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Station Blackout at Browns Ferry Unit One—Accident Sequence Analysis

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Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
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STATION BLACKOUT AT BROWNS FERRY UNIT ONE — ACCIDENT SEQUENCE ANALYSIS

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SUMMARY

This study describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to a hypothetical Station Blackout. This accident would be initiated by a loss of offsite power concurrent with a failure of all eight of the onsite diesel-generators to start and load; the only remaining electrical power at this three-unit plant would be that derived from the station batteries. It is assumed that the Station Blackout occurs at a time when each of the Browns Ferry units is operating at 100% power so that there is no opportunity for use of the batteries for units 2 or 3 in support of unit 1.

The design basis for the 250 volt DC battery system at Browns Ferry provides that any two of the three unit batteries can supply the electrical power necessary for shutdown and cooldown of all three units for a period of 30 minutes with a design basis accident at any one of the units. It is further provided that the system voltage at the end of the 30 minute period will not be less than 210 volts, and that all DC equipment supplied by this system must be operable at potentials as low as 200 volts.

It is clear that the 250 volt system was not designed for the case of a prolonged Station Blackout, and the period of time during which the DC equipment powered by this system could remain operational under these conditions can only be estimated. It would certainly be significantly longer than 30 minutes since all three batteries would be available, the equipment is certified to be operable at 200 volts, and there would be no design basis accident. In response to AEC inquiry in 1971, during the period of plant construction, TVA estimated that the steam-driven High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, which use DC power for turbine control and valve operation, could remain operational for a period of four to six hours. A period of four hours has been assumed for this study.*

Within 30 seconds following the inception of a Station Blackout, the reactor would have scrammed and the reactor vessel would be isolated behind the closed main steam isolation valves (MSIV's). The initial phase of the Station Blackout extends from the time of reactor vessel isolation until the time at which the 250 volt DC system fails due to battery exhaustion. During this period, the operator would maintain reactor vessel water level in the normal operating range by intermittent operation of the RCIC system, with the HPCI system available as a backup. Each of these water-injection systems is normally aligned to pump water from the condensate storage tank into the reactor vessel via a feedwater line.

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*It should be noted that the events subsequent to failure of the 250 volt DC system are relatively insensitive to the time at which this failure occurs. This is because the only parameter affecting the subsequent events which is dependent upon the time of failure is the slowly-varying decay heat.

As part of a requested review of the results of this study, TVA performed a battery capacity calculation which shows that the unit batteries can be expected to last as long as seven hours under blackout conditions. In Appendix G, the effect of a battery lifetime of seven vice four hours upon study results is shown to be limited to timing.

The operator would also take action during the initial phase to control reactor vessel pressure by means of remote-manual operation of the primary relief valves; sufficient stored control air would remain available to permit the desired remote-manual valve operations for well over four hours. The Control Room instrumentation necessary to monitor reactor vessel level and pressure and for operation of the RCIC and HPCI systems would also remain available during this period.

There is no Emergency Operating Instruction for a Station Blackout at Browns Ferry. However, the existing written procedure for operator action following reactor isolation behind closed MSIV's provides for the reactor vessel level control and pressure control described above. The primary relief valves would actuate automatically to prevent vessel over pressurization if the operator did not act; the purpose of pressure control by remote-manual operation is to reduce the total number of valve actuations by means of an increased pressure reduction per valve operation and to permit the steam entering the pressure suppression pool to be passed by different relief valves in succession. This provides a more even spacial distribution of the transferred energy around the circumference of the pressure suppression pool.

The initial phase of a Station Blackout has been analyzed in this study by use of a relatively simple computer code developed specifically for this purpose. This coding uses the Continuous Systems Modeling Program (CSMP) language of the IBM computer system to simulate the response of Browns Ferry Unit 1 to postulated operator actions. The analysis shows, because of the loss of the drywell coolers, that it is necessary for the operator to begin to reduce the reactor vessel pressure to about 0.791 MPa (100 psig) within one hour of the inception of the Station Blackout. This depressurization reduces the temperature of the saturated fluid within the reactor vessel and thereby decreases the driving potential for heat transfer into the drywell, yet keeps the vessel pressure high enough for continued operation of the RCIC system steam turbine. With this action, the drywell average ambient temperature can be kept below 149°C (300°F) throughout the initial phase of a Station Blackout; tests have shown that both the drywell structure and the equipment located therein can be expected to survive temperatures of this magnitude.

The analysis also reveals an important second reason for operator action to depressurize the reactor vesel early in the initial phase of a Station Blackout. This depressurization removes a great deal of steam and the associated stored energy from the reactor vessel at a time when the RCIC system is available to inject replacement water from the condensate storage tank and thereby maintain the reactor vessel level. Subsequently, when water injection capability is lost for any reason, remote-manual relief valve operation would be terminated and there would be no further water loss from the reactor vessel until the pressure has been restored to the setpoint [7.72 MPa (1105 psig)] for automatic relief valve actuation. Because of the large amount of water to be reheated and the reduced level of decay heat, this repressurization would require a significant period of time. In addition, the subsequent boiloff* would begin from a very high

*The term "boiloff" is used to signify a monotonic decrease in reactor vessel water level due to intermittent loss of fluid through the primary relief valves without replacement.

vessel level because of the increase in the specific volume of the water as it is heated and repressurized. Thus, an early depressurization will provide a significant period of valuable additional time for preparative and possible corrective action before core uncovery after injection capability is lost.

One design feature of the HPCI system logic was brought into question by the analysis. The existing logic provides for the suction of the HPCI system pump to be automatically shifted from the condensate storage tank to the pressure suppression pool upon high sensed suppression pool level. During a Station Blackout, this would occur after about three hours when the average suppression pool temperature has reached about 71°C (160°F).* Since the lubricating oil for the HPCI turbine is cooled by the water being pumped, this would threaten the viability of the HPCI system.

The rationale for the automatic shift in HPCI pump suction on high sensed pool level is not explained in the literature available to this study. There is no corresponding provision for a shift of the RCIC pump suction on high pool level, and a separate logic is provided for an automatic shift of the HPCI pump suction should the supply of water from the condensate storage tank become exhausted. For these reasons, it is recommended that the desirabilty of the automatic shift of HPCI pump suction on high sensed suppression pool level be reexamined.

The plant response during the initial phase of a Station Blackout can be summarized as an open cycle. Water would be pumped from the condensate storage tank into the reactor vessel by the RCIC system as necessary to maintain level in the normal operating range. The injected water would be heated by the reactor decay heat and subsequently passed to the pressure suppression pool as steam when the operator remote-manually opens the relief valves as necessary to maintain the desired reactor vessel pressure. Stable reactor vessel level and pressure control is maintained during this period, but the condensate storage tank is being depleted and both the level and temperature of the pressure suppression pool are increasing. However, without question, the limiting factor for continued removal of decay heat and the prevention of core uncovery is the available of DC power.

The sequence of events used for the fission product release analysis for a prolonged Station Blackout was established by this study under the assumption that no independent secondary equipment failures would occur. Dependent secondary failures, i.e., those caused by the conditions of a Station Blackout, were included in the development of the sequence, but with the assumption that the operator does take action to depressurize the reactor vessel and thereby prevent drywell temperatures of the magnitude that could severely damage the equipment therein. The point is important because the operator would probably be reluctant to depressurize; current training stresses concern for high suppression pool temperatures (based on LOCA considerations†) and the operator would recognize that suppression pool cooling is not available during a Station Blackout.

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*It is expected that any accident sequence resulting in high suppression pool level would also produce an associated high pool temperature.

†There is currently no written procedure for the case of Station Blackout.

This study has established that the possible dependent secondary failure of a stuck-open relief valve would not reduce the reactor vessel pressure below that necessary for operation of the RCIC steam turbine during the period in which DC power would remain available. However, during this period all of the energy transferred to the pressure suppression pool would pass through the tailpipe of this one relief valve and would be concentrated near its terminus. This would reduce the steam-quenching capacity of the pool to that provided by the effective volume of water surrounding this one tailpipe. Local boiling in this area would pass most of the relief valve discharge directly to the pool surface and could result in early containment failure by overpressurization.

The question of suppression pool effectiveness in quenching relief valve steam discharge in cases which involve highly localized energy transfer from the reactor vessel is important to a complete analysis of any severe accident in which the pressure suppression pool is used as a heat sink. This question has not been resolved, at least in the non-proprietary literature, and has been recently adopted as a dissertation topic for a University of Tennessee doctorial candidate working with the Severe Accident Sequence Analysis (SASA) project. Pending the results of this work, pool-averaged temperatures are used in this study.

The MARCH code has been used in support of the analysis of the second phase of a Station Blackout, i.e., the period after the 250 volt DC system fails because of battery exhaustion. The existing versions of MARCH are too crude to permit modeling plant response to a series of postulated operator actions such as those previously discussed for the initial phase of a Station Blackout. Therefore, in the event sequence modeled by the MARCH code, the reactor vessel remains pressurized with pressure control by automatic relief valve actuation and level control by automatic operation of the HPCI system. As before, averaged suppression pool temperatures are used, and it is assumed that injection capability is lost after four hours, when the unit battery is exhausted.

In the MARCH event sequence, the reactor vessel water level is in the normal operating range and the vessel is pressurized at the four-hour point when boiloff begins due to loss of injection capability.* The MARCH results predict core uncovery 62 minutes after the beginning of boiloff, followed by the inception of core melting 53 minutes later. The model provides that the melted core slumps down to the bottom of the reactor vessel and this results in a predicted failure of the reactor vessel bottom head at approximately three hours after injection capability is lost. The subsequent breaching of the primary containment because of failure of the electrical penetration modules by overtemperature is predicted at about four and one-half hours after the inception of boiloff. The detailed

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*These conditions at the beginning of boiloff are similar to those predicted by the previously discussed sequence which models the plant response to operator actions during the initial phase because in that sequence, the reactor vessel would have to repressurize before boiloff could begin. The difference is in the timing. If injection capability were lost at the four-hour point, the boiloff would begin immediately if the vessel is pressurized, but would be delayed if the vessel were depressurized.

thermo-hydraulic parameters needed for the analysis of fission product release from the fuel rods and the subsequent transport of fission products to the environment were taken from the MARCH results for this sequence.

An estimate of the magnitude and timing of the release of the noble gas, cesium, and iodine-based fission products to the environment is provided in Volume 2 of this study. Under the conditions of a Station Blackout, fuel rod cladding failure would occur while the reactor vessel was pressurized. The distribution of both solid and gaseous fission products within the Browns Ferry Unit 1 core at the present time has been obtained through use of the ORIGEN2 code; the results show that internal gas pressure under Station Blackout conditions could not increase to the magnitude necessary to cause cladding failure by rod burst. Accordingly, the cladding has been assumed to fail at a temperature of 1300°C.

STATION BLACKOUT AT BROWNS FERRY UNIT ONE —

ACCIDENT SEQUENCE ANALYSIS

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ABSTRACT

This study describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to Station Blackout, defined as a loss of offsite power combined with failure of all onsite emergency diesel-generators to start and load. Every effort has been made to employ the most realistic assumptions during the process of defining the sequence of events for this hypothetical accident. DC power is assumed to remain available from the unit batteries during the initial phase and the operator actions and corresponding events during this period are described using results provided by an analysis code developed specifically for this purpose. The Station Blackout is assumed to persist beyond the point of battery exhaustion and the events during this second phase of the accident in which DC power would be unavailable were determined through use of the MARCH code. Without DC power, cooling water could no longer be injected into the reactor vessel and the events of the second phase include core meltdown and subsequent containment failure. An estimate of the magnitude and timing of the concomitant release of the noble gas, cesium, and iodine-based fission products to the environment is provided in Volume 2 of this report.

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1. INTRODUCTION

The Tennessee Valley Authority (TVA) operates three nearly identical reactor units at the Browns Ferry Nuclear Plant located on the Tennessee River approximately midway between Athens and Decatur, Alabama. The General Electric Company and the Tennessee Valley Authority jointly participated in the design; TVA performed the construction of each unit. Unit 1 began commercial operation in August 1974, Unit 2 in March 1975, and Unit 3 in March 1977.

Each unit comprises a boiling water reactor steam supply system furnished by the General Electric Company. Each reactor is designed for a power output of 3440 MW_t, for a corresponding generated 1152 MW_e; the maximum power authorized by the operating license is 3293 MW_t, or 1067 net MW_e. The primary containments are of the Mark I pressure suppression pool type; the drywells are light bulb shape and both torus and drywell are of steel vessel construction. The three units share a reactor building-secondary containment of the controlled leakage, elevated release type.

Safety systems for each unit include a Reactor Protection System, a Standby Liquid Control System for Poison injection, and the Emergency Core Cooling Systems: High-Pressure Coolant Injection (HPCI), Automatic Depressurization (ADS), Residual Heat Removal (RHR), and Core Spray (CS). The Reactor Core Isolation Cooling (RCIC) system is also provided for the removal of post-shutdown reactor decay heat as a consequence limiting system.

Several components and systems are shared by the three Browns Ferry units. A complete description of these shared features is given in the Final Safety Analysis Report;[1] the shared Safeguards systems and their supporting auxiliary equipment are listed in Table 1.1. With the assumption that the interfaces with the other two units do not interfere with the operation of any shared system as applied to the needs of the unit under study, the existence of the shared systems does not significantly complicate the analysis of an accident sequence at any one unit.

The results of a study of the consequences at Unit 1 of a Station Blackout (loss of all AC power) at the Browns Ferry Nuclear Plant are presented in this report. Section 2 provides a description of the event and discussion of the motivation for consideration of this event. The normal recovery from a Station Blackout is described in Sect. 3, the computer model used for the normal recovery analysis is discussed in Sect. 4, and the instrumentation available to the operator is described in Sect. 5. The actions which the operator should take to prolong the period of decay heat removal are discussed in Sect. 6, and computer predictions of the behavior of the thermal-hydraulics parameters during the period when a normal recovery is possible are displayed in Sect. 7.

A Severe Accident by definition proceeds through core uncovery, core meltdown, and the release of fission products to the surrounding atmosphere. The equipment failures which have the potential to extend a Station Blackout into a Severe Accident are discussed in Sect. 8, and the Severe Accident sequences which would follow during a prolonged Station Blackout are presented in Sect. 9. The consequences of each of these sequences after the core is uncovered differ only in the timing of events; the actions which might be taken by the operator to mitigate the consequences of the Severe Accident are discussed in Sect. 10. The instrumentation available following the loss of injection capability during the period in which severe core damage occurs is described in Sect. 11.

The conclusions of this Station Blackout analysis and the implications of the results are discussed in Sect. 12. This includes consideration of the available instrumentation, the level of operator training, the existing emergency procedures, and the overall system design.

Appendix A contains a listing of the computer program developed to model operator actions and the associated system response during the period when normal recovery is possible. The MARCH code was used for analyses of the severe accident sequences; the modifications made to this code are described in Appendix B and an input listing is provided in Appendix C.

The pressure suppression pool is the key to the safe removal of decay heat from an isolated Boiling Water Reactor, but no satisfactory method

Table 1.1 Shared safeguards systems and their auxiliary support systems

System	Quantity	Function
Standby AC Power Supply System	Four diesel generators each coupled as an alternate source of power to four independent shutdown boards for Units 1 and 2. Four additional diesel generators serve as alternate power sources to four Unit 3 shutdown boards.	Supply emergency power during loss of offsite power conditions
250V DC Power Supply System	Three batteries, one per unit. The 250V DC power supplies for the DC-powered redundant services of each unit are normally derived from separate batteries. A fourth battery is provided for common station service.	Supply DC power when battery chargers are not functioning
Control Rod Drive Hydraulic System	A spare control rod drive hydraulic pump is shared between Units 1 and 2.	Provide driving force for normal rod movement
Reactor Building Closed Cooling Water System	Two pumps and two heat exchangers per unit with one common spare pump and heat exchanger for all three units	Provide cooling water to reactor auxiliary equipment
Gaseous Radwaste System	System is unitized except for a common discharge plenum, ducting, and the 600-ft stack	Obtain elevated discharge from the Standby Gas Treatment System
Station Drainage System	One common drain header into which each of the three unit reactor building floor drainage pumps discharge	Pass reactor building drainage to receiver tanks
Standby Coolant Supply System	Completely shared system	Provide means for core cooling following a complete failure of the Residual Heat Removal (RHR) cooling complex
RHR Service Water System	Completely shared system	Provide an assured heat sink for long-term removal of decay heat when the main condensers are not available for any reason
Emergency Equipment Cooling Water System	Completely shared system	Distribute raw cooling water to equipment and auxiliary systems which are required for shutdown of all three units
Standby Gas Treatment System	Completely shared system used only when an abnormal activity release occurs	Filter and exhaust the air from a unit zone and the refueling zone, the refueling zone only, or the entire secondary containment

for the analysis of the response of the torus and pool under severe accident conditions currently exists, at least in the non-proprietary literature. The nature of the problem and the measures being taken at ORNL to improve the analysis capability are discussed in Appendix D.

In the event of a Station Blackout, the reactor vessel will be isolated behind the closed Main Steam Isolation Valves (MSIVs) with pressure maintained by periodic blowdowns through the relief valves to the pressure suppression pool. Makeup water to maintain the reactor vessel level must be injected either by the High Pressure Coolant Injection (HPCI) or the Reactor Core Isolation Cooling (RCIC) systems. The operation of these important systems is discussed in Appendices E and F.

As a portion of their review of the draft of this report, the Electrical Engineering Branch at TVA performed a battery capacity calculation which shows that the unit batteries at the Browns Ferry plant can be expected to provide power for as long as seven hours under station blackout conditions. Since a battery life of four hours was assumed for the calculations of this work, a new Appendix G has been added in which all of the possible failure modes other than battery exhaustion that were considered in Sections 3, 7, and 8 are re-examined for applicability to the period between four and seven hours after the inception of a Station Blackout.

The magnitude and timing of the release of the fission product noble gases and various forms of iodine to the atmosphere are discussed in Volume 2. This includes the development of release rate coefficients for the escape of these fission products from fuel as a function of temperature, specification of the various chemical forms, determination of the optimum strategy for the modeling of precursor/daughter exchange, and the study of the particular auxiliary systems involved to determine the applicable fission product release pathways. These results are incorporated into a vehicle for the calculation of the transport of the individual fission products from the reactor core, through the primary and secondary containments to the atmosphere, control volume by control volume.

The primary sources of information used in the preparation of this report were the Browns Ferry Nuclear Plant (BFNP) Final Safety Analysis Report (FSAR), the Nuclear Regulatory Commission 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*. Additional information was gathered by means of one visit to the BFNP and several visits to the TVA Power Operations Training Center at Soddy-Daisy, Tennessee. The excellent cooperation and assistance of Tennessee Valley Authority personnel in the gathering of information necessary to this study are gratefully acknowledged.

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*The setpoints for automatic action used in this study are the safety limits as given in the FSAR. 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.

2. DESCRIPTION OF STATION BLACKOUT

A Station Blackout is defined as the complete loss of AC electrical power to the essential and nonessential switchgear buses in a nuclear power plant.[2] At the Browns Ferry Nuclear Plant, a Station Blackout would be caused by a loss of offsite power concurrent with the tripping of the turbine generators and a subsequent failure of all onsite diesel generators to start and load. After these events, the only remaining sources of electrical power would be the battery-supplied 250 volt, 48 volt, and 24 volt DC electrical distribution systems; AC power would be limited to the instrumentation and control circuits derived from the feedwater inverter or the unit-preferred and plant-preferred motor- generator sets which are driven by the DC systems.

The reliability of offsite power at Browns Ferry has been excellent. There are two independent and separated sources of 161 KV offsite power to the plant, each from a different nearby hydroelectric station. In the near future, offsite power will become even more reliable. System modifications will permit any of the three reactor units at Browns Ferry, in the event of a generator trip, to receive reverse power from the TVA 500 KVA grid which these units normally feed.

However, should all offsite power be lost together with the tripping of the turbine generators, there remain eight diesel generators at Browns Ferry which are designed to automatically start and load whenever normal AC power is lost. By design, all equipment required for the safe shutdown and cooldown of the three Browns Ferry units can be powered by six* of these diesel units, even with the assumption of a design basis accident on any one unit.[3]

Therefore, a Station Blackout is an extremely unlikely event. Nevertheless, the consequences of a Station Blackout should be studied. It is important to recall that one Browns Ferry unit suffered a loss of electrical power to all of its emergency core-cooling systems during the March 22, 1975 electrical cable tray fire.[4] To determine if Browns Ferry and similar nuclear plants can recover from the loss of other combinations of AC powered machinery, it will prove most efficient to first consider the plant response to a loss of all AC powered equipment, as in a Station Blackout.

This study considers the effect on Unit 1 at Browns Ferry of a Station Blackout which begins with a loss of offsite power, associated turbine generator trip, and failure of all diesels to start. Barring further equipment failures, the battery powered systems have the capability to supply the power necessary to operate the high-pressure water injection systems to maintain the reactor in a stable cooled state for several hours. The first seven sections of this report pertain to the methods which can be used to prolong this period of water injection capability and decay heat removal for as long as possible.

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*With operator action, any six diesel generators would be satisfactory. However, for adequate short term shutdown and cooldown response without operator action, the six operable diesel generators would have to comprise three of the four provided for the Unit 1/Unit 2 complex and three of the four provided for Unit 3.

However, injection capability would ultimately be lost during an extended Station Blackout, either through exhaustion of the installed battery capacity or earlier, through secondary failures of vital equipment. Once injection capability is lost, the Station Blackout would become a Severe Accident with inevitable core uncovery. The remainder of this report proceeds from an assumption that the Blackout does develop into a Severe Accident, involving core meltdown and subsequent fission product release to the atmosphere.

3. NORMAL RECOVERY

The ability to inject cooling water into an isolated reactor vessel for a significant period of time under Station Blackout conditions at Browns Ferry is provided by the Reactor Core Isolation Cooling (RCIC) and the High Pressure Coolant Injection (HPCI) systems. A normal recovery from a Station Blackout is defined as the restoration of AC power prior to the loss of these cooling water injection systems; no core damage is involved, because as long as the RCIC or HPCI system is operable, the reactor vessel water level can be controlled within normal operating limits. The control of water level and other aspects of the response of Browns Ferry Unit 1 during the period of a Station Blackout in which a normal recovery is possible are discussed in this section. Equipment failures which might ultimately lead to loss of injection capability are discussed in Section 8 and the sequence of events following this loss is discussed in Sections 9 through 11.

A Station Blackout at Browns Ferry would be initiated by a total loss of offsite power and the concomitant reactor scrams, turbine-generator trips, and Main Steam Isolation Valve (MSIV) closures. At each of the three units, the reactor would be shut down and isolated behind the closed MSIV's within 30 seconds. If the on-site diesel-generators fail to successfully start and load, the conditions for a complete Station Blackout are established. The remainder of this discussion will concern the subsequent events at Unit 1 under Station Blackout conditions.

Immediately following MSIV closure, decay heat generation will cause the reactor vessel pressure to increase to the setpoints of as many relief valves as are required to terminate the pressure increase. The affected relief valves would open to pass steam and the associated energy directly from the reactor vessel to a terminus near the bottom of the pressure suppression pool. The emerging steam would be condensed by the mechanism of heat transfer to the suppression pool water; the temperature of the pool water would begin a monotonic increase since there would be no means for suppression pool cooling under Station Blackout conditions.

The events up to this point would have all occurred automatically, leaving the reactor vessel isolated behind the closed MSIV's while decay heat generation adds energy which must be removed. The two major considerations requiring immediate attention are the means of reactor vessel level control and pressure control.

Level Control. Two independent systems, HPCI and RCIC, are provided for water injection into an isolated reactor vessel at high pressure. Both require only DC electrical power for operation and comprise steam-turbine driven pumps; the steam is taken from the main steam piping upstream of the MSIV's and the turbine exhaust steam is discharged into the pressure suppression pool. Both systems are normally aligned for the pumping of water from the condensate storage tank into the reactor vessel via a connection into a feedwater line, and are automatically initiated if the vessel water level decreases to 12.1 m (476 in.) above the bottom of the vessel. This automatic initiation point is 2.16 m (85 in.) below the normal operating level of 14.25 m (561 in.) and 2.95 m (116 in.) above the top of the active fuel in the core. The turbines in both systems will trip if the vessel water level rises to 14.78 m (582 in.).

The HPCI system has an injection capacity of 0.315 m^3/s (5000 GPM), and is capable of cycling the reactor vessel water level over the 2.69 m (106 in.) range between the HPCI automatic initiation point and the HPCI turbine trip point without operator action. The design and operation of this important system are described in detail in Appendix E.

The RCIC system has an injection capacity of 0.038 m^3/s (600 GPM). Operator action is required for long-term level control using this system since once the turbine has been tripped, it will not automatically restart. This important system is discussed in detail in Appendix F.

Both of these high-pressure injection systems will isolate, i.e., their steam supply valves will automatically shut if the reactor vessel pressure becomes too low to permit turbine operation. The purpose is to prevent excessive leakage from the seals of an immobile turbine; the HPCI setpoint is 0.793 MPa (100 psig), while the RCIC system isolates at 0.448 MPa (50 psig). These systems will also isolate if the ambient temperature in the vicinity of their turbines reaches 93.3°C (200°F); this is to protect against a steam leak in the system piping.

Immediately following the inception of the Station Blackout, the core water void collapse caused by both the scram and the pressure increase following MSIV closure would result in a rapid drop in reactor vessel water level. The level would decrease to a point beneath the range of level indication available in the Control Room under Station Blackout Conditions [13.41 to 14.94 m (528 to 588 in.) above the bottom of the vessel]. Standard Browns Ferry Emergency Operating Instructions for immediate action following reactor scram and MSIV closure call for the operator to manually initiate both the HPCI and the RCIC systems. When the level has been restored into the indicating range, the operator would turn off the HPCI system. Subsequently the operator would be able to maintain the reactor vessel water level by intermittent remote-manual operation of the RCIC system alone, as illustrated in Section 7.

For this analysis of the sequence of events during a Station Blackout, it is intended to employ the most realistic assumptions concerning equipment operation and operator actions. Accordingly, it will be assumed that the operator does take the actions described above to control level over the long term using the RCIC system. However, it should be recalled that the HPCI system is capable of automatic level control should the operator fail to act.

The condensate storage tank contains enough stored water to replace that lost from the reactor vessel in the form of steam passed to the pressure suppression pool for well over eight hours. The tank capacity is 1419.4 m^3 (375,000 gallons) of which 511.0 m^3 (135,000 gallons) is a guaranteed reserve for the low pressure emergency cooling systems (which are inoperable during a Station Blackout) and for the HPCI and RCIC systems. This reserve is guaranteed because the pumps of these emergency injection systems take suction on the bottom of the condensate storage tank whereas all other demand is taken from a standpipe within the tank; there is a 511.0 m^3 (135,000 gallon) capacity below the standpipe entrance.

As discussed in Appendices E and F, the pump suctions for each of the two high-pressure injection systems can be remote-manually shifted to the pressure suppression pool by the Control Room operator. This would be useful if the condensate storage tank source failed for any reason during about the first three hours of a Station Blackout. After this, the pressure suppression pool water would be so hot (as illustrated in Section 7) that the RCIC or HPCI system turbine lubricating oil, which is cooled by the water being pumped, would overheat; this would significantly threaten the continued operation of these systems.

Pressure Control. Overpressure protection for the isolated reactor vessel is provided by thirteen two-stage Target Rock primary relief valves, with a roughly even circumferential distribution of the tailpipe discharges into the pressure suppression pool. These valves are located on the main steam lines upstream of the MSIV's and are capable of automatic actuation requiring no external source of power other than reactor vessel steam pressure. Four of these valves are set for automatic actuation at 7.722 MPa (1105 psig), four are set for 7.791 MPa (1115 psig), and the remaining five are set for 7.860 MPa (1125 psig). After automatic actuation, each valve will reseat following a 0.345 MPa (50 psi) drop in vessel pressure.

Each of the primary relief valves can also be remote-manually operated from the Control Room. This requires the availability of pressurized control air for physical operation of the relief valve as well as the availability of DC power for actuation of the solenoid valve which opens to admit the control air to the relief valve operator.

The pressurized air for remote-manual relief valve operation is provided by the Drywell Control Air system. The two air compressors in this system normally operate intermittently as necessary to maintain the pressure in two 1.61 m^3 (57 ft^3) receiver tanks between the limits of 0.689 and 0.793 MPa (85 and 100 psig). Under the conditions of a Station Blackout, the air compressors would be inoperable but there would be an initial supply of stored pressurized air available in the receiver tanks.

Six of the primary relief valves are associated with the Automatic Depressurization System (ADS). (This system, which is designed to automatically reduce the reactor vessel pressure during a LOCA so that the high-capacity, low-pressure injection systems can operate would not be functional during a Station Blackout.) For improved ADS system reliability, each of the six ADS relief valves is fitted with an individual air accumulator sized to permit five valve operations without replenishment.

Therefore, under Station Blackout conditions, an initial supply of stored control air is provided for all thirteen primary relief valves by the drywell control air system receivers. An additional special stored supply is provided for the six valves associated with the ADS system; this special air supply alone will permit 30 valve operations, i.e., five remote-manual operations per ADS valve.

For pressure control of an isolated reactor vessel, standard procedure is for the operator to remote-manually operate the primary relief valves in succession as necessary to cycle the vessel pressure between about 7.688 and 6.309 MPa (1100 and 900 psig). This avoids the automatic actuation of the relief valves [lowest setpoint: 7.722 MPa (1105 psig)] and greatly reduces the total number of relief valve actuations since the pressure reduction associated with an automatic valve actuation is only 0.345 MPa (50 psi). Lowering the number of actuations decreases the opportunity for a relief valve to stick open, and permits the operator to evenly distribute the decay heat energy transferred to the pressure suppression pool.

As in the case of reactor vessel level control, it is intended that the most realistic assumptions concerning pressure control be used in this study. Accordingly, it will be assumed that the operator does take the actions in regard to pressure control as described above. However, it should be noted that without operator action, the pressure would cycle between the limits of 7.377 and 7.722 MPa (1055 and 1105 psig); unfortunately, it is probable that this would occur through the repeated operation of the same relief valve causing severe localized heating in the vicinity of the tailpipe discharge in the suppression pool.

Short-term Plant Status. Within minutes after the inception of a Station Blackout, both reactor vessel level control and pressure control will be achieved with or without operator action. An open cycle will exist, with water pumped from the condensate storage tank into the reactor vessel, converted to steam there by reactor decay heat, and subsequently passed to the pressure suppression pool and condensed. Satisfactory reactor cooling is provided, but both the pressure suppression pool level and temperature are increasing, as is the drywell ambient temperature due to loss of the drywell coolers. The reserve DC power stored in the batteries is being dissipated.

There is a 250V DC battery for each unit at Browns Ferry, plus a station battery for common loads. Each unit battery is designed to provide a guaranteed supply of direct current for the first 30 minutes of a design basis accident. Under the less severe conditions of a Station Blackout, and assuming prudent actions by the operator to conserve battery potential, DC power should be available for the first 4-6 hours of a Station Blackout.[5]

Long-Term Considerations. Once DC power is lost, the HPCI and RCIC systems will be inoperable and cooling water can no longer be injected into the reactor vessel. With prudent and realistic operator actions, this should not occur until 4-6 hours* after the inception of a Station Blackout. However, there are other considerations which must be properly addressed during this 4-6 hour period during which all efforts would be made to accomplish the restoration of AC power.

First, since all plant ventilation is stopped during a Station Blackout, the ambient temperature near the RCIC turbine may reach 93.3°C (200°F), causing an automatic isolation of the RCIC system. This is unlikely since the RCIC system will be operated only intermittently, and the turbine is not located in a closely confined space. If isolation does occur, the isolation signal can be overridden by the placing of rubber insulation between the appropriate relay contacts in the auxiliary instrumentation room, or the HPCI system can be used as a backup.

Second, a relief valve might stick open although it should be noted that the currently installed two-stage Target Rock valves are much less prone to this malfunction than their three-stage predecessors. A stuck-open relief valve would produce difficulties with localized suppression pool heating, but reactor vessel pressure would not decrease to the point that the HPCI or RCIC system turbines could not be used. This eventuality is discussed in more detail in Section 8.2.

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*Recent battery capacity calculations have indicated that DC power may remain available for as long as seven hours.

A third consideration would be the continued availability of condensate storage tank water for injection into the reactor vessel. As previously discussed, there is a guaranteed minimum stored supply of 511.0 m^3 (135,000 gallons) available to the HPCI or RCIC systems. As illustrated in Section 7, only about 359.6 m^3 (9[illegible],000 gallons) would be used during the first five hours of a Station Blackout. It can be concluded that an adequate supply of condensate storage tank water is provided for injection during the period in which DC power will remain available.

A fourth consideration is the availability of sufficient stored drywell control air to permit the desired number of remote-manual relief valve actuations during the period of Station Blackout. The accumulators provided for the six relief valves associated with the ADS system are sized to permit five operations per valve, or a total of 30 actuations. As illustrated in Section 7, less than 25 relief valve actuations will be required during the first five hours of a Station Blackout. It can be concluded that there is enough stored air in the ADS relief valve accumulators alone to permit the desired number of relief valve operations during the period in which DC power is available. In addition, the stored air in the Drywell Control Air system receivers provides an ample backup to the accumulator supply.

A fifth consideration concerns the steady heatup of the pressure suppression pool water with no means of pool cooling available during Station Blackout. The pressure suppression pool of approximately 3785 m^3 (one million gallons) of water is contained within a torus of 33.99 m (111.5 ft) major diameter and 9.45 m (31.0 ft) minor diameter. The torus surrounds the drywell and reactor vessel as shown in Fig. 3.1.

The large heat sink afforded by the pressure suppression pool water can be effectively utilized during a Station Blackout only as long as the steam being discharged into the pool via the relief valves is condensed.

If the local temperature of the water surrounding the relief valve tailpipe terminus in the pool is excessive, condensation oscillations may occur causing gross unstable vibrations of the torus assembly and pressurization of the torus airspace due to the escape of saturated steam from the water surface. This effect is discussed in detail in Appendix D. For the "T-quencher" type of discharge header which is installed at the terminus of each relief valve tailpipe at Browns Ferry Unit 1, the condensation oscillations are not expected to occur if the local pool temperature is limited to 93.3°C (200°F) or equivalent to 87.8°C (190°F) average pool temperature.[6]

The T-quencher dischargers on the tailpipes of the thirteen primary relief valves are distributed approximately evenly around the torus, as shown in Fig. 3.2. This diagram is posted on the Browns Ferry Unit 1 control room panel which contains the relief valve operating switches. During a Station Blackout, it would be incumbent on the operator in his efforts to manually control reactor vessel pressure to ensure that oppositely located primary relief valves are operated in turn so that the energy input to the pressure suppression pool is evenly distributed.

As illustrated in Section 7, the average pressure suppression pool temperature will have reached approximately 82.2°C (180°F) five hours after the inception of a Station Blackout. It can be concluded that with operator action to alternate the relief valve discharges, the pressure suppression pool temperature will not reach the point when condensation

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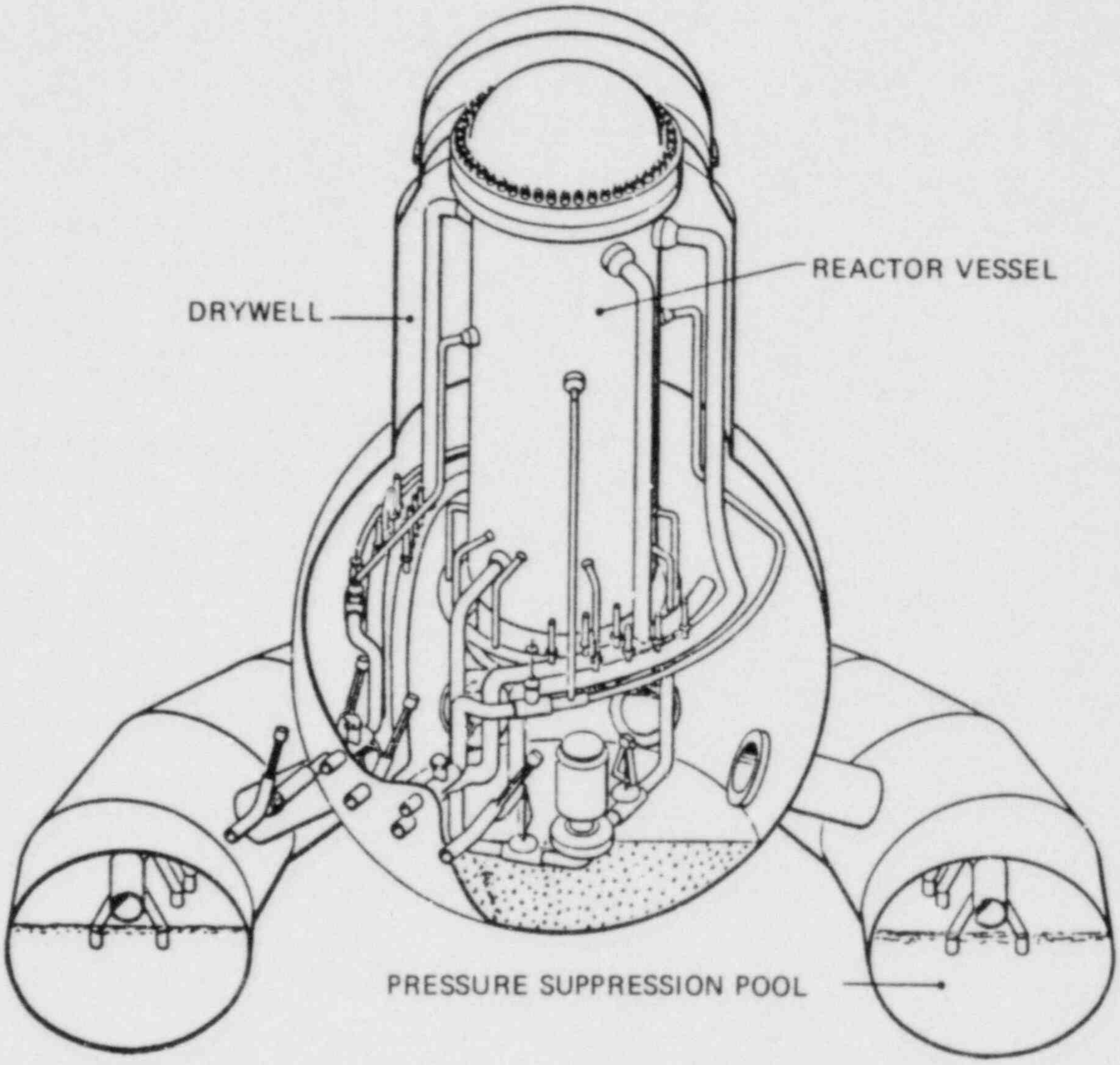

Fig. 3.1 Arrangement of drywell and torus.

oscillations might occur during the first five hours of a Station Blackout.

As a related matter, the HPCI system is equipped with logic which will automatically shift the suction of the HPCI pump to the pressure suppression pool if the indicated pressure suppression pool level increases to a certain point, as discussed in Appendix E. Depending on the initial water level in the pool, this will occur following an increase in pool volume of between 257.4 and 370.9 m^3 (68,000 and 98,000 gallons), or sometime between two and four hours into the Station Blackout. As previously discussed, the HPCI system is not expected to be utilized but merely to serve as a backup to the RCIC system. However, to preserve the HPCI

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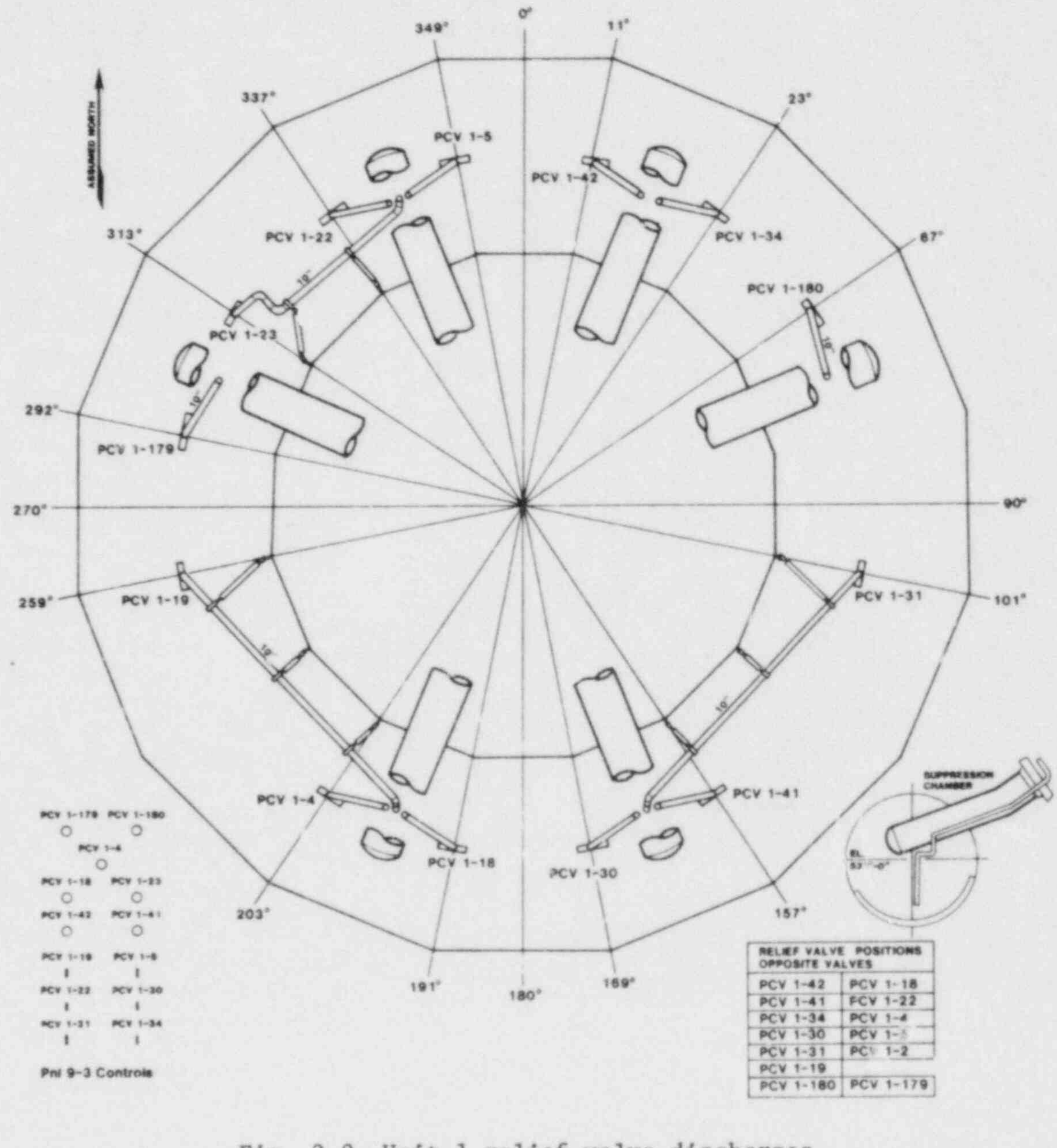

Fig. 3.2 Unit 1 relief valve discharges.

system as a viable backup, the operator should take action to maintain the pump suction on the condensate storage tank. This can be done by racking out the motor-operated breakers for the suction valves, and will prevent the excessive HPCI system lubricating oil temperatures which would result from the pumping of the hot pressure suppression pool water.

The sixth and most important consideration for long term stability following a Station Blackout is the heatup of the drywell atmosphere. The ten drywell coolers immediately fail on loss of AC power, while the rate of heat transfer to the drywell at that instant from the hot [287.8°C (550°F)] reactor vessel and associated piping is about 0.98 MW (3.35×10^6 Btu/h). Under this impetus, the drywell temperature rapidly increases. After a few minutes, the drywell temperature will have significantly increased so that the rate of heat transfer from the reactor vessel is reduced while a substantial rate of heat transfer from the drywell atmosphere to the relatively cool drywell liner has been established, as discussed in Section 4.

The design temperature for the drywell structure and the equipment located therein is 138.3°C (281°F). The response of this structure and of certain safety components within the drywell to higher temperatures has been previously investigated[7] and the results are summmarized below:

It was determined that the drywell liner will not buckle under liner temperatures as high as 171.1°C (340°F) nor would this temperature produce higher than allowable stress intensity.

The drywell electrical penetrations were purchased with a specified short term (15 min) temperature rating of 162.8°C (325°F) and a long term rating of 138.3°C (281°F).

It was determined that the stresses in the drywell piping penetrations would not exceed the allowable stress intensities at a temperature of 171.1°C (340°F).

A DC solenoid control valve used for the remote-manual operation of the primary relief valves was tested at 148.9°C (300°F) for ten hours with 118 actuations during the test period. An AC-DC solenoid control valve for the Main Steam Isolation valves was tested at 148.9°C (300°F) for 7 hours with 83 actuations each on AC, DC, and the AC-DC combination. The maximum temperature achieved during each of these solenoid tests was 153.3°C (308°F).

The electrical cable which feeds the safety equipment inside the drywell has a 600-V rating with cross-linked polyethelene insulation. It has seven conductors of number 12 wire with no sheath. It is estimated that the ten-hour temperature rating is in excess of 160.0°C (320°F).[7]

The results of these tests and investigations indicate that it can be confidently predicted that the primary relief valves will remain operable and the drywell penetrations will not fail if the drywell ambient temperature is prevented from exceeding 148.9°C (300°F) during the normal recovery phase of a Station Blackout.

With the operator acting to maintain reactor vessel level by use of the RCIC system and to control vessel pressure between 7.688 and 6.309 MPa (1100 and 900 psig) by remote-manual relief valve actuation as previously discussed, the drywell ambient temperature would reach 148.9°C (300°F) in one hour (curve A of Fig. 3.3). Before this occurs, the operator should act to reduce the reactor vessel pressure by blowdown to the pressure suppression pool. This will reduce the temperature of the saturated liquid within the reactor vessel and thereby reduce the driving potential for heat transfer into the drywell.

The vessel pressure reduction would be by remote-manual actuation of the primary relief valves, and should proceed at the Browns Ferry Technical Specifications limit of a rate equivalent to a 55.6°C/h (100°F/h) decrease in saturated liquid temperature. The response of the drywell ambient temperature to such a reactor vessel depressurization begun at one hour after the inception of a Station Blackout is shown by curve B of Fig. 3.3.

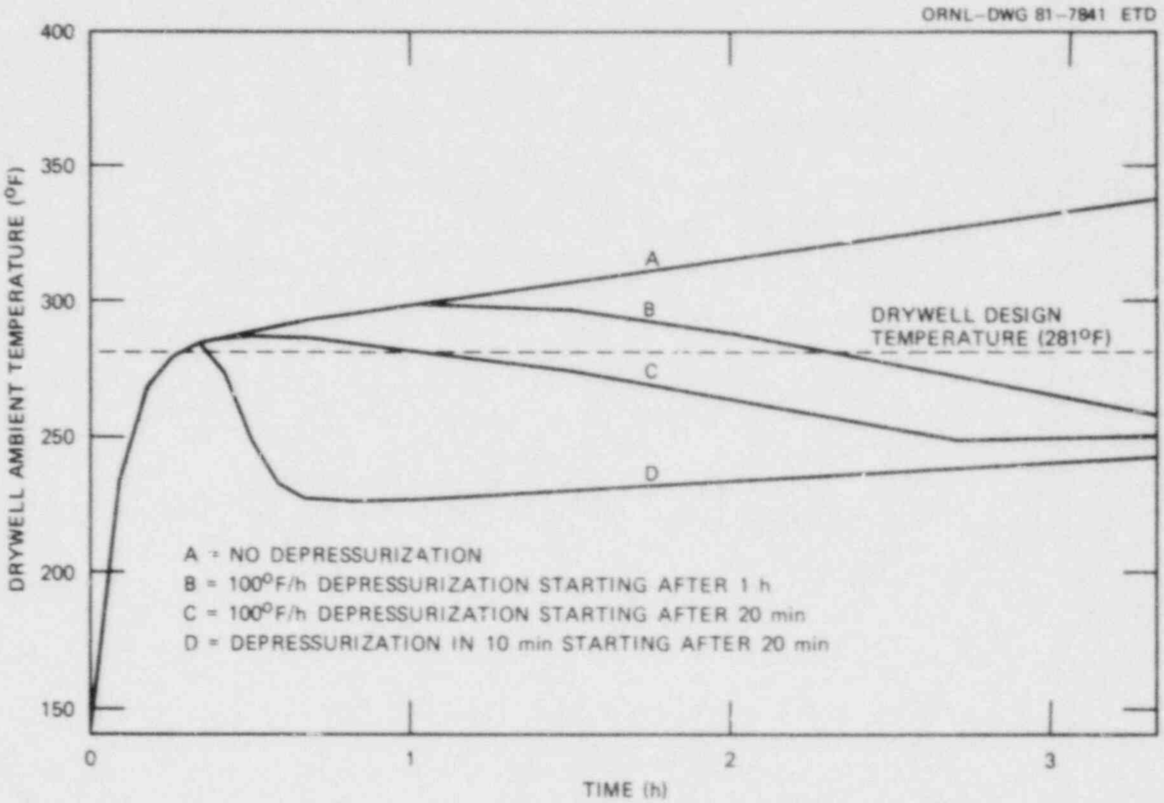

Fig. 3.3 Average drywell temperature during a station blackout.

Once the reactor vessel is depressurized, it is recommended that the operator manually control the vessel pressure between the limits of 0.965 and 0.621 MPa (125 and 75 psig). This pressure range is high enough to permit continued operation of the RCIC system, yet the corresponding average saturated liquid temperature [170°C (338°F)] is low enough to keep the drywell ambient temperature well below 148.9°C (300°F).

As discussed above, a reactor vessel depressurization begun one hour into the Station Blackout would limit the maximum average drywell ambient temperature to 148.9°C (300°F). Alternatively, a depressurization at the Technical Specifications limit of 55.6°C/h (100°F/h) begun 20 minutes after the inception of the Station Blackout would limit the maximum average drywell ambient temperature to about 141.7°C (287°F); the subsequent drywell temperature response is shown by curve C of Fig. 3.3.

Curve D of Fig. 3.3 represents the average drywell ambient temperature response to a rapid depressurization (within ten minutes) of the reactor vessel begun at 20 minutes after the inception of the Station Blackout. This would violate the Technical Specifications limit for reactor cooldown. It can be seen that the drywell temperature only briefly exceeds the design value of 138.3°C (281°F), and is relatively quickly brought down to below 115.5°C (240°F).

The curves of Fig. 3.3 show that the maximum drywell average temperature can be kept below 148.9°C (300°F) provided that the operator takes action to depressurize within one hour following the loss of drywell coolers concomitant with a Station Blackout.* A depressurization at the Technical Specifications limit of 55.6°C/h (100°F/h) will produce a gradual reduction of the drywell ambient temperature; a rapid depressurization will produce a faster drywell temperature reduction, but is probably not necessary. Certainly, if the operator perceives developing difficulties with the in-drywell equipment such as the relief valve solenoids, he should convert an ongoing 55.6°C/h (100°F/h) depressurization into a more rapid one.

Summary. Assuming no independent secondary failures of equipment, reactor vessel level control and pressure control can be maintained during a Station Blackout for as long as the DC power supplied by the unit battery lasts. This is expected to be a period of from four to six hours, and this portion of the Station Blackout severe accident sequence is graphically displayed with commentary in Section 7. If AC power is regained at any time during this period, a completely normal recovery would follow.

If AC power is not regained, and DC power is lost, remote-manual operation of the primary relief valves would no longer be possible and the reactor vessel pressure would increase to 7.722 MPa (1105 psig), the lowest setpoint for automatic actuation of the relief valves. Thereafter, the reactor vessel pressure would cycle between 7.722 and 7.377 MPa (1105 and 1055 psig) while the vessel level steadily decreased because injection capability would be lost as well. The sequence of events following the loss of injection capability is described in Sections 9 through 11.

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*If the backup HPCI system were used for injection, the average reactor vessel pressure after depressurization would have to be maintained at about 1.14 MPa (150 psig), which corresponds to an average saturated liquid temperature of 185°C (365°F). This is only 15°C (25°F) higher than the average temperature associated with RCIC operation and would not significantly increase the drywell temperatures shown in Fig. 3.3.

4. COMPUTER MODEL FOR SYSTEM BEHAVIOR PRIOR TO LOSS OF INJECTION CAPABILITY

The purpose of this chapter is to describe the BWR-LACP (BWR-Loss of AC Power) computer code that was developed to calculate the effects of operator actions prior to loss of injection capability after Station Blackout. Topics covered include: general features, simplifying assumptions, solutions to more important modeling problems, and model validation efforts.

BWR-LACP is a digital computer code written specifically to predict general Browns Ferry thermal-hydraulic behavior following a Station Blackout event (defined in Sect. 2). The code consists of the differential and algebraic equations of mass and energy conservation and equations of state for the reactor vessel and containment. The code was written in the IBM CSMP (Continuous System Modeling Program)[8] Language to minimize programming and debugging time. The listing in Appendix A specifies all required input parameters.

Some of the important variables calculated by the BWR-LACP code include:

1. Reactor vessel levels (above the fuel as well as in the downcomer annulus),
2. Reactor vessel pressure,
3. Total injected water volume,
4. Safety-relief valve (SRV) flow rates,
5. Containment pressures and temperatures, and
6. Suppression pool water level and temperature.

These variables may be calculated for arbitrary time periods following the loss of AC power — typically several hours. Run specific input includes:

1. Injection flow vs time or a law to represent operator control of reactor vessel level with injection flow.
2. SRV opening(s) vs time or a law to represent operator control of reactor vessel pressure with SRV opening.
3. Initialization parameters: initial time elapsed since reactor trip, initial reactor vessel pressure and level, and initial containment pressures, temperatures, and suppression pool level.

Specific characteristics of the Station Blackout event allowed simplifying assumptions to be made, or required specific assumptions for adequate description of system response. These assumptions are discussed below:

1. Reactor trip at time zero: heat input from the reactor is assumed to consist of decay heat and is calculated from the ANS standard[9].

2. No fuel temperature calculation: the fuel remains covered for transients considered here and 100% of the decay heat is assumed to be transferred directly to water.

3. Density of steam-water mixture calculated by drift-flux correlation: the low steaming rate of decay heat operation requires a model, such as the drift-flux model[10] that can allow relative slip between steam and water.

4. Phase separation is likely in the standpipes: the low steaming rates of decay heat operation allow phase separation to occur[11] if the 2-phase level is below the steam separators (in standpipes or core outlet plenum).

5. Recirculation pumps trip at time zero: these pumps require AC power for operation. During the loss of AC power transients, the water in the recirculation piping is treated as an inactive volume.

6. Steam flow through the SRV's is calculated by ratioing the flow at rated conditions:

$$W = W_r \sqrt{(\rho P)/(\rho_r P_r)}$$

where,

W_r = rated SRV flow at pressure P_r and density ρ_r
W = SRV flow at pressure, P, and density ρ.

This relationship is rigorously true for choked flow of an ideal gas with constant C_p/C_v ratio[12] and should give reasonable results for choked flow of dry steam.

7. A realistic estimate of SRV relieving capacity is calculated by taking into account the ASME code rule which requires nominal valve flow to be only 90% of the flow measured at 103% of the nominal actuation pressure. For the two-stage Target Rock SRV's at Browns Ferry, the resulting estimate of actual valve capacity is:

W_r = 121.2 kg/s @ 7.79 MPa

(= 960,000 lb/h @ 1115.0 psig).

8. Steam flowing to RCIC and HPCI turbines is calculated by ratioing the steam flowing at rated conditions:

$$W = W_r*(P/P_r)*(\Delta H_r/\Delta H)$$

where,

W_r = turbine flow required at conditions P_r and ΔH_r (available enthalpy difference for isentropic expansion between P_r and $P_{exhaust}$) to pump rated injection flow.
W = turbine flow required at conditions P and ΔH (isentropic expansion between P and $P_{exhaust}$) to pump the same (rated) injection flow.

This equation predicts a linear relationship between steam pressure and turbine flow providing steam pressure is high enough to establish choked flow across the turbine nozzles. Turbine steam flow is zeroed when the turbine is tripped.

Several modeling problems had to be solved in order to adequately simulate behavior of the system following loss of AC power:

1. Reactor vessel steaming rate,
2. Reactor vessel natural circulation flow,
3. Suppression pool water temperature(s), and
4. Drywell air temperature.

These particular problems are highlighted because of their importance in determining the overall results and to point out approximations that were made in formulating the solutions.

Reactor vessel steaming rate is defined here as the rate at which steam flows from the liquid regions in the reactor vessel into the steam volume in the upper part of the vessel. When pressure is constant the steaming rate equals steam production rate in the reactor core which is a function of core inlet enthalpy, flow, and pressure. However, when pressure is changing, the steaming rate is influenced by the significant amount of water that normally exists in the steam-water mixture above the core in the core outlet plenum, standpipes and steam separators. This water adds to the steaming rate when pressure is decreasing by flashing. It is assumed to subtract from the steaming rate when pressure is increasing by condensing part of the core steam production. In modeling this effect, it was assumed that this water remains at saturation at all times. The net effect on the steaming rate was calculated using the following relationship:

$$W_n = W_c - \frac{dP}{dt} * \frac{dh_f}{dP} * \frac{M_w}{(h_g - h_f)}$$

where

W_n = net steam production rate
W_c = core steam production rate
P = reactor vessel pressure
h_f = saturated fluid enthalpy
h_g = saturated vapor enthalpy
M_w = mass of water in core boiling region and above the core in outlet plenum, standpipes, and steam separators.

The inherent natural circulation capability of BWRs is an important feature that should be considered even in calculation of thermal-hydraulic transients at decay heat level. The natural circulation flow path is from the area outside the standpipes and steam separators (referred to here as the downcomer annulus), down through the jet pump diffusers into the lower plenum and then into the reactor. As it flows up through the core the fluid becomes saturated and begins to boil. The steam/water mixture flows up through the outlet plenum and standpipes to the steam separators, where the water is returned to the downcomer annulus and the steam flows through the steam dryers to the steam dome. The natural circulation rate is dependent on a number of variables including: water level in the downcomer annulus, water-steam level above the core, core steaming rate, and density of the water-steam mixture in and above the core. One phenomenon that makes the calculation more difficult is that when water level gets low enough the recirculation of water from the steam separators ceases and water flows from the downcomer annulus only as required to counter-balance water lost by boil-off from the core.

The equation used to calculate core inlet flow is:

$$K_f W_{ci}^2 = \rho_{dc} * L_{dc} - \bar{\rho}_{tp} * L_{tp}$$

where

K_f = empirically determined friction coefficient
W_{ci} = core inlet flow
ρ_{dc} = downcomer water density
L_{dc} = downcomer water level (ref. to zero at bottom of active fuel)

$\bar{\rho}_{tp}$ = elevation-averaged density of water-steam mixture in and above core
L_{tp} = level of steam-water mixture above bottom of active fuel.

The value of the friction coefficient, K_f, was calculated from the natural circulation curve on Fig. 3.7-1 of the BFNP FSAR by forcing agreement with the predictions of this equation at the 30% thermal power point. This procedure resulted in a reasonably good approximation of the flow at other points, as shown on Fig. 4.1.

The rate of recirculation of water back to the downcomer annulus is calculated as a function of the water-steam mixture level in the steam separators:

$$W_{recir} = (W_{ci} - W_n) * f(L_{tp})$$

where

W_{recir} = water recirculated back to downcomer annulus
$f(L_{tp})$ = empirical function of 2-phase level
W_n = net steam production rate
W_{ci} = core inlet flow.

The function $f(L_{tp})$ is 1.0 at normal level in the steam separators, decreasing to 0.0 when level is below the lower edge of the steam separators.

Calculation of local temperature of suppression pool water is necessary for simulation of those events that require the assumption of extended SRV discharge into one local area of the pool. For the normal recovery accident sequences discussed in this report it was assumed that the operators would follow the Browns Ferry procedures for reactor vessel isolation which require manual cyclic sequential operation of the SRVs

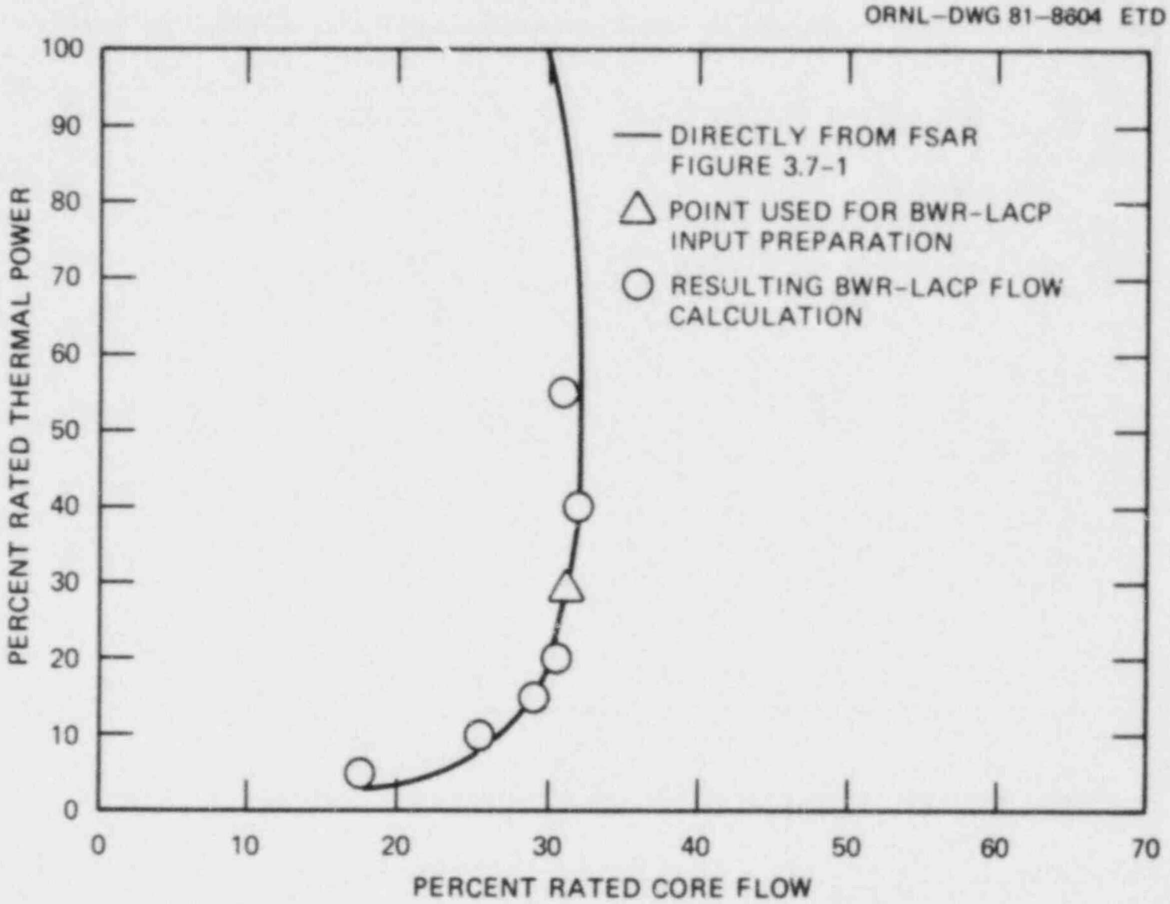

Fig. 4.1 Natural circulation flow.

such that the SRV discharge is distributed throughout the pool. Therefore, it wasn't necessary to calculate local temperatures, and the model was programmed to calculate only the whole-pool volume-averaged temperature. It was assumed that the pool would quench all of the SRV discharge as long as the average pool temperature is below the boiling point. The SRV T-quencher dischargers are located at a depth below the midpoint between pool top and bottom and thus tend to "see" a temperature close to or below the average temperature. The possible occurrence and consequences of non-homogeneous suppression pool temperatures are discussed in Appendix D.

During normal full power operation, plant measurements have shown that the Browns Ferry Unit 1 drywell coolers remove about 1960 kW [$6.7(10)^6$ Btu/h] of thermal energy from the drywell atmosphere. This heat comes from various sources, including heat transfer from the hot reactor vessel and steam lines and from the operation of AC powered equipment inside the drywell. During a Station Blackout event, the AC powered equipment is lost so only about half of the heat source remains; accordingly, the initial drywell heat load following loss of AC power was taken to be 980 kW [$3.35(10)^6$ Btu/h]. This heat load Q_{dw} is assumed to be proportional to the difference between reactor coolant temperature (T_{rc}) and drywell atmosphere temperature (T_{dw}). Therefore,

$$Q_{dw} = 980*(T_{rc} - T_{dw})/(T_{rc} - T_{dw})_o \ .$$

To test the validity of the 980 kW heat load, an independent calculation was performed under the following assumptions:

1. Total heated surface area is 1115 m**2 (12,000 sq ft). This includes 539 m**2 (5800 sq ft) for the reactor vessel, 372 m**2 (4000 sq ft) for the steam lines, and 204 m**2 (2200 sq ft) for other heated piping inside the drywell.
2. 90% of the heated surface is insulated. This is a rough estimate to account both for uninsulated piping and for imperfectly installed insulation.
3. Total heat loss through insulated surfaces is 0.25 kW/m**2 (80 Btu/h sq ft). This is a good nominal estimate for MIRROR insulation when total hot surface to ambient temperature difference is about 220°C (400°F).
4. Radiant heat loss from uninsulated surface is figured using emmissivity = 0.8 (i.e., unpolished surface)
5. Convective heat loss from uninsulated surface figured using the relation Nu = 0.13 (Gr*Pr)**0.3333 (as discussed in a following paragraph).

The results were as follows:

$$Q_{ins,total} = 281 \text{ kW}$$
$$Q_{unins,convective} = 170 \text{ kW}$$
$$Q_{unins,radiant} = 375 \text{ kW}$$
$$Q_{total} = 826 \text{ kW}$$

This shows that the heat losses assumed here for Unit 1 are realistic and perhaps slightly conservative. The Station Blackout analysis presented in the Browns Ferry FSAR (response to AEC Question 14.2) assumes a much more conservative heat loss of 2000 kW.

In addition to heat transfer from hot surfaces, the rate of steam escaping directly to the drywell must be known in order to calculate drywell pressure and temperature. Figure 9.2-2 of the Browns Ferry FSAR specifies a total drywell drain sump input during normal power operation of 22 m**3 (5800 gallons) per day or 4 gpm as measured at the normal sump temperature of about 38°C (100°F). Recirculation pump seals and control rod drive flange seals are identified as the major source of this leakage. This 4 gpm leak rate is believed to be realistic for Browns Ferry Unit 1 during a Station Blackout. The fraction of the leak that flashes to steam was calculated in terms of saturation properties:

$$X_{flash} = (h_f@P_r - h_f@P_{dw})/(h_g@P_{dw} - h_f@P_{dw})$$

where h_f and h_g are saturated liquid and vapor enthalpies and P_r and P_{dw} are reactor vessel and drywell pressures.

Drywell temperature was calculated using a very simple noding scheme — the model assumes that the drywell atmosphere is at a uniform temperature. This is equivalent to assuming that effective natural circulation paths will develop for heat transfer from hot surfaces, to drywell atmosphere, to cold surfaces (such as the drywell liner). Such natural circulation probably exists in the lower part of the drywell where the main steam isolation valves and drywell coolers are located. However it is likely that temperatures in certain areas near the top of the drywell could be much hotter than the single atmosphere temperature this model calculates.

Heat transfer between drywell atmosphere and liner was calculated using the relation

Nu = 0.13*(Gr*Pr)**0.333

where

Nu = Nusselt Number (hL/k)
Gr = Grashof Number ($\rho^2 g\beta L^3 \Delta T/\mu^2$)
Pr = Prandtl Number ($\mu C_p/K$) .

This formula is also used for the same purpose in the MARCH and CONTEMPT-LT codes.[13,14]

Internal heat transfer resistance of the steel drywell liner is low relative to the resistance between liner and the atmosphere. This allows the use of one temperature to represent the whole thickness and it also means that the entire ~364,000 kg (800,000 lbs) of drywell metal is effectively available as a heat sink during drywell heat-up. The heat sink effect of the drywell liner is important because without this heat sink the drywell atmosphere would very quickly approach primary coolant temperature after loss of the drywell coolers.

Various code comparison activities were pursued in order to test the validity of the BWR-LACP simulation:

1. Comparisons of BWR-LACP results to calculations reported in the FSAR.
2. Comparison to results computed by the RELAP-IV computer code.

The first FSAR comparison was the "Loss of Auxiliary Power — All Grid Connections" transient reported on Figs. 14.5—12 and 14.5—13 of the BFNP FSAR. Plant conditions assumed by the FSAR analysis include:

1. Loss of all external grid connections at time = 0.
2. All pumps tripped at time = 0.
3. Reactor trip and main steam isolation at about time = 2 s.

The BWR-LACP calculation was started at time equals 20 s with reactor vessel level and pressure equal to those shown for t = 20 s on FSAR Figs. 14.5—12 and 14.5—13. Figure 4.2 compares the code and FSAR predictions of reactor vessel pressure and core inlet flow between 20 and 50 s. During this period the core inlet flow is decreasing because steam production is falling off rapidly; reactor vessel pressure is being controlled between the assumed 7.59 MPa (1100 psia) and 7.41 (1075 psia) SRV set and reset points. Results are reasonably close. Figure 4.3 shows BWR-LACP and FSAR calculations of reactor vessel level from 20 to 2200 s. During this period, the vessel level decreases slowly until the mass addition rate of the 0.0378 m**3/s (600 gpm) injection flow exceeds the mass loss rate due to steam produced by decay heat. The results show similar behavior. The FSAR minimum level is 11.76 m (463 in.) at time equals 1600 s and the BWR-LACP predicted minimum level is 11.56 m (455 in.) at time equals 1200 s.

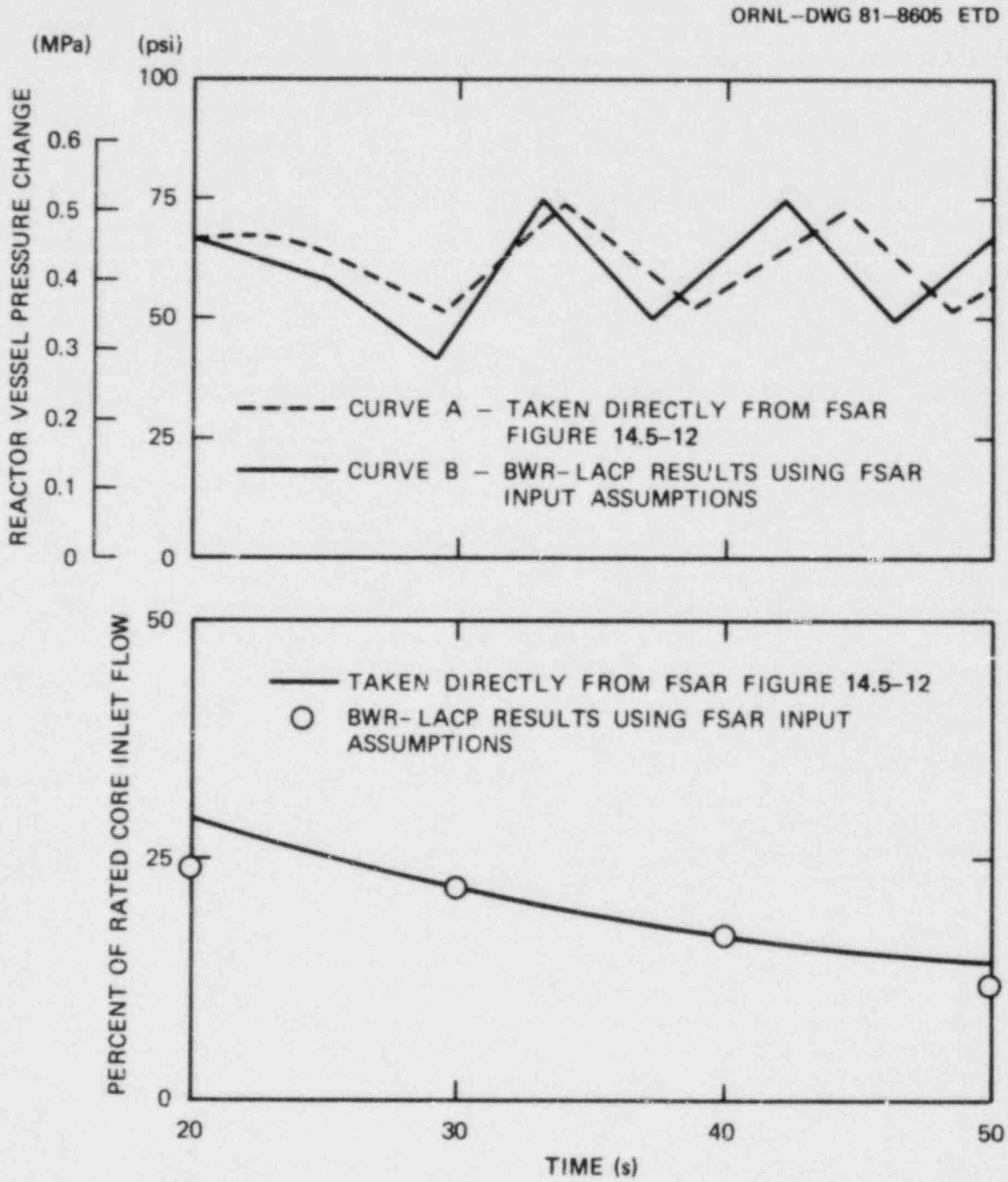

Fig. 4.2 Comparison of FSAR and BWR-LACP results for loss of auxiliary power transients.

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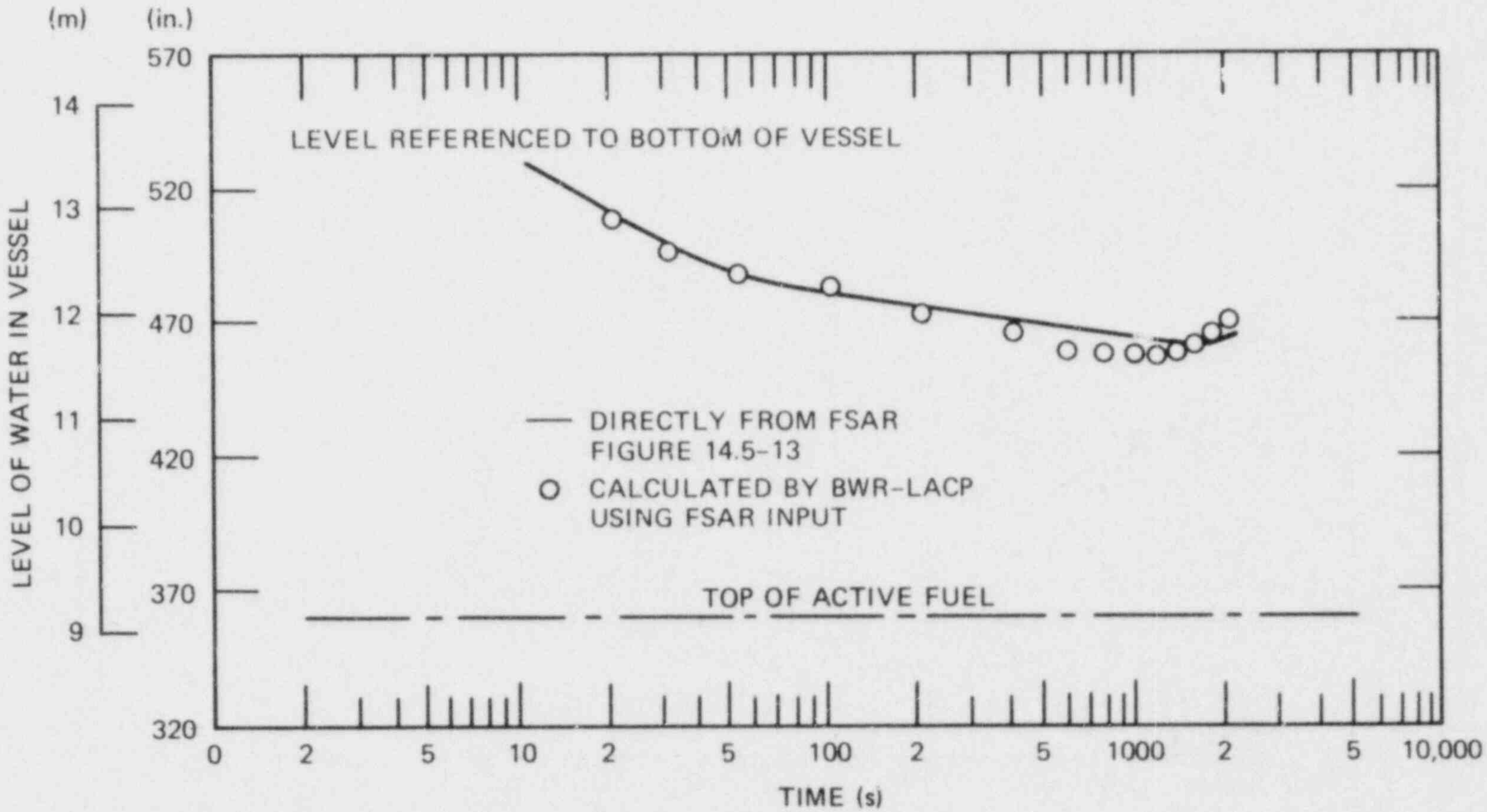

Fig. 4.3 Comparison of FSAR and BWR-LACP results for reactor vessel level.

Comparison of FSAR and BWR-LACP calculations of drywell temperature was performed for the "Loss of All AC Power" results shown on Fig. Q.14.2-1 of the BFNP FSAR. Plant conditions assumed by the FSAR calculations include:

1. Initial power = 100%.
2. Loss of all AC power at time = 0.
3. Drywell perfectly insulated on outside.
4. Heat capacity of interior equipment and structures is negligible.
5. Energy input from primary surfaces is initially 2000 kW.
6. Heat transfer coefficient between atmosphere and wall is 11.36 w/m**2°C (? Btu/h ft^2 °F) without steam condensation.
7. Drywell coolers lost at time = 0 and not restarted.
8. Steam leak = 0

The conditions assumed for this analysis are of the same order as those assumed for the "Normal Recovery" part of the Station Blackout as discussed in this report. Comparisons of various predictions of drywell atmosphere temperature to the FSAR result are shown on Fig. 4.4. The drywell temperature climbs rapidly during the first 5 to 10 min, before heat

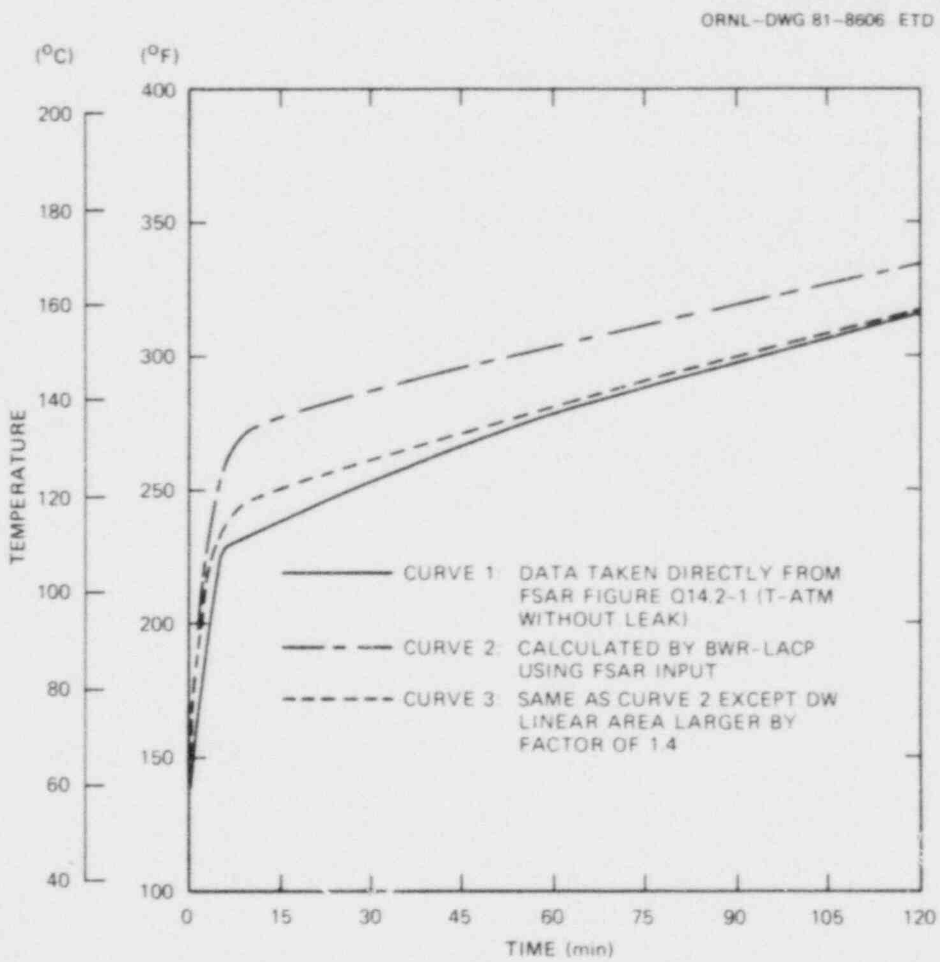

Fig. 4.4 Drywell atmosphere temperature: comparison of FSAR and ORNL BWR-LACP results.

transfer to the drywell liner is established. After the first 10 min, the temperature climbs much more slowly, with the metal liner acting as the heat sink. Curve 2 on Fig. 4.4 was calculated using as input to the containment model the FSAR assumptions as listed above. The resulting curve is similarly shaped, but about 17°C (30°F) above the FSAR prediction. In order to estimate the possible magnitude of input parameter difference, the calculation was repeated with identical input except a 40% increase in drywell liner area. Results are shown on Curve 3 and are very close to the FSAR prediction.

A boil-off transient was selected for comparison to RELAP-IV and BWR-LACP predictions. The RELAP-IV calculation was prepared and run by Idaho National Engineering Laboratory. It starts at nominal full power plant conditions. During the first several seconds the reactor trips and the main steam isolation valves close. Initially, several SRVs are required to control vessel pressure, but after about 40 seconds one SRV is sufficient. Injection flow is zero throughout. The BWR-LACP calculation was intialized at 30 seconds because it is programmed only for decay heat operation. Figure 4.5 compares the results of the calculations of reactor

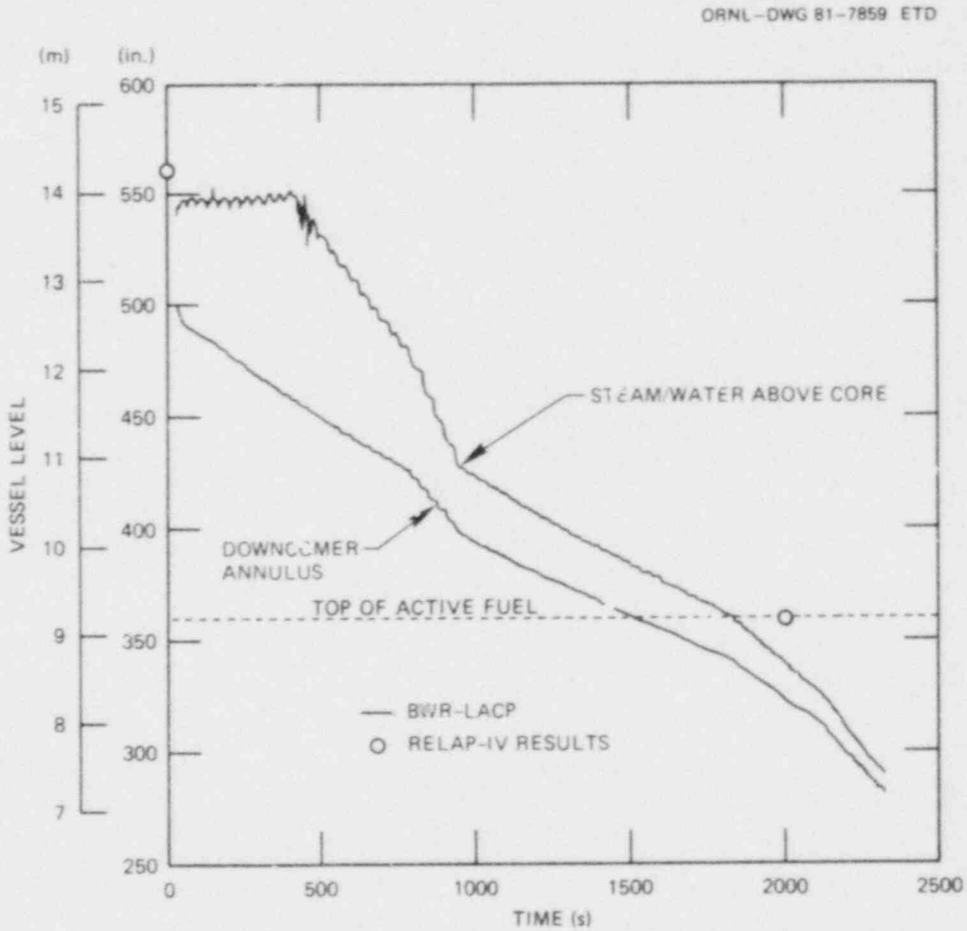

Fig. 4.5 Comparison of RELAP-IV and BWR-LACP results for reactor vessel level.

vessel level. The steam/water level above the core is shown for both codes. Results are very similar, with RELAP predicting 33 minutes and BWR-LACP predicting 30 minutes to begin uncovering active fuel. Figure 4.6 shows RELAP and BWR-LACP results for reactor steam pressure control. Results are similar although RELAP predicts a longer SRV cycle (about 60 seconds) than BWR-LACP (about 38 seconds).

Results presented in this Section show that the BWR-LACP code is capable of providing reasonable predictions of overall thermal-hydraulic

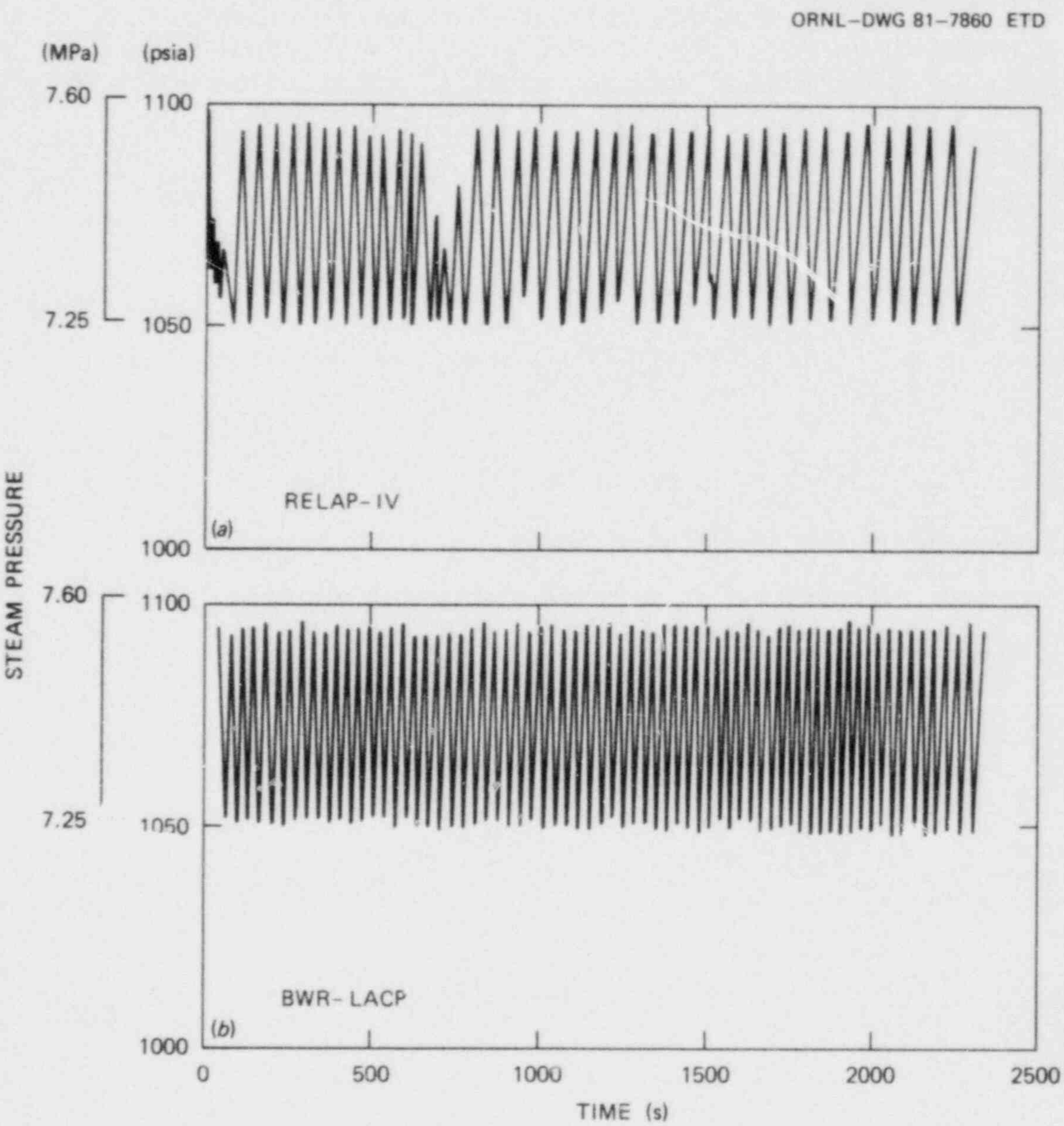

Fig. 4.6 Comparison of RELAP-IV and BWR-LACP results for reactor vessel pressure.

variables such as reactor vessel levels and pressure and containment temperature for extended periods after reactor trip prior to system damage due to loss of injection capability. Features of the code that have contributed to its utility in the analysis of the Normal Recovery portion of Station Blackout include:

1. The code is specifically designed for BWRs; therefore, parameter changes are straightforward and easily made.
2. The code is designed for flexibility to model the plant response to different operator actions.
3. It is locally available and has quick turnaround time.

5. INSTRUMENTATION AVAILABLE DURING STATION BLACKOUT AND NORMAL RECOVERY

For some time following a Station Blackout, the reactor vessel water level can be maintained in the normal range by operation of the HPCI and/or RCIC systems, and the pressure can be maintained in any desired range by remote-manual relief valve operation. Restoration of AC power at any time during this period of stable level and pressure control would permit complete and normal recovery without core damage. The Control Room instrumentation available and necessary to monitor the plant status during the period of stable control following the inception of a Station Blackout is discussed in the following paragraphs.

1. Electrical system status. The Station Blackout would be clearly discernible by the loss of much of the control room indication and the loss of normal Control Room lighting. Emergency Control Room lighting is available. Numerous instruments would indicate the loss of voltage, amperage, and frequency on the electrical supply and distribution boards.

2. Reactor vessel water level. Two channels of Control Room instrumentation would provide indication of the reactor vessel water level over a range between 13.41 and 14.94 m (528 and 588 in.) above the bottom of the vessel. This range includes the normal operating level of 14.15 m (561 in.) and extends over the upper portion of the steam separators, a distance of 4.27 to 5.79 m (14 to 19 ft) above the top of the active fuel. Mechanical Yarway indication available outside of the Control Room would provide an additional range from the low point of the Control Room indication, 13.41 m (528 in.) above vessel zero down to a point 9.47 m (373 in.) above vessel zero, which is 0.33 m (13 in.) above the top of the active fuel in the core.

The level instrumentation derives the reactor water level by comparing the head of water within the downcomer region of the reactor vessel to the head of water within a reference leg installed in the drywell. With the loss of the drywell coolers, the drywell ambient temperature will increase significantly, heating the reference leg water to above-normal temperatures. This will reduce the density of the water in the reference leg, causing an error in the indicated water level, which will be too high. For example, if the drywell ambient temperature increases from its normal range of 57.2 to 63.6°C (135 to 150°F) to 171.1°C (340°F), the indicated level can be as much as 0.76 m (30 in.) too high.[15] As discussed in Sect. 3, the drywell ambient temperature is expected to reach about 148.9°C (300°F). However, the lower boundary of reactor vessel level indication is 4.27 m (168 in.) above the top of the active fuel. Therefore, if the operator maintains the level in the indicating range, a 0.76 m (30 in.) error is not significant as far as the goal of keeping the core covered is concerned.

3. Reactor vessel pressure. Two channels of pressure instrumentation would provide indication of reactor vessel pressure over a range of 8.274 MPa (0 to 1200 psig).

4. Main steamline flow. Two channels of steam flow instrumentation with a range of 2016 kg/s (0 to 16 x 10^6 lbs/h) would provide verification that steam flow had ceased following Main Steam Isolation Valve (MSIV) closure.

5. Feedwater flow. One channel of flow instrumentation with a range of 1008 kg/s (0 to 8 x 10^6 lbs/h) for each of the three feedwater pumps would indicate the decay of feedwater flow following loss of steam to the feedwater turbines.

6. Neutron flux. The power range meters fail on loss of AC power. The source range and intermediate range monitors remain operational, but the detectors for these systems are withdrawn from the reactor core during power operation and could not be reinserted under Station Blackout conditions. Nevertheless, the operator could verify the reactor scram which occurs when the Station Blackout is initiated by observing the decay of the indicated levels on these monitors.

7. Control rod position indication. The control rod position indication system remains operational so that the operator could verify that the control rods had inserted with the scram.

8. Relief valve position. There is no provision for direct indication of the actual position of any primary relief valve, even under normal operating conditions. Thus the operator has no indication of automatic actuation of a specific relief valve other than the recorded tailpipe temperatures available on charts behind the control room panels, or the relief valve acoustic monitors, all of which would be inoperable under Station Blackout conditions. However, remote-manual actuation of a relief valve is accomplished by energizing its DC solenoid operator; lights on the control panel for each valve indicate whether or not these solenoids are energized, and this capability is maintained during Station Blackout.

9. RCIC instrumentation and controls. The RCIC system can be operated and monitored under Station Blackout conditions.

10. HPCI instrumentation and controls. The HPCI system can be operated and monitored under Station Blackout conditions.

11. Condensate Storage Tank Level. The condensate storage tank level indication [range: 9.75 m (0 to 32 ft)] remains operational so that the operator can determine the remaining amount of water available for reactor vessel injection via the RCIC (or HPCI) system.

12. Drywell Pressure. Drywell pressure instrumentation [range: 0.55 MPa (0 to 80 psia)] remains operational so that the operator can monitor the increase in drywell pressure due to the ambient temperature increase following loss of the drywell coolers.

13. Pressure Suppression Pool Water Level. Indication of torus level [range: −0.63 to 0.63 m (−25 to +25 in.)] remains available allowing the operat to monitor the increasing pool level caused by both the relief valve blowdowns and the increasing pool temperature.

The above instrumentation is powered during Station Blackout either by DC power directly from the installed batteries or by AC power indirectly obtained from the battery systems. The sources of AC power during Station Blackout are the feedwater inverter and the unit-preferred and plant-preferred systems where single-phase 120V AC power is provided by generators driven by emergency battery-powered DC motors.

It is important to note that Control Room temperature instrumentation for the drywell and the pressure suppression pool does not remain operational under Station Blackout conditions. However, the ambient temperatures at various points in the drywell would be available at an indicator-meter and a recorder mounted on panel 9-47, which is located on the back of the Control Room panels and is accessible from outside of the Control

Room. Also, using a portable self-powered potentiometer, station personnel can monitor local pressure suppression chamber and drywell temperatures which are sensed by installed thermocouple elements.

Following the restoration of normal AC power, either by recovery of the offsite power sources or by the delayed but finally successful loading of the onsite diesel-generators, the Instrumentation and Control buses would be re-energized, restoring all normal Control Room instrumentation for use in monitoring the normal recovery.

6. OPERATOR ACTIONS DURING STATION BLACKOUT AND NORMAL RECOVERY

In the event of a Station Blackout, the immediate result would be a turbine trip, reactor scram, and Main Steam Isolation Valve (MSIV) closure, which would all occur automatically with no operator action; the reactor would be shut down and isolated behind the closed MSIVs within 30 seconds following the initiating loss of offsite power. Immediately following the isolation, the reactor vessel pressure would increase to the setpoints of as many relief valves as are required to terminate the pressure increase. The affected relief valves would open to pass steam directly from the reactor vessel to a terminus beneath the water level in the pressure suppression pool so that the released steam is condensed.

The core water void collapse caused by both the scram and the pressure increase following MSIV closure would result in a rapid drop in reactor vessel water level to some point below the lower limit of level indication available in the Control Room under Station Blackout conditions [13.41 m (528 in.) above vessel zero]. Standard Browns Ferry Emergency Operating Instructions for immediate action following reactor scram and MSIV closure call for the operator to manually initiate both the HPCI and the RCIC systems. The combined injection flow of 0.353 m^3/s (5600 GPM) would rapidly restore the level into the indicating range and the operator would secure the HPCI system when the indicated level reached a point equivalent to about 13.72 m (540 in.) above vessel zero. The level would then continue to increase due to the remaining 0.038 m^3/s (600 GPM) of RCIC system flow and the heatup and expansion of the injected water, but at a slower rate.

For pressure control, standard procedure is for the operator to remote-manually operate the primary relief valves as necessary to cycle the reactor vessel pressure between 7.69 and 6.31 MPa (1100 and 900 psig). This prevents the automatic actuation of the relief valves [lowest setpoint: 7.72 MPa (1105 psig)], and since the pressure reduction following automatic actuation is only 0.34 MPa (50 psi), greatly reduces the total number of relief valve actuations. The purpose is to decrease the opportunity for a relief valve to stick open, and to allow the operator to alternate the relief valves which are in effect passing the decay heat energy to the pressure suppression pool; this more evenly distributes the pool heatup.

Throughout the period of Station Blackout, the operator would operate the RCIC system intermittently as necessary to maintain the reactor vessel level in the indicating range, i.e., between 13.41 and 14.94 m (528 and 588 in.) above vessel zero. Thus, it is expected that the operator would turn on the RCIC system at an indicated level of about 13.61 m (538 in.) and turn the system off at about 14.68 m (578 in.). The periods of RCIC system operation would become less frequent as the decay heat intensity subsides during a prolonged Station Blackout; after the first hour, the operator would find it necessary to run the RCIC system for only about one-third of the time. This is illustrated in Sect. 7.

As previously discussed, the operator would control reactor vessel pressure by remote-manual operation of the relief valves. It is important that the operator take action to begin to lower the reactor vessel pressure to about 0.79 MPa (100 psig) within 60 min following the inception of a Station Blackout. This is necessary because of the loss of drywell cooling concomittant with a Station Blackout, and would be accomplished by remote-manual actuation of the primary relief valves. Analysis indicates that the average ambient drywell temperature will not exceed 148.9°C (300°F) if this action is taken within one hour. It should not be necessary for the cooldown rate to exceed the Technical Specifications limit of 55.6°C/h (100°F/h) for reactor vessel cooldown, but the operator should increase the cooldown rate if temperature-induced problems with the operation of the relief valves are perceived. This reactor vessel depressurization is necessary to preclude an excessive ambient temperature in the drywell, designed for 138.3°C (281°F). Excessive drywell ambient temperature could cause serious and expensive damage to drywell equipment and from a safety standpoint, would threaten the primary relief valve manual actuation solenoids and the integrity of the primary containment.

When the operator acts to reduce the reactor vessel pressure from 7.00 to 0.79 MPa, (1000 psig to 100 psig) the corresponding reduction in saturation temperature within the vessel is from about 285 to 170°C (545 to 338°F); this reduces the driving potential for heat transfer into the drywell and limits the maximum drywell average ambient temperature following the Station Blackout to about 148.9°C (300°F). The RCIC system steam turbine is capable of operation with an inlet pressure of 0.79 MPa (100 psig) and is often run under these conditions. It is reasonable to expect the operator to act to manually control vessel pressure between the limits of 0.97 and 0.62 MPa (125 and 75 psig) during a prolonged period of Station Blackout while DC power remains available. Analysis shows that this will not require an excessive number of relief valve actuations.

The operator must alternate the relief valves used for pressure control so that the suppression pool water is evenly heated by the condensing steam. Otherwise, the local water temperature surrounding the tailpipe of an overused relief valve may become too high for effective steam condensation.

It should be recalled that there are temperature sensors located near the HPCI and RCIC turbines which are designed to detect steam leaks and consequently shut down these systems by closure of their primary containment inboard and outboard steam supply isolation valves. The setpoint is 93.3°C (200°F) and it is not unreasonable to suggest that this space temperature might be reached during a prolonged Station Blackout when these systems are operated without benefit of space coolers, although neither of these systems is closely confined. If this occurs, the operator would have to take action to override these system isolation signals. This can be easily done in the auxiliary instrument room by the placement of rubber insulation between the temperature-activated relay contacts.

It is important to note that no emergency procedure for Station Blackout at Brown's Ferry currently exists. The operator actions discussed in this section are those indicated by the ORNL analysis of the casualty. In summary, these show that during a prolonged Station Blackout the operator should maintain reactor vessel level within the indicating range by intermittant operation of the RCIC system and should control pressure between 0.62 and 0.97 MPa (75 and 125 psig). This range of control insures sufficient pressure to run the RCIC turbine, but keeps the reactor vessel temperature as low as possible to minimize drywell heatup.

There are ample supplies of Condensate Storage Tank water available for injection, and the period of stable level and pressure control following a Station Blackout can be maintained for a minimum of four hours provided that independent secondary equipment failures do not occur. Beyond four hours, the availability of battery-supplied DC power is in question, and the pressure suppression pool water temperature will have increased to the point where the successful condensation of the relief valve steam discharges begins to be in doubt.*

The required frequency of RCIC system operation and the manual actuations of relief valves by the operator during the period of Station Blackout are shown on the plots of thermal-hydraulic parameters as functions of time included in Section 7.

During the period of Station Blackout, the 0.038 m^3/s (600 GPM) RCIC capacity is sufficient for vessel level control, and the HPCI system need not be used. However, if the RCIC system should malfunction, the HPCI system can be used as a backup. It is important to recall that the suction of the HPCI pump will automatically shift from the condensate storage tank to the pressure suppression pool when the pressure suppression pool level increases to an indicated level of 0.18 m (+7 in.) as discussed in Appendix E. This level will be reached after about three hours of Station Blackout, when the pressure suppression pool temperature will have increased to about 71.1°C (160°F). Since the HPCI turbine lubricating oil is cooled by the water pumped, and because the high water temperature may provide an inadequate net positive suction head, the operator should take action to prevent this shift in pump suction under Station Blackout conditions. This can be done by manually racking out the circuit breakers for the DC-motor-operated suction valves.

When AC power is restored, the operator must take action to implement long-term cooling for the removal of decay heat. It is important that he recognize that the increased drywell pressure caused by the heatup of the drywell atmosphere following the loss of the containment coolers combined with the reduction of vessel pressure form the classic indication of a LOCA, i.e., high containment pressure [setpoint 0.012 MPa (2 psig)] and low vessel pressure [setpoint 3.21 MPa (450 psig)]. Unless the operator takes action to prevent it, these signals will cause both the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) system and the Core Spray system to automatically actuate as soon as AC power is restored. These systems would then inject a combined flow of approximately 2.271 m^3/s (36,000 GPM) into the reactor vessel entirely unnecessarily. This flow would quickly fill the vessel and raise the pressure to the shutoff head of the Core Spray and RHR pumps. Several actions might be taken by the operator to prevent this occurrence; for example the high drywell pressure signal can be overridden, or the core spray and RHR system pumps can be turned off before AC power is restored. The important thing is that the operator anticipate this occurrence.

The normal recovery from a Station Blackout would be established when the operator has manipulated the RHR system into the shutdown cooling mode and the suppression pool cooling mode following the restoration of AC power.

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*See Appendix G for analysis of the case where DC power from the unit batteries is assumed to remain available for seven hours.

7. COMPUTER PREDICTION OF THERMAL-HYDRAULIC PARAMETERS FOR NORMAL RECOVERY

7.1 Introduction

The purpose of this chapter is to present the results of BWR-LACP calculations (see Sect. 4) of system behavior for two different portions of the Station Blackout:

1. Normal Recovery: DC power from the unit battery remains available during this period; therefore RCIC and HPCI injection flows are assumed available throughout, and
2. Loss of 250 vdc Batteries: The batteries are assumed to fail after four hours, causing failure of RCIC and HPCI injection, ultimately leading to uncovering of the core about eight hours after the blackout.

The normal recovery portion of the Station Blackout is discussed in detail in Sects. 3 and 6. Operator actions assumed here are consistent with those sections. Results were calculated to five hours* for normal recovery in order to provide some overlap with the MARCH code severe accident calculations (see Sects. 9 and 10), which assume that the boil-off begins after four hours from a fully pressurized condition. The Loss of 250 vdc Batteries results are given in this section in order to provide an estimate of system behavior possible after loss of dc power if the plant is depressurized during the normal recovery period.

7.2 Conclusions

Major conclusions drawn from the calculated results are given below, succeeded by a detailed discussion of code input assumptions and transient results.

7.2.1 Normal Recovery — Conclusions

If power were recovered within five hours of the inception of the Station Blackout, a normal recovery would be possible. System parameters are within acceptable ranges after five hours:

1. The 250 vdc batteries, by assumption, last the full five hours.
2. Reactor vessel level is within the normal control range, about 5.08 m (200 in.) above the top of active fuel.
3. Reactor vessel pressure is being controlled at about 0.69 MPa (100 psia).
4. About 354 m^3 (93,500 gal.) of water have been pumped from the Condensate Storage Tanks, which had an assured capacity of 511 m^3 (135,000 gal.) before the blackout.
5. Suppression pool temperature is about 82°C (180°F), but this should not be a problem for the T-quencher type of SRV discharge header piping.
6. Containment pressures are elevated to about 0.17 MPa (25 psia), well below the 0.53 MPa (76.5 psia) design pressure.

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*These results are extended to seven hours in Appendix G.

7. Drywell atmosphere temperature is below the 138°C (281°F) design temperature.

7.2.2 Loss of 250 vdc Batteries — Conclusions

The results of this transient show very clearly how fuel damage is postponed because the reactor vessel was depressurized early in the Station Blackout:

1. The repressurization time after loss of the 250 vdc batteries is greater than one hour, during which time there is no significant coolant loss from the reactor vessel.
2. When the boil-off does begin, it takes a much longer time to uncover fuel because of the higher starting inventory of water.

Although fuel damage is significantly delayed, the ability to avoid ultimate fuel damage is compromised because of the elevated drywell temperature experienced after loss of the 250 vdc batteries. As discussed in Sect. 3, a containment temperature of 149°C (300°F) would not prevent normal recovery. This temperature is reached about 40 min. after loss of the batteries. At about four hours after the battery loss, the fuel is beginning to be uncovered, and the drywell temperature is above 191°C (375°F). This elevated temperature may cause failure of the drywell electrical penetrations and may fail the solenoid operators necessary for operation of the SRVs, inner isolation valves, and containment cooler dampers (which fail closed on loss of AC power). Even if electrical power were fully restored at this point, considerable operator ingenuity would be required to effect a normal recovery if the MSIV and SRV solenoid operators have failed.

In addition, the chances for a normal recovery are compromised by the elevated suppression pool temperatures experienced at the end of the transient. Condensation oscillation during SRV discharge (see Appendix D), which can damage the suppression pool pressure boundary, becomes more likely at higher pool temperatures. Steam discharge without oscillation from the Browns Ferry T-quencher type discharge piping is assured up to 88°C (190°F), but average pool temperature at the end of the Loss of 250 vdc Batteries calculation is above 93°C (200°F).

7.3 Normal Recovery

7.3.1 Normal Recovery — Assumptions

Assumptions specific to programming and input preparation for the normal recovery calculation include:

1. The calculation begins 30 s following the Station Blackout initiation with the reactor tripped and the main steam isolation valves closed. Values for system parameters at the 30-s point are taken from the results reported by Sect. 14.5.4.4 of the Browns Ferry FSAR. The calculation ends at the 5 hour point.
2. The RCIC system is used in an on-off mode to control vessel level between 13.7 m (538 in.) and 14.7 m (578 in.) above vessel zero. The

HPCI system is actuated only when level is more than 0.25 m (10 in.) below the 13.4-m (528-in.) lower limit of control room indication under Station Blackout conditions. Suction for both systems is from the condensate storage tank (CST). The operators actuate HPCI and RCIC 1 min. after the blackout.

3. Following 1 min. of automatic SRV actuation, reactor pressure is controlled by remote-manual actuation of the SRVs. Control range during the first hour is 7.52 MPa (1090 psia) to 6.31 MPa (915 psia). After 1 hour the reactor is depressurized to about 0.79 MPa (115 psia) at a rate consistent with the 55.6°C/h (100°F/h) cooldown rate limit. After depressurization, pressure is controlled between 0.62 MPa (90 psia) and 0.86 MPa (125 psia).
   Manual sequential SRV action is used to spread SRV discharge around the suppression pool and thereby avoid localized heating of the pool.
4. The assumed reactor system leakage into the drywell is 0.0089 m^3/s (4 gpm).

7.3.2 Normal Recovery — Results

Results for normal recovery are shown on Figs. 7.1 through 7.9. Each system variable is discussed below.

7.3.2.1 Reactor vessel pressure. Figure 7.1 shows reactor vessel pressure. During the first minute, reactor pressure cycles between 7.72 MPa (1120 psia) and 7.38 MPa (1070 psia) on automatic SRV actuation. After 1 min. the operator opens one SRV to lower pressure to 6.31 MPa (915 psia). Since the HPCI is running, pressure continues to decrease to below 6.21 MPa (900 psia) until the HPCI is shut down. During the first hour a single SRV is cycled open and shut seven times to keep pressure in the desired range.

After one hour, the depressurization begins. Initially, intermittent operation of one SRV is sufficient to meet the target depressurization rate [i.e., 55.6°C (100°F)/h]. As pressure decreases, the rate of decrease begins to slow until finally an additional SRV is opened. For choked flow of dry steam, the mass flow rate is linearly proportional to reactor pressure . Therefore an extra valve is required at the lower pressure. When depressurization is complete, the operator controls pressure between 0.62 and 0.86 MPa (90 and 125 psia).

After depressurization, the steam flow through one SRV is nearly equal to the core steaming rate, resulting in much slower pressure change. Under these conditions, the most rapid pressure change is caused by the RCIC system. When RCIC comes on the 600 gpm injection of cold water tends to lower the core steaming rate, causing pressure to decrease. This effect can be seen in Fig. 7.1. At 256 min., the RCIC comes on, reducing pressure until the open SRV is shut. When the RCIC is shut off 23 min. later, pressure increases rapidly until the SRV is reopened.

Steam flow from the reactor vessel is shown in Fig. 7.2. The large spikes are due to SRV actuation. Also included in the total steam flow are the smaller amounts of steam flowing to the RCIC and HPCI turbines when they are running.

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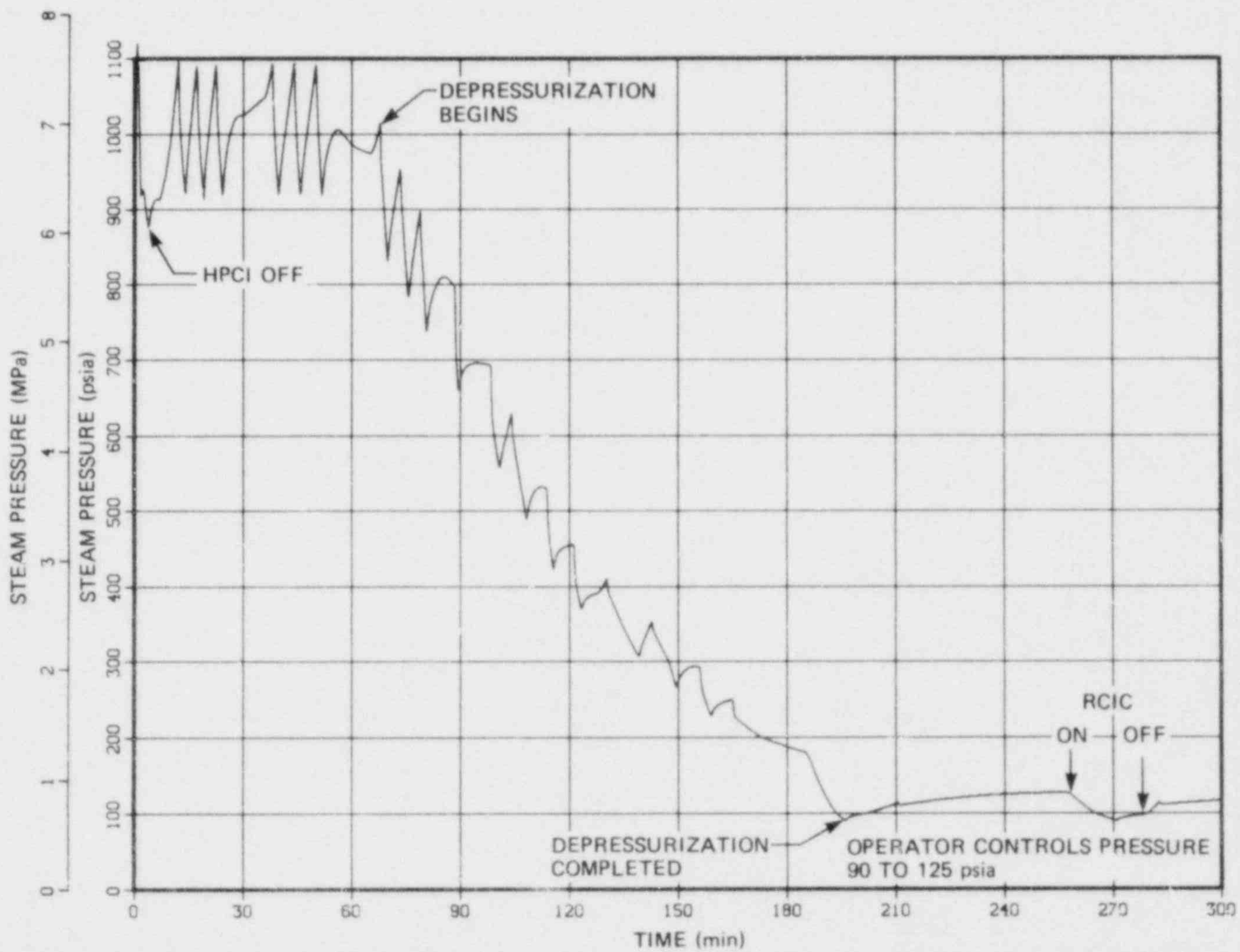

Fig. 7.1 Normal recovery after station blackout - reactor vessel pressure.

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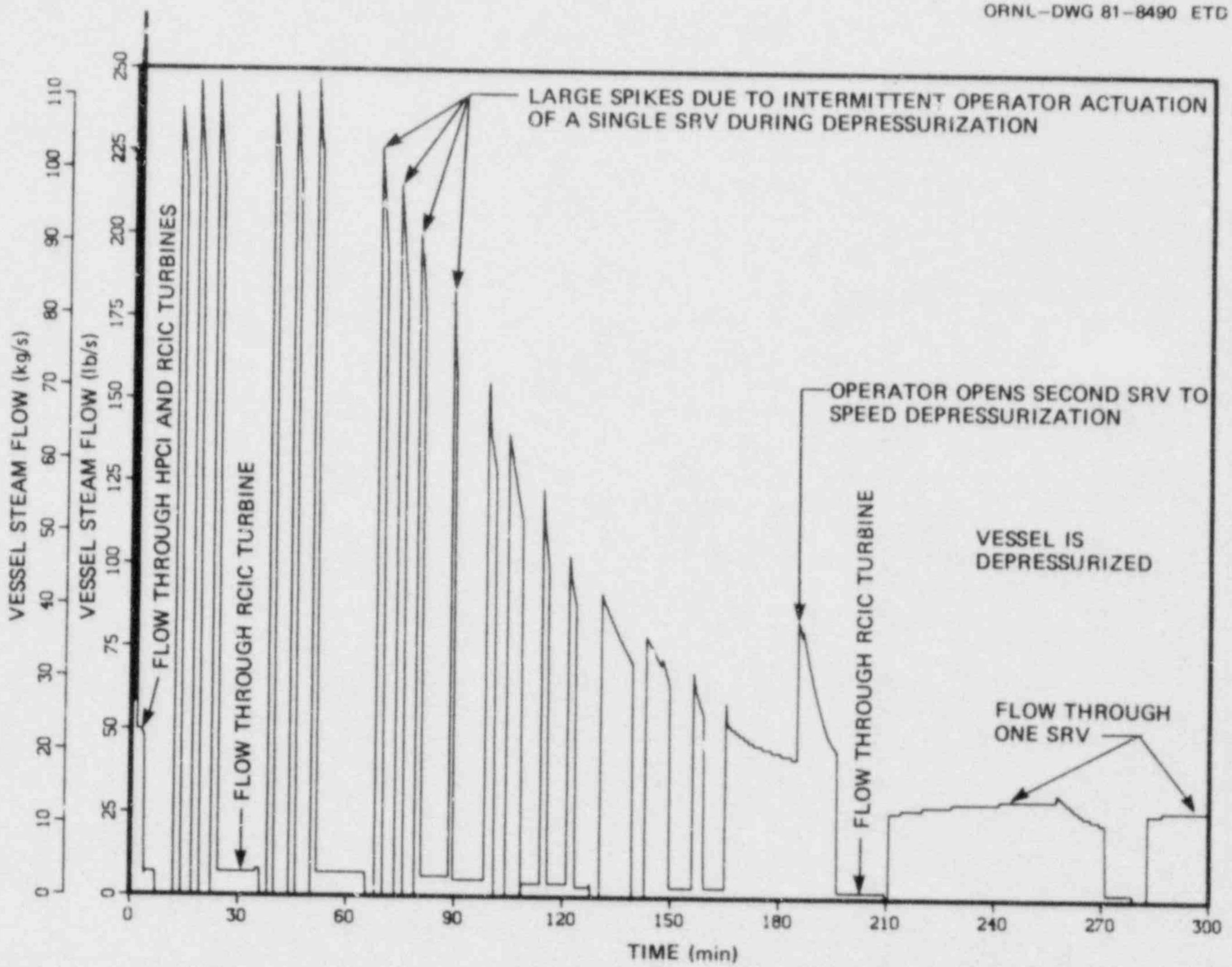

Fig. 7.2 Normal recovery after station blackout - vessel steam flow.

7.3.2.2 Reactor vessel level — normal recovery. Figure 7.3 shows vessel water level (distance above vessel zero) in the region outside the core outlet plenum, standpipes, and steam separators. This is the level measured by vessel level instrumentation and available in the control room. As discussed in Sect. 4, there is a corresponding level of two-phase mixture in and above the core. For an undamaged core in a mild transient such as Station Blackout, the level of steam/water mixture in and above the core will be higher than the downcomer level whenever downcomer level is below the top of the steam separators.

Downcomer level is initialized at 12.7 m (500 in.) and, at first, decreases rapidly until the operator initiates injection via the RCIC and HPCI systems. When level recovers to 13.7 m (540 in.), the operator shuts off the HPCI system, and when level reaches 14.7 m (578 in.), the RCIC injection is shut off. Level continues to increase due to the continuing heatup of the large quantity of cold water injected by the RCIC and HPCI systems and due to formation of additional voids in and above the core as the steaming rate increases. The level peaks at 15.2 m (600 in.). This is about 0.30 m (12 in.) above the top of the range of level indication available in the control room during Station Blackout.

Control Room level indication during a Station Blackout is limited to a range of 3.66 m (0 to 60 in.) with instrument zero corresponding to 13.41 m (528 in.) above vessel zero, which is the bottom of the reactor vessel. The basic cycle of RCIC actuation by the operator when level reaches an indicated 0.25 m (10 in.) and shut-off when level reaches an indicated 1.27 m (50 in.) is repeated eight times on Fig. 7.3. The fine structure is caused by action of the SRVs. When the SRV opens, vessel level first swells due to increased steaming, but then begins to decrease due to inventory loss. When the SRV shuts, level at first continues to decrease due to decrease in steaming rate, but then begins to recover due to heatup of subcooled liquid still within the downcomer.

Injection flow from the CST to the reactor vessel is shown in Fig. 7.4. The operator controls the RCIC and HPCI systems in an off-on mode in order to maintain vessel level within the desired control band as specified in Para. 7.3.1.2. The HPCI system is only run during the initial few minutes. The periods of RCIC system operation become less frequent as the decay heat subsides with time. Total injected flow (i.e., total amount of water taken from the CST) is shown in Fig. 7.5. The injected total increases rapidly at first during the brief period of combined HPCI and RCIC operation; the increase is much slower during the subsequent intermittent periods of RCIC operation.

7.3.2.3 Suppression pool level — normal recovery. When steam flows from the reactor vessel to the suppression pool, it is condensed by the colder pool water. Suppression pool level increases both because of the extra mass of the condensed steam and because of the temperature increase of the original pool water. Suppression pool level is shown in Fig. 7.6, referenced to instrument zero of the torus. The torus has a 9.45 m (31 ft) inside diameter and instrument zero is 0.1 m (4 in.) below the midplane. (A zero level indicates the torus is about half filled with water.) The pool level is normally maintained between −0.051 and −0.152 m (−2 and −6 in.) indicated; an initial pool level of −0.102 m (−4 in.) was assumed for this analysis. The pool condenses steam both from the

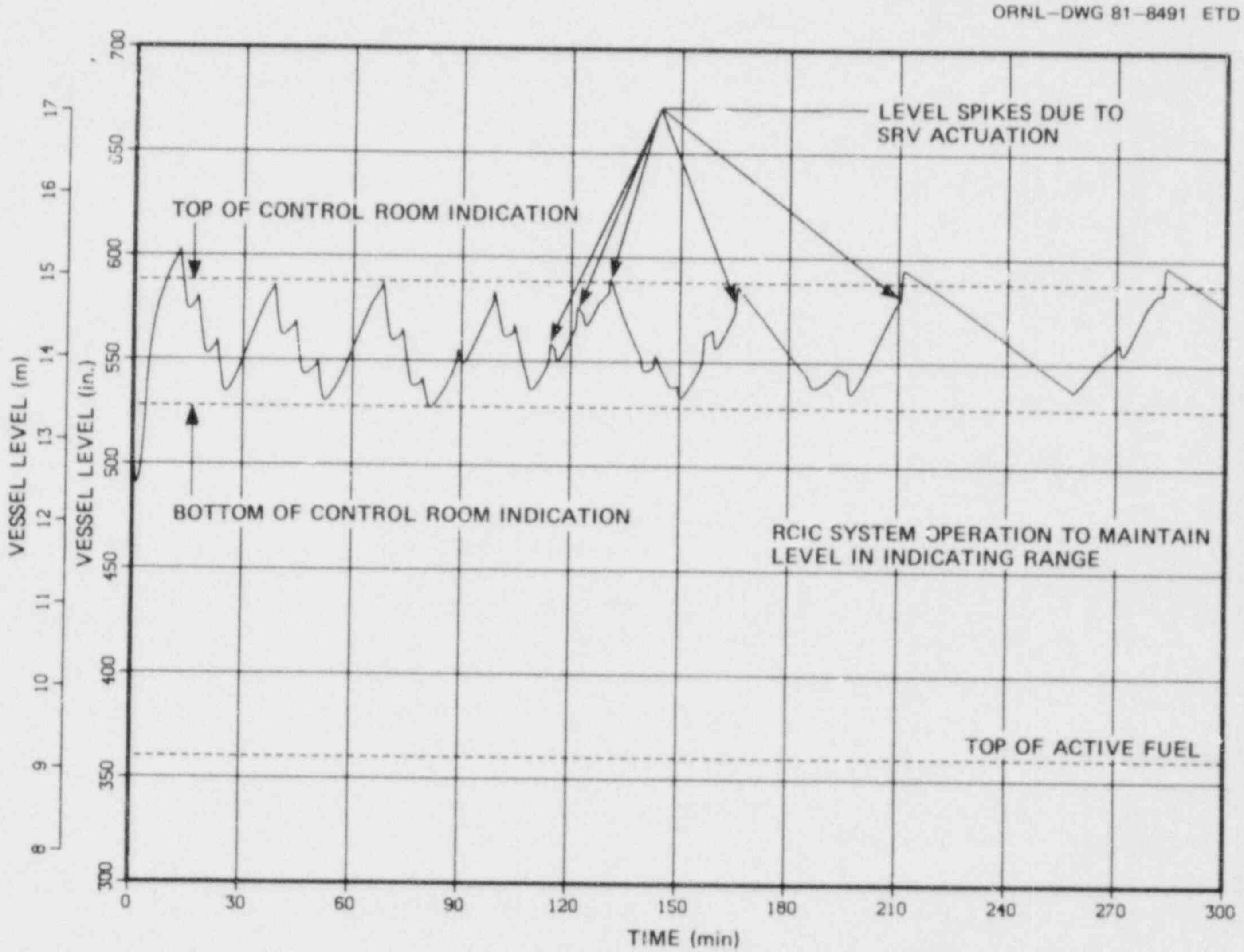

Fig. 7.3 Normal recovery after station blackout - reactor vessel level.

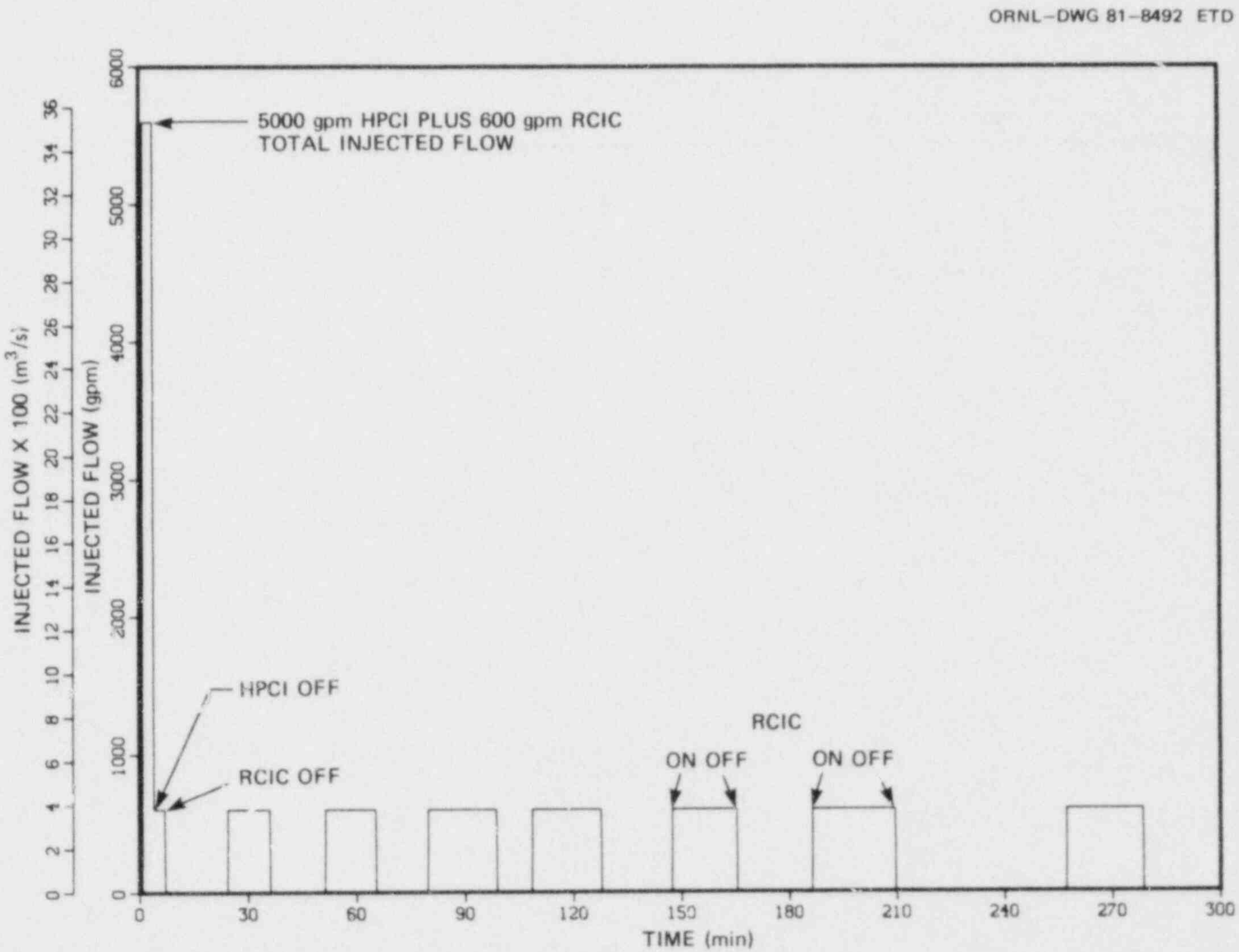

Fig. 7.4 Normal recovery after station blackout - injection flow.

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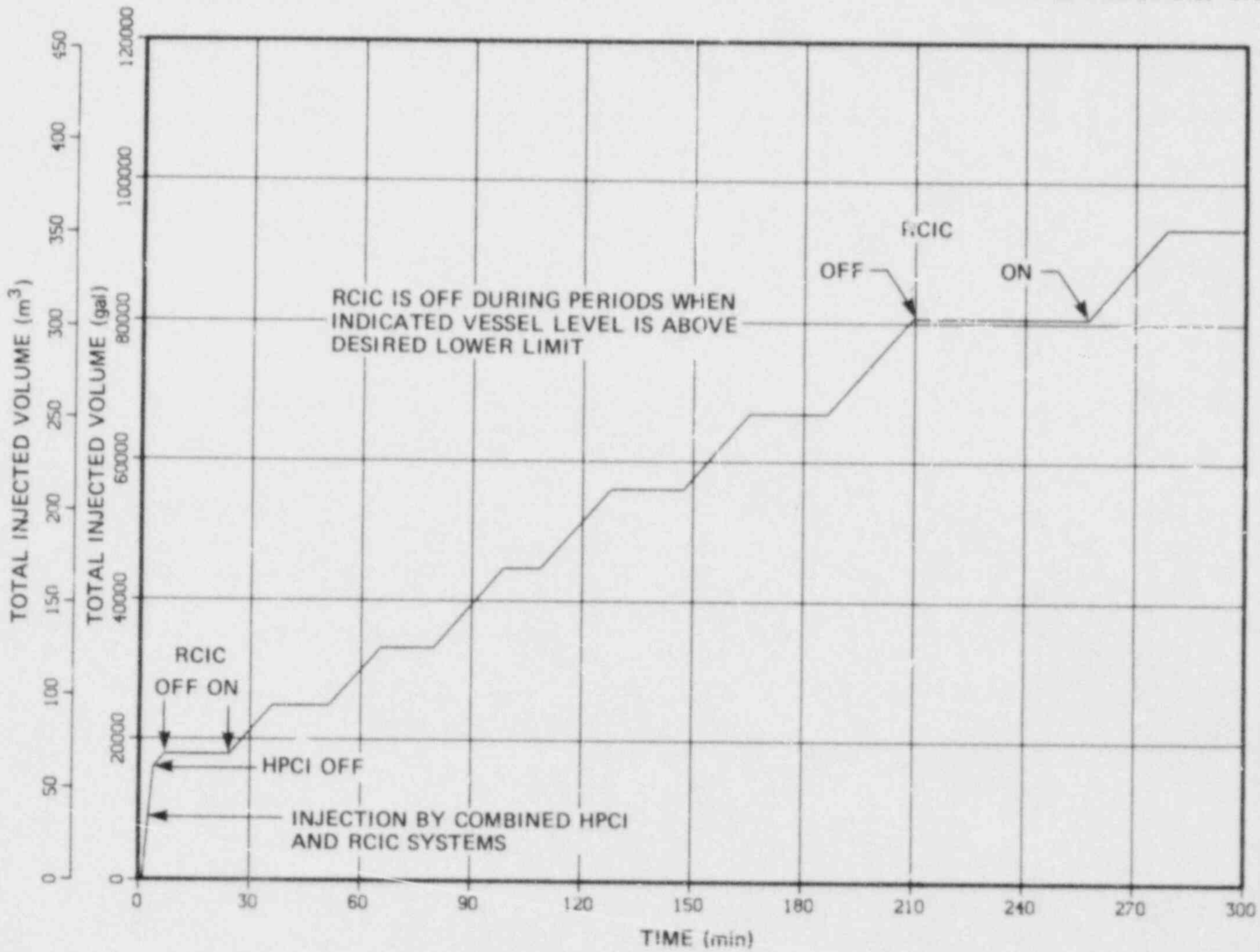

Fig. 7.5 Normal recovery after station blackout - total injected flow.

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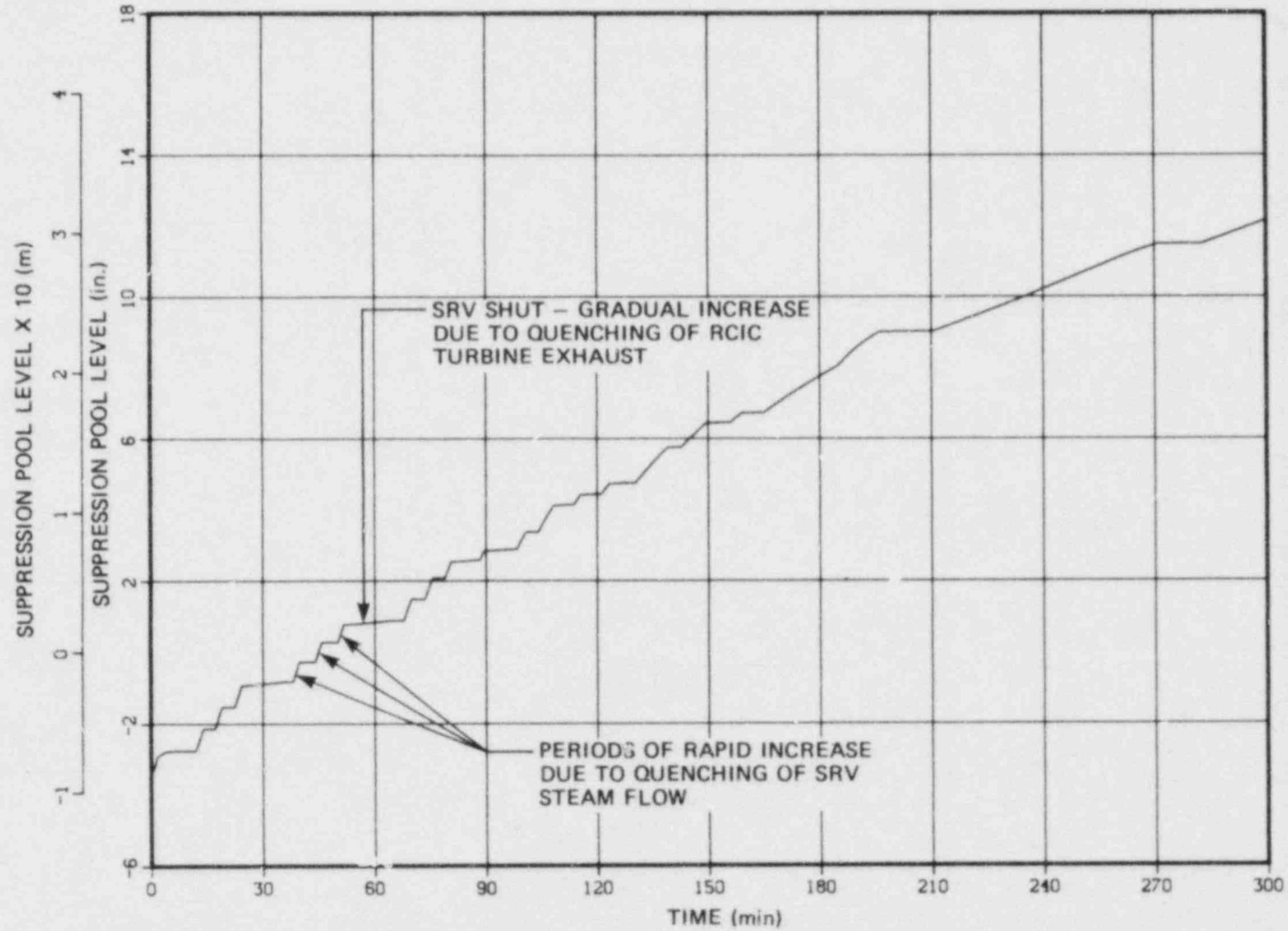

Fig. 7.6 Normal recovery after station blackout - suppression pool level.

SRVs and the RCIC and HPCI turbine exhaust. The effect of RCIC turbine exhaust shows up in Fig. 7.6 as a very slow background rate of increase against the dominant effect of the SRVs.

7.3.2.4 Suppression chamber temperatures — normal recovery. Suppression chamber temperatures are shown in Fig. 7.7. For all practical purposes, the average pool temperature responds solely to condensation of reactor vessel steam. Therefore, this curve has almost the same shape as the pool level curve. The pool atmosphere can be influenced by mass and heat transfer from the pool water and by mass transfer from the drywell atmosphere.

During the first 30 min. the suppression chamber atmospheric temperature increases rapidly because of mixing with the inflow of hotter gases from the expanding drywell atmosphere. Drywell temperature (Fig. 7.8) increases rapidly at first due to loss of the drywell coolers. As a result, the drywell atmosphere expands into the suppression chamber atmosphere through the pool water via the eight vent pipes, the single ring header, and the 96 downcomer pipes.

In the long term, mass transfer from the drywell ceases and the suppression chamber temperature continues to increase due to heat transfer and evaporation from the suppression pool surface.

7.3.2.5 Containment pressures — normal recovery. The suppression chamber and drywell pressures are shown in Fig. 7.9. The drywell and suppression chamber atmospheres are coupled in the following manner:

1. When suppression chamber pressure exceeds drywell pressure by 0.00345 MPa (0.5 psi), the vacuum breaker valves will open and allow flow of suppression chamber atmosphere into the drywell.
2. When drywell pressure exceeds suppression chamber pressure by more than about 0.0121 MPa (1.75 psi), this will clear the 1.22 m (4 ft) of water from the downcomer pipes and allow drywell atmosphere to bubble up through the pool water into the suppression chamber.

Immediately after the Station Blackout, drywell pressure increases rapidly due to the concurrent rapid heatup of the drywell atmosphere. Even after the drywell atmosphere reaches its peak temperature and starts down, the drywell pressure continues to increase under the influence of the assumed 0.0089 m^3/s (4 gpm) leak of hot reactor coolant, some of which flashes to steam. Throughout the first 2 hours, conditions in the drywell bring the suppression chamber pressure up at about the same rate by expansion (via the downcomers) into the suppression chamber. After the first two hours, suppression chamber pressure is increasing faster than dictated by the drywell because of increasing rates of heat transfer and evaporation from the overheated suppression pool. After about 190 min. the pressure is high enough to open the vacuum breakers and cause expansion of suppression chamber atmosphere back into the drywell.

7.3.2.6 Drywell temperature — normal recovery. Drywell atmosphere temperature response is shown in Fig. 7.8. The initial rate of increase is very rapid because the drywell coolers are lost at time zero, and about half of the 100% power drywell heat load is assumed to remain. The rate of increase becomes much slower when drywell temperature is high enough

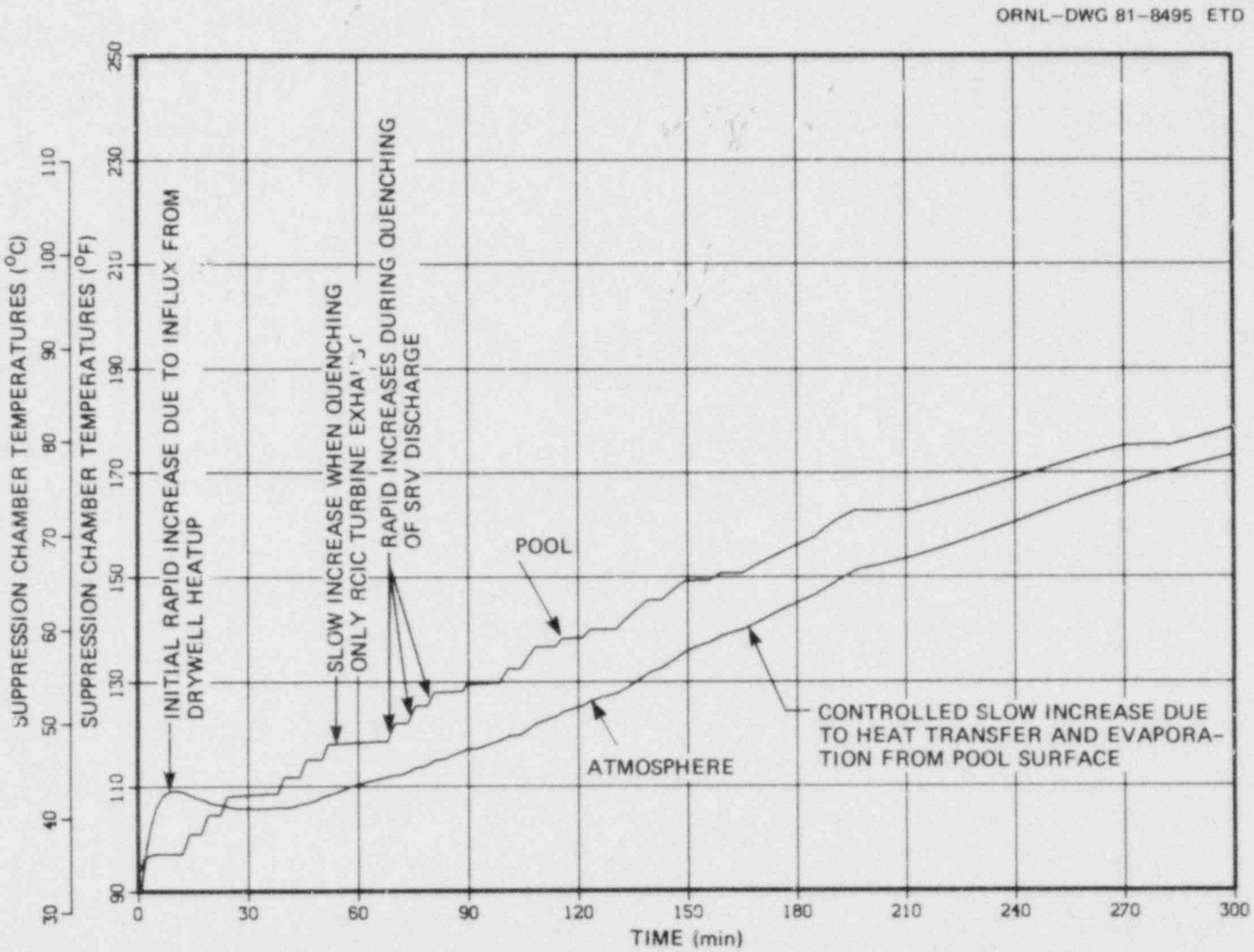

Fig. 7.7 Normal recovery after station blackout - suppression chamber temperatures.

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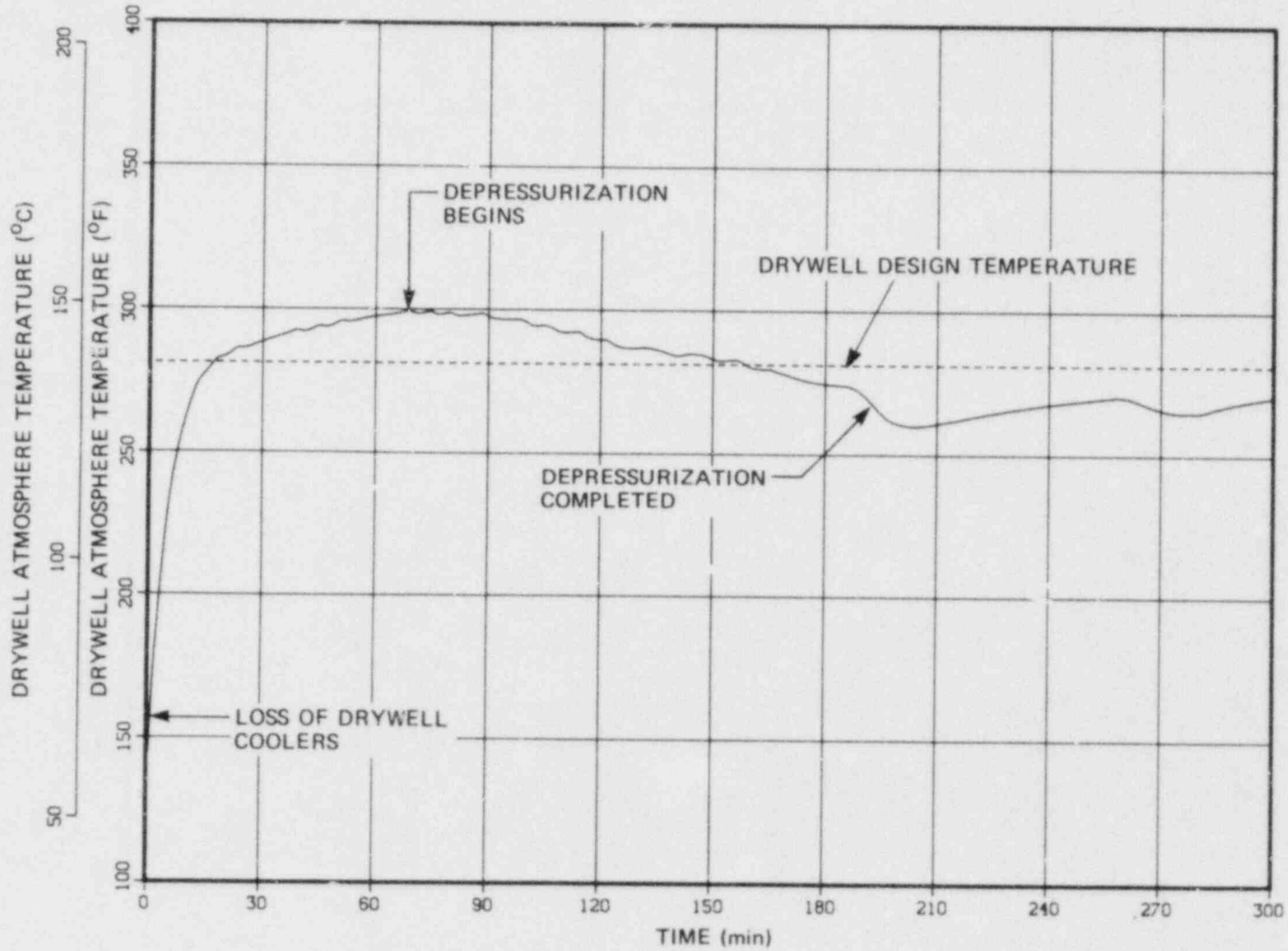

Fig. 7.8 Normal recovery after station blackout - drywell temperatures.

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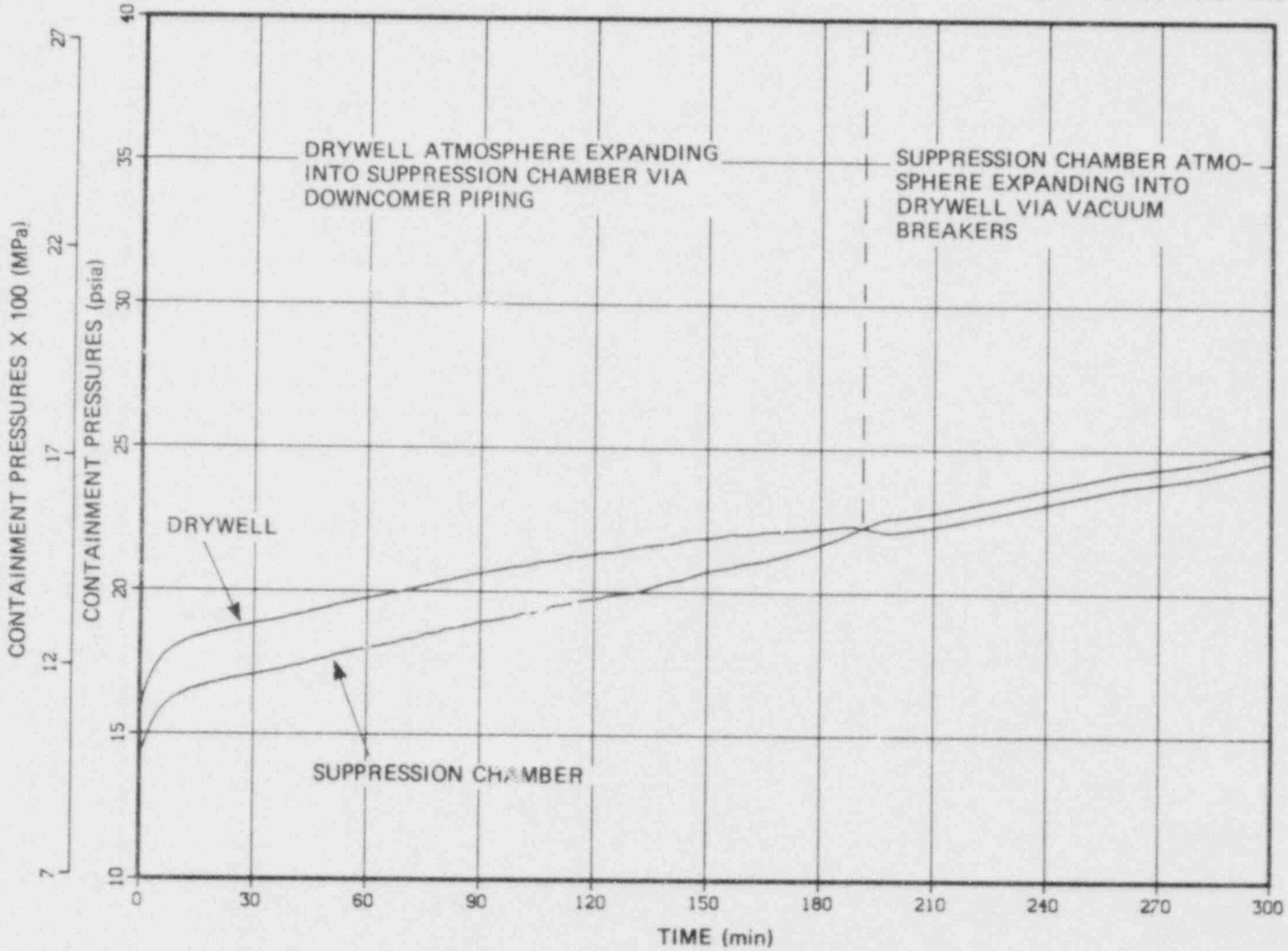

Fig. 7.9 Normal recovery after station blackout - containment pressures.

to establish natural convection heat transfer to the drywell liner (see discussion in Sect. 4). Drywell temperature begins decreasing after about 60 min. because a reactor coolant system (RCS) depressurization is begun at this point. The depressurization reduces saturation temperature within the RCS, hence surface temperatures of the reactor vessel and piping.

7.4 Loss of 250 vdc Batteries

7.4.1 Loss of 250 vdc Batteries — Assumptions

Assumptions specific to programming and input preparation for the Loss of 250 vdc Batteries calculation include:

1. When the batteries are lost, injection capability is lost; in addition, the SRVs can be opened only by automatic actuation.
2. Under automatic actuation, the lowest set SRV will open at about 7.72 MPa (1120 psia) and close at about 7.38 MPa (1070 psia).
3. The calculation begins 4 h after the Station Blackout and proceeds to about 8 h.
4. Initial values of system parameters, except reactor vessel level, are equal to those calculated at the 4 h point in the normal recovery results (see Figs. 7.1 through 7.9). Reactor vessel level is assumed to begin at 14.2 m (558 in.), the midpoint of the control range assumed for the normal recovery calculation.

7.4.2 Loss of 250 vdc Batteries — Results

Results for the Loss of 250 vdc Batteries case are shown in Figs. 7.10 through 7.16. Each system variable is discussed below.

7.4.2.1 Reactor vessel pressure. Reactor vessel pressure is shown in Fig. 7.10. During the first 90 min., pressure climbs slowly from the 0.86 MPa (125 psia) initial value to the 7.72 MPa (1120 psia) actuation point of the lowest set SRV. The rate of increase is slow because the decay heat after 4 hours is small compared to the thermal inertia of liquid water within the vessel.

After repressurization, the pressure is controlled between 7.72 MPa (1120 psia) and 7.38 MPa (1070 psia) by one SRV, actuating automatically. As shown in Fig. 7.11, no steam flows from the vessel until pressure reaches the SRV automatic actuation setpoint.

7.4.2.2 Reactor vessel level. Reactor vessel level is shown in Fig. 7.12. No injection is provided at any time in this transient, since DC power is assumed lost.

The initial response of level is a brief but rapid decrease caused by closing of the previously open SRV at time zero. Within several minutes level begins slowly rising and peaks at about 15.88 m (625 in.). Level swells during repressurization because the large mass of water in the vessel is heated from about 165°C (330°F) to about 288°C (550°F) with about a 25% increase in specific volume, and there is no coolant loss (other than the assumed leakage) during repressurization.

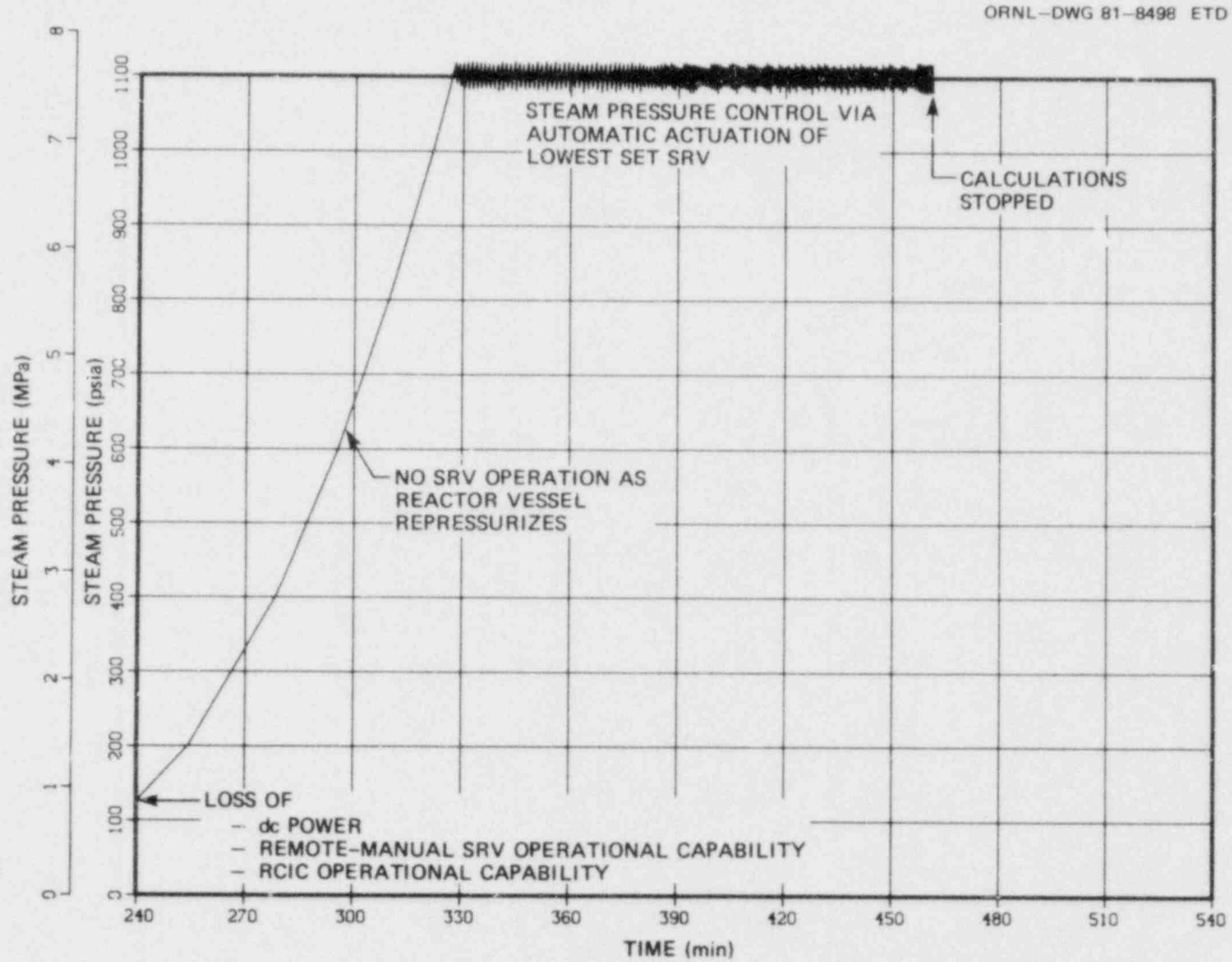

Fig. 7.10 Loss of 250 VDC batteries - reactor vessel pressure.

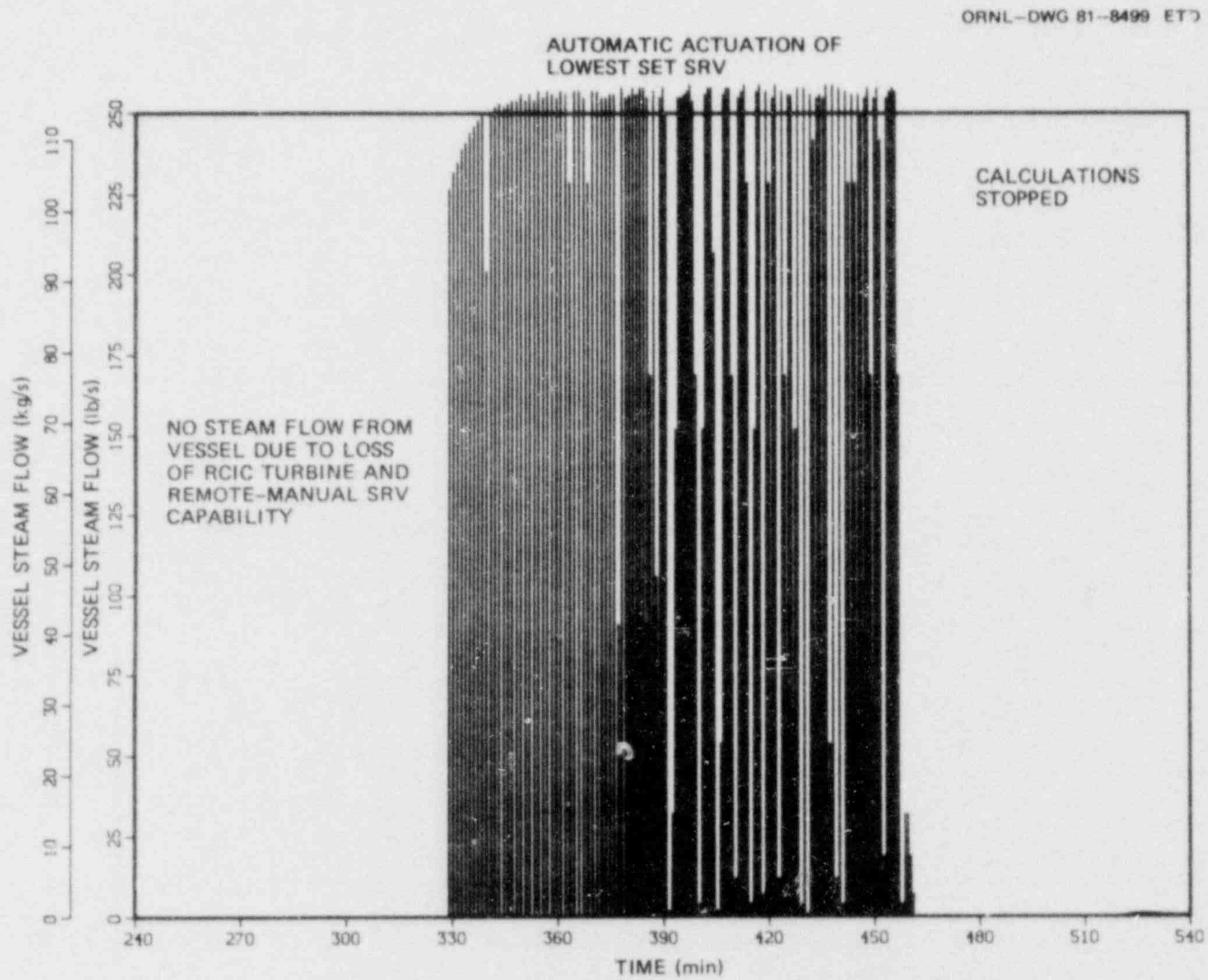

Fig. 7.11 Loss of 250 VDC batteries - vessel steam flow.

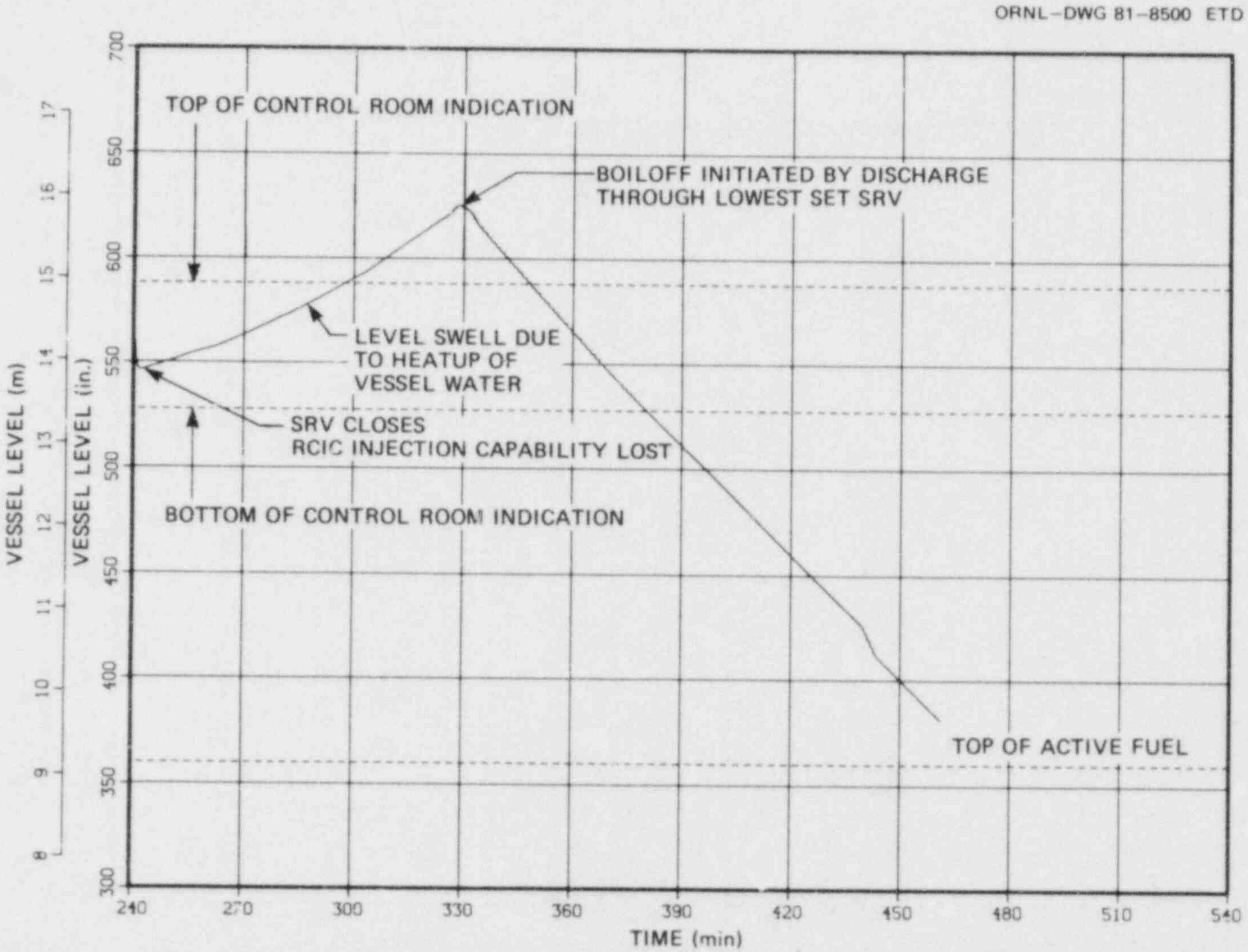

Fig. 7.12 Loss of 250 VDC batteries - vessel level.

After repressurization, the SRVs begin to discharge steam so level decreases throughout the remainder of the transient. Fuel is beginning to be uncovered at the end of the transient.

7.4.2.3 Suppression pool level. The suppression pool level is shown in Fig. 7.13. During the first 90 min., there is no SRV discharge, so pool level is constant. After SRV discharge resumes, the pool level begins increasing steadily.

7.4.2.4 Suppression chamber temperatures. Figure 7.14 shows suppression pool and suppression chamber atmosphere temperature. Pool temperature, like pool level, is constant for the first 90 min. and then begins a steady increase. Atmosphere temperature continues to rise during the first hour due to heat transfer and evaporation from the pool surface. Atmosphere temperature rise accelerates slightly between the first and second hours due to an influx from the expanding drywell atmosphere. After about 2 hours, the mass interchange ceases, and the suppression chamber atmosphere temperature is responding solely to the increasing pool temperature.

7.4.2.5 Containment pressures. Figure 7.15 shows containment pressures. During the first 90 min., drywell pressure increases rapidly due to heatup of the drywell atmosphere during repressurization. This brings drywell pressure from slightly below suppression chamber pressure to about 0.0125 MPa (1.8 psi) above, allowing some drywell atmosphere to expand into the suppression chamber. After about 2 hours, the drywell and suppression chamber pressures are increasing independently at about the same rate, with little mass interchange.

7.4.2.6 Drywell atmosphere temperature. Drywell atmosphere temperature is shown in Fig. 7.16. During the repressurization, the coolant temperature within the reactor vessel increases by over 111 C° (200 F°), thereby increasing the drywell heat source. Atmosphere temperature rises to new quasi-equilibrium values, with increased natural circulation heat transfer from the atmosphere to the drywell liner. After repressurization, the coolant temperature becomes approximately constant, and the temperature of the drywell atmosphere rises more slowly at a rate limited by the thermal inertia of the liner.

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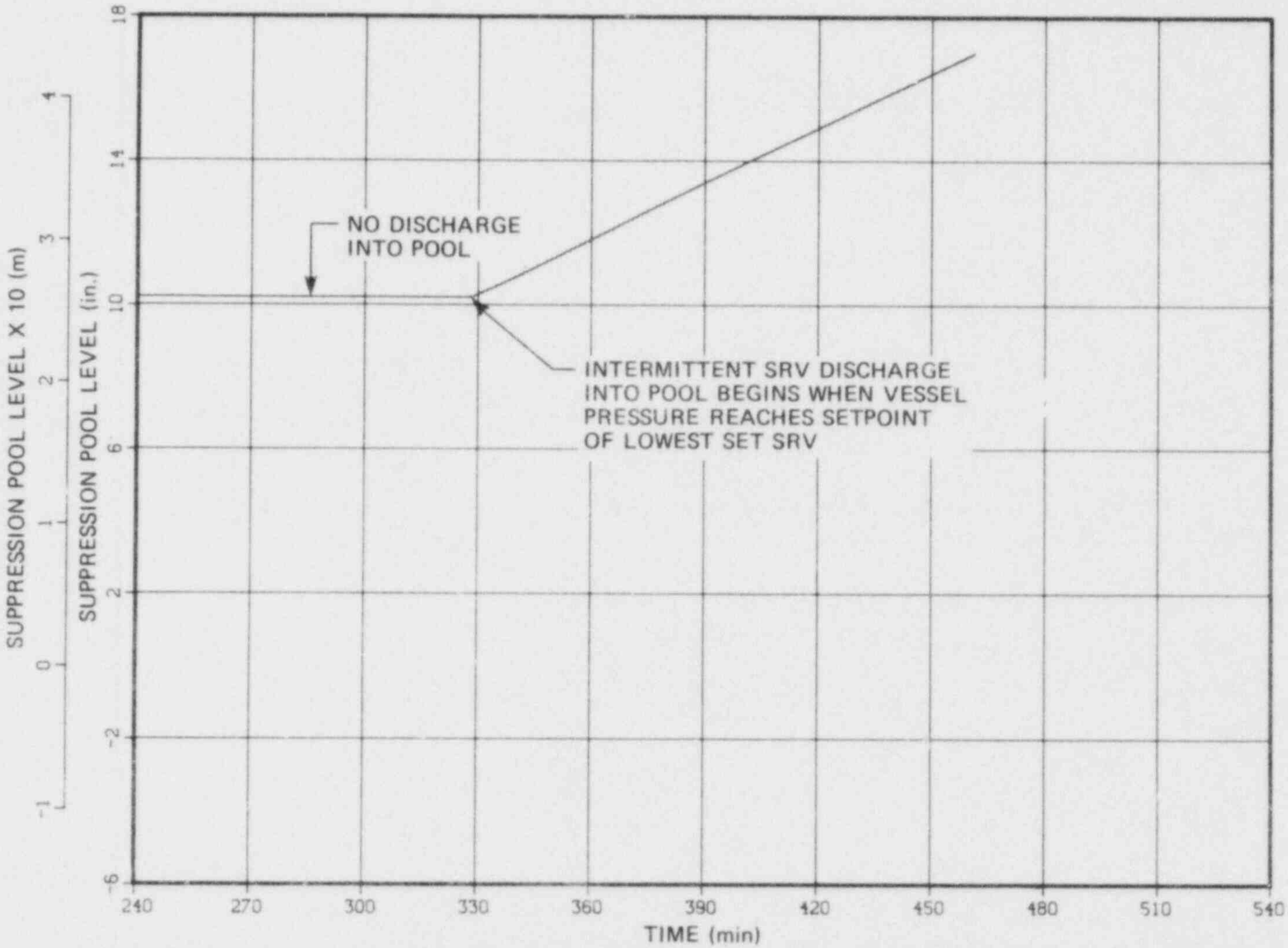

Fig. 7.13 Loss of 250 VDC batteries - suppression pool level.

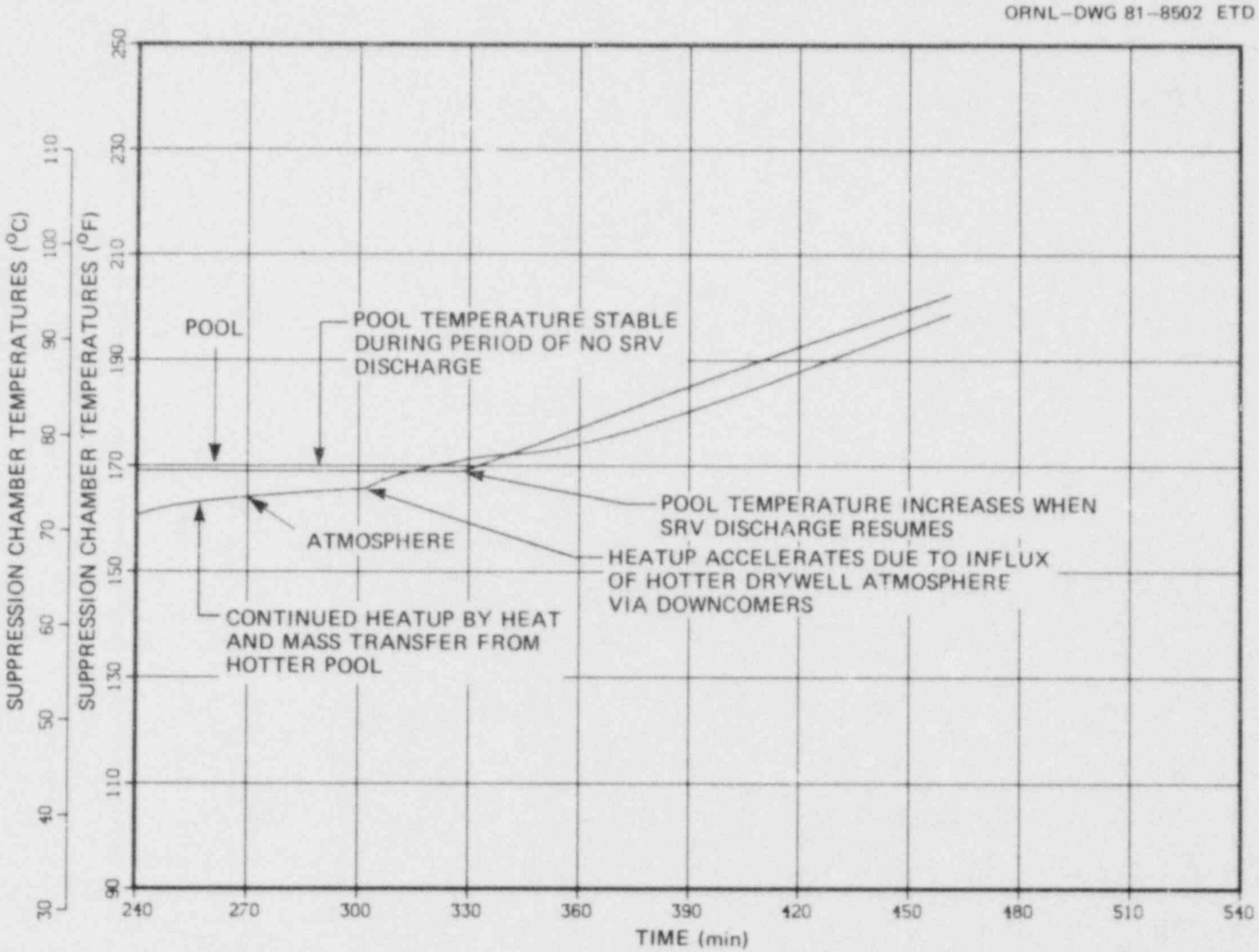

Fig. 7.14 Loss of 250 VDC batteries - suppression chamber temperatures.

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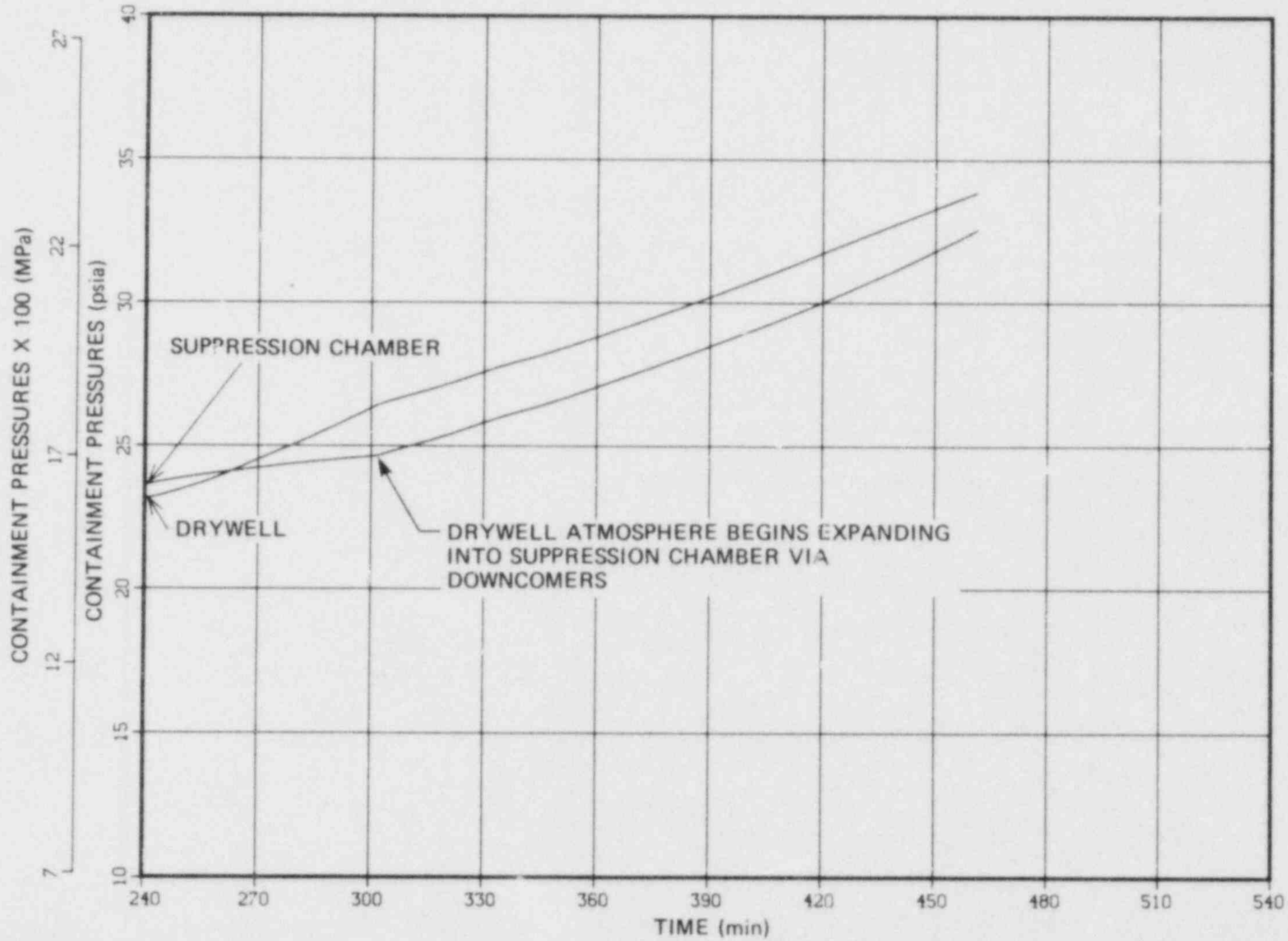

Fig. 7.15 Loss of 250 VDC batteries – containment pressures.

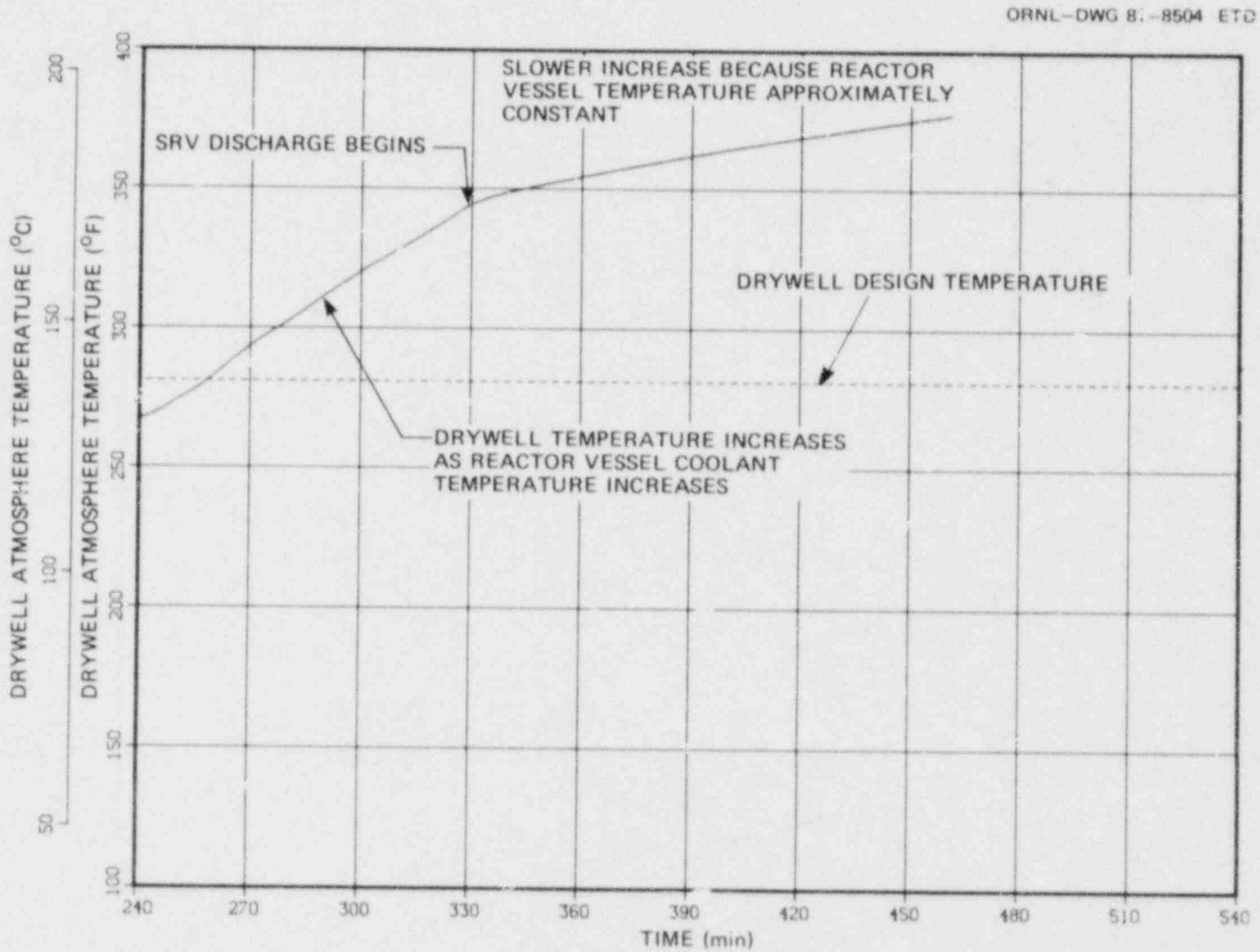

Fig. 7.16 Loss of 250 VDC batteries - drywell atmosphere temperature.

8. FAILURES LEADING TO A SEVERE ACCIDENT

As discussed in Sect. 3, control can be maintained over reactor vessel pressure and water level at Browns Ferry Unit 1 for a significant period of time during a Station Blackout provided the installed equipment operates as designed and the operator takes the required actions. The first objective of this Section is to briefly discuss the other sequences which might occur given significant secondary independent equipment failures or improper operator actions. Subsequently, the equipment failures which have the potential to directly convert the relatively stable period of level and pressure control of the isolated reactor vessel during the early stages of a Station Blackout into a Severe Accident will be discussed in detail.

The several sequences which are considered most probable in a Station Blackout during the period in which DC power remains available are displayed in the "event tree" of Fig. 8.1. Each of these sequences is discussed below.

Sequence 1: Following the complete loss of AC power, the operator takes remote-manual control of the RCIC system and maintains the reactor vessel water level within the limits of the available Control Room indication. Since the RCIC System is being used for level control, the status of the HPCI system is not a factor in this sequence. The operator also controls the reactor vessel pressure by remote-manual operation of the primary relief valves and takes action to begin depressurization of the reactor vessel within one hour; this lowers the temperature of the saturated liquid within the vessel and thereby decreases the driving potential for heat transfer into the drywell atmosphere. As a result of this depressurization, the maximum average ambient temperature in the drywell is limited to 148.9°C (300°F), as discussed in Sect. 3. The operator will be able to maintain reactor vessel level and pressure control for as long as DC power is available from the Unit Battery. This is considered to be the most realistic sequence should a Station Blackout occur, and served as the basis for the plots of thermohydraulic parameters as functions of time which were presented in Sect. 7. This sequence will also serve as the starting point for the degraded accident behavior discussed in Section 9 in which the Station Blackout develops into a Severe Accident when DC power is lost so that cooling water can no longer be injected into the reactor vessel.

Sequence 2: This sequence differs from Sequence 1 only in that the operator does not act to depressurize the reactor vessel. The average ambient drywell temperature continues to increase beyond the one-hour point, reaching 176.7°C (350°F) about four hours after the inception of the Station Blackout (cf. Curve A of Fig. 3.3). This excessive temperature existing over a period of several hours can cause serious damage to the equipment located within the drywell and a breaching of the primary containment due to failure of the drywell electrical penetration assembly seals. If the DC solenoid operators for the primary relief valves fail because of the excessive temperature, the operator will lose remote-manual relief valve control. If the solenoid operators for the inboard Main Steam Isolation Valves also fail, the operator will be unable to depressurize the reactor vessel through the primary relief valves even after AC

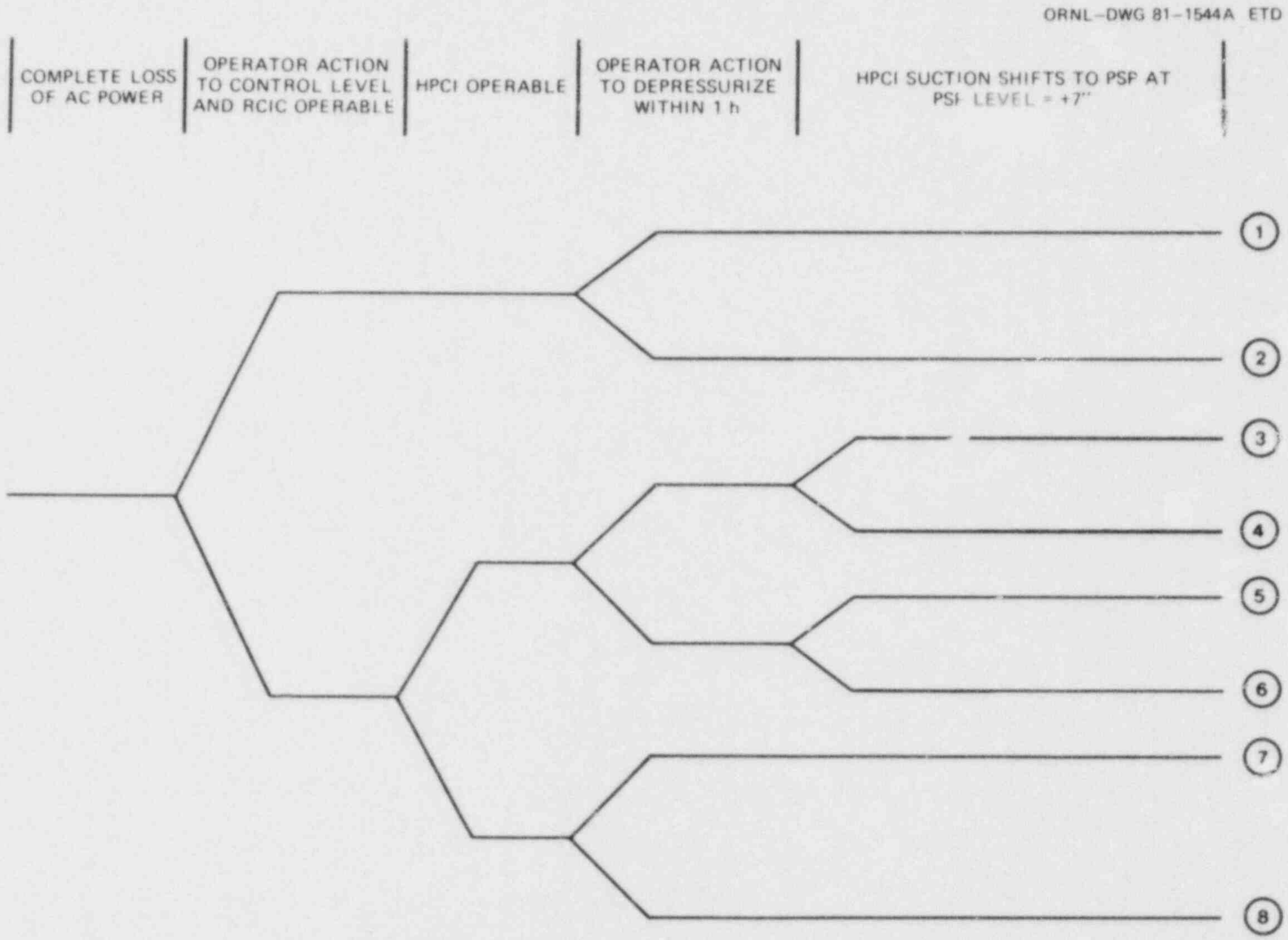

Fig. 8.1 Event tree for initial phase of a station blackout in which DC power remains available.

power is restored. There is no reason to accept these risks to the drywell integrity and installed equipment; the operator should take action to depressurize, as in Sequence 1.

Sequence 3: Either the RCIC system is inoperable or the operator does not choose to employ it for long-term reactor vessel level control. The HPCI System is operable, and either automatically functions as necessary to cycle the reactor vessel level between 12.1 and 14.8 m (476 and 582 in.) above vessel zero, or is remote-manually controlled by the operator to maintain the water level within the indicating range, 13.4 to 14.9 m (528 to 588 in.). The operator controls the reactor vessel pressure by remote-manual operation of the relief valves and takes action to begin depressuration within one hour. Up to this point, this sequence is similar to Sequence 1, except that the injection of condensate storage tank water into the reactor vessel is by the HPCI System instead of the RCIC System.

When the indicated water level in the pressure suppression pool increases to 0.18 m (+7 in.), the HPCI pump suction is automatically shifted from the condensate storage tank to the pressure suppression pool. Depending on the initial suppression pool level, the necessary amount of water to cause this shift will have been transferred from the reactor vessel via the primary relief valves at some time between two and four hours after the inception of a Station Blackout. Since the lubricating oil for the HPCI turbine is cooled by the water being pumped, and the average suppression pool temperature will be about 71.1°C (160°F) at this point, an early failure of the HPCI system is threatened.

Sequence 4: This sequence is the same as Sequence 3, except that the operator has recognized the difficulties with the elevated suppression pool temperature and has taken action to prevent the shift in HPCI pump suction. The operator will be able to maintain reactor vessel level and pressure control for as long as DC power remains available from the Unit Battery. The continuing increase in suppression pool level will not cause any significant problems during the remaining period in which DC power would remain available.

It should be noted that pressure suppression pool temperature instrumentation would be inoperable during a Station Blackout. Operator training and the Station Blackout Emergency Operating Instruction (which does not now exist) must provide for operator understanding of the need to prevent a shift of the HPCI pump suction to the overheated pressure suppression pool.

Sequence 5: This sequence represents the case of Station Blackout with no operator action and no secondary independent equipment failures. The HPCI System automatically cycles the reactor vessel water level between the HPCI system initiation point of 12.1 m (476 in.) and the HPCI turbine trip point of 14.8 m (582 in.) above vessel zero. Over the long term, one primary relief valve would repeatedly operate as necessary to maintain the reactor vessel pressure between about 7.722 and 7.377 MPa (1105 and 1055 psig).

There are three problem areas with this case of no operator action:

1. the difficulties with high drywell ambient temperature as discussed in connection with Sequence 2,
2. the increased probability of loss of injection capability after the undesirable shift of the HPCI pump suction to the overheated pressure suppression pool as discussed in connection with Sequence 3, and

3. the decrease in suppression pool effectiveness caused by the highly localized heating of the pool water surrounding the discharge of the one repeatedly operating relief valve. The possible consequences are discussed in Appendix D.

Sequence 6: This sequence differs from Sequence 5 only in that the HPCI pump suction is maintained on the condensate storage tank, either by operator action, or by fortuitous failure of the pertinent HPCI system logic. This reduces the probability of HPCI system failure due to inadequate lube oil cooling during the period of Station Blackout in which DC power is still available. The difficulties with high drywell ambient temperature and localized suppression pool heating as discussed in connection with Sequence 5 remain.

Sequence 7: The RCIC system is not available because of equipment failure, or lack of operator action. The HPCI system has failed because of a system malfunction. The represents the worst case of loss of injection capability while DC power remains available. A boiloff of the initial reactor vessel water inventory begins immediately after the complete loss of AC power, leading to a relatively quick core uncovery and subsequent meltdown. In this sequence, the core uncovery is hastened by a depressurization of the reactor vessel either by means of a stuck-open relief valve or because of inappropriate operator action; the depressurization increases the rate of loss of the irreplaceable reactor vessel water inventory.

Sequence 8: This sequence is the same as Sequence 7, except that there is no reactor vessel depressurization. The uncovery of the core would begin about one-half hour after the loss of AC power. The plant response to Sequences 7 and 8 will be discussed further in Section 9.

The preceding discussion of alternate sequences again shows that a Station Blackout at the Browns Ferry Nuclear Plant will not evolve into a Severe Accident as long as reactor vessel water injection capability is maintained. The remainder of this section contains a discussion of the methods by which this injection capability might be lost. There are three main areas of concern for the viability of the injection systems: Station Blackout induced direct failure of the HPCI and RCIC systems, a stuck-open relief valve which might reduce reactor vessel pressure below that necessary for HPCI and RCIC turbine operation, and the ultimate loss of DC power. Each of these will be discussed in turn.

8.1 Induced Failure of the HPCI and RCIC Systems

The design and the principles of operation of the HPCI and RCIC systems are discussed in Appendices E and F, respectively. Each of these systems has the capacity to provide the necessary reactor vessel water level control during a Station Blackout; both must fail to cause a total loss of injection capability.

The failure modes considered in this Section will be limited to those induced by the conditions of a Station Blackout. Independent secondary failures, whose occurrence during a Station Blackout would be purely coincidental, have been considered in other studies[16] and will not be discussed here.

The first threat to the continued operation of the HPCI and RCIC systems under Station Blackout conditions is due to the loss of the Reactor Building cooling and ventilation systems. The major components of the HPCI and RCIC systems are located in the basement of the Reactor Building, where the average temperatures during a Station Blackout would be significantly affected by the temperature of the pressure suppression pool. The HPCI and RCIC systems are designed for continuous operation at an ambient temperature of 64.4°C (148°F). As was shown in Section 7, the average pressure suppression pool temperature would reach 71.1°C (160°F) after about three hours under Station Blackout conditions. However, it is expected that the ambient temperature in the vicinity of the HPCI and RCIC systems would significantly lag the increasing suppression pool temperature; it is estimated that the design temperature of 64.4°C (148°F) would not be exceeded for at least four hours. In any event, exceeding the design ambient temperature would not assure failure of the HPCI and RCIC systems.

A more probable cause of HPCI or RCIC system failure is the automatic isolation of these systems due to the tripping of the temperature sensing circuits designed to detect steam leaks in the system piping. As described in Appendices E and F, each of these systems has a set of four trip logics with four temperature sensors per logic. The 16 sensors are physically arranged in four groups placed near the HPCI or RCIC equipment, with the four sensors in each group arranged in a one-out-of-two taken twice trip logic. If the trip setpoint of 93.3°C (200°F) is reached, an automatic closure signal is sent to the RCIC or HPCI primary containment inboard and outboard steam supply valves, and the system turbine is tripped. Since the inboard steam supply valve for each of these systems is AC-motor-operated, it would not close, but the system would be effectively isolated due to closure of the outboard (DC-motor-operated) steam supply valve.

As shown in Section 7, it is expected that the RCIC turbine would be operated only intermittently during a Station Blackout, while the HPCI system would serve only as a backup in the event of RCIC system failure. Also, the RCIC equipment is not located in a closely confined space, and some natural convection within the reactor building will certainly occur. Nevertheless, it is conceivable that the local temperature in the vicinity of the steam leak detection sensors could reach 93.3°C (200°F) during RCIC system operation after the average space temperature has increased to over 60°C (140°F). If this occurs, the resulting RCIC system isolation signal can be overriden in the auxiliary instrumentation room and the steam supply valves reopened and the turbine trip reset. Thereafter, the HPCI system and the RCIC system, which are located on opposite sides of the suppression pool torus, might be alternately used for reactor vesssel level control so as to reduce the heatup in the vicinity of the RCIC system.* Thus, the tripping of the temperature sensing circuits is not expected to lead to a total loss of injection capability.

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*The HPCI room has not been designed to permit natural circulation and the HPCI system has much more surface area for heat transfer into its surrounding space. Therefore, it is expected that the heatup associated with HPCI operation would be more severe than that associated with RCIC operation.

Another challenge to the viability of the HPCI and RCIC systems would be posed by the loss of control air pressure during a prolonged Station Blackout. The plant control air system is supplied from three air receivers with a combined capacity of 22.60 m^3 (798.0 ft^3). Under Station Blackout conditions, the air compressors which normally run intermittently as necessary to maintain the receiver pressure in the range of 0.66 to 0.86 MPa (80 to 105 psig) would be inoperable; the stored inventory of pressurized air would gradually be lost due to valve operations and leakage.

Control air is used in the HPCI and RCIC systems to operate the steam supply line drain isolation valves. In each of these systems, the two primary containment isolation valves in the steam line to the turbine are normally open so that the steam supply piping is kept at elevated temperatures; this permits rapid turbine startup on demand. The water formed by the steam which condenses in this line when the turbine is not operating is removed to the main condenser via a thermostatic steam trap. The two air-operated drain isolation valves close to prevent the flow of water from the steam trap to the main condenser when the primary containment is to be isolated; the closing signal is automatically generated on low reactor water level [12.10 m (476.5 in.) above vessel zero].

The steam supply line drain isolation valves fail closed on loss of control air, which would eventually occur during a prolonged Station Blackout. This should not cause serious difficulties for the RCIC system which would be run intermittantly so that the accumulation of water in the steam supply piping between runs would not be excessive. However, a substantial amount of water would collect in the HPCI steam supply piping if this system is not operated over a long period following the loss of control air pressure. Subsequent initiation of the HPCI system might cause turbine damage. However, it should be noted that the HPCI and RCIC turbines are two-stage, non-condensing Terry turbines, which are of very rugged construction and are designed for use under emergency conditions, and the operator could take action to run the HPCI turbine for short periods to clear the lines of water as necessary. Thus the loss of control air during a prolonged Station Blackout should not lead to a total loss of injection capability.

A third challenge to the viability of the water injection systems should affect the HPCI system only. This challenge would occur because of the HPCI system logic which provides for an automatic shifting of the HPCI pump suction from the condensate storage tank to the pressure suppression pool when the indicated pressure suppression pool level reaches 0.18 m (+7 in.). As previously discussed, this would occur between two and four hours after the inception of a Station Blackout, when the pressure suppression pool temperature has increased to about 71.1°C (160°F). Since the turbine lubricating oil is cooled by the water being pumped, and the oil cooler is designed for a maximum inlet water temperature of 60.0°C (140°F), the oil would become overheated, possibly leading to failure of the turbine bearings. This can be avoided if the operator takes action to maintain suction on the condensate storage tank for any HPCI system operation. There is an ample supply of condensate storage tank water available for injection, and the continuing increase in pressure suppression pool level will in no way threaten the continued operation of the injection systems during the remaining period in which DC power remains available.

The effects of increased temperature in the vicinity of the HPCI and RCIC turbines, of the loss of control air, and of an automatic shift of the HPCI pump suction to an overheated pressure suppression pool under Station Blackout conditions have been examined in this subsection. None of these Station-Blackout-induced events are expected to lead to a total loss of injection capability by means of a direct failure of the HPCI and RCIC systems, but the operator should be aware of the potential for failure and be prepared to take the appropriate corrective actions when required.

8.2 Stuck-Open Relief Valve

Since both the RCIC and the HPCI steam turbines are driven by steam from the reactor vessel, question arises as to whether enough steam pressure would be maintained in the event of a stuck-open relief valve to permit the operation of these injection systems. Each of these systems will automatically isolate on low reactor vessel pressure to prevent the escape of large quantities of steam to the atmosphere through the gland seals of an immobile turbine; the trip setpoint is 0.793 MPa (115 psia) for the HPCI turbine and 0.448 MPa (65 psia) for the RCIC turbine. The isolation is similar to that which occurs if the sensed equipment space temperature reaches 93.3°C (200°F) as previously described.

The reactor vessel pressure as a function of time during a Station Blackout in which one relief valve sticks open as a result of the initial relief valve liftings is shown in Fig. 8.2. The points for this figure were calculated using the computer program described in Sect. 4; this program was also used in the development of the figures included in Sect. 7.

The reactor vessel steam pressure shown in Fig. 8.2 decreases very rapidly during about the first six minutes of the Station Blackout. This rapid decrease is due both to the stuck-open valve and to the injection of relatively cold water from the condensate storage tank by the combined operation of the HPCI and RCIC systems at a rate of 0.353 m^3/s (5600 GPM). The operator uses both of these systems as shown in Fig. 8.3 as he strives to bring the reactor vessel level back into the Control Room indicating range. Once the level has been restored into the operating range, the operator turns off the HPCI system; the subsequent pressure decrease is at a much slower rate. (The mass flow through the relief valves is approximately proportional to reactor vessel pressure, as explained in Section 4).

When the reactor vessel indicated level is near the top of the indicating range, the operator turns off the 0.038 m^3/s (600 GPM) RCIC system; this occurs at about 29 minutes into the Station Blackout. The result as shown on Fig. 8.2 is that the reactor vessel pressure begins to increase; more steam is being generated by decay heat within the vessel than the stuck-open relief valve can remove. At about time 48 minutes, the indicated reactor vessel level has decreased to the point where the operator again turns on the RCIC system. This pattern is continued over the five-hour period shown in Fig. 8.2; the steam pressure increases during the periods when the RCIC system is off and decreases when the RCIC

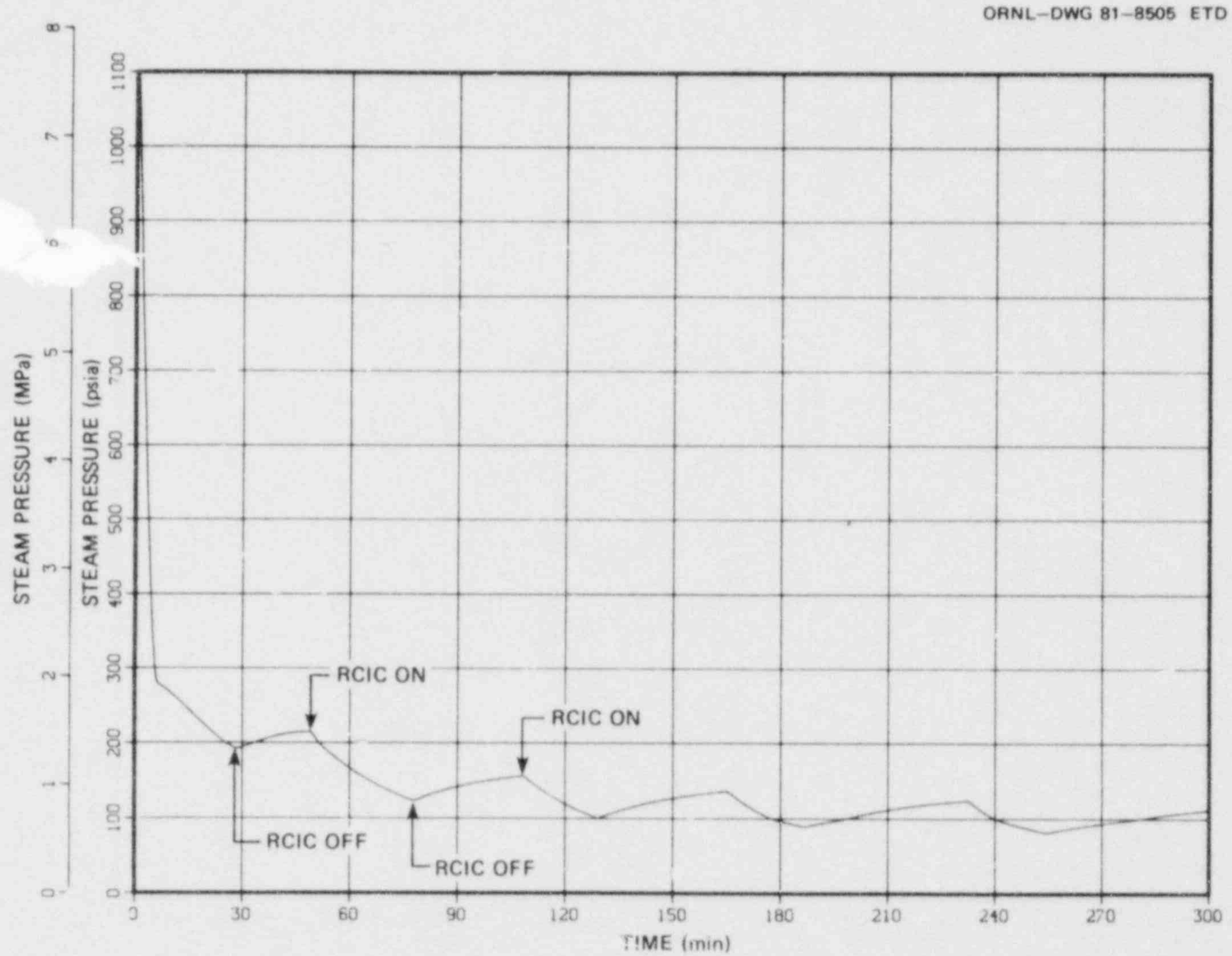

Fig. 8.2 Reactor vessel pressure response to a stuck-open relief valve during a station blackout.

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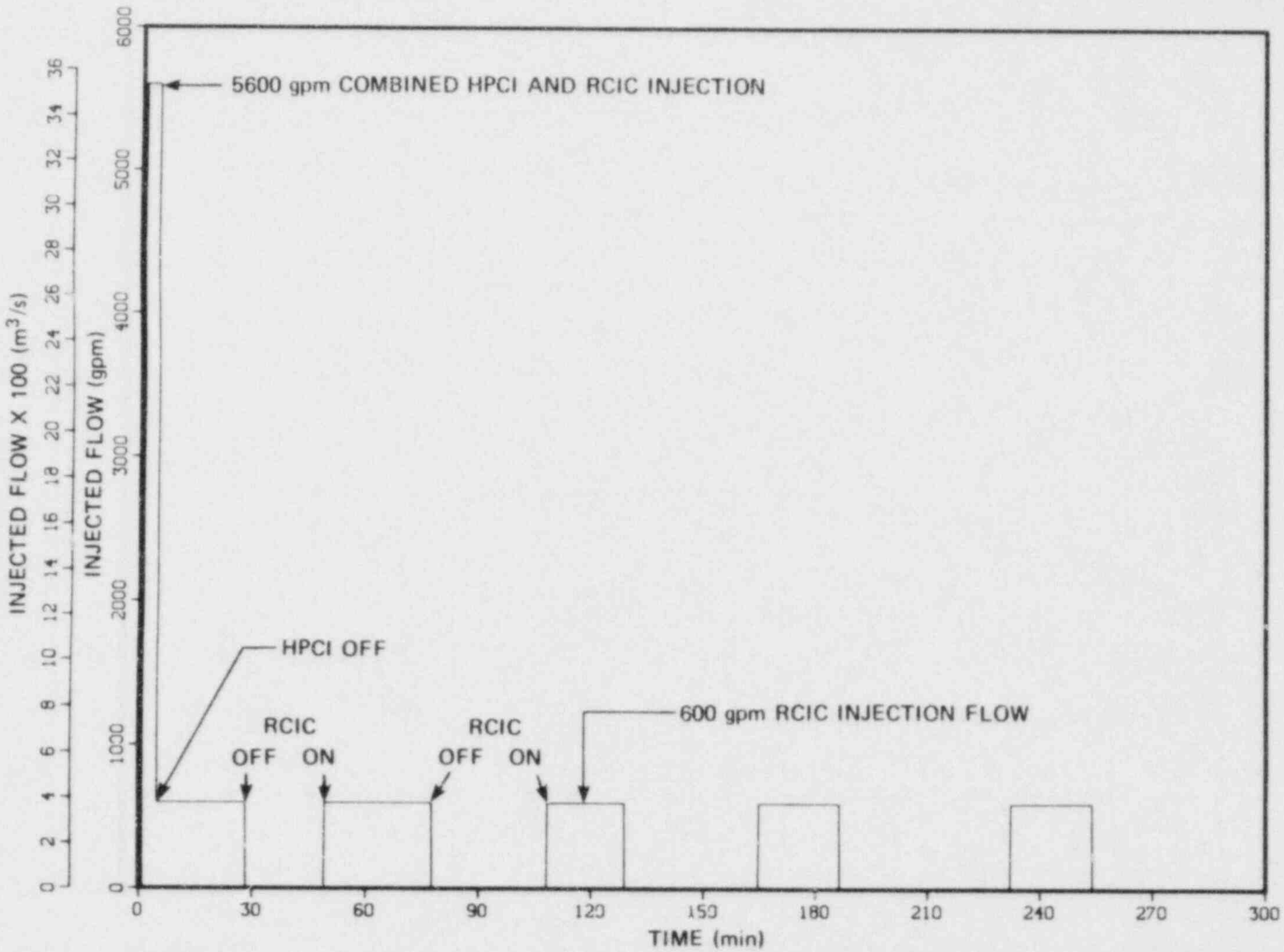

Fig. 8.3 Reactor vessel water injection with a stuck-open relief valve during a station blackout.

system is operating. The average steam pressure decreases with time as the power generation due to decay heat slowly decreases.

As shown in Fig. 8.2, the average reactor vessel steam pressure will have decreased to the point where continued HPCI system operation is questionable [0.793 MPa (115 psia)] after about two hours of the Station Blackout. However, the RCIC system should remain operational for over five hours. The minimum pressure reached during this period would be about 0.586 MPa (85 psia) whereas the RCIC system is operational at steam pressures as low as 0.448 MPa (65 psia).

It can be concluded that reactor vessel level control can be maintained for at least five hours (or until DC power is lost) during a Station Blackout with one stuck-open relief valve.

8.3 Loss of 250-Volt DC power

There are eight 250-volt DC battery systems at the Browns Ferry Nuclear Plant. Each system consists of a 120 cell lead-acid battery, a battery charger, and the associated distribution equipment. Four of these battery systems provide control power to the four 4160-volt AC shutdown boards and would not be used during a Station Blackout. The fifth 250-volt DC system provides power for common plant and transmission line control functions and would supply drive power for a 120-volt AC plant preferred motor-generator set during a Station Blackout. The 120-volt AC plant preferred system loads include common plant equipment such as the communications room, the CO_2 fire protection system, the sequential events recorder, the computer clock, and the stack gas monitors.

Each of the Browns Ferry units is provided one of the three remaining 250-volt DC battery systems as a source of power for unit control functions and for certain unit DC motor loads. During a Station Blackout, each unit battery would also provide drive power for a 120-volt AC unit preferred motor-generator set. The 120-volt AC unit preferred system loads are listed in Table 8.1. The major 250-volt DC loads which would be supplied by each of the unit batteries during a Station Blackout are listed in Table 8.2. It is possible to feed the 250-volt distribution system of one unit from another units' battery, but there is no provision for supplying DC power into any of the unit distribution systems from the shutdown board control power or plant battery systems discussed in the preceding paragraph.

A total loss of injection capability at each unit will certainly occur during a prolonged Station Blackout at the time when the unit battery, which supplies 250 volt DC logic and valve-control power to the RCIC and HPCI systems becomes exhausted.

The design basis for the Browns Ferry 250V-DC Power Supply and Distribution System provides that:[17]

> "Battery capacity shall be adequate so that any two unit batteries can supply for 30 minutes, without chargers available, the DC power required to operate the engineered safeguards systems on any one reactor unit in the event of a design basis accident as well as the DC power required for the safe shutdown and cooldown of the other two units with a final terminal voltage of 210 volts."

Table 8.1 Unit preferred system loads

1. Containment Isolation panel
2. Feedwater Control panels
3. Reactor Manual Control panel
4. Drywell Ventilation and Reactor Building Closed Cooling Water System Control panel
5. Feedpump Turbine Control panels
6. Rod Position Information System
7. EHC Control Unit
8. Unit Computer and Rodworth Minimizer panels
9. Reactor Water Cleanup panel

Table 8.2 Major unit 250-volt DC loads

1. Turbine Building Distribution Board
2. 480-volt Shutdown Board Control Power
3. Emergency DC Lighting
4. Unit Preferred AC Motor-Generator
5. Circuit Breaker Board 9-9 (Feedpump Turbine Controls)
6. Reactor Motor-Operated Valve Boards
   a) Primary Relief Valve DC Solenoids
   b) Main Steam Isolation Valve DC Solenoids
   c) Recirculation Motor-Generator Set Emergency Oil Pumps
   d) Backup Scram Valves
   e) RHR Shutdown Isolation Valves
   f) Engineered Safeguards Logic Power Supplies
   g) HPCI System Controls, Valves, and Auxiliaries
   h) RCIC System Controls, Valves, and Auxiliaries

"The engineered safeguards systems that are supplied from the 250 volt DC system shall be designed to operate at a minimum of 200 volts."

It nas previously been estimated[5] that under the less severe conditions of a Station Blackout, with all unit batteries avaiable, and assuming prudent actions by a well-trained operator to conserve battery potential by minimizing DC loads, the necessary 250 volt DC power for HPCI or RCIC system operation would remain available during the first four to six hours of a Station Elackout.

In summary, it is reasonable to expect that control can be maintained over reactor vessel pressure and water level at each of the Browns Ferry Units during a Station Blackout for as long as the 250 volt DC power remains available. This period will depend on the operator's ability to conserve the unit battery potential by minimizing the DC loads, but is expected to last from four to six hours.* During this period the operator may have to take other actions to maintain the operability of the HPCI and RCIC systems as discussed in subsection 8.1.

*A battery life of seven hours is considered in Appendix G.

9.0 ACCIDENT SEQUENCES RESULTING IN CORE MELTDOWN

9.1 Introduction

This section deals with the various accident sequences which might occur during a complete Station Blackout (CSB) at the Browns Ferry Nuclear Plant which can lead to core meltdown.[18] Event trees are presented for what are considered to be the six most probable sequences. The characteristics and event timing for each of these sequences were determined by use of the MARCH code, and include consideration of the operator's role in reactor vessel pressure and level control. The progression of core damage and containment failure following core uncovery will be discussed in this section; an event tree for operator key actions in mitigating the accident progression will be discussed in Sect. 10.

Of the sequences modelled by MARCH and presented in this section, the sequence TB′* is deemed to be most representative of the events which would occur after core uncovery following a four-hour period during which the availability of DC power permitted reactor vessel level and pressure control. For this analysis, it is assumed that no independent secondary equipment failures occur, and this sequence was chosen to serve as the basis for the fission product transport analysis discussed in Volume 2 of this report.

9.2 Accident Phenomenology

9.2.1 Accident Progression Resulting in Core Melt

Upon a loss of offsite and onsite AC power, a number of reactor safety systems respond immediately and automatically. First of all, a full load rejection (i.e., fast closure of the turbine control valves) occurs, followed by the tripping of the recirculation pumps and the main condenser cooling water pumps. With the load rejection, the scram pilot valve solenoids are deenergized and the control rods start to move toward the fully inserted position. The main steam isolation valves (MSIVs) begin to close, resulting in a rapid reactor vessel pressure increase. These events are closely followed by a main turbine trip (i.e., closure of the turbine stop valves) and the tripping of the feedwater turbines.

After the automatic actions described above, core flow is provided by natural circulation and excess reactor vessel pressure is relieved by steam blowdown through the safety/relief valves (SRVs) into the pressure suppression pool. If any SRV fails to reclose after actuation, the reactor vessel will continue to depressurize with an accompanying loss of

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*The nomenclature TB′ follows that used in the Reactor Safety Study,[19] where T denotes a transient event and B′ denotes failure to recover either offsite or onsite AC power within about 1 to 4 hours. In effect, TB′ is a loss of decay heat removal (TW) sequence with the loss of both offsite and onsite AC power as an initiating event.

water inventory similar to that occurring in a small break LOCA. According to the Reactor Safety Study,[19] the probability of a stuck-open relief valve (SORV) event is estimated to be 0.10 with an error spread of 3. However, this estimate was based upon consideration of three-stage SRVs such as those originally installed at the Browns Ferry site. with the two-stage relief valves of improved design which have a recently been installed at the Browns Ferry Plant, the probability of a SORV event is substantially reduced and is estimated to be less than 10^{-2}. The occurrence of a sequence involving a SORV, designated sequence TPB′, is included in the event trees of this section.

The reactor vessel water level decreases rapidly during the initial moments of a Station Blackout due both to the void collapse caused by the increased pressure and to the water inventory lost through the SRVs. When the water level sensed by the wide range detector reaches the low water level setpoint (Level 2), the DC-powered HPCI and RCIC systems are automatically initiated, with their turbine-driven pumps supplied by steam generated by the decay heat. The HPCI and RCIC systems are assumed to remain operational until the unit battery is exhausted at four hours into the Station Blackout. Following the loss of these injection systems, the reactor vessel level decreases until the core is uncovered about one hour later. Subsequently, the core begins to melt.

The failure probability for HPCI has been estimated[19] to be 9.8×10^{-2} with an error spread of 3 and for RCIC has been estimated to be 8×10^{-2} with an error spread of 3. The overall failure probability of both HPCI and RCIC to provide makeup water during the sequence TB′, is designated TUB′ and has been estimated[19] to be 2×10^{-3} with an error spread of about 4. The failure of both HPCI and RCIC together with a SORV occurrence is designated TUPB′.* Sequences TUB′ and TUPB′ have been included in the event trees of this section.

9.2.2 Containment Failure Modes

Any sequence resulting in core melt will eventually lead to containment failure if electrical power is not restored before the reactor vessel fails. According to the Reactor Safety Study,[19] BWR containment failure could occur in the following categories:

- α — Containment failure due to steam explosion in vessel,
- β — Containment failure due to steam explosion in containment,
- γ — Containment failure due to overpressure - release through reactor building,
- γ' — Containment failure due to overpressure-release direct to atmosphere,
- δ — Containment isolation failure in drywell,
- ϵ — Containment isolation failure in wetwell,
- ζ — Containment leakage greater than 2400 volume percent per day.

Containment failure by overpressurization was considered to be the dominant failure mode for a BWR inerted containment in the Reactor Safety

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*TUB′ and TUPB′ are in effect TQUV and TPQUV sequences with the loss of both offsite and onsite AC power as an initiating event.

Study.[19] It was postulated that failure would occur when the steel liner and the inner layer of reinforcement were stressed to a level between the yield and the ultimate tensile strength. For this reason, the study provided an upper and lower bound for the containment failure pressure [i.e., failure was assumed to occur between 1.34 and 1.21 MPa (195 and 175 psia)].

Other failure modes resulting in high initial leakage from the containment have historically been considered to be of less significance from the standpoint of radiological release because they would not lead to gross containment failure by overpressurization and it was believed that the leaked radioactivity would be significantly reduced by the filtration and absorption capacity of the standby gas treatment system.

However, failure modes involving high initial containment leakage have been found to be important for the Station Blackout sequences considered in this study, in which the standby gas treatment system would be inoperable as a consequence of the loss of AC power. Gross containment failure due to overpressure is discounted for the following reasons:

1. Containment failure due to a steam explosion following core melt has been found to be highly unlikely at the Zion plant,[20] and recent experiments at SANDIA[21] have shown that corium does not undergo violent explosions upon interaction with water. Therefore, containment failure due to steam explosions is not considered in this study.
2. The containment at Browns Ferry has an inert atmosphere. This means it is highly unlikely that the containment would fail as a result of overpressurization caused by hydrogen burning, and this failure mode is not considered in this study.

Containment failure by deterioration of the drywell electric penetration assembly (EPA) seals due to high ambient temperature resulting from core meltdown would occur before the containment pressure increase caused by suppression pool heatup reached failure proportions. Accordingly, EPA seal deterioration has been identified as the dominant failure mode in the BWR.[18,22] However, for sequences in which excessive amounts of superheated steam and noncondensibles are discharged into the suppression pool within a short period of time, the wetwell might fail before the drywell due to forces of steam jet impingement and condensation oscillations.[18] These forces are discussed further in Appendix D.

In the case of failure of the drywell EPA, the initial high leakage would not prevent an ultimate gross containment failure. The drywell temperature would continue to increase, causing the EPA seals to further decompose and to finally be blown out of the containment wall.

If, on the other hand, failure occured in the wetwell, radiological consequences from the releases would be less than those from a drywell failure because a large fraction of the fission products would have been dissolved or deposited in the suppression pool and therefore not released from the containment.

As will be shown, the postulated mode of containment failure by decomposition of the electrical penetration seals due to overheating would occur at a pressure approximately 30% less than that considered necessary for containment failure in the Reactor Safety Study.[19]

9.3 Event Trees

9.3.1 Event Trees for Accident Sequences Resulting in Core Melt

It has been concluded in this study that the sequence of events during a prolonged complete Station Blackout (CSB) at the Browns Ferry Nuclear Plant (BFNP) would most probably be described by one of the following titles:

1. CSB + HPCI/RCIC (TB′)
2. CSB + HPCI/RCIC + SORV (TPB′)
3. CSB + Manual RCIC & SRV (T_vB')
4. CSB + Manual RCIC & SRV + SORV (T_vPB')
5. CSB + No HPCI/RCIC (TUB′)
6. CSB + No HPCI/RCIC + SORV (TUPB′)

As indicated in this listing, a stuck-open relief valve (SORV) corresponds to a small break LOCA. The terminology "Manual RCIC & SRV" refers to the assumption that the operator controls level and pressure by remote-manual operation of the RCIC system and the relief valves, respectively. The phrase "No HPCI/RCIC" means that both of these water-injection systems are assumed to be unavailable due to mechanical failure.

The event tree identifying these six accident sequences is shown in Fig. 9.1. Sequence No. 1, traced by the dotted line in this figure, is the sequence TB′ which involves no independent secondary equipment failures and was selected as the basis for the fission product transport analysis discussed in Volume 2. Other possible but much less likely sequences such as failure of reactor scram are shown on Fig. 9.1 but were not considered in this study.

9.3.2 Core Damage Event Tree

Without restoration of offsite or onsite AC power and with the consequent inevitable loss of DC power, any of the numbered sequences shown on Fig. 9.1 will lead to core uncovery and subsequent core melting, reactor vessel failure, and corium-concrete interaction. A tree for the events subsequent to core uncovery is shown in Fig. 9.2, with the path used for the fission product transport analysis of Vol. 2 identified by a dotted line.

9.3.3 Containment Event Tree

A containment event tree for the occurrences following core melt is given in Fig. 9.3. The particular path which has been chosen for the fission product transport studies is again indicated by the dotted line path.

It should be noted that if AC power is restored after core melt, but before a failure of the reactor vessel occurs, the low-pressure Emergency Core Cooling Systems (ECCS) would function and a breach of the containment in the drywell or wetwell areas may be averted.

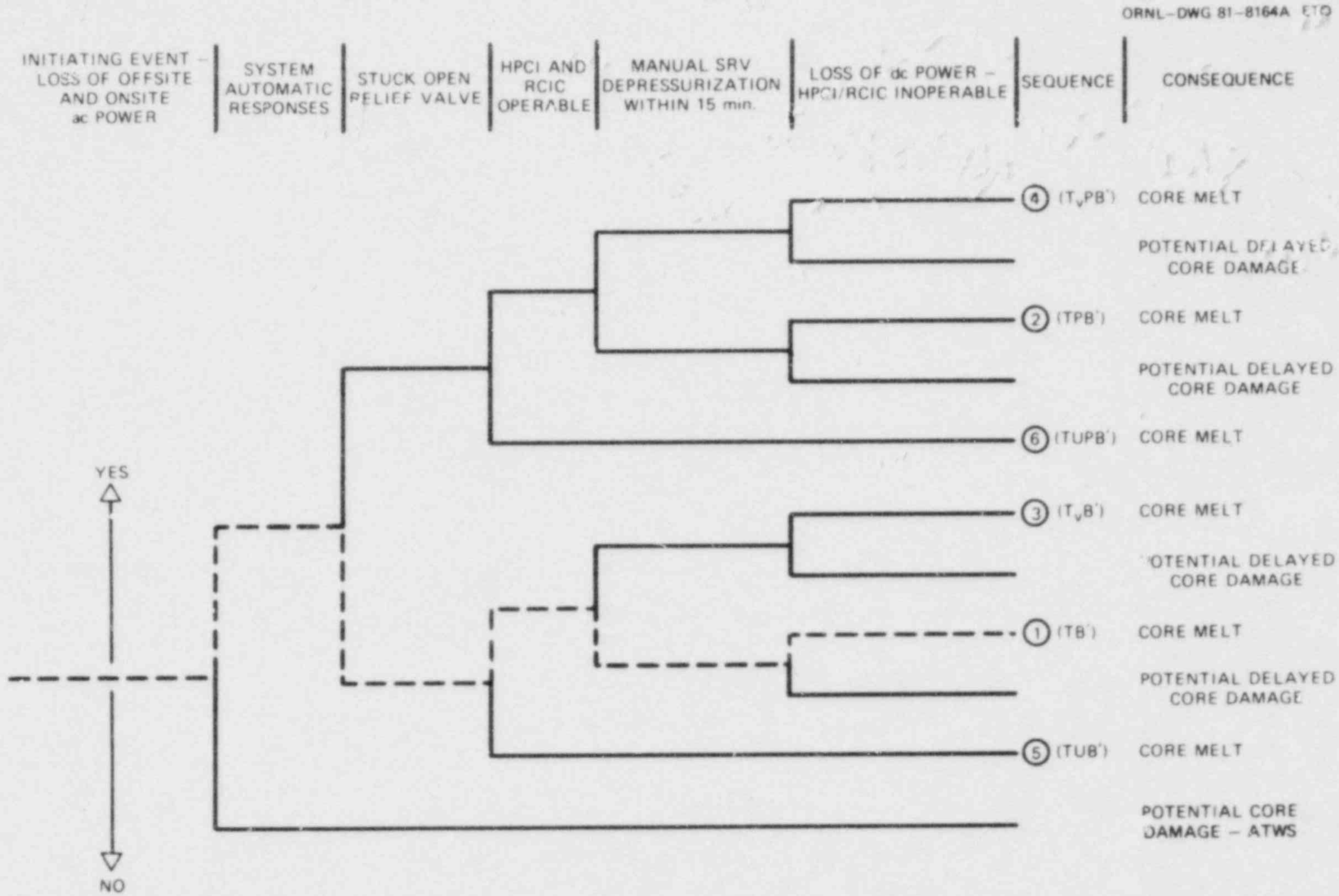

Fig. 9.1 Event tree for accident sequences resulting in core melt.

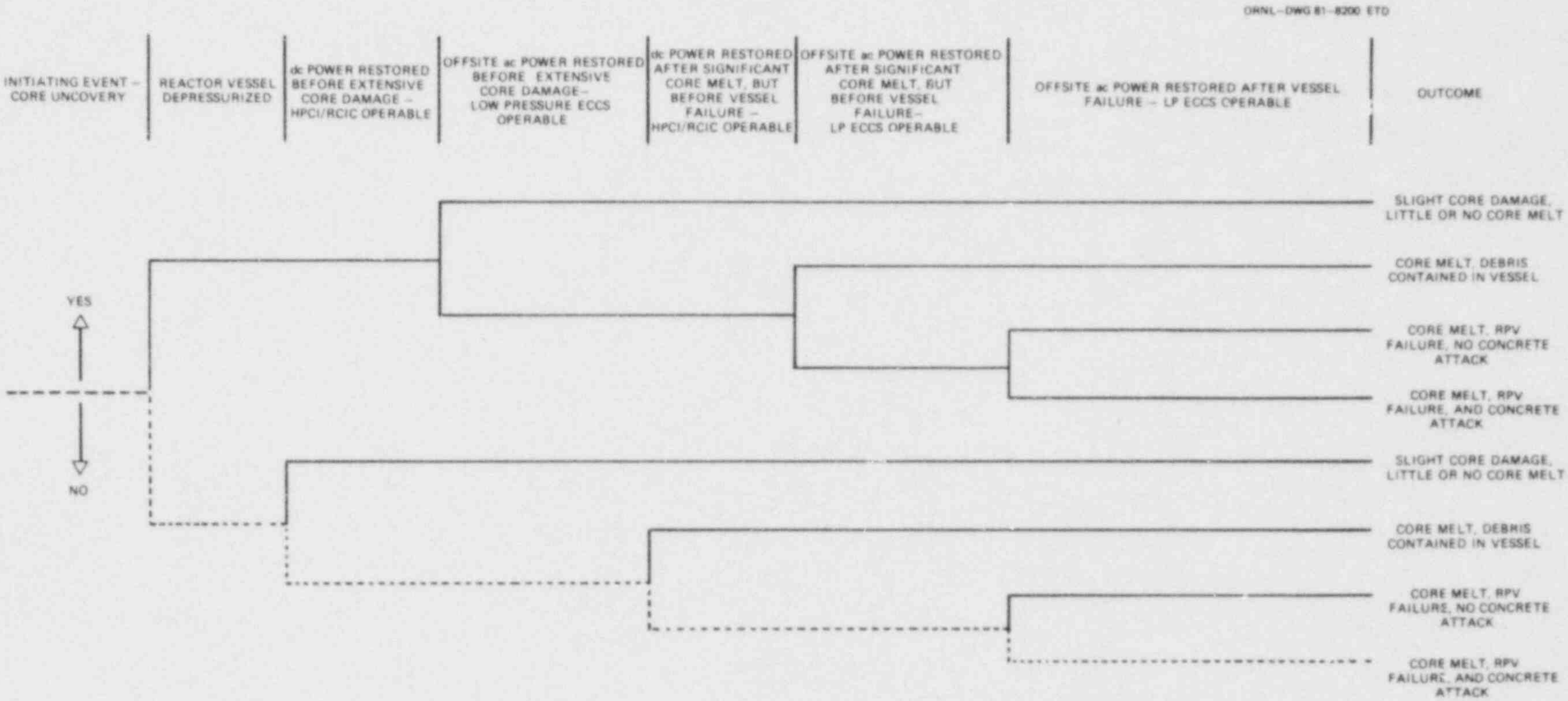

Fig. 9.2 Core damage event tree.

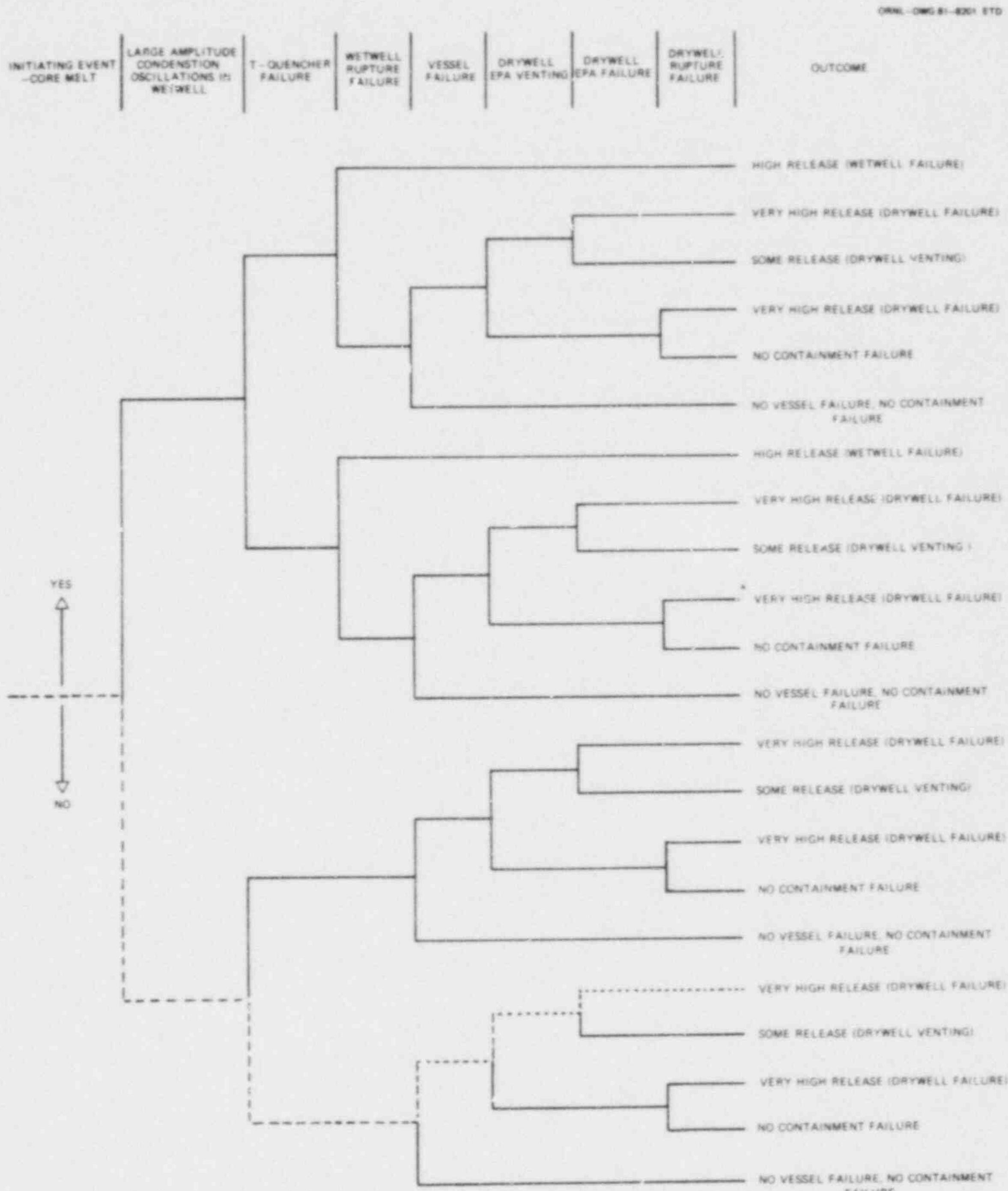

Fig. 9.3 Containment event tree.

9.4 Accident Sequences

9.4.1 MARCH computer code

Detailed calculations for the accident sequences have been performed with the MARCH computer code[13], version 1.4B. This version is based on version 1.4 at Brookhaven National Laboratory, but contains a different subroutine ANSQ for the decay heat power calculations and modifications to the subroutine HEAD which were necessary to improve the prediction of vessel failure time.[18]

In the MARCH code, the fission product decay heat source term is based on ANS Standard ANS-5.1 (1973)[23] which does not account for the decay of U-239 or Np-239. The new standard ANS-5.1 (1979)[24] provides decay parameters for U-239 and Np-239, but does not account for the decay power generated from these or the numerous other heavy nuclides (actinides) which exist in power reactors. The new subroutine ANSQ used in this study (Cf. Appendix B) is based on the actinide decay heat source in a BWR following a depletion of 34,000 MWd/t. The actinide heating calculations[25] were performed using the EPRI—CINDER code and include all significant actinides from Tl-208 through Cm-246. Furthermore, the initial fission energy release from the fuel is assumed to last 3 seconds rather than 5 seconds as used originally in the MARCH code. This has been found to yield better agreement with test data in implementing the REDY code at GE.[26]

A comparison of MARCH calculations for the sequence TB´ as performed by versions 1.4 and 1.4B are given in Figs. 9.4 through 9.11 for the time- dependence of (1) the maximum core temperature, (2) the water-steam mixture level, (3) the containment wall temperature, and (4) the containment volumetric leak rate. It should be noted that the times predicted by version 1.4B for core uncovery and core melt are less by approximately 18 and 26%, respectively, and that the maximum containment wall temperature is increased by more than 120%. Other comparisons have shown that the 1.4B version produces results in better agreement with the core uncovery time calculated by the RELAP4/Mod 7 Code for the sequence TUB´ at INEL.[27]

9.4.2 Accident progression signatures

This section contains the accident progression signatures for the six most probable sequences following complete station blackout (CSB). The early part of the plant transient characteristics follow that contained in the Browns Ferry Nuclear Plant Final Safety Analysis Report. All other accident signatures have been based on the results calculated by the MARCH computer code.[13]

9.4.2.1 CSB + HPCI (TB´). The accident progression signature for the sequence TB´ is given in Table 9.1. This sequence has been chosen for the fission product transport studies presented in Volume 2.

9.4.2.2 CSB + HPCI + SORV (TPB´). The accident progression signature for the sequence TPB´ is given in Table 9.2. The SORV event has the same effect as a small break LOCA, ensuring that the reactor vessel

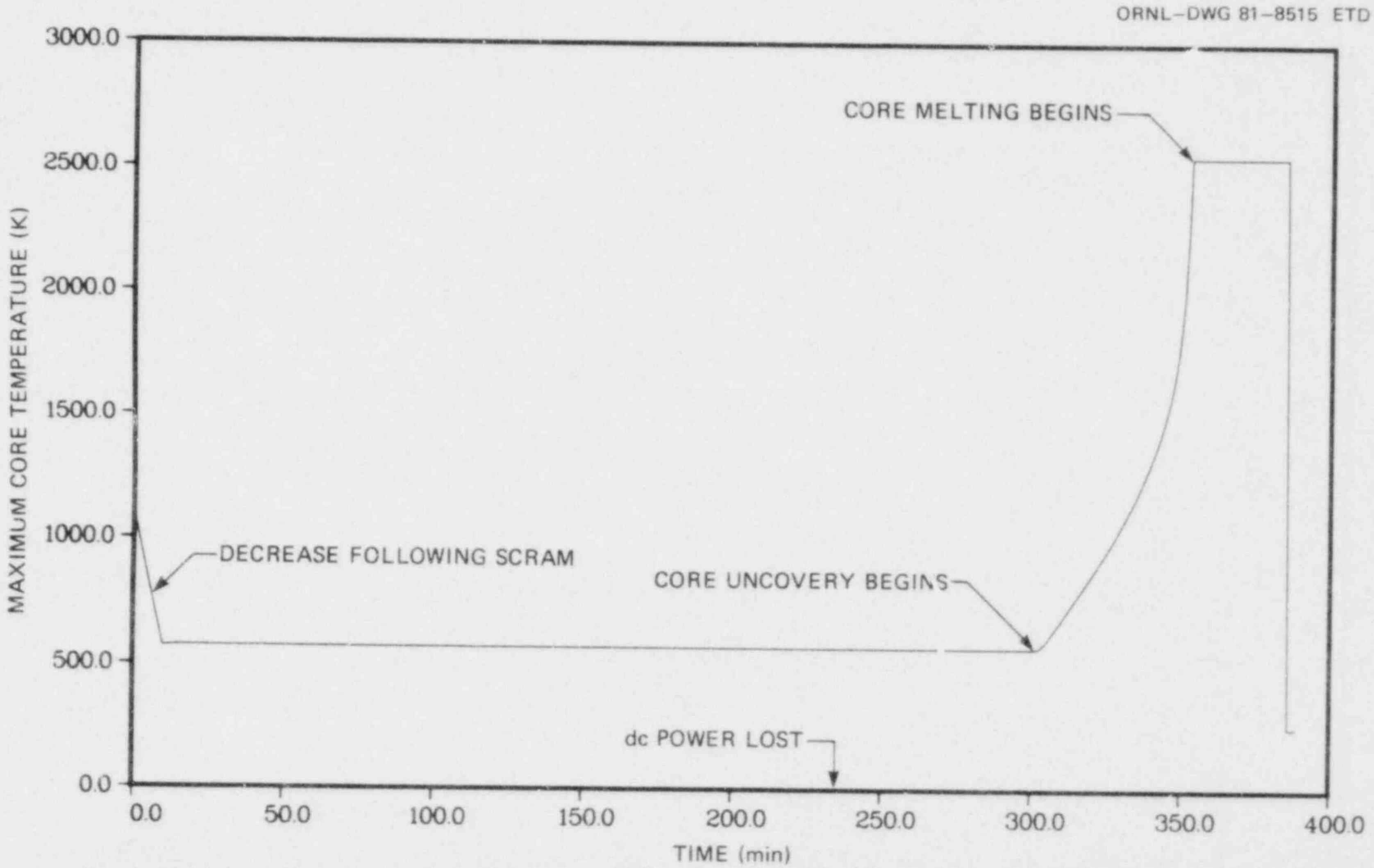

Fig. 9.4 Maximum core temperature (MARCH/MOD 1.4B).

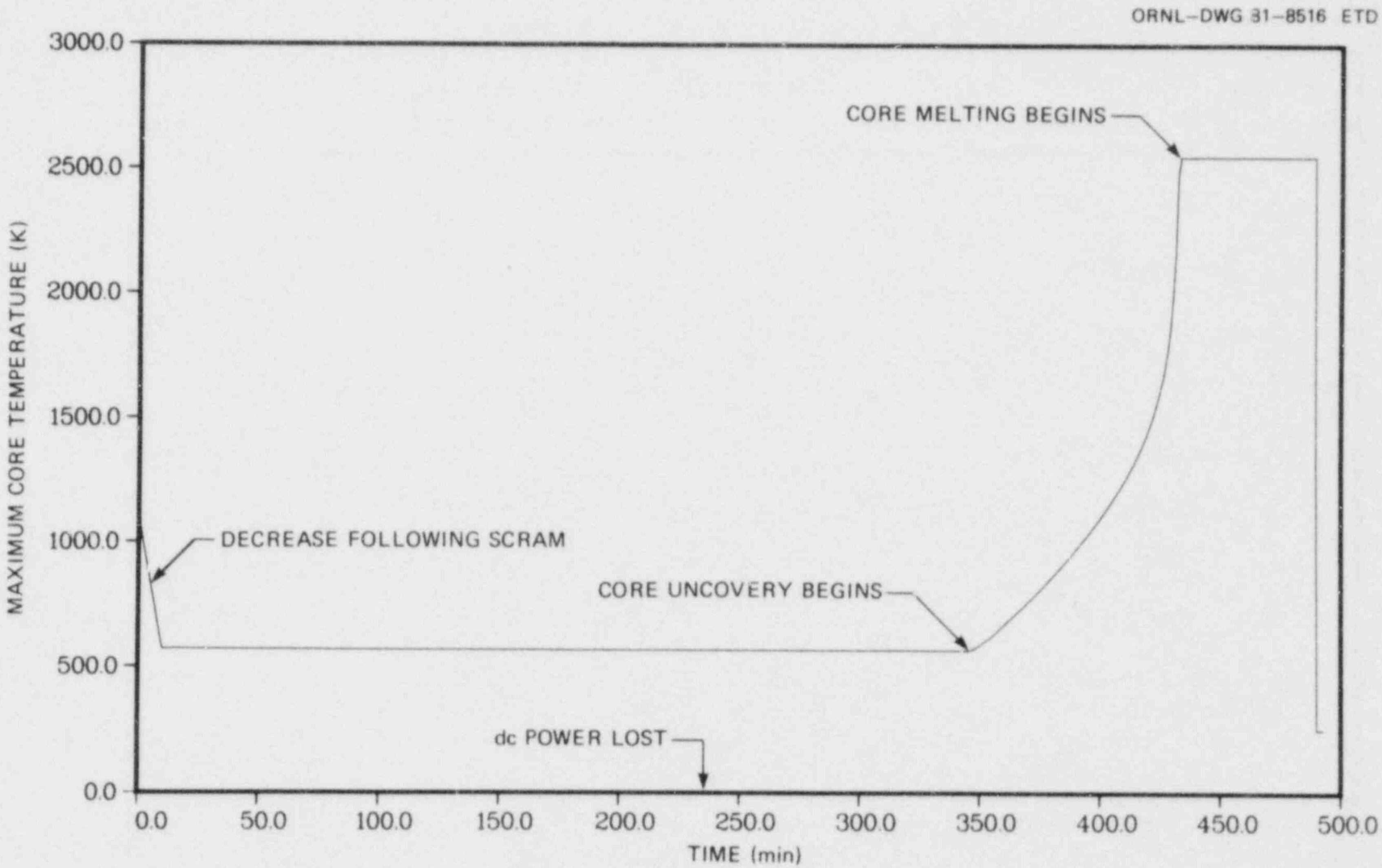

Fig. 9.5 Maximum core temperature (MARCH/MOD 1.4).

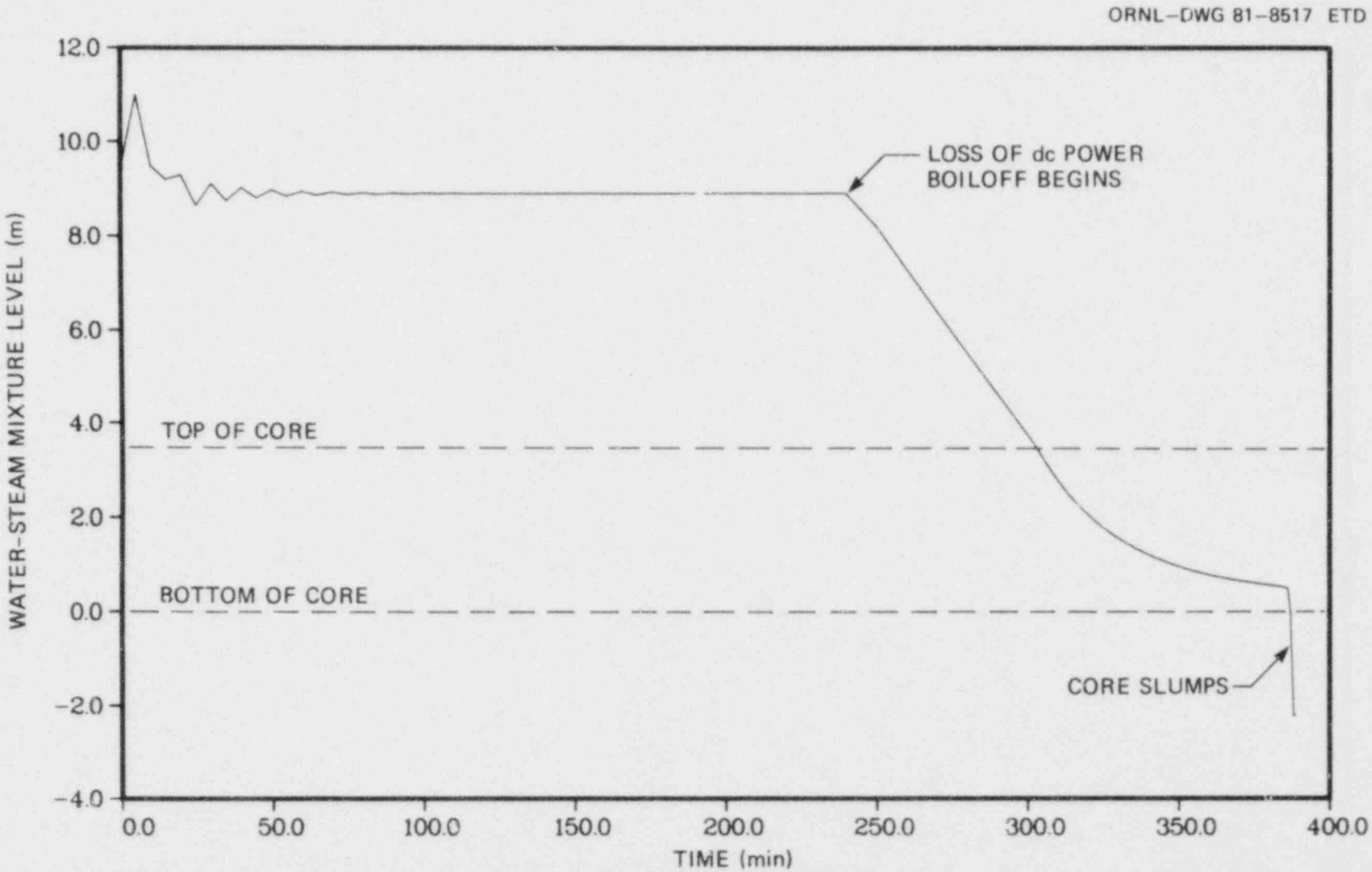

Fig. 9.6 Water-steam mixture level (MARCH/MOD 1.4B).

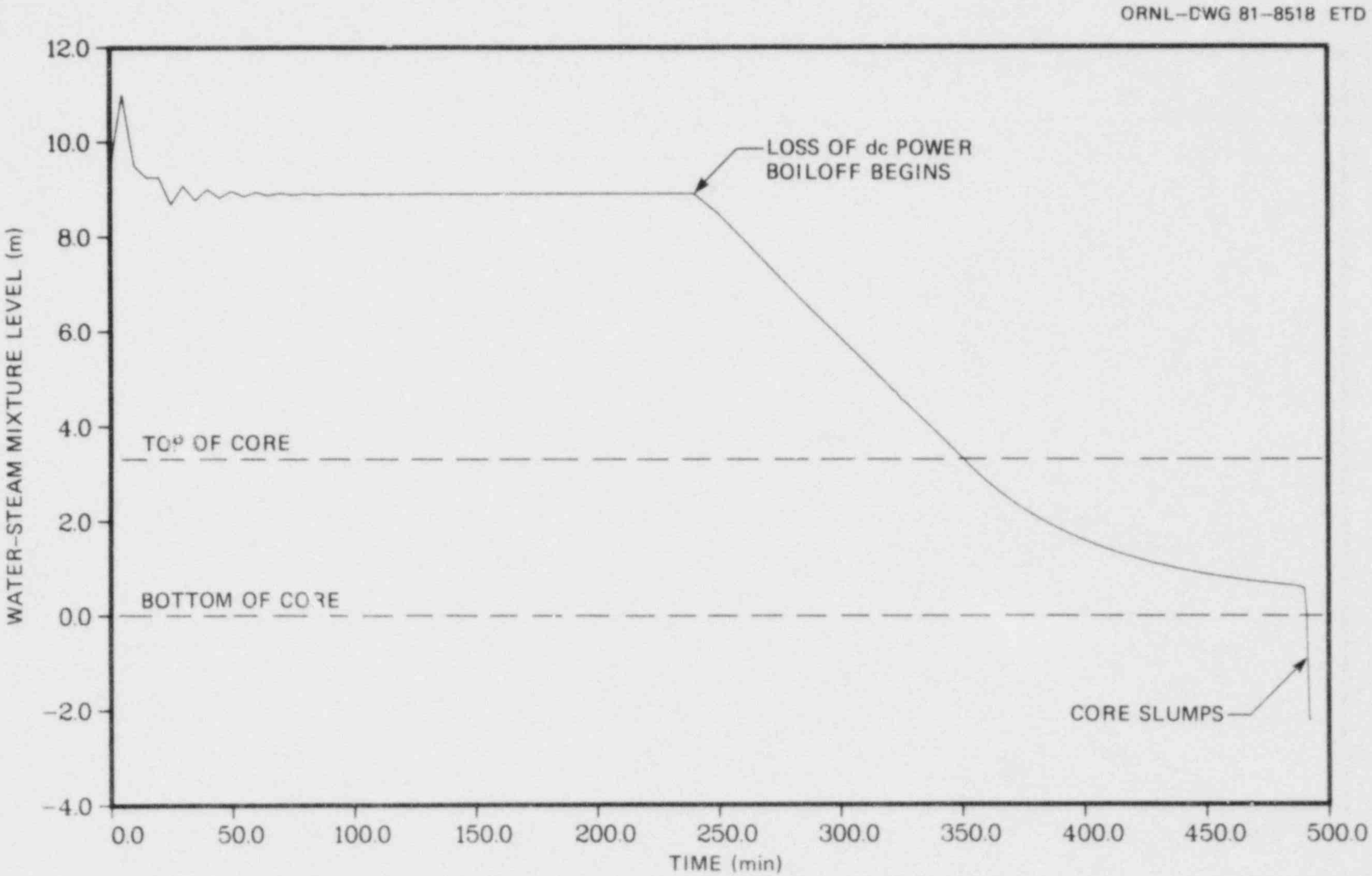

Fig. 9.7 Water-steam mixture level (MARCH/MOD 1.4).

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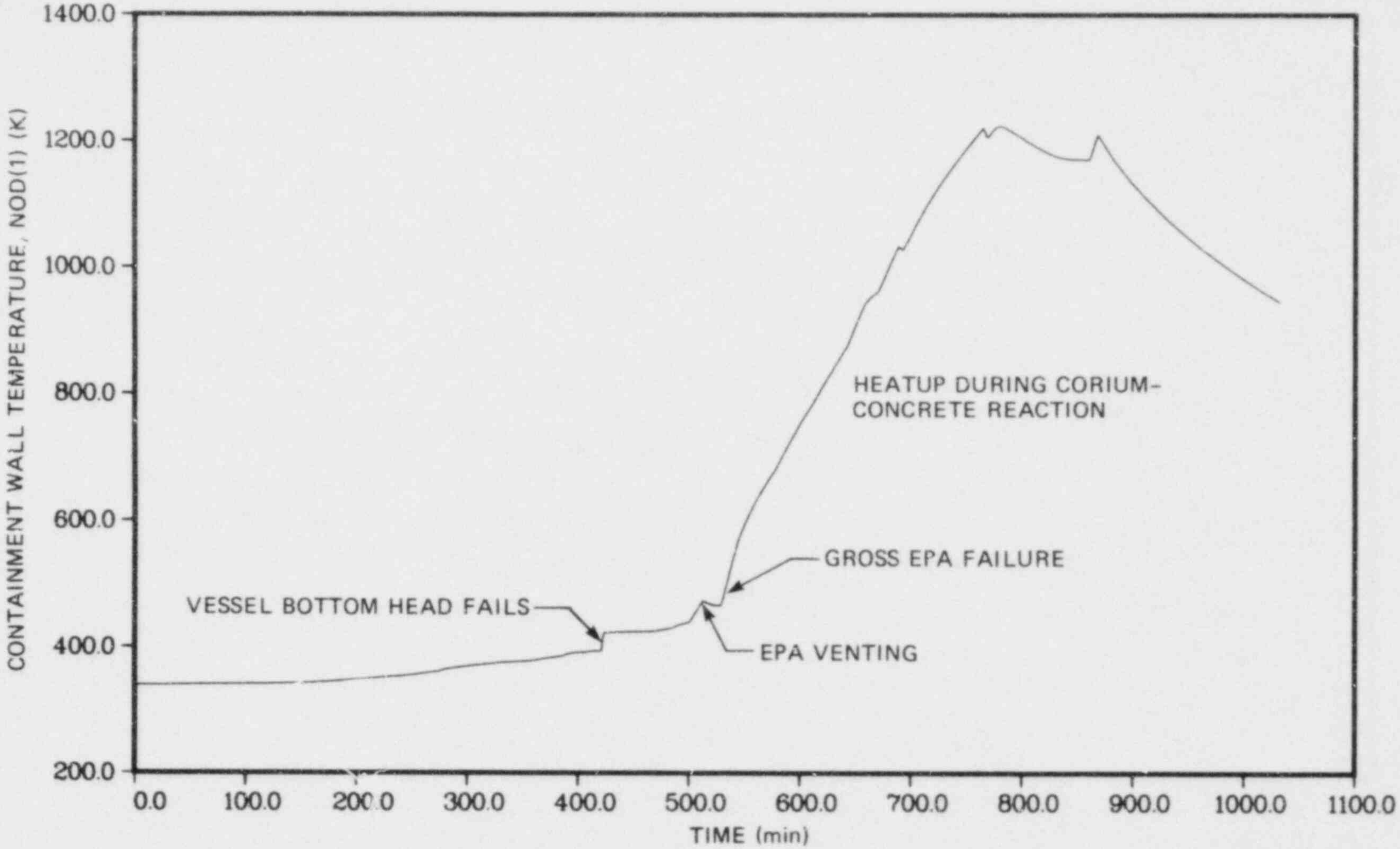

Fig. 9.8 Containment wall temperature (MARCH/MOD 1.4B).

ORNL-DWG 81-8520 ETD

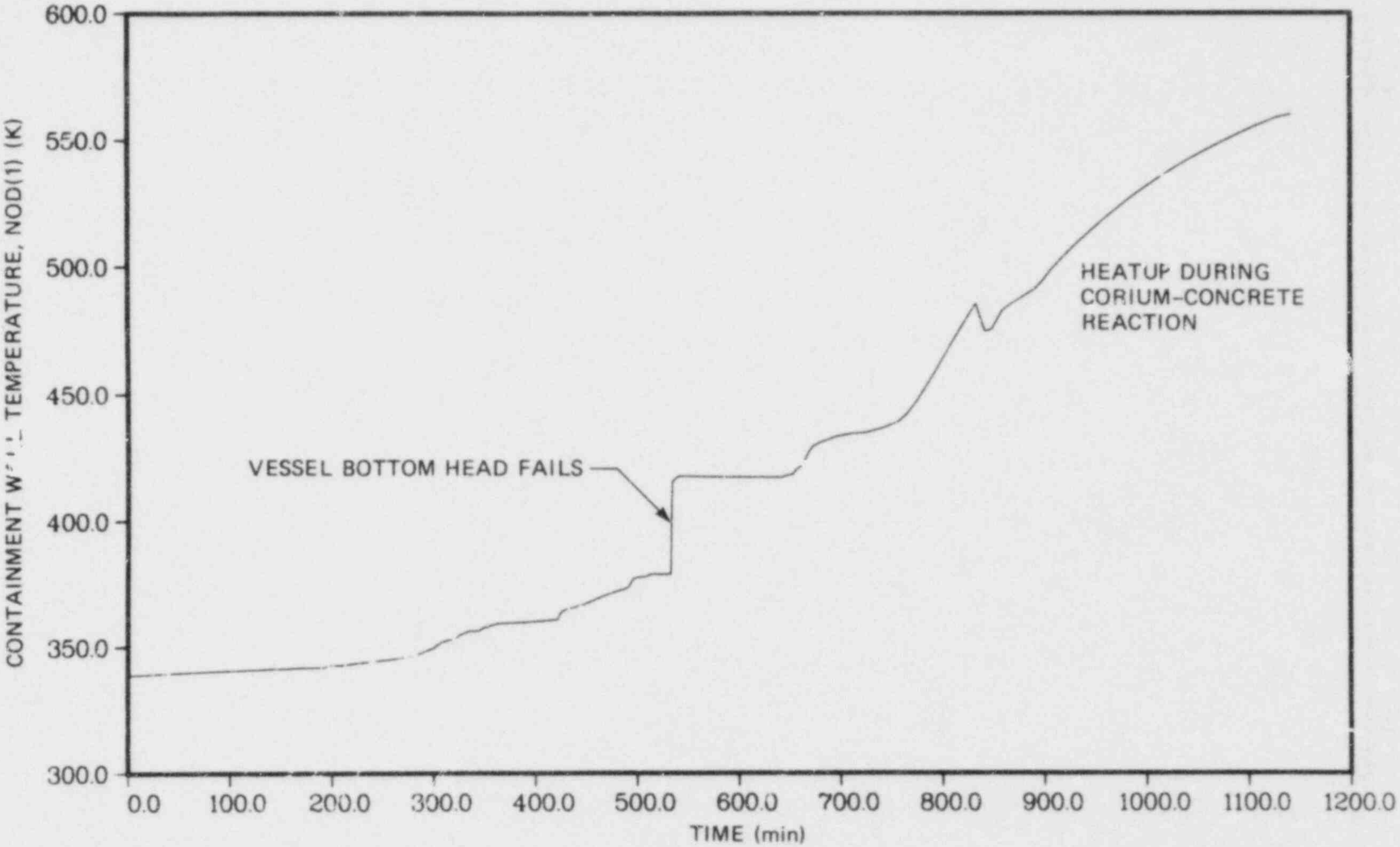

Fig. 9.9 Containment wall temperature (MARCH/MOD 1.4).

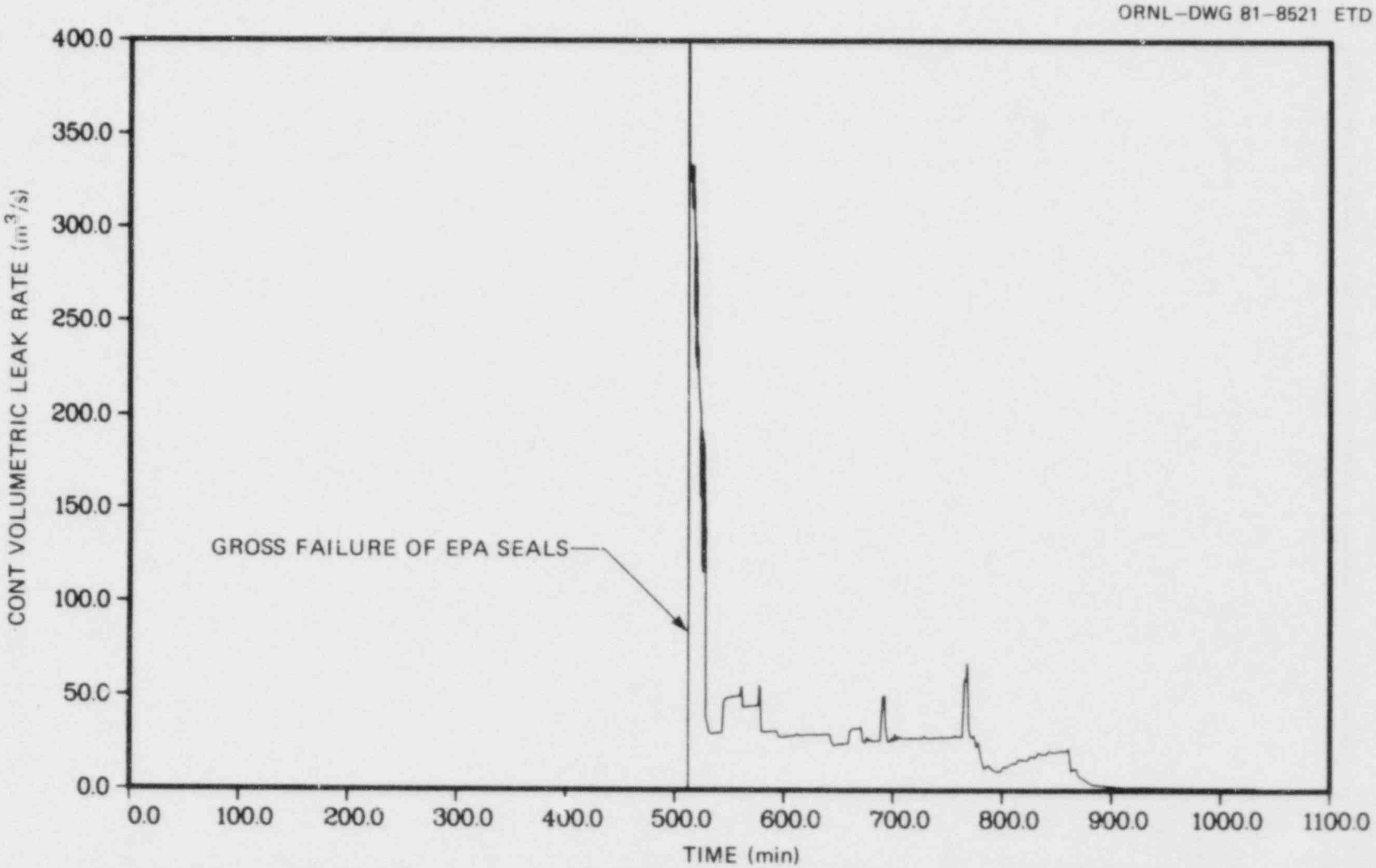

Fig. 9.10 Containment leak rate (MARCH/MOD 1.4B).

ORNL-DWG 81-8522 ETD

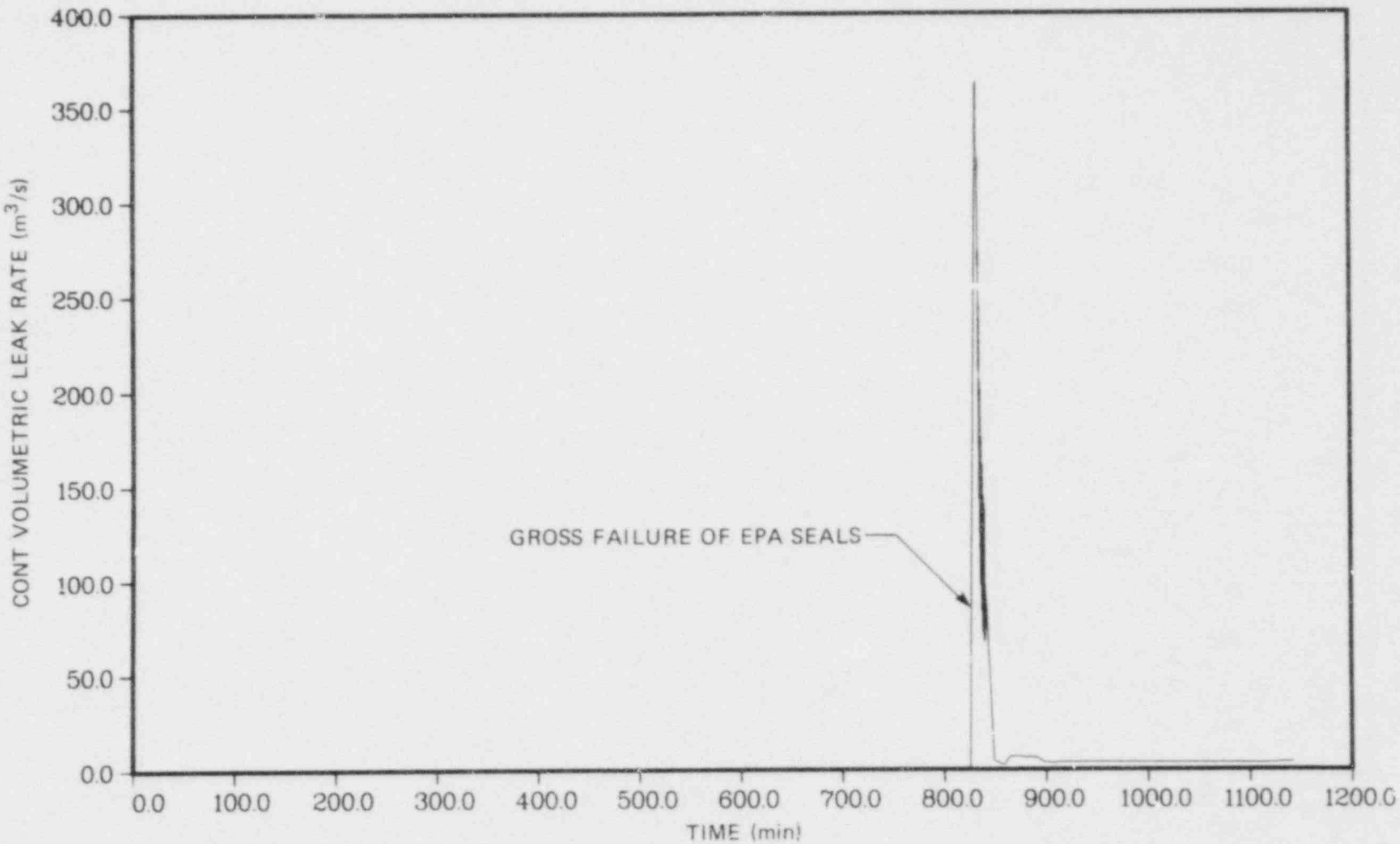

Fig. 9.11 Containment leak rate (MARCH/MOD 1.4).

Table 9.1

Browns Ferry Nuclear Plant: Complete Station Blackout
Sequence of Events

CSB + HPCI/RCIC

(TB´)

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.

Time (sec)	Event
3.0	Turbine trips off (turbine stop valves fully closed).
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Seven (7) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 0.51 m (20 in.) from the previous momentary 1.02 m (40-in.) drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 7 S/RVs are completely closed.
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 5.98 m (235.50 in.) above Level 0, or 5.00 m (196.44 in.) above TAF.
22.0	Suppression pool water average temperature rises to 35.13°C (95.24°F) in response to the first S/RV pops.
29.0	All 4 S/RVs are completely closed.

Time (sec)	Event
29.7	Two out of 13 S/RVs start to open.
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 35.3°C (95.54°F).
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 5.50 m (216.00 in.) above Level 0, or 4.50 m (176.94 in.) above TAF.
90.0	Suppression pool water average temperature is approximately 35.4°C (95.72°F).
101.0	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the subsequent cyclings of HPCI and RCIC injections.
625	Wide range sensed water level reaches low water level setpoint (Level 2), i.e., 4.18 m (164.50 in.) above Level 0 at 2/3 core height, or 2.96 m (116.50 in.) above TAF.
625	HPCI and RCIC systems are automatically turned on. The HPCI and RCIC turbine pumps are driven by steam generated by decay heat. System auxiliaries are powered by the 250 V dc system.
655	HPCI and RCIC flows enter the reactor pressure vessel at 315 ℓ/s (5000 gpm) and 38 ℓ/s (600 gpm), respectively, drawing water from the condensate storage tank.
12.5 min.	Narrow range sensed water level reaches Level 8 setpoint, i.e., 6.86 m (270.00 in.) above Level 0, or 5.64 m (222.00 in.) above TAF.

Time (sec)	Event
12.5 min.	HPCI and RCIC trip off.
20 min.	Drywell and wetwell temperatures exceed 70°C (158°F) and 42°C (108°F), respectively.
26.5 min.	Wide range sensed water level reaches Level 2 setpoint and HPCI automatically restarts. (RCIC does not automatically restart.)
27.0 min.	HPCI flow enters the RPV.
29.0 min.	Narrow range sensed water level reaches Level 8 setpoint and HPCI trips off again. The HPCI system, driven by steam generated by decay heat, turns on and off automatically between sensed Levels 2 and 8 until the batteries run out.
80 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The HPCI/ RCIC systems are not isolated.
240 min.	The HPCI pump stops when the batteries run out.
260 min.	Wide range sensed water level reaches Level 2 setpoint. Drywell and wetwell temperatures are 85°C (185°F) and 87°C (188°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	27.82	3.68×10^3	7.68×10^7	4.37×10^6
Hydrogen	0	0	0	0

Time (sec)	Event
302 min.	Core uncovery time. Steam-water mixture level is at 3.58 m (11.73 ft) above bottom of the core.
320 min.	Average gas temperature at top of core is 485°C (904°F). Drywell and wetwell temperatures and pressures are 101°C (213°F) and 0.21 MPa (31 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	18.75	2.48×10^3	5.40×10^7	3.07×10^6
Hydrogen	6.65×10^{-7}	8.80×10^{-5}	3.34	1.90×10^{-1}

Time (sec)	Event
340 min.	Average gas temperature at top of core is 821°C (1509°F). Drywell and wetwell temperatures and pressures are 103°C (218°F) and 0.23 MPa (33 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	11.75	1.56×10^{3}	3.76×10^{7}	2.14×10^{6}
Hydrogen	1.52×10^{-3}	2.00×10^{-1}	1.10×10^{4}	6.25×10^{2}

Time (sec)	Event
355 min.	Core melting starts.
389 min.	Water level in vessel drops below bottom grid elevation.
390 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
392 min.	The corium slumps down to vessel bottom.
394 min.	Debris starts to melt through the bottom head.
426 min.	Vessel bottom head fails, resulting in a pressure increase of 0.34 MPa (49 psia).
426.04 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 1433°C (2611°F) initially. Internal heat generation in metals and oxides are 1.05×10^{7} and 1.95×10^{7} watts, respectively.
503.27 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 204°C (400°F) and start to vent through the primary containment at a leak rate of 118 ℓ/s (250 ft^3/min).
513.59 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	4.61	610	1.59×10^{5}	9052
Hydrogen	0.11	15	0	0
CO_2	1.01	133		
CO	2.35	311		

The leak rate through the drywell penetration seals is ~3.04×10^{4} ℓ/s (6.44×10^{4} ft^3/min).

Time (sec)	Event
613 min.	Drywell and wetwell pressures are at 0.10 MPa (~14.7 psia) and temperatures are 661°C (~1222°F) and 98°C (~209°F), respectively. The leak rate through the containment failed area is ~2.96×10^4 ℓ/s (~6.27×10^4 ft^3/min).
695 min.	Drywell and wetwell temperatures are 623°C (1154°F) and 97°C (207°F), respectively. The leak rate through the containment failed area is ~6.47×10^4 ℓ/s (~1.37×10^5 ft^3/min).
1028 min.	Drywell and wetwell temperatures are 614°C (~1138°F) and 97°C (~207°F), respectively. The leak rate through the containment failed area is ~1.34×10^3 ℓ/s (~2.83×10^3 ft^3/min).

Table 9.2

Browns Ferry Nuclear Plant: Complete Station Blackout
Sequence of Events

CSB + HPCI/RCIC + SORV (Small Break LOCA)

(TPB´)

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.

Time (sec)	Event
3.0	Turbine trips off (turbine stop valves fully closed).
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Seven (7) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 0.51 m (20 in.) from the previous momentary 1.02 m (40-in.) drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 6 S/RVs are completely closed. One S/RV is stuck open (SORV); this has the same effect as a small break LOCA of equivalent break area of 0.015 m^2 (0.1583 ft^2).
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 5.98 m (235.50 in.) above Level 0, or 5.00 m (196.44 in.) above TAF.

Time (sec)	Event
22.0	Suppression pool water average temperature rises to 35.13°C (95.24°F) in response to the first S/RV pops.
29.0	All 4 S/RVs are completely closed.
29.7	Two out of 13 S/RVs start to open.
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 35.3°C (95.54°F).
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 5.50 m (216.00 in.) above Level 0, or 4.50 m (176.94 in.) above TAF.
90.0	Suppression pool water average temperature is approximately 35.4°C (95.72°F).
101.0	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the subsequent cyclings of HPCI and RCIC injections.
625	Wide range sensed water level reaches low water level setpoint (Level 2), i.e., 4.18 m (164.50 in.) above Level 0 at 2/3 core height, or 2.96 m (116.50 in.) above TAF.
625	HPCI and RCIC systems are automatically turned on. The HPCI and RCIC turbine pumps are driven by steam generated by decay heat. System auxiliaries are powered by the 250 V dc system.
655	HPCI and RCIC flows enter the reactor pressure vessel at 315 ℓ/s (5000 gpm) and 38 ℓ/s (600 gpm), respectively, drawing water from the condensate storage tank.
12.5 min.	Narrow range sensed water level reaches Level 8 setpoint, i.e., 6.86 m (270.00 in.) above Level 0, or 5.64 m (222.00 in.) above TAF.
12.5 min.	HPCI and RCIC trip off.

Time (sec)	Event
20 min.	Drywell and wetwell temperatures exceed 70°C (158°F) and 50°C (122°F), respectively.
	Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	35.04	4.63×10^3	9.87×10^7	5.61×10^6
Hydrogen	0	0	0	0

Time (sec)	Event
25.5 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The HPCI/ RCIC systems are not isolated.
26.5 min.	Wide range sensed water level reaches Level 2 setpoint and HPCI automatically restarts. (RCIC does not automatically restart.)
27.0 min.	HPCI flow enters the RPV.
29.0 min.	Narrow range sensed water level reaches Level 8 setpoint and HPCI trips off again. The HPCI system, driven by steam generated by decay heat, turns on and off automatically between sensed Levels 2 and 8 until the batteries run out.
240 min.	The HPCI pump stops when the batteries run out.
261 min.	Wide range sensed water level reaches Level 2 setpoint. Drywell and wetwell temperatures are 100°C (212°F) and 96°C (205°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	10.45	1.38×10^3	2.92×10^7	1.66×10^6
Hydrogen	0	0	0	0

Time (sec)	Event
315.07 min.	Core uncovery time. Steam-water mixture level is at 3.53 m (11.58 ft) above bottom of the core.
321.5 min.	Average gas temperature at top of core is 211°C (412°F). Drywell and wetwell temperatures and pressures are 106°C (223°F) and 0.24 MPa (35 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	22.66	3.00×10^3	6.33×10^7	3.60×10^6
Hydrogen	0	0	0	0

Time (sec)	Event
384 min.	Average gas temperature at top of core is 1046°C (1915°F). Drywell and wetwell temperatures and pressures are 114°C (237°F) and 0.28 MPa (41 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	2.46	325.0	8.30×10^6	4.72×10^5
Hydrogen	0.021	2.78	1.14×10^5	8.19×10^3

Time (sec)	Event
387.77 min.	Core melting starts.
418.77 min.	Water level in vessel drops below bottom grid elevation.
420.77 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
421.67 min.	The corium slumps down to vessel bottom.
422.12 min.	Debris starts to melt through the bottom head.
515.18 min.	Vessel bottom head fails, resulting in a pressure increase of 0.34 MPa (49 psia).
515.20 min.	Debris starts to boil water from containment floor.
515.20 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 204°C (400°F) and start to vent through the primary containment at a leak rate of 105 ℓ/s (222 ft^3/min).
515.21 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 1771°C (3219°F) initially. Internal heat generation in metals and oxides are 1.01×10^7 and 1.86×10^7 watts, respectively.
579.24 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	0.63	83.33	1.59×10^5	9052
Hydrogen	0.22	29.10	0	0
CO_2	2.08	275.13		
CO	4.63	612.44		

The leak rate through the drywell penetration seals is ~2.53×10^5 ℓ/s (5.35×10^5 ft^3/min).

Time (sec)	Event
681.88 min.	Drywell and wetwell pressures are at 0.10 MPa (~14.7 psia) and temperatures are 674°C (~1245°F) and 98°C (~209°F), respectively. The leak rate through the containment failed area is ~3.61 × 10^4 ℓ/s (~7.65 × 10^4 ft^3/min).
810 min.	Drywell and wetwell temperatures are 1006°C (1843°F) and 97°C (207°F), respectively. The leak rate through the containment failed area is ~3.10 × 10^4 ℓ/s (~6.57 × 10^4 ft^3/min).
1128 min.	Drywell and wetwell temperatures are 591°C (~1095°F) and 97°C (~207°F), respectively. The leak rate through the containment failed area is ~1.27 × 10^3 ℓ/s (~2.70 × 10^3 ft^3/min).

is depressurized during core uncovery and melt. However, the pressure remains high enough so that the HPCI turbine can be operated for as long as the battery lasts. Compared with the sequence TB′, the SORV results in a ~30 min. delay in the core melt, ~90 min. delay in vessel failure, and ~65 min. delay in the containment failure.

9.4.2.3 <u>CSB + Manual RCIC & SRV (T_VB')</u>. The accident progression signature for the sequence T_VB' is given in Table 9.3. In this sequence, it is assumed that the HPCI system is inoperable. The operator remote-manually opens one SRV after 15 minutes of Station Blackout to depressurize the vessel in order to lower the vessel temperature and to prepare for use of the low pressure ECCS injection systems upon restoration of AC power. At the same time, the operator attempts to maintain a constant vessel water level by manually controlling the RCIC injection. It is noted that the fluid loss through the SRV causes the core to become momentarily uncovered, but it refloods immediately with the continued RCIC water injection. This short period of core uncovery is believed not to cause any core damage at that stage.

9.4.2.4 <u>CSB + Manual RCIC & SRV + SORV (T_VPB')</u>. The accident progression signature for the sequence T_VPB' is given in Table 9.4. In this sequence, it is assumed that an SORV event occurs at the beginning of the transient, and the operator remote-manually opens a SRV which causes the reactor vessel to depressurize at a faster rate. Compared with the sequence T_VB', the two sequences have produced very similar results with respect to the times of core melt, vessel failure, and containment failure.

9.4.2.5 <u>CSB + No HPCI/RCIC (TUB′)</u>. The accident progression signature for the sequence TUB′ is given in Table 9.5. The events of this sequence as calculated by MARCH have been compared with results from the RELAP4/MOD7 code[27] up to the time of core uncovery. With the modifications incorporated in MARCH version 1.4B, the core uncovery times predicted by the two codes are in agreement.

9.4.2.6 <u>CSB + No HPCI/RCIC & SORV (TUPB′)</u>. The accident progression signature for the sequence TUPB′ is given in Table 9.6. This is the severest of the six Station Blackout signatures studied, and includes a prediction of core uncovery at 17 minutes after the initiating loss of AC power. The mitigating action which might be taken by the operator in response to this and the other accident progression signatures will be discussed in Section 10.

9.4.2.7 <u>Summary of key events</u>. A comparison of the predicted times to key events following core uncovery for each of the six sequences as calculated by MARCH version 1.4B is given in Table 9.7.

9.4.2.8 <u>Key results for sequence TB′</u>. Key results from the MARCH calculations for the sequence TB' which is used as the base case for the fission product transport calculations in Volume 2 are presented in Figs. 9.12 through 9.21.

Table 9.3

Browns Ferry Nuclear Plant: Complete Station Blackout
Sequence of Events

CSB + Manual RCIC & SRV

(T_VB')

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.

Time (sec)	Event
3.0	Turbine trips off (turbine stop valves fully closed).
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Seven (7) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 0.51 m (20 in.) from the previous momentary 1.02 m (40-in.) drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 6 S/RVs are completely closed. One S/RV is stuck open (SORV); this has the same effect as a small break LOCAL of equivalent break area of 0.015 m^2 (0.158? ft^2).
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 5.98 m (235.50 in.) above Level 0, or 5.00 m (196.44 in.) above TAF.
22.0	Suppression pool water average temperature rises to 35.13°C (95.24°F) in response to the first S/RV pops.

Time (sec)	Event
29.0	All 4 S/RVs are completely closed.
29.7	Two out of 13 S/RVs start to open.
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 35.3°C (95.54°F).
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 5.50 m (216.00 in.) above Level 0, or 4.50 m (176.94 in.) above TAF.
90.0	Suppression pool water average temperature is approximately 35.4°C (95.72°F).
101.0	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the subsequent RCIC injections.
625	Wide range sensed water level reaches low water level setpoint (Level 2), i.e., 4.18 m (164.50 in.) above Level 0 at 2/3 core height, or 2.96 m (116.50 in.) above TAF.
625	Operator manually controls RCIC injection to maintain constant vessel water level. The RCIC turbine pump is driven by steam generated by decay heat. System auxiliaries are powered by the 250 V dc system.
655	RCIC flows enter the reactor pressure vessel at 38 ℓ/s (600 gpm) drawing water from the condensate storage tank.
15 min.	Operator manually opens one SRV to depressurize the vessel.
20 min.	Drywell and wetwell temperatures exceed 76°C (169°F) and 50°C (122°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	829.75	1.10×10^5	2.32×10^8	1.32×10^7
Hydrogen	0	0	0	0

Time (sec)	Event
21.14 min.	Core uncovery time.
22.0 min.	Core refloods.
30 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The RCIC system is not isolated.
240 min.	The RCIC pump stops when the batteries run out.
266.3 min.	Wide range sensed water level reaches Level 2 setpoint. Drywell and wetwell temperatures are 99°C (210°F) and 100°C (212°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	19.16	2.53×10^3	5.20×10^7	2.96×10^6
Hydrogen	0	0	0	0

347 min.	Core uncovers again.
366 min.	Average gas temperature at top of core is 491°C (916°F). Drywell and wetwell temperatures and pressures are 113°C (236°F) and 0.28 MPa (40 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	9.26	1.22×10^3	2.97×10^7	1.69×10^6
Hydrogen	4.09×10^{-5}	5.41×10^{-3}	222.28	12.64

386 min.	Average gas temperature at top of core is 855°C (1571°F). Drywell and wetwell temperatures and pressures are 115°C (239°F) and 0.29 MPa (41 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	5.05	6.68×10^2	1.81×10^7	1.03×10^6
Hydrogen	1.68×10^{-2}	2.23	1.35×10^5	7.70×10^3

Time (sec)	Event
395.3 min.	Core melting starts.
449.3 min.	Water level in vessel drops below bottom grid elevation.
451.2 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
452 min.	The corium slumps down to vessel bottom.
452.9 min.	Debris starts to melt through the bottom head.
539.3 min.	Vessel bottom head fails, resulting in a pressure increase of 0.0047 MPa (0.68 psia).
539.3 min.	Debris starts to boil water from containment floor.
539.3 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 204°C (400°F) and start to vent through the primary containment at a leak rate of 118 ℓ/s (250 ft^3/min).
539.3 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 1750°C (3182°F) initially. Internal heat generation in metals and oxides are 9.99×10^6 and 1.84×10^7 watts, respectively.
601.05 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	4.70	621.51	1.59×10^5	9052
Hydrogen	0.14	18.27	0	0
CO_2	1.29	170.23		
CO	2.88	381.21		

The leak rate through the drywell penetration seals is ~5.33×10^4 ℓ/s (1.13×10^5 ft^3/min).

718.8 min.	Drywell and wetwell pressures are at 0.10 MPa (~14.7 psia) and temperatures are 700°C (1293°F) and 98°C (~209°F), respectively. The leak rate through the containment failed area is ~5.18×10^4 ℓ/s (~1.10×10^5 ft^3/min).

Time (sec)	Event
821.5 min.	Drywell and wetwell temperatures are 737°C (1359°F) and 93°C (199°F), respectively. The leak rate through the containment failed area is ~4.23×10^4 ℓ/s (~8.96×10^4 ft^3/min).
1127.5 min.	Drywell and wetwell temperatures are 468°C (~875°F) and 86°C (~188°F), respectively. The leak rate through the containment failed area is ~4.79×10^4 ℓ/s (~1.02×10^4 ft^3/min).

Table 9.4

Browns Ferry Nuclear Plant: Complete Station Blackout
Sequence of Events

CSB + Manual RCIC & SRV + SORV (Small Break LOCA)

($T_V PB'$)

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.

Time (sec)	Event
3.0	Turbine trips off (turbine stop valves fully closed).
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Operator manually opens one S/RV to depressurize the vessel.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 6 S/RVs are completely closed. One S/RV is stuck open (SORV). This has the same effect as a small break LOCA of equivalent break area of 0.0147 m^2 (0.1583 ft^2).
17.0	Neutron flux drops below 1% of initial full power level.
30	Operator manually controls RCIC injection to maintain constant vessel water level. The RCIC turbine pump is driven by steam generated by decay heat. System auxiliaries are powered by the 250 V dc system.
60	RCIC flows enter the reactor pressure vessel at 38 ℓ/s (600 gpm) drawing water from the condensate storage tank.

Time (sec)	Event
10.52 min.	Core uncovery time. Steam-water mixture level is at 3.56 m (11.68 ft) above bottom of the core.
11.52 min.	Core refloods.
20 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The RCIC system is not isolated.
240 min.	The RCIC pump stops when the batteries run out.
270 min.	Wide range sensed water level reaches Level 2 setpoint. Drywell and wetwell temperatures are 100°C (212°F) and 102°C (216°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	19.50	2.58×10^3	5.29×10^7	3.01×10^6
Hydrogen	0	0	0	0

Time (sec)	Event
337 min.	Core uncovers again.
356 min.	Average gas temperature at top of core is 313°C (595°F). Drywell and wetwell temperatures and pressures are 113°C (236°F) and 0.28 MPa (40 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	12.25	1.62×10^3	3.64×10^7	2.07×10^6
Hydrogen	3.05×10^{-8}	4.03×10^{-6}	0.115	6.53×10^{-3}

Time (sec)	Event
376.4 min.	Average gas temperature at top of core is 650°C (1202°F). Drywell and wetwell temperatures and pressures are 115°C (239°F) and 0.29 MPa (41 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	7.20	9.52×10^2	2.43×10^7	1.38×10^6
Hydrogen	1.60×10^{-3}	0.21	1.07×10^4	6.09×10^2

Time (sec)	Event
396.36 min.	Core melting starts.
450.4 min.	Water level in vessel drops below bottom grid elevation.
452.3 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
453.1 min.	The corium slumps down to vessel bottom.
454. min.	Debris starts to melt through the bottom head.
542.5 min.	Vessel bottom head fails, resulting in a pressure increase of 0.005 MPa (0.7 psia).
542.5 min.	Debris starts to boil water from containment floor.
542.5 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 240°C (400°F) and start to vent through the primary containment at a leak rate of 104 ℓ/s (221 ft^3/min).
542.5 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 1766°C (3210°F) initially. Internal heat generation in metals and oxides are 1.00×10^7 and 1.83×10^7 watts, respectively.
596.4 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment. Mass and energy addition rates into the drywell are:

	Mass Rate		Energy Rate	
	(kg/s	(lb/min)	(w)	(Btu/min)
Steam	7.69	1017	1.59×10^5	9052
Hydrogen	0.038	5.08	0	0
CO_2	2.58	342		
CO	0.80	160		

The leak rate through the drywell penetration seals is ~9.20×10^4 ℓ/s (1.95×10^5 ft^3/min).

Table 9.5

Browns Ferry Nuclear Plant: Complete Station Blackout
Sequence of Events

CSB + No HPCI/RCIC

(TUB′)

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.
3.0	Turbine trips off (turbine stop valves fully closed).

Time (sec)	Event
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Seven (7) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 0.51 m (20 in.) from the previous momentary 1.02 m (40-in.) drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 7 S/RVs are completely closed.
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 5.98 m (235.50 in.) above Level 0, or 5.00 m (196.44 in.) above TAF.
22.0	Suppression pool water average temperature rises to 35.13°C (95.24°F) in response to the first S/RV pops.
29.0	All 4 S/RVs are completely closed.
29.7	Two out of 13 S/RVs start to open.

Time (sec)	Event
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 35.3°C (95.54°F).
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 5.50 m (216.00 in.) above Level 0, or 4.50 m (176.94 in.) above TAF.
90.0	Suppression pool water average temperature is approximately 35.4°C (95.72°F).
101.0	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the sequence.
625	Wide range sensed water level reaches low water level set-point (Level 2), i.e., 4.18 m (164.50 in.) above Level 0 at 2/3 core height, or 2.96 m (116.50 in.) above TAF.
625	HPCI and RCIC systems are not turned on because they are assumed to be unavailable.
20 min.	Suppression pool water average temperature reaches 46°C (114°F).
33 min.	Core uncovery time. Steam-water mixture level is at 3.54 m (11.61 ft) above bottom of the core.
40 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The HPCI/RCIC systems are not affected. Drywell and wetwell temperature are 72°C (162°F) and 55°C (130°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	33	4.36×10^3	9.25×10^7	5.26×10^6
Hydrogen	6×10^{-9}	8.62×10^{-7}	2.92×10^{-2}	1.66×10^{-3}

Time (sec)	Event
60 min.	Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	15.6	2.07×10^3	5.06×10^7	2.88×10^6
Hydrogen	2.8×10^{-3}	3.76×10^{-1}	2.15×10^4	1.22×10^3

Time (sec)	Event
70 min.	Core melting starts.
80 min.	Drywell and wetwell temperatures are 75°C (167°F) and 63°C (145°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	5.68	7.51×10^2	2.22×10^7	1.26×10^6
Hydrogen	0.19	2.53×10^1	2.29×10^6	1.30×10^5

Time (sec)	Event
96 min.	Water level in vessel drops below bottom grid elevation.
97 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
99 min.	The corium slumps down to vessel bottom.
101 min.	The debris is starting to melt through the bottom head. Drywell and wetwell temperatures are 97°C (207°F) and 71°C (159°F), respectively. Meanwhile, local pool water temperature at the discharging bay exceeds 149°C (300°F). Steam condensation oscillations could accelerate due to the continuous discharge of superheated noncondensable gases into the suppression pool. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	18.6	5.46×10^3	5.42×10^7	3.08×10^6
Hydrogen	6.8×10^{-2}	8.93	3.59×10^5	2.04×10^4

Time (sec)	Event
129 min.	Vessel bottom head fails, resulting in a pressure increase of 0.34 MPa (49 psia).
129.03 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 1546°C (2815°F) initially. Internal heat generation in metals and oxides are 1.36×10^7 and 2.50×10^7 watts, respectively.

Time (sec)	Event
165 min.	Drywell and wetwell temperatures are 141°C (286°F) and 74°C (166°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	5.46	722.83	1.59×10^5	9052
Hydrogen	3.3×10^{-2}	4.38	0	0
CO_2	2.58	341.88		
CO	0.69	91.35		

Time (sec)	Event
190 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 204°C (400°F) and start to vent through the primary containment.
193 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment.
219 min.	Drywell and wetwell pressures are at 0.10 MPa (14.7 psia). Drywell and wetwell temperatures are 598°C (1109°F) and 78°C (173°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	0.70	92	1.59×10^5	9052
Hydrogen	0.24	32	0	0
CO_2	2.32	307		
CO	5.03	666		

The leak rate through the containment failed areas is $\sim 2.90 \times 10^5$ ℓ/s ($\sim 6.15 \times 10^5$ ft^3/min).

Time (sec)	Event
250 min.	Drywell and wetwell temperatures are 675°C (1247°F) and 78°C (173°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	6.84	905	1.59×10^5	9052
Hydrogen	0.25	33	0	0
CO_2	1.53	203		
CO	5.25	695		

Time (sec)	Event
	The leak rate through the containment failed area is $\sim 4.91 \times 10^4$ ℓ/s ($\sim 1.04 \times 10^5$ ft^3/min).
309 min.	Rate of concrete decomposition is $\sim 4.65 \times 10^4$ gm/s. Rate of heat added to atmosphere is $\sim 1.20 \times 10^4$ kW.
367 min.	Drywell and wetwell pressures are at 0.10 MPa ($\sim$14.7 psia) and temperatures are 854°C (1570°F) and 77°C (171°F), respectively. The leak rate through the containment failed area is $\sim 3.94 \times 10^4$ ℓ/s ($\sim 8.35 \times 10^4$ ft^3/min).
733 min.	Drywell and wetwell temperatures are 546°C ($\sim$1014°F) and 77°C (170°F), respectively. The leak rate through the containment failed area is $\sim 2.12 \times 10^3$ ℓ/s ($\sim 4.50 \times 10^3$ ft^3/min).

Table 9.6

Browns Ferry Nuclear Plant: Complete Station Blackout
Sequence of Events

CSB + No HPCI/RCIC & SORV (Small Break LOCA)

(TUPB´)

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.
3.0	Turbine trips off (turbine stop valves fully closed).

Time (sec)	Event
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Seven (7) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 0.51 m (20 in.) from the previous momentary 1.02 m (40-in.) drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 6 S/RVs are completely closed. One S/RV is stuck open (SORV); this has the same effect as a small break LOCA of equivalent break area of 0.015 m^2 (0.1583 ft^2).
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 5.98 m (235.50 in.) above Level 0, or 5.00 m (196.44 in.) above TAF.
22.0	Suppression pool water average temperature rises to 35.13°C (95.24°F) in response to the first S/RV pops.
29.0	All 4 S/RVs are completely closed.
29.7	Two out of 13 S/RVs start to open.

Time (sec)	Event
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 35.3°C (95.54°F).
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 5.50 m (216.00 in.) above Level 0, or 4.50 m (176.94 in.) above TAF.
90.0	Suppression pool water average temperature is approximately 35.4°C (95.72°F).
101.0	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the sequence.
625	Wide range sensed water level reaches low water level set-point (Level 2), i.e., 4.18 m (164.50 in.) above Level 0 at 2/3 core height, or 2.96 m (116.50 in.) above TAF.
625	HPCI and RCIC systems are not turned on because they are assumed to be unavailable.
17.2 min.	Core uncovery time. Steam-water mixture level is at 3.62 m (11.88 ft) above bottom of the core.
20 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The HPCI/RCIC systems are not affected. Drywell and wetwell temperature are 73°C (163°F) and 55°C (130°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	60	7.94×10^{3}	1.69×10^{8}	9.61×10^{6}
Hydrogen	8.53×10^{-13}	1.13×10^{-10}	3.19×10^{-6}	1.81×10^{-7}

Time (sec)	Event
40 min.	Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	10.30	1.36×10^{3}	3.41×10^{7}	1.94×10^{6}
Hydrogen	2.38×10^{-4}	3.15×10^{-2}	1.61×10^{3}	91.56

Time (sec)	Event
56.6 min.	Core melting starts.
60 min.	Drywell and wetwell temperatures are 75°C (167°F) and 63°C (145°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	2.73	3.61×10^2	1.23×10^7	6.99×10^5
Hydrogen	0.49	6.48×10^1	7.05×10^6	4.01×10^5

Time (sec)	Event
78 min.	Water level in vessel drops below bottom grid elevation.
79 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
81 min.	The corium slumps down to vessel bottom.
81.5 min.	The debris is starting to melt through the bottom head. Drywell and wetwell temperatures are 82°C (180°F) and 71°C (159°F), respectively. Meanwhile, local pool water temperature at the discharging bay exceeds 149°C (300°F). Steam condensation oscillations could accelerate due to the continuous discharge of superheated noncondensable gases into the suppression pool.
101 min.	Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	1.25	165.35	3.47×10^6	1.97×10^5
Hydrogen	4.45×10^{-4}	0.06	1.03×10^3	58.58

Time (sec)	Event
142.5 min.	Vessel bottom head fails, resulting in a pressure increase of 0.34 MPa (49 psia).
152.5 min.	Debris starts to boil water from containment floor.
162.5 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 2013°C (3655°F) initially. Internal heat generation in metals and oxides are 2.43×10^7 and 1.26×10^7 watts, respectively.

Time (sec)	Event
162.5 min.	Drywell and wetwell temperatures are 128°C (262°F) and 74°C (166°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s)	(lb/min)	Energy Rate (w)	(Btu/min)
Steam	0.057	7.54	1.59×10^5	9052
Hydrogen	0	0	0	0

Time (sec)	Event
167.8 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 204°C (400°F) and start to vent through the primary containment.
175.2 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment.
185.3 min.	Drywell and wetwell pressures are at 0.10 MPa (14.7 psia). Drywell and wetwell temperatures are 314°C (598°F) and 78°C (173°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s)	(lb/min)	Energy Rate (w)	(Btu/min)
Steam	1.65	218.26	1.59×10^5	9052
Hydrogen	0.025	3.31	0	0
CO_2	5.79	765.88		
CO	0.526	69.58		

The leak rate through the containment failed areas is $\sim 3.00 \times 10^4$ ℓ/s ($\sim 6.36 \times 10^4$ ft^3/min).

Time (sec)	Event
206 min.	Drywell and wetwell temperatures are 610°C (1130°F) and 78°C (173°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s)	(lb/min)	Energy Rate (w)	(Btu/min)
CO_2	1.83	242		
Hydrogen	0.20	26	0	0
Steam	1.36	180	1.59×10^5	9052
CO	4.15	549		

Time (sec)	Event
	The leak rate through the containment failed area is ~2.94×10^4 ℓ/s (~6.24×10^4 ft^3/min).
222.5 min.	Rate of concrete decomposition is ~4.46×10^4 gm/s. Rate of heat added to atmosphere is ~3.71×10^4 kW.
254.5 min.	Drywell and wetwell pressures are at 0.10 MPa (~14.7 psia) and temperatures are 746°C (1375°F) and 77°C (171°F), respectively. The leak rate through the containment failed area is ~5.54×10^4 ℓ/s (~1.17×10^5 ft^3/min).
501 min.	Drywell and wetwell temperatures are 815°C (~1500°F) and 77°C (170°F), respectively. The leak rate through the containment failed area is ~2.34×10^4 ℓ/s (~4.96×10^4 ft^3/min).

Table 9.7. Predicted times to key events

Sequence	Event	Accident progression time (min)								
		Core uncover	Core reflood	Core uncover again	Start melt	Core slump	Vessel failure	Wetwell failure[a]	Drywell EPA vent[b]	Drywell failure[c]
1	CSB + HPCI/RCIC	302			355	392	426		503	514
2	CSB + HPCI/RCIC + SORV	315			388	419	515		515	580
3	CSB + Manual RCIC/SRV	21	22	347	395	449	539		539	601
4	CSB + Manual RCIC/SRV + SORV	11	12	337	396	453	543		543	596
5	CSB + No HPCI/RCIC	33			69	95	128	130[d]	190	193[e]
6	CSB + No HPCI/RCIC + SORV	17			57	78	143	145[f]	168	175

[a] Wetwell failure is due to forces of steam jet impingement and condensation oscillations resulting from excessive thermal stratification in the suppression pool.

[b] Drywell electric penetration assembly seals start to vent when the ambient temperature exceeds 204°C (400°F).

[c] Drywell electric penetration assembly seals become decomposed and are blown out of the containment when the ambient temperature exceeds 260°C (500°F).

[d] Best estimate value.

[e] Containment would fail at 288 min. using WASH-1400 failure criterion of static pressurization at 190 psia.

[f] Best estimate value.

Figure 9.12 shows the reactor vessel pressure distribution; it remains approximately constant during the first 240 minutes when the steam mass flow through the SRV discharge into the suppression pool (Fig. 9.13) is balanced by HPCI injection (Fig. 9.14). The pressure increases after the HPCI injection has stopped upon the assumed loss of dc power at 4 h into the transient.

The mass of water and the mass of steam within the reactor vessel are shown in Figs. 9.15 and 9.16 respectively. The relatively rapid decrease in steam inventory beginning at about 305 minutes as shown in Fig. 9.16 is due to depletion in the production of hydrogen (Fig. 9.17) by the Zirconium-water reaction (Fig. 9.18) before the onset of core melting (Fig. 9.19). The large amount of energy released by the Zr-H_2O reaction (Fig. 9.20) before the core slumps into the lower head causes an increase in the steaming rate and the reactor vessel pressure.

The large amount of energy release in the predicted corium-water interaction after reactor vessel failure would cause a significant containment pressure spike. The vertical concrete penetration resulting from corium attack is shown in Fig. 9.21.

9.5 Containment Responses

As previously discussed, containment failure could occur either in the drywell or in the wetwell if a core meltdown accident were to occur at the Browns Ferry Nuclear Plant. The radiological consequences of a wetwell failure would be less severe than those of a drywell failure, because a significant fraction of the released fission products would be dissolved or deposited in the suppression pool.

9.5.1 Drywell Response

Drywell failure would occur in the EPA when the elastomeric sealing materials undergo degradation and lose sealing integrity at ambient temperatures greater than 204°C (400°F)[18, 22]. This does not preclude the possibility of an earlier failure of the wetwell by overpressurization. An EPA typical of those installed in the Browns Ferry unit 1 drywell liner is shown in Fig. 9.22.

The masses of steam and hydrogen accumulated in the drywell and wetwell following core melt in the base case TB′ are shown in Figs. 9.23 through 9.26. For this case, wetwell failure by overpressurization is not preducted to occur.

The drywell temperature responses for the sequences TB′ and TUB′ are shown in Figs. 9.27 and 9.28 respectively; the general trends are very similar in the two sequences, except that high temperatures are reached much sooner in sequence TUB′, as would be expected.* The corresponding drywell pressure responses for these two sequences are shown in Figs. 9.29 through 9.36.

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*In sequence TUB′, it is assumed that the HPCI and RCIC systems are unavailable from the inception of the Station Blackout. Thus boiloff, core uncovery, and the subsequent events would occur sooner.

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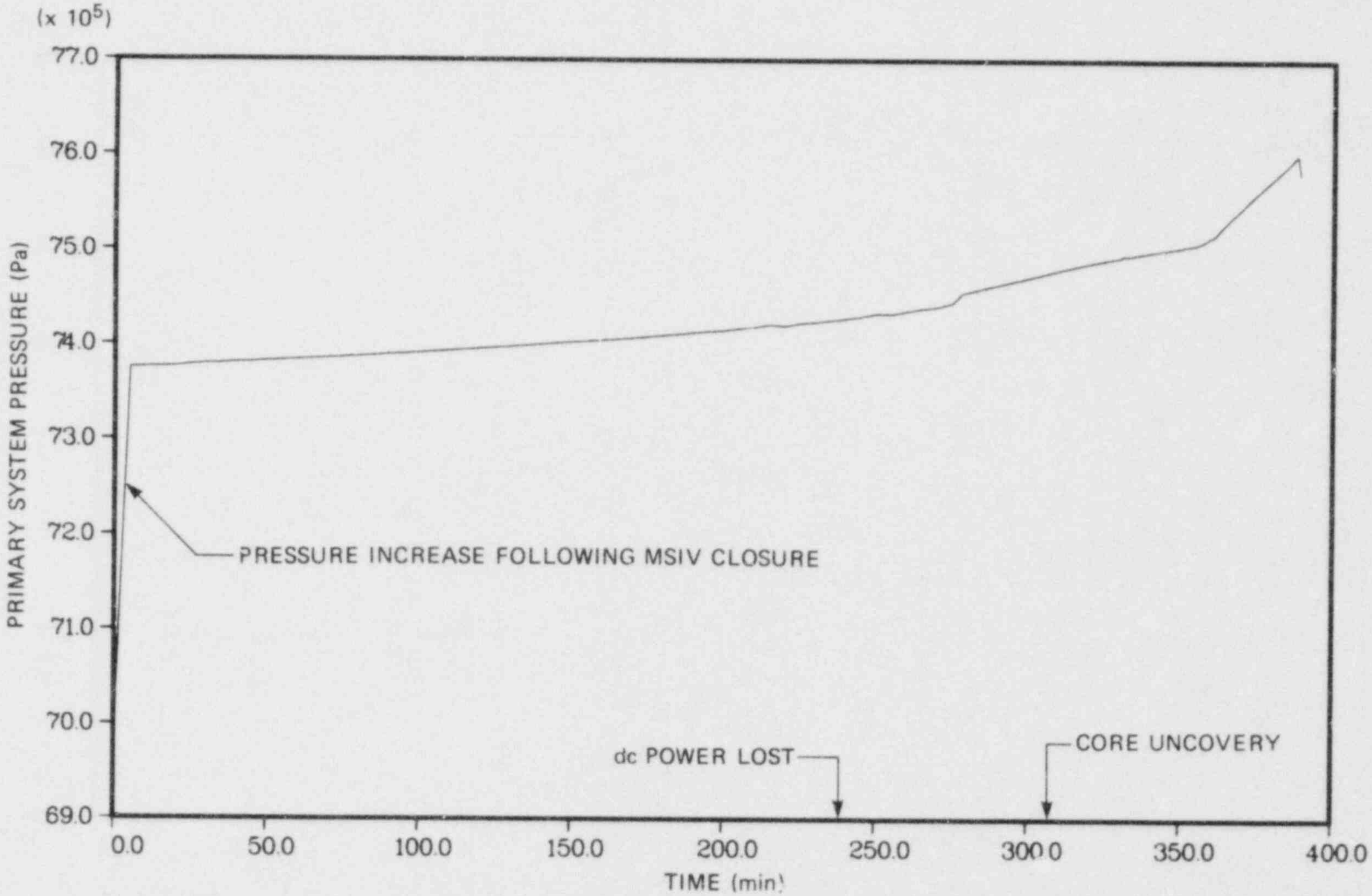

Fig. 9.12 Reactor vessel pressure.

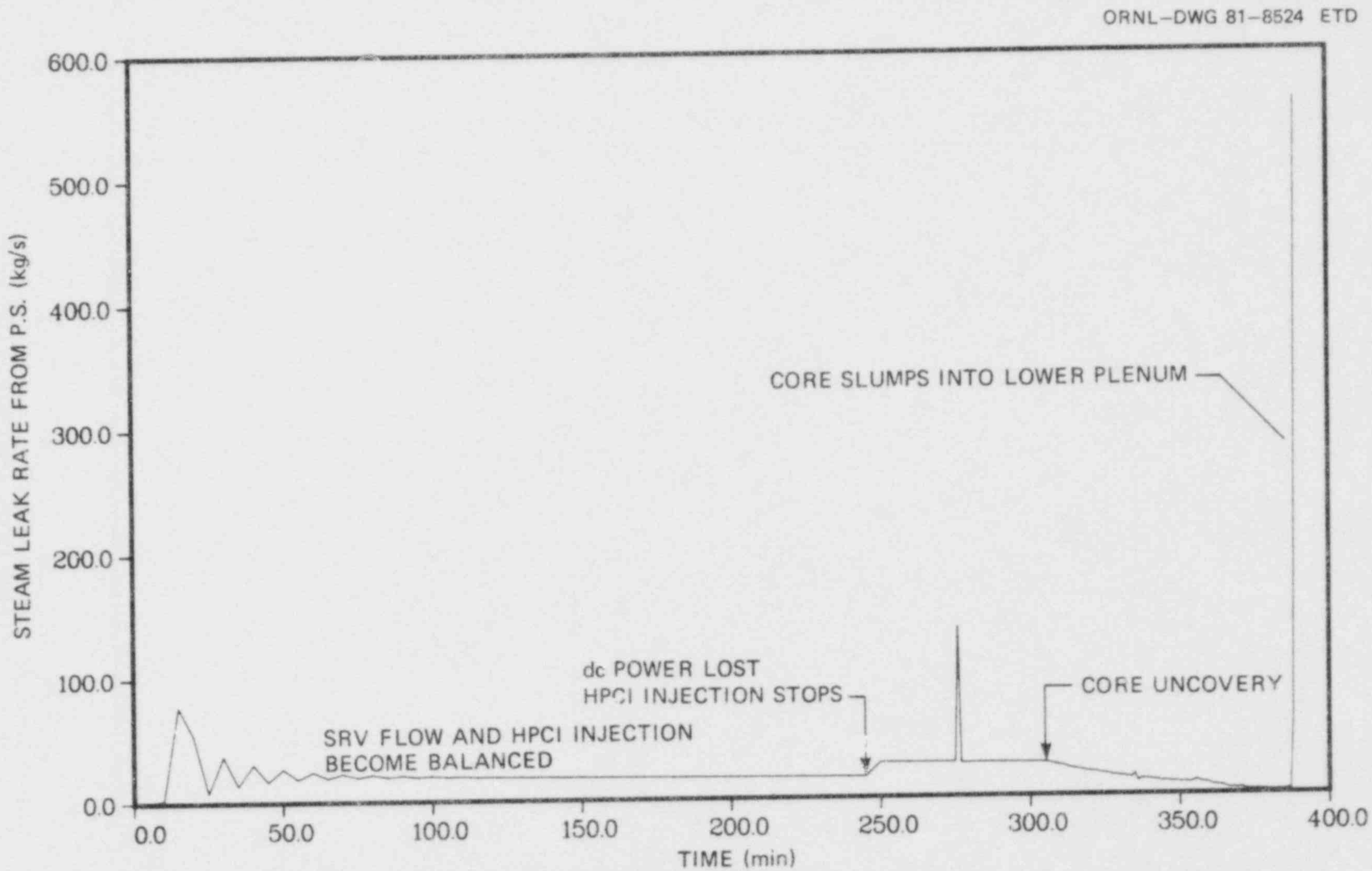

Fig. 9.13 Steam mass flow rate from primary system.

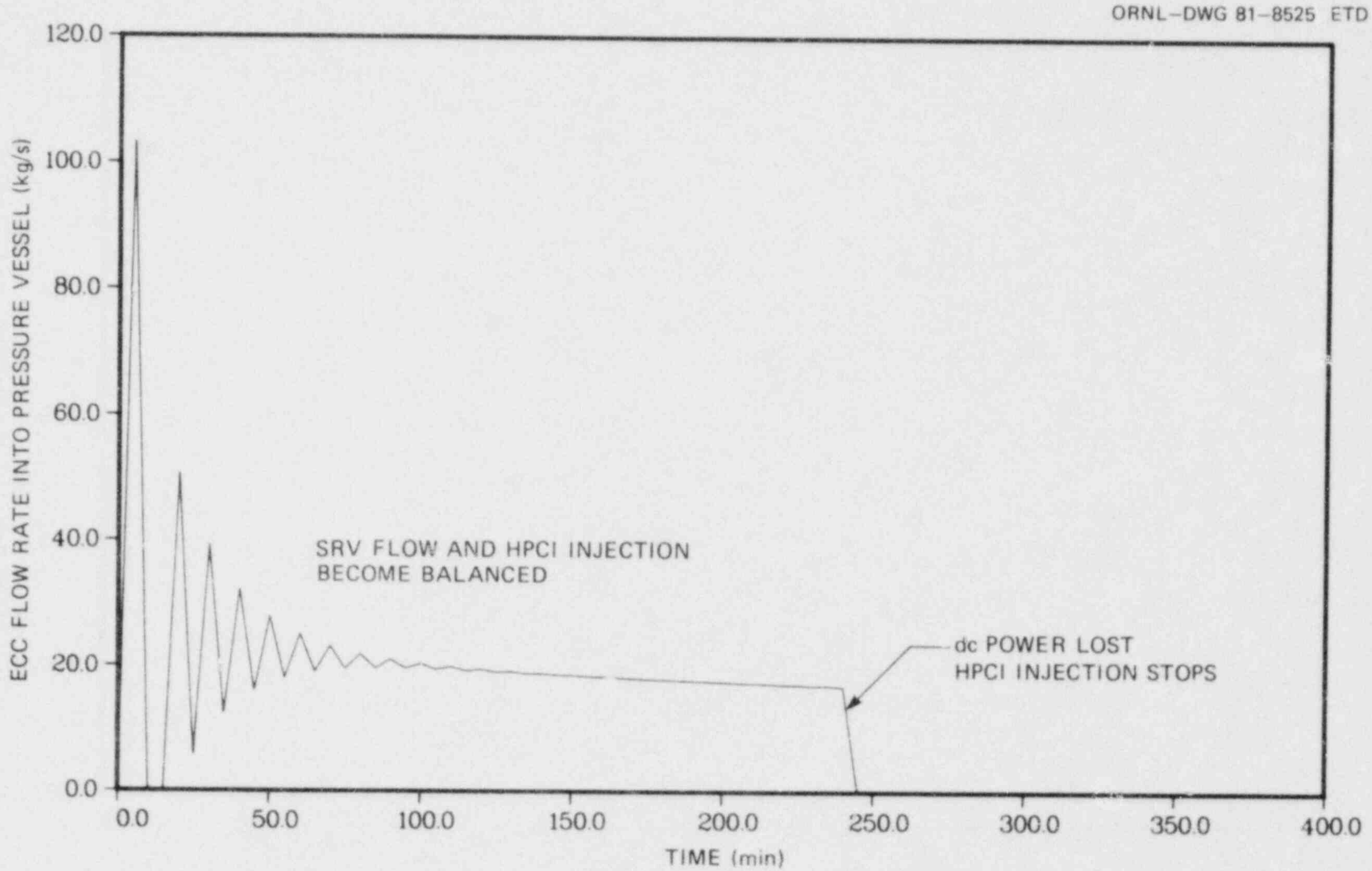

Fig. 9.14 HPCI flow rate into pressure vessel.

ORNL-DWG 81-8526 ETD

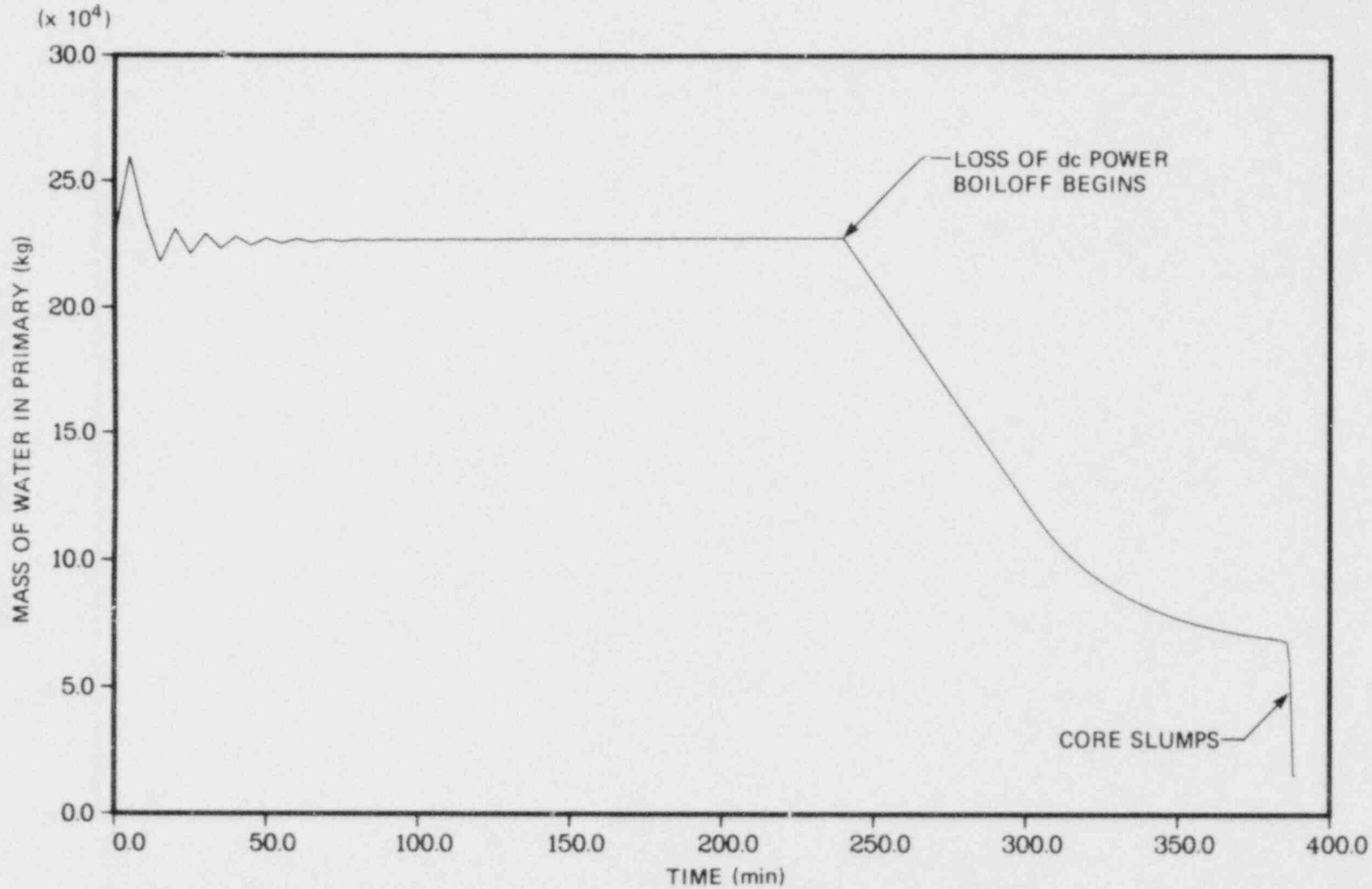

Fig. 9.15 Mass of water in primary system.

ORNL-DWG 81-8527 ETD

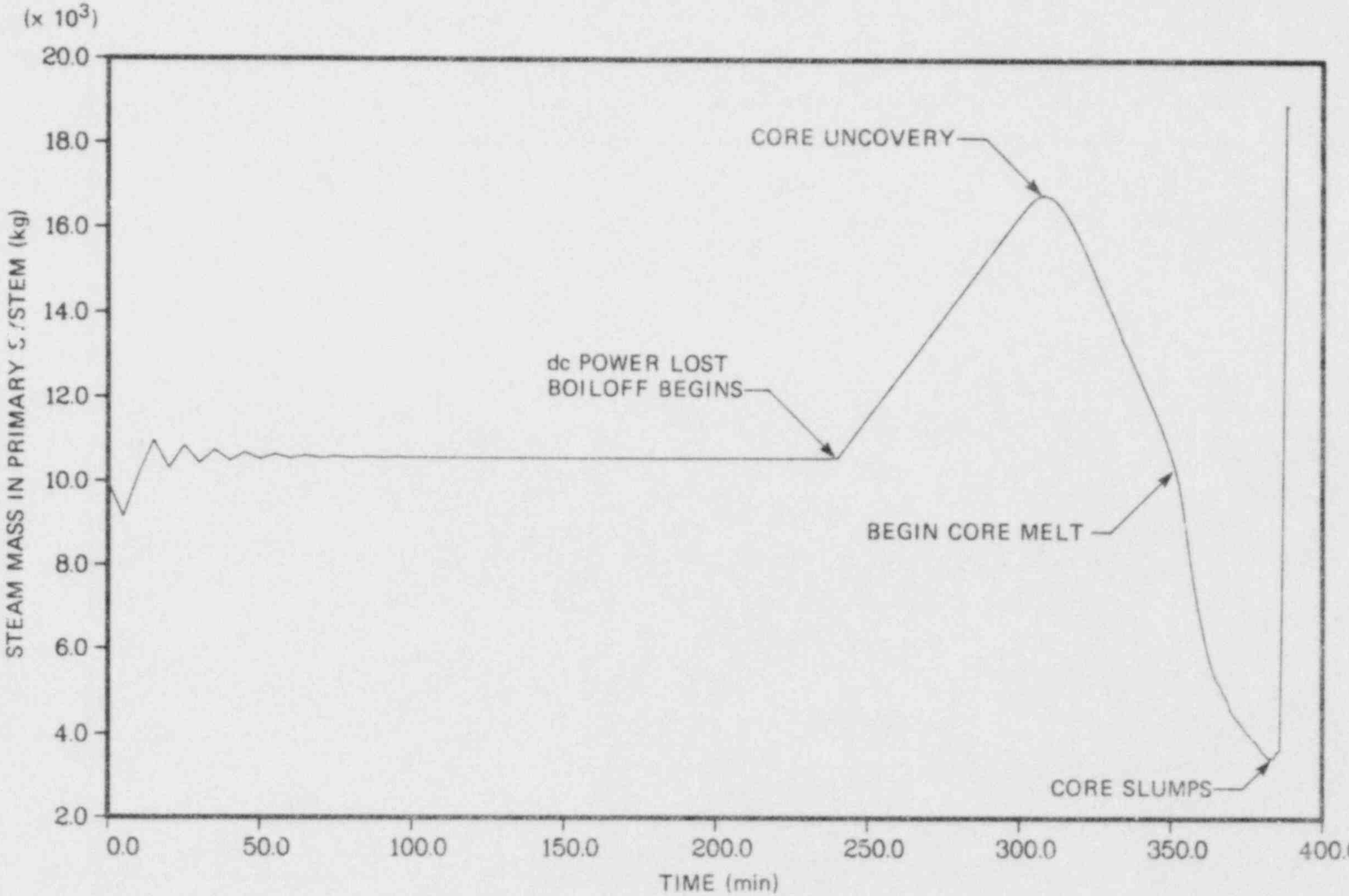

Fig. 9.16 Steam mass in primary system.

ORNL-DWG 81-8528 ETD

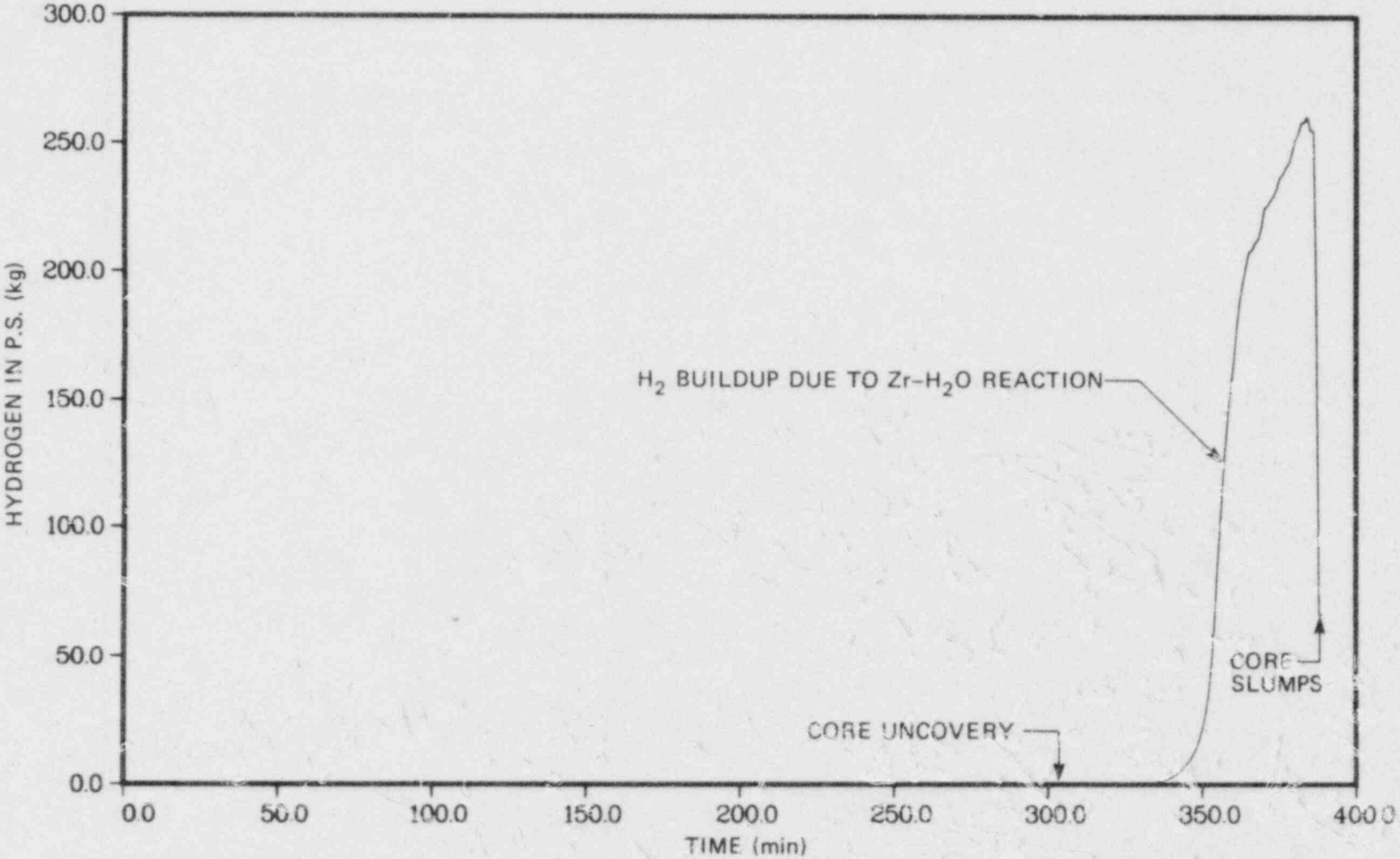

Fig. 9.17 Hydrogen mass in primary system.

ORNL-DWG 81-8529 ETD

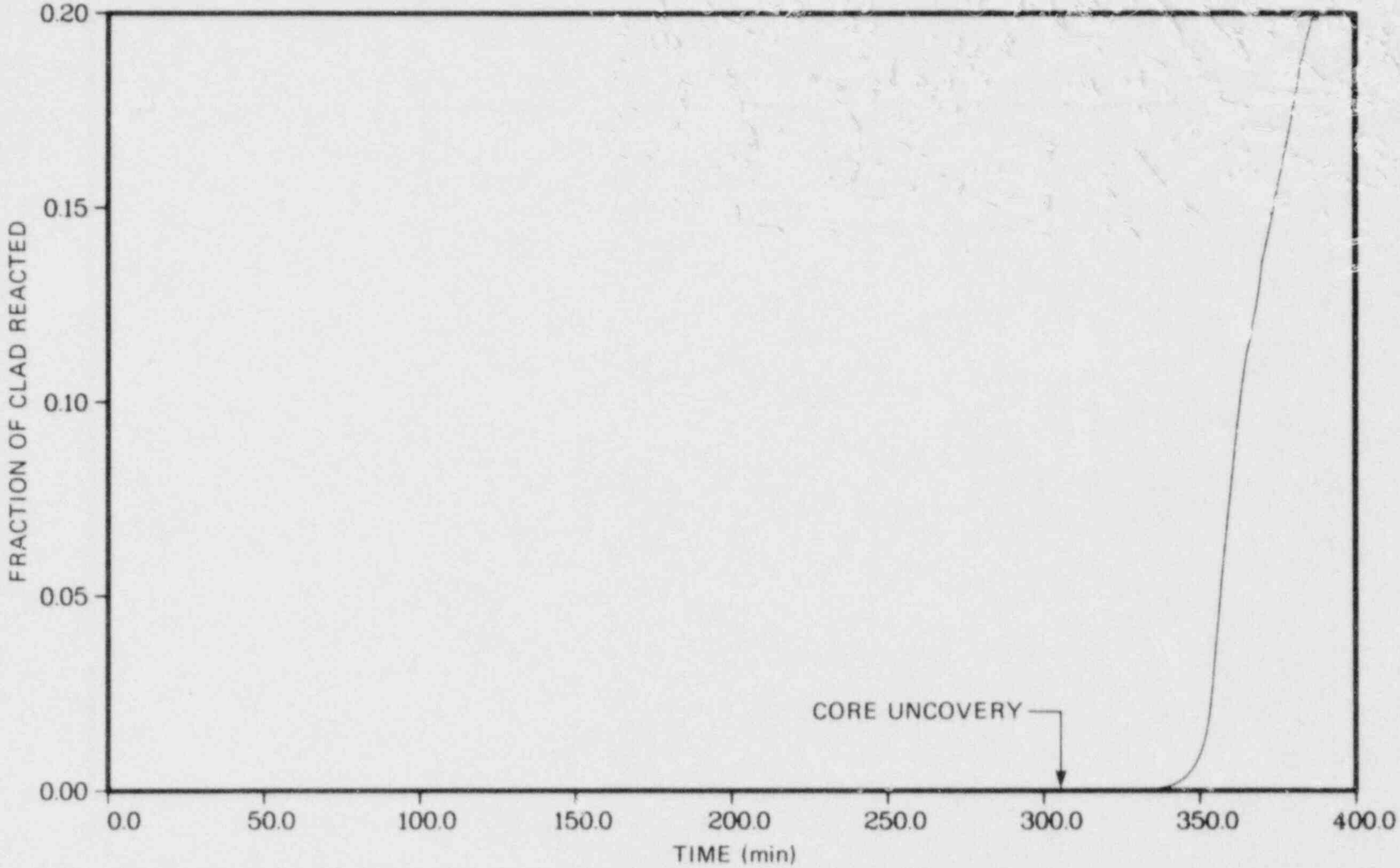

Fig. 9.18 Fraction of clad reacted.

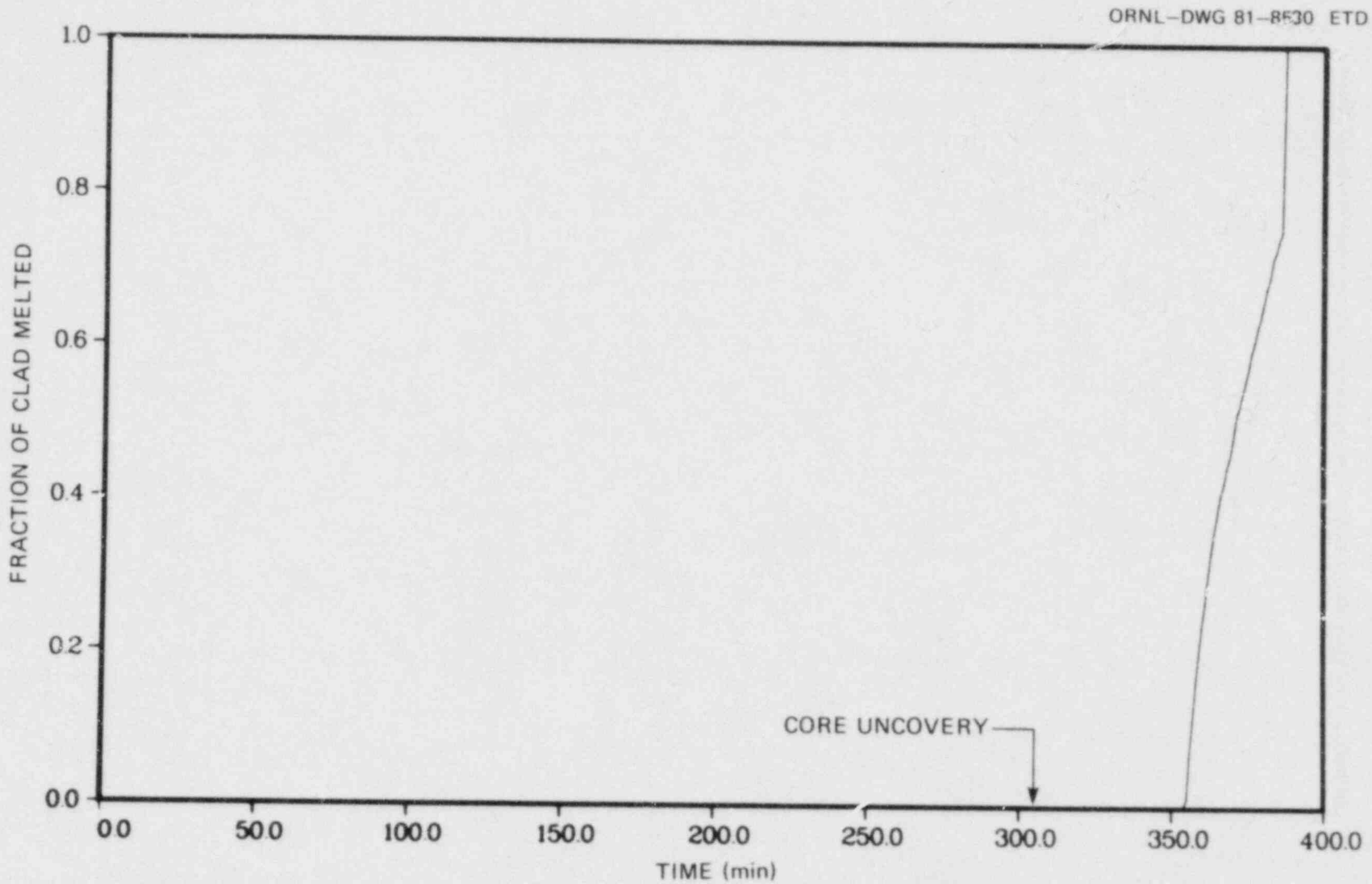

Fig. 9.19 Fraction of core melted.

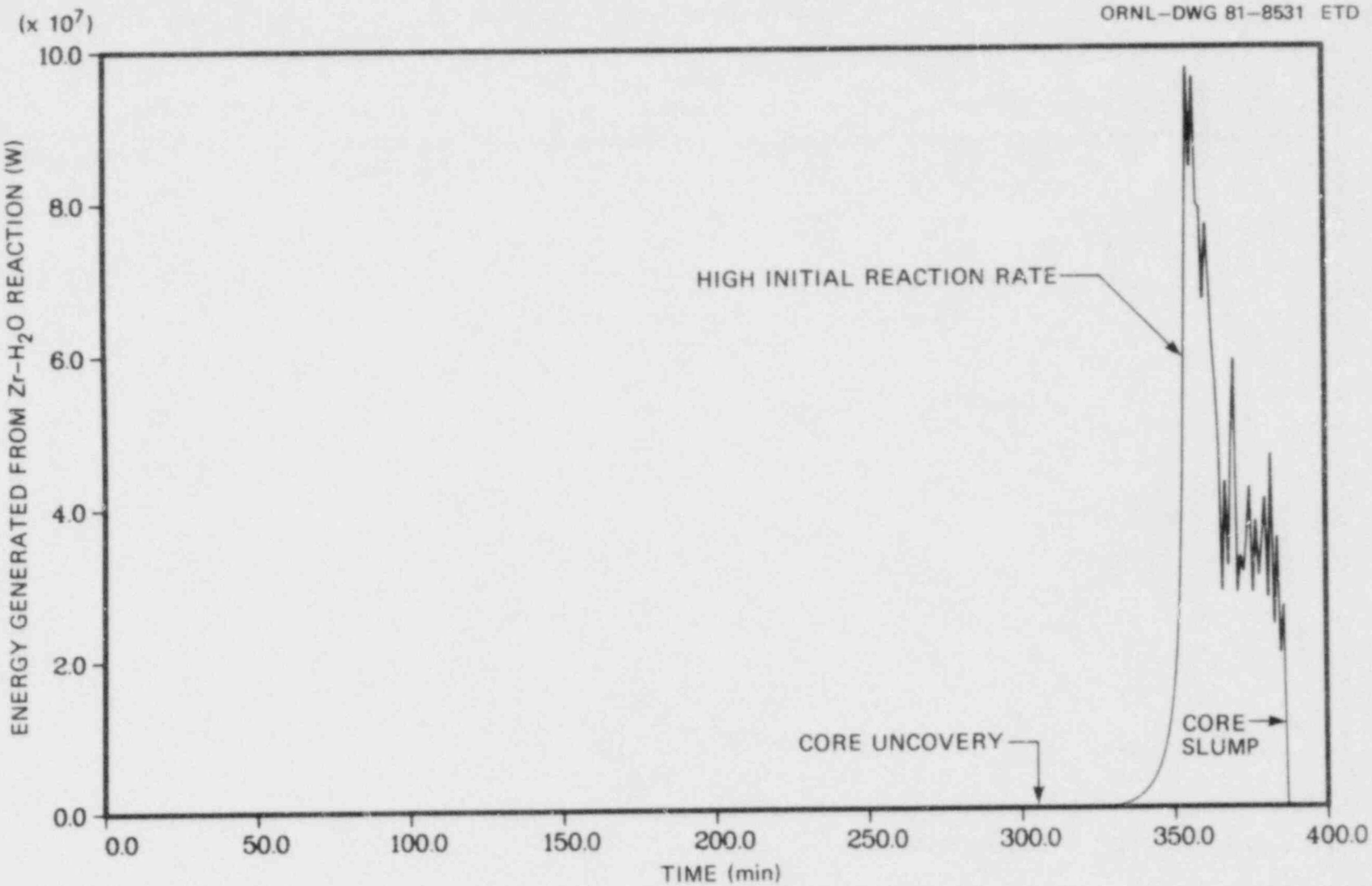

Fig. 9.20 Energy generated from $Zr-H_2O$ reaction.

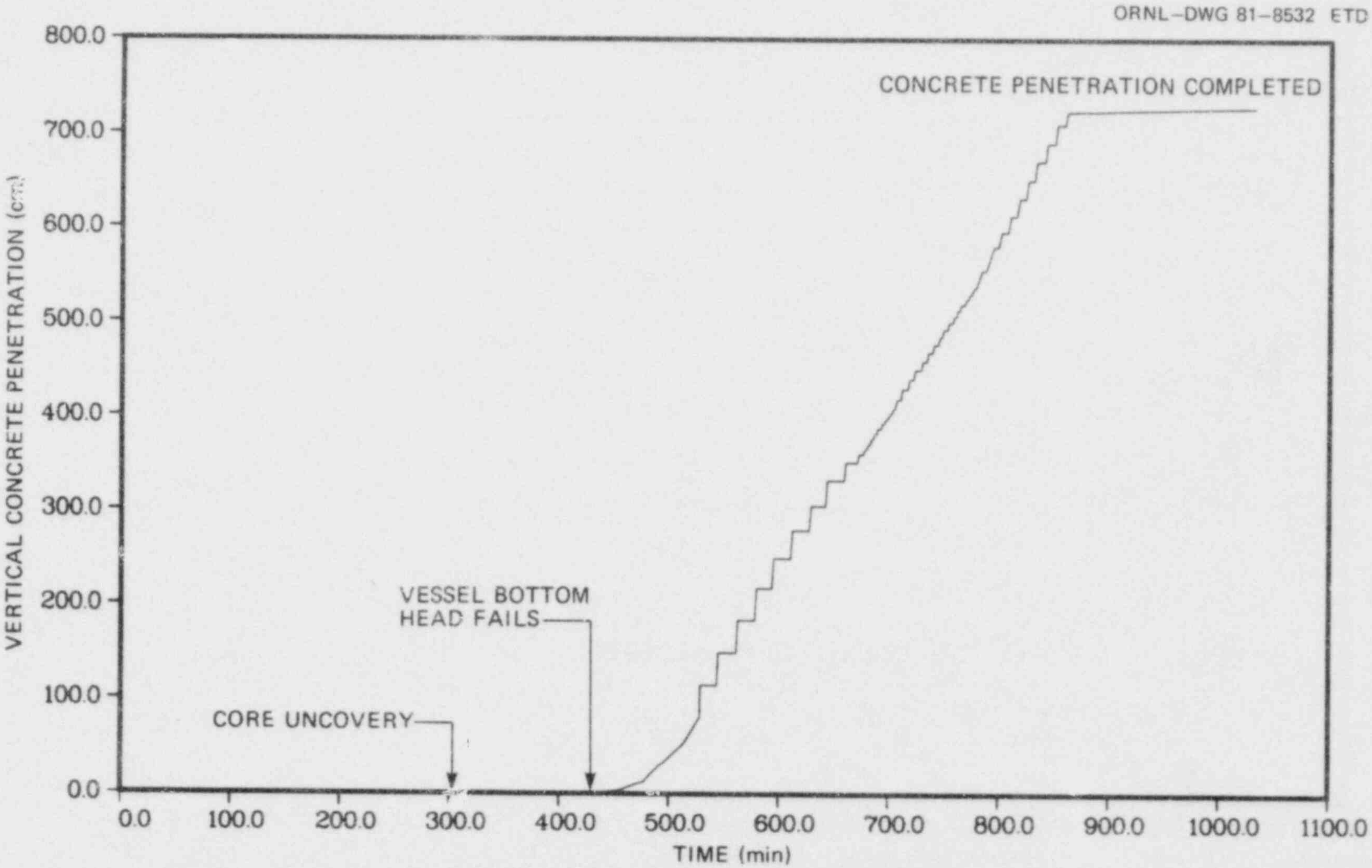

Fig. 9.21 Vertical concrete penetration.

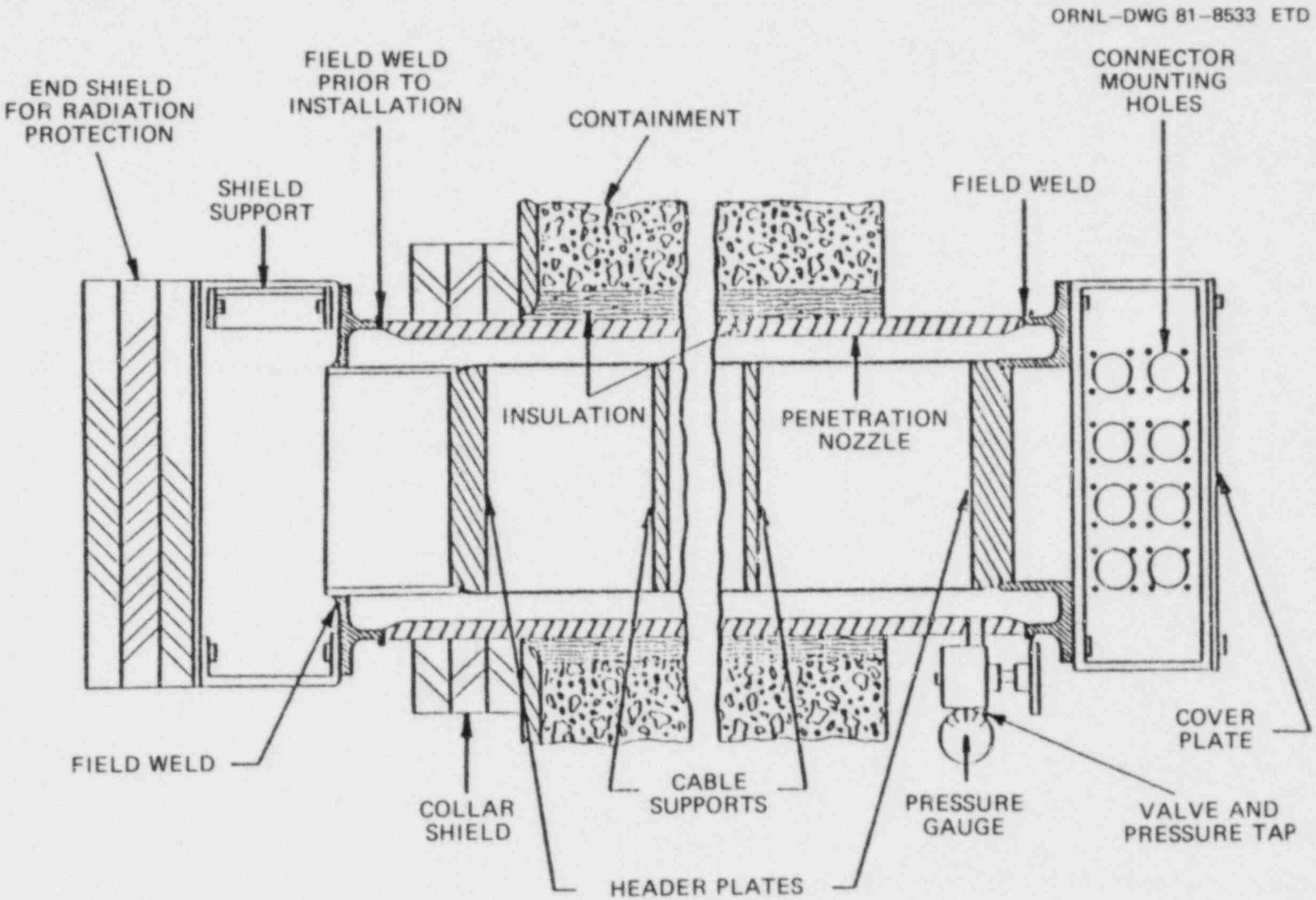

Fig. 9.22 Typical electrical penetration assembly canister.

ORNL-DWG 81-8534 ETD

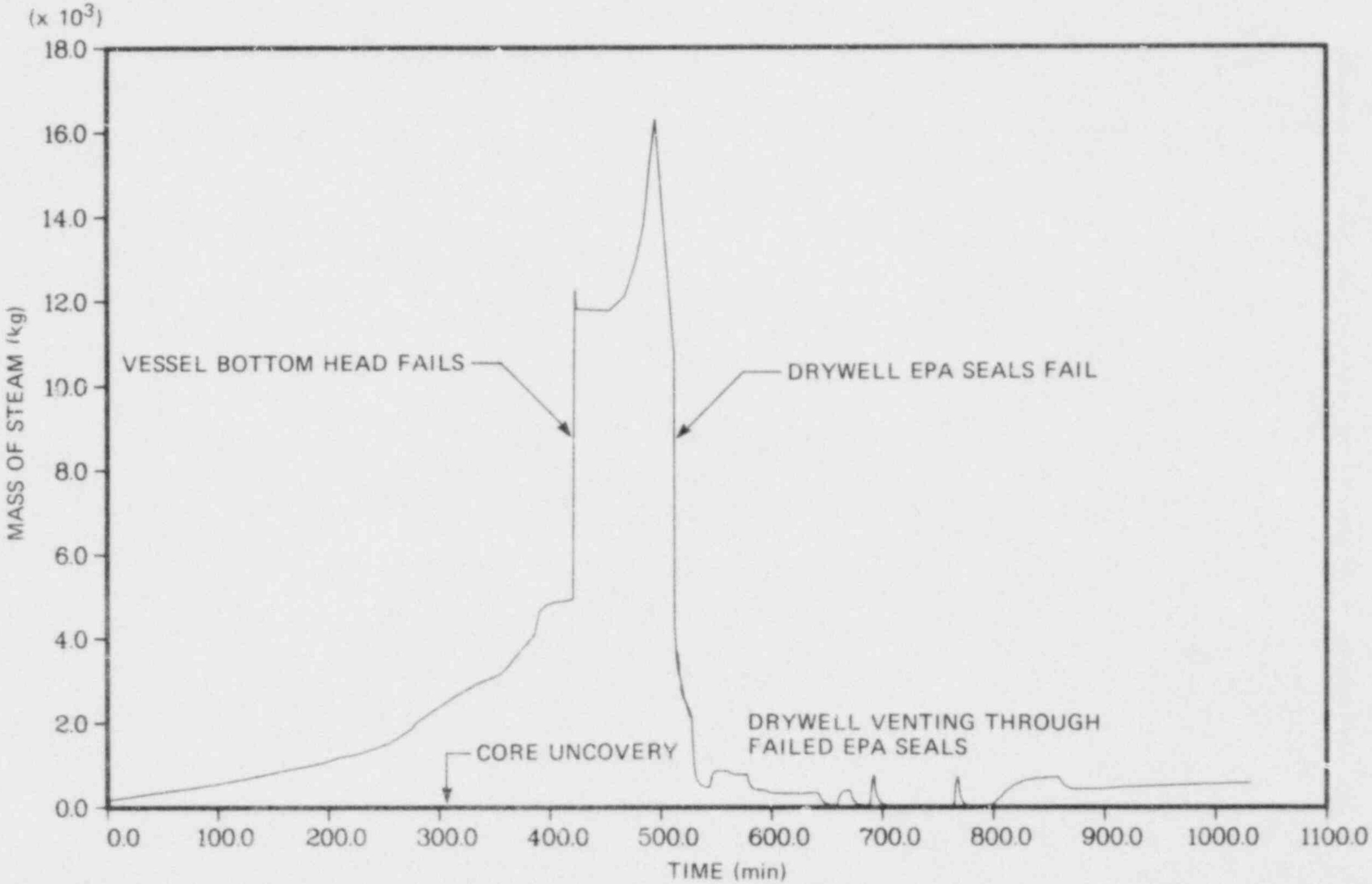

Fig. 9.23 Mass of steam in drywell.

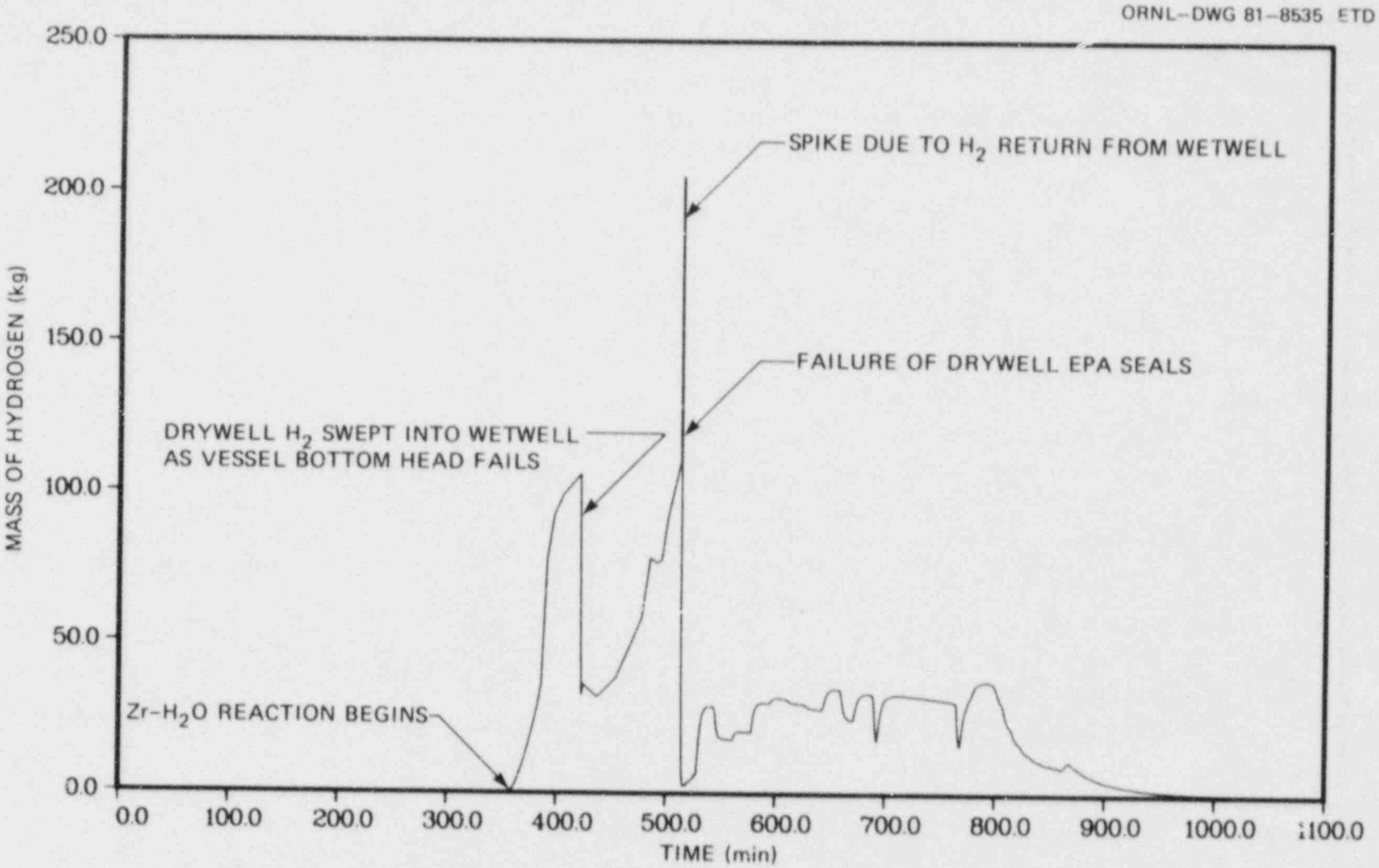

Fig. 9.24 Mass of hydrogen in drywell.

ORNL–DWG 81–8536 ETD

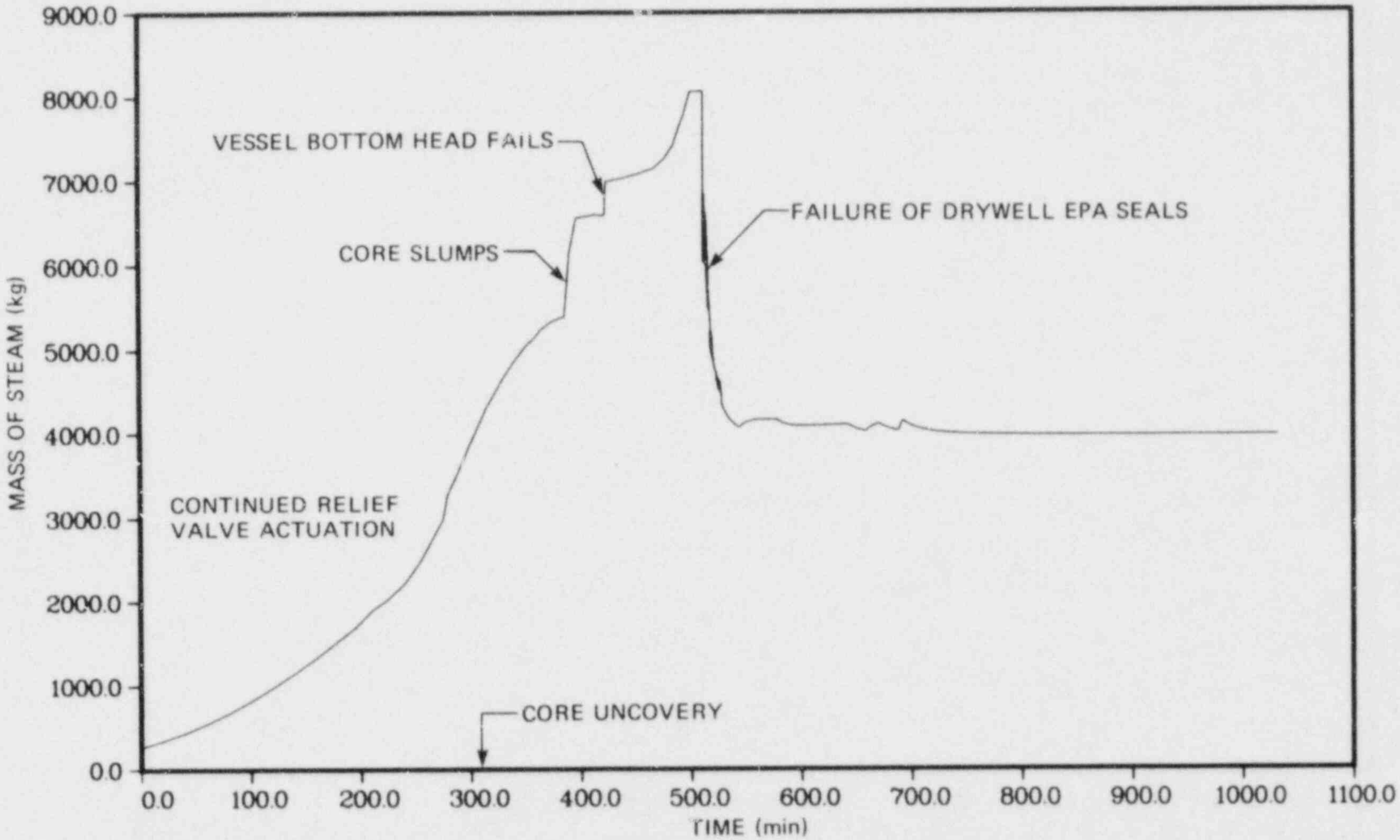

Fig. 9.25 Mass of steam in wetwell.

ORNL-DWG 81-8537 ETD

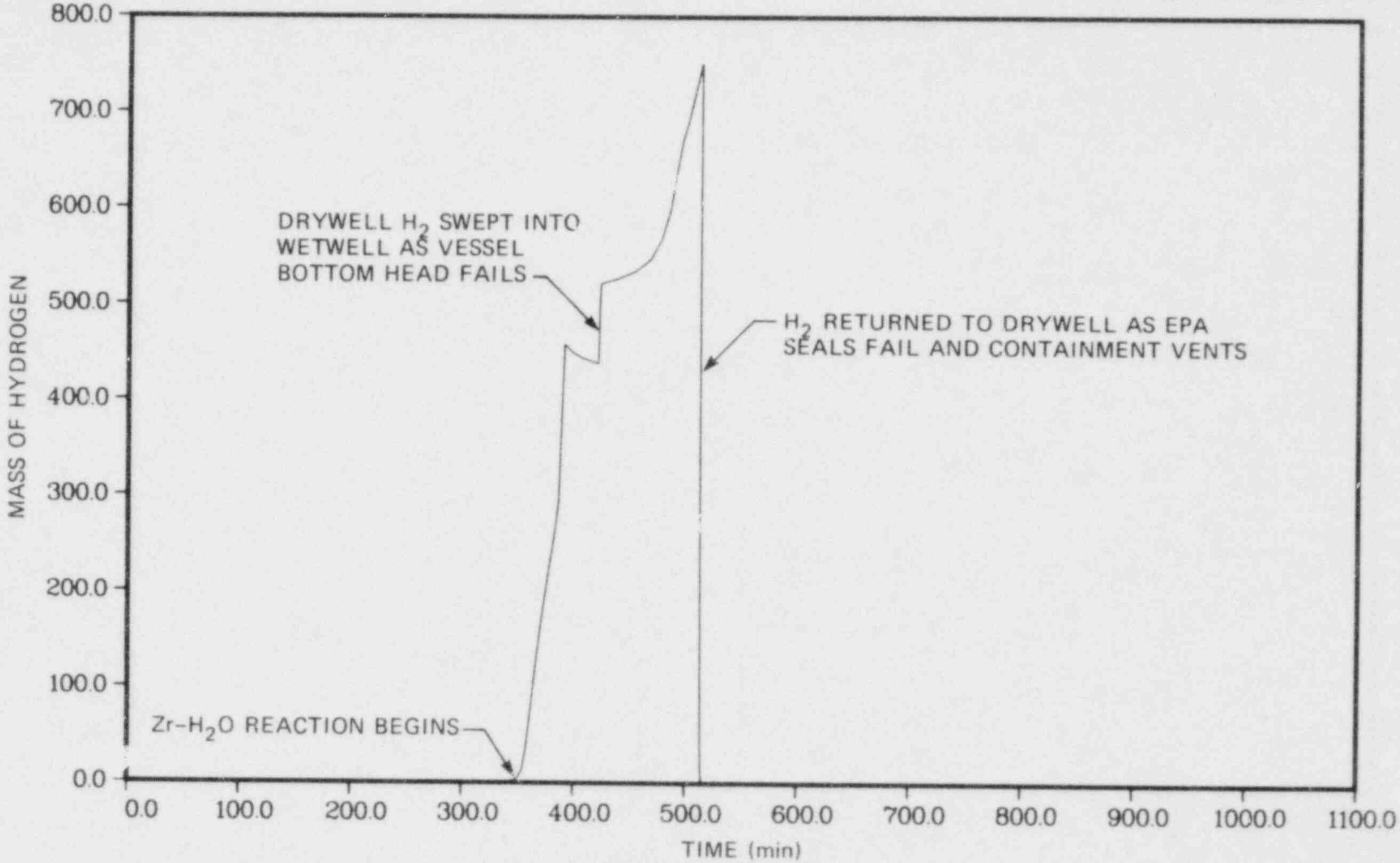

Fig. 9.26 Mass of hydrogen in wetwell.

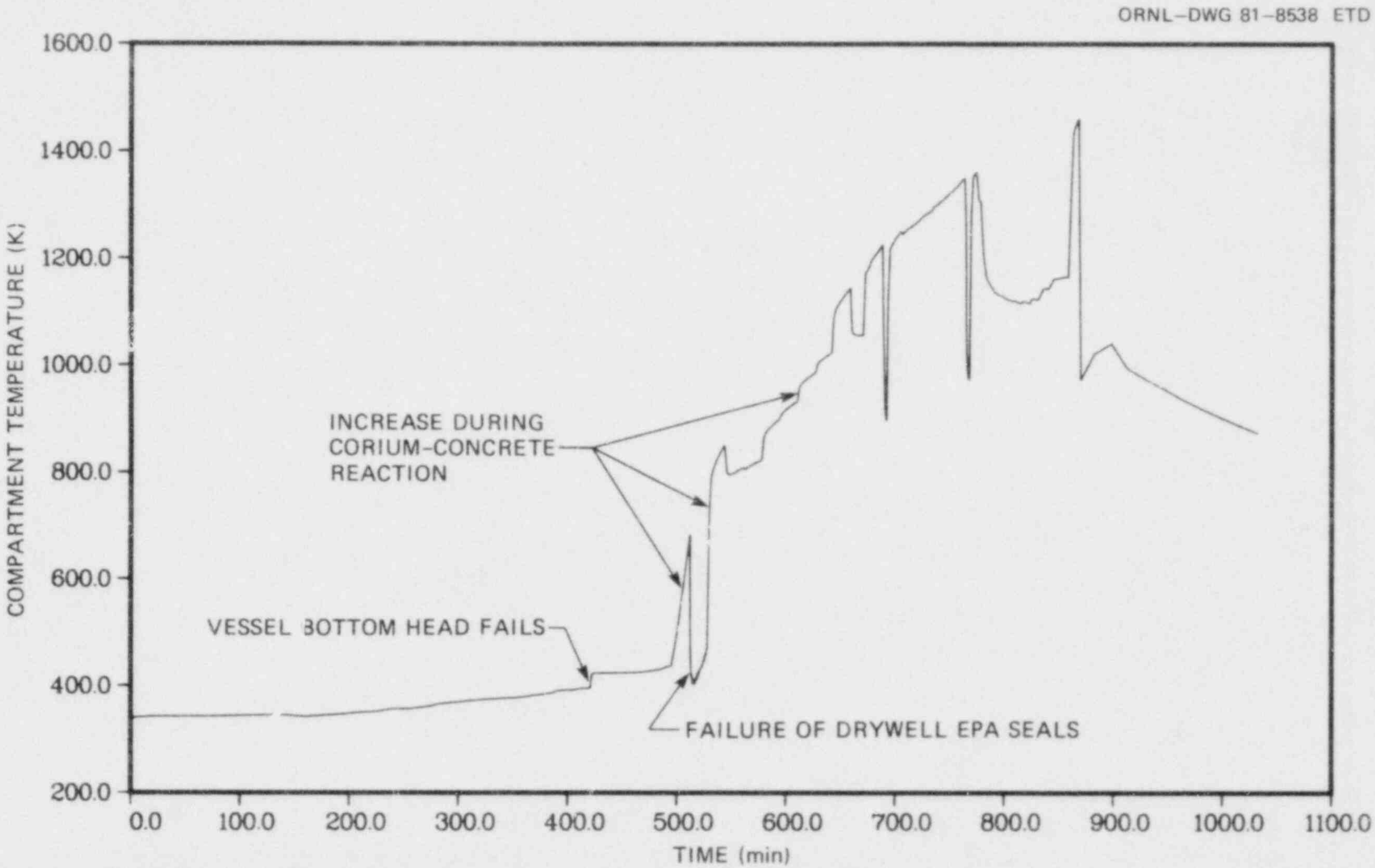

Fig. 9.27 Drywell temperature distribution (TB´).

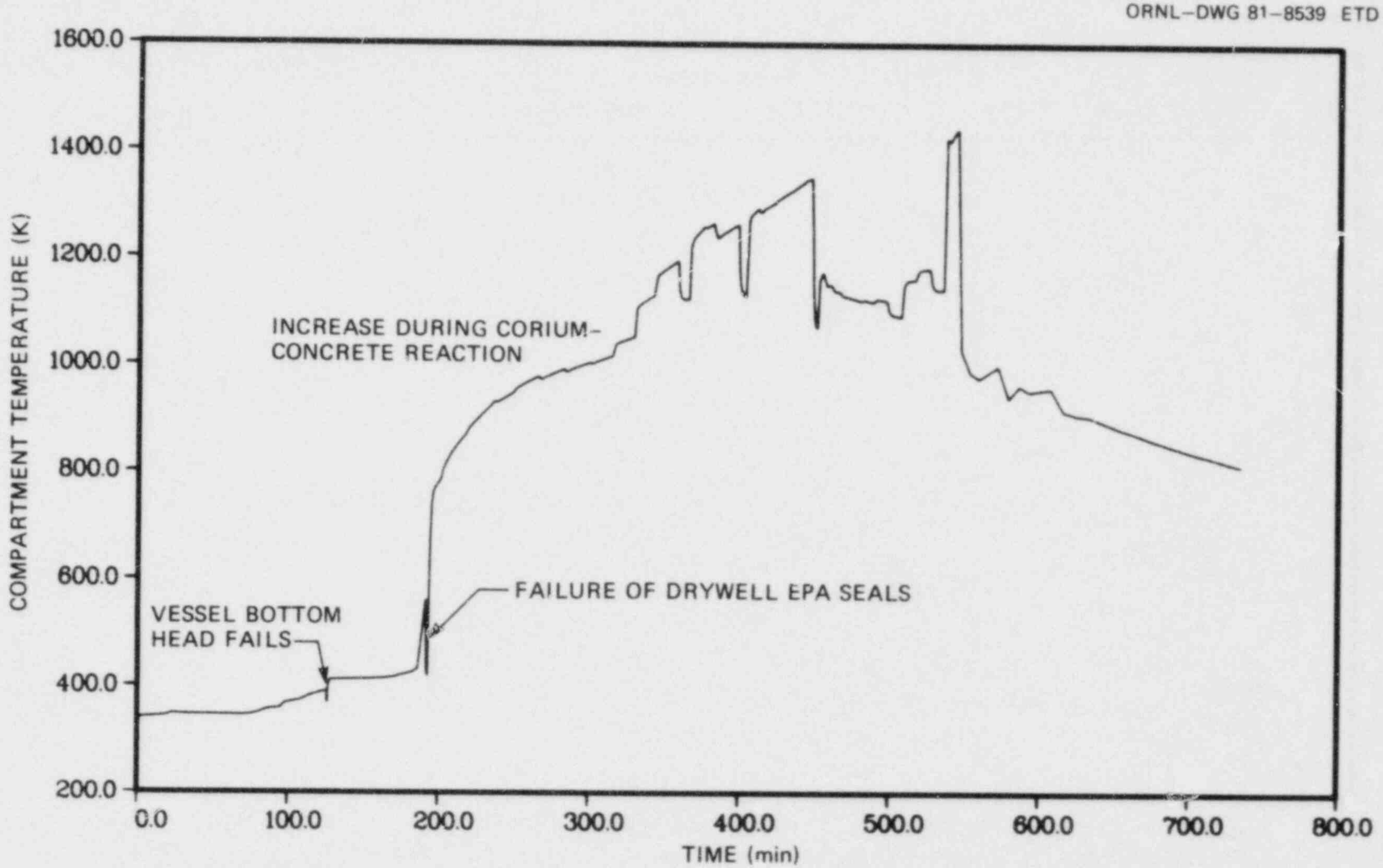

Fig. 9.28 Drywell temperature distribution (TUB′).

ORNL-DWG 81-8540 ETD

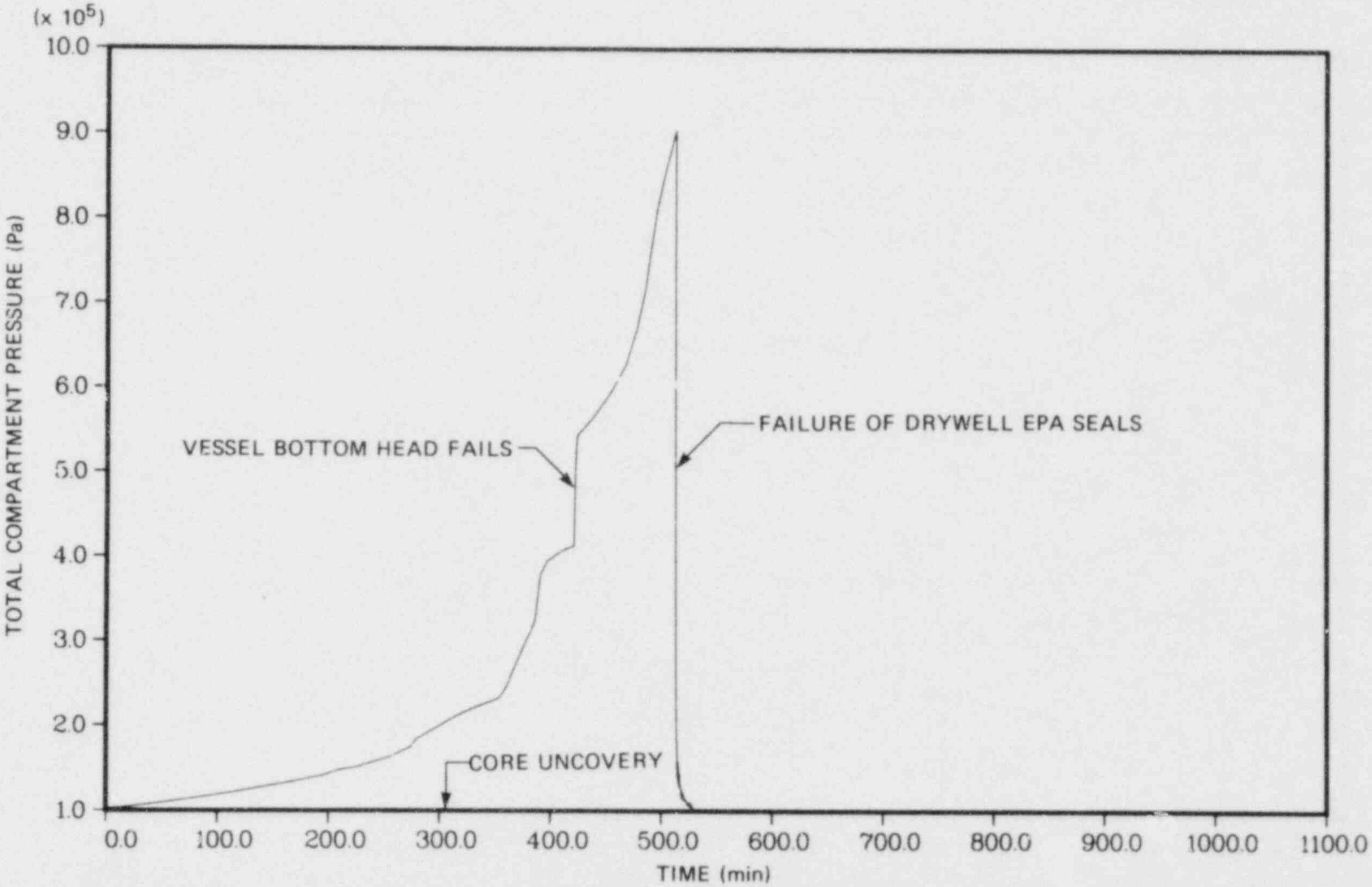

Fig. 9.29 Drywell total pressure (TB′).

ORNL-DWG 81-8541 ETD

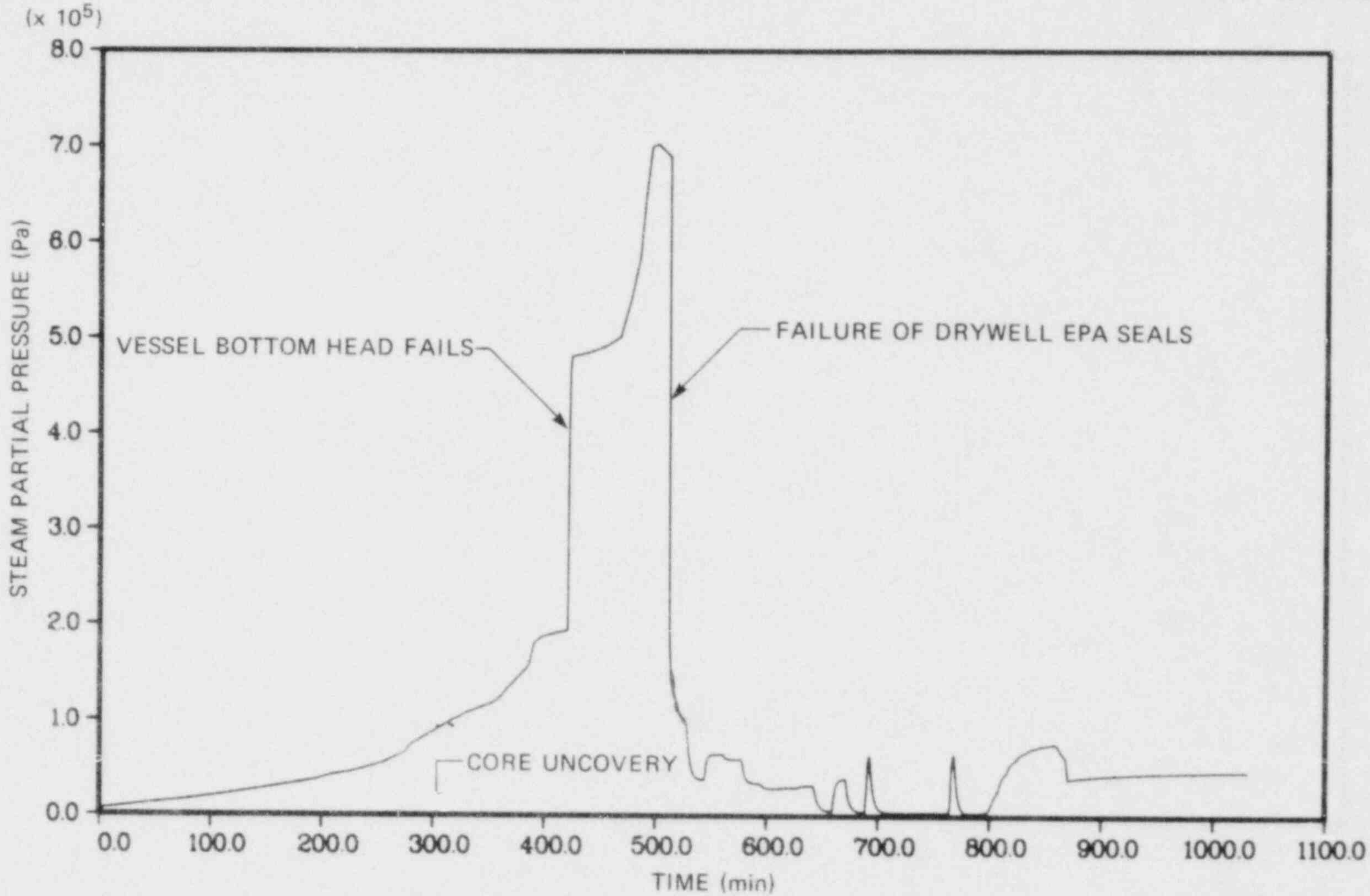

Fig. 9.30 Drywell steam partial pressure (TB′).

ORNL-DWG 81-8542 ETD

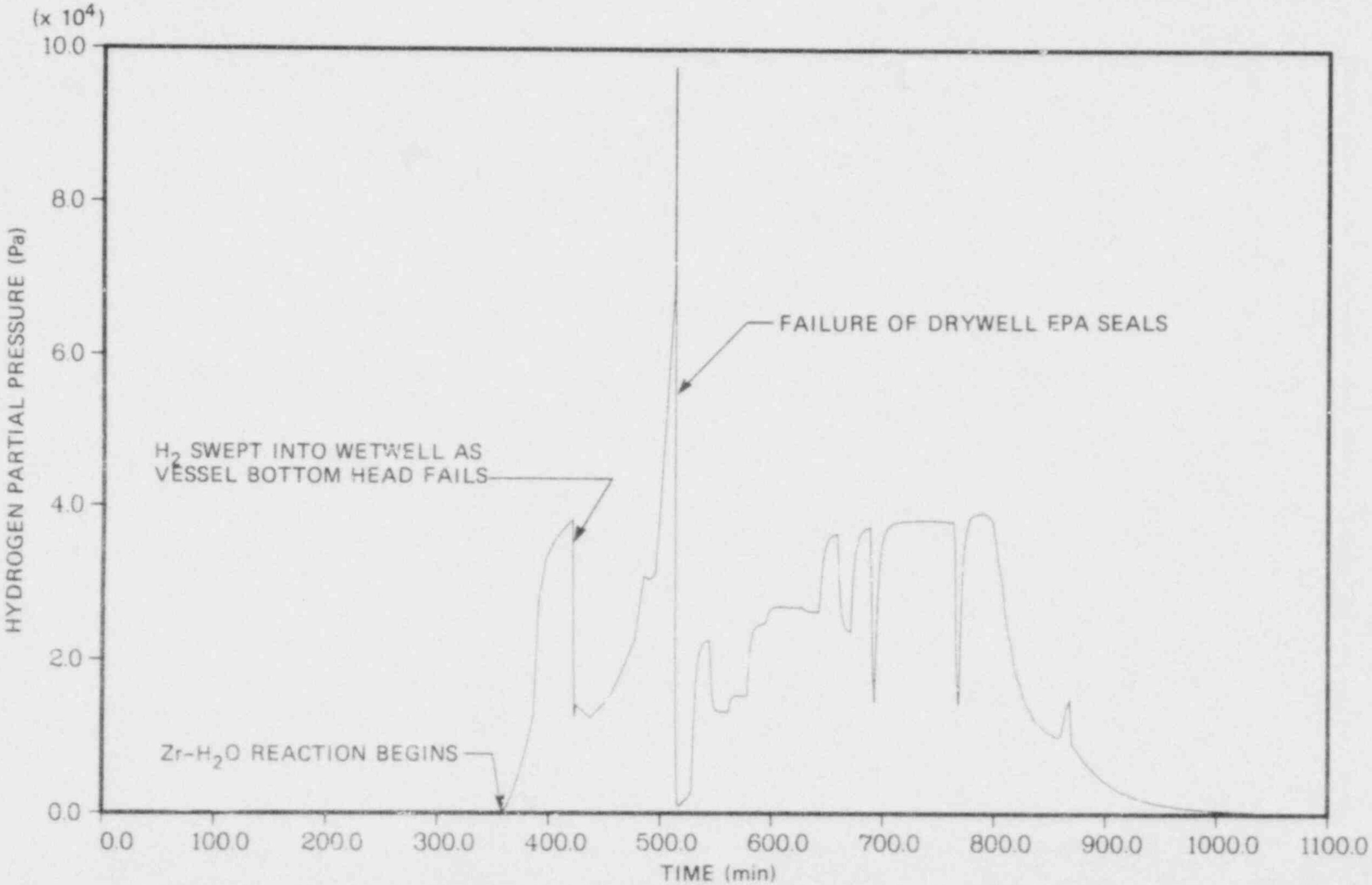

Fig. 9.31 Drywell hydrogen partial pressure (TB′).

ORNL-DWG 81-8543 ETD

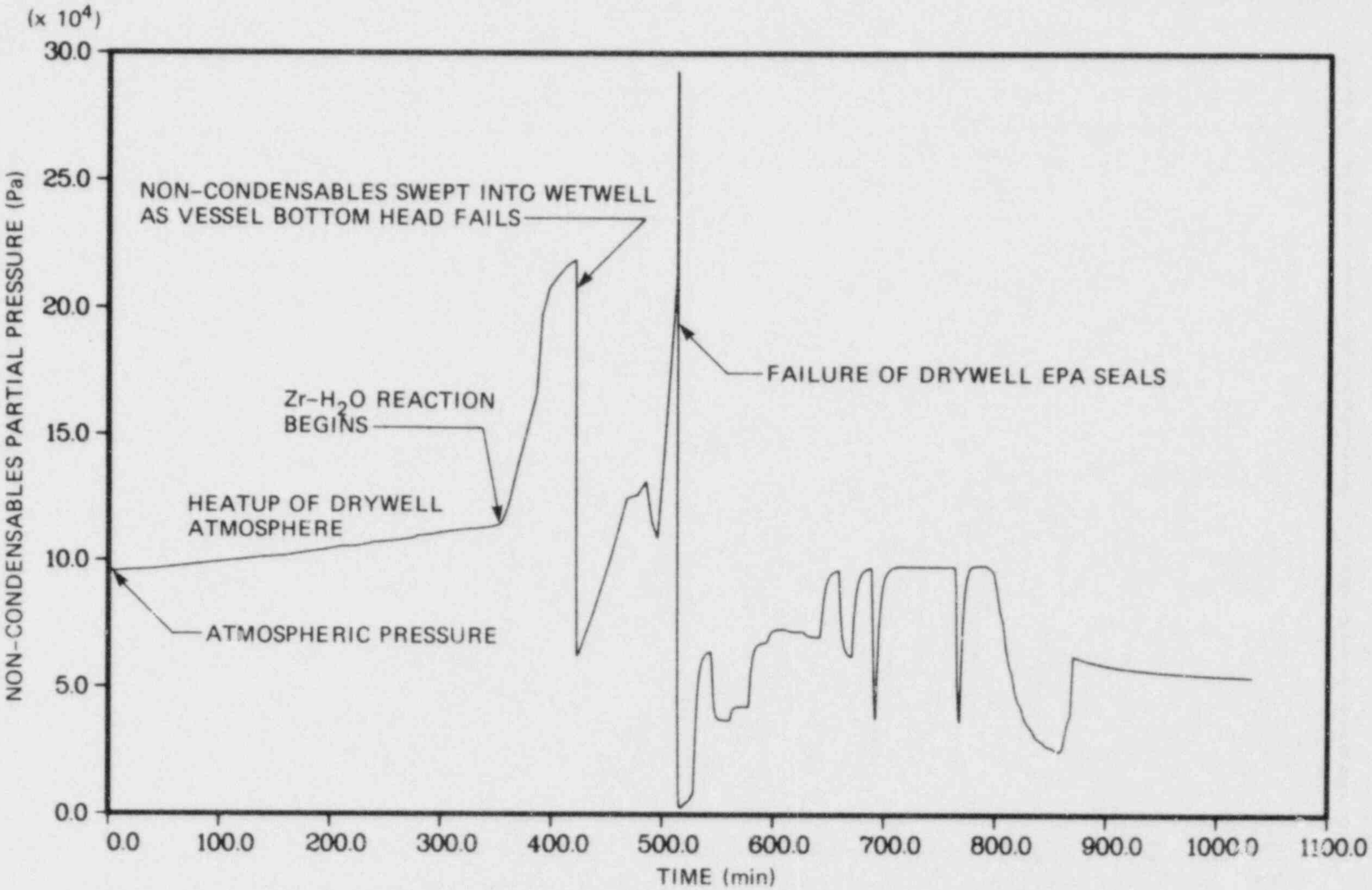

Fig. 9.32 Drywell non-condensables partial pressure (TB´).

ORNL-DWG 81-8544 ETD

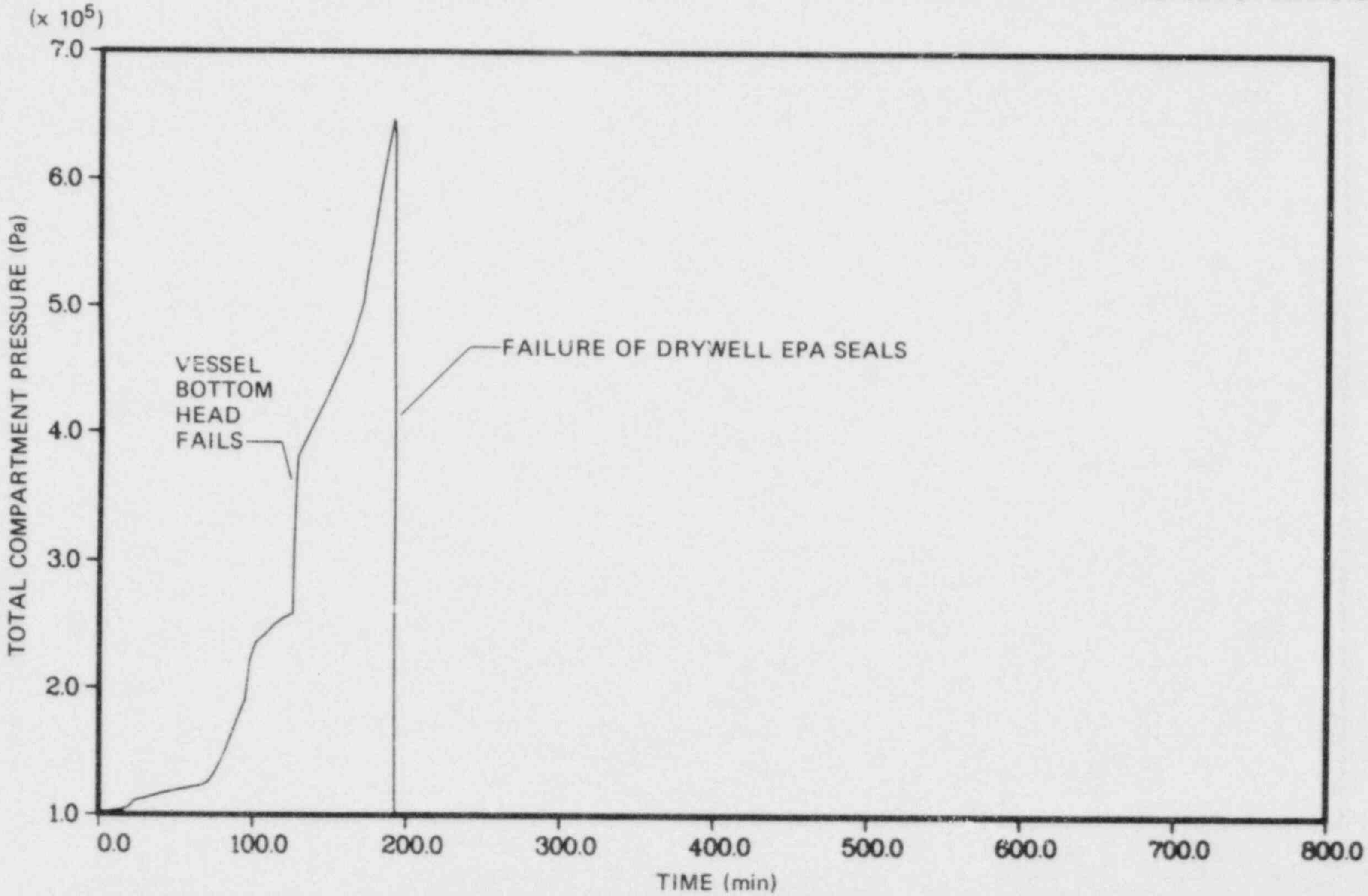

Fig. 9.33 Drywell total pressure (TUB′).

ORNL–DWG 81–8545 ETD

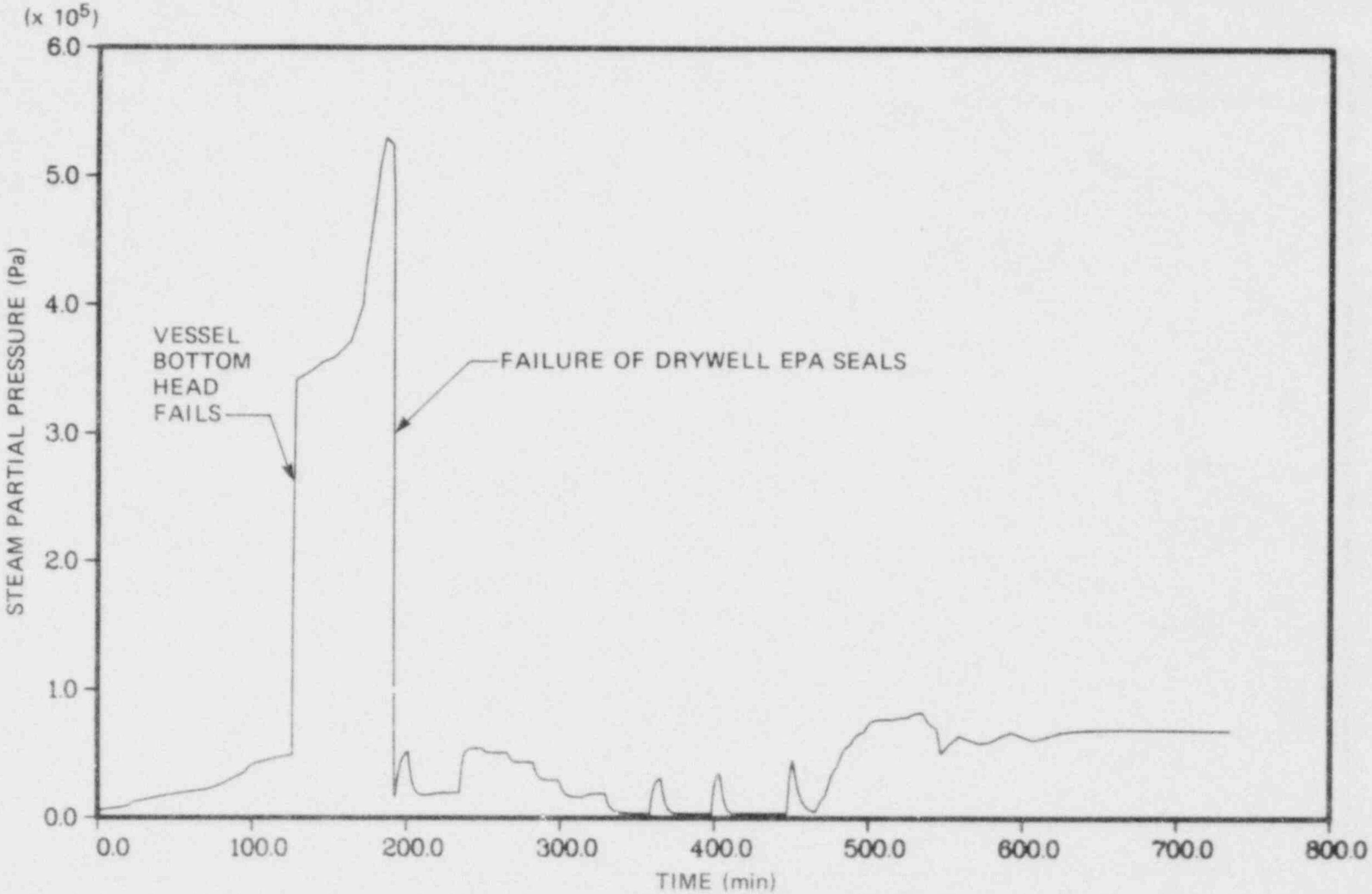

Fig. 9.34 Drywell steam partial pressure (TUB′).

ORNL-DWG 81-8546 ETD

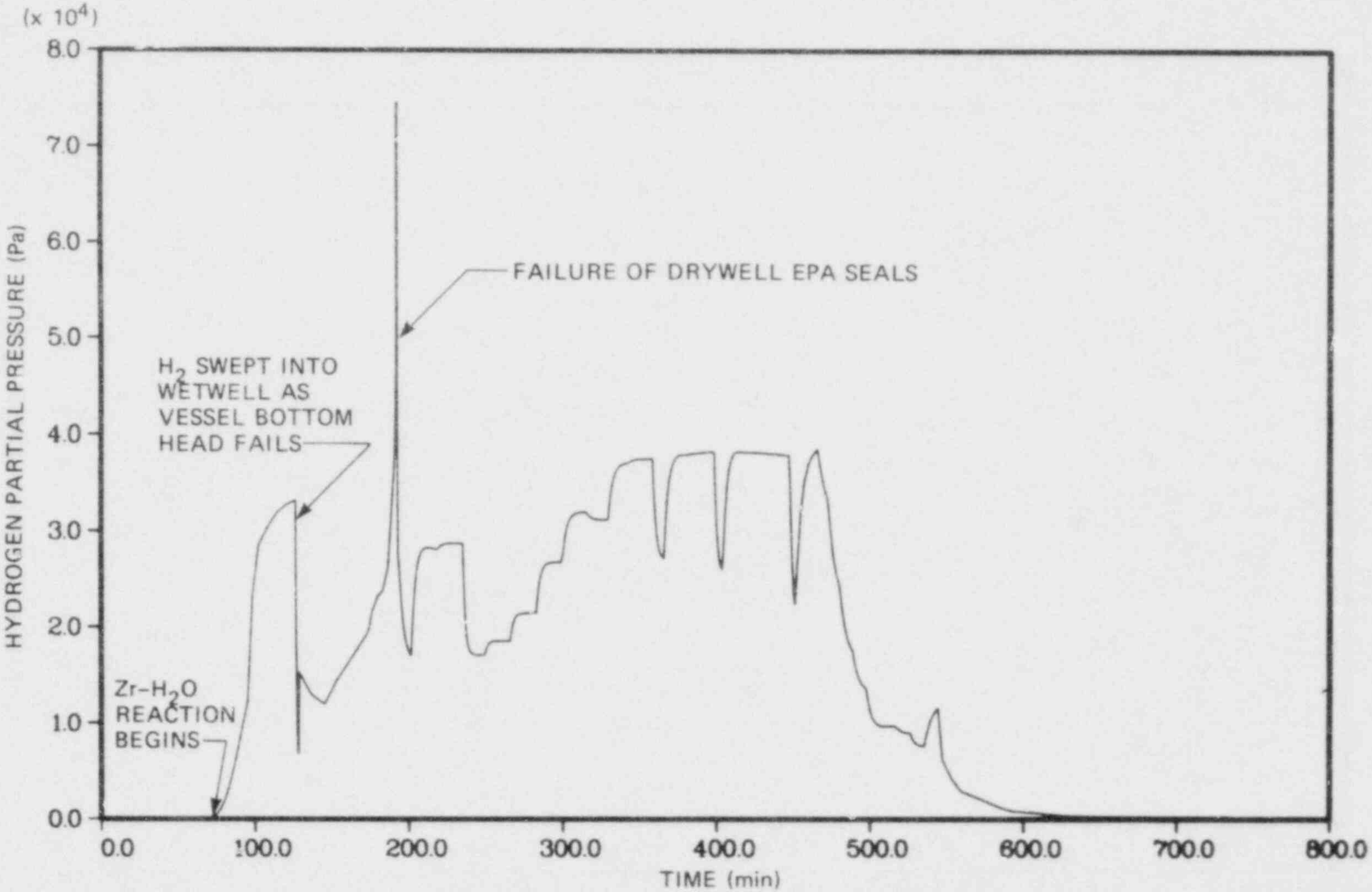

Fig. 9.35 Drywell hydrogen partial pressure (TUB´).

ORNL-DWG 81-8547 ETD

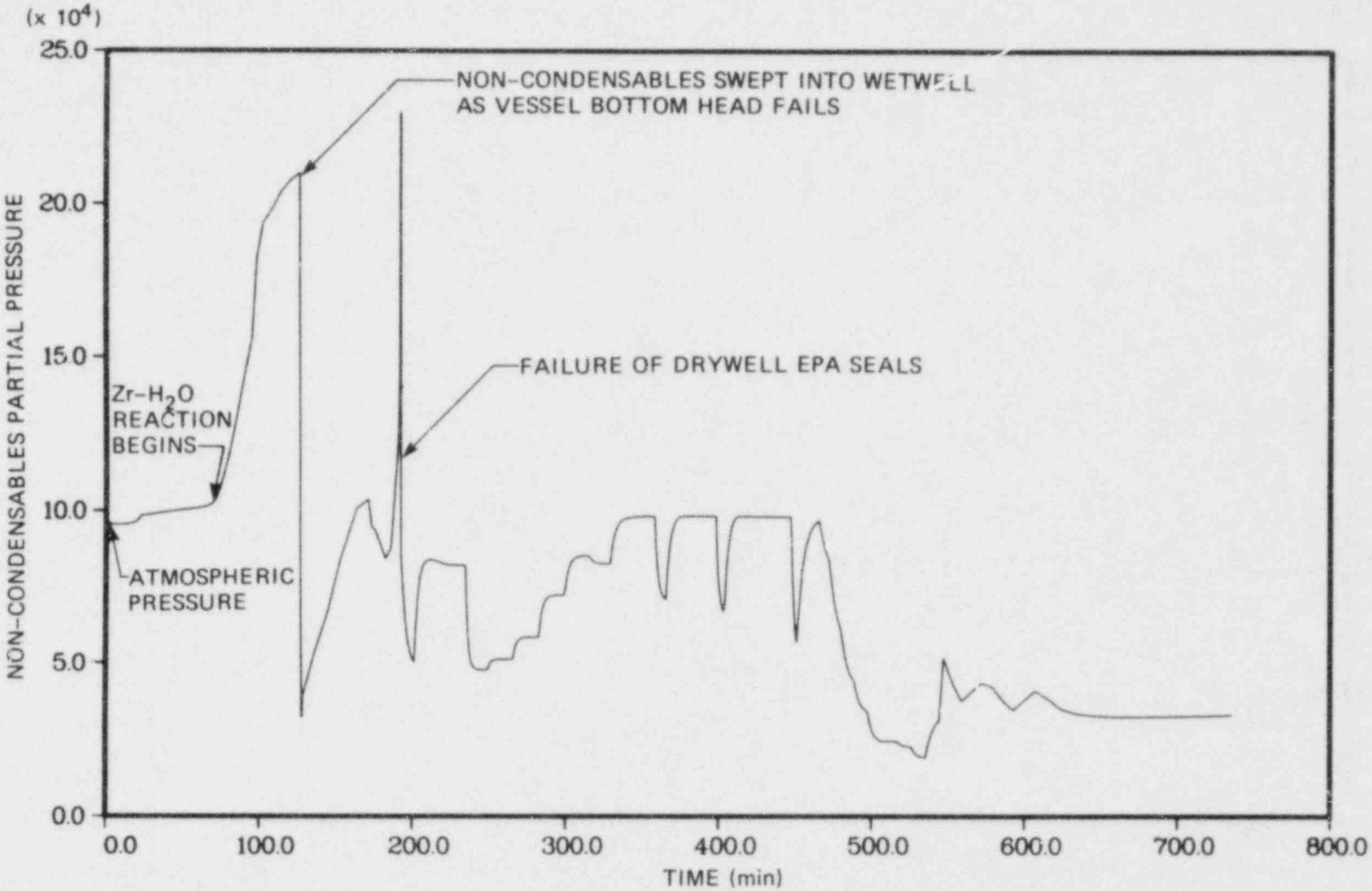

Fig. 9.36 Drywell non-condensables partial pressure (TUB´).

As the drywell ambient temperature exceeds 204°C (400°F), the EPA elastomeric sealing materials start to deteriorate and venting of the drywell begins. When the ambient temperature has exceeded 260°C (500°F), the EPA elastomeric sealing materials have decomposed to such an extent that the EPA seals are blown out of the containment wall by the elevated pressure within the drywell. This greatly increases the containment leakage (Fig. 9.10).

9.5.2 Wetwell response

The condensation of the SRV steam discharge within the pressure suppression pool is generally accomplished without undue pressure stresses on the containment wall. If, however, conditions are such that complete condensation of the steam discharge does not occur, pressure loads significantly in excess of design limits can result, leading eventually to wetwell rupture.

Relief valve steam discharge into the pressure suppression pool is accompanied by pressure oscillations of varying characteristics which are functions of the steam mass flux and local pool water temperatures as well as the type of sparger device installed at the discharge line terminus.

It has been observed that condensation instability can occur when a submerged pipe vent discharges steam at flow rates higher than critical (sonic discharge) with a sufficiently high ambient pool temperature- the so-called "Wurgassen effect". The threshold of instability is characterized by an increase in the amplitude of pressure oscillations which normally accompany condensation at supercritical flow rates.

At present, ramshead-type sparger devices are installed on the relief valve tail pipe discharge lines at the Browns Ferry Plant. The ramshead device consists of two 90° elbows welded back-to-back to form a modified T-junction, and provides an improvement in condensation performance over that of a straight vertical pipe. The horizontal discharge from the ramshead allows the rising convection currents and induced secondary flows to circulate cooler water around the steam plumes rather than stagnate against a downward flow of steam. However, small scale tests have shown that it also has the potential to produce unstable condensation and concomitant large pressure stresses when sufficiently high pool water temperature is reached during supercritical (sonic) discharge. For this reason, certain limitations on plant operation are established in the Browns Ferry Technical Specifications to preclude any possible steam discharge through the ramshead devices at pool temperatures greater than 77°C (170°F).

In the near future, the ramshead devices at the Browns Ferry Plant will be replaced by T-quencher sparger devices similar to that shown in Fig. D-2 of Appendix D. Tests have shown that the T-quencher spargers produce lower loads during the initial air-clearing upon relief valve actuation and permit smooth condensation of the steam discharge at pool temperatures up to 93°C (200°F). For this study, it was assumed that the SRV discharge lines terminate in T-quencher spargers.

Although the use of T-quenchers considerably improves the steam condensation characteristics related to SRV steam discharges, the constricted flows introduced by the T-quencher have caused insufficient thermal mixing in the suppression pool. The resultant thermal stratification in the pool can produce much higher local water temperature in the vicinity of the T-quenchers although the average pool temperature remains quite low.

The problem of thermal stratification has been found to be significant for the sequences TUB′ and TUPB′, in which the HPCI and RCIC systems are assumed inoperative. Without injection capability, core uncovery and subsequent degradation occur early in the Station Blackout when the level of decay heat is relatively high so that a large amount of superheated steam and noncondensibles is discharged from the T-quenchers into the suppression pool in a short period of time. The average suppression pool temperature is plotted as a function of time for sequences TB' and TUB' in Figs. 9.37 and 9.38 respectively.

In sequence TUB′, it is realistically assumed that without operator action, reactor vessel pressure control over the long term would be by repeated cycling of the same relief valve.* Because of the high steam mass flux into the suppression pool bay in which the discharging T-quencher is located, significant thermal stratification would be expected. MARCH computations show that the difference between the local and average suppression pool temperatures can be estimated to increase from about 5°C at the beginning of the transient to about 40°C 100 minutes later. This means that the suppression pool would lose its condensation effectiveness; the resulting pressure loads from the SRV discharge of steam and noncondensibles would rapidly increase, leading to a possible rupture of the wetwell which could occur before the overtemperature-induced failure of the drywell.

The drywell pressure signatures for the sequences TB′ and TUB′ are shown in Figs. 9.29 through 9.36. The general trends are again very similar, except for the timing of events. The pressure peaks for the sequence TB′ are generally higher than those for the sequence TUB′; this is attributed to the presence of a larger water inventory as a result of the four-hour period of HPCI injection during sequence TB′.

The wetwell temperature signatures are given in Figs. 9.37 and 9.38, and the average pressure distributions are given in Figs. 9.39 through 9.46 for sequences TB′ and TUB′. The plotted pressures do not include the effect of pressure waves resulting from condensation instabilities or the vapor jet plumes emanating from the T-quencher spargers.

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*Although four of the thirteen SRVs share the lowest nominal setpoint of 7.72 MPa (1105 psig), practical considerations dictate that there would be one lowest-set valve.

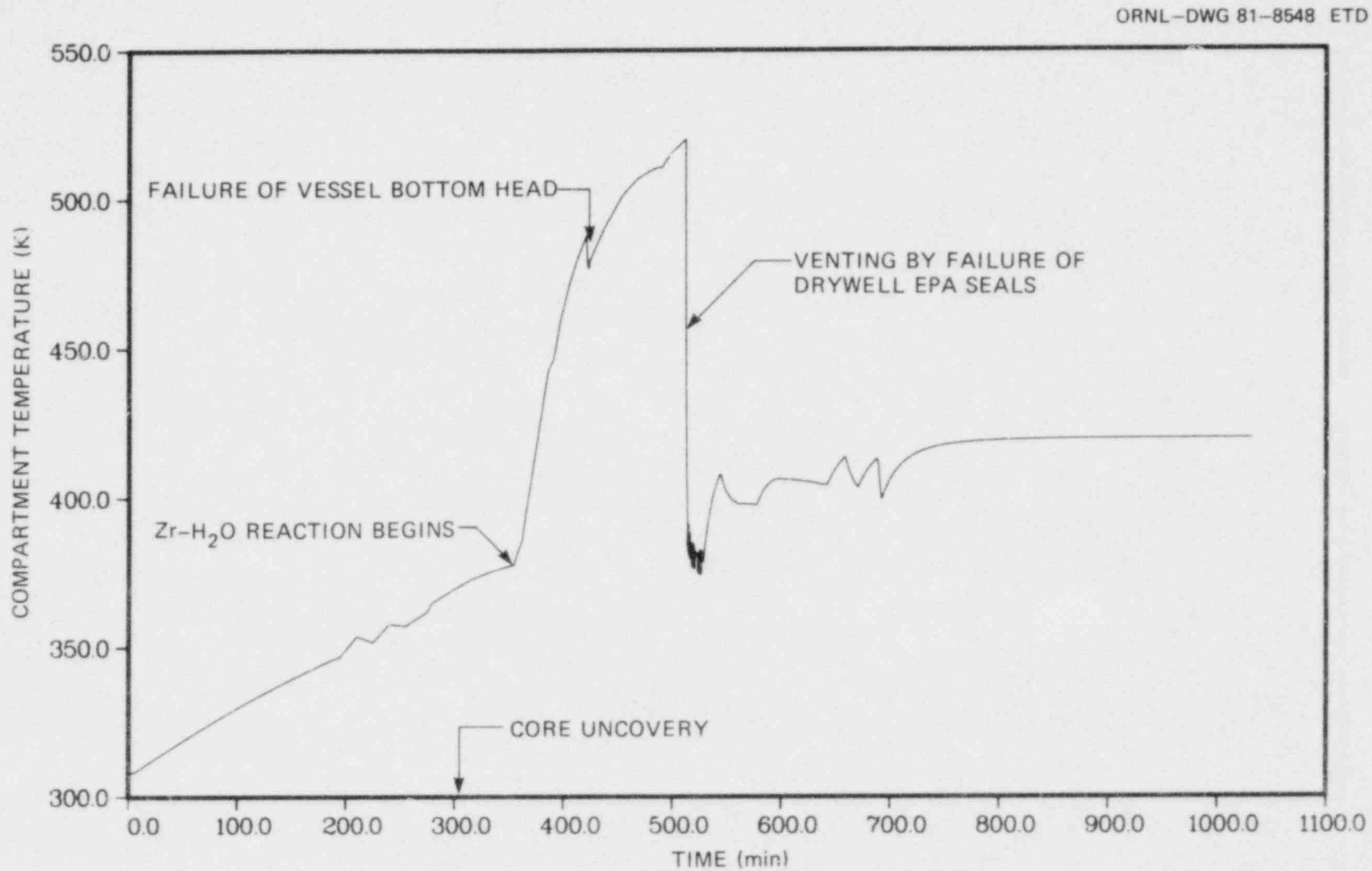

Fig. 9.37 Wetwell temperature distribution (TB′).

ORNL–DWG 81–8549 ETD

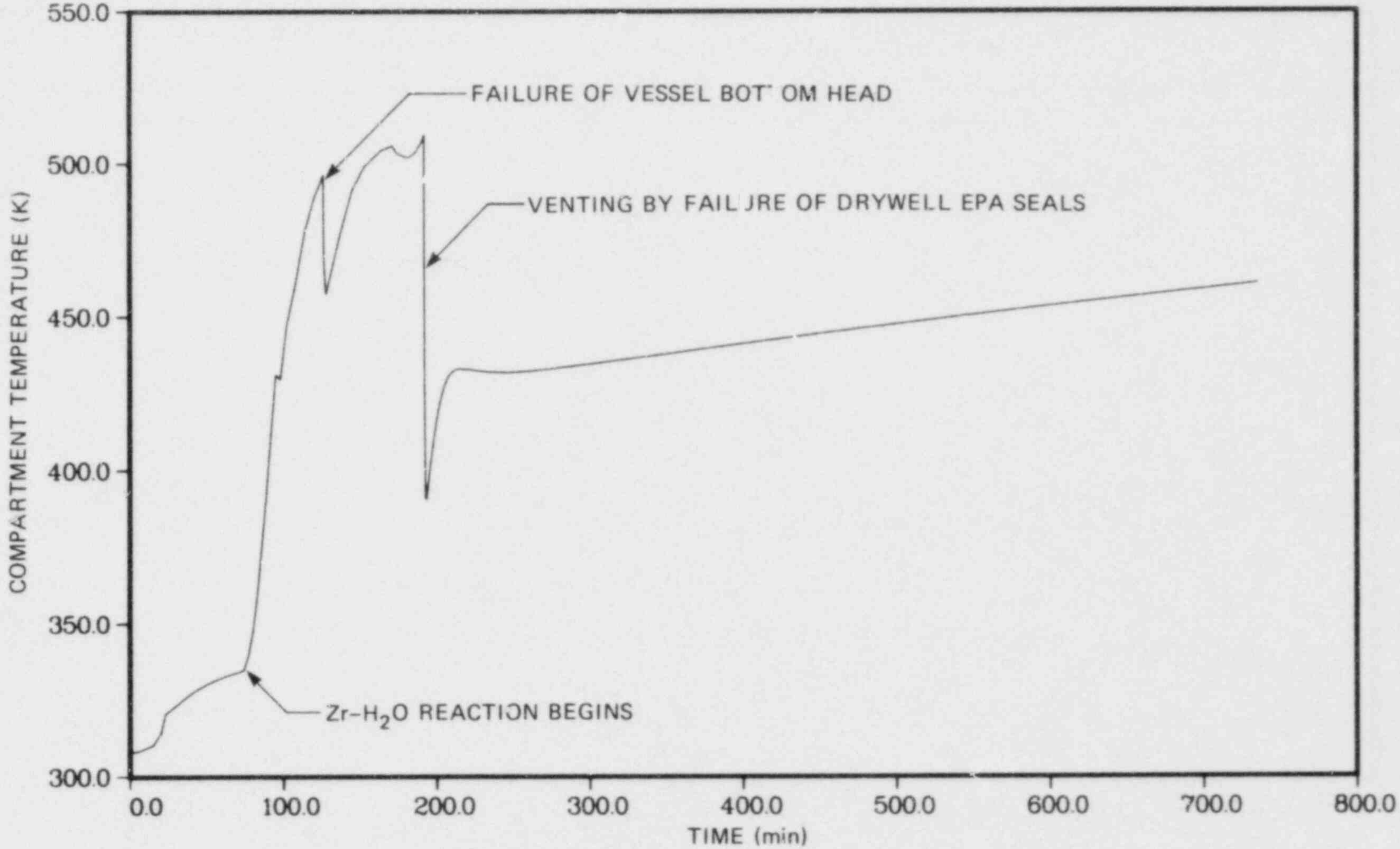

Fig. 9.38 Wetwell temperature distribution (TUB´).

ORNL-DWG 81-8550 ETD

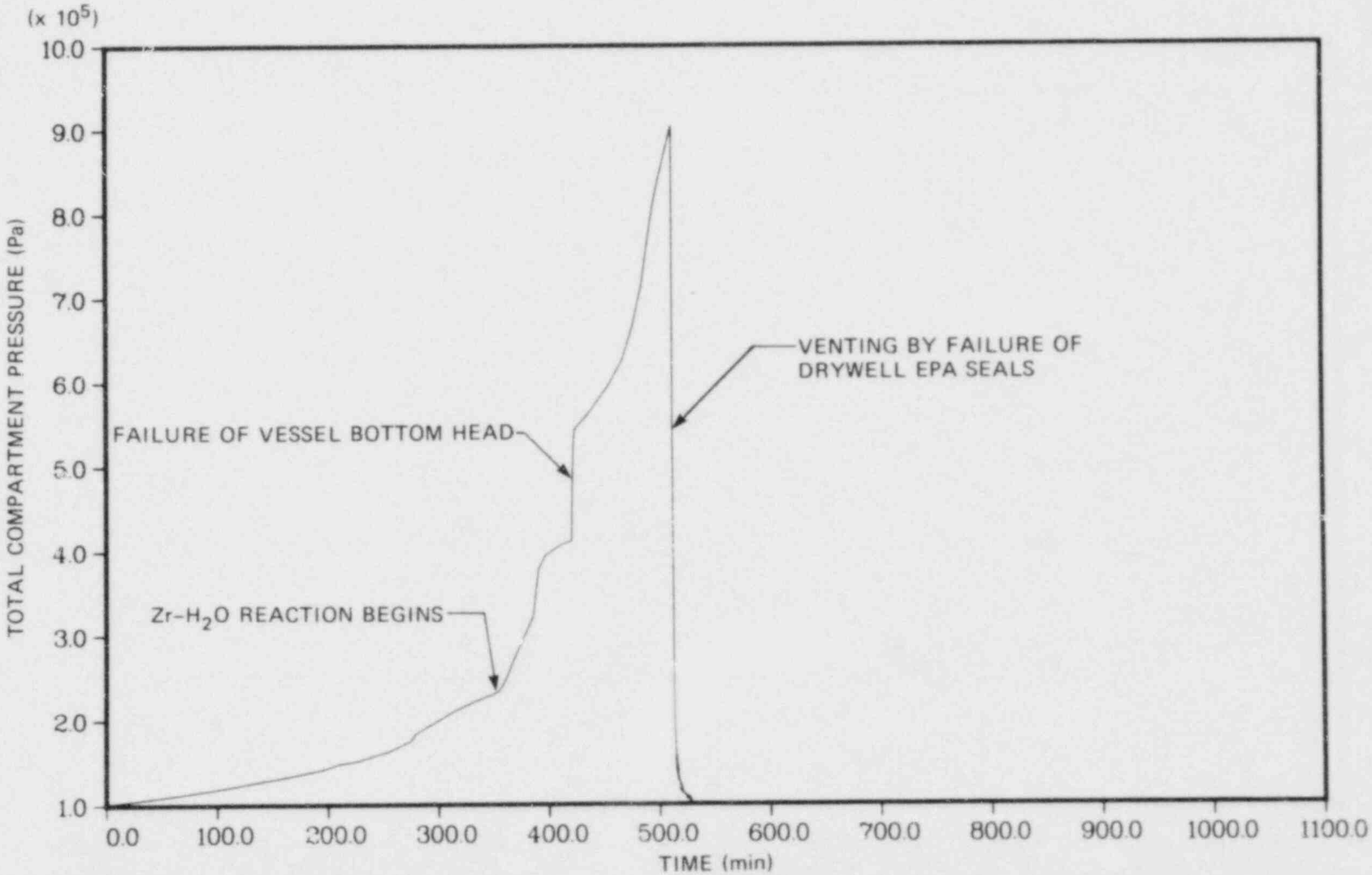

Fig. 9.39 Wetwell total pressure (TB′).

ORNL-DWG 81-8551 ETD

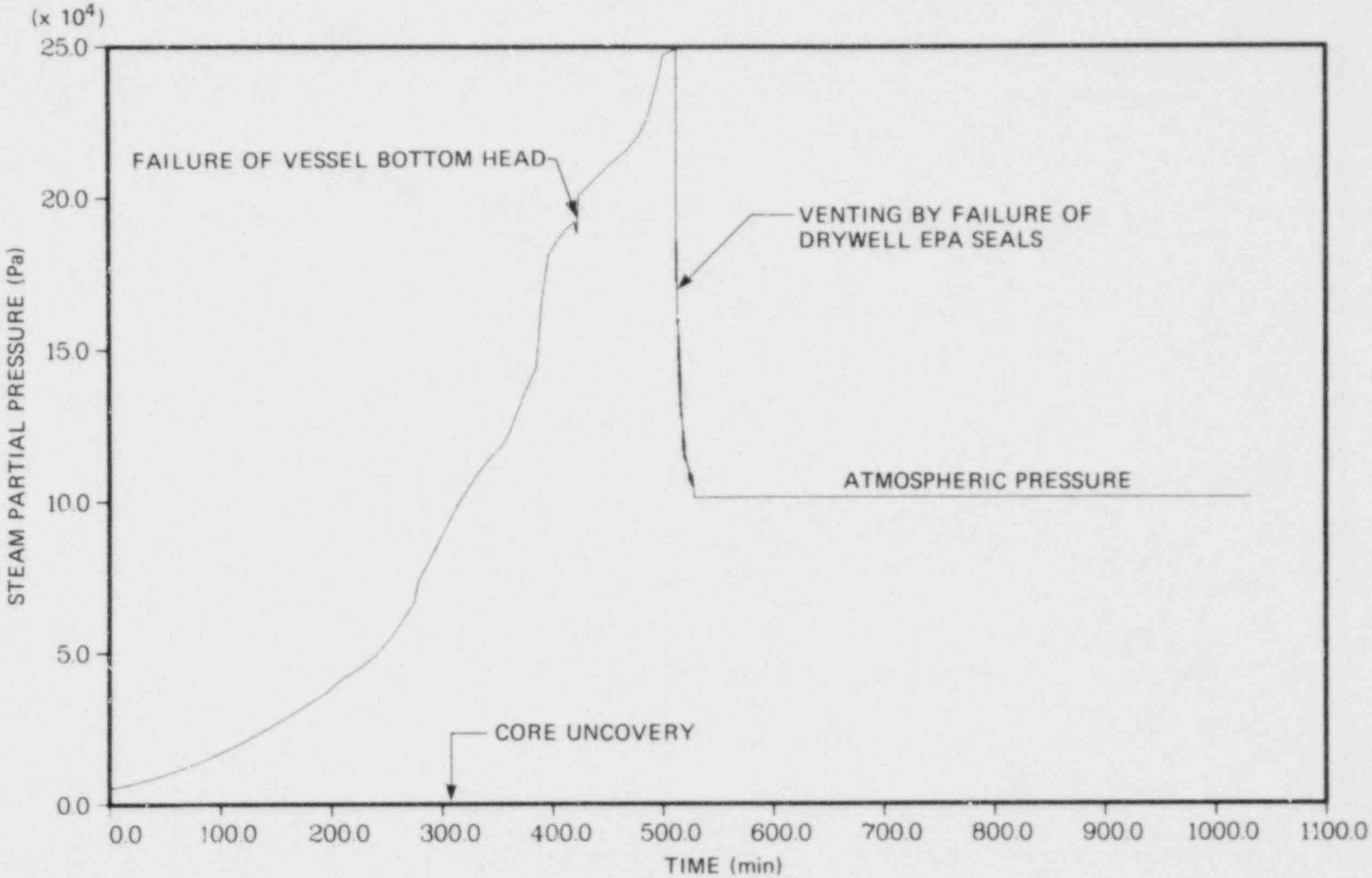

Fig. 9.40 Wetwell steam partial pressure (TB′).

ORNL-DWG 81-8552 ETD

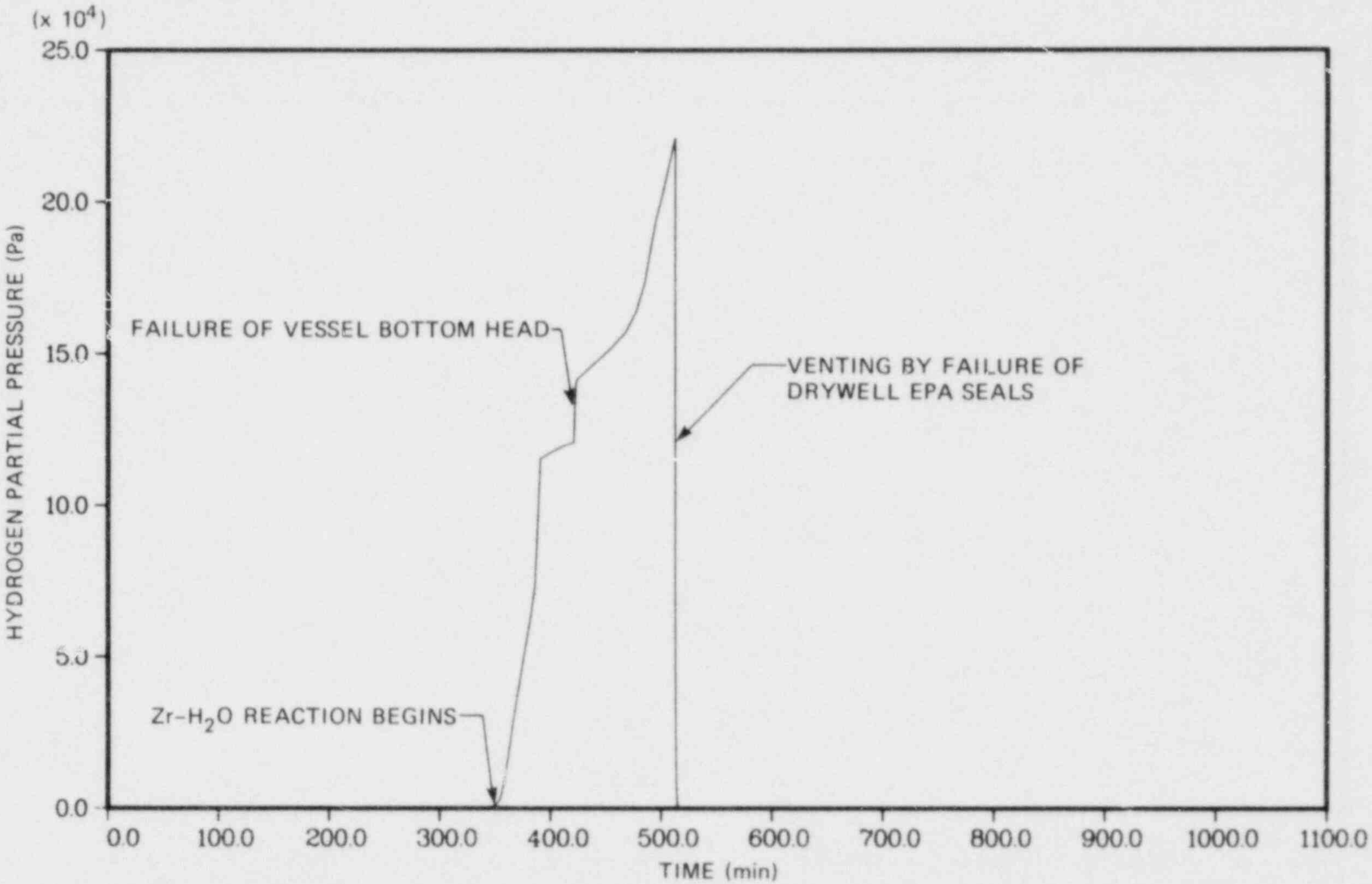

Fig. 9.41 Wetwell hydrogen partial pressure (TB´).

ORNL-DWG 81-8553 ETD

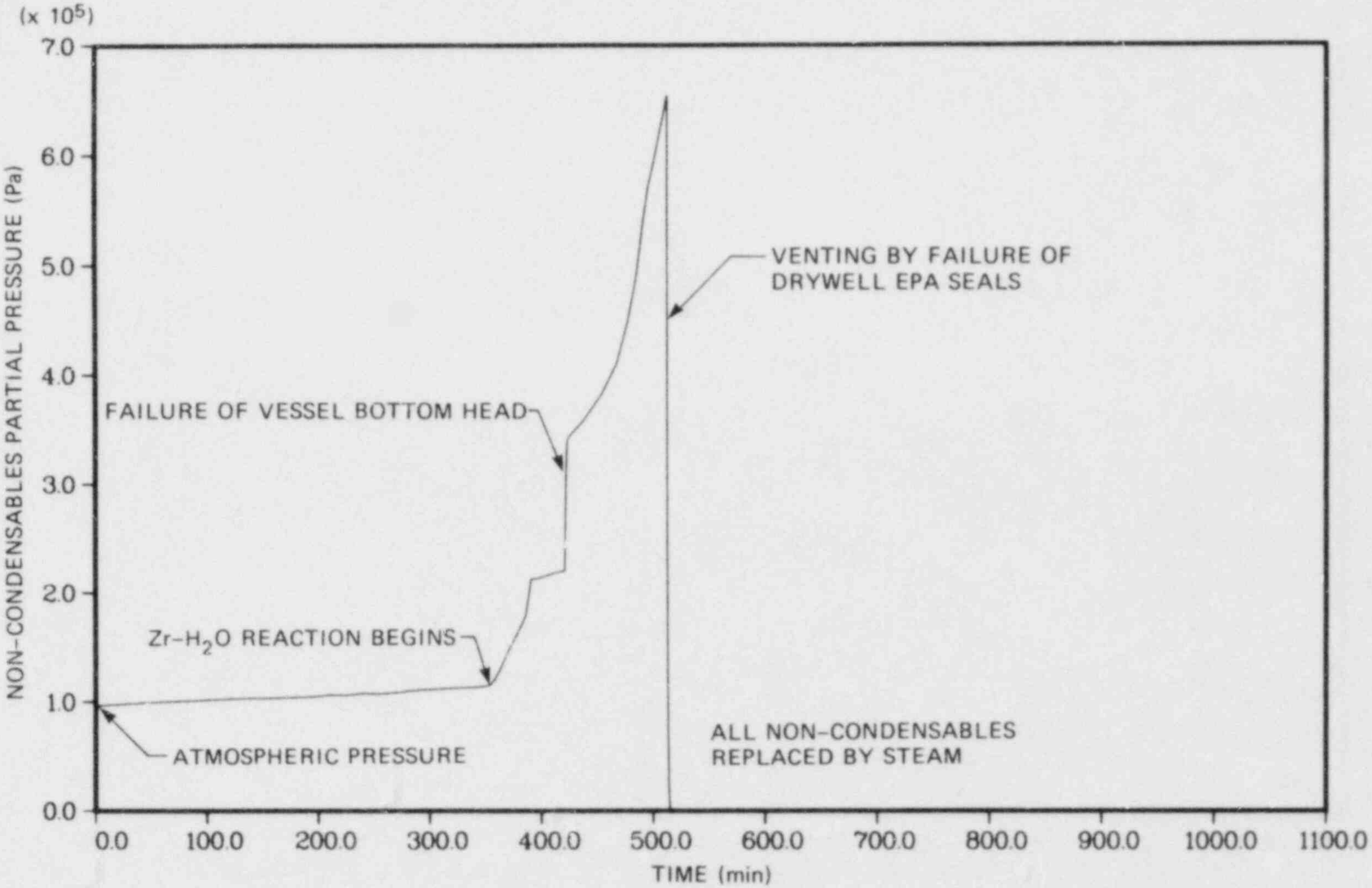

Fig. 9.42 Wetwell non-condensables partial pressure (TB′).

ORNL-DWG 81-8554 ETD

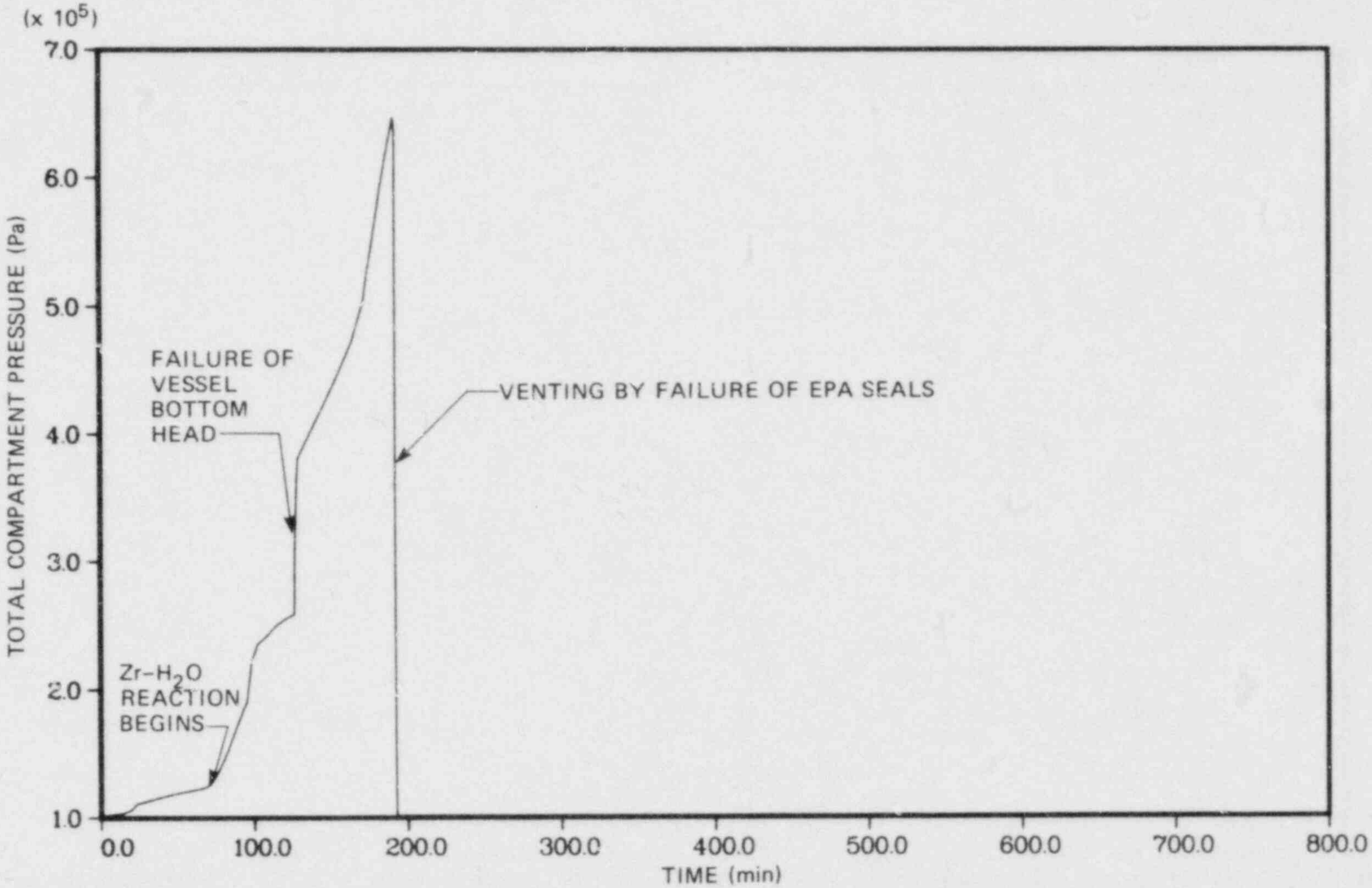

Fig. 9.43 Wetwell total pressure (TUB′).

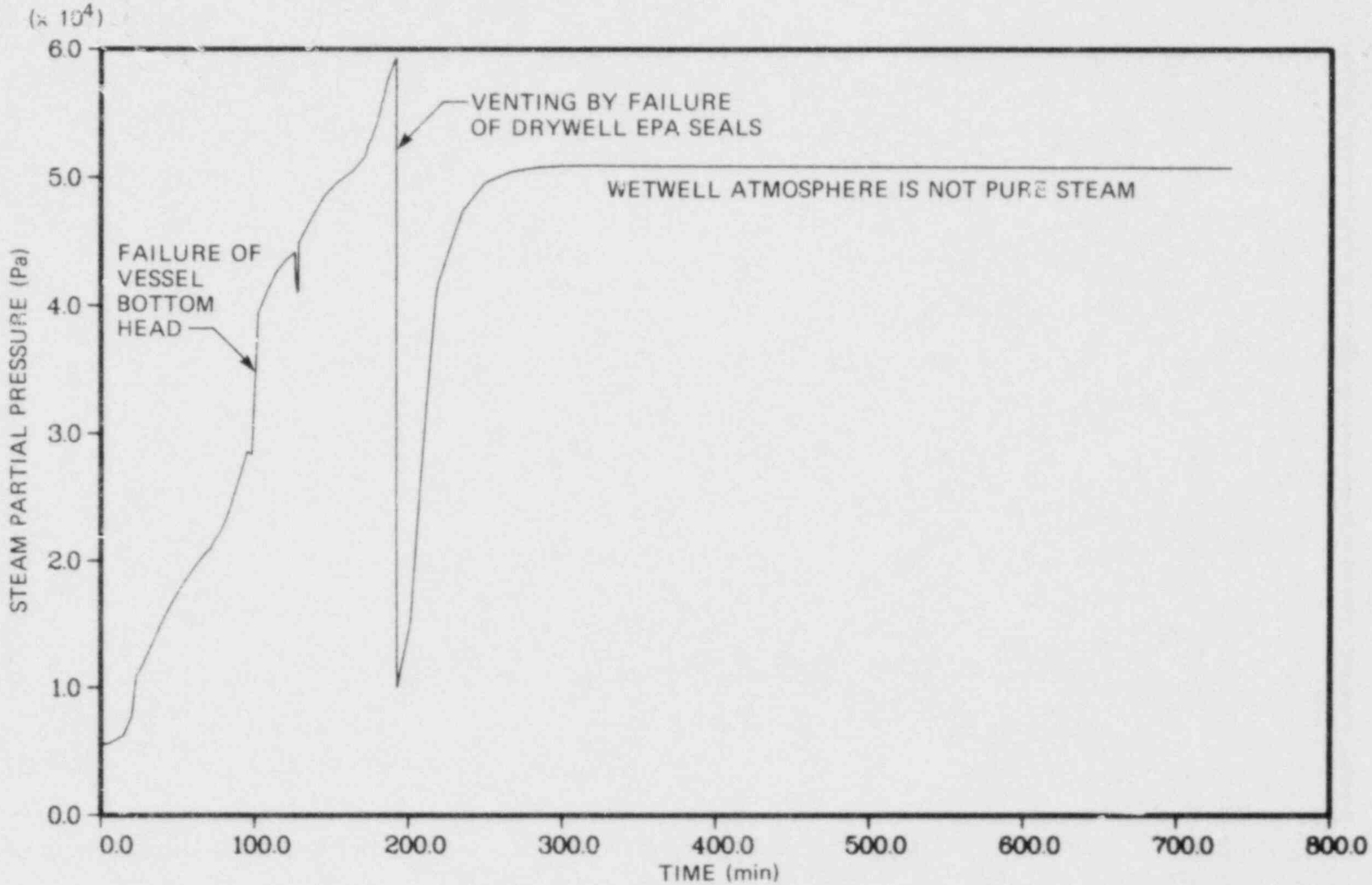

Fig. 9.44 Wetwell steam partial pressure (TUB′).

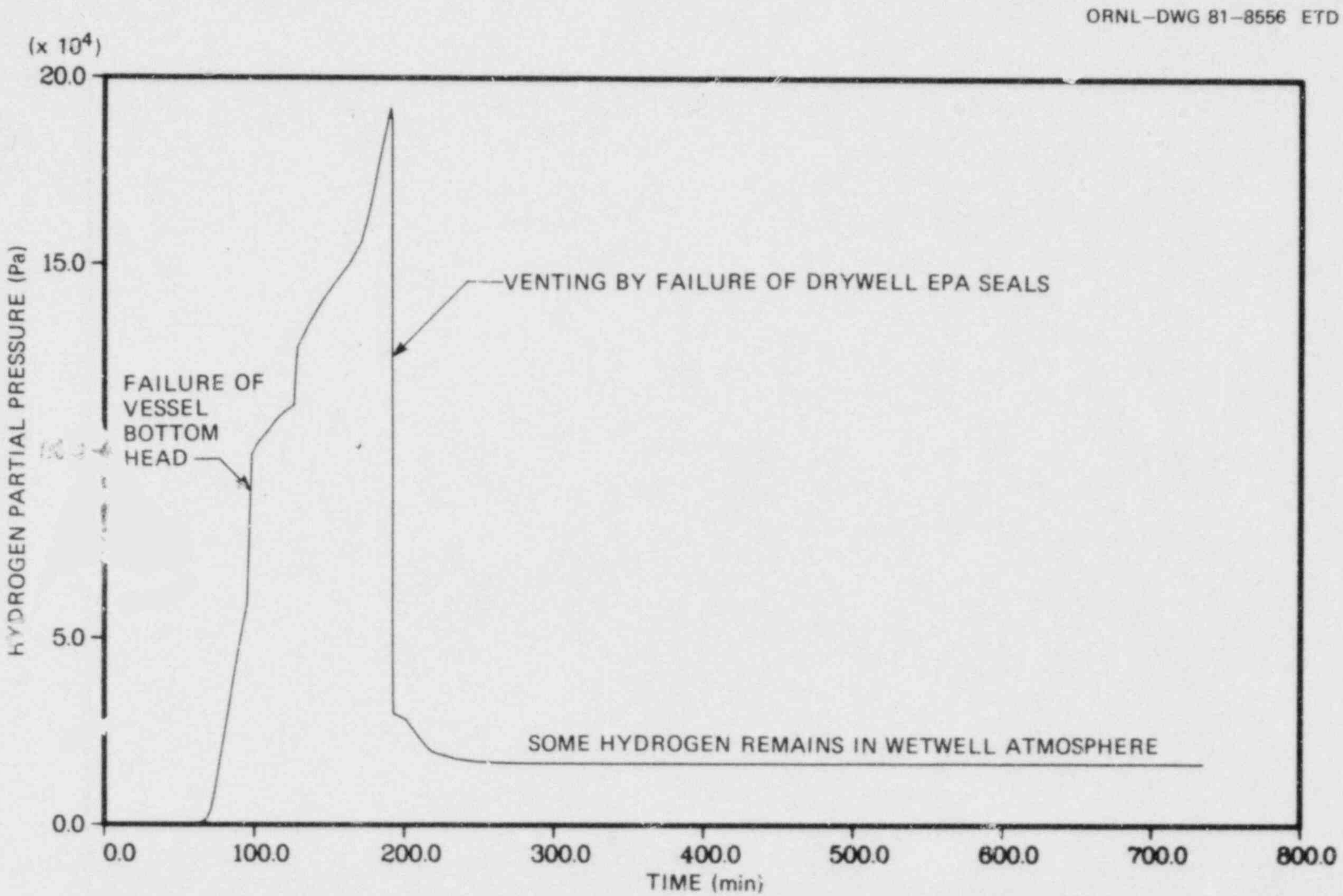

Fig. 9.45 Wetwell hydrogen partial pressure (TUB′).

ORNL-DWG 81-8557 ETD

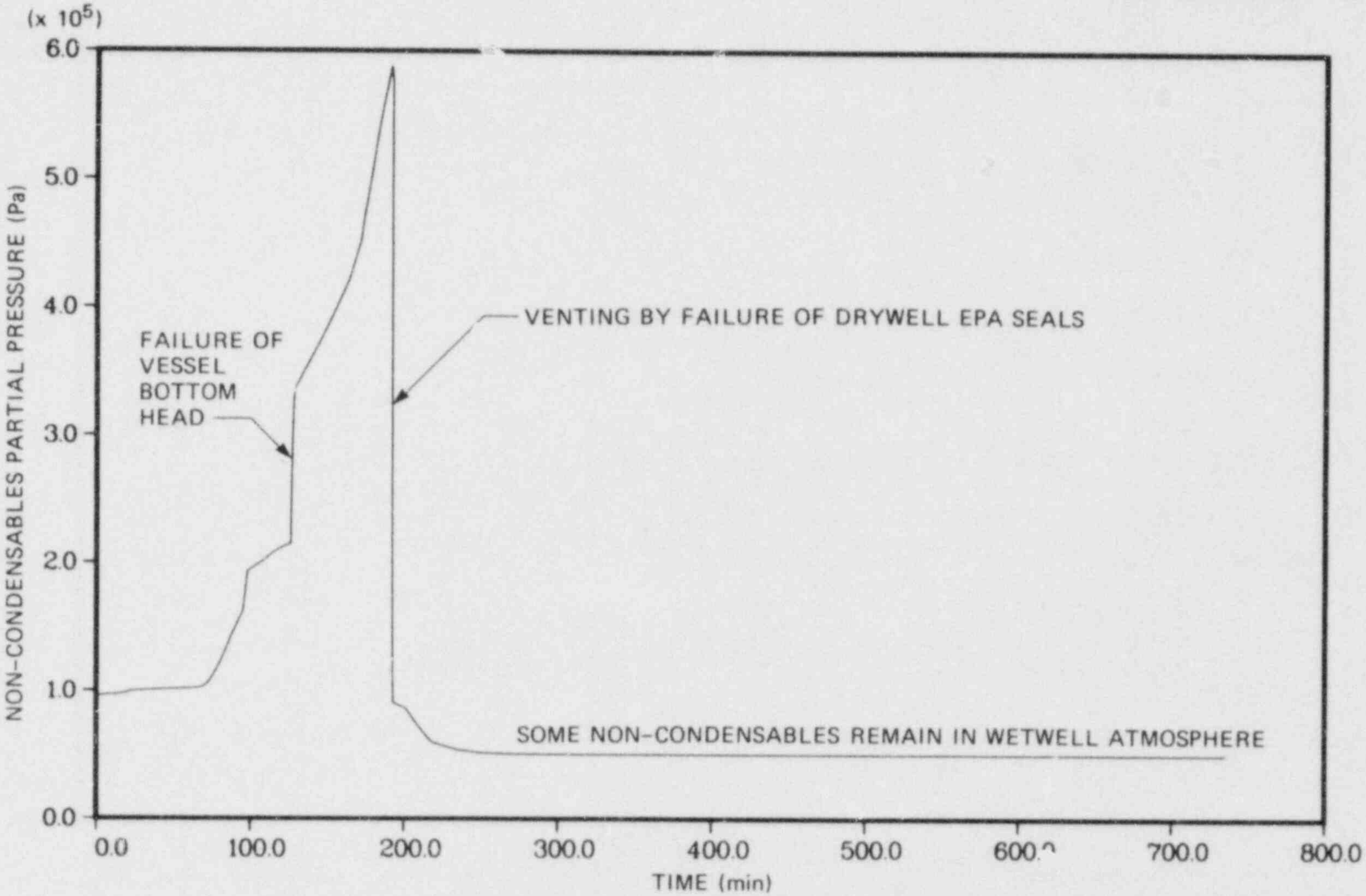

Fig. 9.46 Wetwell non-condensables partial pressure (TUB′).

10. PLANT STATE RECOGNITION AND OPERATOR MITIGATING ACTIONS

10.1 Introduction

The accident signatures developed in Sect. 9 provide baseline information for six possible sequences leading to core meltdown that might occur during a Station Blackout. The objectives of this chapter are to examine these same sequences from the operator's standpoint in order to identify and evaluate potential preventive and corrective actions keyed to the time windows available.

As discussed in Sect. 9, a number of plant safety systems would function automatically during a Station Blackout. The automatic safety system responses include reactor scram, vessel pressure control by SRV operation, and level control through cycling of the HPCI system. Thus, assuming no independent secondary failures, the reactor could be maintained in a safe stable state without operator action until DC power is lost at an assumed four hours into the blackout. After the loss of DC power, operator action would be crucial in attempting to avert the impending core degradation.

10.2 Plant State Recognition

To enable the operator to better identify and evaluate potential mitigating actions during a Station Blackout, the accident progression is categorized into a number of plant states. The available time windows for each plant state are determined from the baseline information provided in Sect. 9. The location of the plant states within each sequence is shown in Fig. 10.1, with the sequence chosen for the fission product transport analysis of Volume 2 indicated by a dashed line.

Plant State 1 (0-s). This is the initial state at the moment of loss of offsite and onsite power, and is common to all of the blackout sequences.

Plant State 2 (0-625 s). This state represents the period of automatic plant safety system response and is common to the six sequences considered in detail in this report.

Plant State 3 (625 s-240 min). This state is applicable to sequences TB′ and TPB′. It is characterized by the availability of adequate decay heat removal by relief valve operation and the HPCI and RCIC injection systems, which are dependent on DC power from the unit battery.

Plant State 3A (625 s-240 min). This state is applicable to sequences T_VB' and T_VPB'. It is characterized by operator action to control level by operation of the RCIC system (only) and to rapidly depressurize the reactor vessel by remote-manual relief valve actuation. The loss of fluid through the relief valve exceeds the capacity of the RCIC system for a short period of time, which causes a momentary core uncovery early in these sequences.

Plant State 3B (625 s-28 min). This state is applicable to sequences TUB′ and TUPB′. In these sequences, the HPCI and RCIC systems are assumed to be inoperable from the inception of the blackout, because of equipment failure. This state is characterized by decreasing water level.

Plant State 4 (240-295 min). This state is applicable to sequences TB′ and TPB′. This is the period immediately following the loss of DC power,

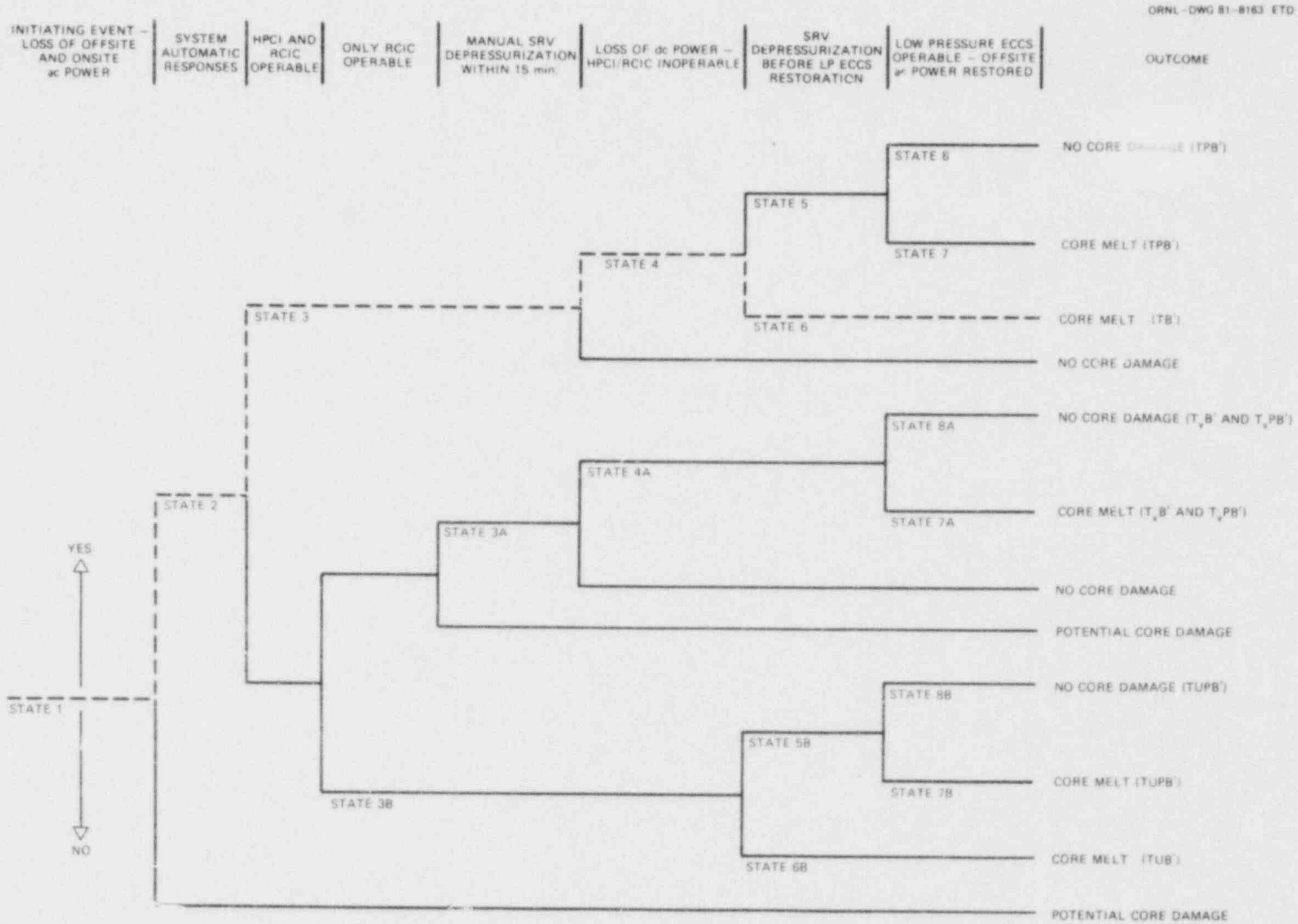

Fig. 10.1 Operator key action event tree.

power, characterized by the unavailability of the HPCI and RCIC systems and a steadily decreasing reactor vessel water level.

Plant State 4A (240-305 min). This state is applicable to sequences T_VB' and T_VPB', and follows the loss of DC power. This state is characterized by a depressurized reactor vessel, unavailability of the HPCI and RCIC systems, and a steadily decreasing vessel water level.

Plant State 5 (295-305 min). This state is applicable to sequence TPB′ and is characterized by a stuck-open relief valve, so that the reactor vessel is depressurized.

Plant State 5B (28-38 min). This state is applicable to sequence TUPB′. This sequence involves the combined secondary failures of inoperative HPCI and RCIC systems and a stuck-open relief valve (SORV) from the inception of the blackout. It is characterized by a rapidly decreasing water level.

Plant State 6 (295-355 min). This state is applicable to sequence TB′ only. It is characterized by boiloff of the reactor vessel water inventory due to automatic relief valve actuation with the reactor vessel fully pressurized, and terminates with the beginning of core melt.

Plant State 6B (28-70 min). This state is applicable to sequence TUB′. It is characterized by a pressurized reactor vessel, core uncovery, and the beginning of core melt.

Plant State 7 (305-360 min). This state is applicable to sequence TPB′. It is characterized by a depressurized reactor vessel, unavailability of the low pressure ECCS injection systems, decreasing water level, core uncovery, and the beginning of core melt.

Plant State 7A (305-360 min). This state is applicable to sequences T_VB' and T_VPB' is identical to state 7.

Plant State 7B (38-360 min). This state is applicable to sequence TUPB′ and is identical to state 7.

Plant State 8 (305-360 min). This state is applicable to sequence TPB′. It is characterized by a depressurized reactor vessel and the recovery of AC power, which permits the restoration of normal vessel water level.

Plant State 8A (305-360 min). This state is applicable to sequences T_VB' and T_VPB' and is identical to state 8.

Plant State 8B (38-360 min). This state is applicable to sequence TUPB′ and is identical to state 8.

10.3 Operator Key Action Event Tree

For sequences TUB′ and TUPB′, and after the loss of DC power in the case of sequences TB′, TPB′, T_VB', and T_VPB', mitigating action by the operator is essential if core degradation is to be avoided and the plant brought to a safe stable state.

Upon restoration of AC power, the operator's primary responsibility is to ensure vessel depressurization and operation of the low pressure injection systems.

The operator key action event tree is shown in Fig. 10.1.

10.4 Operator Mitigating Actions

In addition to continued efforts to restore electrical power, the recommended actions after the loss of water injection capability when boiloff and core damage become inevitable include the use of portable pumps, perhaps provided by fire engines, to flood the drywell in an attempt to preclude melt-through of the reactor vessel. If successful, this would keep the degraded core inside the vessel and prevent gross containment failure and the corresponding major releases of radioactivity. For the sequences TUB′ and TUPB′, in which injection capability is assumed lost at the inception of the blackout, a drywell flooding rate of 3,200 ℓ/s (50,000 gpm) would be necessary to flood the drywell to the level of the reactor vessel core prior to reactor vessel melt-through.

11. INSTRUMENTATION AVAILABLE FOLLOWING LOSS OF 250 VOLT DC POWER

Reactor vessel level and pressure control can be maintained during a Station Blackout for as long as 250 volt DC power from the unit battery remains available, as discussed in Sect. 8. The instrumentation available during this initial phase of a Station Blackout was discussed in Sect. 5. It is the purpose of this section to discuss the instrumentation which would remain operational after the unit battery is exhausted; this final phase of a Station Blackout would constitute a Severe Accident because there would be no means of injecting water into the reactor vessel to maintain a water level over the core.

In addition to the 250 volt unit battery system, there are two smaller battery systems which supply power to Control Room instrumentation and alarm circuits. The first of these is a 24 volt DC system which supplies power to the Source Range and Intermediate Range neutron flux monitors as well as radiation monitors for the off-gas, RHR Service Water, Liquid Radwaste, Reactor Building Closed Cooling Water, and Raw Cooling Water systems, none of which would be operational during a Station Blackout. The 24 volt batteries for this system are designed to supply the connected loads for a period of 3 hours without recharging. Since the 250 volt system batteries are expected to last for a period of four to six hours, it is unlikely that the 24 volt system would remain available after the 250 volt DC unit distribution system has failed.

The second of the smaller battery systems is a 48 volt DC power supply and distribution system for the operation of the plant's communication and annunciator systems. This system comprises three batteries, one of which supplies the plant communications system while the remaining two batteries are for the annunciator system. However, the system design provides that the total station annunciator load can be supplied from one battery. The 48 volt DC system batteries are capable of supplying the connected loads for a period of eight hours without recharging. This is well beyond the period of expected operation of the 250 volt DC systems and assures continued availability of the plant communications system. The efficacy of the annunciator system depends upon the availability of the power supplies to the signal transmission systems of the various sensors as well as the 48 volt DC system. Most of the alarm annunciator capability will be lost when the unit battery fails.

There is reason to believe that plant preferred 120 volt AC single phase power would remain available for a significant period of time after failure of the unit battery. Plant preferred power is obtained from the plant 250-volt DC system during a Station Blackout by means of a DC-motor, AC generator combination. The plant battery is similar to each of the unit batteries, but would be more lightly loaded during a Station Blackout. The major loads on the plant battery are the turbo-generator oil pumps and seal-oil pumps for the three Browns Ferry Units. These pumps could be stopped within one hour after the inception of a Station Blackout since the turbo-generators will have stopped rolling and there is no jacking capability under Station Blackout conditions.

Assuming that plant preferred power does remain available during the period of core uncovery following the loss of the unit battery, two sources of valuable information powered by this system and located outside of the Control Room would remain. The first is an indication of the temperatures at various points within the drywell provided both by a recorder and an indicator-meter for which the displayed drywell temperature can be selected by a set of toggle-switches.

The second information source provided by plant-preferred power consists of the temperatures at various points on the surface of the reactor vessel as provided by 46 attached copper-constantan thermocouples. Unfortunately, the indicating range for the instruments reporting the thermocouple responses is only 315.6°C (0–600°F). However, when the instrument responses for the thermocouples attached near or on the bottom of the reactor vessel are pegged at the high end, the operator can be sure that the core is uncovered, and the only fluid within the vessel is superheated steam. Both the drywell temperature indication and the reactor vessel thermocouple recorder are mounted on panel 9–47, which is located on the back of the Control Room Panels.

Information concerning the reactor vessel water level would also be available outside the Control Room from the mechanical Yarway indication (which requires no electrical power) located at the scram panels on the second floor of the reactor building; this would provide direct indication of the reactor vessel water level over the range from 14.94 to 9.47 m (588 to 373 in.) above vessel zero. The lower boundary of this Yarway indicating range is 0.33 m (13 in.) above the top of the active fuel in the core.

There would be no indication of reactor vessel pressure after loss of the unit battery. However, it is expected that the vessel pressure would be maintained in the range of 7.722 to 7.377 MPa (1105 to 1055 psig) by repeated automatic actuation of a primary relief valve for as long as the reactor vessel remains intact.

It should be noted that the emergency lighting for the Control Room is supplied from the 250 volt DC system; after failure of this system hand-held lighting for the Control Room would be necessary. The door security system is supplied by plant preferred power and would remain operable as long as the plant 250 volt DC system is functional. The area radiation monitors located throughout the plant are powered from the Instrumentation and Control buses and would not be operational from the inception of a Station Blackout.

12. IMPLICATIONS OF RESULTS

The purpose of this section is to provide a discussion of the present state of readiness at the Browns Ferry Nuclear Plant to cope with a Station Blackout. The discussion will include consideration of the available instrumentation, the level of operator training, the existing emergency procedures, and the overall system design.

12.1 Instrumentation

The availability of Control Room instrumentation during the period of a Station Blackout in which 250 volt DC power remains available has been discussed in Sect. 5. The most important parameters while injection capability remains are the reactor vessel level and pressure, and these would be adequately displayed in the Control Room during this period. This instrumentation available after the loss of DC power was discussed in Sect. 11.

The existing instrumentation at Browns Ferry Unit 1 which would be important during a Station Blackout is summarized below, with additional information concerning the power supplies. The instrumentation is discussed in the order given in reference 28, and is located in the Control Room unless otherwise indicated.

I. Core
   A. Core Exit Temperature — not available at Browns Ferry.
   B. Control Rod Position — the rod position indicating system is powered by the 120V AC unit-preferred system, which would be available during a Station Blackout for as long as 250V—DC power remains available from the unit battery.
   C. Neutron Flux — The Source Range and Intermediate Range monitors are powered by the 24 volt DC battery system, which is designed to supply the connected loads for a period of 3 hours under the conditions of a Station Blackout. The Source and Intermediate Range detectors are withdrawn from the core during power operation to a point 0.61 m (2 ft) below the bottom of the active fuel to increase their active life, and these detectors could not be reinserted during a Station Blackout. However, the relative changes in neutron flux following reactor scram will produce a proportional change in Source Range meter (range: 0.1 to 10^6 CPS) response when the detector is fully withdrawn.

II. Reactor Coolant System
   A. RCS Pressure — one channel of reactor vessel pressure powered by the feedwater inverter and another channel powered by the unit-preferred system, each with a range of 8.274 MPa (0 to 1200 psig), would be available during a Station Blackout for as long as the unit battery continues to supply 250 volt DC power.
   B. Coolant Level in the Reactor — two channels of reactor vessel level instrumentation, one each powered by the feedwater inverter and the unit-preferred system would be available during Station Blackout for as long as the 250 volt DC unit battery

system remains functional. These two channels each have a range of 1.524 M (0 to 60 in.) and provide indication of the reactor vessel water level between 13.41 and 14.94 m (528 and 588 in.) above the bottom of the vessel, a range which extends over the upper portion of the steam separators. Additional level information would be available outside of the Control Room, as follows:

- Two channels of Yarway level indication over the range from 9.47 to 14.94 m (373 to 588 in.) above the bottom of the vessel are available at the backup control panel located in the Backup Control Room. This instrumentation is powered from the unit-preferred system; the lower limit of this indication is 0.33 m (13 in.) above the top of the active fuel in the core.
- The Yarway instruments are located at the scram panel on the second floor of the reactor building and mechanically display the reactor vessel water level over the same range as that electrically transmitted to the backup control panel. This level indication at the Yarway instruments requires no electrical power, and would remain available even after all DC power was lost.

C. Main Steamline Flow — two channels of steam flow instrumentation powered by the feedwater inverter would provide an indication of total steam flow [range: 2016 kg/s (0 to 16 × 10^6 lbs/h)] during a Station Blackout while the unit battery remains functional.

D. Main Steamline Isolation Valves' Leakage Control System Pressure — not applicable to Browns Ferry.

E. Primary System Safety Relief Valve Positons, Including ADS — There is no provision for indication of the actual position of any primary relief valve, including those assigned to the ADS system. Under normal operating conditions, the operator can identify a leaking relief valve by means of the recorded tailpipe temperatures available on charts behind the Control Room panels, or by the recently installed (December, 1980) acoustic valve monitors. Both of these leakage detection systems would be inoperable under Station Blackout conditions.

On the other hand, remote-manual actuation of a relief valve is accomplished by energizing the associated DC solenoid operator, and each valve has lights on the control panel which indicate whether or not the solenoid for that valve is energized. The capability for this indirect indication of successful remote-manual valve actuation is maintained during the period of a Station Blackout in which 250 volt DC power is available from the unit battery.

F. Radiation Level in Coolant — not available at Browns Ferry.

III. Containment

A. Primary Containment Pressure — Drywell pressure instrumentation with a range of 0.55 MPa (0-80 psia) powered by the unit-preferred system will be available during a Station

Blackout while the unit battery remains functional. A set of 12 vacuum breakers ensures that the pressure in the pressure suppression pool torus cannot exceed the drywell pressure by more than 0.003 MPa (0.5 psi).

B. Containment and Drywell Hydrogen Concentration — Instruments for this purpose are provided for each Browns Ferry Unit, but would not be available during a Station Blackout.

C. Containment and Drywell Oxgyen Concentration — Instruments for this purpose are provided, but as in the case of the hydrogen monitors, are powered from the Instrumentation and Control buses which would not be available during a Station Blackout.

D. Primary Containment Isolation Valve Position — During a Station Blackout, valve position indication is maintained for the Main Steam Isolation Valves (MSIV's) and for the systems which remain operable, i.e., the HPCI and the RCIC systems. However, valve position indication for the primary containment isolation valves in the low-pressure injection systems which are powered by sources not available during a Station Blackout would be lost.

E. Suppression Pool Water Level — This instrument, powered by the unit-preferred system, would indicate the water level in the torus with a range of 1.27 m (−25 to +25 in.) as long as the unit battery is functional.

F. Suppression Pool Water Temperature — This instrumentation would not be available during a Station Blackout.

G. Drywell Pressure — This is synonymous with "Primary Containment Pressure" — see "A" above.

H. Drywell Drain Sumps Level — not indicated. The frequency of drain pump operation for each of the two 1.893 m^3 (500 gal.) drain sumps in the drywell is indicated in the Control Room under normal conditions, but would not be operable during a Station Blackout.

I. High-Range Containment Area Radiation — not available at Browns Ferry.

IV. Power Conversion Systems

A. Main Feedwater Flow — one channel of flow instrumentation with a range of 1008 kg/s (0 to 8 × 10^6 lbs/h) and powered by the unit-preferred system is provided for each of the three feedwater pumps. This instrumentation would remain operable as long as the unit battery is functional.

B. Condensate Storge Tank Level — The condensate storage tank level indication [range: 9.75 m (0 to 32 ft)] is powered by the unit-preferred system and would remain operational until the unit battery is exhausted.

V. Auxiliary Systems

A. Steam Flow to RCIC — This instrumentation is powered by the Instrumentation and Control buses and would not be available during a Station Blackout.

B. RCIC Flow — The RCIC pumped flow is indicated over a range of 0.044 m^3/s (0–700 GPM) on the flow controller, which is powered by the unit-preferred system and would be available as long as the unit battery is functional.

VI. Radiation Exposure Rates — The permanently-installed radiation monitors located throughout the plant are powered from the Instrumentation and Control buses and would not be operational during a Station Blackout.

The available Control Room instrumentation is adequate to monitor the plant response to a Station Blackout during the estimated four-hour period during which 250 volt DC power would remain available from the unit battery. After the unit battery is exhausted, virtually all Control Room instrumentation would be lost and water could no longer be injected into the reactor vessel to make up for that lost to the pressure suppression pool through relief valve actuation. Consequently, the reactor vessel water level would slowly decrease until the core became uncovered.

The most important single parameter during the period of decreasing water level following loss of the unit battery is the reactor vessel level, which could be monitored at the Yarway instruments on the second floor of the Reactor Building as described in part III B above. In addition, the drywell ambient temperatures and the reactor vessel surface temperatures would remain available as long as the plant-preferred 250 volt-DC system is functional, as described in Sect. 11.

12.2 Operator Preparedness

There is currently no specific training provided Browns Ferry Operators in regard to plant response or required operator actions in the event of a complete Station Blackout.

A Station Blackout casualty can be run at the TVA Browns Ferry Unit 1 simulator by selecting the pre-programmed loss-of-offsite-power casualty and then having the simulator operator immediately punch the diesel-generator manual trips in the control room. Since the instrumentation in the simulated control room is powered from several diverse sources representative of those at the actual plant, the resulting effect closely models a Station Blackout. This procedure was developed and tested by TVA simulator operating personnel as an effort in cooperation with this study, but has not been used for operator training.

There is no Emergency Operating Instruction to cover a Station Blackout at the Browns Ferry Nuclear Plant, but there are Emergency Operating Instructions for the loss of individual subsections of the overall AC system.

With two exceptions, the current level of operator training and the existing Emergency Operating Instructions are believed adequate to prepare the operator to cope with a Station Blackout during the initial phase in which 250 volt-DC power would remain available. The first exception involves the need to depressurize the reactor vessel within the first hour. The Technical Specifications provide that the reactor vessel must be depressurized to less than 1.48 MPa (200 psig) if the pressure suppression pool temperature exceeds 48.9°C (120°F); this is based on a maximum permissible temperature of 76.7°C (170°F) for the pool so that complete condensation of steam is ensured in the event of a LOCA. The Emergency

Operating Instructions and the Operator Training manuals establish this requirement clearly.

During a Station Blackout, suppression pool temperature indication is not available and the operator may be reluctant to depressurize the reactor vessel, knowing that suppression pool cooling is not operable. However, depressurization is necessary for two reasons. The first is to preclude an excessive drywell ambient temperature by reducing the temperature of the saturated fluid within the reactor vessel, as discussed in Sect. 3. The second reason is to provide additional time between the loss of injection capability and core uncovery; as explained in Sect. 7, no coolant is lost from the reactor vessel during the significant period of time required for the reactor vessel to repressurize to the set point for automatic relief valve actuation.

With an early depressurization, a great deal of energy and coolant mass is dumped to the pressure suppression pool early in the Station Blackout when the HPCI and RCIC systems are available to restore the lost coolant and maintain vessel level in the normal operating range. Should the Station Blackout continue to the point that the HPCI and RCIC systems fail due to exhaustion of the unit battery, a significant amount of time is required after injection failure for the reactor decay heat to add the energy required for vessel repressurization. Thus early reactor vessel depressurization will provide valuable additional time for corrective action before core uncovery.

The second exception to the current level of operator training and the Emergency Operating Instructions with regard to a Station Blackout involves the need for the operator to reduce the load upon the 250 volt DC system as much and as quickly as possible to prolong the life of the unit battery. There is a need for a procedure which lists all of the loads upon the unit battery, indicates whether or not each load is significant, the purpose of each load, and the consequences of removal.

Operator actions during a Station Blackout should be directed toward keeping the vessel level in the normal operating range, early vessel depressurization, and reduction of the battery load. All of these actions will increase the time available in which maintenance actions can be taken to restore AC power before core damage occurs. The recommendations of this study concerning operator preparations can be summarized as:

1. Operator training and the Emergency Operating Instructions should explain that reactor vessel depressurization reduces the driving head for heat transfer into the drywell during a casualty, and provides a significant additional margin of time before core uncovery occurs once injection capability is lost. The present operator training implies that the only reason for depressurization is to satisfy the Technical Specification, written in anticipation of a LOCA.
2. The plant batteries are provided as energy-storage systems for temporary use when the normal AC power supply systems are unavailable. However, during a Station Blackout or other casualty in which the normal AC supply to a battery charger is unavailable for a significant period of time, the operator is faced with the necessity of opening the power-supply breakers to the less-important loads to prolong the availability of power to the more essential equipment. It is recommended that the priority of the battery loads be established in advance of the casualty, by means of a procedure which indicates the

recommended order of removal of loads if the loss of AC power is perceived to be long-term.

3. Recovery procedures should be developed to provide a detailed method for recovery of vital power supplies and equipment in a safe, efficient manner upon restoration of AC power following a Station Blackout.

12.3 System Design

The existing system design provides sufficient instrumentation and equipment to maintain decay heat removal capability for several hours during a Station Blackout. The only questionable feature of the existing design with regard to the Station Blackout sequence is the provision for automatic shifting of the HPCI pump suction to the pressure suppression pool on high sensed pool level. As discussed in Sect. 8.1, this can lead to failure of the HPCI system because of the high temperature of the pressure suppression pool water at the time the shift in pump suction occurs.

Separate provision is made for an automatic shift of the HPCI pump suction if the normal condensate storage tank water source becomes exhausted. Thus it appears that the automatic high 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 Technical Specifications, and it should be noted that there is no corresponding provision for the RCIC system. It is recommended that the desirability of an automatic shift in HPCI pump suction on high suppression pool water level be reexamined and if found necessary, that the basis be included in operator training.

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APPENDIX A

LISTING OF PLANT RESPONSE CODE BWR-LACP

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$$$CONTINUOUS SYSTEM MODELING PROGRAM  III   V1M3   TRANSLATOR OUTPUT$$$

  LABEL  -------------------------BWR-LACP CODE------------------------

  LABEL  CONTAINMENT MODEL WITH 1-NODE POOL
*
*------------MACRO SUBPROGRAM FOR POOL WATER TEMP CALCULATION------------
 MACRO XNQI,TI,WEI,MWI,LWI=W(TL,TR,LWL,LWR,TWIO,FWI,FSVI,FTEI,FDI,FSI)
*  MACRO TC CALCULATE MASS ENERGY BALANCES FOR SP WATER NODE I
*  INPUTS :
*  TL,TR=TEMPS OF WATER IN ADJACENT NCDES
*  LWL,LWR=LEVELS OF ADJACENT NODES
*  TWIO=INITIAL WATER TEMP CF NCDE I
*  FWI=FRACTION OF TOTAL SP WATER VOL IN NCDE I(CONSTANT)
*  FSVI=FRACTION OF SAFETY VALVE FLOW DIRECTED TO NODE I
*  FTEI=FRACTION OF TURBINE EXHAUST DIRECTED TO NODE I
*  FDI=FRACTION OF PUMP DUSCHARGE DIRECTEC TC NODE I
*  FSI=FRACTION OF PUMP SUCTION TAKEN FROM NODE I
*
*
*  MASS BALANCE
*
*  WSSPW=PUMP SUCTION FLCW(TOTAL FROM SP)
*  WDSPW=PUMP DISCHARGE FLOW(TOTAL TO SP)
*  OUTPUTS:
*  XNQI=FRACTIN
*  XNQI=FRACTCON OF STM DISCH TC NODE I NOT QUENCHED
*  XNQI=FRACTION OF STM DISCH TO NODE I NOT QUENCHED
*  TI=AVG. WATER TEMP OF NODE I (DEG-F)
*  WEI=RATE OF EVAPORATION FROM NODE I (LB/SEC)
*  MWI=MASS OF WATER IN NODE I (LB)
*  LWI=WATER LEVEL OF NODE I (IN)
*
*  VWSPO=INITIAL TOTAL SP WATER VOLUME(CU.FT.)
      MWIO=FWI*VWSPO*(63.9-.019*TWIO)
     CMWI=INTGRL(0.,XQI*(FSVI*WSSV+FTEI*WSTE)+FINI-FSI*WSSPW-WEI)
      FINI=WEQL+WEQR+FWI*WCSPG+FDI*WDSPW
      MWI=MWIO+CMWI
      VWI=MWI/(63.9-.019*TI)
      LWI=AFGEN(SPLEV,VWI/FWI)
*
*  CALC. QUENCH FRACTION FROM ASSUMPTION OF NO QUENCH WHEN NODE VAPOR
*  PRESSURE EQUAL TC TOTAL GAS PRESSURE
      PVAPI=NLFGEN(SPFOT,TI)
      XQI=LIMIT(0.,1.,(PTSPG-PVAPI-DPQZ)/DPQR)
      XNQI=(1.-XQI)
*  SPFOT(TI)=SAT. PRESS AS F OF TI
*  DPQZ=DIFFERENCE(PSI) OF PTSPG OVER PVAPI REQUIRED FOR ANY
*  QUENCHING TO TAKE PLACE
*  DPZR=RANGE(PSI) CF PTSPG OVER PVAPI CVER WHICH QUENCHING
*  GOES FROM 0.0 TO 100PERCENT
*
*  EVAPORATICN RATE BASED ON EQUN.13-33, KREITH'S 'HEAT TRANSFER'
*  PSSPG=STM. PART PRESSURE ON SP GAS
*  ASSPW=SURF AREA OF SP WATER(FT.SQ)
      WEI=(WEIN/WEID)*CCMPAR(PVAPI,PSSPG)
      WEIN=ASSPW*FWI*HSI*12.3*(PVAPI-PSSPG)*PTSPG*VGSP
      WEID=(PNSPG*2.-PVAPI)*TGSPR*(MSSPG+MNSPG)
      HSI=5.83E-05*((ABS(TI-TGSP))**.333)
*
*  EQUALIZATION FLOWS FROM ACJACENT NODE(S)
*  FWE=EQ. FLOW CONSTANT (LB/SEC/(INCH LEVEL DIFF) )
      WEQL=FWE*(LWL-LWI)
      WEQR=FWE*(LWR-LWI)
*  ENERGY BALANCE
*  HEAT TRANSF RATES TO METAL AND ATMCSPHERE NEGLEGIBLE
```

```
*  WITH RESPECT TO MASS-MIXING ENERGY TRANSFER
    CMHWI=INTGRL(0.,XQI*(FSVI*WSSV*HST+WSTE*FTEI*HSTE)+E1I+E2I)
      E1I=(TLX-32.)*WEQL+(TRX-32.)*WECR+WML*(TL-TI)+WMR*(TR-TI)
      E2I=FWI*WCSPG*HFGSP+FDI*WDSPW*HD-FSI*WSSPW*(TI-32.)-WEI*1105.
      TLX=TL*COMPAR(LWL,LWI)+TI*COMPAR(LWI,LWL)
      TRX=TR*COMPAR(LWR,LWI)+TI*COMPAR(LWI,LWR)
*
*  CALC OF MIXING FLOWS(THEY DID NOT APPEAR IN THE MASS BALANCE BECAUSE
*  ZERO NET MASS TRANSFER IS ASSUMED
      WML=KMIX*SQRT(ABS(TL-TI))
      WMR=KMIX*SQRT(ABS(TR-TI))
*
*
*  TEMPERATURE CALCULATION ASSUMES CONSTANT SPECIFIC HEAT=1.0 FOR WATER
      TI=32. + ((TWIO-32.)*MWIO + CMHWI)/MWI
*
 ENDMACRO
*
*
     INITIAL
*
*----------INITIALIZATION FOR REACTOR VESSEL CALCULATION---------------
*
*                          CONSTANTS FOR REACTOR MODEL
*
*  ACOP=CORE OUTLET PLENUM FLOW AREA(FT**2)
*  ACOR=CORE FLOW AREA(FT**2)
*  ART=RISER TUBE FLOW AREA(FT**2)
*  CPO=PRE-TRIP CORE POWER(TOTAL) (MWTH)
*  EQXO=FRACTION OF WAY TO SATURATION THAT STEAM CONTACT RAISES
*       INJECTION WATER IF DC LEVEL IS AT LEVEL OF JET PUMP SUCTION
*  FFLASH=FRACTION FLASHED/SEC PER BTU/LB ABOVE SATURATION
*  HCIO=INITIAL CORE INLET ENTHALPY(BTU/LB)
*  HINJIN=ENTHALPY OF INJECTION FLUID(BTU/LB)
*  LBOT=HEIGHT OF STM SEP BOTTOM(FT)
*  LDCO=INITIAL DOWNCOMER LEVEL(FT) (HEIGHT ABOVE BOT. OF ACT. FUEL)
*  LHEDER=HEIGHT OF FW HEADER ABOVE BOAF(FT)
*  LOP=AVG. LENGTH OF CORE OUTLET PLENUM(FT)
*  LRT=AVG. LENGTH OF RISER TUBES(FT) (STANDPIPES)
*  PCOR=CORE HEAT TRANSFER PERIMETER(FT)
*  PO=INITIAL REACTOR VESSEL PRESSURE(PSIA)
*  RCICMX=NOMINAL RCIC FULL FLOW(LB/SEC)
*  TAULEN=STABILITY TIME CONSTANT FOR REGION AVERAGE HEAT FLUX CALC(SEC)
*  TCFUEL=TIME CONSTANT TO ACCOUNT FOR RESIDUAL HEAT IN CORE FOR
*   INITIALIZAT.ON CLOSELY FOLLOWING SCRAM
*  TO=TIME AFTER TRIP THAT TRANSIENT INITIATES(SEC)
*  VOLP=LOWER PLENUM VOLUME(FT**3)
*  VIDC=VOLUME OF DOWNCOMER BETWEEN BOAF AND PUMP DIFFUSER EXIT(FT**3)
*  WINJO=INITIAL INJECTION FLOW(LB/LEC)
*
*  REFERENCE PARAMETERS FOR NATURAL CIRC LOSS COEFF CALCULATION
*  CPR=RATED CORE POWER(MWTH)
*  HREF=REFERENCE CORE INLET ENTHALPY(BTU/LB)
*  LDCR=REFERENCE DOWNCOMER LEVEL(I.E. NORMAL LEVEL AT STM SEP MIDDLE)
*  PRELR=REFERENCE RELATIVE CORE POWER
*  PRR=REFERENCE REACTOR VESSEL PRESSURE
*  RHOO=DENSITY OF STEAM(LB/FT**3) IN R.V. AT INITIAL PRESSURE=PO
*  WGUESS=GUESS ON FLOW(LB/SEC) FOR ITERATIVE SOLUTION FOR INITIAL FLOW
*  WREF=REFERENCE FLOW(LB/SEC)
*  XREF=REFERENCE FRACTIONAL QUALITY
*
*  PARAMETERS FOUND IN FUNCTION SUBPROGRAMS
*  VDIFF=JET PUMP VOL BELOW BOAF(FT**3)
*  VFREE=TOTAL STM. VOL. IN R.V.(LESS LOWER PLENUM,CORE,CORE OUTLET PLENUM
*  AND STM. SEPARATORS) AND MAIN STEAMLINES TO ISOLATION VALVES(FT**3)
*  VJET=VOL BETWEEN JP SUCTION AND BOAF(FT**3)
*  VANN=VOL BETWEEN TOP OF CORE OUTLET PLENUM AND JP SUCTION(FT**3)
*  VSSOP=VOL BETWEEN TOP OF CO PLENUM AND BOTTOM OF STM SEP(CU.FT.)
*  WRATED=STM. FLOW THUR 1 SRV (LB/SEC) WHEN PRESSURE=PRATED (PSIA)
*  XKVGJ=EMPIRICLE CONSTANT USED IN CALCULATION OF DRIFT VELOCITY
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*  XL1=0=ZERO POINT FOR DC HEIGHT(FT):  COINCODES WITH BOAF
*  XL2=HEIGHT AT JET PUMP SUCTION(FT)
*  XL3=HEIGHT AT TOP OF CORE OUTLET PLENUM(FT)
*  XL4=HEIGHT AT BOTTOM OF STEAM SEPARATORS(FT)
      CONSTANT  ACOR=82., ACOP=234., ART=42.?
      CONSTANT  CPO=3293., EQXO=.5, FFLASH=.001
      CONSTANT  LHEDER=23.4, LOP=5.58, LRT=10.0
      CONSTANT LBOT=24.85
      CONSTANT  PCOR=5518., PRATED=1095.
      CONSTANT  TAULEN=.75, TCFUEL=9.5
      CONSTANT  VFREE=13000., VCLP=3350., V1DC=192.,WRATED=259.
*
*  CARD TO INITIALIZE AT ARBITRARY TIME POINT
      CONSTANT  HCIO=312.05,LDCO=28.63,PO=125.,TO=14430.,RHOGO=.2788
*  INITIAL CORE FLOW GUESS DEPENDS ON LDCO AND PRELO
      CONSTANT  WGUESS=8310.
*  REFERENCE PARAMETERS FOR CALCULATION OF NATURAL CIRCULATION
*  FLOW RESISTANCE COEFFICIENT
      CONSTANT  CPREF=3293., PRELR=.32, PRR=1020.,...
                LDCR=27.58, XREF=.133, WREF=9111., HREF=522.
*  RUN CONTROL PARAMETERS
      CONSTANT  HINJIN=58.,RC?CMX=82.9876
*
*                        FUNCTION SUBPROGRAMS
*  CAVOID(XIN,XCUT,WTOTAL,TSAT,RHOF,RHOG)
*  DKFUN(T)=P/PO AS A FUNCTION OF TIME AFTER SHUTDOWN
*  QREGAV(BOTTOM,TOP,KAPPA,PEKAVG)=(AVG HT. FLUX FROM A TO B)/(CORE AVG)
*  RHOTP(RHOF,RHOG,VOID)=2-PHASE DENSITY
*  XLENDC(VOLDC)=DOWNCOMER LEVEL AS FUNCTION OF LIQUID VOLUME
*  VOID(QUALITY,TOTAL FLOW,FLOW AREA,TSAT,RHOF,RHOG)=POINT VOID FRACTION
*  VOLDC(LDC)=DOWNCOMER HEIGHT ABOVE BOT OF ACT. FUEL
*
*     STEAM TABLE CSMP INTERPOLATION FUNCTIONS
*  VSATF(PRESS)=SAT FLUID SPECIFIC VOL(FT**3/LB)
      FUNCTION  VSATF=15.,.0167, 50.,.0173, 100.,.0177, 200.,.0184,...
        400.,.0193, 600.,.0201, 800.,.0209, 1000.,.0216, 1200.,.0223,...
        1400.,.0231
*
*  VSATG(PRESS)=SAT GAS SPECIFIC VOL(FT**3/LB)
      FUNCTION  VSATG=15.,26.3, 50.,8.51, 75.,5.81,100.,4.43,...
        150.,3.01,200.,2.29,400.,1.16, 600.,.77,800.,.569,...
       1000.,.446,1200.,.362,1400.,.302
*
*  HSATF(PRESSURE)=SAT FLUID ENTHALPY(BTU/LB)
      FUNCTION HSATF=15.,181., 50.,250., 100.,298., 200.,355.,...
        400.,424., 600.,472., 800.,510., 1000.,543., 1200.,572.,...
        1400.,599.
*
*  HSATG(PRESSURE)=SAT GAS ENTHALPY(BTU/LB)
      FUNCTION HSATG=15.,1151., 50.,1174., 100.,1187., 200.,1198.,...
        400.,1205., 600.,1204., 800.,1199., 1000.,1193., 1200.,1185.,...
        1400.,1175.
*
*  TSATM(PRESSURE)=SAT MIXTURE TEMPERATURE(DEG-F)
      FUNCTION TSATM=15.,213., 50.,281., 100.,328., 200.,382.,...
        400.,445., 600.,486., 800.,518., 1000.,545., 1200.,567.,...
        1400.,587.
*
*  VSC(ENTHALPY)=SUBCOOLED FLUID SPECIFIC VOL(FT**3/LB)
      FUNCTION  VSC=21.4,.016, 121.,.0163, 221.,.0169, 272.,.0174,...
        376.,.0185, 431.,.0193, 488.,.0203, 549.,.0217, 562.,.0221,...
        575.,.0224
*  CALCULATION OF INITIAL SATURATION PROPERTIES
      RHOFO=1./(AFGEN(VSATF,PO))
      HFO=AFGEN(HSATF,PO)
      HGO=AFGEN(HSATG,PO)
      TSATO=FUNGEN(TSATM,2,PO)
*  SUBCOOLED WATER DENSITY
      RHOSCO=1./(AFGEN(VSC,HCIO))
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*
*  CALCULATION OF FLOW RESISTANCE COEFFICIENT
*  NEEDS:  REFERENCE CONDITIONS FOR NATURAL CIRCULATION:CPREF,PRELR,PRR,XREF,
*                                                       WREF, HREF, LDCR
*  RETURNS:  LOSS COEFFICIENT:  KLOSSU
*  CALCULATION OF REFERENCE DENSITIES AND SATURATION PROPERTIES
      HFR=AFGEN(HSATF,PRR)
      TSATR=FUNGEN(TSATM,2,PRR)
      RHOFR=1./(AFGEN(VSATF,PRR))
      RHOGR=1./(FUNGEN(VSATG,2,PRR))
      RHOSCR=1./(AFGEN(VSC,HREF))
      QTOTR=CPREF*PRELR*947.8
*  THIS EXP FOR LSC ASSUMES AVG POWER=CORE AVG IN SC REGION
      LSCR=12.*WREF*(HFR-HREF)/QTOTR
      LBR=12.-LSCR
      RSCAVR=RHOFR/2.+RHOSCR/2.
*  FUNCTION SUBPROGRAM CAVOID RETURNS AVERAGE CORE BOILING REGION
*  VOID FRACTION FROM THE GIVEN INPUTS
      CAVR=CAVOID(0.,XREF,WREF,TSATR,RHOFR,RHOGR)
      RHOBR=RHOTP(RHOFR,RHOGR,CAVR)
      ROCORR=RSCAVR*LSCR/12.+RHOBR*LBR/12.
      RHOPR=RHOTP(RHOFR,RHOGR,OPVR)
*  FUNCTION VOID RETURNS POINT VOID FRACTION FROM GIVEN INPUTS
      OPVR=VOID(XREF,WREF,ACOP,TSATR,RHOFR,RHOGR)
      RORTR=RHOTP(RHOFR,RHOGR,RTVR)
      RTVR=VOID(XREF,WREF,ART,TSATR,RHOFR,RHOGR)
      RDP=LDCR*RHOSCR/144.-RORTR*LRT/144.-...
        RHOPR*LOP/144.-ROCORR*12./144.
      KLOSSU=RDP/(WREF*WREF)
*
*  INITIAL RELATIVE POWER
*
      PRELO=(DKFUN(TO)+.94*EXP(-TO/TCFUEL))
*
      QTOTO=PRELO*CPO*947.8
*
*  CALCULATION OF INITIAL TOTAL FLOW BY ITERATIVE SOLUTION, WITH
*  STARTING FLOW GUESS INPUT BY USER
      WCIO=IMPL(WGUESS,.001,WCAL)
      XOU=(QTOTO/WCIO+HCIO-HFO)/(HGO-HFO)
      XO=LIMIT(0.001,1.000,XOU)
*  THIS EXP FOR LSC ASSUMES AVG POWER=CORE AVG IN SC REGION
      LSCO=LIMIT(0.,12.,12.*WCIO*(HFO-HCIO)/QTOTO)
      LBO=12.-LSCO
*  CALCULATION OF INITIAL DENSITIES
      RSCAVO=RHOFO/2.+RHOSCO/2.
      WBOO=WCIO*XO
*  VOID AT BOTTOM AND TOP OF BOILING REGION CALC. BY DFVOID
*  FUNCTION SUBPROGRAM, THEN AVERAGED TO GIVE CORE AVERAGE VOID
      CAVINO=DFVOID(ACOR,WCIO,0.,TSATO,RHOFO,RHOGO)
      CAVEXO=DFVOID(ACOR,WCIO-WBOO,WBOO,TSATO,RHOFO,RHOGO)
      CAVO=(CAVINO+CAVEXO)/2.
      RHOBO=RHOTP(RHOFO,RHOGO,CAVO)
      ROCORO=RSCAVO*LSCO/12.+RHOBO*LBO/12.
      ROPO=RHOTP(RHOFO,RHOGO,OPVO)
      OPVO=DFVOID(ACOP,WCIO-WBOO,WBOO,TSATO,RHOFO,RHOGO)
      RORTO=RHOTP(RHOFO,RHOGO,RTVO)
      RTVO=DFVOID(ART,WCIO-WBOO,WBOO,TSATO,RHOFO,RHOGO)
*  NET GRAVITY PRESS DROP(PSI) AVAILABLE FOR UNRECOV. DROP ACROSS CORE
      LDCX=LIMIT(0.,LDCR,LDCO)
      UNRECO=LDCX*RHOSCO/144. + (12.+LRT+LOP-LDCX)*RHOGO/144.-...
        RORTO*LRT/144.-ROPO*LOP/144.-ROCORO*12./144.
      DPO=LIMIT(.0001,1000.,UNRECO)
      WCAL=SQRT(DPO*RHOBO/(KLOSSU*RHOBR))
*  END ITERATIVE LOOP
*
*
*  DOWNCOMER AND LOWER PLENUM INITIALIZATION
*  VOLUME OF WATER IN DOWNCOMER NODE GIVEN BY FUNCTION
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*  VOLDC( ) AS A FUNCTICN OF WATER LEVEL
      MDCO=VOLDC(LDCO)*RHOSCO
      MHDCO=MDCO*HCIO
      MLPO=VOLP*RHCSCO
      MHLPO=MLPO*HCIO
*
      G=4.75*ART*RHOGO
      MTOTO=(LSCO*RSCAVO+LBO*RHOBO)*ACOR+LCP*ACOP*ROPO+LRT*ART*RORTO+G
*
*  PRESSURE CALCULATION INITIALIZATION
      RHOSVR=1./FUNGEN(VSATG,2,PRATED)
      UCSRV=WRATED/SQRT(PRATED*RHOSVR)
      MSTO=(VFREE-VOLDC(LDCO))*RHOGO
      UMSTO=MSTO*(HGO-PO*144./(778.*RHOGO))
*
*
*
*---------INITIALIZATION FOR SUPPRESSION POOL CALCULATION--------------
*
*
*                       CONSTANTS FOR CONTAINMENT MODEL
*
*  ADMET=HEAT TRANS AREA BET DW MET AND DW ATMOS(FT**2)
*  APMET=HEAT TRANSF AREA BET SP MET AND SP ATMOS
*  ASSPW=AREA OF POOL WATER SURFACE(SQ.FT.)
*  BDWSPO=CU.FT/SEC/PSI FLOW WHEN DOWNCOMERS CLEARED
*  BSPDWO=CU.FT/SEC/PSI OF FLOW WHEN VALVE OPEN
*  CDMET=MASS*SPEC HEAT OF DW METAL(BTU/DEG-F)
*  CPAIR=MASS*SPEC HEAT OF AIR IN SP CHAMBER ROOM
*  CPMET=MASS*SPEC HEAT OF SP METAL IN CONT WITH GAS
*  DM=AMOUNT OF STM. QUENCHED(UNIFORMLY ARCUND POOL) IF THERE
*  IS ELAPSED TIME BETWEEN NOMINAL START AND INITIALIZATION OF THE RUN
*  DPQR=RANGE(PSI) OVER WHICH QUENCH FRACTION GOES FROM 1 TO 0.
*   WITH INCREASING VAPOR PRESSURE
*  DPQZ=MARGIN(PSI) ABOVE SATURATION FOR COMPLETE QUENCHING
*  FDI,FSI=FRACTION OF POOL COOLING DISCHARGE AND SUCTION TO POOL NODE I
*  FWE=FLOW BETWEEN S.P. NODES(LB/SEC) PER INCH OF WATER LEVEL DIFFERENCE
*  FWI=FRACTION OF POOL WATER CONTAINED IN POOL NODE I
*  FSVI=FRACTION OF TOTAL SRV FLOW DISCHARGED TO POOL NODE I
*  FTEI=FRACTION OF TURBINE EXHAUST DISCHARGED TO POOL NODE I
*  FLSPG, FLDWG=FRACTION OF ATM. LEAKED TO RX-BLDG. PER SECOND
*  GCH=GAS CONSTANT OF H2
*  GCM=GAS CONSTANT OF M(MISC.)
*  GCN=GAS CONSTANT OF N2
*  GCS=GAS CONSTANT OF H2O VAP(PSI*FT**3/LB*DEG-R)
*  HSRREF=DELTA TO REREFERENCE STM ENTHALPY FROM ASME STEAM TABLES
*  TO PERFECT GAS EXP. THAT HAS H=0.0 AT 0 DEG-R
*  HSTE=ENTH OF TURBINE EXHAUST(BTU/LB)
*  HUMDWO= INITIAL DW GAS HUMIDITY (PERCENT)
*  HUMSPO=INITIAL SP GAS HUMIDITY (PERCENT)
*  KMIX=EMPIRICLE CONSTANT FOR NAT. CIRC MIXING FLOW BET. POOL NODES
*     LBASE=NCMINAL STARTING LEVEL OF POOL(IN. FROM INST. 0.)
*  MHSPGO,MHDWGO=INITIAL MASSES OF H(LBS)
*  MMSPGO,MMDWGO=INITIAL MASSES OF M (LBS)
*  PDCVP=PRESS. DIFF NECESSARY TO CLEAR THE VENT
*  PIPES FOR FLOW FRCM DW TO SP
*  PTDWGO= INITIAL TOTAL PRESS OF DW GAS (PSIA)
*  PTSPGO=INITIAL TOTAL PRESS OF SP GAS (PSIA)
*  QRVHLO=REACTOR(+PIPING) HEAT LOSSES (MW) FOR TEMP DIFF=DTRVHL(DEG-F)
*  TAUFDW=AVERAGE RESIDENCE TIME OF FOG IN DW(SEC)
*  TAUFSP=AVERAGE RES. TIME OF FOG IN SP
*  TAUVRV=TIME CONSTANT FOR SP VAC RELIEF VALVE TO LIFT(SEC)
*  TBASE=NCMINAL STARTING TEMP CF POOL(DEG-F)
*  TBII=PROVISION FOR BIASING STARTING TEMP CF POOL NODE I
*  TD=DISCH. TEMP(F) OF POOL COOLING FLOW
*  TDMETO=INITIAL DRYWELL METAL TEMP (F)
*  TGDWO=INITIAL DW GAS TEMP(DEG-F)
*  TGSPO=INITIAL SP GAS TEMP (DEG-F)
*  TPAIRO=INITIAL SP CHAMBER ROOM TEMP
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*  TPMETO=INITIAL TEMP : SP METAL IN CONTACT WITH SP GAS
*  VGDW=TOTAL FREE VOLUME OF DW (FT**3)
*  VTSP=TOTAL FREE VOLUME OF SP (FT**3)
*  WDLEAK=LEAK RATE(LB/SEC) OF SAT. WATER FROM RV TO DRYWELL
*  WDSPW=DISCHARGE FLOW OF POOL COOLING(LB/SEC)
*  WSSPW=SUCTION FLOW OF POOL COOLING(LB/SEC)
*
*
*  GEOMETRY AND PHYSICAL CHARACTERISTICS
      CONSTANT  ADMET=1.65E04,APMET=1.7E04,ASSPW=10860.
      CONSTANT  BDWSPO=2500.,BSPDWO=2000.
      CONSTANT  CDMET=8.33E04,CPAIR=3200.,CPMET=5.82E04
      CONSTANT  DPQR=2.,DPQZ=2.
      CONSTANT  GCH=5.361,GCM=.2436,GCN=.3829,GCS=.5955
      CONSTANT  HSTE=915.,HSRREF=-854.5,PDCVP=1.75
      CONSTANT  TAUVRV=3.,TAUFDW=30.,TAUFSP=15.
      CONSTANT  VGDW=159000.,VTSP=267600.
*
*  POOL NODALIZATION
      CONSTANT FD1=1.,FS1=1.,FW1=1.,FSV1=1.,FTE1=1.,FWE=5000.,KMIX=950.
*
*  INITIALIZATION
      CONSTANT  DM=0.,HUMDWO=18.4,HUMSPO=79.4,LBASE=10.23
      CONSTANT  MHSPGO=0.,MHDWGO=0.,MMSPGO=0.,MMDWGO=0.
      CONSTANT  PTDWGO=23.15,PTSPGO=23.65,TBASE=169.1,TBI1=0.
      CONSTANT TGDWO=268.,TGSPO=161.,TDMETO=212.,TPMETO=145.1,TPAIRO=155.
*
*  RUN CONTROL
      CONSTANT  FLDWG=0.,FLSPG=0.,TD=90.,QRVHLO=1.,DTRVHL=404.
      CONSTANT  WDLEAK=.68,WSSPW=0.,WDSPW=0.
      HD=TC-32.
*
*  INITIALIZATION CALC. FOR DW AND SP MASS AND ENERGY BALANCES
*  ASSUME ZERO INITIAL HYDROGEN AND MISC. GAS
*
*----------FUNCTION TABLES
*  FUNCTION SPLEV : SP LEVEL(INCHES FROM INSTRUMENT ZERO) AS A FUNCTION A
*  FUNCTION OF VOLUME(CU.FT)
      FUNCTION  SPLEV=0.,-182.,20222.,-134.,74646.,-62.,...
             117344.,-14.,130129.,0.,139300.,10.,...
             182563.,58.,242176.,130.,267611.,190.
*
*
*  FUNCTION STFOSV: SATURATION TEMP AS A FUNCTION OF SPECIFIC
*  VOLUME
      FUNCTION STFOSV=2.83,363.5,4.65,324.1,7.65,288.2,12.2,257.6,...
      20.1,228.,31.2,204.,44.7,185.6,76.4,160.5,158.9,129.6,...
           333.6,101.7,641.5,79.6,1235.,59.3
*
*  FUNCTION SPFOT : SATURATION PRESSURE AS A FUNCTION OF TEMPERATURE
      FUNCTION SPFOT=59.3,.25,79.6,.5,101.7,1.0,132.9,2.4,...
              160.5,4.8,185.6,8.5,203.9,12.5,...
              228.,20.,250.34,30.,288.2,56.,324.1,95.,363.5,160.,...
          381.8,200.,417.3,300.,444.6,400.,467.,500.,486.,600.,...
         503.,700.,525.,850.,544.6,1000.,556.,1100.,567.,1200.
*
*
*
*----------PRELIMINARY CALCULATIONS
      SPDWO=NLFGEN(SPFOT,TGDWO)
      SPSPO=NLFGEN(SPFOT,TGSPO)
      PSDWGO=(HUMDWO/100.)*SPDWO
      PSSPGO=(HUMSPO/100.)*SPSPO
*
      PNSPGO=PTSPGO-PSSPGO
      PNDWGO=PTDWGO-PSDWGO
*
*
      MNSPGO=PNSPGO*(VTSP-VWSPO)/(GCN*(TGSPO+460.))
      MNDWGO=PNDWGO*(VGDW)/(GCN*(TGDWO+460.))
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```
*
*  VWSPO=INITIAL TOTAL WATER VOL IN SP
*
      MSSPGO=PSSPGO*(VTSP-VWSPO)/(GCS*(TGSPO+460.))
      MSDWGO=PSDWGO*VGDW/(GCS*(TGDWO+460.))
*
*
      UMSPGO=(TGSPO+460.)*(MNSPGO*(.2475-.1851*GCN)+...
         MSSPGO*(.45-4.89/(TGSPO+460.)-.1851*GCS) )
      UMDWGO=(TGDWO+460.)*(MNDWGO*(.2475-.1851*GCN)+...
         MSDWGO*(.45-4.89/(TGDWO+460.)-.1851*GCS) )
*
*  DEPTH LBASE SPECIFIED: INCHES FROM INST ZERO(4 IN BELOW CENTER)
*  TBASE IS NOMINAL STARTING TEMP OF POOL
*  TBI(I) PROVIDES FOR BIASING INITIAL TEMPS IN MULTI-NODE MODEL
      VII=2.462*AII
      AII=(LBASE-4.)*SQRT(34580.-LBASE*LBASE+8.*LBASE)+...
          34596.*ARSIN( (LBASE-4.)/186. ) + 54343.
      MW1II=FW1*VII*(63.9-.019*(TBASE+TBI1))
      MII=MW1II
      DTBASE=(1190.-(TBASE-32.))*DM/(MII+DM)
      TIC1=TBASE+DTBASE+TBI1
      VWSPO=(MW1II+FW1*DM)/(63.9-.019*TIC1)
      VGSPO=VTSP-VWSPO
*
*
    DYNAMIC
*------------DYNAMIC PORTION OF REACTOR VESSEL CALCULATION--------------
*
*--------RUN CONTROLS FOR INJECTION CONTROL
      LDCVZ=LDC*12.+216.
      LDDEL=REALPL(500.,2.00,LDCVZ)
      RCICD=0.0
      HPCID=0.0
      WINJ=RCICMX*(RCICD+8.33*HPCID)*STEP(30.)
*---------RUN CONTROLS FOR VESSEL PRESSURE CONTROL
      POPEN=1120.
      PSHUT=1080.
*
*
*  CALCULATION OF DYNAMIC SATURATION PROPERTIES
      RHOF=1./(AFGEN(VSATF,P))
      RHOG=1./(FUNGEN(VSATG,2,P))
      HF=AFGEN(HSATF,P)
      HG=AFGEN(HSATG,P)
      TSAT=FUNGEN(TSATM,2,P)
*  INTERMEDIATE CALCULATIONS: AVERAGE HEAT FLUXES
*  'DKFUN' RETURNS DECAY HEAT AS A FUNCTION OF TIME SINCE SCRAM
*  'QREGAV' RETURNS AVERAGE NORMALIZED POWER, GIVEN LOCATION OF TOP
*  AND BOTTOM OF REGION
      T=TO+TIME
      TMIN=T/60.
*
       PREL=(DKFUN(T)+.94*EXP(-T/TCFUEL) )
*
      QTOT=PREL*CPO*947.8
      QFCOR=QTOT/(12.*PCOR)
      AVMSC=QREGAV(0.,LSCD)
      AVMB=QREGAV(LSCD,LSCD+LBD)
      QFSC=AVMSC*QFCOR
      QFB=AVMB*QFCOR
      LSCD=REALPL(LSCO,TAULEN,LSC)
      LBD=REALPL(LBO,TAULEN,LB)
*  DOWNCOMER ANNULUS CALCULATION
*  NEEDS: DENSITIES: RHODC,RHOLP
*         FLOWS INTO DOWNCOMER: WINJ, WRECIR, WINJAS
*         INJECTION ENTHALPY:  HINJIN
*         FLOWS OUT OF DOWNCOMER: WCI, WFLDC, WFLLP
*         REACTOR VESSEL PRESSURE: P
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*  RETURNS: DOWNCOMER HEIGHT(ABOVE BOTTOM OF ACTIVE FUEL) : LDC
*           ENTHALPY INTO LOWER PLENUM: HDC
*
*  INTERMEDIATE CALCULATIONS: MASS RATE OF ASPIRATION FROM STEAM SPACE
*  DUE TO INJECTION FLOW; FLASHING RATE FROM DC MASS; RECIRC FLOW RATE
      EQXX=EQXO*(LHEDER-LDC)/LHEDER
      EQX=LIMIT(0.,1.,EQXX)
      HINJFI=HINJIN+(HF-HINJIN)*EQX
      GP=7.23*WINJ
      WINTOT=INTGRL(0.,WINJ)
      GT=(WINTOT*7.481)/62.11
      WINJAS=(HF-HINJIN)*EQX*WINJ/(HG-HF)
      WFLDCX=MDC*FFLASH*(HDC-HF)
      WFLDC=LIMIT(0.,9999.,WFLDCX)
*
*  MASS AND ENERGY BALANCES(B1 IS A LOGIC SIGNAL TO STOP THE
*  INTEGRATION WHEN LDC.LT.0(BELOW BOTTOM OF ACTIVE FUEL)
      CMDC=INTGRL(0.,DMDC)
      MDC=MDCO+CMDC
      DMDC=B1*(WRECIR+WINJ+WINJAS-RHODC*(WCI+WFLLP)/RHOLP-WFLDC)+WEXPC
      CMHDC=INTGRL(0.,DMHDC)
      MHDC=MHDCO+CMHDC
      DMHDC=B1*(HF   *WRECIR+WINJ*HINJIN+HG*WINJAS-...
        HDC*(WCI+WFLLP)*RHODC/RHOLP-WFLDC*HG)
      HDC=MHDC/MDC
      RHODC=1./AFGEN(VSC,HDC)
*
*  DC LEVEL IS CALCULATED BY FUNCTION SUBPROGRAM XLENDC(VOLUME)
      VDC=MDC/RHODC
      LDC=XLENDC(VDC)
      LDL=LIMIT(0.,LDCR,LDC)
*  LOWER PLENUM CALCULATION
*  NEEDS: SAME AS DOWNCOMER
*  RETURNS:  ENTHALPY AT CORE INLET(HLP)
*  INTERMEDIATE CALCULATIONS:  LOWER PLENUM FLASHING RATE, LP WATER VOL
*  (VOLPW--USED FOR LOGIC CONTROL ONLY SINCE THE LOWER PLENUM IS
*  MODELED AS A CONSTANT VOLUME UNLESS DOWNCOMER IS EMPTY) :
      HLP=MHLP/MLP
      VOLPW=MLP/RHOLP
      RHOLP=1./AFGEN(VSC,HLP)
      XWFLLP=(HLP-HF)*MLP*FFLASH
      WFLLP=LIMIT(0.,9999.,XWFLLP)
*
*  LOGIC CONTROL
*
      X1B1=LDC-.001
      X2B1=VOLPW-1.01*VCLP
      B1N=NOR(X1B1,X2B1)
      B1=NOT(B1N)
*
*  MASS AND ENERGY BALANCES
*
      DMLP=(WCI+WFLLP)*(RHODC/RHOLP)*(B1)-WCI-WFLLP+...
        B1N*(WINJ+WINJAS)
      CMLP=INTGRL(0.,DMLP)
      MLP=MLPO+CMLP
      DMHLP=HDC*(WCI+WFLLP)*RHODC*B1/RHOLP+...
        B1N*(WINJ*HINJIN+HG*WINJAS)-WCI*HLP-WFLLP*HG
      CMHLP=INTGRL(0.,DMHLP)
      MHLP=MHLPO+CMHLP
*  CORE INLET FLOW CALC.(FORCE BALANCE)
*  NEEDS:  DENSITIES: RHODC,RHOLP,RSCAV,RHCB,ROP, RORT
*          REACTOR VESSEL PRESSURE : P
*          FLOW OUT OF BOILING REGION: WBO
*          REGION LENGTHS: LDC,LSC,LB
*  RETURNS: FLOW INTO CORE: WCI
*
     CMTOT=INTGRL(0.    ,(WCI-WRECIR-WTOST+WFLLP))
      MTOT=MTOTO+CMTOT
      WBO=(QFB*PCOR*LBD)/(HG-HF)
```

```
      WRECIR=LIMIT(0.,10000.,(WCI-WTOST)*FRECIR)+WCOLL
      WCOLL=LIMIT(0.,10000.,(LINT-30.)*3000.)
      FRECIR=LIMIT(0.,10.,(LINT-LBCT)/(LDCR-LBCT))
      LINT=LSCD+LB+LOPTP+LRTTP
      WFLOP=0.0
      WFLBR=0.0
      WFLRT=0.0
      MCOREM=12.*ACOR*RHOG
      MOPEM=LOP*ACOP*RHOG
      MRTEM=(LRT+4.75)*ART*RHOG
      WBRAV=WCIG
      RSCAV=(RHOF+RHOLP)/2.
      WCIG=REALPL(WCIO,3.0,WCI)
*   CALCULATE DENSITIES
      CAVIN=DFVOID(ACOR,WCIG,0.0   ,TSAT,RHOF,RHOG)
      CAVEX=DFVOID(ACOR,WCIG-WBO,WBO             ,TSAT,RHOF,RHOG)
      CAV=(CAVIN+CAVEX)/2.
      RHOB=RHOTP(RHOF,RHOG,CAV)
      OPV=DFVOID(ACOP,WCIG-WBO,WBO              ,TSAT,RHOF,RHOG)
      RHOOP=RHOTP(RHOF,RHOG,OPV)
      RTV=DFVOID(ART,WCIG- WBO,WBO                    ,TSAT,RHOF,RHOG)
      RHORT=RHOTP(RHOF,RHOG,RTV)
      BUVIN=DFVOID(ACOR,WCIG,WFLLP,TSAT,RHOF,RHOG)
      BUVEX=DFVOID(ACOR,WCIG-WBO,WTOST,TSAT,RHOF,RHOG)
      BUAVG=(BUVIN+BUVEX)/2.
      ROBUB=RHOTP(RHOF,RHOG,BUAVG)
*
      DELH1=LIMIT(1.,400.,HF-HLP)
      DLSCX=(-2.)*(LSCX*QFSC*PCCR-WCI*LIMIT(0.,400.,HF-HLP) )/...
          (RHOLP*ACOR*DELH1)
      LSCX= INTGRL(LSCO,DLSCX)
      LSC=LIMIT(0.,12.,LSCX)
      MSC=RSCAV*LSCD*ACOR
*
*   FIND THE TWO PHASE LENGTHS AND REGION MASSES
*
      LB=LIMIT(0.,12.-LSCD,LBU)
      LBU=(MTOT-MCOREM+MSCDUM-MOPEM-MRTEM-MSC)/(ACOR*(RHOB-RHOG))
      LCOV=LIMIT(0.,12.,LBCOVU+LSCD)
      LBCOVU=LBU*(RHOB-RHOG)/(ROBUB-RHOG)
      MSCDUM=ACCR*RHOG*LSCD
      MB=LB*RHOB*ACOR
      MCOR=MSC+MB+(12.-LB-LSCD)*RHOG*ACOR
    LOPTP=LIMIT(0.,LOP,(MTOT-MCCR-MRTEM-MOPEM)/(ACOP*(RHOOP-RHOG)))
      MOP=(RHOOP*LOPTP+(LOP-LOPTP)*RHOG)*ACOP
      MRT=MTOT-MOP-MCOR
      LRTTP=(MRT-MRTEM)/(ART*(RHORT-RHOG))
      WPOO=(LB*(1.-CAV)*ACOR+LOPTP*(1.-OPV)*ACOP+...
            LRTTP*(1.-RTV)*ART)*RHOF
*
*   TABULATE PRESS DROPS AND CALCULATE WCI
*
      PCOBOT=(MCOR/ACOR+MOP/ACOP+LRTLIM*RHORT+(10.-LRTLIM)*RHOG)/144.
      LRTLIM=LIMIT(0.,10.,LINT-17.52)
      LOSSIN=LDL*RHODC/144.+(LOCR-LDL)*RHOG/144.-PCOBOT
      XDUM=LIMIT(1.E-9,5.,LOSSIN)
      WCI=SQRT(XDUM*RHOB/(KLOSSU*RHOBR))
      WCIO=REALPL(WCIO,10.,WCI)
*
*   REACTOR VESSEL PRESSURE CALCULATION
*
*   STEAM MASS BALANCE
      CMST=INTGRL(0.,DMST)
      MST=MSTO+CMST
      DMST=WTOST+WEVAP+WFLDC-WEXPC-WCOND-WINJAS-WSTC
      WTOST=LIMIT(0.,1000.,WBO+WFLLP-WPOO)
      WCOND=0.0
      WEVAP=0.0
      WEXPC=LIMIT(0.,100.,1.67E-04*MST*(HGST-HST))
      HGST=AFGEN(HSATG,PST)
```

```
      WPOO=WPOO*DHF/(HG-HF)
      HFD=REALPL(HFO,2.0,HF)
      DHF=(HF-HFD)/2.
*
      VST=VFREE-VOC
      DVST=DERIV(0.,VST)
      VSVST=VST/MST
*  VST=TOTAL VAPOUR SPACE VOLUME
*  VSVST=SPECIFIC VOLUME OF STEAM
*
*
*  TOTAL ENERGY BALANCE
      CUMTO=INTGRL(0.,DUMTO)
      UMTO=UMSTO+CUMTO
      DUMTO=WTOST*HG+(WFLDC+WEVAP)*HG-...
      WEXPC*HF-WCOND*HG-WINJAS*HST-WSTC*HST-DVST*P*144./778.
      HST=UMTO/MST+PSTG*VSVST*144./778.
*  STEAM PRESSURE CALCULATED BY FUNCTION SUBPROGRAM PFVH--STEAM PRESSURE
*  AS A FUNCTION OF SPECIFIC VOLUME AND ENTHALPY
*  PFVH REQUIRES AN INITIAL GUESS OF PRESSURE, PSTG
      PST=PFVH(HST,VSVST,PSTG)
      PSTG=REALPL(PO,2.0,PST)
      P=PST
*
*  SAFETY RELIEF VALVE MODEL--TWO VALVES MODELED--WSSV=TOTAL SRV FLOW
      LOGREL=LOGV1+LOGV2
      LOGV1=REALPL(0.,.62,LOGPV1)
      LOGV2=REALPL(0.,.62,LOGPV2)
      LOGPV1=RST(X1V1,X2V1,0.)
      LOGPV2=RST(X1V2,X2V2,0.)
      X1V1=COMPAR(PSHUT,P)
      X1V2=COMPAR(POPEN-25.,P)
      X2V1=COMPAR(P,POPEN)
      X2V2=COMPAR(P,POPEN+25.)
      WSSV=UCSRV*SQRT(PSTG/VSVST)*LOGREL
*
*  WSTE=STEAM TURB EXHAUST FLOW(RCIC)
*  ENTHALPY IS ASSUMED=915. BUU/LB
      WSTE=(RCICD*(.00613*PST+1.1)+HPCID*(.0387*PST+7.81))*STEP(30.)
*  WSTC IS TOTAL STEAM FLOW RATE FROM REACTOR VESSEL TO CONTAINMENT
      WSTC=WSTE+WSSV
*
*-----------DYNAMIC PORTION OF CONTAINMENT CALCULATION-------------------
*
*  CALLS TO SP WATER MACRO--ONLY ONE NECESSARY FOR THE SINGLE NODE POOL MODEL
      XNQ1 ,TP1 ,WE1 ,MW1 ,LW1 =W(TP1,...
         TP1 ,LW1,LW1 ,TIC1 ,FW1 ,FSV1 ,FTE1 ,FD1 ,FS1 )
      VWSP=MW1/(63.9-.019*TP1)
      MWSP=MW1
      GWSP=7.481*VWSP
*
      TWSPAV=(63.9-MWSP/VWSP)/.019
       LWSPAV=AFGEN(SPLEV,VWSP)
*
*  INTERFACE THE SP WATER TO THE SP GAS
*
      XNQDW=XNQ1
      WSSVNQ=WSSV*FSV1*XNQ1
      WSTENQ=WSTE*FTE1*XNQ1
      WTESPW=WE1
*
*
*  INTERFACE VARIABLES : RV TO SUPPRESSION POOL
*  WSSV=STM. FLOW FROM RV THRU RELIEF VALVES(LB/SEC)
*  WSTE=STM  FLOW FROM RV THRU TURBINES(RCIC+HPCI)
*  WHSV=HYDROGEN FLOW FROM RV THRU RELIEF VALVES(LB/SEC)
*  HHM=HYDROGEN ENTHALPY AT MIXED RV TEMP(BTU/LB) (REF TO 0. DEG R)
*  HSTE=ENTHALPY OF TURBINE EXHAUSE(ASME)
*  TM=MIXTURE TEMP OF RV STEAM SPACE
*
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*  INTERFACE VARIABLES : RV TO DRYWELL
*  WSDWR=STEAM FLOW, RELEASE TO DRYWELL(LB/SEC)
*  HSDWR=ENTHALPY OF WSDWR(ASME)
*  WHDWR=HYDROGEN FLOW, RELEASE TO DRYWELL(LB/SEC)
*  HHDWR=ENTHALPY OF WHDWR(REF TO 0.0 DEG R)
*  WMDWR=MISC. FLOW RELEASED TO DRYWELL
*  HMDWR=ENTHALPY OF WMDWR(0.0 DEG R)
*
*  INTERFACE VARIABLES : SP WATER TO SP GAS
*  WSSVNQ=TOTAL NON-QUENCHED RELIEF VALVE FLOW FROM RV TO SP VIA RV'S
*  WSTENQ=TOTAL NON-QUENCHED TURBINE EXHAUST FLOW
*  WCSPG=TOTAL CONSECSATE FLOW, SPG TO SPW
*  WTESPW=TOTAL EVAPORATION RATE, SPW TO SPG
*  HSTSPW=ENTHALPY OF STEAM(REF O. AT O. DEG) AT MASS-WEIGHTED SP WATER
*        TEMP, TWSP
*  HCSPG=ENTHALPY OF CONDENSED SP STEAM(ASME)
*
*  MASS BALANCES FOR SP GAS AND DW GAS
*
*  INPUT FROM OTHER CALCULATIONS:
*  VWSP=SP TITAL WATER VOL (FT**3) (VARIABLE)
*  WSSNQ= TOTAL NON-QUENCHED STEAM FLOW (LB/SEC)
*  WHSV= H FLOW FROM THE RV (THROUGH RELIEFS)
*  WSDWR= FLOW RATE OF STEAM RELEASED DIRECT TO DRYWELL
*  WHDWR= FLOW RATE OF HYDROGEN RELEASED TO DRYWELL
*  WMDWR= FLOW RATE OF MISC. RELEASED DIRECT TO DRYWELL
*
*
*  DEFINITIONS
*
*  MSSPG= MASS STM IN SPG
*  MHSPG= MASS H IN SPG
*  MNSPG= MASS N IN SPG
*  MMSPG=MASS M IN SPG
*
*  FLOWS = LB/SEC UNLESS OTHERWISE SPECIFIED
*  WSSVNG= TOTAL NON-QUENCHED STM FROM SV'S ANT TURB, EXHAUST
*  WHSV= TOTAL HYDROGEN FLOW FROM RELIEF VALVES
*  DWSP= DESIGNATES FLOW FROM DW TO SP
*  SPDW= DESIGNATES FLOW FROM SP TO DW
*  SPL= SP GAS LEAKAGE
*  B= BULK FLOW RATE (FT**3/SEC)
*  WCSPG= TOTAL CONDENSATE FLOW FROM SP GAS
*  WCDWG= TOTAL CONDENSATE FLOW FROM DW GAS
*
*
*  SP GAS MASS BALANCE:
 MSSPG=INTGRL(MSSPGO,WSSVNQ+WSDWSP+WSTENQ+WTESPW-WSSPDW-WSSPL-WCSPG)
      MHSPG=INTGRL (MHSPGO,WHSV+WHDWSP-WHSPDW-WHSPL)
      MNSPG= INTGRL(MNSPGO,WNDWSP-WNSPL-WNSPDW)
      MMSPG=INTGRL (MMSPGO,WMDWSP-WMSPL-WMSPDW)
*
      BSPL=VGSP*FLSPG
      WSSPL=BSPL*(MSSPG/VGSP)
      WHSPL=BSPL*(MHSPG/VGSP)
      WNSPL=BSPL*(MNSPG/VGSP)
      WMSPL=BSPL*(MMSPG/VGSP)
*  BSPL=BULK LEAKAGE FROM SP GAS
*  FLSPG=FRACTION OF TOTAL SP GAS VOLUMES LEAKED PER SECOND
*  VGSP=VOLUME OF SP GAS, VWSP=SP WATER VOLUME
      VGSP=VTSP-VWSP
*
*  NOTE: 0.5 PSI IS NECESSARY TO OPEN VAC RELIEF VALVE
      BSPDWX=LIMIT(0.,1.E06,BSPDWO*(PTSPG-PTDWG-.5))
      BSPDW=REALPL(0.,TAUVRV,BSPDWX)
      WSSPDW=BSPDW*(MSSPG/VGSP)
      WHSPDW=BSPDW*(MHSPG/VGSP)
       WNSPDW=BSPDW*(MNSPG/VGSP)
      WMSPDW=BSPDW*(MMSPG/VGSP)
*  BSPDWO= (FT**3/SEC)/PSI OF PRESS DIFFERENCE WHEN
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*  VAC RELIEF VALVE IS OPEN
*
      BDWSPX=LIMIT (0.,1.E06,BDWSPO*(PTDWG-PTSPG-PDCVP))
      BDWSP=REALPL(0.,TAUVRV,BDWSPX)
      WSDWSP=BDWSP*(MSDWG/VGDW)*XNQDW
      WHDWSP=BDWSP*(MHDWG/VGDW)
      WNDWSP=BDWSP*(MNDWG/VGDW)
      WMDWSP=BDWSP*(MMDWG/VGDW)
*  BDWSPO=(FT**3/SEC)/PSI WHEN DW PRESS IS GREAT ENOUGH
*  TO CLEAR THE VENT PIPE DOWNCOMERS
*  PDCVP=PRESS DIFF. NECESSARY TO CLEAR THE VENT PIPES
*
      WCSPG=WWCSPG+WVCSPG
      WVCSPG=(MSSPG/TAUFSP)*(1.-100./HUMSP)*COMPAR(HUMSP,100.)
*  WWCSPG=RATE OF COND. ON SPG WALL
*  WVCSPG=RATE OF COND. FROM SPG VOLUME
*
*  DRYWELL MASS BALANCE:
      MSDWG=INTGRL(MSDWGO,WSDWR+WSSPDW-WSDWSP-WSDWL-WCDWG)
      MHDWG=INTGRL(MHDWGO,WHDWR+WHSPDW-WHDWL-WHDWSP)
      MNDWG=INTGRL(MNDWGO,WNSPDW-WNDWSP-WNDWL)
      MMDWG=INTGRL(MMDWGO,WMDWR+WMSPDW-WMDWSP-WMDWL)
*
      WCDWG=WWCDWG+WVCDWG
      WVCDWG=(MSDWG/TAUFDW)*(1.-100./HUMDW)*COMPAR(HUMDW,100.)
*
*  HYDROGEN AND MISC RELEASES SET TO ZERO
*  FFRACT=FRACTION OF HOT LIQUID LEAK THAT FLASHES TO STEAM
      FFRACT=(AFGEN(HSATF,PST)-AFGEN(HSATF,PTDWG))/...
          (AFGEN(HSATG,PTDWG)-AFGEN(HSATF,PTDWG))
      WSDWR=WDLEAK*FFRACT
      HSDWR=AFGEN(HSATG,PTDWG)
      WHDWR=0.
      WMDWR=0.
      HHDWR=0.
      HMDWR=0.
*
* WWCDWG=RATE OF COND. ON DWG WALL
*  WVCDWG=RATE OF VOLUME COND. IN DRYWELL
*
*  ENERGY BALANCES FOR DW AND SP GAS
*
*  INPUT FROM OTHER CALCULATIONS:
*  HMDWR=ENTHALPY OF MISC. RELEASED DIRECT TO DRYWELL
*  HHDWR=ENTHALPY OF HYDROGEN RELEASED DIRECT TO DRYWELL
*  HHRV=ENTHALPY OF HYDROGEN IN REACTOR VESSEL
*  HSRV=ENTHALPY OF STEAM IN REACTOR VESSEL
*  HSDWR=ENTHALPY OF STEAM RELEASED DIRECT TO DRYWELL
*  HSTE=ENTHALPY OF STEAM FROM TURBINE EXHAUST
*  NOTE THAT THE ABOVE 3 ARE REF. TO ZERO AT 32.F WATER
*  (ASME STM TABLES), BUT ARE RE-REF TO ZERO
*  AT ZERO DEG-R FOR CONTAINMENT CALC.
*  QVSPG,QVDWG=VOLUME HEAT SOURCE (FROM F.P.'S)
*  QRVHL=HEAT LOSS (THROUGH INSULATION) FROM RV TO SP GAS
*  QLSPG=HEAT LOSS FROM SP GAS TO SP LINER
*  QLDWG=HEAT LOSS FROM DW GAS TO DW LINER
*
*  DEFINITIONS:
*
*  TGSP=TEMP OF HOMOGENIZED SP GAS (DEF-F)
*  TGDW=TEMP OF  HOMOGENIZED DW GAS (DEG-F)
*  UMSPG=TOTAL INTERNAL ENERGY OF SP GAS (BTU)
*  UMDWG=TOTAL INTERNAL ENERGY OF DW GAS (BTU)
*  MASS FLOWS DEFINED IN MASS BALANCE SECTION
*
*  SP GAS ENERGY BALANCE:
      CUMSPG=INTGRL(0.0,DUMSPG)
      UMSPG=UMSPGO+CUMSPG
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```
 DUMSPG=EPSSP+EPHSP+EPNSP+EPMSP+QVSPG-QLSPG-MWORK+QSSPG
      EPSSP=WSSVNQ*(HSSVNQ)+WSTENQ*(HSTE+HSRREF)+...
      WSDWSP*(HSTGDW)-(WSSPDW+WSSPL)*HSTGSP-...
      WCSPG*(HFGSP+HSRREF)+WTESPW*HSTSPW
      MWORK=DVDTPG*PTSPG*.1851
      DVDTPG=(VGSP-VGSPD)/5.
      VGSPD=REALPL(VGSPO,5.,VGSP)
*  WSSVNQ=TOTAL SRV FLOW THAT EXITS SURFACE OF POOL
*  WSTENQ=TOTAL NON-QUENCHED STM FROM TURBINE EXHAUST
*  HSRREF=DELTA-H TO REREF. STEAM ENTHALPY
*  HFGDW=SAT FLUID ENTH AT DW TEMP(ASME)
*  HFGSP=SAT FLUID ENTH AT SP TEMP(ASME)
*  HSSVNQ=ENTHALPY OF NON-QUENCHED STEAM ENTERING SP GAS(REF TO 0.
*         AT 0. DEG-R)
      HSSVNQ=HST+HSRREF
      HSTGSP=.45*TGSPR-4.89
      HSTSPW=.45*(TWSPAV+460.)-4.89
      HSTGDW=.45*TGDWR-4.89
*
*
      HFGSP=TGSP-32.
      HFGDW=TGDW-32.
*
      EPHSP=WHDWSP*HHTGDW+WHSV*HHRV-(WHSPDW+WHSPL)*HHTGSP
      HHTGSP=3.466*TGSPR-40.
      HHTGDW=3.466*TGDWR-40.
*
      EPNSP=HNTGDW*WNDWSP-(WNSPDW+WNSPL)*HNTGSP
      HNTGSP=.2475*TGSPR
      HNTGDW=.2475*TGDWR
*
      EPMSP=HMTGDW*WMDWSP-(WMSPDW+WMSPL)*HMTGSP
      HMTGSP=.21*TGSPR-20.8
      HMTGDW=.21*TGDWR-20.8
*
*  DRYWELL ENERGY BALANCE
      CUMDWG=INTGRL(0.0,DUMDWG)
      UMDWG=CUMDWG+UMDWGO
      DUMDWG=EPSDW+EPHDW+EPNDW+EPMDW+QVDWG+...
      QRVHL-QLDWG
*
      QSSPG=(TWSPAV-TGSP)*ASSPW*5.3E-05*((ABS(TWSPAV-TGSP))**.33)
      QLSPG=QPMWET+QPMDRY
      QVDWG=0.
      QVSPG=0.
      QRVHL=QRVHLO*948.*(TSAT-TGDW)/DTRVHL
      QLDWG=QDM
*
*
      EPSDW=(HSDWR+HSRREF)*WSDWR+HSTGSP*WSSPDW-HSTGDW*(WSDWSP+WSDWL)-..
      HFGDW*WCDWG
*
      EPHDW=HHDWR*WHDWR+HHTGSP*WHSPDW-HHTGDW*(WHDWSP+WHDWL)
*
      EPNDW=HNTGSP*WNSPDW-HNTGDW*(WNDWSP+WNDWL)
*
      EPMDW=HMDWR*WMDWR+HMTGSP*WMSPDW-HMTGDW*(WMDWSP+WMDWL)
*
*
*  SOLUTION FOR DW AND SP GAS TEMPERATURES
*  ITERATION IS NOT NECESSARY SINCE THE S,N,H,AND M
*  ENTHALPIES ARE ASSUMED LINEAR WITH TEMP:
*  H(N2)=.2475T, H(H2O)=.45T-4.89, H(M)=.21T-20.8, H(H)=3.466T-40.
*
      TGSPR=(UMSPG+4.89*MSSPG+20.8*MMSPG+40.0*MHSPG)/...
          (MNSPG*(.2475-.1851*GCN)+MSSPG*(.45-.1851*GCS)+...
          MMDWG*(.21-.1851*GCM)+MHSPG*(3.466-.1851*GCH) )
      TGDWR=(UMDWG+4.89*MSDWG+20.8*MMDWG+40.*MHDWG)/...
      (MNDWG*(.2475-.1851*GCN)+MSDWG*(.45-.1851*GCS)+...
      MMDWG*(.21-.1851*GCM)+MHDWG*(3.466-.1851*GCH))
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```
      TGSP=TGSPR-460.
      TGDW=TGDWR-460.
*   TOTAL AND PARTIAL PRESSURES(PSIA) CALC. FROM TEMP(F)
      PNSPG=MNSPG*GCN*TGSPR/VGSP
      PHSPG=MHSPG*GCH*TGSPR/VGSP
      PSSPG=MSSPG*GCS*TGSPR/VGSP
      PMSPG=MMSPG*GCM*TGSPR/VGSP
      PTSPG=PNSPG+PHSPG+PSSPG+PMSPG
*
      PNDWG=MNDWG*GCN*TGDWR/VGDW
      PHDWG=MHDWG*GCH*TGDWR/VGDW
      PSDWG=MSDWG*GCS*TGDWR/VGDW
      PMDWG=MMDWG*GCM*TGDWR/VGDW
      PTDWG=PNDWG+PHDWG+PSDWG+PMDWG
*
*   HUMIDITIES AND DEWPOINTS
      HUMSP=100.*PSSPG/NLFGEN(SPFOT,TGSP)
      HUMDW=100*PSDWG/NLFGEN(SPFOT,TGDW)
      TDEWSP=NLFGEN(STFCSV,VGSP/MSSPG)
      TDEWDW=NLFGEN(STFCSV,VGDW/MSDWG)
*
*
      BDWL=VGDW*FLDWG
      WSDWL=BDWL*(MSDWG/VGDW)
      WHDWL=BDWL*(MHDWG/VGDW)
      WNDWL=BDWL*(MNDWG/VGDW)
      WMDWL=BDWL*(MMDWG/VGDW)
*
*   CALCULATION OF SP AND DW WALL METAL TEMP ASSUMPTIONS:
*      1.CONSIDER ONLY METAL SURFACE IN CONTACT WITH GAS
*      2.NO CONDENSING UNLESS METAL TEMP BELOW DEWPOINT
*   STM. CONDENSATION ON WALLS IS AIR-LIMITED AND IS CALCULATED
*   FROM ECUN. III.8.26 OF MARCH MANUAL(NUREG/CR-1711)
      QPM=QPMDRY+QPMWET-QPGAIR
      QPMDRY=(TGSP-TPMET)*5.3E-05*((ABS(TGSP-TPMET))**.3333)*APMET
   QPMWET=COMPAR(TDEWSP,TPMET)*(TDEWSP-TPMET)*APMET*...
         .0185*((MSSPG/MNSPG)**.707)
      TPMET=INTGRL(TPMETO,QPM/CPMET)
      QDM=QDMDRY+QDMWET
      QDMDRY=(TGDW-TDMET)*5.3E-5*((ABS(TGDW-TDMET))**.3333)*ADMET
      QDMWET=COMPAR(TDEWDW,TDMET)*(TDEWDW-TDMET)*ADMET*...
          .0185*((MSDWG/MNDWG)**.707)
      TDMET=INTGRL(TDMETO,QDM/CDMET)
*
*   CALC OF CONDENSATION RATE ON DW AND SP WALLS. APPROXIMATE VALUE
*   OF 900 BTU/LB IS USED FOR (HG-HF)
      WWCSPG=QPMWET/900.
      WWCDWG=QDMWET/900.
*
*   CALCULATION OF POOL ROOM AIR TEMP
      QPAIR=QPGAIR+QPWAIR
      QPWAIR=5.3E-05*((ABS(TWSPAV-TPAIR))**.3333)*APMET*(TWSPAV-TPAIR)
      QPGAIR=5.3E-5*((ABS(TPMET-TPAIR))**.333)*APMET*(TPMET-TPAIR)
      TPAIR=INTGRL(TPAIRO,QPAIR/CPAIR)
*
*   NOTE THAT THE TERM APMET IS THE SAME IN BOTH EQUATIONS
*   BECAUSE IT IS FOR ONE HALF OF TOTAL METAL SURFACE AREA
*
*   THESE CALCULATIONS ARE TO BE USED FOR AVERALL THERMO
*   CONSERVATION CHECK
      DMSP=INTGRL(0.,WSTC)
      HEATIN=INTGRL(0.,QTOT)
      DMHRV=INTGRL(0.,WINJ*HINJIN-WSTC*HST)
*
      LINTVZ=LINT*12.+216.
   PRINT LDCVZ,LSC,LB,P,WTOST,WSTC,TWSPAV,LWSPAV,GP,TGSP,TGDW,WCID
     TIMER  DELT=.25, FINTIM=20.000, PRDEL=10.
      METHOD RECT
      NOSORT
```

```
      CALL DEBUG(1,0.0)
      CALL CEBUG(1,3600.)
      CALL DEBUG(1,7600.)
      CALL DEBUG(1,1C800.)
      CALL CEBUG(1,14400.)
      CALL CEBUG(1,17999.)
      END
      STOP
```

APPENDIX B

MODIFICATION TO MARCH SUBROUTINE ANSQ

In the MARCH code, the calculation of decay heating is carried out in subroutine ANSQ. For this study, the calculation was modified to include the actinide decay heat source in a BWR following a burnup of 34,000 MWd/τ. A listing of the revised subroutine follows:

```
      Subroutine ANSQ (ANS·TIME·TAP)
C
C      ANSQ calculates the decay heat using ANS correla-
C      tions.

      ANS1=ANS2=0.0
      TVAR=TIME*60.0
      IF (TVAR·GT·10.0) GO TO 10
      ANS1=0.06950—0.001592*(ALOG(TVAR))
      IF (TVAR·LT·3.) ANS1=1.
      GO TO 40
10    IF (TVAR·GT·150.0) GO TO 20
      ANS1=0.069241—0.0069355*(ALOG(TVAR))
      GO TO 40
20    IF (TVAR·GT·4.E6) GO TO 30
      ANS1=4.0954E-2—2.8324E-3*(ALOG(TVAR))
      GO TO 40
30    ANS1=4.6893E-3-2.3800E-4*(ALOG(TVAR))
40    TVAR=(TIME+TAP)*60.0
      IF (TVAR·GT·10.0) GO TO 50
      ANS2=6.950E-2—1.592E-3*(ALOG(TVAR))
      GO TO 80
50    IF (TVAR·GT·150.0) GO TO 60
      ANS2=6.9241E-2—6.9355E-3*(ALOG(TVAR))
      GO TO 80
60    IF (TVAR·GT·4.E6) GO TO 70
      ANS2=4.0954E-2—2.8324E-3*(ALOG(TVAR))
      GO TO 80
70    ANS2=4.6893E-3—2.3800E-4*(ALOG(TVAR))
80    ANS-ANS1—ANS2
      RETURN
      END
```

APPENDIX C

MARCH INPUT FOR ACCIDENT SEQUENCE TB′

```
 BROWNS FERRY CSB + HPCI/RCIC
$NLMAR
 ITRAN=1,
 IBRK=0,
 ISPRA=1,
 IECC=2,
 IBURN=0,
 NINTER=100,
 IPDTL=7,
 IPLOT=3,
 IU=3,
 VOLC=416700.0,
 DTINIT=0.01,
 TAP=2.62E06,
$END

$NLINTL
$END

STEEL     CONCRETE
DRYWELL1  DRYWELL2  CONC SHELLMISC STEELMISC CONC.
C
 $NLSLAB
  NMAT=2,
  NSLAB=3,
  NOD=1,4,13,
  DEN(1)=486.924,157.481,
  HC(1)=.1137,.23817,
  TC(1)=25.001,.80024,
  IVL=1,1,2,
  IVR=1,1,2,
  NNO1=3,9,4,
  MAT1=1,2,1,
  MAT2=1,2,1,
  SAREA=18684.,5358.,15982.,
  X(1)=0.,.01,.02083, X(4)=0.,.01,.03,.07,.15,.31,.63,1.27,2.5,
  X(13)=0.,.01,.03,.0625,
  TEMP=12*150.,4*95.,
 $END

$NLECC
 PUHIO=0.001,
 UHIO=3.11E04,
 PACMO=0.001,
 ACMO=3.11E04,
 PHH=1120.,
 WHH1=-5000.0,
 PSIS=1120.,
 WSIS1=0.0,
 PLH=1120.,
 WLH1=0.0,
 STPHH=240.,
 RWSTM=3.11E06,
 ECCRC=0.64,
 CSPRC=1.0,
 DTSUB=-100.,
 WTCAV=100.,
 TRWST=95.0,
$END

 $NLECI
 $END

 $NLCSI
 $END
```

```
$NLCOOL
$END

$NLMACE
 NCUB=2,
 NRPV1=2,
 NRPV2=1,
 ICECUB=-1,
 DTPNT=20.0,
 IDRY=-1,
 IWET=2,
 WPOOL=7.801E06,
 TPOOL=95.0,
 VDRY=3.839E03,
 VTORUS=257700.0,
 WVMAX=5.146E05,
 NSMP=-2,
 NSMP2=2,
 NCAV=-1,
 VCAV=4789.1,
 VFLR=15.0,
 AVBRK=292.0,
 CVBRK=4.04,
 VC(1)=159000.0,257700.0,
 AREA(1)=1.6399E03,1.098E04,
 HUM(1)=0.2,1.0,
 TEMPO(1)=150.,95.,
 N=10,
 NS(1)=1,1,1,3,3,2,2,2,2,2,
 NC(1)=1,1,1,1,1,1,1,2,2,2,
 NT(1)=1,2,3,-7,-7,-7,-7,-7,-7,
 C1(1)=1.E6,1.E6,0.,400.,500.,139.7,189.7,139.7,189.7,159.7,
 C2(1)=0.,1.333E5,7.59106,5.9297,.583,5.9297,.583,5.9297,.583,.583,
 C3(1)=95.,0.,1192.5,.00694,20.97,.00694,20.97,6.94E-4,.0833,6.94E-3,
 C4(1)=0.0,0.0,0.0,14.7,0.0,14.7,0.0,14.7,14.7,14.7,
 KT(1,2)=1,
 KT(2,1)=1,
 STPECC=240.,
$END

$NLBOIL
 NNT=37436,
 NR=35908,
 NDZ=50,
 ISTR=3,
 ISG=0,
 IMWA=1,
 IHR=1,
 WDED=3.50E04,
 QZERO=1.1242E10,
 H=12,
 HO=28.,
 DC=15.59,
 ACOR=104.833,
 ATOT=287.898,
 WATBH=97000.,
 D=.04692,
 DF=.04058,
 DH=0.056,
 CLAD=.005594,
 XOO=8.33E-06,
 RHOCU=68.783,
 TGOO=546.,
 PSET=1050.0,
 CSRV=3380.,
 FDCR=-.5,
```

```
DPART=0.0208333,
 FZMCR=0.05,
 FZOCR=0.08,
 FZOS1=0.1,
 WFE2=7992.,
 TFEOO=546.,
 FULSG=0.0,
 PVSL=1000.,
 TCAV=1210.,
 ABRK=0.0,
 YBRK=45.0,
 DTPNTB=5.0,
 DTPN=-5.,
 VOLP=2.459E04,
 VOLS=9.638E03,
 WCST=3.11E06,
 F(1)=0.1,0.25,0.47,0.[illegible],0.84,0.96,1.13,1.27,1.295,1.27,1.24,1.21,1.195,
 F(14)=1.15,1.11,1.08,1.05,1.03,1.02,1.016,1.017,1.05,1.06,1.061,1.062,
 F(26)=1.061,1.06,1.059,1.059,1.06,1.07,1.075,1.095,1.11,1.12,1.185,1.215,
 F(38)=1.25,1.26,1.24,1.21,1.15,1.09,1.0,0.87,0.76,0.6,0.41,0.21,0.1,
 PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,0.9665,0.408,
 VF(1)=10*0.1,
 TT=6*546.,
 CM=1824.,7992.,8760.,2460.,5550.,24900.,
 AH=740.,263.,9225.,400.,7000.,700.,
 DD=1.,1.,1.,.17,.02,.546,
 AR=150.,263.,165.,0.,-10.,-20.,
$END

$NLHEAD
 WZRC=140397.0,
 WFEC=30447.73,
 WUO2=361837.0,
 WGRID=66750.,
 WHEAD=175927.08,
 DBH=20.915,
 THICK=0.52198,
 COND=8.0005,
 E1=.8,
 E2=.5,
$END

$NLHOT
 IHOT=100,
 DP=0.25,
 FLRMC=3360.,
$END

$NLINTR
 CAYC=0.01524,
 CPC=1.30,
 DENSC=2.375,
 TIC=308.16,
 FC1=0.441,
 FC2=0.108,
 FC3=0.357,
 FC4=0.027,
 RBR=0.135,
 R0=322.6,
 R=6000.0,
 HIM=0.2,
 HIO=0.09,
 WALL=1000.,
$END
```

Appendix D
PRESSURE SUPPRESSION POOL MODEL

D.1 Introduction

The primary containment of each of the Browns Ferry Nuclear Plant units is a Mark I pressure suppression pool system. The safety objective of the Mark I containment is to provide the capability, in the event of an accident, to limit the release of fission products to the environment.[29] The *key* to the safety objective of this system lies in the performance of the pressure suppression pool (PSP). The PSP is designed to rapidly and completely condense steam released from the reactor pressure vessel, to contain fission products released from the vessel, and to serve as a source of water for the emergency core cooling systems. If the PSP should fail to properly perform during an accident sequence, then there is a high probability that fission products will be released to the environment.

In the past few years, a major concern of the NRC, the BWR vendor, and the utilities who operate the early generation BWRs has been scenarios[30] where it has not been conclusively shown that the PSP will meet the performance requirements stated above. Specifically, there is a lack of information on condensation oscillations and the resulting loads they place on the PSP. In addition, there is the question of pressure suppression pool thermal stratification and the resulting impact of this phenomenon on the condensing ability of the PSP.

The problem of condensation oscillations* includes the analysis of both loads which originate at the downcomer exit during a large break loss of coolant accident (LOCA) and loads derived from the safety relief valve (SRV) discharge during normal blowdown to the PSP. The condensation oscillation problem involves coupling an analysis of the momentum transport within the PSP to a stress analysis of the torus walls.

The problem of pool thermal stratification is of interest because thermal layering of the water near a SRV discharge point can limit the ability of the PSP to condense the steam. Subsequently, this can lead to increased intensity of the condensation oscillation loads and possibly to overpressurization of the torus. Furthermore, the temperature distribution in the torus determines (to some extent) the distribution of fission products in the PSP during an accident. Thus, an analysis of fission product transport to the environment will depend on an analysis of the temperature distribution in the PSP.

D.2 Purpose and Scope

The purpose of this work is to study the dynamics of the Mark I pressure suppression pool (PSP). Knowledge of the response of the PSP is

---

*In the literature, when the condensation instability occurs outside the relief valve tailpipe it is termed "condensation oscillation." If it occurs inside the pipe, the phenomenon is called "chugging." Here, "condensation oscillations" includes both of these effects.

desired for transients ranging from a normal SRV discharge to a full-blown, large break LOCA. The primary objective of this study is to improve the available BWR pressure suppression pool analysis techniques. A secondary objective is to learn as much as possible about the complex phenomena involved in each of the transients (phenomena such as pool swell, condensation oscillation, chugging, and thermal stratification).

An analysis of the PSP dynamics will necessarily involve accounting for the two phase flow of steam jets into subcooled water and the ensuing transport of mass, momentum, and energy by the mechanism of condensation. Because of the complex geometry of the PSP, (toroidal geometry plus large, submerged, complicated flow obstructions plus an air-water interface) the analysis tool used in this study will be a state-of-the-art, 3-D thermal hydraulics code.

Pending completion of this work, it will be necessary to assume pool-averaged conditions for accident analyses.

D.3 Description of the System

The Mark I containment system consists of the drywell, the pressure suppression pool, the vent system connecting the drywell and PSP, a containment cooling system, isolation valves, and various service equipment. Figure D-1 shows the arrangement of the drywell, PSP, and vent system within the Reactor Building.

The drywell is a steel pressure vessel with a spherical lower portion and a cylindrical upper portion. It is designed for an internal pressure of 0.531 MPa (62 psig) at a temperature of 138°C (281°F). Normal environment in the drywell during plant operation is an inert atmosphere of nitrogen at atmospheric pressure and a temperature of about 57°C (135°F).

The vent system consists of 8 circular vent pipes which connect the drywell to the PSP. The vent pipes are designed to conduct flow from the drywell to the PSP (in the event of a LOCA) with minimum resistance, and to distribute this flow uniformly in the pool. The vent pipes are designed for an internal pressure of 0.531 MPa (62 psig) with a temperature of 138°C (281°F); they are also designed to withstand an external pressure of 0.014 MPa (2 psi) above internal pressure.

The pressure suppression pool is a toroidal shaped steel pressure vessel located below the drywell. The PSP contains about 3823 m^3 (135,000 ft^3) of water and has an air space above the water pool of 3370 m^3 (119,000 ft^3). Inside the PSP, extending around the circumference of the torus, is a 1.45 m (4.75 ft) diameter vent header. The 8 drywell vents connect to this vent header. Projecting down from the vent header are 96 downcomer pipes which terminate 1.22 m (4 ft) below the surface of the water. At 13 unevenly distributed positions around the PSP, discharge lines from the safety relief valves extend through the vent pipes and terminate in a T-quencher device located near the bottom of the pool. Figure D-2 shows a cross section of the PSP and the relative locations of the vent pipe, vent header, downcomer, SRV discharge line, and the T-quencher, which has been rotated 90° for the purpose of illustration. Near the bottom of the PSP, a 0.762 m (30-in.) suction header (ring header) circumscribes the torus and connects to the pool at four locations. At Browns Ferry, the RHR, HPCI, core spray, and RCIC systems are supplied from this header.

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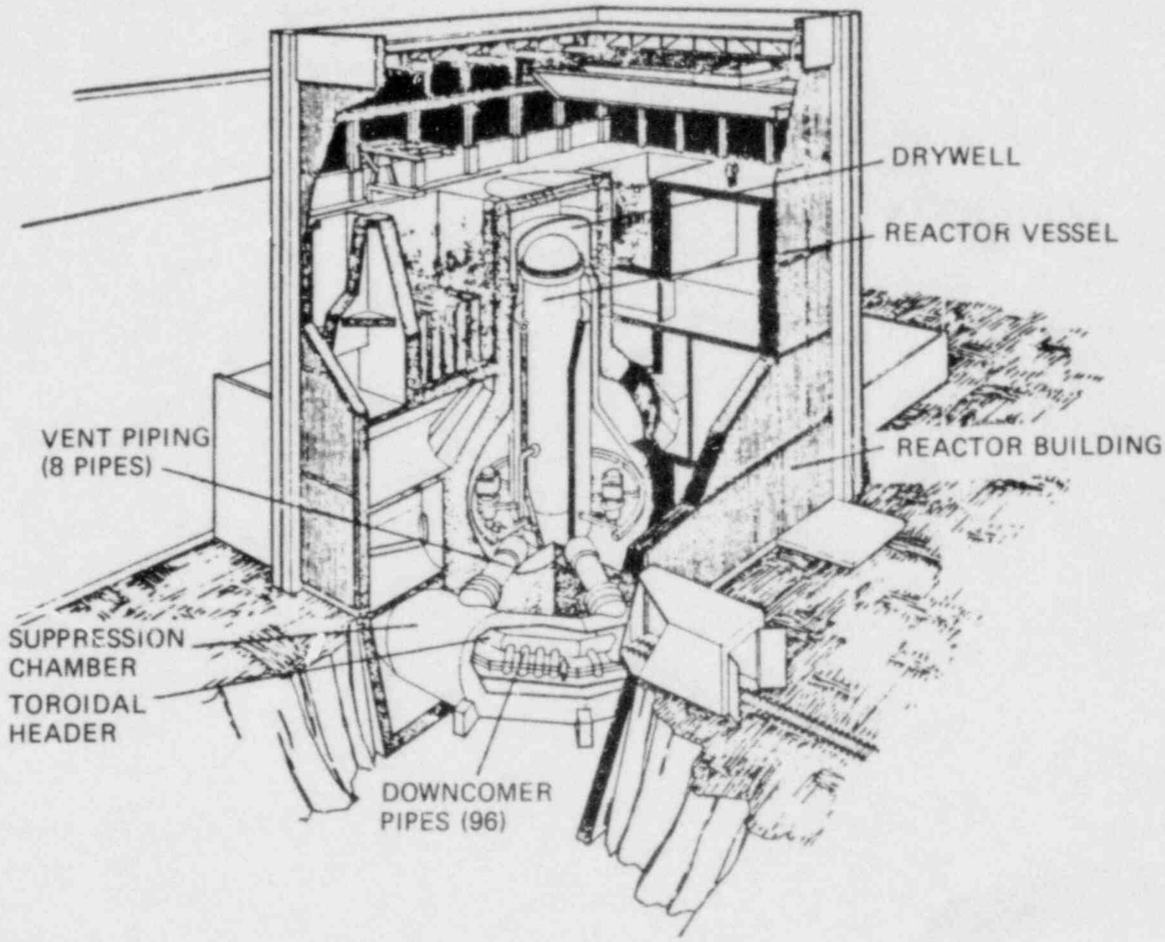

Fig. D.1 BWR reactor building showing primary containment system enclosed.

The torus which contains the pressure suppression pool is designed to essentially the same requirements as the drywell liner, i.e., a maximum internal pressure of 0.531 MPa (62 psig) at 138°C (281°F), but neither the drywell nor the torus is designed to withstand the stresses which would be created by a significant internal vacuum. To ensure that a significant vacuum can not occur in the drywell, vacuum breaker valves are installed, which will open to permit flow from the PSP airspace into the drywell whenever the suppression pool pressure exceeds the drywell pressure by more than 3447 Pa (0.5 psi). Additional vacuum breaker valves with the same setpoints are installed to permit flow from the Reactor Building into the PSP airspace, to prevent a significant vacuum there.

D.4 Identification of the Phenomena

The thermal hydraulic phenomena associated with a BWR pressure suppression pool are mainly those dealing with the PSP response during two

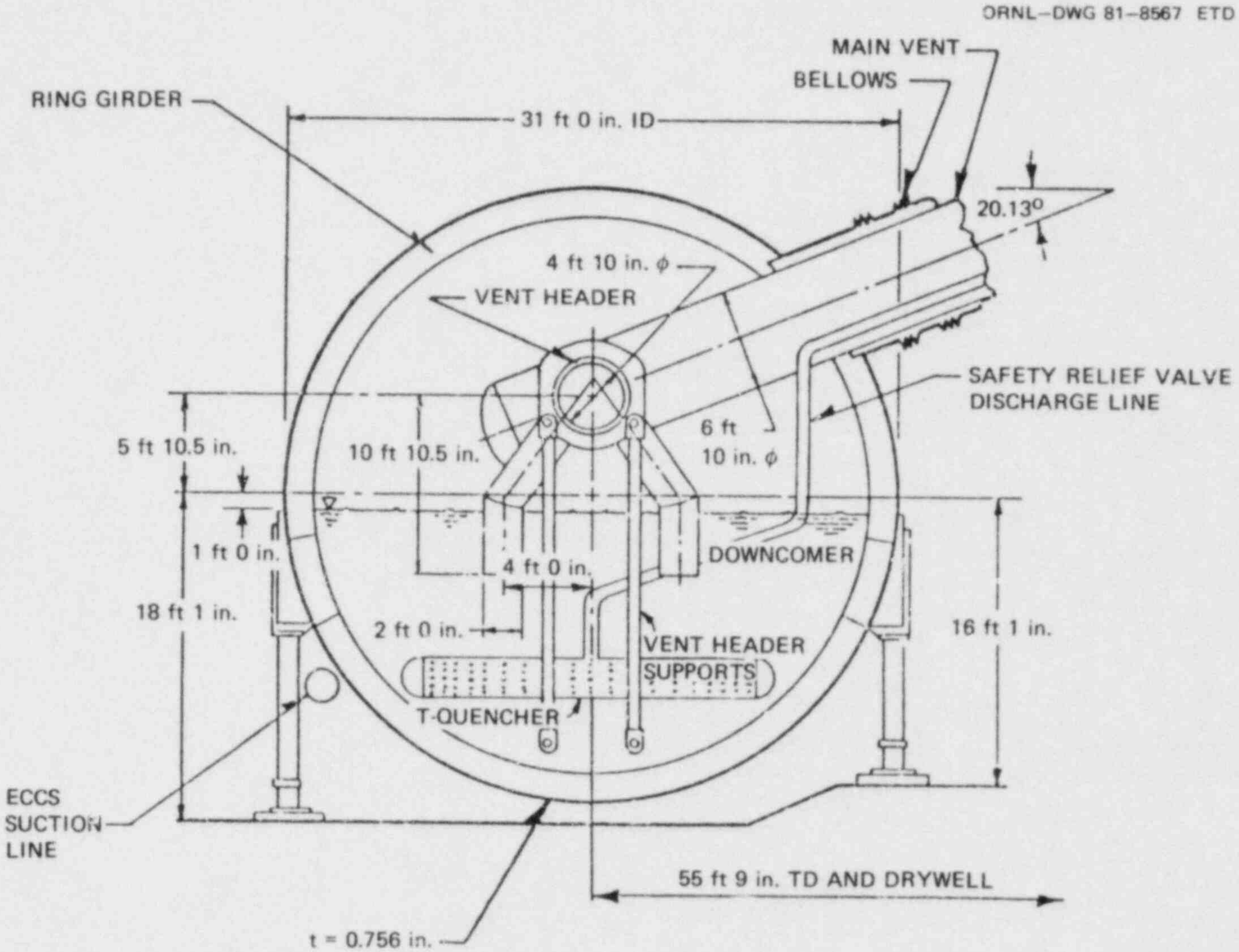

Fig. D.2 Browns Ferry Mark I containment pressure suppression pool.

types of transients: (1) LOCA-related phenomena and (2) SRV-discharge phenomena.

A. LOCA — Related Phenomena

Immediately following the pipe break in a LOCA, the drywell pressure and temperature increase very quickly. The pressure increase forces water standing in the downcomer to accelerate rapidly into the PSP and impinge on the torus wall. Following the slug of water, air that was in the vent pipes and drywell is forced into the PSP. This forms a bubble of air at the downcomer exit which expands into the suppression chamber and causes the pool to swell. As the air bubble rises into the torus airspace, the water will experience a gravity-induced fallback and phase separation will again occur.

The pool swell transient described above lasts on the order of 3 to 5 s.[30] It has been studied by several authors, both experimentally[31–33] and numerically.[34,35] The consensus is that pool swell impingement and drag loads induced during a LOCA are conservatively estimated and acceptable.

Immediately following the pool swell transient, an air/steam mixture will flow into the PSP. Early in this process, when the mass flow rate is high, the injected steam condenses at an unsteady rate causing periodic oscillations in the pressure and flow. However, since the mass flux is high enough to maintain the steam/water interface outside the downcomer, the overall condensation proceeds at a regular rate. This phenomenon is known as condensation oscillation. It is characterized by a steady, periodic variation in the pressure which forces local structures within the torus to vibrate in phase with the oscillations. Condensation oscillation has been studied experimentally[36,37] and analytically;[37,38] however, the basic driving mechanism for the pressure resonance has not been identified.[39]

When the air/steam flow through the downcomer decreases to the point where the condensation rate outside the pipe exceeds the steam flow exiting the pipe, the steam bubble collapses very rapidly. This results in a large drop in the steam pressure and the steam-water interface rushes up into the downcomer. Once there, the interface is warmed by condensing steam and the condensation rate begins to decrease. At some point, the steam pressure will rise, and the interface is pushed out of the downcomer to form an irregularly shaped bubble at the pipe exit. The bubble begins to collapse; and the entire process, known as chugging, repeats. Chugging is characterized by rapid, irregular interface accelerations and pressure oscillations that cause large loads on the torus structure.

The chugging phenomenon is very similar to the condensation oscillation problem. It has also been studied in detail, with analysis methods which range from manometer-like models that attempt to predict the gross motion of the interface, to probabilistic models that attempt to predict internal chugging.[40–42] The central problems which plague analysis of the chugging phenomenon are (1) high uncertainty in the basic condensation rates involved and (2) lack of understanding of the triggering mechanism for bubble collapse.

Both of the condensation phenomena (chugging and condensation oscillation) involve transient, stochastic, turbulent, two phase flow. Because of the complexity of these problems, no accurate assessment has been made of the loads involved.[30] Consequently, the analysis of PSP response to condensation oscillations must rely on data from experiments which model plant behavior. Thus, there is a need for improvement of the analysis capability in this area.

B. SRV — Discharge Phenomena

When a SRV actuates, water and air initially in the discharge line are immediately accelerated into the PSP. This results in air-clearing loads much the same as the pool swell loads discussed earlier. These loads are of no major consequence because they can be adequately "scaled" from the results of laboratory experiments.

Following the air-clearing phase, steam is injected at high velocity into the PSP. Experience has shown that, depending on the discharge device used, condensation oscillations can occur as steam bubbles exit the pipe and collapse in the bulk fluid. As the steam bubbles collapse, severe pressure oscillations are induced on the surrounding structures. Current practice is to limit the severity of these oscillations by using a T-quencher device.[43] The T-quencher is a section of pipe (in the form of a "T") with holes strategically drilled in the arms to enhance local mixing. The T-quencher has been shown to be very effective in eliminating the severity of the pressure pulses, provided the local bulk fluid temperature is sufficiently low.

This is the factor that establishes a limit on the effectiveness of the T-quencher — the local fluid temperature. If it becomes too high, the PSP will not be able to rapidly condense all the steam. Some of it will escape into the air space above the pool and pressurize the torus. When the suppression pool pressure becomes 3447 Pa (0.5 psi) greater than the drywell, the vacuum breakers will open and steam will escape into the drywell (forcing the temperature and pressure up). As more steam is discharged to the PSP, the drywell and torus pressure will continue to increase.

It is concern for this increase in drywell pressure (possibly resulting in overpressurization) that has led to the establishment of strict procedures for the sequential ordering of SRV blowdowns to the PSP. Current practice involves sequentially venting SRVs on opposite sides of the torus; this prevents excessive local temperatures near any discharge point.

More recently, concern has been expressed over the long term response of the PSP during a prolonged Station Blackout. During part of this scenario, the operator loses manual control of the SRVs. The long term result is that a single SRV will continually open into the PSP. The local fluid temperature will monotonically rise, resulting in pressurization of the torus and possible condensation oscillations. The potential exists for rupture of the torus due to overpressure coupled with violent pressure oscillations.

For example, this potential exists in the particular case of a Station Blackout with loss of the HPCI and RCIC systems. At about 100 min into the transient, local pool temperatures at the discharge bay are postulated (c.f. Section 9) to be greater than 149°C (300°F). At that time, steam condensation oscillations are expected to accelerate due to the excessive temperature and the continuous discharge of superheated, non-condensible gases (from hydrogen generated in the melting core) into the PSP. These extreme conditions in the PSP yield a high probability for rupture of the torus.

Along with the temperature rise assocated with SRV discharge comes a gravity induced thermal stratification of the pool. This phenomenon is of interest because the layering tends to remain long after SRV discharge is complete. Since the stratification remains for a significant period of time, the validity of the current design basis for Mark I PSPs becomes questionable. This is because a fundamental assumption in virtually all the transient analyses is that the pool is thoroughly mixed and at a uniform temperature following SRV blowdown. The impact of thermal stratification on the performance of the PSP remains to be evaluated.

Thermal stratification is also of interest because the temperature distribution in the PSP affects the fission product distribution in the torus. Should a breach of primary containment occur, transport of the fission products to the environment will depend, to some extent, on the thermal stratification in the torus. Thus, an analysis of fission product transport is coupled with the thermal stratification problem.

To the best of our knowledge, no information is available in the open literature concerning systematic analysis of the thermal stratification problem. There is therefore the need for research in this area.

D.5 Pool Modeling Considerations

Model requirements

The prediction of suppression pool system behavior during the course of a Severe Accident is an exceedingly difficult task. The previously described phenomena comprise a complex set of physical processes for which few detailed analytical models currently exist. These phenomena can be roughly divided into two types, i.e., thermal hydraulic phenomena and fluid-structure interaction phenomena. The initial efforts at ORNL are directed toward the development of the thermal hydraulic model. A brief survey was conducted to identify existing multi-dimensional thermal hydraulic analysis codes which might be applied to the problem. The results of the survey are shown in Table D1 which presents a brief summary of the candidate codes.

Due to the nature of the previously described phenomena, it was felt that any model employed for suppression pool analysis should possess the following characteristics:

1. Appropriate hydrodynamics
    a. two phase
    b. non-equilibrium
    c. free field format (equations and constitutive relationships independent of hydraulic diameter)
    d. incorporate a non-condensable gas field
    e. employ multi-dimensional geometry (3 dimensions desirable)
2. Employ appropriate constitutive relationships
3. Utilize efficient solution techniques
4. Readily accessable to ORNL staff.

Table D1. Candidate suppression pool analysis codes

Code	Developer	Geometry	T-H characteristics
TRAC (PIA, PD2)	LANL	3D cylindrical	2 fluid nonequilibrium
COBRA-TF	BPNL	3D cartesian	3 field nonequlibrium
COMMIX-2	ANL	3D	2 fluid nonequilibrium
BEACON/Mod2	INEL	2D cartesian cylindrical spherical	2 component vapor 2 phase nonequilibrium

Requirements 1.a and 1.b are imposed by the complex nature of the condensation processes involved in SRV discharge and LOCA blowdown transients. Requirement 1.c is a result not only of the nature of the SRV discharge process, but of the geometric dimensions of the suppression pool. The enormous size and complex structural design of the suppression pool system virtually mandates that free field hydrodynamics be employed. Requirement 1.d is imposed because significant amounts of non-condensable gases will be released from the fuel and generated in metal-water reactions during Severe Accidents. Requirement 1.e. is imposed by the nature of the hydrodynamic phenomena and the complex structural relationship of the SRV discharge lines, vent downcomers, and ring header suction system. Appropriate constitutive relationships (requirement 2) are needed for closure of the hydrodynamic field equations. Many of the relationships employed in existing thermo-hydraulic codes may be invalid for use in the present problem. Requirement 3 is imposed by the fact that it will be necessary to predict the behavior of the suppression pool for periods of time ranging from a few minutes to several hours. It is, therefore, important that all codes employ fast solution techniques in order to minimize computing costs. The fourth characteristic is desired because the computer code selected for the analysis will require local modification as necessary to reflect the unique characteristics of the suppression pool problem.

We are unaware of any existing computer code or model which possesses all of the desirable characteristics outlined above. The initial effort at ORNL is directed toward development of a pool model utilizing the TRAC-PIA[44] vessel module. TRAC (Transient Reactor Analysis Code) — PIA is a best estimate computer code developed at Los Alamos National Laboratory for analysis of large break PWR loss of coolant accidents. Although more recent versions of TRAC are available (i.e., PD2, PF1, and BD1), TRAC-PIA was chosen for this application because it is installed and operational on ORNL's computer system.

While TRAC-PIA does employ a full three dimensional, two fluid, nonequlibrium approach to reactor vessel hydrodynamics, it does not account for free flow or non-condensable gas fields. Additionally, the constitutive relationships employed in TRAC-PIA are based upon data which was originally developed for vertical pipe flow and may not be valid for application to steam jet discharge phenomena.[45] Though these limitations are significant, it is felt that our access to TRAC-PIA will allow us to implement any changes in the constitutive relationship package which might be necessary to reflect the characteristics of the suppression pool phenomena. LANL is currently modifying TRAC to include non-condensable gas field hydrodynamics (TRAC-PF1). ORNL's utilization of TRAC-PIA will allow us to quickly implement this modified version when it becomes available.

We have tentatively chosen the PELE-IC[46] code for future use in analysis of the fluid-structure interaction within the suppression pool. PELE-IC (developed by LLL) couples a two dimensional Eulerian fluid dynamics algorithm to a Lagrangian finite element shell algorithm. The code can couple either a one dimensional or a lumped parameter description of compressible gases, and can employ either cartesian or cylindrical coordinates. PELE-IC employs the basic semi-implicit solution algorithm contained in the SOLA code.[47] The movement of free surfaces is treated in a full donar cell fashion based on a combination of void fractions and interface orientation. The structural motion is calculated by a finite element method, from the applied fluid pressure at the fluid structure interface. The finite element shell structure algorithm uses conventional thin shell theory with transverse shear and provides the fluid module with the resultant position and velocity of the interface. The code is capable of analyzing both vent clearing and condensation related phenomena.

TRAC suppression pool model description

A block diagram of the ORNL-TRAC suppression pool model is shown in Fig. D-3. The model is comprised of the TRAC VESSEL module which represents the pool hydrodynamics, and two FILL and PIPE modules, which represent the SRV discharge flow and T-quencher assembly. A detailed view of

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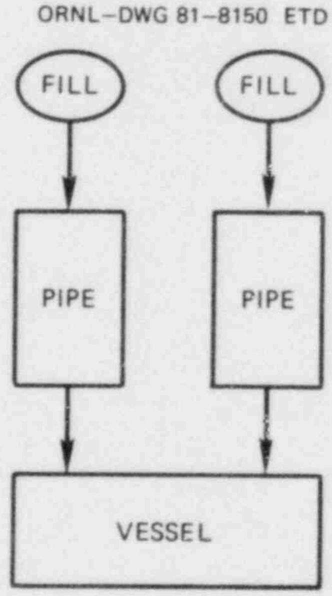

Fig. D.3 Suppression pool model block diagram.

the pool model is shown in Figs. D-4 and D-5. The pool is represented as 112 cells in cylindrical geometry, located at five axial, four azimuthal, and eight radial regions. SRV discharge is assumed to occur as shown in Figs. D-4 and D-5. Half of the SRV discharge flow is directed radially inward from the fifth radial zone, while the other half is directed radially outward. It should be noted that the T-quencher is not modeled in a strictly physical fashion, since the discharge is assumed to occur through a single open ended pipe with a flow area approximately equal to that of the T-quencher device. This approach was chosen for the initial model due to the significant reduction in input preparation time and model complexity associated with this approach. The initial water level is assumed to be 4.47 meters above the pool bottom (i.e., at level 4 in

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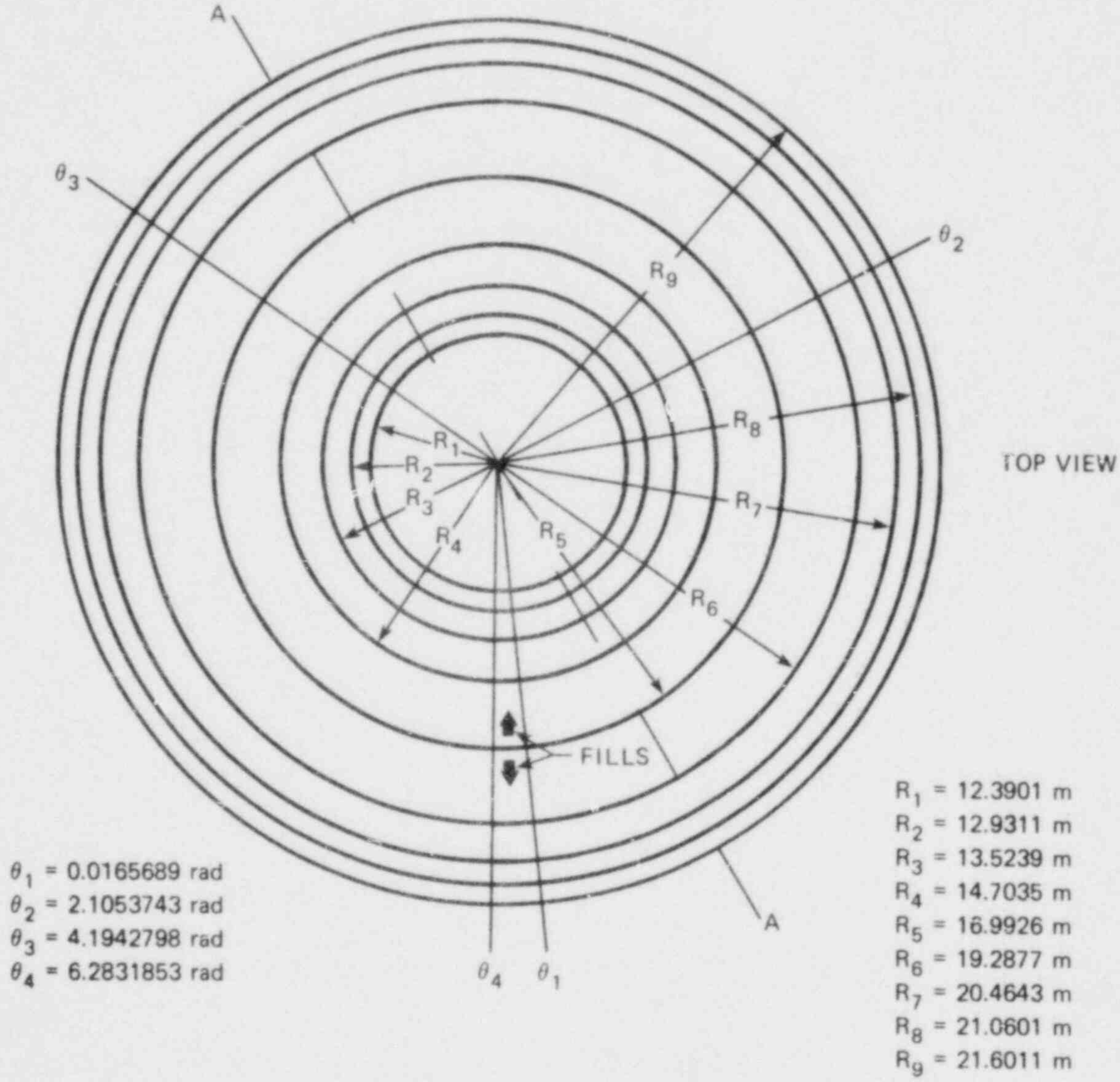

Fig. D.4 Suppression pool model top view.

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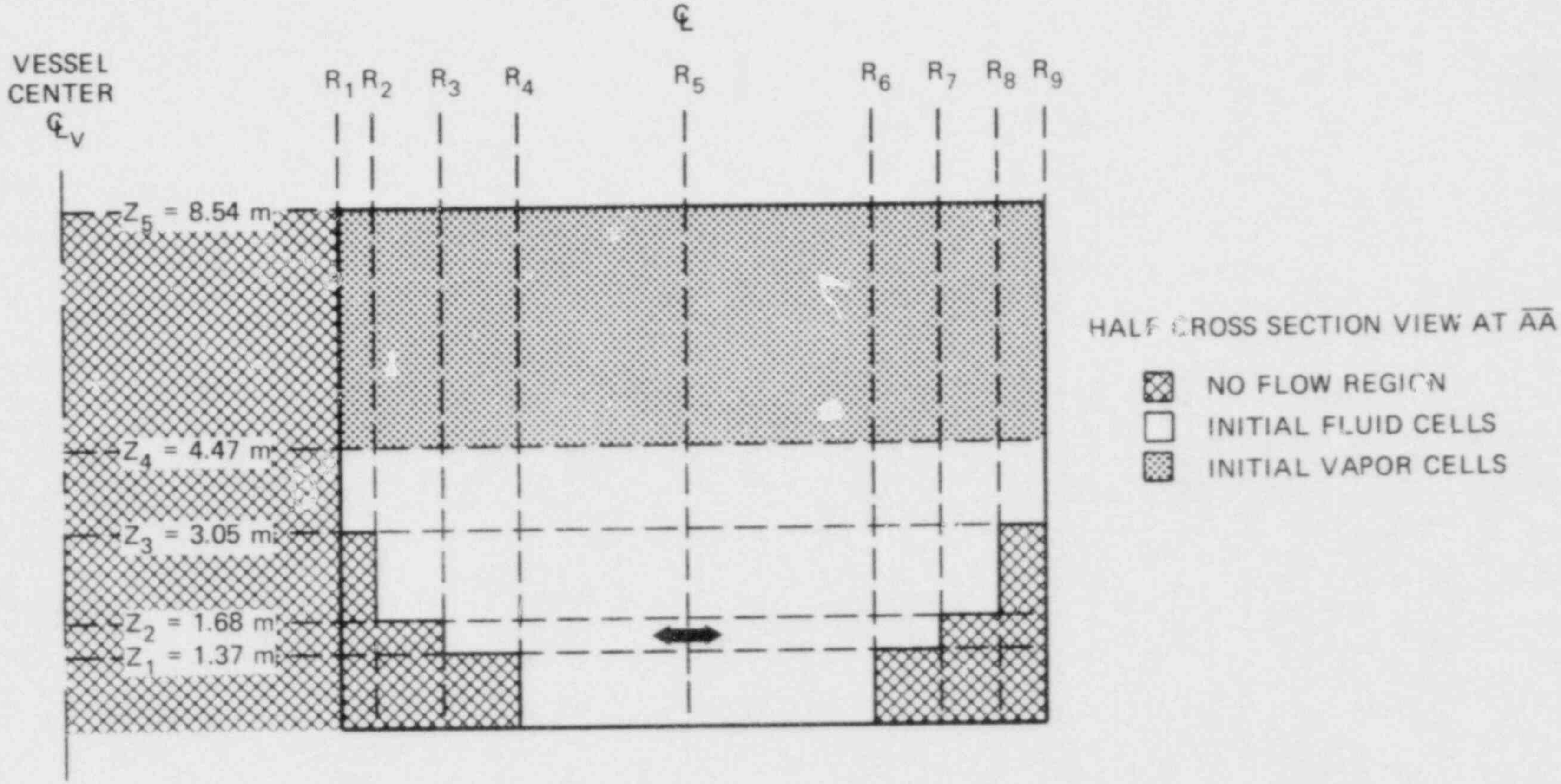

Fig. D.5 Suppression pool model cross section.

Fig. D-5) with all cells in level 5 occupied by vapor. Internal structures (vent header, down comers, etc.) are not modeled.

Current status of suppression pool model

Several preliminary computer runs have been made in which TRAC data base errors have been identified and eliminated. We are currently experiencing problems related to the limitations of TRAC-PIA in accounting for the simultaneous existence of steam (vapor) and perfect gas (nitrogen) within the vessel. The TRAC user must specify that either vapor or air properties be used everywhere within the code. In terms of the pressure suppression pool, this means that the nitrogen gas (which is located in the space above the pool) must be treated as subcooled vapor if one is to inject steam into the pool. It is also possible that the constitutive relationships within TRAC are producing instabilities in the solution due to their dependence upon hydraulic diameter and fluid property information. We are currently working closely with the TRAC User Liaison Section at LANL in an effort to determine whether these problems can be overcome. In the event these problems prove to be insurmountable, we will re-evaluate the remaining code candidates and select an alternative program for the suppression pool thermal hydraulic analysis.

APPENDIX E

A COMPENDIUM OF INFORMATION CONCERNING THE BROWNS FERRY UNIT 1 HIGH-PRESSURE COOLANT INJECTION SYSTEM

E.1 Purpose

The High-Pressure Coolant Injection (HPCI) System is designed to ensure adequate core cooling to prevent damage to fuel in the event of a loss of coolant accident that does not result in rapid depressurization of the reactor vessel. The HPCI System provides water to make up for that which is lost through steam generated by decay heat.[48]

E.2 System Description

The HPCI System (Fig. E-1) consists of a steam turbine driven booster pump — main pump combination and the associated piping and valves. The booster pump can take suction from either the condensate storage tank or the pressure suppression pool. The HPCI pumped flow enters the reactor vessel feedwater line "A" via a thermal sleeve connection. A test line permits testing of the HPCI System at full flow while the reactor is at power, with the main pump discharge routed to the condensate storage tank. A minimum flow line connects the main pump discharge to the pressure suppression pool, as a means of ensuring a flow of at least 0.038 m^3/s (600 GPM) through the pumps. The normal setpoint for HPCI pumped flow is 18.93 m^3/s (5000 GPM).

The HPCI turbine is driven by steam extracted from main steam line "B" upstream of the main steam line isolation valves. The two primary containment isolation valves in the steam line to the HPCI turbine are normally open to keep the piping to the turbine at elevated temperatures to permit rapid system startup (within 25 s of receipt of an initiation signal). The normally closed DC-motor-operated steam supply valves upstream of the HPCI turbine will open against full system pressure within 20 s after receipt of a system initiation signal. Signals from the control system open or close the turbine stop valve. The turbine control valve is physically attached to the HPCI turbine and is positioned by the turbine governor as necessary to maintain the pumped flow at the level set by the operator, normally 0.315 m^3/s (5000 GPM). The turbine exhaust steam is discharged to the pressure suppression pool. The turbine gland seals are vented to a gland seal condenser. A small water flow diverted from the booster pump discharge is used to cool both the turbine lubricating oil cooler and the gland seal condenser. This cooling flow is returned, together with the gland seal condensate, to the booster pump suction. Noncondensible gases from the gland seal condenser are removed via a DC-motor-operated blower to the Standby Gas Treatment System.

A vacuum breaker line (not shown on Fig. E-1) is installed between the torus airspace and the HPCI turbine exhaust line. Its purpose is to prevent water from the pressure suppression pool from being drawn up into the turbine exhaust line as the steam condenses in this line following turbine operation.

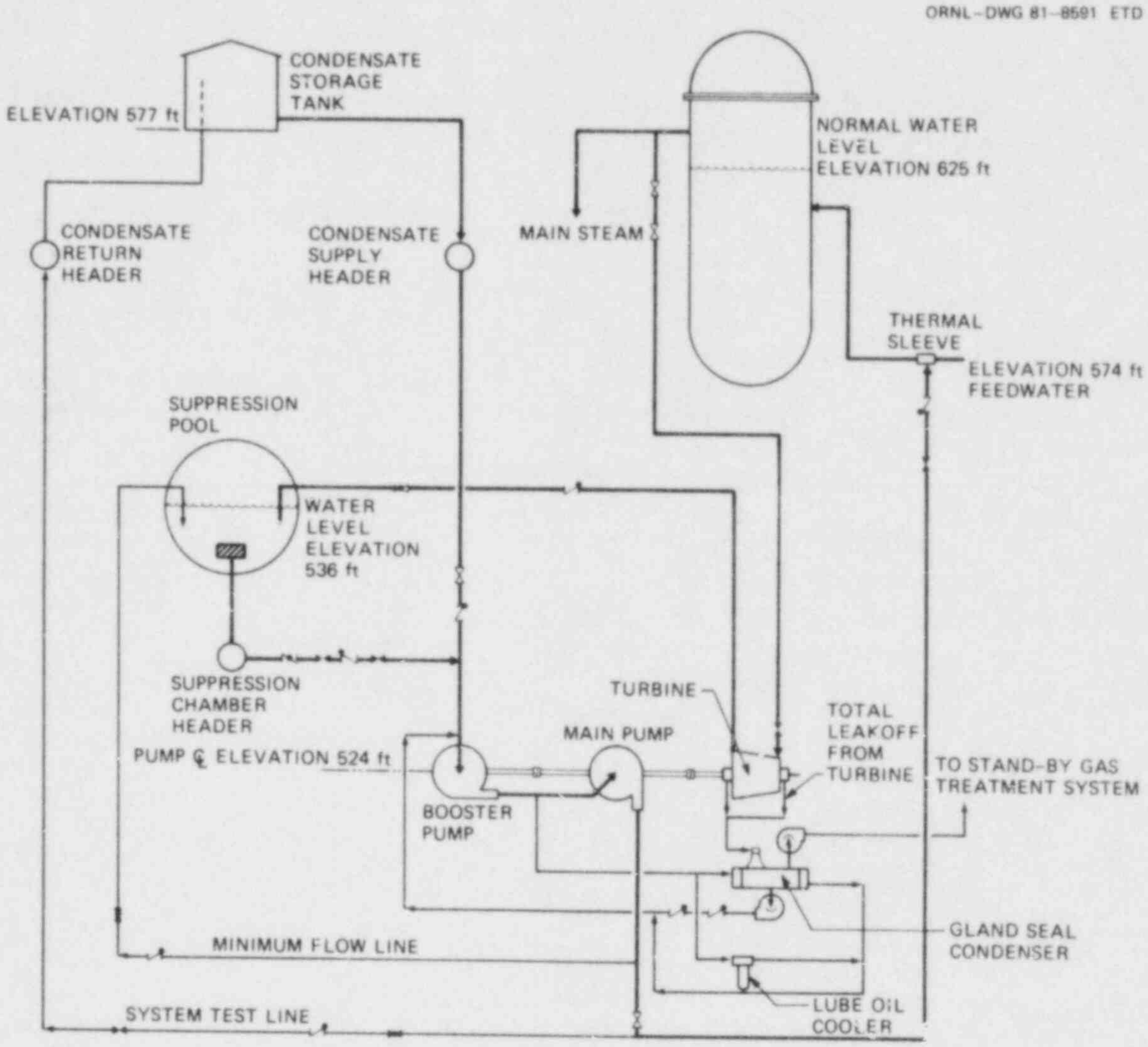

Fig. E.1 High pressure coolant injection system.

All components required for operation of the HPCI System are completely independent of AC power, control air systems, or external cooling water systems, requiring only DC power from the unit battery. On loss of control air, the HPCI steam line drains to the main condensers will fail closed; this is their normal position when the HPCI system is in operation.

The principal HPCI equipment is installed in the reactor building, at a level below that of the pressure suppression pool. The turbine-pump assembly is located in a shielded area so that personnel access to adjacent areas is not restricted during HPCI System operation. The only operating component located inside the primary containment is the normally open AC-motor-operated inboard HPCI steam line isolation valve, which will remain open on loss of power.

E.3 HPCI Pump Suction

The HPCI pump can take suction either from the condensate storage header or from the pressure suppression pool via the suppression pool ring header. The normal lineup is for suction of the reactor-grade water in the condensate storage header.

Each Browns Ferry unit is provided with a 1419.4 m^3 (375,000 gallon) condensate storage tank, which provides a water head to the storage header for that unit. The storage header, which taps into the bottom of the storage tank, feeds the suctions of the high-pressure ECCS Systems, specifically, the pumps for the HPCI and the RCIC systems. The core spray pumps and the RHR pumps can be fed from this source, but are not normally aligned to it. All other demand for condensate storage tank water is fed via a standpipe within the tank; the standpipe height is such that 511.0 m^3 (135,000 gallons) of water is reserved for the ECCS Systems.

It is important to note that the local-manual opening of one normally locked-shut valve will cross-connect the Unit 3 and the Unit 1 condensate storage header; the opening of a second such valve will cross-connect all three condensate storage headers.[49] Thus the Unit 1 ECCS Systems have a minimum of 511.0 and a maximum of 1419.4 m^3 (135,000 to 375,000 gallons) available through normally open valves from the Unit 1 storage tank and these limits can be increased three-fold by manually opening two normally locked-closed valves. Two additional condensate storage tanks, each of 1892.5 m^3 (500,000 gallon) capacity, have recently been installed at the Browns Ferry Nuclear Plant. Thus it is unlikely that HPCI System operation will ever be limited by the availability of condensate storage header water.

The HPCI booster pump suction will be automatically shifted from the condensate storage header to the suppression pool ring header if either:

1. The water level above the booster pump suction falls to an elevation of 168.4 m (551 ft). This would mean that the condensate storage tank had completely drained and there remained just sufficient water in the condensate storage header to effect a transfer before losing net positive suction head (NPSH).

or

2. The pressure suppression pool level increases to an indicated level of 0.18 m (+7 in). Since the normal pool level is maintained between 0.05 and 0.15 m (2 and 6 in.),[50] this implies the addition of between 257.4 and 371.0 m^3 (68,000 and 98,000 gallons) of water to the pool.

The 0.76 m (30 in.) diameter suppression pool ring header lies parallel to and beneath the suppression pool. Water flow from the suppression pool to the ring header is via four 0.76 m (30 in.) diameter downcomer pipes spaced at irregular intervals around the torus. Each downcomer is capped with strainers at the torus end.

The change in HPCI pump suction lineup is accomplished by the opening of two DC-motor-operated valves in the line from the pressure suppression pool to the HPCI booster pump suction followed by the closing of the DC-motor-operated valve in the suction line from the condensate storage header. A check valve in the line from the suppression pool prevents backflow from the condensate storage tank into the suppression pool during the change.

Two pressure switches are used to determine the water head from the condensate storage tank above the booster pump suction and two level switches monitor the suppression pool level. In either case, just one of the two available signals is sufficient to initiate the shift in HPCI booster pump suction. Once the booster pump suction has been automatically shifted to the pressure suppression pool, the operator cannot reposition the valves back to the condensate header suction lineup.

The minimum required net positive suction head (NPSH) for the HPCI booster pump is 6.40 m (21 ft). This requirement is easily satisfied when suction is taken from the condensate storage header which is at an elevation of approximately 8.69 m (28 1/2 ft) above the booster pump centerline.

When the HPCI booster pump suction is shifted to the pressure suppression pool due to a high indicated pool level of 0.18 m (+7 in.), the pool water level is 4.04 m (13.25 ft) above the booster pump centerline. Therefore, the NPSH requirement of 6.40 m (21 ft) can be met as long as the temperature of the pumped water is below 85.0°C (185°F), assuming no containment back pressure.

E.4 System Initiation

Either of two signals will cause an automatic start of the HPCI System. These are:

1. Low reactor water level [12.09 m (476 in.) above vessel zero].
2. High drywell pressure [0.12 MPa (2 psig)].

With either of these conditions, both the suction valve to the condensate storage header and the minimum flow bypass valve will open if they were closed. The pump discharge valve between the main HPCI pump and feedwater header "A" will open, and the two test line isolation valves will close if they were open. The steam supply valve to the turbine will open.*

The DC-motor-driven auxiliary oil pump starts and as the oil pressure increases, the turbine stop and control valves open. Above 1800 rpm, the main oil pump which is driven on the turbine shaft takes over and maintains the oil pressure while the auxiliary oil pump shuts down. The minimum flow bypass valve closes automatically when the increasing HPCI System pump flow exceeds 0.076 m^3/s (1200 GPM).

The time from actuating signal to full flow is less than 25 seconds. The turbine control system will act to maintain a pumped flow of 0.315 m^3/s (5000 GPM) into the feedwater line over a reactor pressure range of from 1.14 to 7.83 MPa (150 to 1120 psig). If desired, the operator in the Control Room can operate the system in the manual or automatic mode at a different controlled flow.

The HPCI turbine operates at between 0.746 to 3.356 MW, (1000 and 4500 horsepower) with a steam demand of between 6.17 to 23.18 kg/s (49,000 and 184,000 lbs/h). System steam and pumped water flows at operating conditions are given in tabular form on Fig. 6.4.1 of the Browns Ferry FSAR.

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*The two normally-open primary containment isolation valves in the steam supply line will not reopen, if closed.

E.5 Turbine Trips

On an HPCI turbine trip, the turbine stop valve closes and the minimum flow bypass valve to the pressure suppression pool closes to preclude drainage from the condensate storage header into the suppression pool. The following conditions will cause turbine trip:

1. High reactor vessel water level [14.78 m (582 in.) above vessel zero].
2. High HPCI turbine exhaust pressure [1.14 MPa (150 psig)].
3. Low HPCI booster pump suction pressure [−0.381 m (−15 in.) Hg].
4. HPCI turbine mechanical overspeed (5000 rpm).
5. Any HPCI isolation signal.
6. Remote manual trip from Control Room.
7. Manual trip lever on the HPCI turbine.

All turbine trips except high reactor water level and HPCI isolation will reset automatically when the initiating condition clears. The high reactor water level signal can be reset manually, or will reset automatically when the reactor water level decreases to the low reactor water level HPCI initiation point, 12.09 m (476 in.) above vessel zero. The HPCI isolation signal must be manually reset.

E.6 System Isolation

The Primary Containment and Reactor Vessel Isolation Control System initiates automatic isolation of appropriate pipelines which penetrate the primary containment whenever certain monitored variables exceed their preselected operational limits. The system is designed so that, once initiated, automatic isolation continues to completion. Return to normal operation after isolation requires deliberate operator action.[51]

An automatic isolation signal for the HPCI System causes the inboard and outboard HPCI steam line isolation valves to close, trips the HPCI turbine, and closes the two motor-operated valves in the suction line from the pressure suppression pool. The inboard steam line isolation valve is AC-motor-operated, the outboard DC-motor-operated. The maximum closing time for these valves is 20 s.

The following conditions cause a HPCI System isolation signal:[52]

1. HPCI System equipment space high temperature. Since high temperature in the vicinity of the steam supply line or other HPCI equipment could indicate a break in the turbine steam supply line, an isolation signal is generated if this temperature exceeds 93.3°C (200°F). This temperature is sensed by four sets of four bimetallic temperature switches. These 16 temperature switches are arranged in four trip systems with four temperature switches in each trip system. The four temperature switches in each trip system are arranged in one-out-of-two taken twice logic.
2. HPCI turbine high steam flow. Since high steam flow could indicate a break in the turbine steam supply line, an isolation signal is generated if the measured steam flow exceeds 150% of design maximum steady state flow. The steam line flow is sensed by two differential pressure switches which monitor the differential pressure across a mechanical element installed in the HPCI turbine steam pipeline. The tripping of either switch at a differential pressure of 0.62 MPa (90 psi) initiates HPCI System isolation.

3. Low reactor pressure. After steam pressure has decreased to such a low value that the HPCI turbine cannot be operated, the steam line is isolated so that steam and radioactive gases will not escape from the HPCI turbine shaft seals into the reactor building. The steam pressure is sensed by four pressure switches from the HPCI turbine steam line upstream of the isolation valves. The switches are arranged in a one-out-of-two taken twice logic. The set point for this isolation signal is 0.793 MPa (100 psig).
4. High turbine exhaust diaphragm pressure. A line tapping off the turbine exhaust line contains two rupture diaphrams in series, with the space between vented to the HPCI equipment space through a flow restricting orifice. The diaphrams are designed for 1.138 MPa (150 psig). If the pressure in the space between the diaphrams exceeds 0.172 MPa (10 psig), a system isolation signal is generated.
5. Manual Isolation. If a HPCI initiation signal is present, the operator can cause HPCI system isolation by pushing a control panel push button.

The low reactor pressure isolation will be automatically reset if reactor pressure is restored; all other isolation signals seal in and the operator must push the HPCI auto isolation circuit reset push-button after the condition has cleared.

E.7 Technical Specifications

1. The HPCI System shall be operable
   a. Prior to startup from cold condition
   b. Whenever there is irradiated fuel in the reactor vessel and the reactor vessel pressure is greater than 0.945 MPa (122 psig), except
   c. If the HPCI System is inoperable the reactor may remain in operation for a period not to exceed 7 days provided ADS, core spray, LPCI mode of RHR and RCIC are all operable.

   If these conditions are not met, an orderly shutdown shall be initiated and the reactor vessel pressure reduced to 0.945 MPa (122 psig) or less within 24 hours.
2. HPCI testing shall be performed as follows:
   a. Simulated automatic actuation test — once per operating cycle
   b. Pump operability — once per month
   c. Motor operated valve operability — once per month
   d. Flow rate at normal reactor operating pressure — once per three months
   e. Flow rate at 1.138 MPa (150 psig) — once per operating cycle
3. Whenever HPCI is required to be operable the piping from the pump discharge to the last flow blocking valve shall be filled. Water flow from the high point vent must be observed monthly. (Purpose is to prevent water hammer when the system is started.)

APPENDIX F

A COMPENDIUM OF INFORMATION CONCERNING THE BROWNS FERRY UNIT 1 REACTOR CORE ISOLATION COOLING SYSTEM

F.1 Purpose

The Reactor Core Isolation Cooling (RCIC) System is designed to ensure that the core is not uncovered in the event of loss of all AC power. The RCIC System provides water to make up for that lost through steam generated by decay heat during reactor isolation.[53] RCIC is a consequence limiting system rather than an ECCS System; the system design is not predicated on any loss of structure accident.[54]

F.2 System Description

The RCIC System (Fig. F-1) consists of a steam-turbine driven pump unit and the associated piping and valves. The RCIC pump can take suction from either the condensate storage tank or the pressure suppression pool and discharges to the reactor vessel feedwater line "B" via a thermal sleeve connection. A full flow system test line permits testing of the RCIC System while the reactor is at power, with the pump discharge routed to the condensate storage tank. A minimum flow line connects the RCIC pump discharge to the pressure suppression pool, as a means of ensuring a flow of at least 0.004 m^3/s (60 GPM) through the pump. The minimum flow bypass valve closes at a system flow of greater than 0.008 m^3/s (120 GPM). The normal setpoint for RCIC pumped flow is 0.038 m^3/s (600 GPM).

The RCIC turbine is driven by steam extracted from main steam line "C" upstream of the main steam line isolation valve. The two primary containment isolation valves in the steam line to the RCIC turbine are normally open to keep the piping to the turbine at elevated temperatures so as to permit rapid system startup (within 30 seconds of an initiation signal). The normally closed DC-motor-operated steam supply valve just upstream of the RCIC turbine will open against full system pressure within 15 seconds after receipt of a system initiation signal.

The RCIC turbine exhaust steam is discharged to the pressure suppression pool. The turbine gland seals are drained to a barometric condenser, in which the steam is condensed by a water spray. The water spray is provided by a flow diverted from the RCIC pump discharge to pass through the turbine lube oil cooler and subsequently form the spray. The condensate from the barometric condenser is pumped back to the RCIC pump suction. A vacuum pump removes the non-condensibles from the barometric condenser and inserts them into the pressure suppression pool.

A vacuum breaker line (not shown on Fig. F-1) is installed between the torus airspace and the RCIC turbine exhaust line. Its purpose is to prevent pressure suppression pool water from being drawn up into the turbine exhaust line when the remaining exhaust steam condenses after turbine operation and shutdown.

All components normally required for initiating operation of the RCIC System are completely independent of AC power, plant service air, and external cooling water systems, requiring only DC power from a unit battery

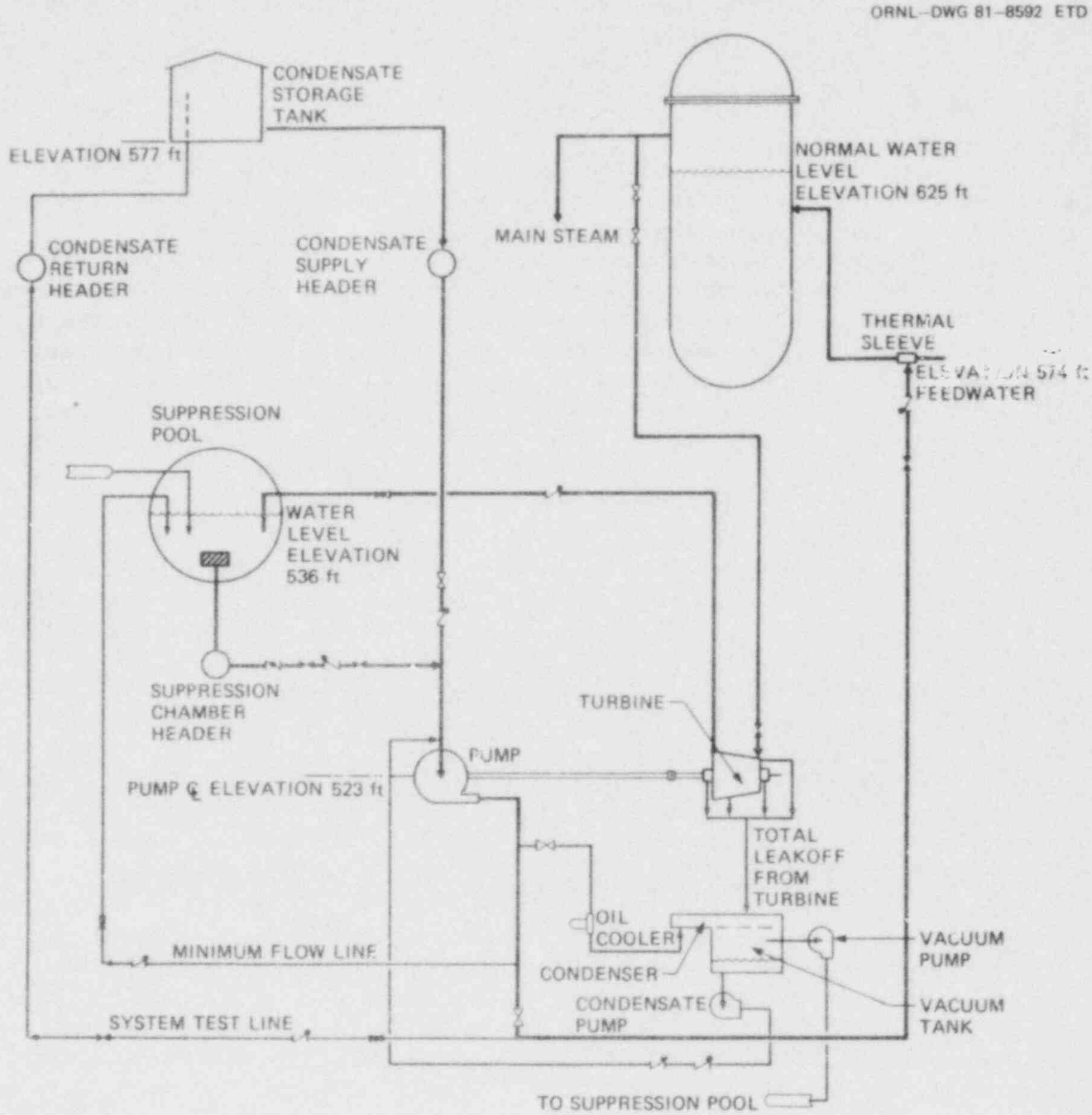

Fig. F.1 Reactor core isolation cooling system.

to operate the valves, vacuum pump, and condensate pump.[55] On loss of control air, the RCIC drain lines to the main condensers will fail closed; this is their normal position when the RCIC System is in operation. The drain functions of these valves is transferred to overseat drain ports in the turbine stop valves.

The principal RCIC equipment is installed in the reactor building at a level below that of the pressure suppression pool. The turbine-pump assembly is located in a shielded area so that access to adjacent areas of the reactor building is not restricted during RCIC System operation. The only operating component located within the Primary Containment is the normally open AC-motor-operated RCIC inboard steam line isolation valve, which will remain open on loss of power.

F.3 RCIC Pump Suction

The RCIC pump can take suction either from the condensate storage header through a single normally-open DC-motor-operated valve or from the pressure suppression pool via the suppression pool ring header.

Each Browns Ferry Unit is provided with a 1419.4 m^3 (375,000 gallon) condensate storage tank, which provides a water head to the storage header for that unit. The storage header, which taps into the bottom of the storage tank, feeds the suctions of the high-pressure ECCS Systems, specifically, the pumps for the HPCI and the RCIC systems. The core spray pumps and the RHR pumps can be fed from this source, but are not normally aligned to it. All other demand for condensate storage tank water is fed via a standpipe within the tank; the standpipe height is such that 511.0 m^3 (135,000 gallons) of water is reserved for the ECCS Systems.

There is no provision for an automatic shifting of the RCIC pump suction from the condensate storage header to the pressure suppression pool. However, if condensate storage tank water is unavailable for any reason, the control room operator can shift the RCIC pump suction to the pressure suppression pool ring header. This is done by remote-manually opening the two DC-motor-operated suction valves to the pool ring header; when these two valves are fully open, the suction valve to the condensate storage header will automatically close.

The minimum required net positive suction head (NPSH) for the RCIC System is 6.10 m (20 ft), which is readily available with suction taken from the condensate storage tank. With suction from the pressure suppression pool, the required NPSH is available for suppression pool temperatures up to 85.0°C (185°F) with no containment back pressure.

F.4 System Initiation

An automatic start of the RCIC System is initiated by low reactor vessel water level at 12.10 m (476.5 in.) above vessel zero. The single normally-closed valve in the steam supply line opens, and steam is admitted to the turbine. (The turbine stop valve and control valve are normally open when the RCIC System is in standby.) The barometric condenser vacuum pump starts, and the condensate pump will act to automatically control the water level in the condenser. The RCIC turbine speed and pump flow are automatically maintained by the steam flow controller. Turbine lube oil is supplied by a shaft-driven oil pump. The single normally-closed valve in the pump discharge line to feedwater header "B" automatically opens, and the minimum flow bypass valve closes automatically when pump flow exceeds 0.008 m^3/s (120 GPM).

If the RCIC System is in an abnormal lineup when an initiation signal is received, the system will realign:

a. If the normally open pump discharge valve is closed, it will open.
b. If the normally open pump suction valve from the condensate storage header is closed, it will open *provided* at least one of the suppression pool suction valves is not fully open.

c. If the normally closed valve in the system full flow test line, which returns RCIC pump discharge water to the pressure suppression pool, is open, it will close.
d. If the system logic mode is in test, it will automatically switch to automatic flow control to maintain turbine speed and pump flow.

The time from actuating signal to full flow is less than 30 seconds. The turbine control valve is governed by a speed controller which compares the measured (tachometer) turbine speed to a speed demand provided by the flow controller. In the automatic mode, the speed demand is established by comparison of the measured pump discharge flow to the set point signal, which, though variable, is normally set at 0.038 m^3/s (600 GPM). In the manual mode, the operator positions a potentiometer to produce a direct speed demand signal to the speed controller.

The RCIC System is designed to provide a full flow of 0.038 m^3/s (600 GPM) at reactor pressures from 1.138 to 7.826 MPa (165 to 1135 psia). With the reactor vessel at the higher pressure, the RCIC turbine delivers approximately 0.373 MW (500 horsepower) with a steam demand of 3.53 kg/s (28,000 lbs/h). With the reactor vessel at 1.138 MPa (165 psia), the turbine delivers approximately 0.060 MW (80 horsepower) with a steam demand of 0.96 kg/s (7600 lbs/h).[56]

In addition to panel 9–3 in the Control Room, a RCIC pump flow controller is also located at the remote shutdown panel for use in emergencies when the Control Room is not available.

F.5 Turbine Trips

On an RCIC turbine trip, the turbine trip throttle valve (not shown on Fig. 1) closes. The valve in the steam supply line which was opened by the RCIC initiation signal remains open, as does the discharge valve to feedwater header "B". Thus, the RCIC System remains lined up for injection with the exception of the turbine trip throttle valve.

The following conditions will cause the trip throttle valve to close:
1. High reactor vessel water level [14.78 m (582 in.) above vessel zero].
2. Electrical overspeed at 110% rated speed.
3. Mechanical overspeed at 125% rated speed.
4. High RCIC turbine exhaust pressure [0.276 MPa (25 psig)].
5. Low RCIC pump suction pressure [0.381 m (15 in.) Hg vacuum].
6. Any automatic isolation signal.
7. Remote manual trip from the control room.
8. Local manual trip lever.

All turbine trips except the mechanical overspeed operate by de-energizing a solenoid valve which dumps oil, allowing a spring to close the trip throttle valve. When the condition causing the trip clears, the solenoid is re-energized; however, the valve must be manually reopened. The control room operator can reopen the trip throttle valve by running the motor operator to the "close" position which relatches the trip valve to the solenoid. The valve can then be opened by running the motor operator to "open".[54]

If the throttle valve is tripped by the mechanical overspeed trip, it must be manually reset locally at the turbine.[57]

F.6 System Isolation

The Primary Containment and Reactor Vessel Isolation Control System initiates automatic isolation of appropriate pipelines which penetrate the primary containment whenever certain monitored variables exceed their predetermined operational limits. The system is designed so that once initiated, automatic isolation goes to completion. Return to normal operation after isolation requires deliberate operator action.

An automatic isolation for the RCIC System causes the normally-open inboard and outboard steam supply isolation valves to close, the turbine to trip, and the two RCIC suction valves from the pressure suppression pool to close (if they were open). The minimum flow bypass valve to the suppression pool is interlocked to close whenever the turbine is tripped.

The following conditions cause a RCIC system isolation signal:

1. RCIC System equipment space high temperature. Since high temperature in the vicinity of the RCIC equipment could indicate a break in the turbine steam supply line, an isolation signal is generated if this temperature exceeds 93.3°C (200°F). This setpoint is based on the calculated ΔT with a 0.001 m^3/s (15 GPM) steam leak in the space. There are 16 temperature sensors arranged in four trip logics with four sensors in each logic. The 16 sensors are physically arranged in four groups with four sensors in each group. One sensor in each group is in each of the four one-out-of-two taken twice trip logics.
2. RCIC System high steam flow. Since high steam flow could indicate a leak in the turbine steam supply line, an isolation signal is generated if the measured steam flow exceeds 150% of design maximum steady state flow. The steamline flow is sensed by two differential pressure switches which monitor the differential pressure across an elbow installed in the RCIC turbine steam supply pipeline. The tripping of either trip channel at a differential pressure of 11.43 m (450 in.) H_2O initiates RCIC System isolation.
3. Low reactor pressure. After steam pressure has decreased to such a low value that the RCIC turbine cannot be operated, and there is no coolant spray to the barometric condenser, the steam line is isolated to protect against continuous gland seal leakage to the RCIC equipment space. The set point for this isolation signal is a reactor vessel pressure of 0.345 MPa (50 psig). The pressure is sensed by four pressure switches at the RCIC turbine steam line upstream of the isolation valves. The switches are arranged in a one-out-of-two taken twice logic.
4. High turbine exhaust diaphragm pressure. A line tapping off the turbine exhaust line contains two rupture diaphrams in series, with the space between vented to the RCIC equipment space through a flow restricting orifice. The diaphrams are designed for 1.138 MPa (150 psig). If the pressure in the space between the diaphragms exceeds 0.172 MPa (10 psig), a system isolation signal is generated.
5. Manual Isolation. If an RCIC initiation signal is present, the operator can cause RCIC System isolation by pushing a control panel pushbutton.

All isolation signals are sealed in and must be manually reset after the condition causing them has cleared. A control panel pushbutton is provided for this purpose.

F.7 Technical Specifications

1. The RCIC System must be operable prior to startup from a cold condition or whenever there is irradiated fuel in the reactor and the reactor vessel pressure is above 0.945 MPa (122 psig).

   If the RCIC System is inoperable, the reactor may remain in operation for a period not to exceed seven days if the HPCI System is operable during such time.
   If these conditions are not met, an orderly shutdown of the reactor must be initiated and the reactor depressurized to less than 0.945 MPa (122 psig) within 24 hours.
2. RCIC testing shall be performed as follows
   a. Simulated automatic actuation test — once per operating cycle
   b. Pump operability — once per month
   c. Motor operated valve operability — once per month
   d. Flow rate at normal reactor operating pressure — once per three months
   e. Flow rate at 1.138 MPa (150 psig) — once per operating cycle
3. Whenever RCIC is required to be operable the piping from the pump discharge to the last flow blocking valve shall be filled. Water flow from the high point vent must be observed monthly. (Purpose is to prevent water hammer when the system is started.)

APPENDIX G

EFFECT OF TVA-ESTIMATED SEVEN HOUR BATTERY LIFE ON NORMAL RECOVERY CALCULATIONS

Section 7, Computer Prediction of Thermal Hydraulic Parameters for Normal Recovery, is based on the assumption that the 250 vdc unit batteries will fail in four hours under the conditions of Station Blackout. After review of the draft results presented in Sect. 7, the Electrical Engineering Branch at TVA performed a battery capacity calculation to determine how long the Browns Ferry 250 vdc batteries would last under these conditions, and arrived at an estimate of seven hours. This appendix examines the impact of the newly estimated seven hour battery failure time on the results and conclusions of Sect. 7.

Two basic conclusions are presented in Sect. 7.

1. If AC power is recovered any time in the first five hours, then a normal recovery will be possible if the 250 vdc batteries have not failed.
2. If the 250 vdc batteries fail at four hours and the AC power remains unavailable then the time between battery failure and first uncovering of fuel will be at least three hours if the reactor has been previously depressurized as recommended by this report.

If the unit batteries were to last seven instead of four hours, there would be a slight lengthening of the time interval from battery failure to core uncovery due to the slightly lower decay heat. The most important question to be answered regarding the extension of the estimated battery failure time from four to seven hours is whether there is some other system failure caused by the increased temperatures and/or pressures during this period that would complicate or make impossible a normal recovery if AC power were restored.

To answer this question calculations similar to those shown in Figs. 7.1 through 7.9 of Sect. 7 were performed. These new calculations start at an initial point four hours after station blackout and extend to seven hours. The batteries are assumed to last throughout this period and operator actions are the same as detailed in Sect. 7.3.1, Normal Recovery-Assumptions.

Due to the elevated pool temperatures experienced during the period from four to seven hours, it has been necessary to modify the calculation of the fraction of relief valve (SRV) discharge that is quenched in the suppression pool. When the pool temperature is within its normal operational range, the SRV discharge is completely condensed (quenched) in the cool water surrounding the submerged SRV discharge nozzle (T-quencher). However, if the temperature of the water around the T-quencher is sufficiently close to saturation, then the condensation will be less than 100% and some of the steam will reach the suppression pool atmosphere. Monticello tests[58] showed that an approximately 28°C (50°F) difference between bulk and local pool temperature can exist during extended discharge through a single SRV. In this case, one would expect less than 100% quenching to begin at a bulk pool temperature of about 74°C (165°F), corresponding to 100°C (212°F) near the SRV discharge point (provided the pool is at atmospheric pressure). For the purposes of the calculations reported in this appendix, the following provisions were made for the calculation of quench fraction in the pool:

1. The temperature of the water surrounding the T-quencher during SRV discharge is assumed to be 28°C (50°F) higher than the bulk pool temperature.
2. If the vapor pressure of the water surrounding the T-quencher is equal to or greater than the static pressure then 0% of the SRV discharge is quenched; if the vapor pressure is 5.0 psi or more below the static pressure then 100% of the discharge is quenched. Variation of quench fraction is linear between these two points.

This should provide a reasonable upper limit estimate of containment pressurization due to non-quenched SRV discharge. Consideration of this effect was not necessary for the zero to five hour results reported in Sect. 7 because bulk pool temperature does not exceed 82°C (180°F) during the first five hours of a Station Blackout.

Results of the four to seven hours after blackout calculations are shown in Figs. G.1 through G.9. From these results it is concluded that system parameters remain within acceptable ranges during this period:

1. The 250 vdc batteries by TVA estimate last the full seven hours.
2. Reactor vessel level is within the normal control range, about 5.08 m (200 in.) above the top of active fuel.
3. Reactor vessel pressure is being controlled by operator action at about 0.69 MPa (100 psia).
4. At the seven-hour point, about 414 m^3 (109,265 gal) of water have been pumped from the condensate storage tank, which had an assured capacity of 511 m^3 (135,000 gal) before the blackout.
5. At the seven-hour point, average suppression pool temperature is about 92°C (198.2°F), but this should not be a problem for the T-quencher type of SRV discharge header piping.
6. Containment pressures would be elevated to about 0.23 MPa (33 psia), well below the 0.53 MPa (76.5 psia) design pressure.
7. Drywell atmosphere temperature is about equal to the 138°C (281°F) design temperature.

These results show that if power were recovered within seven hours of the Station Blackout a normal recovery would be possible.* The 92°C (198°F) bulk pool temperature after seven hours may cause elevated air temperature in the RCIC and HPCI spaces, but this would not be expected to lead to failure of the HPCI or RCIC, because the lube oil of both units is cooled by the water being pumped -- in this case, water from the condensate storage tank which would not exceed 32°C (90°F). The periods of RCIC operation are very infrequent after the first four hours of Station Blackout, with about an hour between actuations.

The supply of control air for remote-manual operation of the SRVs is sufficient for the full seven hours. As discussed in Chap. 3 of this report, the accumulators provided for the six relief valves associated with the ADS system are sized to permit five operations or a total of 30 actuations. As illustrated in Sect. 7 and this appendix, less than 30 actuations are required during the first seven hours of the Station Blackout.

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*With the assumption of no independent secondary equipment failures.

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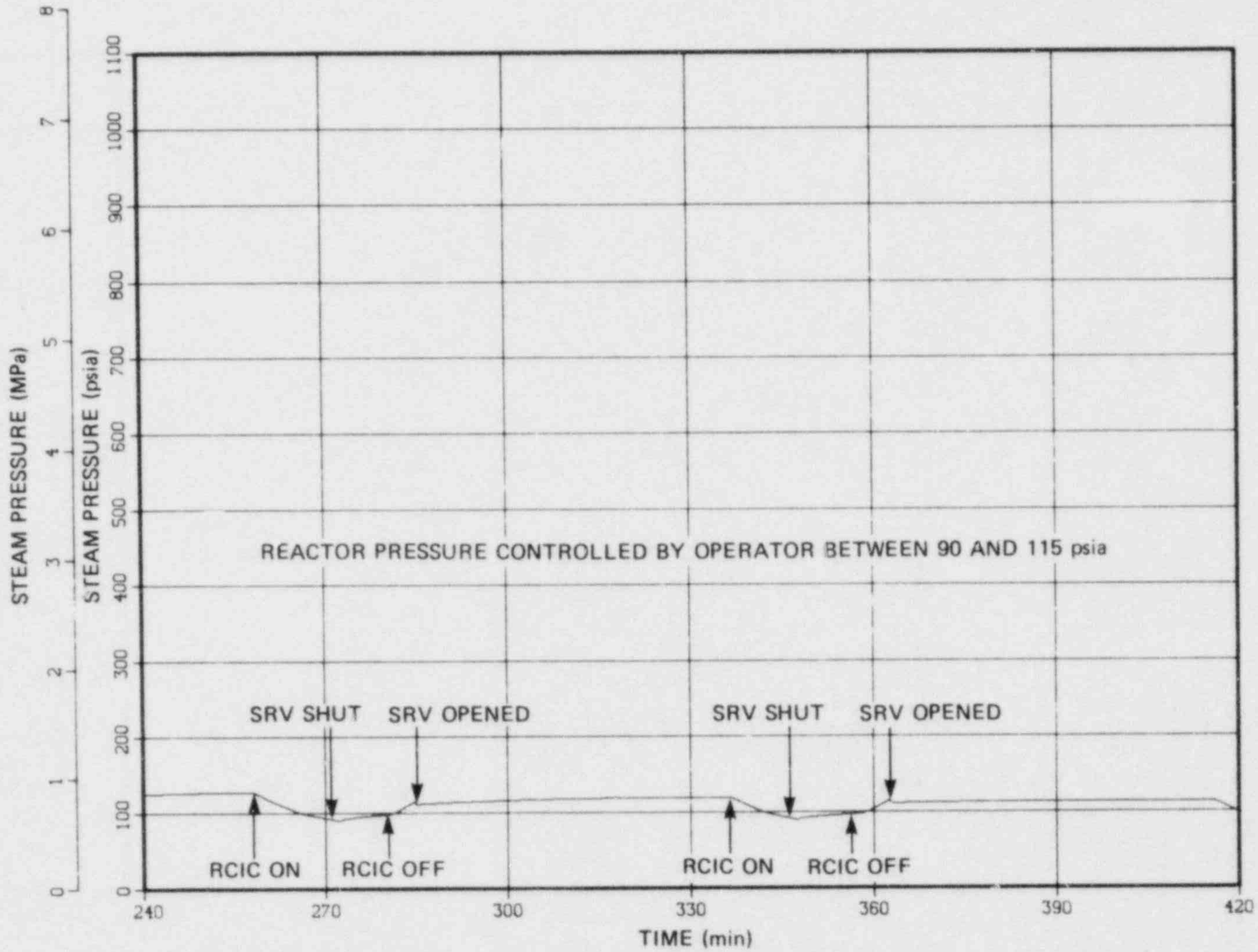

Fig. G.1 Reactor vessel pressure - four to seven hours after station blackout.

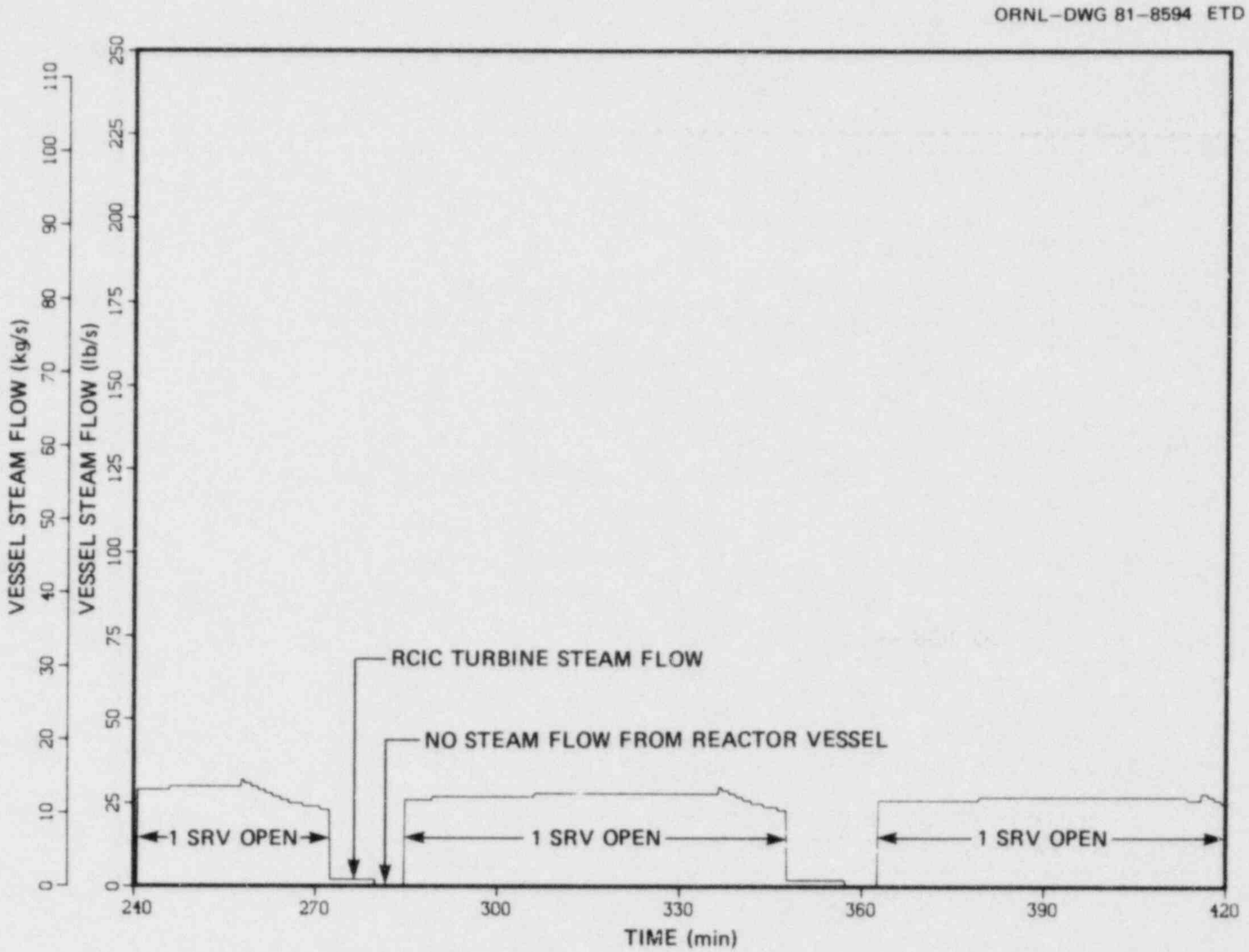

Fig. G.2 Vessel steam flow - four to seven hours after station blackout.

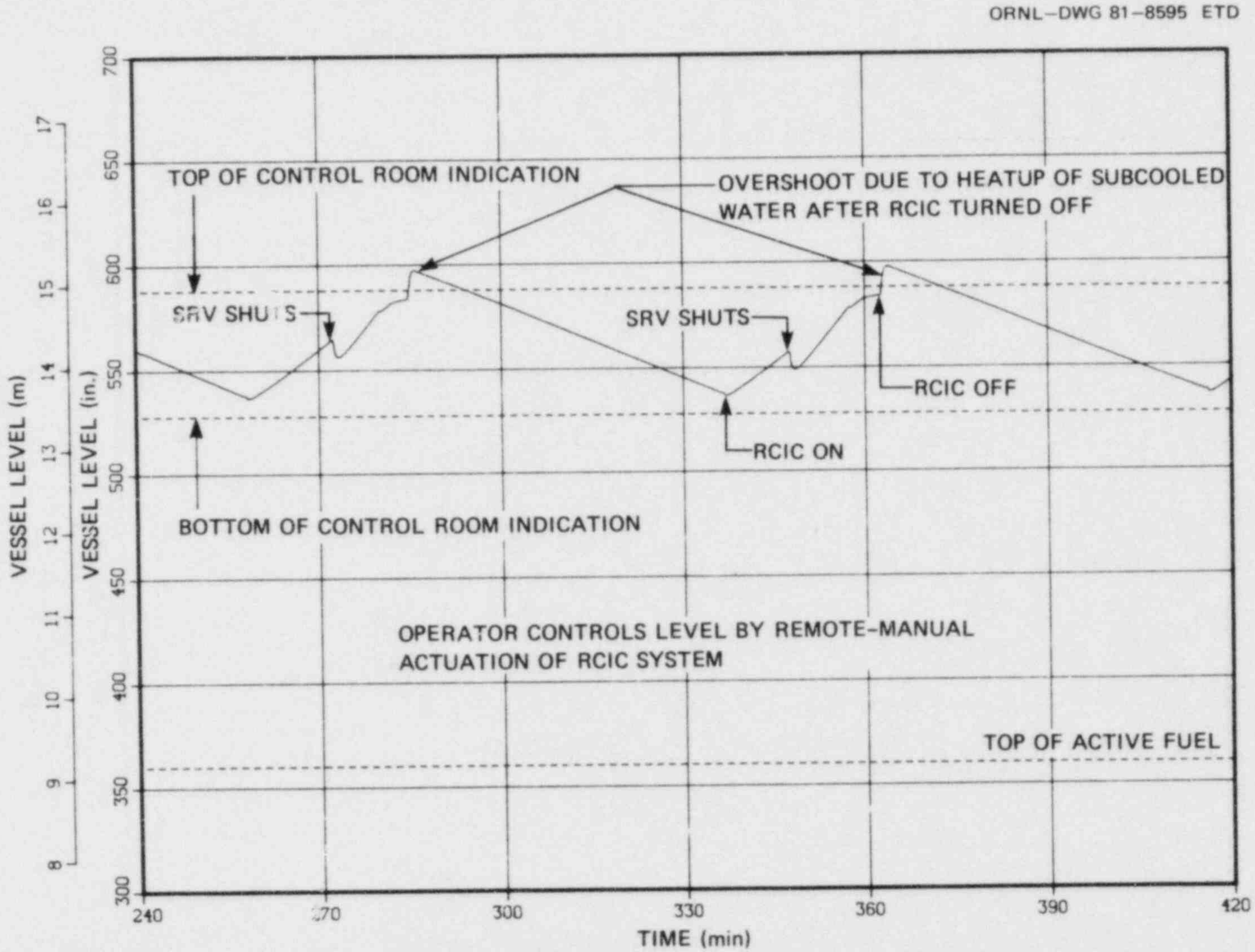

Fig. G.3 Vessel level - four to seven hours after station blackout.

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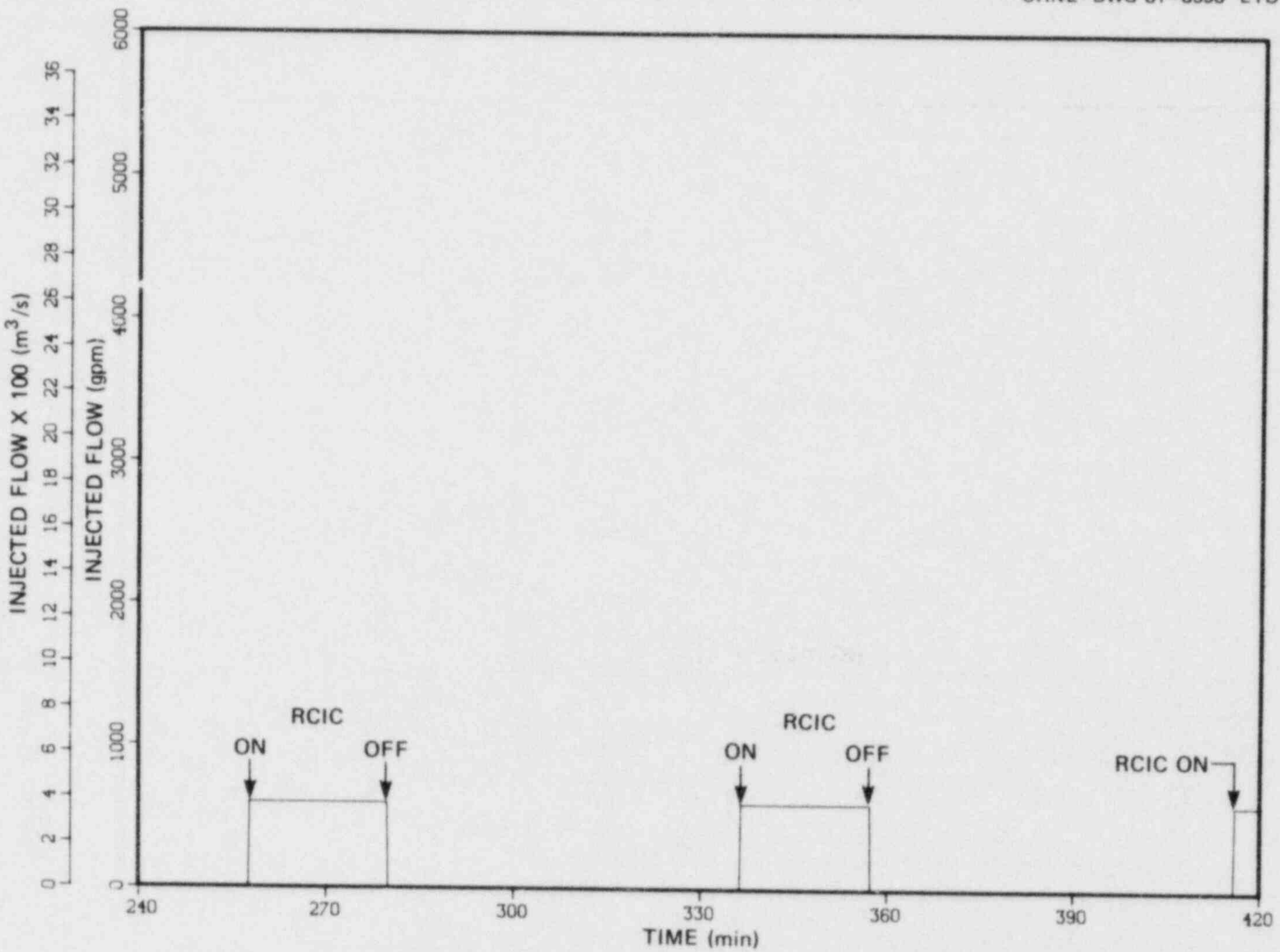

Fig. G.4 Injected flow - four to seven hours after station blackout.

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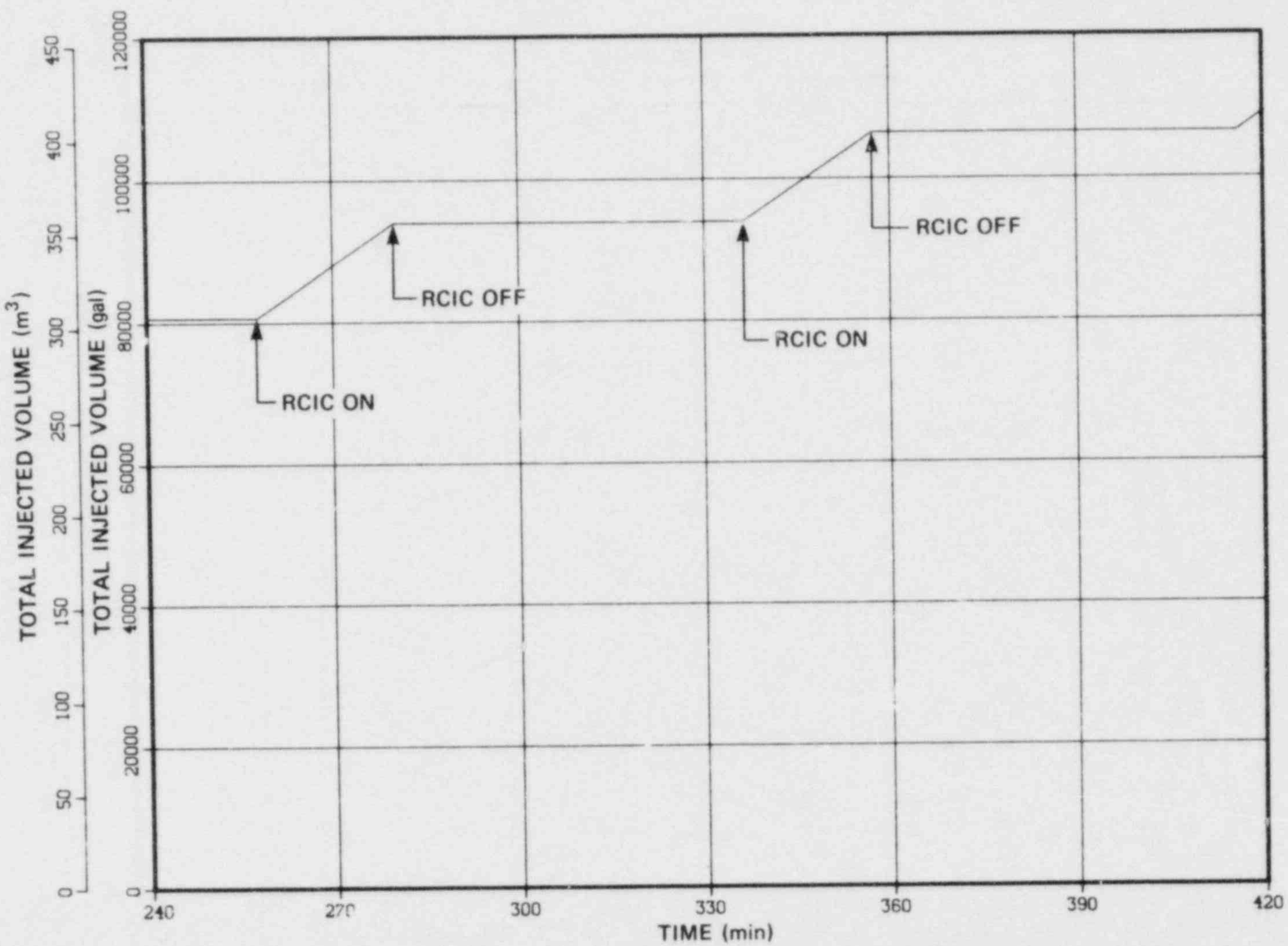

Fig. G.5 Total injected flow (pumped by RCIC from condensate storage tank) - four to seven hours after station blackout.

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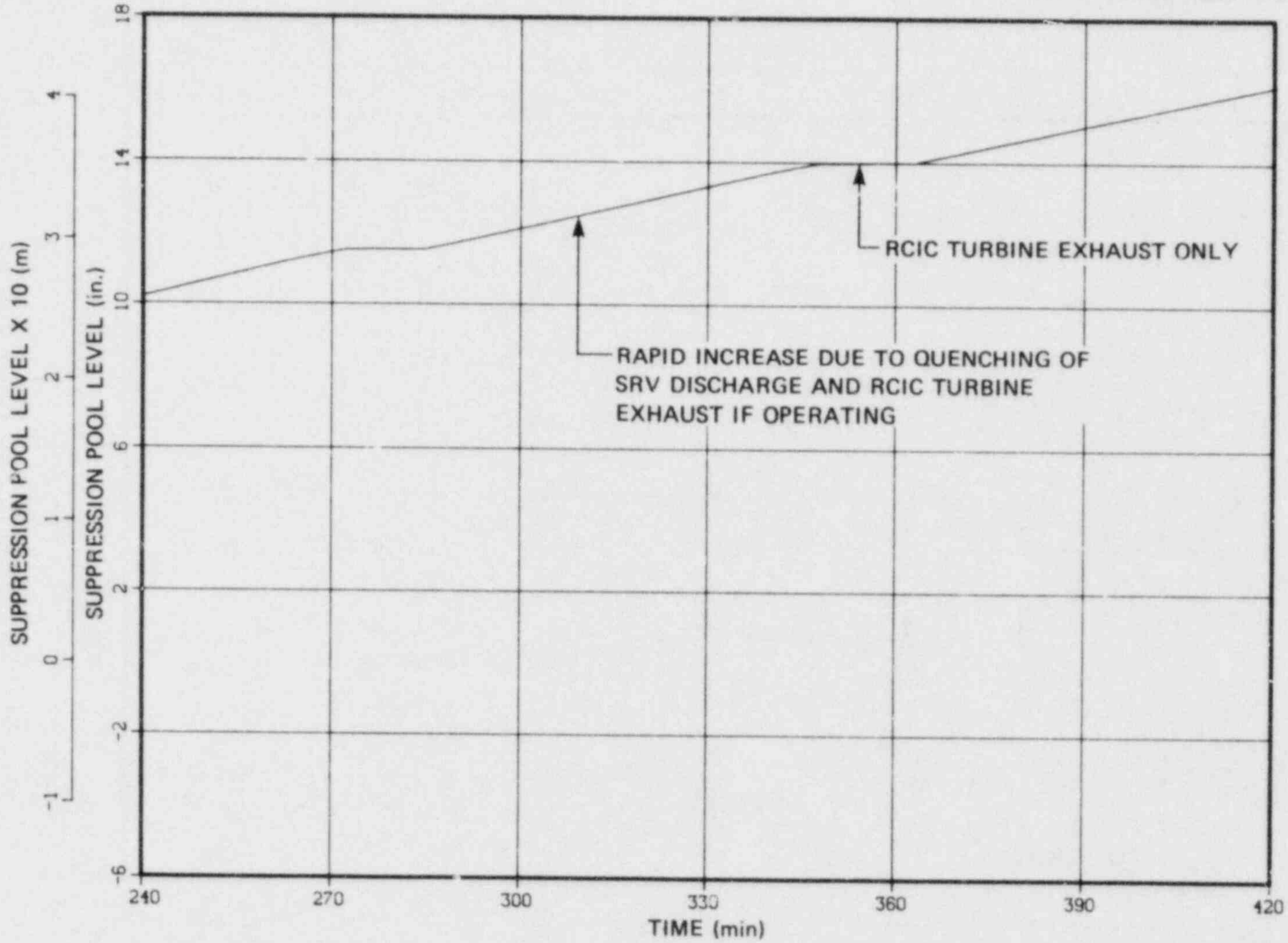

Fig. G.6 Suppression pool level - four to seven hours after station blackout.

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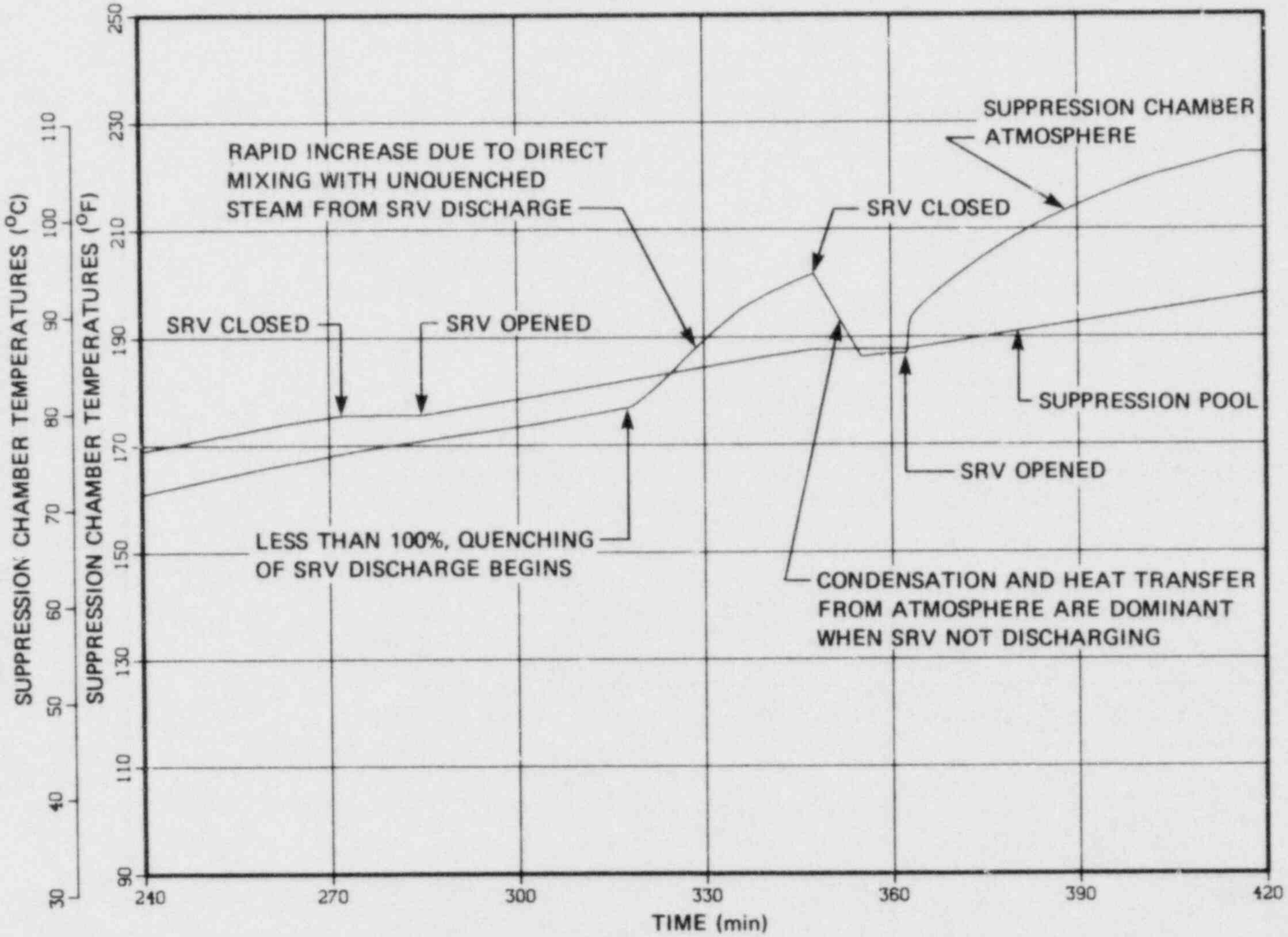

Fig. G.7 Suppression chamber temperatures - four to seven hours after station blackout.

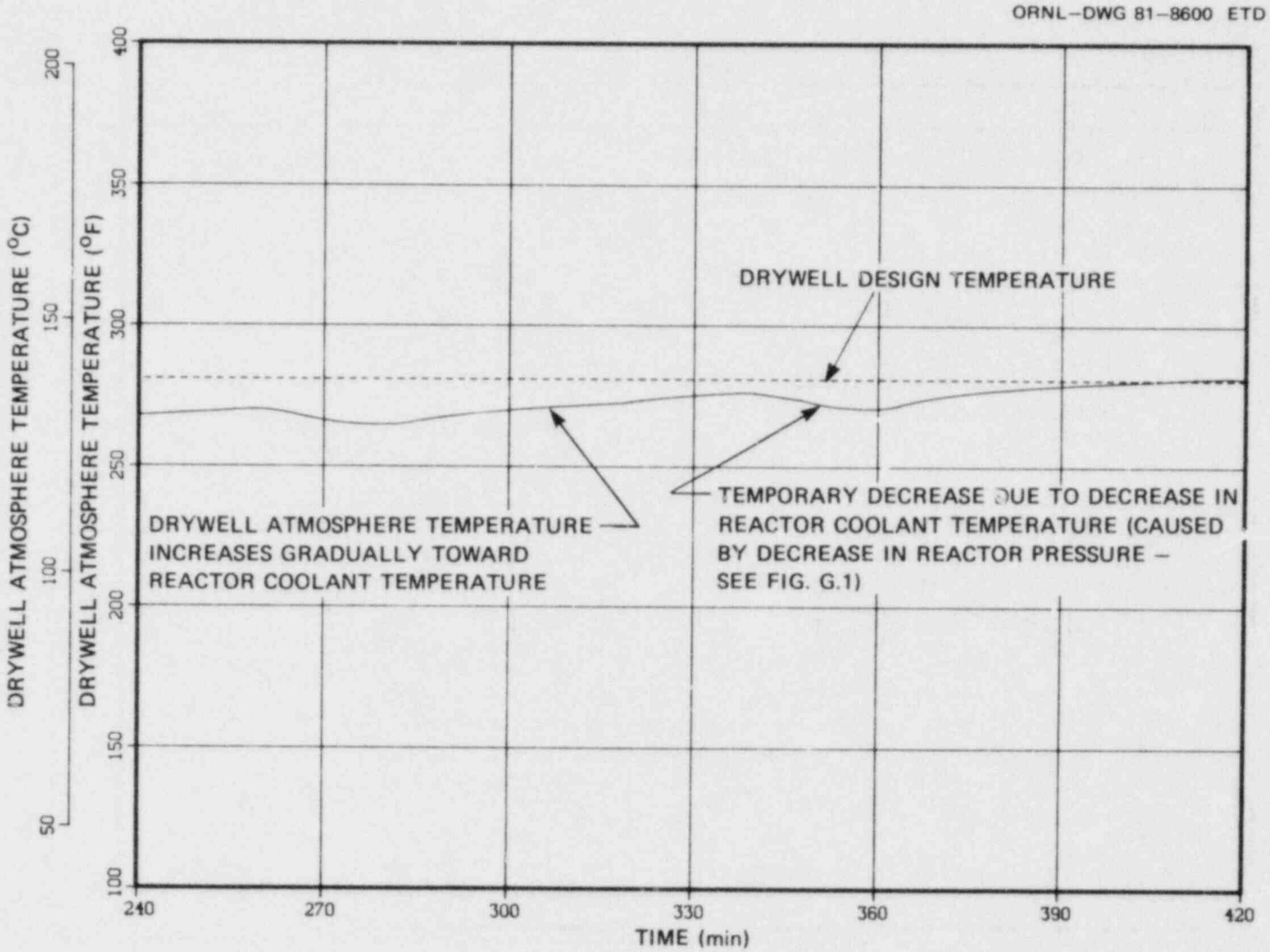

Fig. G.8 Drywell temperatures - four to seven hours after station blackout.

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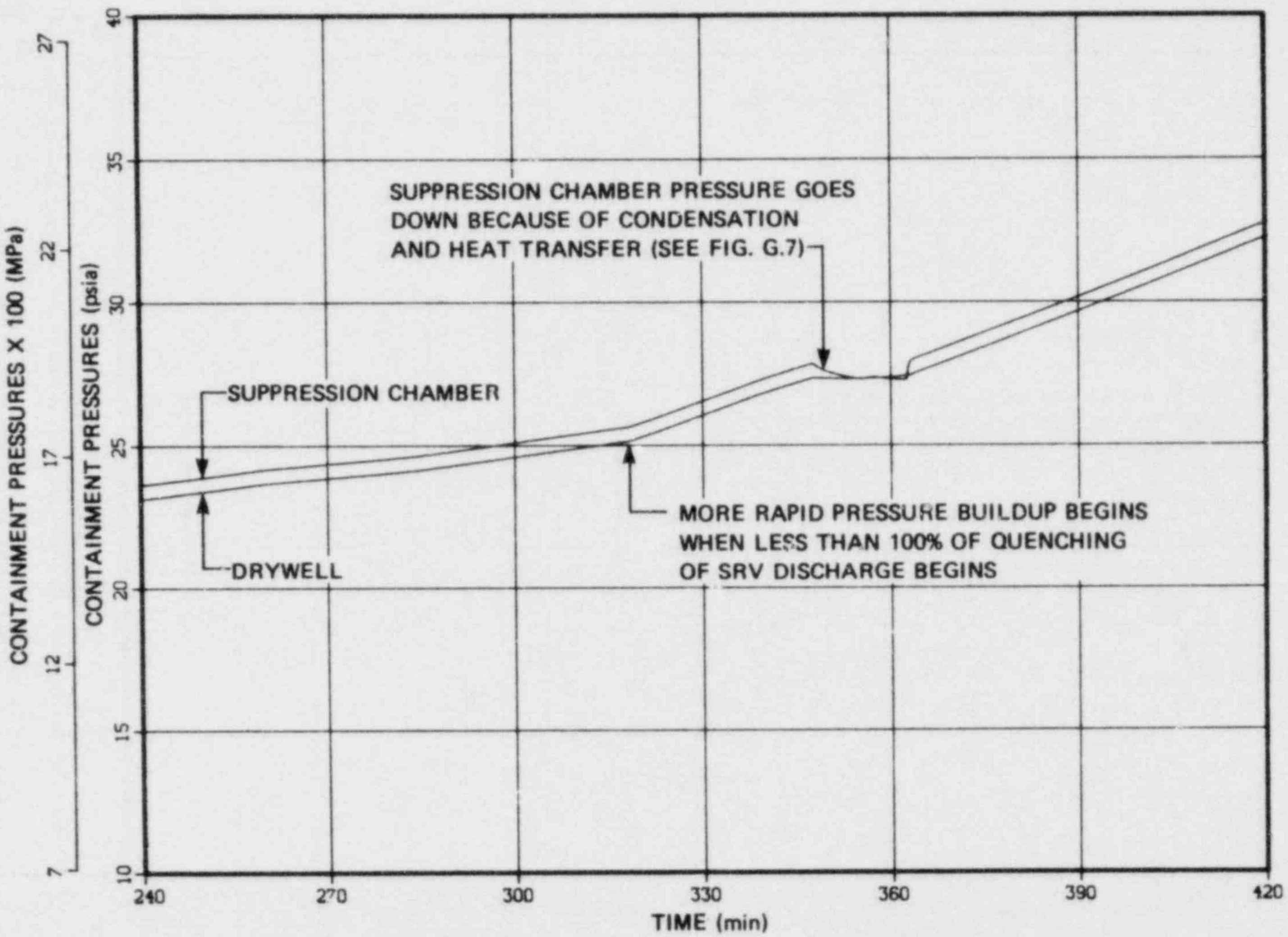

Fig. G.9 Containment pressures - four to seven hours after station blackout.

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