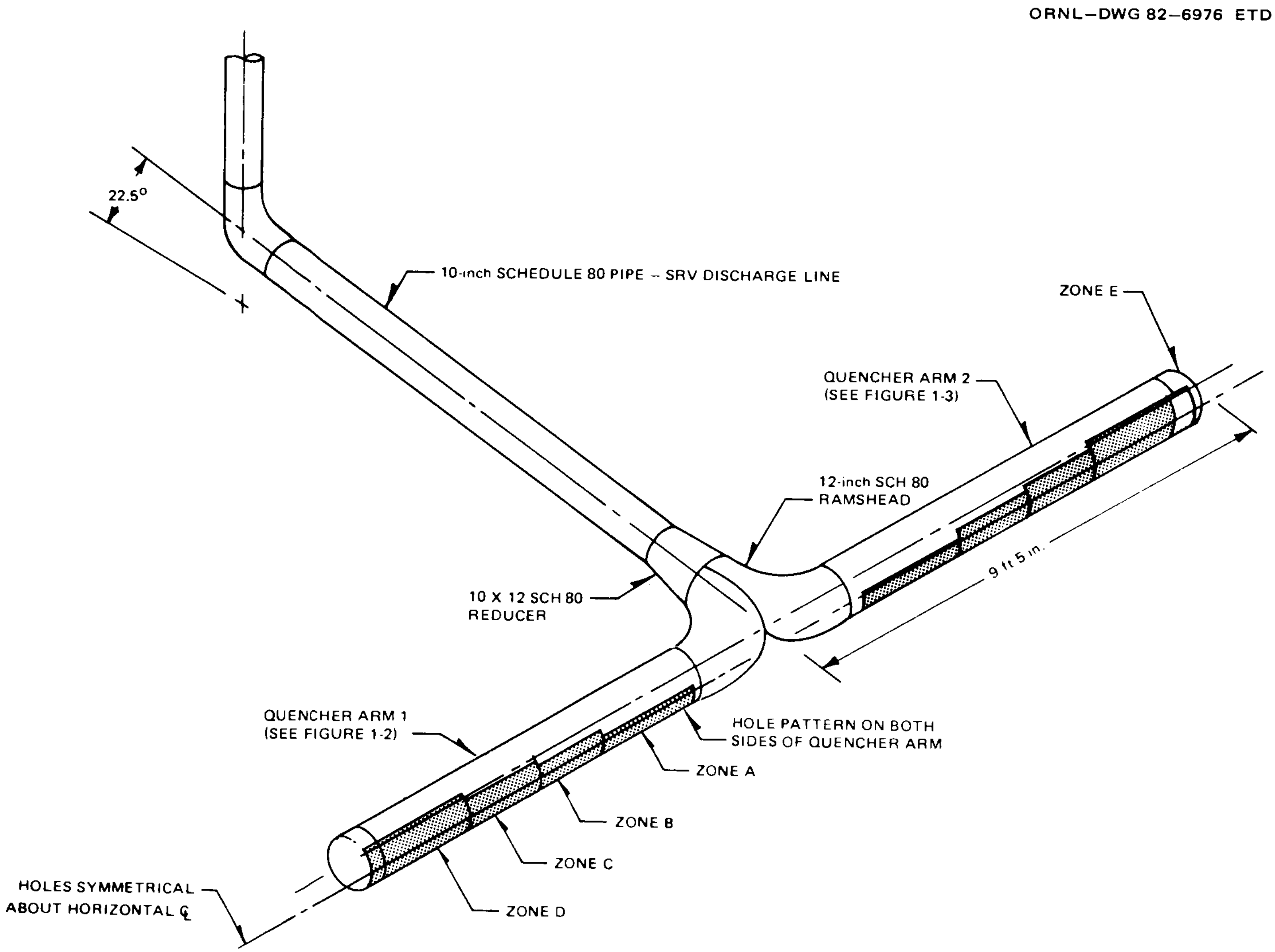

Fig. C.1. Typical T-quencher (from Ref. C.1).

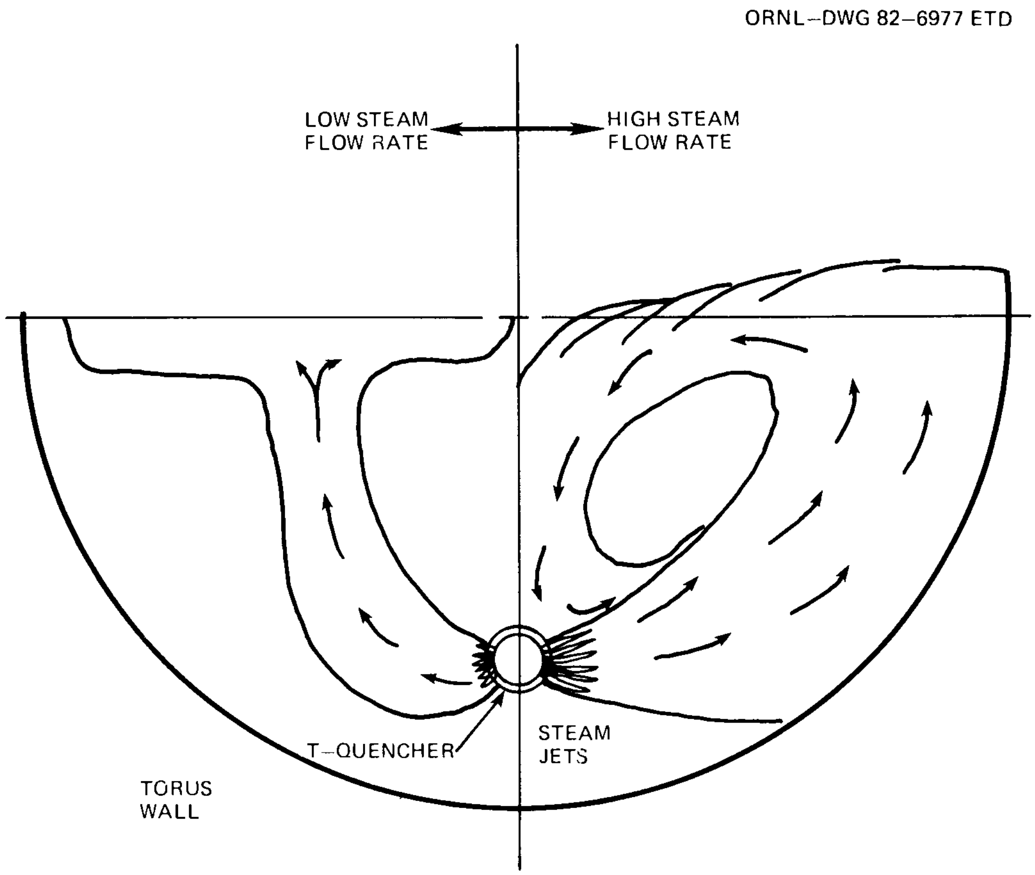

Fig. C.2. Bay D recirculation.

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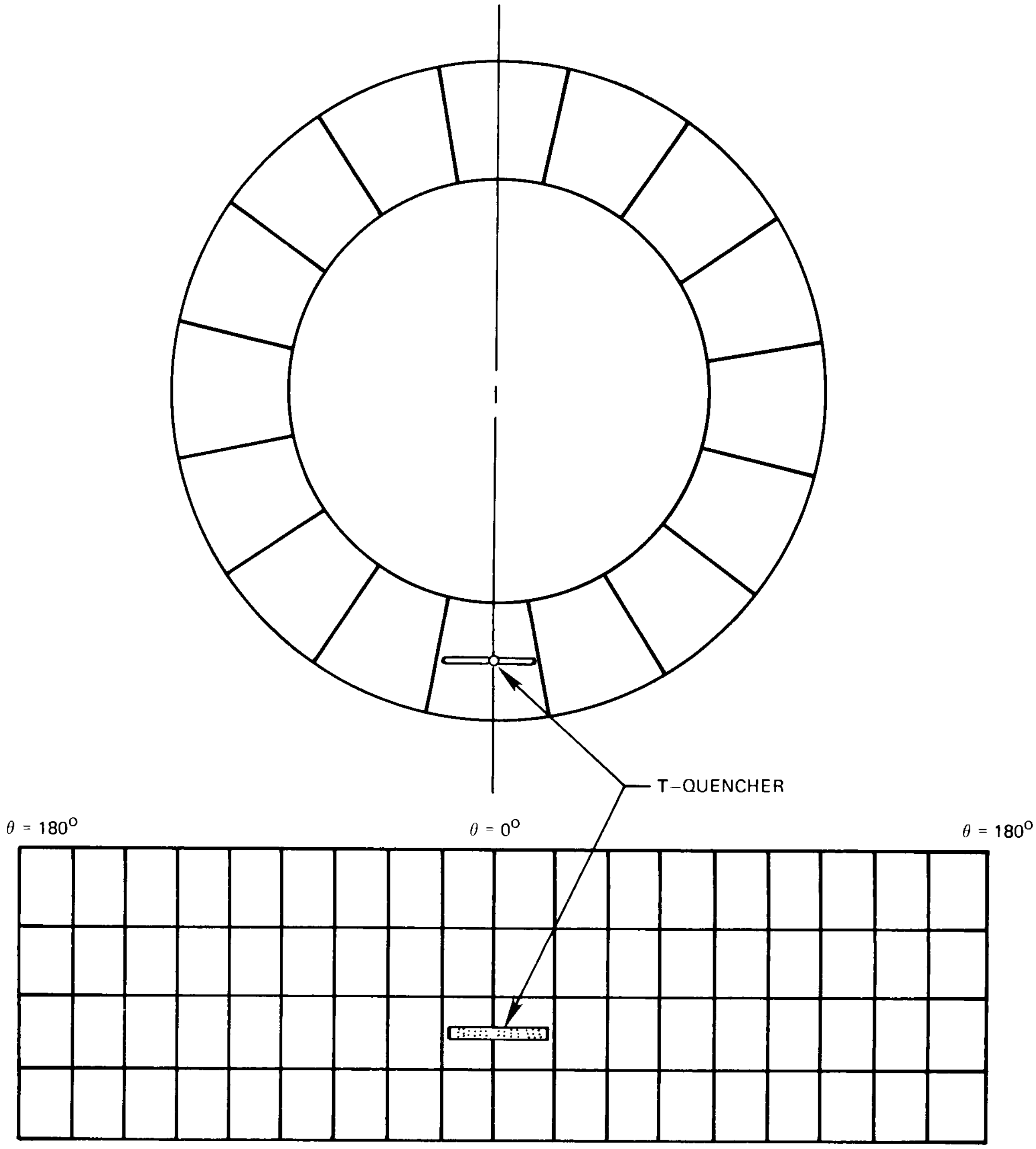

Fig. C.3. Pool model geometry.

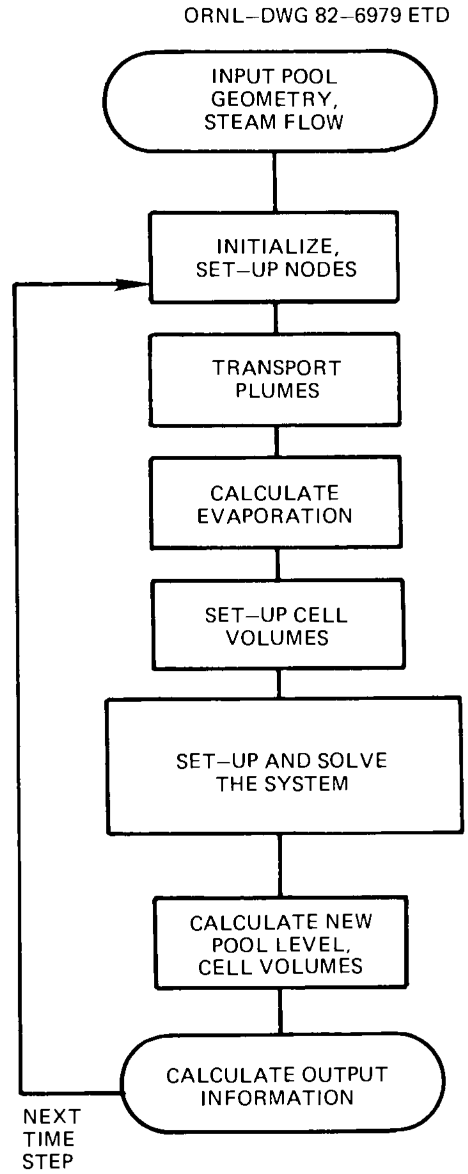

Fig. C.4. Calculational steps in the code.

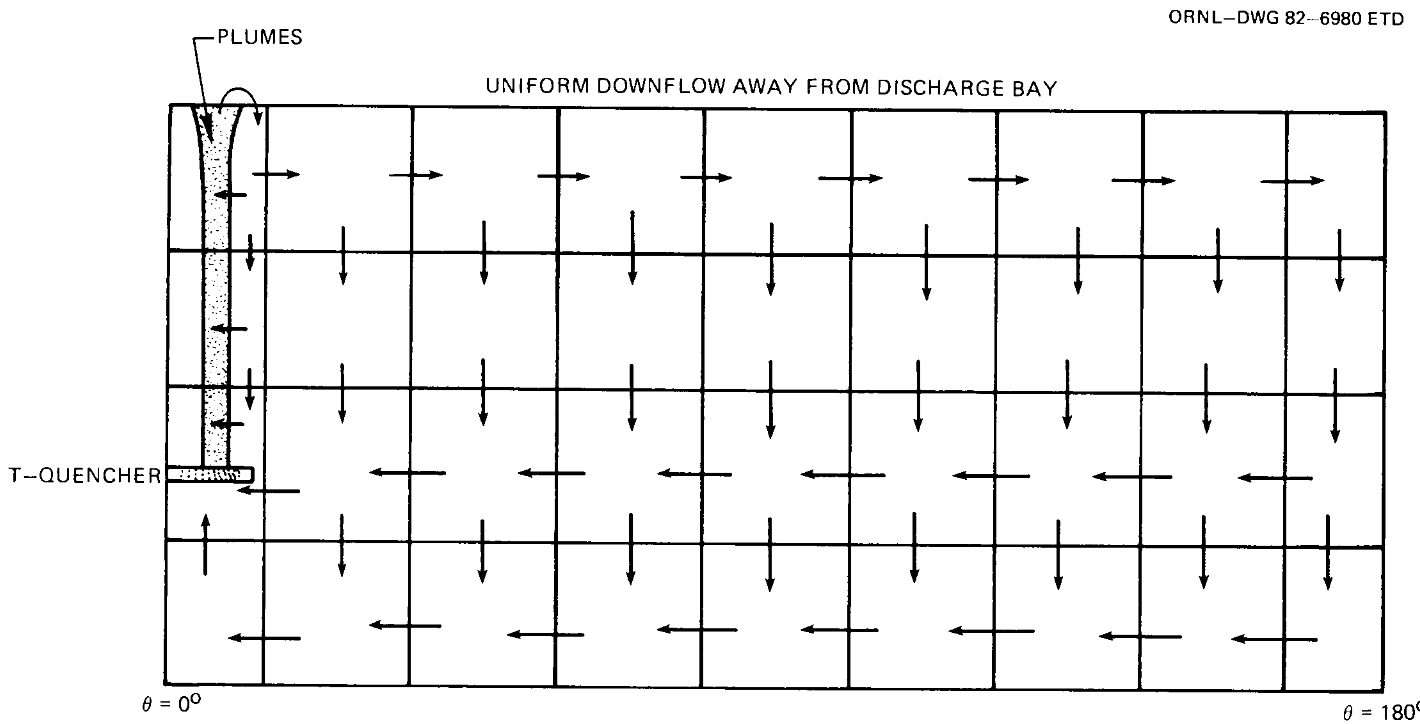

Fig. C.5. T-quencher discharge flow field.

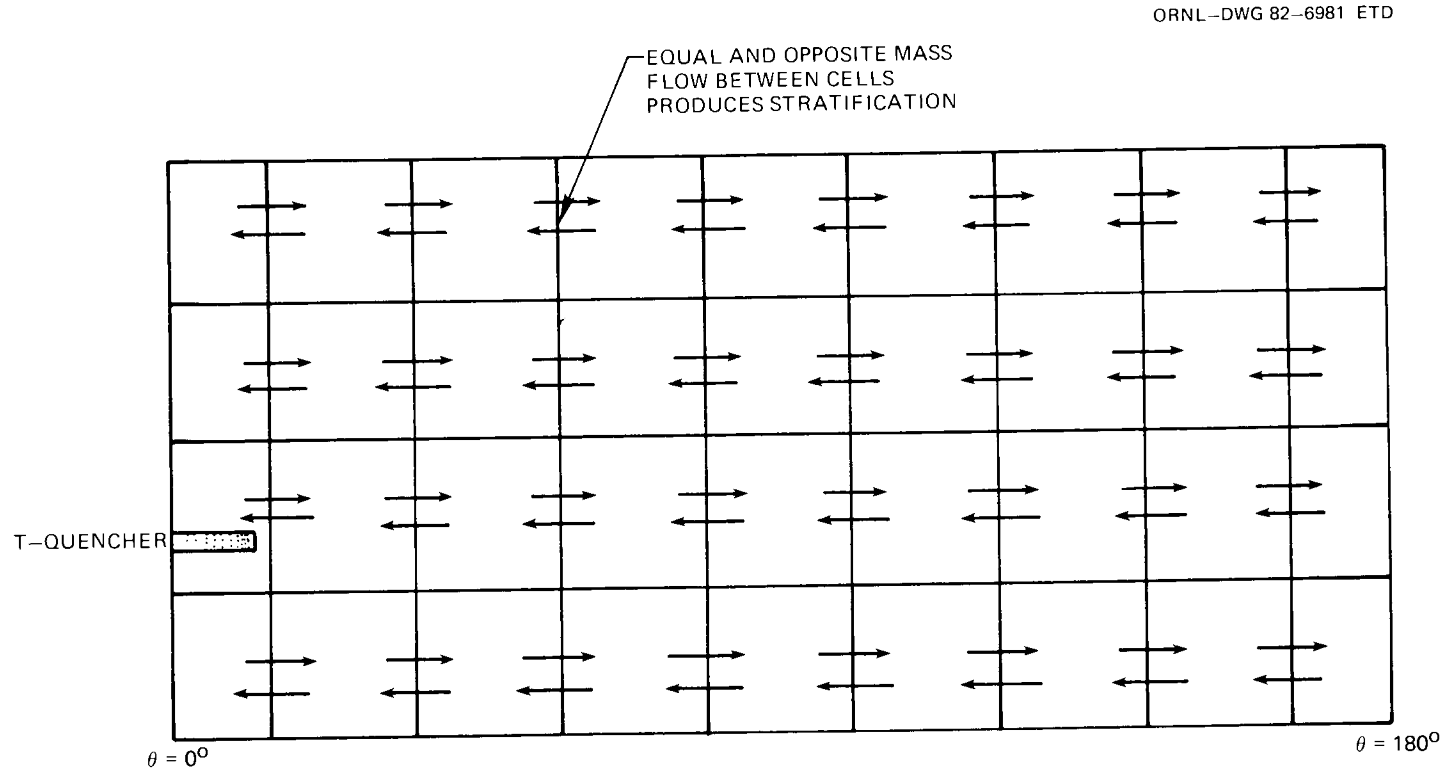

Fig. C.6. Flow field when SRV is not discharging.

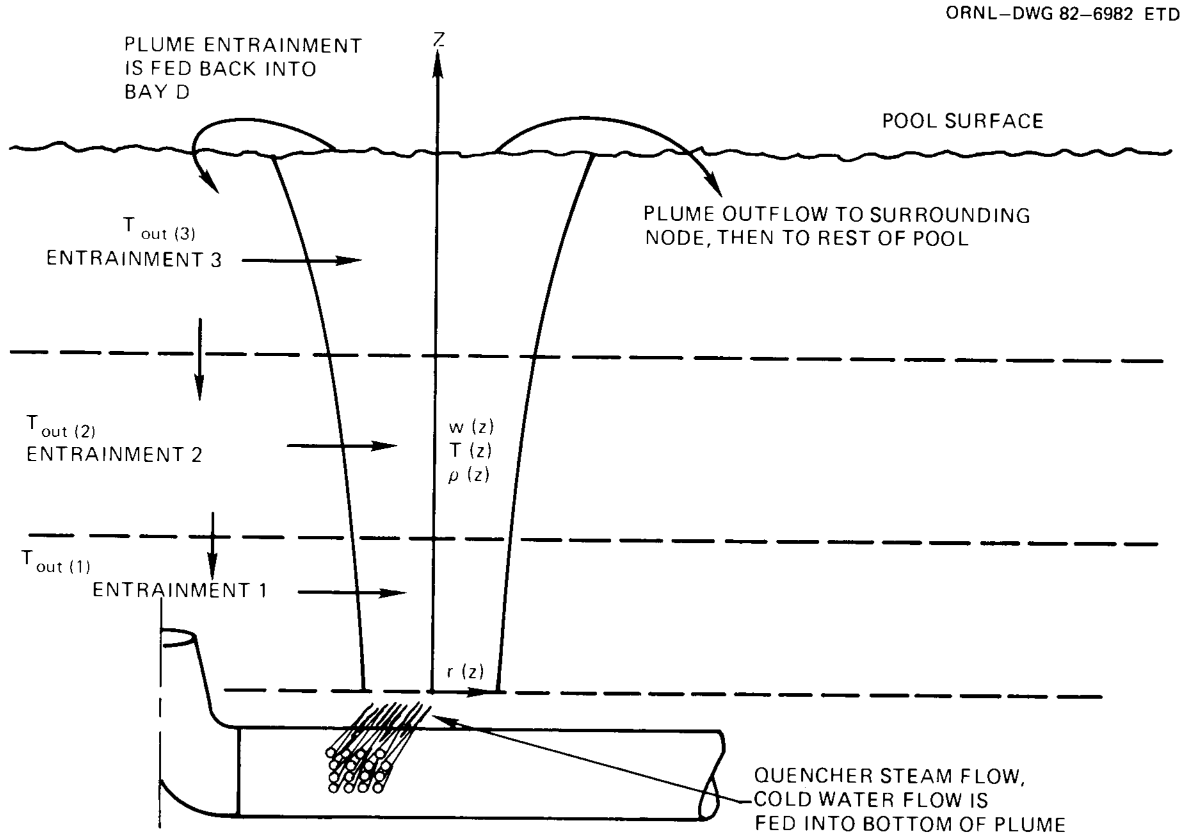

Fig. C.7. Plume transport problem.

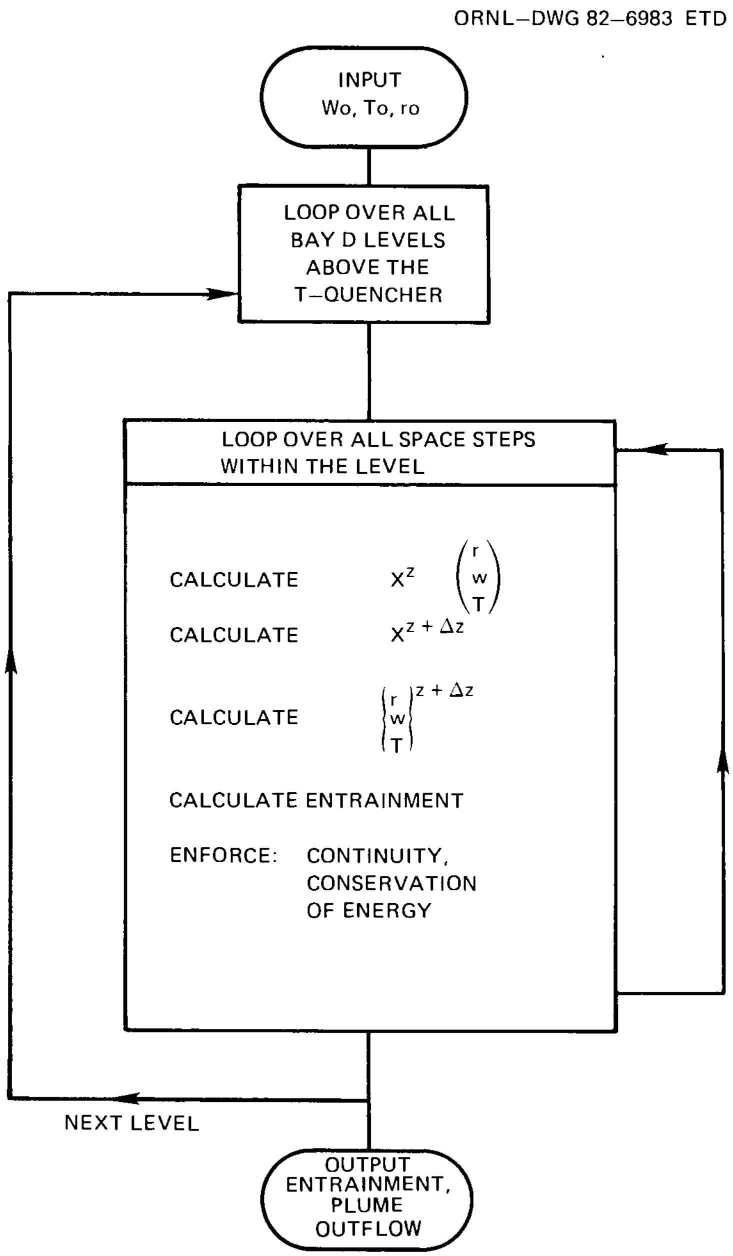

Fig. C.8. Plume transport calculational steps.

Appendix D. MARCH CODE INPUT

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&CHANGE
  TRST = 2600.,
          PRST = 1C00.0,
          CPSTP = 1000.,
          IS = 10,
          IFPM = 10,
          IFPV = 10,
          MEL = -1,
          FDRP = -1.0,
          TMX = -1.0,
          TFX = -1.0,
          IHOTX = 10,
          HIMX = -1.0,
          HIOX = -1.0,
          IGASX = 10,
          WALLX = -1.0,
          PFAIL = -1.0,
          ACBRK = -1.0,
          ID = 0,
          IFISH = -1,
          NCT7 = 1,
          NCRST = 1,
          LST7 = 0,
          IPLOT = 0,
&END
&FPANAL
 TIMEON=9000.0,
&END
&NLMAR
          ITRAN=1,
          IBRK=0,
          ICBRK=1,
          ISPRA=1,
          IECC=2,
          ICE=0,
          NPAIR=0,
          NINTER=18,
          IXPL=0,
          IBURN=0,
          TBURN=0.,
          H2HI=0.,
          H2LO=0.,
          IPDTL=7,
          IPDEF=0,
          IPLOT=3,
          IU=0,
          ICKV=0,
          IFPSM=2,
          IFPSV=2,
   VOLC=278000.0,
          DTINIT=0.02,
```

```
  TIME=2040.0,
          TAP=1.18E6,
&END
&NLINTL
&END
 STEEL  CONCRETE
 DWLINER   DWFLCOR   UPRXPED   LORXPED   WWLINER

&NLSLAB
          NMAT=2,
          NSLAB=5,
          DEN(1)=486.924,157.481,
          HC(1)=0.1137,0.3107,
          TC(1)=25.001,0.881,
          NOD(1)=1,6,17,26,35,
          IVL(1)=1,1,1,1,2,
          IVR(1) = 1,1,1,1,2,
          NNO1(1)=5,11,9,9,5,
          NNO2(1)=0,0,0,0,0,
          MAT1(1)=1,2,2,2,1,
          MAT2(1)=1,2,2,2,1,
          SAREA(1)=18684.0,1640.3,4130.0,1815.0,17050.0,
          HIF(1)=5*0.0,
          DTDX(1)=0.0,0.0,1.0,1.0,0.0,
          X(1)=0.00,0.01,0.03,0.05,0.09375,
          X(6)=0.,0.08,0.25,0.50,0.75,1.0,1.50,2.0,3.0,4.0,4.7326,
          X(17)=0.00,0.10,0.20,0.50,1.146,1.792,2.092,2.192,2.292,
          X(26)=0.0,.25,.75,1.25,1.75,2.25,2.75,3.25,3.49,
          X(35)=0.00,0.0100,0.020,0.040,0.0625,
  TEMP(1)=34*362.,
  TEMP(35)=5*330.,
&END
&NLECC
          PUHIO=0.,
          UHIO=0.,
          PACMO=0.,
          ACMO=0.,
          TMHH=1.E8,
          PHH=1.E6,
          PHLO=100.0,
          WHH1=5000.,
          TMSIS=1.E8,
          PSIS=0.0,
          PSLO=0.,
          WSIS1=0.,
          TMLH=1.E8,
          PLH=1.E6,
          PLLO=50.0,
          WLH1=600.,
          NP = 3,
          TM(1) = 1.E8,  1.E8,  0.,
  STP(1) = 1.E8,  1.E8,  2116.,
          P(1) = 295.,  289.,   1300.,
          WEC(1) = -41429.,   -16346.,   101.8,
          PLO(1) = 0.,   0.,   0.,
          STPHH=1.E8,
```

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            STPLH=1.E8,
            RWSTM=3107859.,
            ECCRC=.01,
            CSPRC=0.,
            DTSUB=-100.0,
            WTCAV=1.0,
            TLHI=0.0,
            TACM=0.,
            TRWST=90.,
            STPECC=1.E8,
&END
&NLECX
&END
&NLCSX
&END
&NLCOOL
            JCOOL=0,
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            CWPR=210000.,
            CTPR=150.,
            CWSR=8587.7,
            CTSR=105.,
            TCOOL=1.E8,
            NCOOL=0,
            QRCOOL=0.,
            PCOOL=0.0,
            POFF=0.0,
&END
&NLMACE
            NCUB=2,
            NRPV1=2,
            NRPV2=1,
            NRPV3=0,
            ICECUB=-1,
  PO=125.,
            FALL=0.,
            HMAX=280.0,
            DTO=0.05,
            DTS=5000.,
   DTPNT=2.,
            IDRY=1,
            IWET=2,
            IBETA=0,
  WPOOL=1.0175200E7,
  TPOOL=340.,
            DCF=1000.0,
   VDRY=533.7,
            VTORUS=257700.,
  WVMAX = 533106.1,
  PRESS(1,2)=1.32,
  PRESS(2,1)=0.5,
            WICE=0.,
            TICE=0.,
            TWTR=0.,
            TWTR2=0.,
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          DCFICE=0.,
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          NSMP2=2,
           WVMAKS=0.,
          NCAV=1,
          VCAV=133.7,
          VFLR=400.,
   FSPRA=1.,
          IVENT=0,
          TVNT1=0.,
          TVNT2=0.,
          AVBRK=0.,
          CVBRK=0.,
          VC(1)=159000.,
  VC(2)=119C00.,
          AREA(1)=1640.3,
  AREA(2)=10980.0,
  HUM(1)=0.265,0.95,
  TEMPO(1)=431.,34C.,
  INERT(1)=1,1,
  N=1,
  NS(1)=                                        2,
  NC(1)=                                        1,
  NT(1)=                                       -7,
  C1(1)=                                    131.7,
  C2(1)=                                     .593,
  C3(1)=                                     10.0,
  C4(1)=                                       0.,
          KT(1,2)=1,KT(2,1)=1,
          STPSPR=1.E8,
&END
&NLBOIL
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          R1 = 1,
          R2 = 10,
          NNT = 45686,
          NR = 44749,
          NDZ = 10,
          ISTR = 3,
          ISG = 0,
   MELMOD = -1,
   IMWA = 1,
   ISTM = 0,
   IHC = 0,
   IHR = 1,
   NDZDRP = 2,
          IMZ = 100,
          FR = 0.0,
          FM = 0.0,
          MWCRNL = 1,
          IFP = 2,
          ISAT = 1,
   IGRID1 = 1,
   IGRID2 = 0,
          KRPS = 0,
          TRPS = 0.0,
```

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          ANSK = 0.0,
          TDK = 0.0,
          YT = 0.0,
          YB = 0.0,
          DTK = 1000.0,
          ICON = 0,
          TMSG1 = 1.0E06,
          TMSG2 = 1.0E06,
TPM = 1.0,
          AB(1)=0.,
          TB(1)=0.,
          TMYBK=0.0,
          YBRK2=1000.,
          TMLEG(1) = 1.0E06,
          WDED = 0.0,
          TPUMP1 = 1.0E06,
          TPUMP2 = 1.0E06,
          QPUMP1 = 0.0,
          QPUMP2 = 0.0,
          TMUP1 = 1.0E06,
          TMUP2 = 1.0E06,
          WMUP1 = 0.0,
          WMUP2 = 0.0,
          HCBOT=18.03,
          VSHEAD=981.4,
          HSHEAD=3.72,
          ASTAND = 42.5,
          HSTAND = 6.81,
          ASEP = 173.8,
          HSEP = 7.73,
TSCT(1) = 1.E6,
TSB(1) = 0.20,
          TALF1 = 1.0E10,
          TALF2 = 1.0E10,
          QZERO = 1.1242E10,
          H = 12.289,
HO = 28.8,
          DC = 15.59,
          ACOR = 108.74,
          ATOT = 261.43,
WATBH = 150516.9,
          D = 0.0424,
          DF = 0.0358,
          DH = 0.0459,
          CLAD = 0.00472,
          XOO = 0.0,
          RHOCU = 81.48,
 HW = 150.0,
TGOO = 553.5,
          CSRV = 0.0,
          TMELT = 4352.0,
          TFUS = 5381.0,
          TFAIL = 2500.,
FDROP=.75,
FCOL = .75,
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 FDCR = -0.50,
         DPART = 0.020833,
         DUO2 = 0.0358,
         FZMCR = 0.05,
         FZOCR = 0.08,
         FZUS1 = 0.10,
 F12 = 0.445,
         WFE2 = 2865.8,
 TFEOO = 553.50,
         WTRSG = 0.0,
         FULSG = 0.0,
         PSG = 0.0,
 PVSL = 1072.,
 TCAV = 653.5,
         YLEG = 16.0,
         ABRK=0.0,
         YBRK=1000.,
 DTPNTB = 5.0,
 DTPN = -3.0
 TPN = 1.0E06,
 VOLP = 21338.4,
 VOLS = 9684.7,
         TMAFW = 1.0E06,
         WAFW = 0.0,
         TAFW = 0.0,
         WCST = 0.0,
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         PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,
         PF(9)=0.9665,0.408,
         VF(1) = 10*0.1,
 TT(1) = 553.50,553.50,553.50,
         CM(1) = 2865.8,9852.3,8712.0,
         AH(1) = 286.6,5085.2,31700.0,
         DD(1) = 0.5,0.5,1.0,
         AR(1) = 286.0,332.9,72.5,
 TT(4) = 553.50,553.50,553.50,
         CM(4) = 2331.0,5259.0,23593.0,
         AH(4) = 400.0,6866.0,687.0,
         DD(4) = 0.17,0.02,0.703,
         AR(4) = 0.0,-7.083,-12.79,
         NVALVE = 13,
         RATFLO = 838900.0,
         RATPRS = 1143.0,
         RATRHO = 2.6083,
         PSET1 = 1105.0,
         TSET2 = 1.0E8,
         PSET2 = 0.0,
 WATMAS=382589.2,
&END
&NLDP
         TDP = 1.0E8,
         NUMVAL = 1,
         PSETDP = 0.0,
&END
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&NLHEAD
            WZRC = 144381.7,
            WFEC = 26980.0,
            WUO2 = 351439.9,
            WGRID = 66750.0,
            WHEAD = 207500.0,
            TMLT = 4135.0,
            DBH = 20.915,
            THICK = 0.7031,
            COND = 6.39,
  E1 = 0.0,
  E2 = 0.0,
  FOPEN = 0.0,
&END
&NLHOT
            IHOT = 100,
            MWR = 1,
            DP = 0.25,
            CCN = 6.39,
            FLRMC = 3360.0,
            WTR = 0.0,
            TPOOLH = 100.0,
            NSTOP = 200,
&END
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            DENSC = 2.375,
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            FC1 = 0.455,
            FC2 = 0.070,
            FC3 = 0.388,
            FC4 = 0.048,
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            RO = 322.6,
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  DPRIN = 300.0,
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            EPSI(1) = 0.5,0.5,
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            IGAS = 1,
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Appendix E: ACRONYMS AND SYMBOLS

ADS	Automatic Depressurization System
ANS	American Nuclear Society
ANSI	American National Standards Institute
BAF	Bottom of Active Fuel
BCL	Battelle Columbus Laboratories
BFNP	Browns Ferry Nuclear Plant
BFNP#1	Browns Ferry Nuclear Plant Unit One
BWR	Boiling Water Reactor
CBP	Condensate Booster Pump
CILRT	Containment Integrated Leak Rate Test
CP	Condensate Pump
CRD	Control Rod Drive
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
EOI	Emergency Operating Instruction
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
GPM	Gallons Per Minute
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
ID	Internal Diameter
INEL	Idaho National Engineering Laboratory
INTER	Core-concrete interaction subroutine of the MARCH code
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
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
ORNL	Oak Ridge National Laboratory
Pa	Pascal
PCV	Pressure Control Valve
PCIS	Primary Containment and Reactor Vessel Isolation Control System
PCS	Power Conversion System
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
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup System
SASA	Severe Accident Sequence Analysis
SBCS	Standby Coolant Supply System
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
SNL	Sandia National Laboratories
SRV	Safety Relief Valve
TAF	Top of Active Fuel
TIP	Traveling Incore Probe
TVA	Tennessee Valley Authority
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 loss of decay heat removal (DHR) capability following scram from full power with the power conversion system unavailable. In accident sequences without DHR capability, the residual heat removal (RHR) system functions of pressure suppression pool cooling and reactor vessel shutdown cooling are unavailable. Consequently, all decay heat energy is stored in the pressure suppression pool with a concomitant increase in pool temperature and primary containment pressure. With the assumption that DHR capability is not regained during the lengthy course of this accident sequence, the containment ultimately fails by overpressurization. Although unlikely, this catastrophic failure might lead to loss of the ability to inject cooling water into the reactor vessel, causing subsequent core uncovery and meltdown. The timing of these events and the effective mitigating actions that might be taken by the operator are discussed in this report.

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