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Loss of Control Air at Browns Ferry Unit One—Accident Sequence Analysis

R. M. Harrington
S. A. Hodge

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

This study describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to a postulated complete loss of plant control air compounded by an assumption of failure-upon-demand of both of the unit emergency high pressure injection systems, Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI). This accident sequence was identified as a possible significant contributor to plant severe accident risk in studies carried out jointly by the Tennessee Valley Authority (TVA) and the firm of Pickard, Lowe, and Garrick in support of the TVA-sponsored probabilistic risk assessment of the Browns Ferry Nuclear Plant.

The loss of plant control air is a safety concern because of the close grouping of the plant control air compressors, the relative unreliability of the high pressure injection systems, and because, at Units 1 and 2, the continuity of the drywell control air supply depends upon the availability of plant control air pressure to hold open the drywell control air compressor suction isolation valves.* The capability for remote operation of the safety relief valves (SRVs), in turn, depends upon the availability of drywell control air pressure. If the compressed air stored in the drywell control air system receivers and in the small individual accumulators attached to the six SRVs associated with the Automatic Depressurization System (ADS) becomes depleted, remote operational capability of the SRVs would be lost, and the reactor vessel could not subsequently be depressurized. Alternatively, if the reactor vessel had previously been depressurized, it would now return to full pressure. The low pressure injection systems, although operational, could not then be used to charge the vessel, and with the assumption that the HPCI and RCIC systems are both failed at Unit 1, a high pressure boil-off and core uncovery would follow.

The postulated total loss of plant control air would not compromise the containment heat removal functions of the Residual Heat Removal RHR) systems because all of the essential valves in this system are motor-operated and do not depend upon the availability of control air.

The BWR-LTAS code, developed at Oak Ridge National Laboratory (ORNL), has been used in this study to predict the timing of accident sequence events and to assess the efficacy of potential operator actions that might be taken to prevent the sequence from degrading into a severe fuel damage accident. Calculations have been performed both for cases with and without operator action.

For the case without operator action, the results of the study demonstrate that the automatic actions of the plant protective systems would provide adequate core cooling for about 16 h after the accident-sequence-initiating loss of plant control air. The reader is reminded that it is essential to the structure of this postulated severe accident sequence that both RCIC and HPCI are assumed to fail on demand, for

*At Unit 3, the drywell control air system is completely independent of the plant control air system.

completely independent causes, at Unit 1.* As a result, the reactor vessel water level falls, and when the conditions for automatic reactor vessel depressurization are met, the reactor vessel is depressurized by the opening of the six ADS SRVs. The automatically-actuated Core Spray and RHR systems† then flood the reactor vessel. The SRVs are assumed, based upon available experimental data, to remain open throughout the transition from steam to water discharge. The low-pressure injection systems keep the reactor vessel flooded for about 6 h, but then the open SRVs are predicted to shut because of insufficient remaining control air pressure. The water-filled reactor vessel repressurizes, steam production resumes, and a bubble is drawn. The 105 gpm (0.0066 m^3/s) of injection provided by the control rod drive (CRD) hydraulic system is insufficient to offset the loss of inventory, so reactor vessel water level steadily decreases and core uncovery begins about 15 h after the automatic reactor scram on low control air pressure.

Available laboratory test data show that the SRVs will remain open during periods of steam discharge and during periods of water discharge, but do not cover the transition. In the very unlikely event that the SRVs cannot remain open when the transition from steam to water discharge occurs, then the reactor vessel would be overpressurized soon after the inception of flooding.

The results of the no-operator-action case study demonstrate that the plant should protect itself adequately and that plenty of time would be available for the operators to initiate mitigating actions. However, the short period of time required for the low-pressure injection systems, once initiated, to complete vessel flooding, ~3 min, dramatizes the rapidity with which the operators must act if overfilling of the reactor vessel is to be avoided (e.g., by tripping 7 of the 8 low-pressure ECCS injection system pumps). This would keep water away from the SRV inlets and the HPCI/RCIC steam supply lines.

Two different operator action cases have been analyzed in this study, also using the BWR-LTAS code. These two cases were selected to encompass the expected range of operator responses. In the first case, the operators attempt to maintain a safe state without depressurizing the reactor vessel. In the second case, the operators act to depressurize the reactor vessel within the first half hour after the automatic reactor scram with the objective of going into shutdown cooling. Nevertheless, after capability for remote SRV operation is lost in the second case, the reactor vessel repressurizes. Therefore, although the two cases start very differently, they reach similar states and the end result is the same for both; eventually, the reactor vessel will be

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*This, of course, also reduces the probability of the accident sequence.

†The condensate booster pumps are assumed failed because of the long period of time that they would have been operated at shutoff head without a recirculation path available; valves in the recirculation path fail closed on loss of plant air.

pressurized, the low-pressure injection systems cannot charge the vessel, and the core will be uncovered unless the operators take action to enhance the cooling water injection into the vessel by the CRD hydraulic system.

At Browns Ferry, the appropriate action to enhance CRD hydraulic system flow would be to initiate flow through the normally isolated Pump Test Bypass Line, which has a direct path into the reactor vessel via a feedwater line. This action, in conjunction with opening the pump discharge throttling valve, would yield an injected flow of about 200 gpm (0.0126 m /s) with one CRD hydraulic system pump running and about 300 gpm (0.0189 m /s) if the operator acts to start the spare pump. The potential for still further augmented flow by initiating injection by the Standby Liquid Control (SLC) system [50 gpm (0.0032 m /s) injection capability] is available in the operator action cases but has not been considered, because the flow available with the CRD hydraulic system via the Pump Test Bypass Line is demonstrated to be sufficient to prevent core uncovery. Such might not be the case for other BWR plants that do not have the Pump Test Bypass Line.*

The main conclusion that can be drawn from the results of the operator-action case studies that are presented here is that the operators can avoid core uncovery by using the CRD hydraulic system as a standby high pressure injection system. Alternatively, the reactor vessel could be maintained at low pressure by using the controls available at the Backup Control Panel to override the high drywell pressure interlock on the shutdown cooling suction valves. Low-pressure injection systems, whose operability does not depend upon the availability of control air, could then be used to keep the core covered.

Recommendations developed from the results of this study include measures that would increase the probability that the operators could achieve a success path using the mitigation measures cited above. However, the models employed in the BWR-LTAS code to estimate the potential for enhanced CRD hydraulic system flow have never been checked against data from actual tests (which currently does not exist). The practicality of a plant test to verify the maximum CRD hydraulic system flow capability should be evaluated, and a test conducted, if feasible. Also, an upgrade of the CRD hydraulic system flow indication range available to the control room operator should be considered. Although much greater flows are possible, the upper limit of the present control panel meter is only 100 gpm (0.0063 m /s), a flow insufficient to prevent core uncovery or significant core damage. Operator procedures and training should make it clear which valves must be opened, specify the location of the valves in the reactor building, and list the sequence in which they should be operated to enhance the CRD hydraulic system flow. The feasibility of overriding the high drywell pressure interlock

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*Injection requirements are discussed in Appendix D. Core uncovery can be prevented by a continuous injection rate of 225 gpm. Rates as low as 170 gpm prevent significant core damage although temporary core uncovery does occur (see NUREG/CR-3179).

that prevents operation of the shutdown cooling suction valves should also be evaluated, and instructions for execution of this mitigation strategy should be included in emergency procedures.

As discussed previously, the threat to reactor safety caused by loss of drywell control air pressure is a result of the consequent inability to operate the safety/relief valves and maintain the reactor vessel depressurized so that the low-pressure systems can be used for injection. It should be noted that the TVA has committed to provide a safety-grade, long-term depressurization capability for the six safety/relief valves associated with the Automatic Depressurization System by installing supply lines from the nitrogen supply trains of the Containment Atmosphere Dilution (CAD) system. This improvement should reduce, to an insignificant level, the probability that Loss of Control Air or any other accident sequence involving loss of the Drywell Control Air System will lead to an inability to depressurize the reactor vessel.

LOSS OF CONTROL AIR AT BROWNS FERRY UNIT ONE — ACCIDENT SEQUENCE ANALYSIS

R. M. Harrington
S. A. Hodge

ABSTRACT

This study describes the predicted response of the Browns Ferry Nuclear Plant to a postulated complete failure of plant control air. The failure of plant control air cascades to include the loss of drywell control air at Units 1 and 2. Nevertheless, this is a benign accident unless compounded by simultaneous failures in the turbine-driven high pressure injection systems. Accident sequence calculations are presented for Loss of Control Air sequences with assumed failure upon demand of the Reactor Core Isolation Cooling (RCIC) and the High Pressure Coolant Injection (HPCI) systems at Unit 1. Sequences with and without operator action are considered. Results show that the operators can prevent core uncovery if they take action to utilize the Control Rod Drive Hydraulic System as a backup high pressure injection system.

1. INTRODUCTION

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

*Previous reports concern Station Blackout (NUREG/CR-2181), Scram Discharge Volume Break (NUREG/CR-2672), Loss of Decay Heat Removal (NUREG/CR-2973), Loss of Injection (NUREG/CR-3179) and ATWS (NUREG/CR-3470) accident sequences.

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

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

This report presents a study of the predicted sequence of events during a postulated complete loss of control air compounded by the failure-on-demand of both the HPCI and the RCIC high pressure injection systems at Unit 1 of the Browns Ferry Nuclear Plant. This accident category was selected for analysis because it has been identified as a possible contributor to total plant risk in the TVA-sponsored Probabilistic Risk Assessment (PRA) of the Browns Ferry Plant. The postulated complete loss of control air begins with the loss of the plant control air system. As described in Chapter 2, the loss of plant control air pressure causes the loss of the drywell control air system at Units 1 and 2. Therefore, a complete loss of control air at two of the units is the ultimate result of the initial loss of the plant control air system.

Even after a complete and sudden failure of the plant control air system, there would not be an immediate reactor scram. The discussion of Chapter 3 demonstrates how the decaying control air pressure would lead to the transient-initiating scrams from full power at all three units, and outlines which systems would fail as the compressed air remaining in the receivers, the Automatic Depressurization System safety/relief valve accumulators, and the distribution lines is depleted. With an associated loss of the drywell control air systems at Units 1 and 2, and an assumed loss of the HPCI and RCIC systems at Unit 1, a Severe Accident would develop at Unit 1.

The principal tool for this analysis of the Loss of Control Air accident sequence is the BWR-LTAS code. This code, developed by R. M. Harrington at ORNL, has also been used in previous ORNL SASA studies to define accident sequence events up to, but not including, permanent core uncovery and/or severe fuel damage. The accident calculations reported here are terminated at permanent core uncovery; however, for sequence branches leading to severe fuel damage, the time of initial permanent core uncovery is reported.

Previous SASA studies have shown that the determination of the effect of operator actions upon the progression of an accident sequence is facilitated if the accident sequence of events is first established for the case without operator action. This procedure is also followed in this study. The loss of control air accident sequence without operator action is the subject of Chapter 4.

The two basic operator action accident sequences discussed in Chapter 5 are intended to encompass the range of likely operator strategies that might be taken to protect the reactor and containment. For the case discussed in Chapter 5.2, the operators attempt to maintain core cooling flow without depressurizing the reactor vessel. For the other case, Chapter 5.3, the operators depressurize the reactor vessel in order to restore vessel water level and implement shutdown cooling with the low-pressure injection systems. The possible effects of failure of the Control Rod Drive Hydraulic System and of stuck-open safety/relief valves are considered for each operator-action sequence.

Uncertainties in the BWR-LTAS calculational model and uncertainties with regard to the assumptions of operator action and the assumed timing of equipment failures are discussed in Chapter 6.

Chapter 7 is devoted to a discussion of the major conclusions of this study and provides a detailed summary of implications concerning the adequacy of system design, plant equipment, and operator training.

The computer code used for the calculations of this study is described in the report "BWR-LTAS: A Boiling Water Reactor Long-Term Accident Simulation Code," NUREG/CR-3764. Primary system calculations for the portion of a severe accident sequence before core uncovery are much simpler for a BWR than for a Pressurized Water Reactor (PWR). For the PWR, consideration must be given to hot leg, pressurizer, steam generator, and cold leg. For the BWR, the low reactor vessel water level that is common to all BWR severe accident sequences would ensure that the reactor vessel is isolated and that the recirculation pumps are tripped; thus the core inlet flow would be a function only of the amount of makeup water injection and the effect of natural recirculation circuits within the reactor vessel. Therefore, sophisticated primary system analyses codes such as RELAP5, RETRAN, or TRAC are usually not necessary for BWR severe accident calculations; fundamental modeling of the processes within the reactor vessel in a properly benchmarked relatively simple code such as BWR-LTAS is sufficient. Appendix A provides a description of the additions and improvements made to BWR-LTAS to provide the special capabilities* needed for the loss of control air calculations.

Appendix B presents the results of BWR-LTAS calculations and analytical calculations made to predict the maximum possible pressure suppression pool temperature after shutdown at Unit 1 in the unlikely event that only one of the four associated residual heat removal system heat exchangers is available for pool cooling. The analytical model developed to permit a closed-form solution for the maximum pressure suppression pool temperature is described in Appendix C. It should be recognized that these results are conservative since there is no reason to believe that all four of the RHR system heat exchangers would not be available during the accident sequence.

Experience has shown that control room operators and other interested personnel often greatly overestimate the minimum required rate of reactor vessel injection flow to keep the BWR core covered during

*Discharge of water through SRVs, heat removal from primary coolant by the shutdown cooling system, failure of remote-manual operability of SRVs by loss of control air pressure, reduced conservatism in drywell and wetwell heat sink models.

accident sequences other than LOCA. This is probably because of their recognition of the enormous capacity of the low-pressure ECCS systems, which are installed to protect against the consequences of large-break LOCA. Nevertheless, the results of this study demonstrate again that the electric motor-driven control rod drive hydraulic system pumps, which have a relatively small capacity, can serve as an effective backup to the steam turbine-driven high-pressure injection systems for accident sequences other than LOCA. Appendix D provides background information concerning the need for reactor vessel injection following reactor scram in non-LOCA situations and the potential of the Control Rod Drive Hydraulic System to satisfy this relatively small requirement.

2. CONTROL AIR SYSTEMS

The purpose of this chapter is to describe the compressed air systems at the Browns Ferry Nuclear Plant and the important dependence of the Drywell Control Air Systems of Units 1 and 2 upon the Plant Control Air System. Without this dependence, the potential for a severe accident sequence initiated by loss of plant control air would not exist, since the accident would be reduced to a loss of feedwater, which the plant is designed to handle.

There are three types of compressed air distributed throughout the Browns Ferry Nuclear Plant: plant service air, plant control air, and drywell control air. Plant service air, used for maintenance and general plant service, is discussed only briefly in this report. Plant control air and drywell control air, on the other hand, are moisture-free, high-quality sources used for the pneumatic operation of valves and instrumentation. The Plant Control Air System is described in this section.

2.1 Plant Control Air System

The Plant Control Air System is shown schematically in Fig. 2.1. Four compressors, each rated at 610 scfm (0.288 m^3/s), are installed at adjacent locations in the Unit 1 Turbine Building. These compressors serve all three units through a common discharge line that feeds three 266 ft^3 (7.532 m^3) receivers. The discharge from the receivers is routed to three individual headers, one for each unit. A cross-connection permitting service air backup to plant control air ties in at the common receiver discharge.

The remainder of this discussion will pertain to loads fed from the Unit 1 plant control air header; Unit 2 and Unit 3 header configurations and loads are similar, except as noted.

The first components fed by the Unit 1 header are the Unit 1 dryer and the standby dryer, which is common to all three units. The flow through the dryer is regulated by an air-operated flow control valve (not shown) that would close automatically upon a loss of air pressure or electrical power. Flow leaving the dryer is filtered and then divided among four distribution headers that serve various loads in the Turbine Building and the Reactor Building.

The Plant Control Air System loads important to plant safety are located in the Reactor Building and are represented in the lower portion (beneath the dashed line) of Fig. 2.1. It should be noted that these loads are served by three distinct air supply lines, each provided with an isolation valve on the turbine building side and a check valve on the reactor building side of the point of entry to the Reactor Building. This is to provide capability for secondary containment isolation in the event of piping breaks in the Turbine Building.

To the left of Fig. 2.1, fed through valve 32-28, are shown the supply to the drywell control air compressor suction isolation valves,

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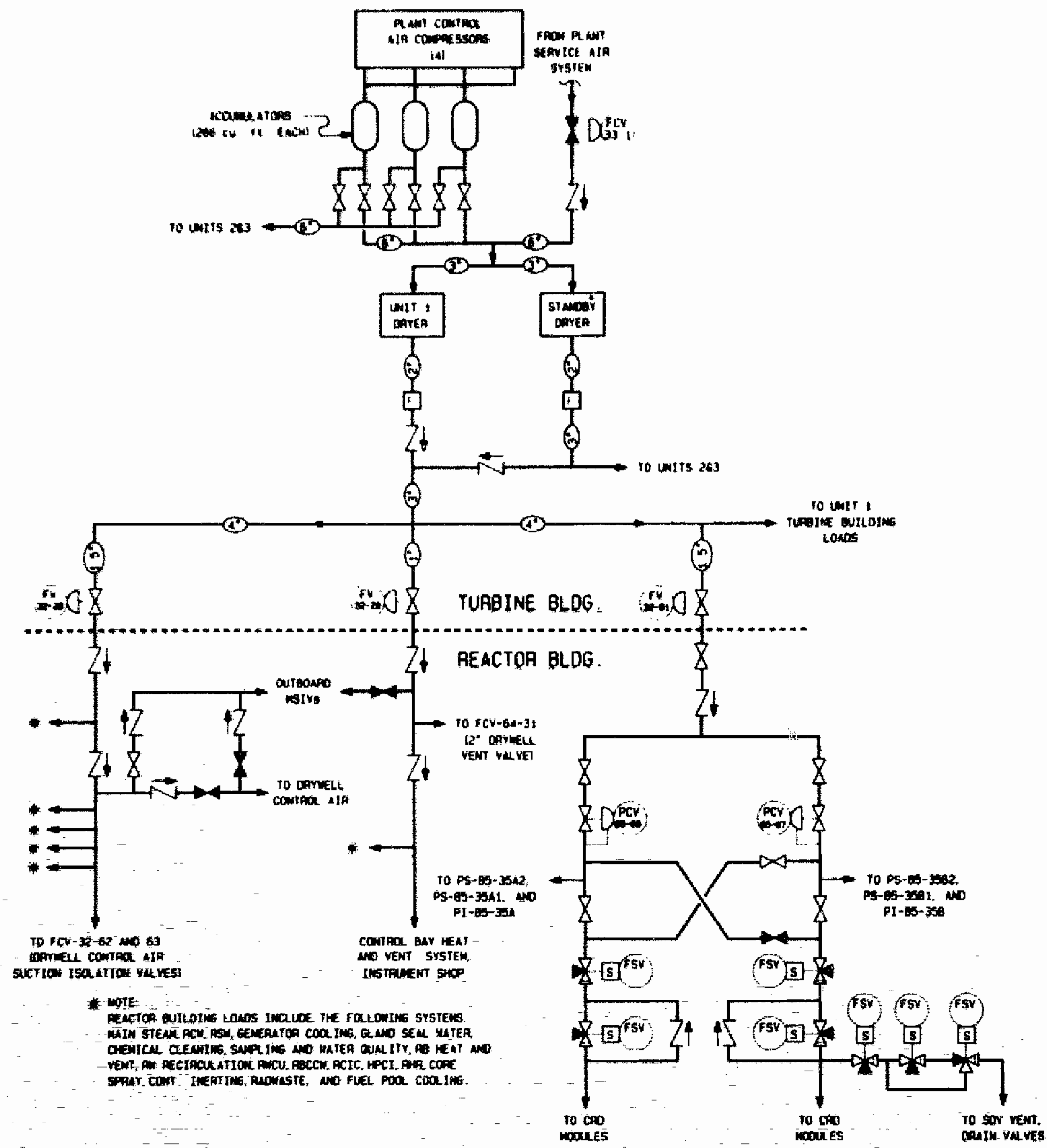

Fig. 2.1. Browns Ferry Plant control air system with emphasis on the portion associated with Unit 1.

the normal connection to the outboard main steam isolation valves (MSIVs), and the normally-shut crosstie to the Drywell Control Air System itself. In the center, fed through valve 32-29, are shown the backup supply to the outboard MSIVs, the supply to the 2-in. drywell venting valves, and the supply to the control bay heating and ventilation system. Other loads fed through valves 32-28 and 32-29, but not specifically indicated on Fig. 2.1, include the 18-in. drywell and wetwell venting valves and various components of the following systems: Main Steam, Raw Cooling Water (RCW), Reactor Service Water (RSW), Reactor Water Cleanup (RWCU), Reactor Building Closed Cooling Water (RBCCW), Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), Residual Heat Removal (RHR), and Core Spray (CS). Additional information concerning these loads is provided in Table 2.1.

The Plant Control Air System supply to the Unit 1 scram system, fed through valve 32-91, is shown on the lower right of Fig. 2.1. The connections to pressure switches (PS) shown just downstream of the pressure control valves (PCVs) are the result of recent modifications that provide a reactor scram signal upon low air pressure. This is to preclude a situation in which the Scram Discharge Volume was prematurely filled because the scram outlet valves associated with several of the individual control rods were individually opened in some haphazard manner as plant control air pressure slowly decayed due to some mishap at its source.

The function of plant control air in the operation of the Reactor Protection System can best be explained by reference to Fig. 2.2, which provides an expanded view of the associated piping configuration. At the upper left, within the dashed square, are shown the air-operated valves within a single hydraulic control unit (HCU). Each of the 185 control blades per unit has its own HCU and individual blade scram is accomplished when the associated scram inlet and scram outlet valves are opened. The Scram Discharge Volume (SDV) catches the water discharge from the above-piston volumes of all of the control rod drive mechanism assemblies as the control blades are driven into the core. The SDV is drained [through the Scram Discharge Instrument Volume (SDIV)] and vented during normal reactor operation but becomes isolated and serves as a catch tank during scram. For additional information concerning the operation of the control rod drive hydraulic system during scram, the reader is referred to Appendix E of NUREG/CR-2672, Volume 1.*

As previously discussed, each HCU comprises the scram inlet valve and the scram outlet valve for its associated control rod drive mechanism. These scram valves are air-operated globe valves with Teflon seats, held closed by control air pressure during normal reactor operation and snapped open by internal springs when air pressure is removed. A schematic of the plant control air supply to the air-operators of these valves is included in Fig. 2.2. As shown, the control air

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*Plant modifications in progress improve the reliability of the scram system, including the provision of two Scram Discharge Instrument Volumes (SDIVs).

Table 2.1. Reactor building loads supplied by Plant Control Air System

I. Loads supplied via valve 32-28

a. purge air for TIP system drive mechanisms

b. backup supply for drywell control air

c. outboard MSIV operators

d. drywell floor drain and equipment drain sump pump outlet inboard and outboard isolation valve operators.

e. drywell equipment drain sump valve operators for pump discharge to radwaste and recycle to heat exchanger.

f. RHR system valve operators for system vents to suppression pool and head spray flow control valve.

g. RCW system valve operators for 1A RBCCW heat exchanger control, 1B recirculation system motor generator and oil coolers, 1B drywell air compressor valve jacket and aftercoolers.

h. valve operator for RBCCW supply to RWCU system non-regenerative heat exchangers

i. RWCU system valve operators

j. valve operators for primary containment ventilation system

k. valve operator for recirculation system sample line outboard containment isolation

l. valve operators for RWCU sample system control network.

m. valve operators for fuel pool cooling system

II. Loads supplied via valve 32-29.

a. alternate supply to outboard MSIVs

b. backup supply for drywell control air

c. valve operators for primary containment venting and inerting

d. RCW system valve operators for 1B and 1C RBCCW heat exchangers.

e. valve operators in the RCIC system to drain the steam supply lines to the main condenser and the barometric condenser to the radwaste drain sump.

f. valve operators in the HPCI system to drain the steam supply lines to the main condenser.

g. valve operators in the Core Spray system to supply the keep full system.

h. RSW components in the Reactor Building such as heating and ventilation system controllers, RBCCW surge tank demineralized water inlet, and the valve operator for the condensate head tank supply to various users.

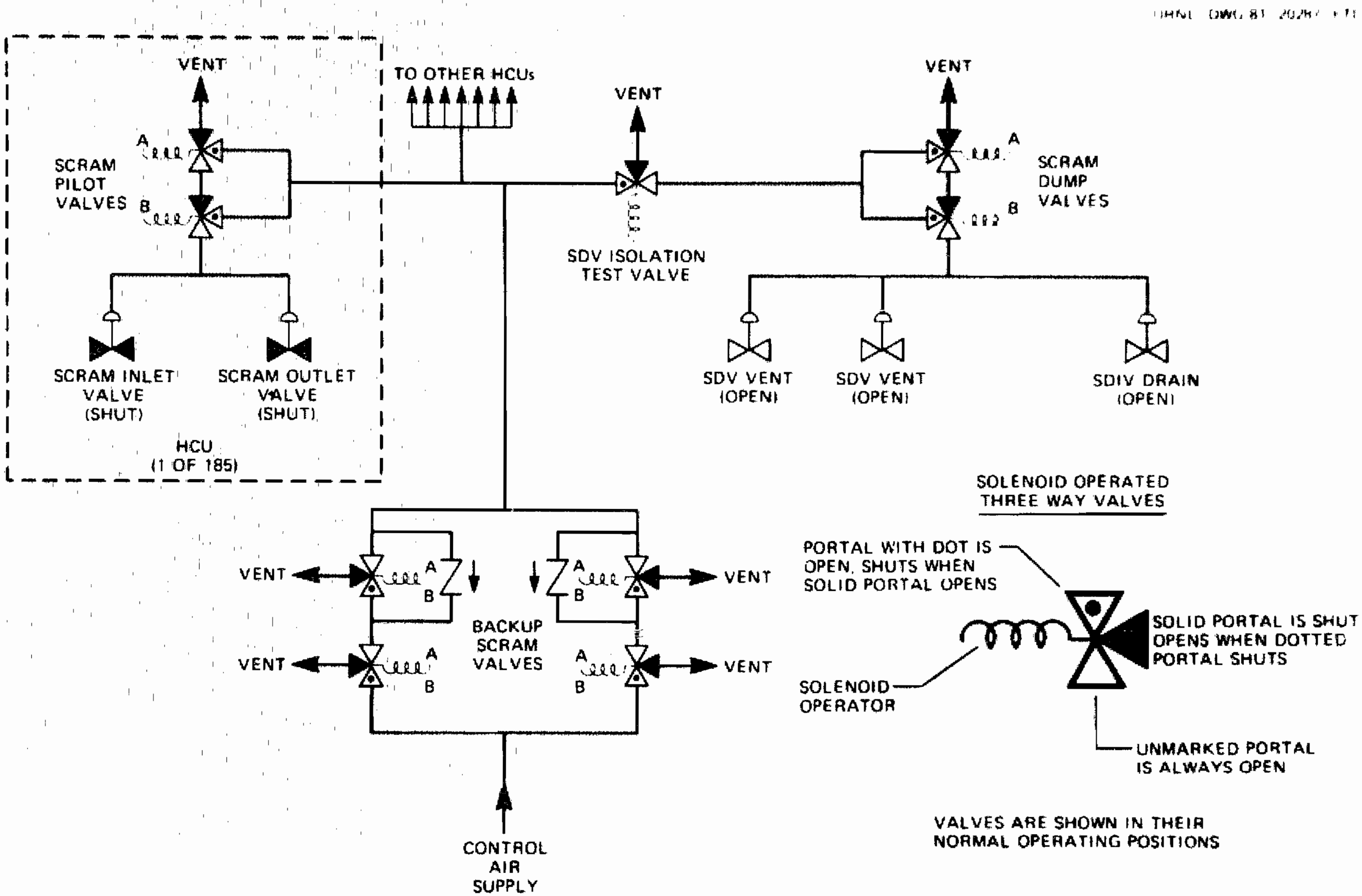

Fig. 2.2. Air operator network for the scram pilot valves and the scram dump valves.

pressure is transmitted through the solenoid-operated backup scram valves and scram pilot valves.

There are two solenoid-operated scram pilot valves in each HCU, each energized from a separate reactor protection system (RPS) bus (A or B) to remain in the position shown in Fig. 2.2. When a scram occurs, both scram pilot valve solenoids are deenergized by the Reactor Protection System and both scram pilot valves reposition so that the air operators of the scram inlet and the scram outlet valves are vented to atmosphere, permitting the scram inlet and outlet valves to be opened by their internal springs. It should be noted that the piping arrangement provides that the scram inlet and outlet valves will remain shut if only one scram pilot valve is deenergized at a time.

In contrast to the scram valves, the SDV vent valves and the SDIV drain valve are held open by control air pressure and are spring-loaded to shut. Each of the scram dump valve solenoids shown on Fig. 2.2 is powered from a separate reactor protection system bus (A or B), and when a scram occurs, both solenoids are deenergized. Upon deenergization, the scram dump valves reposition to vent the air operators of the SDV vent and the SDIV drain valves to atmosphere, permitting these valves to be shut by their internal springs. If only one scram dump valve is deenergized, the SDV vents and the SDIV drain will remain open.

An SDV isolation test valve operable from the control room is provided to permit closure of the SDV vent valves and the SDIV drain valve during normal reactor operation so that excessive leakage through the scram outlet valves can be detected by monitoring the subsequent level increase in the scram discharge instrument volume. The SDV isolation test valve is normally deenergized and aligned as shown in Fig. 2.2. When operated, the solenoid is energized from instrument and control bus A, and the valve repositions to vent the air operators of the SDV vent valves and the SDIV drain valve to atmosphere.

As shown in Fig. 2.2, control air pressure to the air operators of both the scram pilot valves and the scram dump valves is transmitted from the plant control air supply through the backup scram valves. The backup scram valves are not intended to function as an alternate method for rapid scram of all control rods, but do provide assurance that air pressure will be removed from the air operators of the scram inlet and outlet valves in all HCUs and from the SDV vents and SDIV drain valve operators as protection against a common cause failure of the scram pilot valves and scram dump valves.

During normal reactor operation, the backup scram valve solenoids are deenergized and the valves are aligned as shown in Fig. 2.2. Both reactor protection system channels A and B must trip to energize any or all of the backup scram valve solenoids and when this occurs, the backup scram valves realign to vent the control air lines leading to the scram pilot valves and the scram dump valves. Although the backup scram valves all actuate whenever the two reactor protection system channels trip, the operation of any one of these valves would be sufficient to vent the air from the supply line and accomplish a scram. Any scram accomplished solely through action of the backup scram valves would require from 15 to 20 s because of the large volume of air that must be vented through the small valve ports.

All control rod drive (CRD) hydraulic system valves fail in the scrammed position upon loss of electrical power or plant control air, i.e., the scram inlet and outlet valves fail open while the scram discharge volume vents and the scram discharge instrument volume drain fail shut. Thus, in the failed condition, the reactor would be scrammed, and the scram discharge volume and associated piping, after filling, would remain at full reactor pressure.

Under normal plant operating conditions, three of the four plant control air compressors are selected to operate as necessary to maintain system pressure between 85 and 110 psig (0.687 and 0.860 MPa). One compressor is specified to be the lead unit and runs almost continuously. A second compressor starts automatically and loads if system pressure falls below a predetermined setpoint, and the third compressor starts automatically if the sensed plant control air pressure continued to decrease. Practical considerations at the plant determine that the specific compressor loading sequence is adjusted periodically so as to equalize the running times among all four compressors.

The air receivers serve as reservoirs to damp the system response to sudden changes in demand and to reduce the number of air compressor loading cycles during normal operation. The dryers are dual chamber; while one chamber is aligned to the air flowpath, the other is automatically regenerated to remove the accumulated moisture from the desiccant. All four dryers are normally in service.

Each of the plant control air compressors will automatically trip on high discharge air temperature [310°F (427.6 K)], high lube oil temperature [180°F (355.4 K)], or low lube oil pressure [10 psig (0.170 MPa)]. The compressors are cooled by the Raw Cooling Water System, with a backup supply of cooling water available from the Emergency Equipment Cooling Water (EECW) North Header. The compressors are visually inspected once per shift. Each compressor is tested at design capacity once per quarter and torn down, overhauled, and rebuilt once per year.

As previously mentioned, the Plant Service Air System provides backup service to the Plant Control Air System at the common discharge header from the plant control air receivers. The Plant Service Air System comprises one 950 scfm (0.448 m^3/s) and one 590 scfm (0.278 m^3/s) compressor, both designed for continuous service. The service air compressors discharge into a common header that supplies one 266-ft^3 (7.532-m^3) and one 48-ft^3 (1.359-m^3) receiver. Interfacing valve 33-1 (Fig. 2.1) between the Plant Service Air and the Plant Control Air Systems is designed to automatically open if plant control air pressure falls below 90 psig (0.722 MPa) but also can be manually opened at any time from the Unit 1—Unit 2 Control Room.

With the provision of four plant control air compressors to support a load that can be carried by three and with the provision of a backup supply that can be taken from the Plant Service Air System, the availability factor for plant control air has been very high. Nevertheless, the close grouping of all of the plant air compressors and the service air backup connection in the Unit 1 Turbine Building suggests a vulnerability of the entire system to a common-mode failure by means of fire, compressor explosion, or seismic event. Accordingly, it is appropriate to examine the consequences of loss of plant control air pressure.

The joint Unit 1-Unit 2 control room air-conditioning system is operated by plant control air, but a backup air compressor, located in the Unit 1 Reactor Building would maintain operating air pressure if plant control air pressure is lost. The Unit 3 control room air conditioning system is protected by a similar arrangement.

The capability for containment venting, either from the drywell or the wetwell, depends upon the continued availability of plant control air.

Automatic scram would be initiated when the control air pressure sensed in the air headers just upstream of the scram backup valves fell below 60 psig (0.515 MPa). As previously mentioned, this automatic scram is intended to preclude the possibility of premature filling of the scram discharge volume, which otherwise might occur as a result of haphazard individual opening of the scram discharge valves associated with a few control blades. A full scram discharge volume would prevent a subsequent full scram signaled by the Reactor Protection System. The automatic scram on low plant control air pressure, along with continuous monitoring of the scram discharge instrument volume water level, is required by IE Bulletin 80-17, and is not part of the Reactor Protection System.

The outboard MSIVs are held open by plant control air and are closed by combined spring force and control air pressure. If plant control air pressure is lost, the associated pilot valves are repositioned and air stored in accumulators located in the steam and feedwater valve room provides sufficient pressure for rapid closure of the outboard MSIVs. This is expected to occur when plant control air pressure has fallen into the range of 55 to 60 psig (0.481 to 0.515 MPa), or slightly less than the pressure at which automatic scram is initiated.

After reactor scram and closure of the outboard MSIVs, reactor vessel water level could be maintained by use of either of the steam turbine-driven high-pressure injection systems, HPCI or RCIC. These systems remain fully operational upon loss of plant control air since their only air-operated components are valves in the steam supply line drains to the main condenser; these valves fail closed on loss of control air, but this is their normal position during system operation. Nevertheless, the HPCI and RCIC systems have a relatively high failure rate upon demand due to other causes,* and an accident sequence initiated by common mode failure of plant control air combined with independent failure, at one unit, of both HPCI and RCIC on demand is not of such low probability as to be disregarded without further investigation.

Even in the extreme case in which loss of plant control air is compounded by independent failure of both the HPCI and the RCIC systems at one unit, action could be taken by the operator, or the Automatic Depressurization System (ADS) would actuate, to depressurize the reactor vessel of that unit if drywell control air pressure is available to enable safety/relief valve operation. Subsequently, the low-pressure

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*The Browns Ferry Probabilistic Risk Assessment assigns a combined failure probability of 1.5×10^{-2} for these systems.

injection systems could be used to maintain vessel water level. The potential for a loss of the Plant Control Air System to develop into a severe accident sequence with core degradation and the aftermath, derives solely from the necessity of plant control air pressure to maintain the availability of drywell control air at Units 1 and 2.* To understand why, it is necessary to consider the design and operation of the Drywell Control Air System.

2.2 Drywell Control Air System

Each Browns Ferry reactor unit has an independent Drywell Control Air System that provides control air for the air-operated equipment inside the drywell. (Since the drywell atmosphere is intentionally inerted, it should be remembered that the drywell "air" is primarily nitrogen.) Each system comprises two 100 percent capacity compressors, dryers, and receivers located in the Reactor Building. As indicated on Fig. 2.3, each system takes suction on the drywell atmosphere, filters, compresses, cools, and dries the flow, then returns the pressurized air to the drywell to feed the air-operated equipment located within.

Compressor suction within the drywell is via two strainers installed in parallel. After penetrating the drywell liner, the single compressor suction line passes through two remotely-operated isolation valves in series. These valves (32-62 and 32-63) are air-operated gate valves that require air pressure to remain open and are spring-loaded to close. Both valves are included in Primary Containment Isolation System (PCIS) Group II and receive automatic closing signals upon low reactor vessel water level [at 539 in. (13.69 m)] or high drywell pressure [at 2.45 psig (0.118 MPa)]. The operating air supply to these valves is from the Plant Control Air System at Units 1 and 2 and from the Drywell Control Air System at Unit 3.

Each compressor is of the single-stage, single-acting reciprocating type, rated at 9.5 ft^3/min (0.00448 m^3/s) at 100 psig (0.791 MPa) discharge pressure. The compressors are designed for continuous operation. Under normal operating conditions, both compressors are selected for automatic operation. In this mode, one compressor starts when the system pressure drops to 87 psig (0.701 MPa) and operates until the pressure is restored to 100 psig (0.791 MPa). When the pressure again decreases due to system demands, the opposite compressor takes its turn in restoring pressure. Plant experience indicates that about one-half hour elapses between the trip of one compressor and the subsequent start of the other. If the pressure ever falls as low as 83 psig (0.674 MPa), both compressors will operate simultaneously.

The drywell control air compressor water jackets and aftercoolers are cooled by the Reactor Building Closed Cooling Water (RBCCW) System

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*At Unit 3, the drywell control air system is completely independent of the plant control air system.

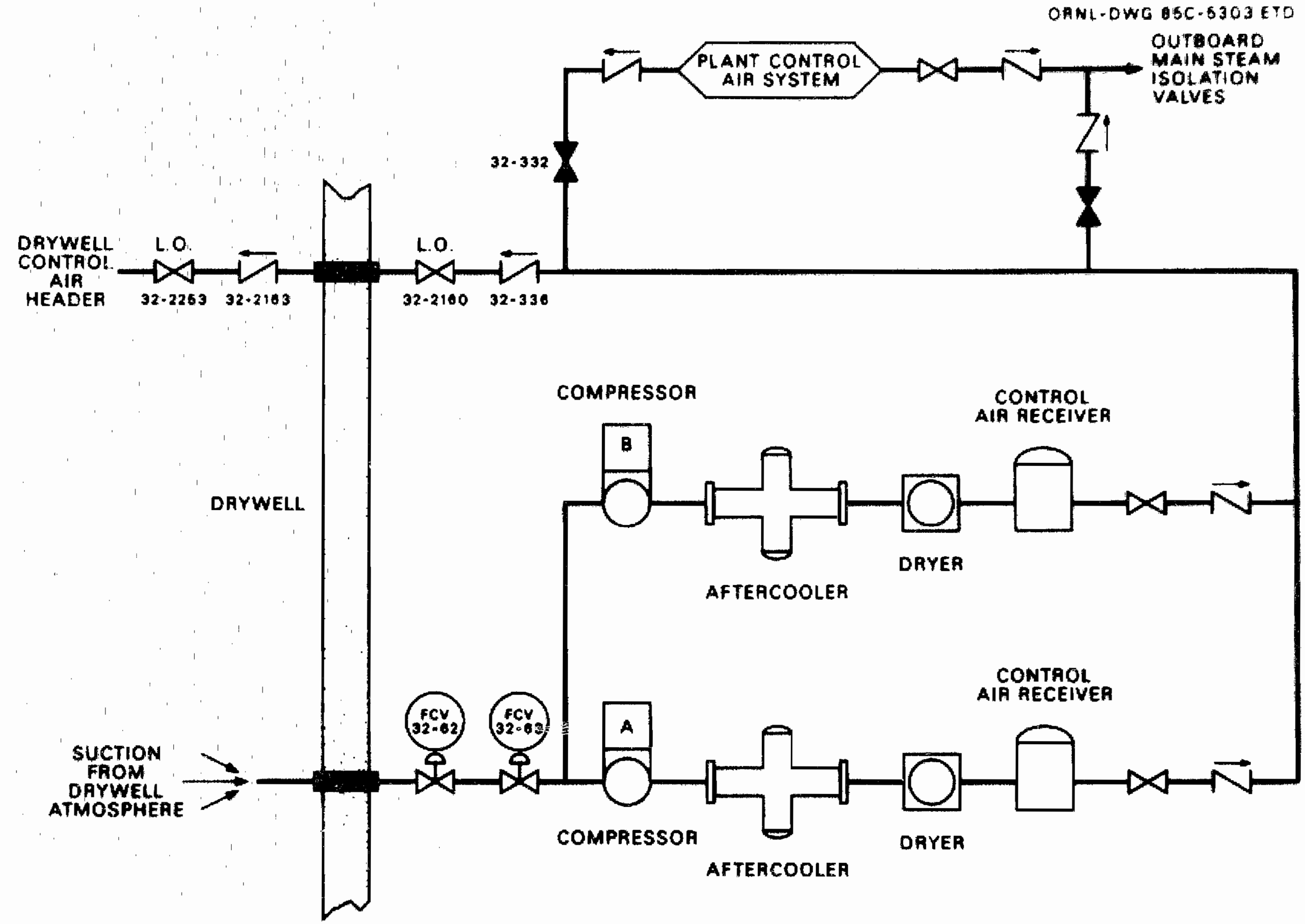

Fig. 2.3. Unit 1 drywell control air system.

which is represented schematically in Fig. 2.4. The RBCCW system heat exchangers are cooled by Raw Cooling Water, with backup from the Emergency Equipment Cooling Water System. As shown, each of the unit RBCCW systems normally supplies many unit loads, but the cooling water supply lines are divided so as to permit the less important loads to be dropped if the water supply becomes limited. Motor-operated valve 70-48 will shut automatically upon loss of offsite power or upon low RBCCW pump discharge pressure at 60 psig (0.515 MPa). As indicated, the water supply to the drywell control air compressors and aftercoolers can be valved off only locally. The inlet valves open automatically when the associated compressor starts and close when the compressor is tripped.

As indicated on Fig. 2.3, the air discharge from each of the two drywell control air compressors is passed through an aftercooler and a dryer and stored in a 57 ft^3 (1.614 m^3) receiver. The outlet of each receiver contains both a manual isolation valve and a check valve. After these valves, the two lines join and pass through a locked-open isolation valve and a check valve (32-336) just before entering the drywell liner penetration.

The original system design provided a flow control valve just upstream of the drywell penetration. However, during the fire of March 22, 1975, the large volume of stored air available in the drywell control air receivers was isolated because of loss of power to the flow control valve. In order to prevent this condition from ever occurring again, the flow control valve has been removed and replaced with the current manual locked-open valve, and a second seismic-qualified check valve (32-2163) was installed inside the drywell. This change complies with the requirement of paragraph 5.2.3.5 (Isolation Valves) of the Browns Ferry FSAR that lines with check valves that open into the primary containment must be provided with at least one AC-powered valve located outside the primary containment or a second check valve within the containment.

Provision is made, by means of the cross-connection lines with locked-shut isolation valves and check valves indicated on Fig. 2.3, to use the Plant Control Air System to provide drywell control air and to use the Drywell Control Air System to provide control air for the outboard main steam isolation valves, should the need arise.

It should be noted that Fig. 2.3 is a simplified representation of the Unit 1 Drywell Control Air System as it now stands, after piping modifications made during the fifth refueling outage. The modern system comprises a second air header that leads from the common receiver discharge into the drywell, with its own liner penetration and associated check valves and manual isolation valves. Within the drywell, the system loads have been divided, by category, between the two headers. The TVA is in the process of making similar modifications at Units 2 and 3. These improvements protect against a loss of all air-operated equipment within the drywell as a result of internal piping failure, but have no effect on the accident sequence that is the subject of this report, in which the Unit 1 Drywell Control Air System would be lost at its source.

The loads supplied by the Drywell Control Air System are listed in Table 2.2. The loss of most of these loads, as would be occasioned by a

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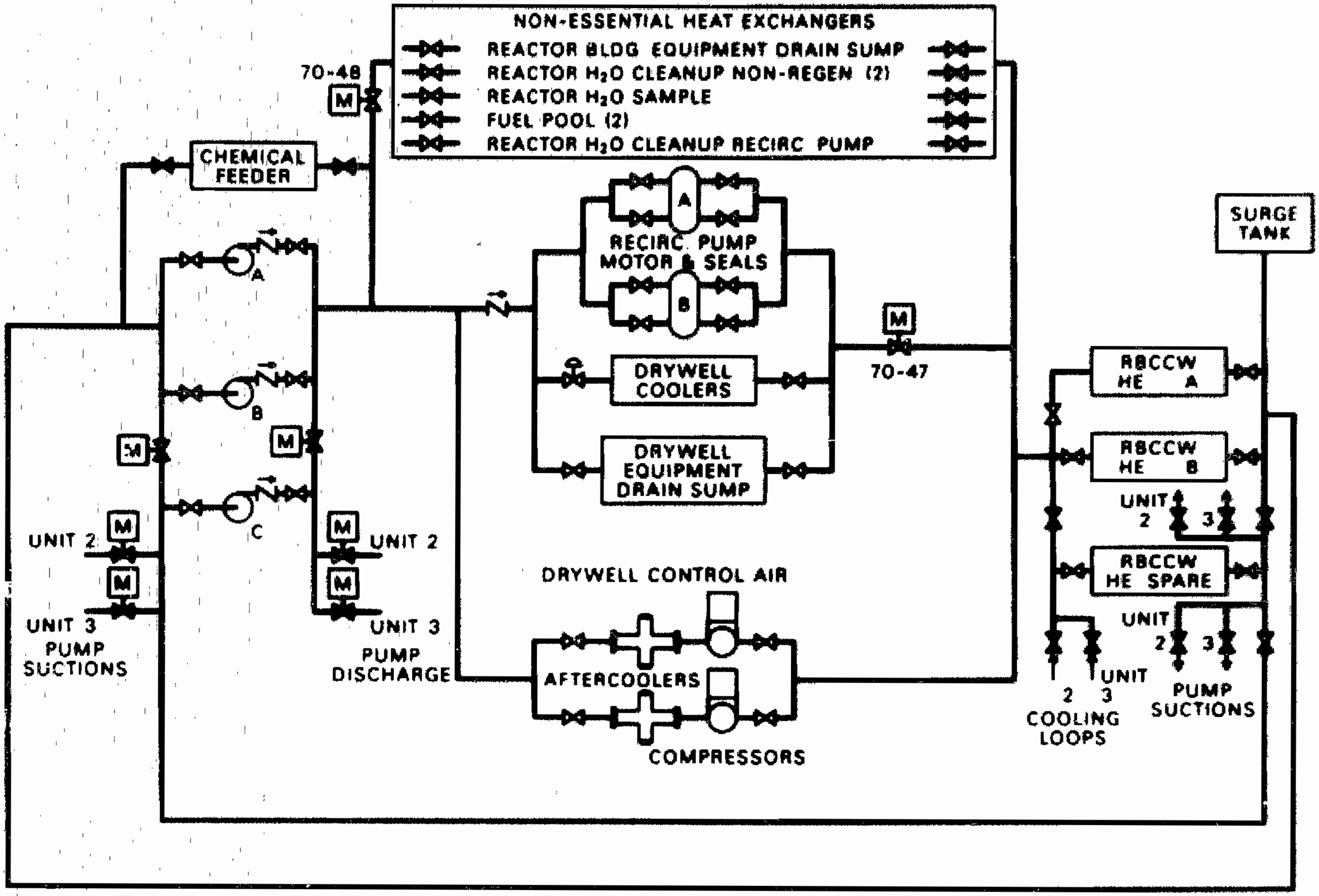

Fig. 2.4. Unit 1 reactor building closed cooling water system.

Table 2.2. Loads supplied by the
Drywell Control Air System

The inboard main steam isolation valves (4)
The safety/relief valves (13)
Valve operators and controls for RBCCW system cooling water to the drywell air coolers (10) and the under-vessel air coolers.
Valve operators for the wetwell-to-drywell vacuum breakers.
Valve operators for the RHR system and core spray system testable check valves.
Valve operators on the drywell equipment drain sump and floor drain sump isolation valves.
Valve operators on the reactor head vent to floor drain and reactor seal cavity drains.

loss of drywell control air pressure, would not be important to the progression of events in an accident sequence, but three of these loads do have significance. First, loss of the effectiveness of the drywell coolers (because their discharge dampers would fail closed) would lead to very high temperatures in the drywell. Second, the inboard main steam isolation valves would fail closed, but this is of less importance if the outboard valves had previously closed because of loss of plant control air pressure. Third, and most importantly, the air supply to the safety/relief valves would be lost, threatening their continued operability.

There are 13 safety/relief valves installed on the four main steam lines emanating from the reactor vessel. These valves are actuated automatically by high reactor vessel pressure or can be operated remotely by control air pressure. In the latter case, the air pressure must be at least 25 psi (0.172 MPa) higher than drywell pressure to open a valve; once open, the valve can be held open as long as the air pressure remains at least 25 psi (0.172 MPa) above the pressure of the drywell atmosphere.

Seven of the safety/relief valves are fed directly from the drywell control air headers and would become unavailable as soon as drywell control air pressure was lost. However, the six safety/relief valves associated with the Automatic Depressurization System (ADS) are equipped with individual one-gallon (0.0038 m^3) air accumulators which, in the event that drywell control air pressure is lost, will provide for up to five valve actuations. The air pressure in each accumulator is continuously monitored by a pressure switch that, on low accumulator pressure [70 psig (0.584 MPa)], will cause annunciation in the Control Room. The drywell control air pressure is also monitored continuously by a pressure switch installed downstream of the system receivers; this switch causes annunciation in the Control Room if the system pressure falls to 80 psig (0.653 MPa).

Each ADS accumulator has sufficient stored air to provide at least five actuations of its associated safety/relief valve if the drywell atmosphere is at ambient pressure. The leakage criteria for each accumulator is 10 psi/h (0.069 MPa/h). The accumulators and associated components were purchased and installed to seismic category I standards and specified to withstand design basis conditions in the drywell. Assuming the maximum permissible rate of accumulator leakage, the capability for valve actuation could be maintained for 6 h.

For accident sequences involving elevated pressures in the drywell, the number of safety/relief valve actuations that could be performed with the stored energy in the accumulators would be reduced. The accumulators are sized to permit two actuations at 35 psig (0.343 MPa) followed by three actuations with the drywell at atmospheric pressure. With the maximum allowable rate of accumulator leakage, sufficient pressure for valve actuation at a drywell pressure of 35 psig (0.343 MPa) could be maintained for 2.5 h.

Each ADS accumulator is tested for leakage once per operating cycle using the pressure decay method. The pressure switches and their associated alarm function are also tested once per operating cycle.

The importance of the ADS safety/relief valve accumulators, in the event that the Drywell Control Air System ceases to function, can be appreciated by recognizing the need to depressurize the reactor vessel, and keep it depressurized, if the low-pressure injection systems are to be used to maintain the reactor vessel water level and keep the core covered. This must be done if the high-pressure injection systems become unavailable for any reason. The sequence of events for the accident scenario initiated by a loss of plant control air with concurrent failure of the HPCI and RCIC systems at Unit 1 is the subject of the following chapters.

3. INITIATING EVENTS

The purpose of this chapter is to discuss the initial events expected to occur as a result of a postulated loss of the Plant Control Air System at the Browns Ferry Nuclear Plant. It is assumed that all three units are operating at the time that plant control air pressure is lost, and that the initiating event is compounded by failure, upon demand, of both of the high-pressure steam turbine-driven systems of Unit 1. Events expected to occur after the point at which the potential for operator actions becomes important in determining the sequence progression are discussed in the subsequent chapters.

3.1 Control Air Failure Modes

Loss of the Plant Control Air System might occur in any of several different ways. Explosion, fire or seismic event might cause extensive local damage with piping breaks and instantaneous loss of pressure. On the other hand, if the piping remains intact but the compressors are lost, the pressure decay in the system would be gradual because of the large storage volume of the accumulators. From the standpoint of analyzing the effects of the loss of plant control air pressure, the rate of pressure decay in the main system piping is of little significance if the overall plant accident sequence is considered as three separate accident sequences, one per unit, each defined to begin at the time of unit reactor scram. [The reader is reminded that recent plant modifications provide an automatic scram if the plant control air pressure sensed at the inlet to the backup scram valves falls to 60 psig (0.515 MPa).] This approach is adopted for this study.

3.2 Accident Sequence Selection

It is important to note, as shown on the lower portion of Fig. 2.1, that in each unit, the reactor building loads supplied by the Plant Control Air System are fed by three distinct headers, each separated from the main system supply piping by a check valve. Thus, assuming loss of plant control air pressure in the main supply piping, the rate of pressure decay downstream of the check valves in the three individual headers will depend upon the load demand and leakage rate of each individual header. When pressure in the header supplying the scram system (lower right of Fig. 2.1) falls to 60 psig (0.515 MPa), automatic reactor scram will occur. When pressure in the header supplying the outboard main steam isolation valves (MSIVs) falls to the 55–60 psig (0.481 to 0.515 MPa) range, the outboard MSIVs will shut, isolating the reactor vessel. Also, importantly, the drywell control air compressor suction isolation valves on Units 1 and 2 will fail shut when plant control air pressure in the associated valve operator supply lines has decayed sufficiently; this will lead to loss of the Drywell Control Air Systems of these units. On Unit 3, the drywell control air compressor suction isolation valves are operated by drywell control air and thus are immune to failure of the plant air system.

It is expected that reactor scram and outboard MSIV closure at a particular unit would both occur at about the same time after loss of plant control air pressure. Analysis indicates that which comes first only affects the accident sequence from the standpoint of the initial water level in the reactor vessel at the time of vessel isolation. If scram is assumed to occur first, then the reactor vessel water level would be significantly increased during the brief period before MSIV closure and the concomitant loss of the feedwater pumps. It does seem most likely that reactor scram would occur first, and this assumption is used for the no-operator-action case discussed in Chapter 4. However, both possibilities are considered in the discussion of the effect of operator actions upon the accident sequence that is presented in Chapter 5.

Although the postulated loss of the Plant Control Air System would directly affect all three Browns Ferry units, it is easy to show that severe accident consequences should be confined to one unit. First, the Unit 3 Drywell Control Air System is immune to failure of plant control air pressure and therefore the Unit 3 reactor vessel could be maintained depressurized at any time to permit continued core coverage by water injected by the low-pressure injection systems. Thus, loss of plant control air should not lead to a severe accident on Unit 3.

Loss of the Plant Control Air System would lead to loss of the Drywell Control Air Systems of Units 1 and 2. The penalty for loss of drywell control air pressure and the subsequent depletion of the air stored in the ADS accumulators is an inability to depressurize, or to maintain a previously established depressurized state of, the reactor vessel. However, reactor vessel water level can easily be maintained by minimal use of either of the high-pressure steam turbine-driven injection systems (HPCI or RCIC) if the vessel remains pressurized. Therefore, loss of plant control air could not lead to a severe accident on Units 1 or 2 unless there is a simultaneous loss of both high-pressure injection systems at one of these units.

The reliability of the HPCI and RCIC systems is such that it is not unreasonable to suppose a simultaneous loss of plant control air and failure-upon-demand of both the HPCI and RCIC systems at one of the three Browns Ferry units. As discussed previously, loss of plant control air plus HPCI and RCIC would not lead to a severe accident on Unit 3, but might on Unit 1 or Unit 2. However, the probability is so remote that both of the high-pressure injection systems would fail at both Units 1 and 2 simultaneously with loss of plant control air that it can be disregarded. Accordingly, this study is based upon an assumption of a loss of plant control air pressure combined with failure of both the HPCI and RCIC systems at Unit 1, and the severe accident consequences are confined to Unit 1 events.

4. LOSS OF CONTROL AIR WITHOUT OPERATOR ACTION

4.1 Introduction

This chapter presents the results of BWR-LTAS calculations of the response of the primary system and containment following loss of plant control air, performed under the assumption that the operators take no action of any kind. Experience gained from previous ORNL SASA program studies has confirmed the obvious: that an understanding of the automatic plant response without operator action is a very useful first step toward determination of the operator actions that should be taken to mitigate the consequences of the initiating event.

As discussed in Chap. 3, loss of plant control air pressure would affect all three Browns Ferry units, but would be expected to lead to loss of the Drywell Control Air Systems only on Units 1 and 2, where the drywell control air compressor suction valves are held open by plant control air. Loss of a unit's Drywell Control Air System, in turn, would lead to loss of ability to maintain the reactor vessel depressurized since the reactor vessel safety/relief valves are opened and held open by drywell control air pressure when remotely operated. Still, the unit steam turbine-driven high-pressure injection systems HPCI and RCIC should be available and both of these have ample capacity to maintain reactor vessel water level.

Unfortunately, the demonstrated reliability of the HPCI and RCIC systems* is such that it is not unreasonable to perform accident sequence calculations based upon the assumption of independent failure of both of these systems upon demand at one unit, following a loss of the Plant Control Air System. It is important to note that this assumption of independent failure of both HPCI and RCIC is also <u>necessary</u>, if it is postulated that an accident sequence initiated by loss of plant control air pressure might degenerate into a Severe Accident at one unit. In this study, the combined failure-upon-demand of both HPCI and RCIC is assumed to occur at Unit 1. Furthermore, it is assumed that both of these systems remain inoperable throughout the accident sequence.

4.2 Events Assuming Nominal Safety/Relief Valve (SRV) Behavior

The results of the BWR-LTAS calculations for the case without operator action are illustrated in Figs. 4.1 through 4.6. The loss of air compressors and the subsequent decay of plant control air pressure to

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*Per the Browns Ferry Probabilistic Risk Assessment, the estimated frequency of combined failure-upon-demand of both of these systems is 1.5×10^{-2}.

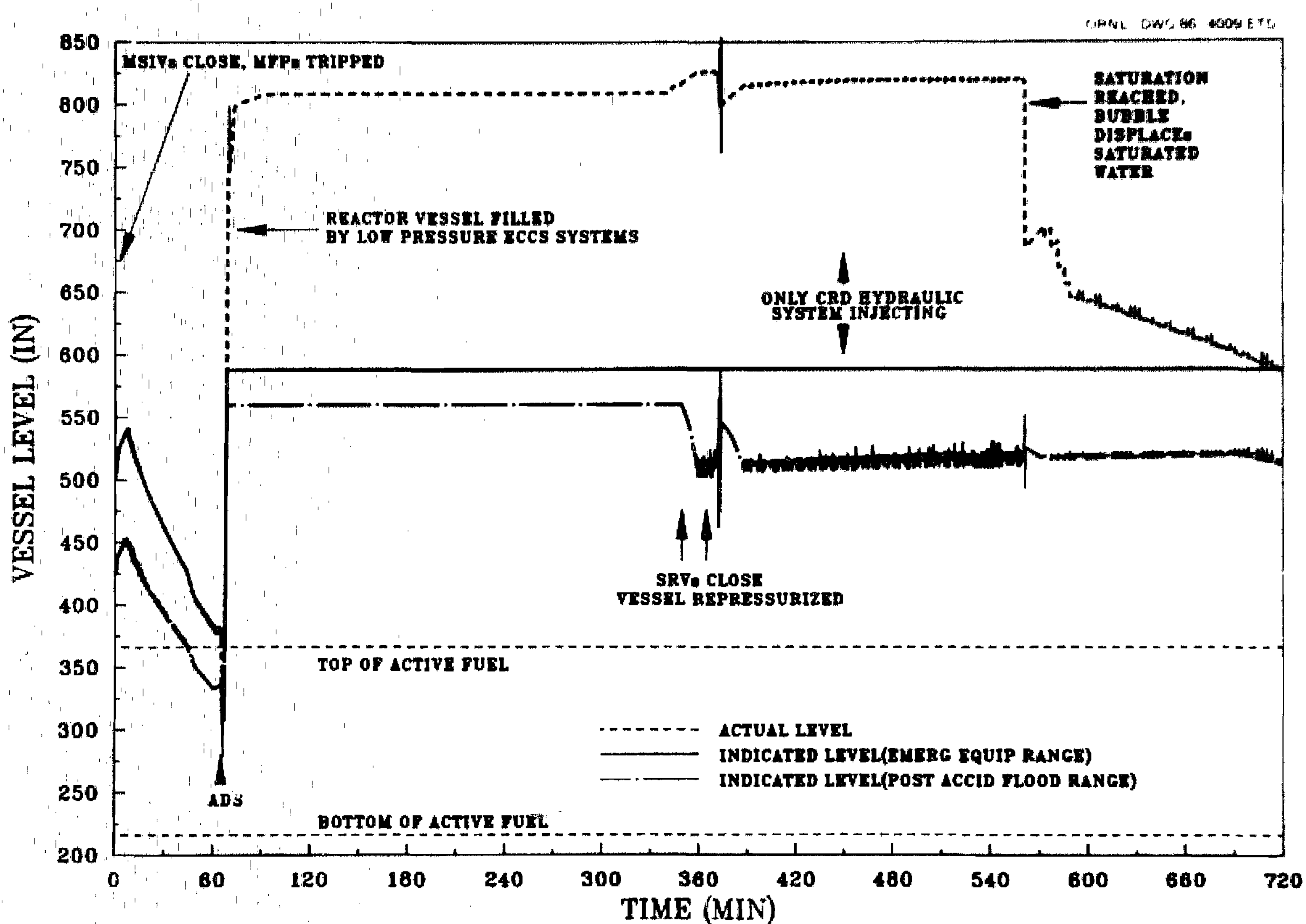

Fig. 4.1. Loss of control air without operator action — reactor vessel water level.

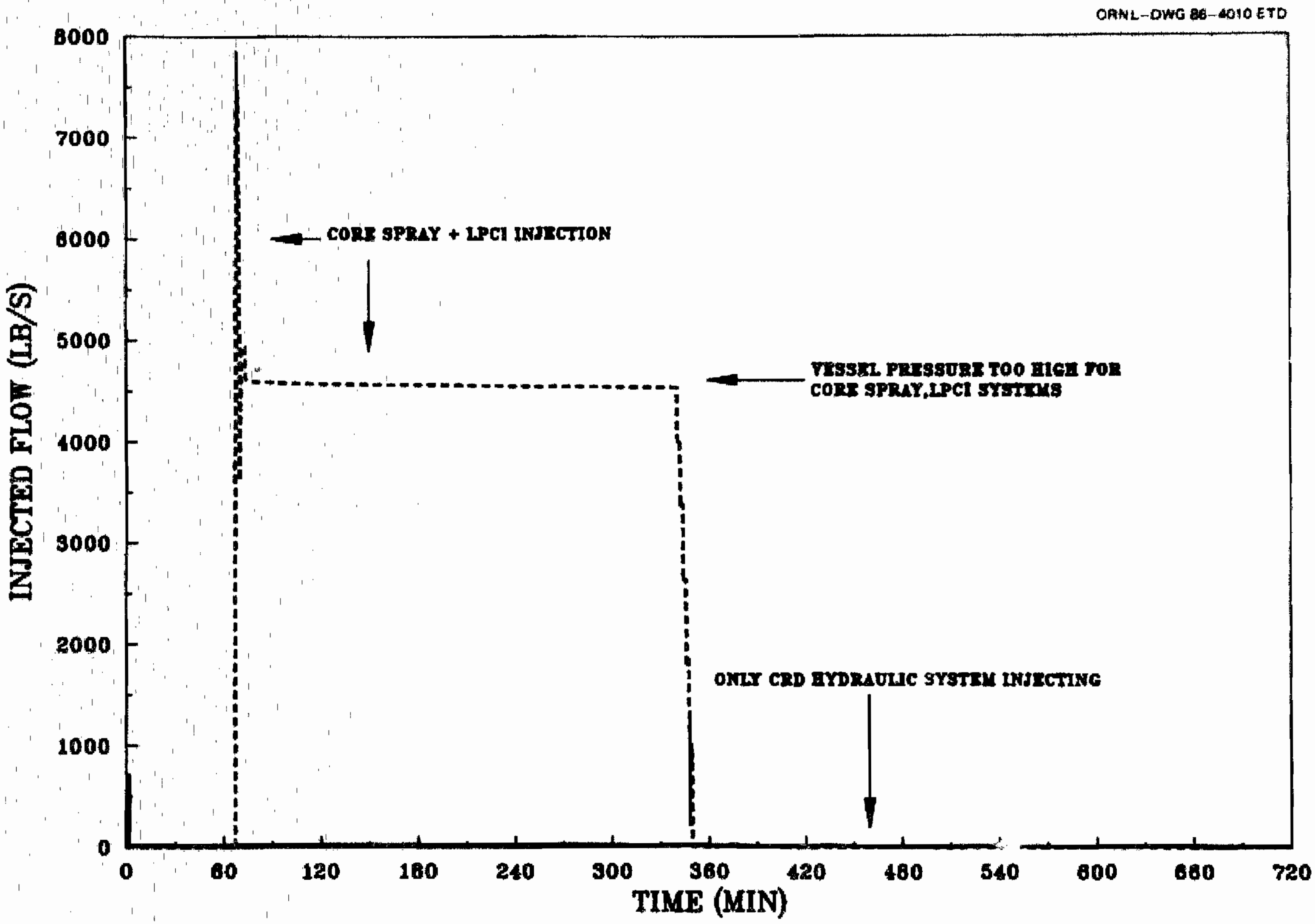

Fig. 4.2. Loss of control air without operator action — injected flow.

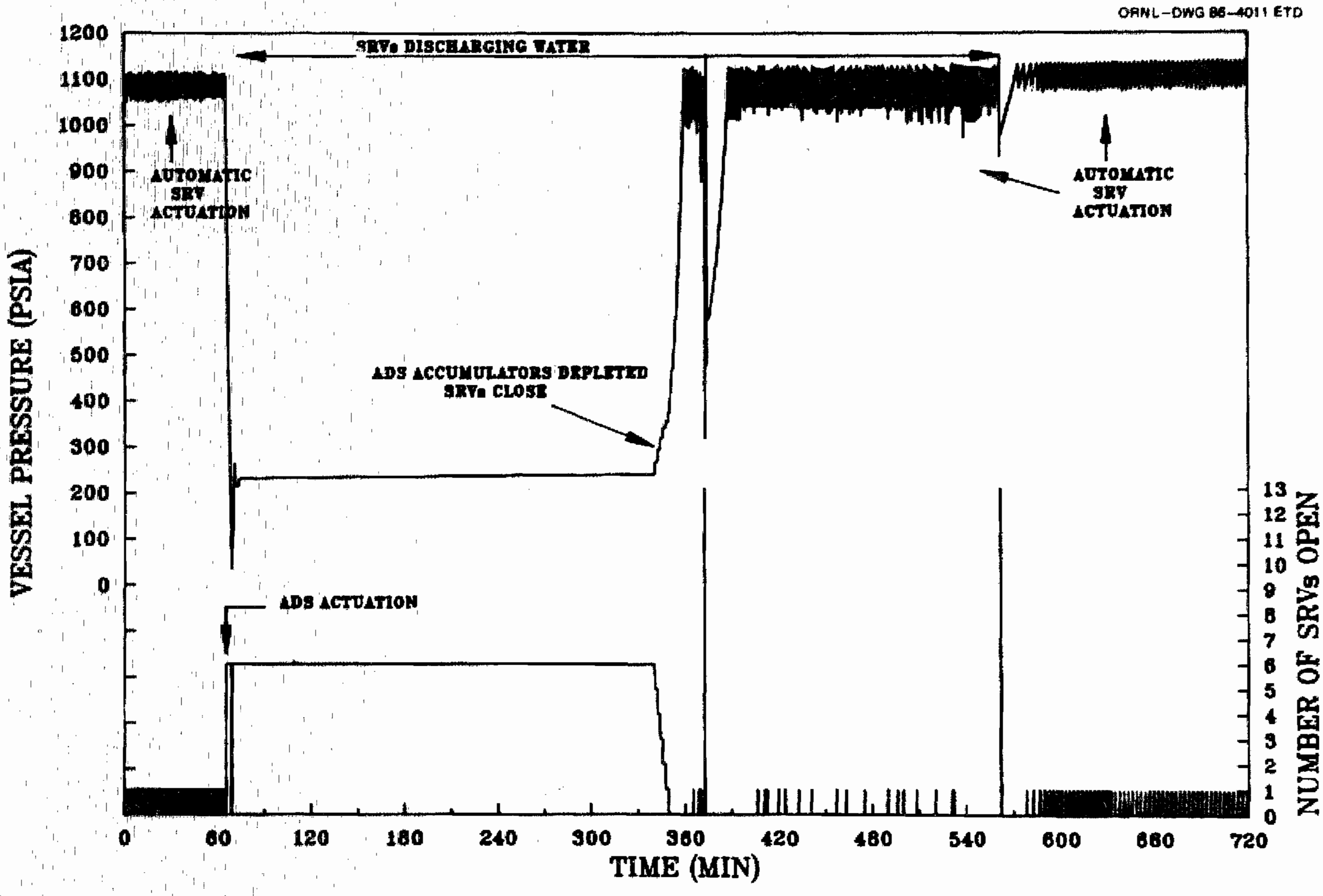

Fig. 4.3. Loss of control air without operator action — reactor vessel pressure and number of open SRVs. The SRV-open intervals during the period when the reactor vessel is full of water and at-pressure (about time 350 to time 565 min) are so brief that many are not recognized by the plotting code.

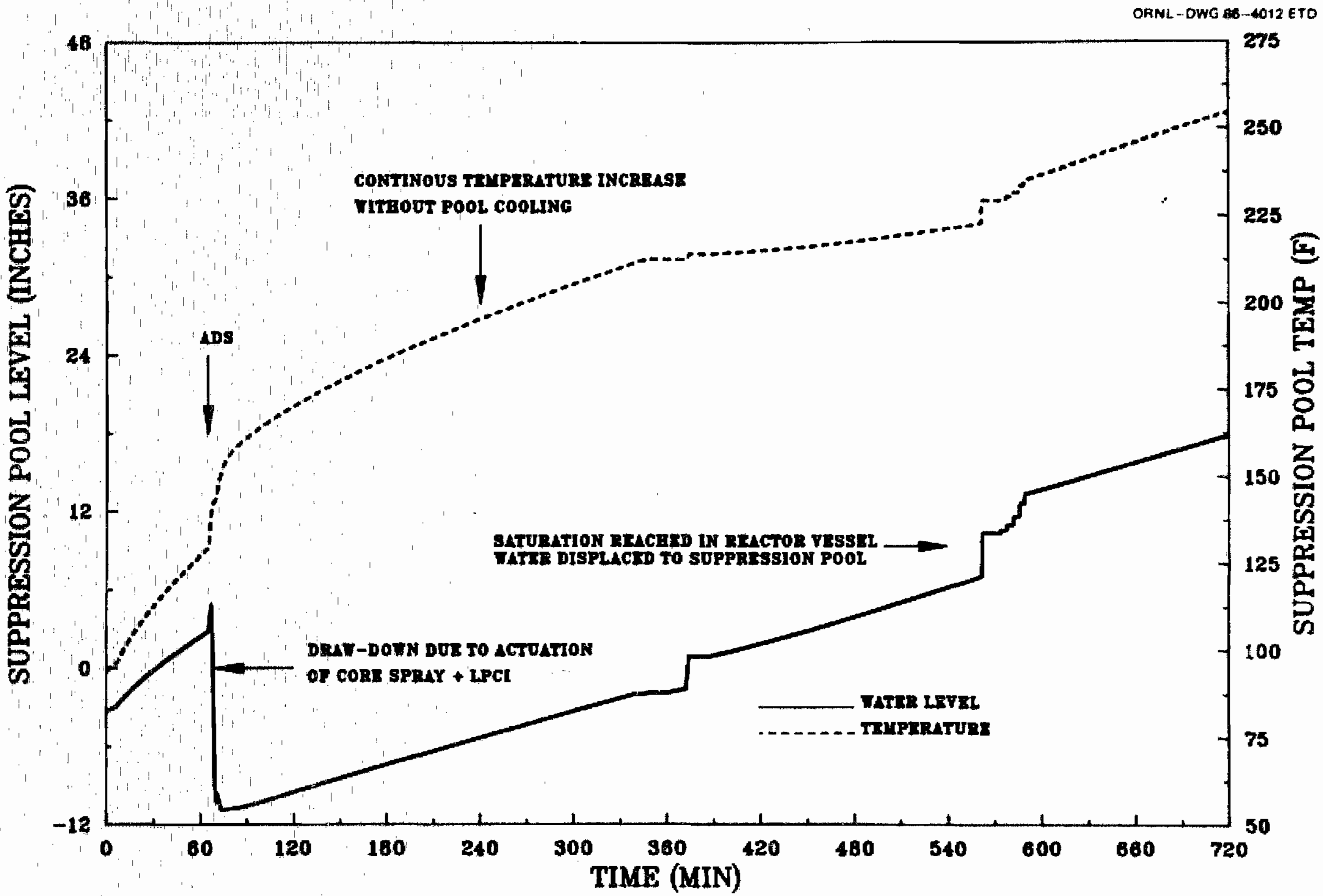

Fig. 4.4. Loss of control air without operator action — pressure suppression pool level and temperature.

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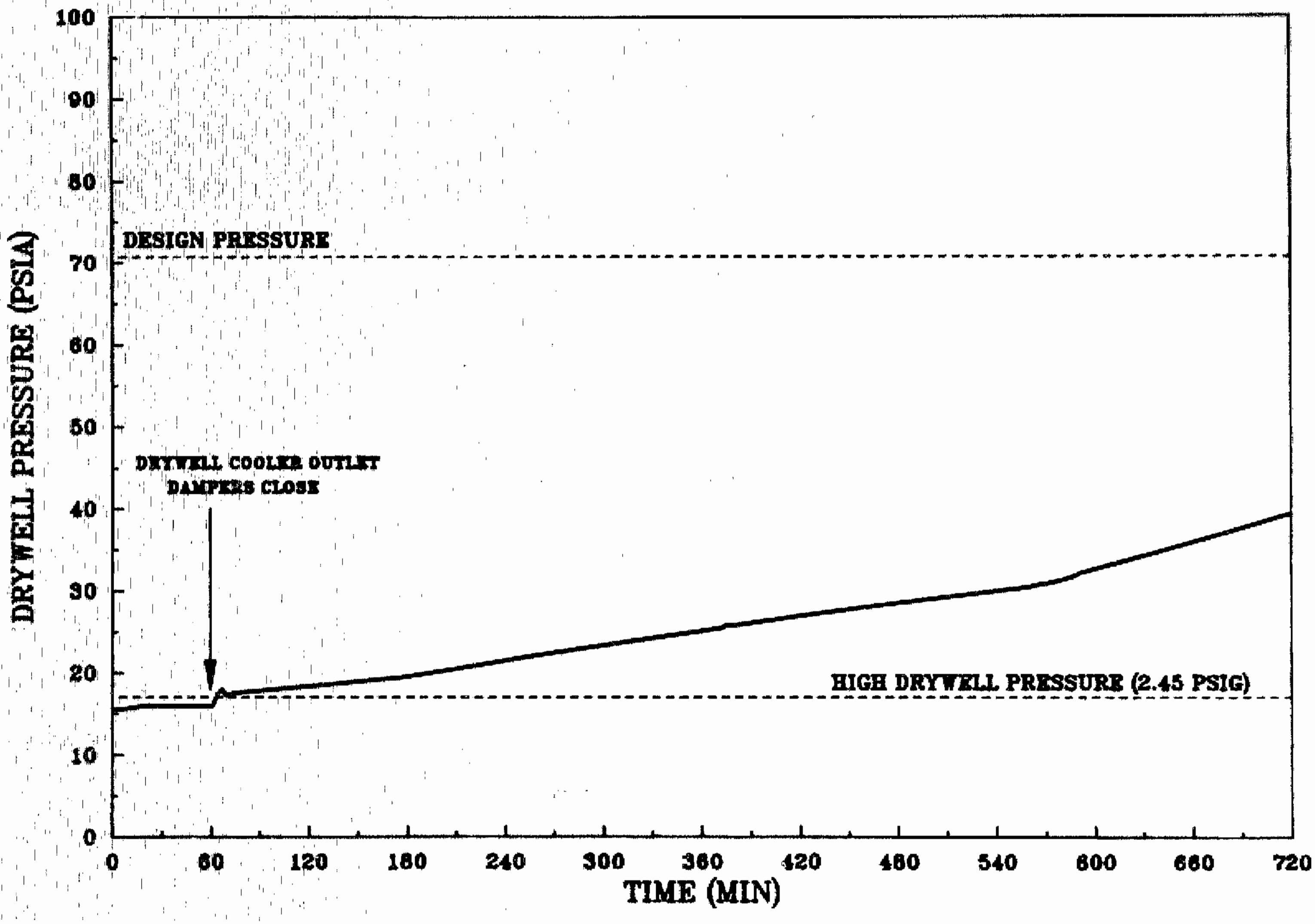


Fig. 4.5. Loss of control air without operator action — drywell pressure.

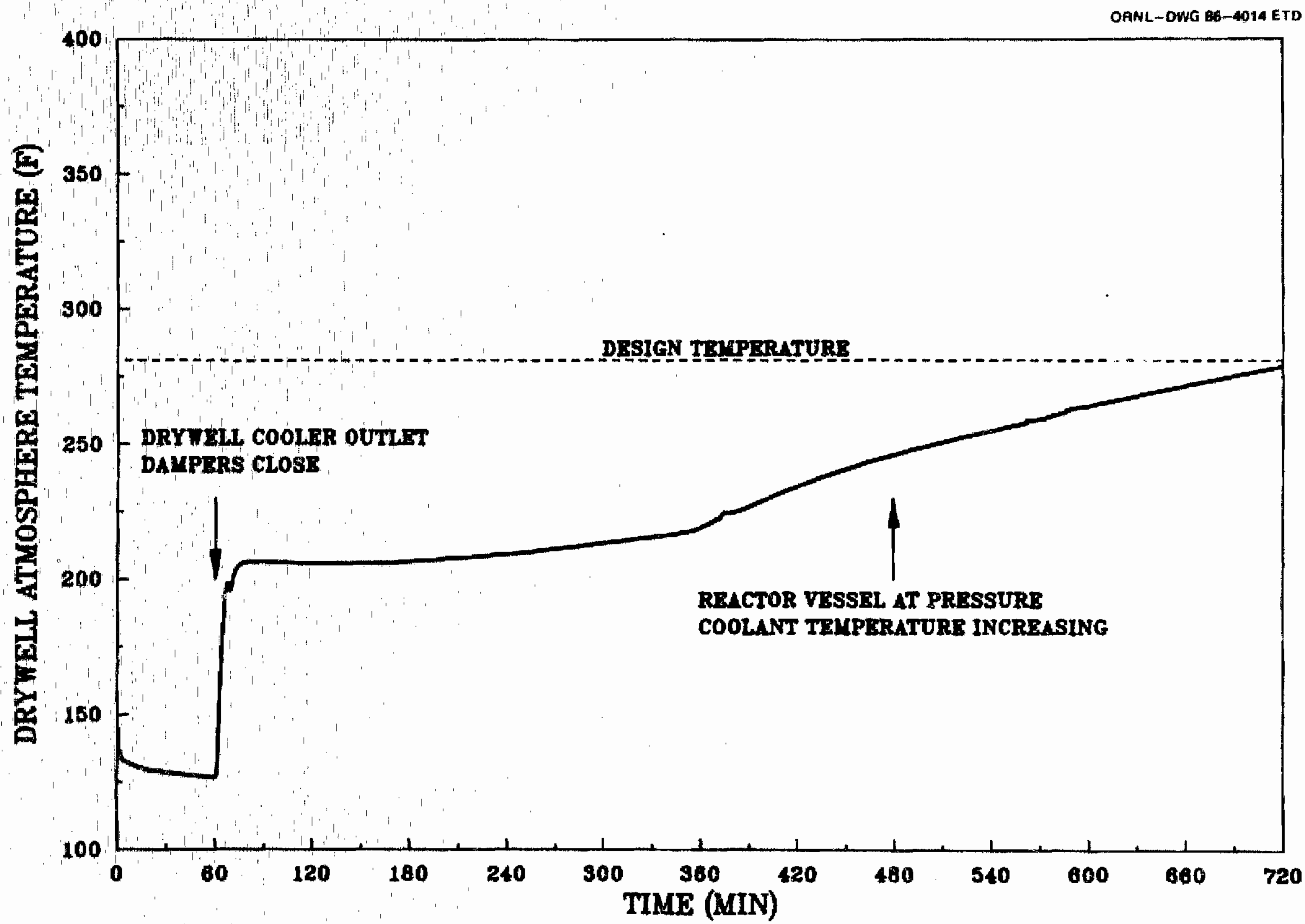


Fig. 4.6. Loss of control air without operator action — drywell atmosphere temperature.

below 60 psig (0.515 MPa) cause the sequence-initiating automatic reactor trip from full power. The plant control air pressure continues to decay and, 2 min later, the outboard MSIVs go shut, causing the steam-driven main feedwater pumps to coast to a stop. Loss of plant control air also causes closure of valves in the recirculation lines from the condensate booster pump discharge to the condensate storage tank; the booster pumps cannot inject into the pressurized reactor vessel but continue to run at shutoff head, zero flow, and begin to overheat. After a brief period of increasing water level as the feedpumps coast down, decreasing downcomer water level (Fig. 4.1) energizes the HPCI and RCIC initiating circuits as it sinks below 476.5 in. (12.10 m) above vessel zero; this occurs 24.5 min after reactor trip but both of the Unit 1 systems, by sequence-defining assumption, fail to operate upon demand.

The timing of the sequence of events is diagrammed in Table 4.1, in which the succession of events is presented in three columns. On the left side, the label "DW,SP" for "Drywell, Suppression Pool" heads the column of events for the containment. In the center, the column of events for reactor vessel pressure control is labeled "RC/P". On the right, events that determine or depend upon the reactor vessel water level are described in the column headed "RC/L". These titles have been selected to conform with the titles used in the BWR Owner's Group Emergency Procedure Guidelines (EPGs). Events that can be identified with a particular time in the accident sequence are represented by narrative descriptions within rectangles; events that would occur repeatedly over a period of time are represented by descriptions enclosed within oval outlines. Dashed lines between the columns show dependence; the event at the arrowhead cannot occur until the event at the tail has occurred.

The ~105 gpm (0.0066 m^3/s) injected into the reactor vessel by the control rod drive (CRD) hydraulic system is insufficient to reverse the downward trend of reactor vessel water level that continues after failure of HPCI and RCIC. At time 60 min, the effectiveness of the drywell fan coolers is lost because the decaying drywell control air pressure is no longer sufficient to hold open the spring-loaded cooler air outlet dampers. Drywell pressure (Fig. 4.5) and drywell temperature (Fig. 4.6) increase sharply after this time.

The Automatic Depressurization System (ADS) initiates the opening of six Safety Relief Valves (SRVs) when drywell pressure increases to its high level setpoint of 2.45 psig (0.118 MPa); this occurs 63 min after the scram and 16 min after the vessel water level has decreased below the ADS permissive setpoint. At the beginning of the depressurization, the collapsed reactor vessel water level is only slightly above the top of the core.

The condensate booster pumps are assumed failed due to lengthy operation at shutoff head and zero flow, but within 2 min after ADS actuation, reactor vessel pressure is sufficiently low so that the core spray and residual heat removal (RHR) system pumps [combined capacity ~50,000 gpm (3.15 m^3/s)] can begin injecting (Fig. 4.2) and they take only 2 min to flood the reactor vessel. This, of course, includes filling of the main steam lines and the introduction of water to the SRVs.

Two-stage pilot-operated Target Rock safety/relief valves, model No. 7567F, similar to those installed at Browns Ferry, have been tested

Table 4.1. Sequence of events for loss of control air without operator action.

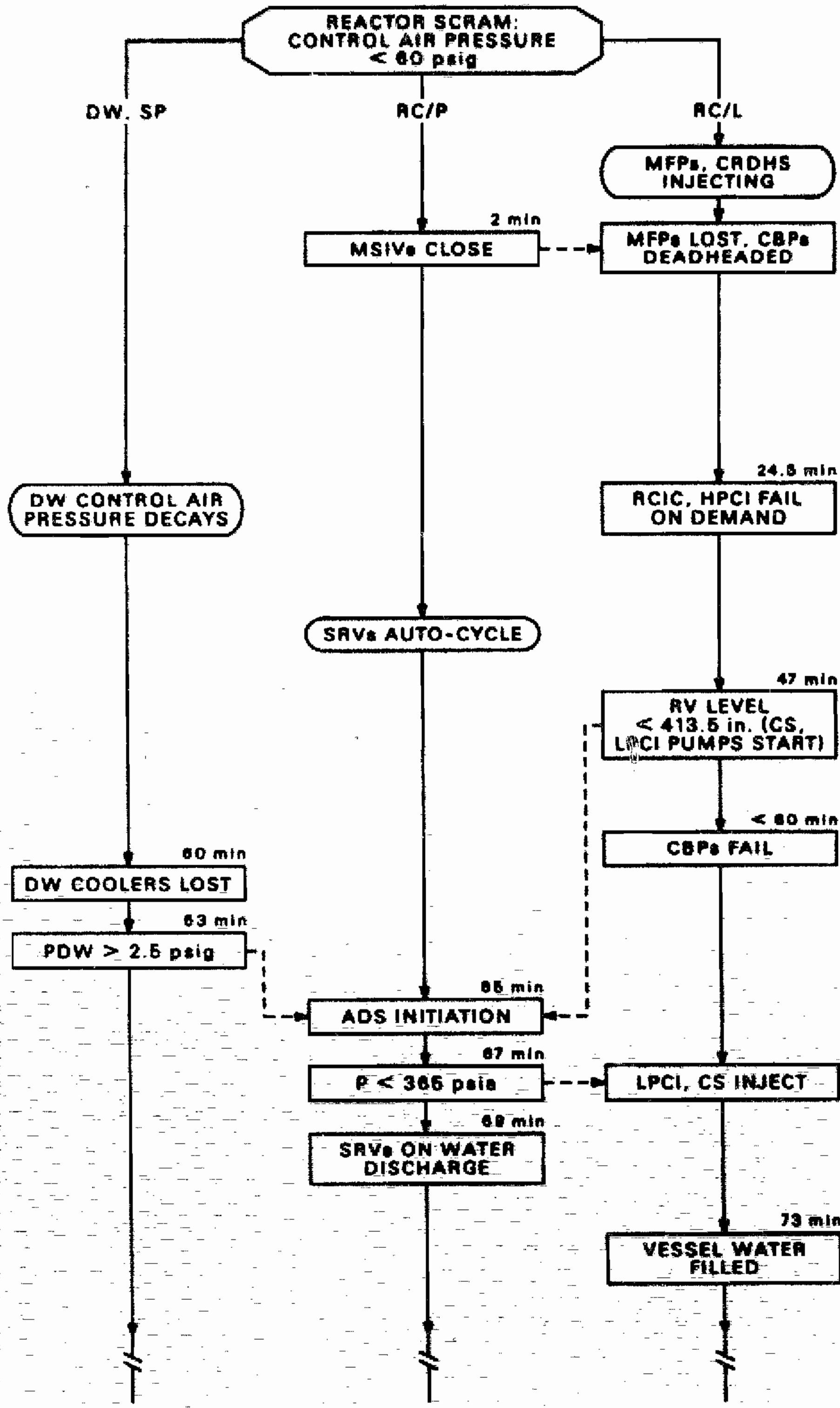
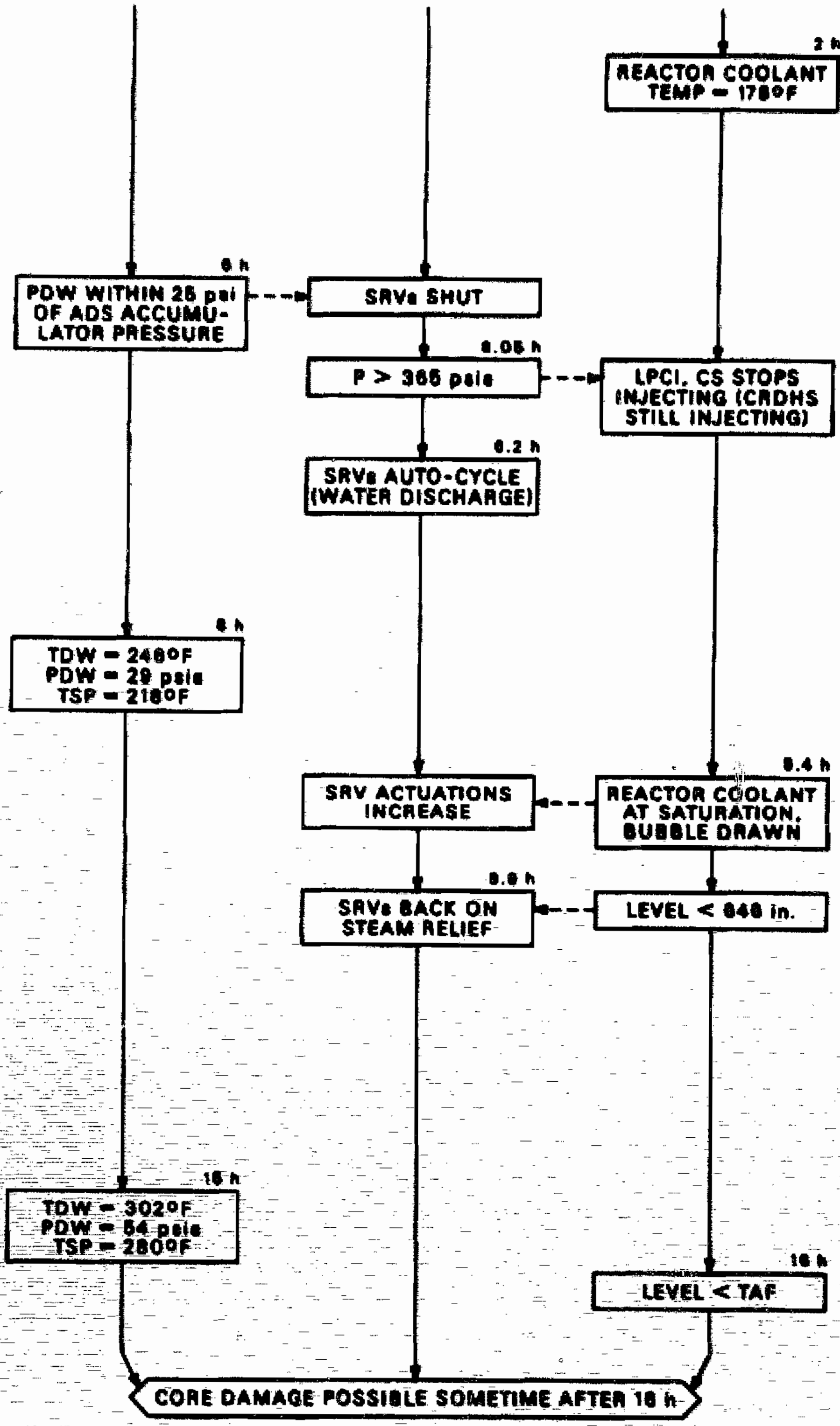


Table 4.1 (continued)



at Wyle Laboratories for the BWR Owners Group.* The test conditions were intended to simulate the valve and piping operating environment that would be expected during the alternate shutdown cooling mode of operation of the RHR system. This is a backup operational mode that could be used in the event that some malfunction prevents the opening of the shutdown cooling supply valves that connect the reactor vessel to the RHR pump suctions. In the alternate shutdown cooling mode, water is injected into the reactor vessel by the RHR system pumps while they take suction on the pressure suppression pool; the water flows through the vessel and is discharged back to the pressure suppression pool through one or more open safety/relief valves.

The results of the Wyle Laboratory tests demonstrate that the Browns Ferry safety/relief valves can be opened, held open, or closed, at will, with single-phase water as the working fluid under nearly steady-state conditions. It seems reasonable to assume that the same SRV behavior would be observed under transient accident conditions in which the working fluid changed in a continuous fashion from steam to water. Under these circumstances, the reactor vessel pressure would equilibrate (Fig. 4.3) at about 250 psig (1.825 MPa), with a continuous reactor vessel through-flow of about 15,000 gpm (0.945 m^3/s), pumped from the pressure suppression pool to the reactor vessel by the low pressure injection systems and returned to the suppression pool via the six open SRVs. Although, without operator action, the RHR system heat exchangers would not be supplied with cooling water, the pressure suppression pool comprises a very large heat sink that could accommodate the reactor decay heat for many hours.

This quasi-equilibrium state with the core amply cooled and only a small rate of increase of pressure suppression pool temperature (Fig 4.4) would continue until about 6 h after scram, when the SRVs would shut on depletion of drywell control air pressure and the decline of the stored air pressure in the six individual ADS SRV accumulators to within 25 psi (0.172 MPa) of the drywell atmosphere pressure. After this point, the SRVs could no longer be voluntarily held open. The reactor vessel would repressurize and a steam bubble would form within. If there were still no operator action, a very slow boiloff would ensue with one SRV intermittently open due to high internal reactor vessel pressure, leading to core uncovery about 16 h after the reactor scram.

4.3 Variations of the No-Operator-Action Sequence

Since the reactor vessel would be filled with relatively cold water during the accident sequence without operator action, the performance of the SRVs when liquid water is introduced is of paramount importance to the sequence outcome. For the study results discussed in Sect. 4.2, it was assumed that the SRVs would remain open during the period when the

*The test results are proprietary information.

fluid discharge changed from steam to water. The available experimental evidence supports this assumption.

However, if the relief valves were to shut at the time that liquid water is introduced, then the reactor vessel and associated piping would constitute a closed solid system, initially at about 300 psig (2.17 MPa), with an embedded large heat source. Fluid expansion would quickly lead to pressures that would force a pressure-relieving opening somewhere in the isolated primary system.

A second variation of the no-operator-action case involves the occurrence of a stuck-open relief valve at the time of the increase in reactor vessel pressure always occasioned by a reactor scram that is accompanied by MSIV closure. The stuck-open valve would actually have beneficial consequences, since it would totally remove the potential for rupture of an isolated, solid, system. The reactor vessel would remain at low pressure after loss of drywell control air, and calculations indicate that the flow of liquid water through one open SRV would be sufficient to remove the decay heat generated by the shutdown core. The low-pressure injection systems would continue to circulate the combined reactor vessel and pressure suppression pool water mass for at least 24 h without reaching excessive temperature. It should be recognized that this variation of the Unit 1 no-operator-action sequence is characterized by ample opportunity for mitigative operator action and therefore is very similar to the Loss of Decay Heat Removal accident sequence previously reported as NUREG/CR-2973.

5. LOSS OF CONTROL AIR WITH OPERATOR ACTION

The general subject of this and the previous chapter of this report is the progression of postulated accident sequences, initiated by loss of plant control air at the Browns Ferry Nuclear Plant, that are compounded by the assumption of failure, upon demand, of both of the steam turbine-driven, high-pressure injection systems at Unit 1.* In the discussion of Chapter 4, the operators were assumed to do absolutely nothing. In this chapter, the operators are assumed to act, and it is shown that the extent of operator actions plays a dominant role in determining the subsequent sequence of events. The discussion is focused upon the events at Unit 1, which, as a result of the assumed failure of the associated HPCI and RCIC systems, is the only unit at which operator action might be necessary to prevent the situation from deteriorating into a Severe Accident.

The two cases presented in this Chapter were selected with the intent that they would bracket the range of possible operator actions, particularly with respect to the question of whether or not to depressurize the reactor vessel. The BWR-LTAS calculations for the at-pressure case (Section 5.2) assume that the operators do not depressurize. The calculations for the depressurized case (Section 5.3) assume that the operators initiate an emergency depressurization shortly after the reactor scram. The depressurized case begins to look like the at-pressure case after 6 h, when stored air to hold the SRVs open is no longer available and the reactor vessel returns to full pressure. In both cases, the possible progression to a severe accident hinges on the same question: can/will the operators prevent core uncovery by acting to enhance the flow provided by the CRD hydraulic system? Appendix D of this report addresses the minimum injection needed to prevent core uncovery; the required flow is within the capability of the CRD hydraulic system [supplemented if necessary by the Standby Liquid Control (SLC) system] with appropriate operator action. That the operators might, in compliance with an event-specific procedure, follow some intermediate path, for example, by depressurizing at 100°F/h (55.5 K/h) instead of at the emergency rate, is irrelevant to the conclusions of this chapter; therefore, such cases have not been simulated.

5.1 Basic Considerations for Operator Action

Loss of the Plant Control Air System would confront the Unit 1 control room operator with reactor scram and closure of the outboard main

*The reader is reminded that the drywell control air systems of Units 1 and 2 would both be lost following loss of the Plant Control Air System. Thus both of these units would be susceptible to an accident-compounding loss of the high-pressure injection systems. However, the probability of independent failure of both high-pressure injection systems at both of these units is incredibly small. We have assumed that both systems are lost at Unit 1.

steam isolation valves (MSIVs). As a result of the scram, the control rod drive (CRD) hydraulic system would inject cooling water to the reactor vessel at about twice its normal rate, but this would not be equivalent to the rate at which the water inventory is boiled away by core decay heat until many hours after scram. Accordingly, the reactor vessel water level would quickly fall to the setpoint for automatic initiation of the HPCI and RCIC systems, but it has been assumed that both of these high-pressure injection systems fail, upon demand, at Unit 1. What actions should the operators take? This is an important consideration that, following the practice previously established in ORNL SASA program accident studies, can be divided into four areas, each based upon one of the four major goals of operator action. These are: reactivity control, reactor vessel water level control, reactor vessel pressure control, and containment temperature and pressure control. Each of these is discussed in turn in the following subsections.

5.1.1 Reactivity control

The condition of low plant control air pressure would by itself generate a signal for automatic reactor scram. Low plant control air pressure would also cause closure of the outboard MSIVs and this would trigger an additional, completely independent, automatic scram signal. Even if these scram signals both fail to produce an automatic scram, the physical reality of low plant control air pressure would cause all of the control blades to move into the core. The reasons for this are explained in Section 2.1 of Chapter 2, but it is important to note here that no operator action would be necessary to achieve reactivity control in an accident sequence initiated by loss of plant control air pressure.

5.1.2 Reactor vessel level control

As discussed previously, it is assumed in this study that the loss of plant control air pressure, which affects all three units, is accompanied by a failure, upon demand, of both the HPCI and the RCIC systems at Unit 1. With loss of the reactor feed pumps as a result of MSIV closure, the only remaining high-pressure injection source at Unit 1 would be that provided by the CRD hydraulic system. This system injects control blade cooling water at a rate of 60 gpm (3.79×10^{-3} m^3/s) during normal reactor operation, but this injection rate is nearly doubled, automatically, by the basic characteristics of the system design, when the reactor is scrammed (see Appendix E of NUREG/CR-2672).

Browns Ferry EPG-based emergency procedure EOI-1 directs the operators to maintain the reactor vessel downcomer water level above 413.5 in. (10.5 m) (referenced to vessel zero) by use of the following high pressure systems: the feedwater/condensate system, the CRD hydraulic system, and the HPCI and RCIC systems. This minimum allowed vessel water level is 47.5 in. (1.2 m) above the top of active fuel. In the accident sequences of this chapter, only the CRD hydraulic system is available and the vessel water level would decrease to below 413.5 in. (10.5 m). In this event, the level restoration procedure of EOI-1 directs the operators to start pumps and verify injection valve operability in two or more of the following low-pressure injection systems: Condensate System, RHR System, or Core Spray System. Of course, two of these injection systems would provide much more injection than needed, but injection will not begin until the reactor vessel is depressurized and starting pumps in two systems is a prudent measure; unneeded pumps can be tripped after level restoration.

After verifying the availability of low pressure injection and if water level continues to decrease, the operators are instructed to depressurize the reactor vessel in order to restore water level with low pressure injection systems. For the depressurized operator action case of this chapter, the BWR-LTAS calculations show that the operators can restore and maintain level by using as few as one of the eight available low pressure pumps (four RHR pumps plus four core spray pumps). The condensate/condensate booster pumps would normally be the first choice for low pressure injection, but this injection source is compromised by the loss of control air, which incapacitates the condensate system minimum flow recirculation valves and the startup bypass valve.

The automatic CRD hydraulic system injection rate after reactor scram is about 112 gpm (7.07×10^{-3} m^3/s) while the reactor vessel is at 1015 psia (7 MPa), not enough to prevent core uncovery since the initial mass generation rate of steam by decay heat is much greater than the mass equivalent of this. The information provided in Appendix D demonstrates, however, that the CRD hydraulic system alone can provide sufficient injection to keep the core covered if the operator takes the actions required to enhance the rate of CRD hydraulic system injection flow. The necessary actions include starting the backup CRD hydraulic system pump, fully opening the pump discharge throttle valves, and other measures that are plant specific. A discussion of the actions that might be taken at a Browns Ferry unit is provided in NUREG/CR-3179. The important point here is that, if the operator knows how to maximize the CRD hydraulic system flow, the need for reactor vessel depressurization can be averted.

If the reactor vessel is depressurized, then the operator must take action to prevent overfill since the low-pressure injection systems have an enormous capacity and pumps not previously started by the operators will start automatically. Overfill of the reactor vessel would result in flooding of the inlets to the SRVs and the steam lines leading to the HPCI and RCIC system turbines. The operator does have the capability to turn off selected low-pressure system pumps, once they have started. In the discussion of Section 5.3, it is assumed that the operators skillfully use just one core spray pump, run intermittently, to maintain reactor vessel water level.

5.1.3 Reactor vessel pressure control

Would the Unit 1 control room operators take action to depressurize the reactor vessel immediately upon recognition of the failure-upon-demand of the HPCI and RCIC systems, or would the operators take action to enhance the flow injected by the CRD hydraulic system? If the latter action is chosen, the operators, after enhancing the flow, would have to stand firm while the reactor vessel water level continued to decrease and approached the top of the core, recognizing that the water level would ultimately recover as the core decay heat and the corresponding demand for injected flow subsided.*

The Browns Ferry EPG-based emergency procedure EOI-1, which is applicable after reactor scram and MSIV closure, requires remote-manual actuation of different SRVs in succession in order to minimize opening/closing cycles of the SRVs and to distribute the SRV discharge around the circumference of the pressure suppression pool so as to "minimize torus hot spots." However, when the drywell control air pressure is decaying, unnecessary remote-manual SRV actuation would deplete the precious temporary supply of pressurized air held in the six ADS SRV accumulators and the drywell control air receivers (see Section 2.2). Therefore, the Browns Ferry Operating Instruction that addresses Loss of Drywell Control Air Pressure (OI 32, Sect, V.C and V.D) directs the operators to minimize manual SRV actuation "to conserve accumulated air." The BWR-LTAS calculations discussed in this chapter are based on the assumption of automatic SRV actuation, which requires no control air, until depressurization is initiated. ORNL work on pressure suppression pool temperature distribution [D.H. Cook, "Pressure Suppression Pool Thermal Mixing," NUREG/CR-3472 (ORNL/TM-8906), October 1984] has shown that the "torus hot spot" problem is not as severe as previously supposed.

If the operators do choose to take manual action to depressurize the reactor vessel and exercise skill to use only an appropriate portion of the available low-pressure system injection capability, normal reactor vessel level control could be maintained, but only temporarily. Once the air stored in the ADS accumulators is depleted, about 6 h after reactor scram, the reactor vessel could not be maintained depressurized. Subsequently, the reactor vessel would repressurize and the low-pressure injection systems could not be used.

5.1.4 Containment pressure and temperature control

Loss of plant control air pressure and drywell control air pressure does not threaten the capability for pressure suppression pool cooling, but the air-coolers for the drywell atmosphere would fail early in the accident sequence because the dampers on the outlet side would fail closed on loss of drywell control air pressure. Without cooling, the drywell atmosphere rapidly heats up due to the presence of a large heat source in a small containment. This would lead to a very high drywell atmosphere temperature, particularly for cases in which the reactor vessel remained pressurized with an internal saturation temperature of 550°F (561 K).

Primary containment spray would remain available as an option and could be used in this accident sequence to reduce the drywell-wetwell pressure at any time. The Browns Ferry EPG-based emergency procedure

*For additional information, see "The Effect of Small-Capacity, High-Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One," NUREG/CR-3179, and Appendix D of this report.

EOI-2, Section 5.4, requires initiation of the drywell sprays if the drywell temperature exceeds its design value [281°F (411.5 K)], providing that the "suppression chamber" temperature is not in the forbidden region of the "Drywell Spray Initiation Pressure Limit" curve (Chart 5.4.6, EOI-2). However, the peak suppression chamber temperatures would be in the permissible region of the curve, so spray initiation would be allowed by procedure.

In the absence of spray, the average drywell atmosphere temperature would exceed the design value after 10.8 h, but 2 h later, it would be only slightly higher. There is a real question as to whether or not the use of containment spray would be justified for this accident sequence.

Pressures high enough to threaten primary containment integrity never arise in this accident sequence. At 12 h after the initiating reactor scram, the primary containment pressure, without the use of sprays, is only 26 psia (0.179 MPa).

Browns Ferry emergency procedure EOI-2, Section 5.3, requires that the operators "place all available RHR in the torus cooling mode" should the suppression pool temperature exceed 95°F (308 K). The calculations of this Chapter assume that one RHR heat exchanger is placed in the pool cooling mode 30 min after scram. The calculated results show that acceptable pool cooling is achieved even if only one of the four available RHR heat exchangers is employed.

Emergency procedure EOI-2, Section 5.3, also specifies a limit for reactor vessel pressure as a function of suppression pool temperature, the "Heat Capacity Temperature Limit" depicted by Chart 5.3.6 of the procedure. In the latter stages of the accident sequences discussed in this Chapter, the suppression pool temperature, with only one of the four RHR heat exchangers utilized, increases into the range in which reactor vessel depressurization would be required; however, this happens after 6 h, when there is no longer sufficient drywell control air pressure to permit remote-manual operation of the SRVs. Therefore, the specified depressurization cannot take place.

5.2 Cases in which the Reactor Vessel Remains Pressurized

5.2.1 Systems function as designed

In the first operator action case studied, the operators attempt to maintain a safe shutdown state without depressurizing the reactor vessel. The BWR-LTAS calculation is initialized at a point 30 s after reactor scram, with vessel water level 500 in. (12.7 m) above vessel zero. This is a typical post-scram water level as predicted by more sophisticated analysis codes such as RELAP. (BWR-LTAS is not intended to accurately simulate the very rapidly developing events that occur during the several seconds that immediately follow a scram from full power.) The BWR-LTAS results for this calculation are provided in Figs. 5.1 through 5.6.

The system transient for the first operator action case is initiated by an automatic scram from full reactor power on detected control

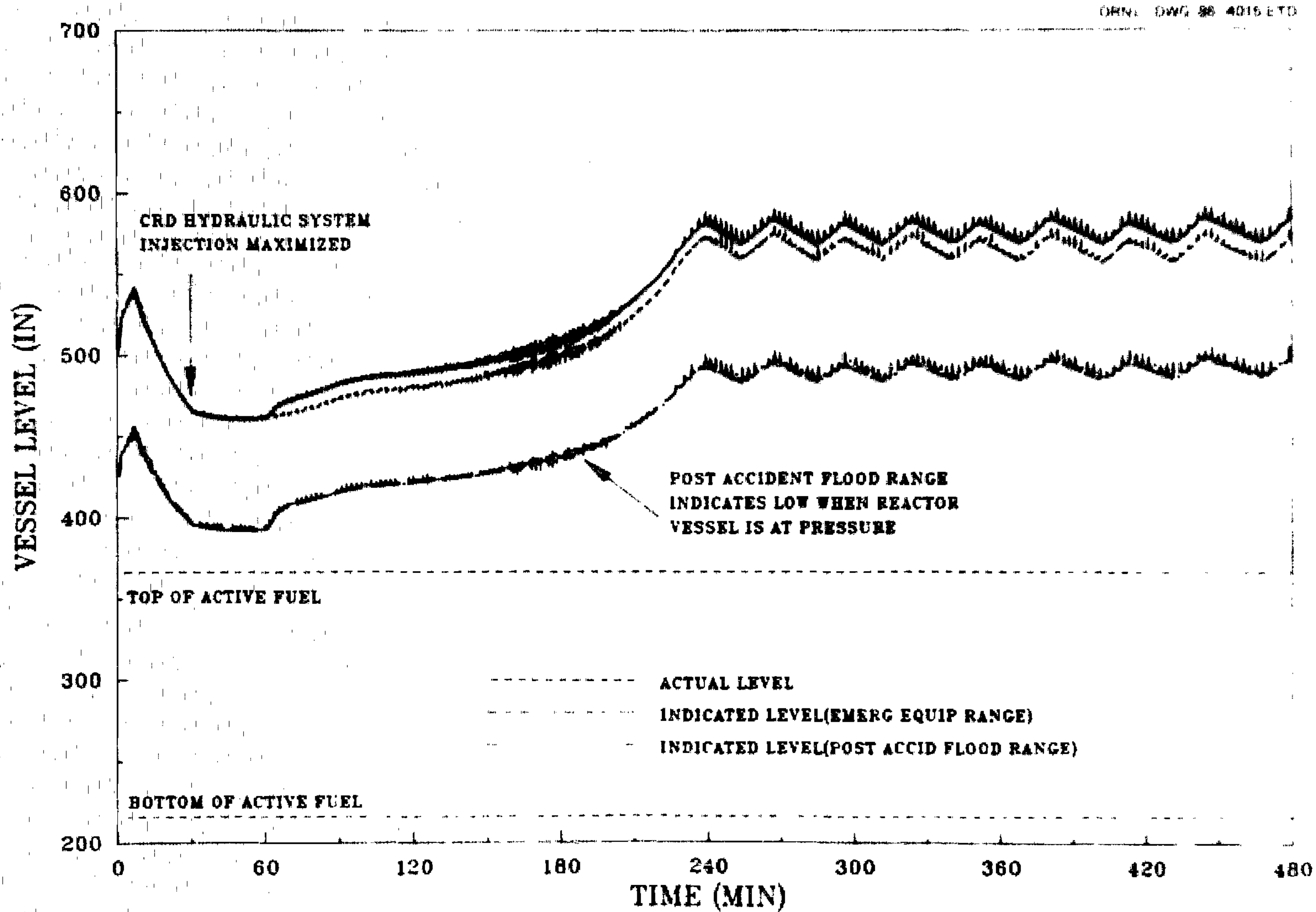


Fig. 5.1. Loss of control air operator action case with reactor vessel maintained at pressure — reactor vessel water level.

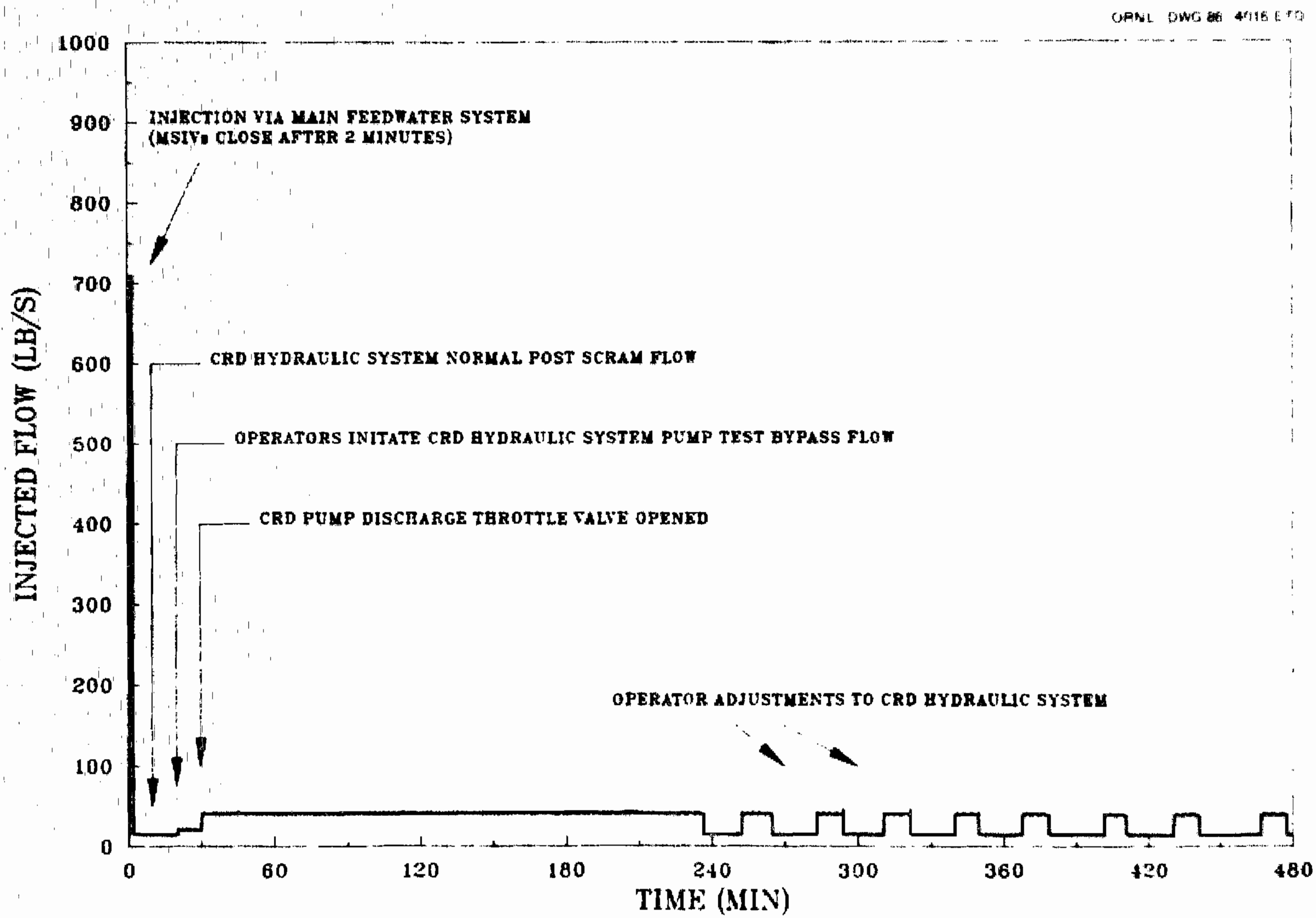


Fig. 5.2. Loss of control air operator action case with reactor vessel maintained at pressure — injected flow.

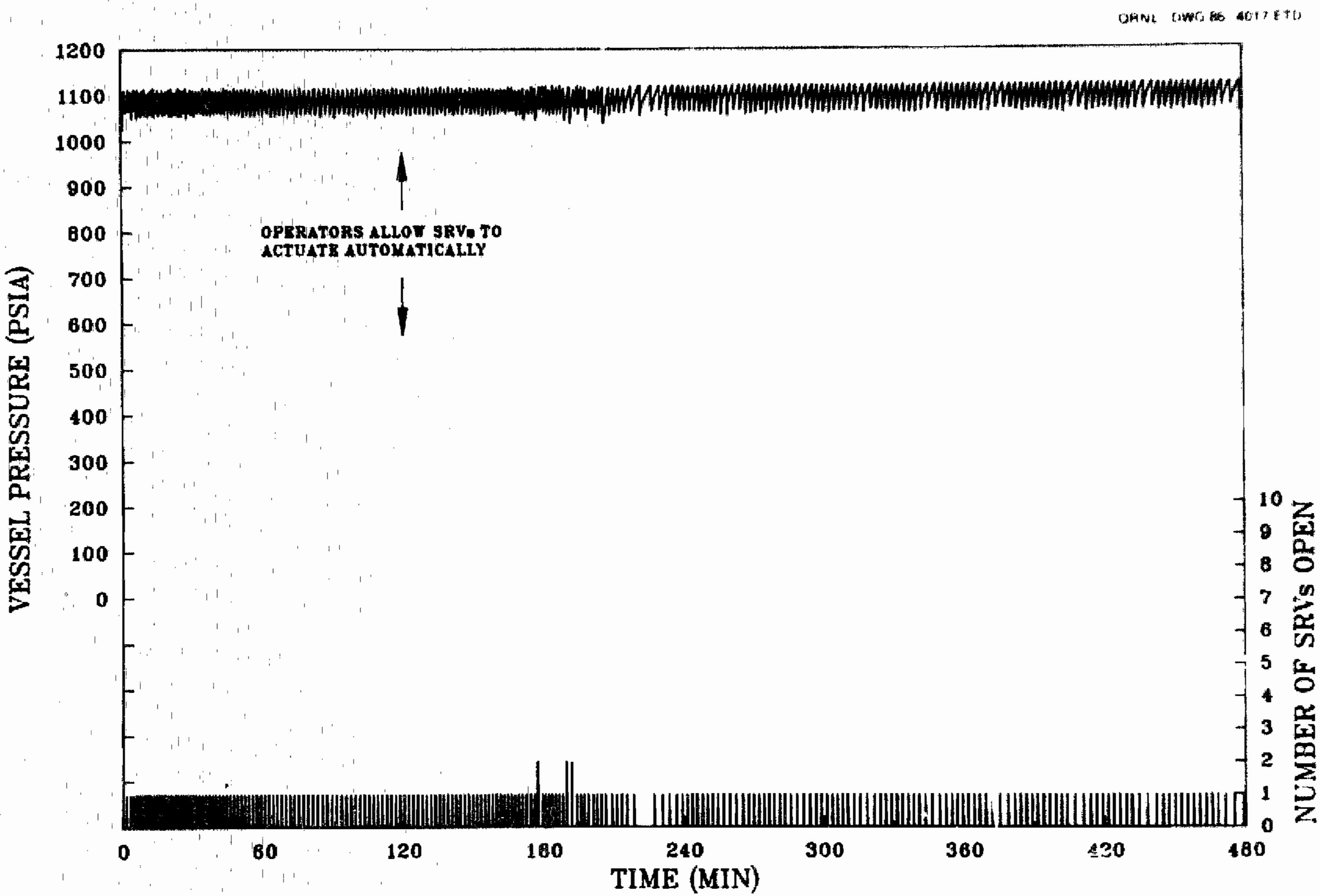


Fig. 5.3. Loss of control air operator action case with reactor vessel maintained at pressure — reactor vessel pressure and number of open SRVs.

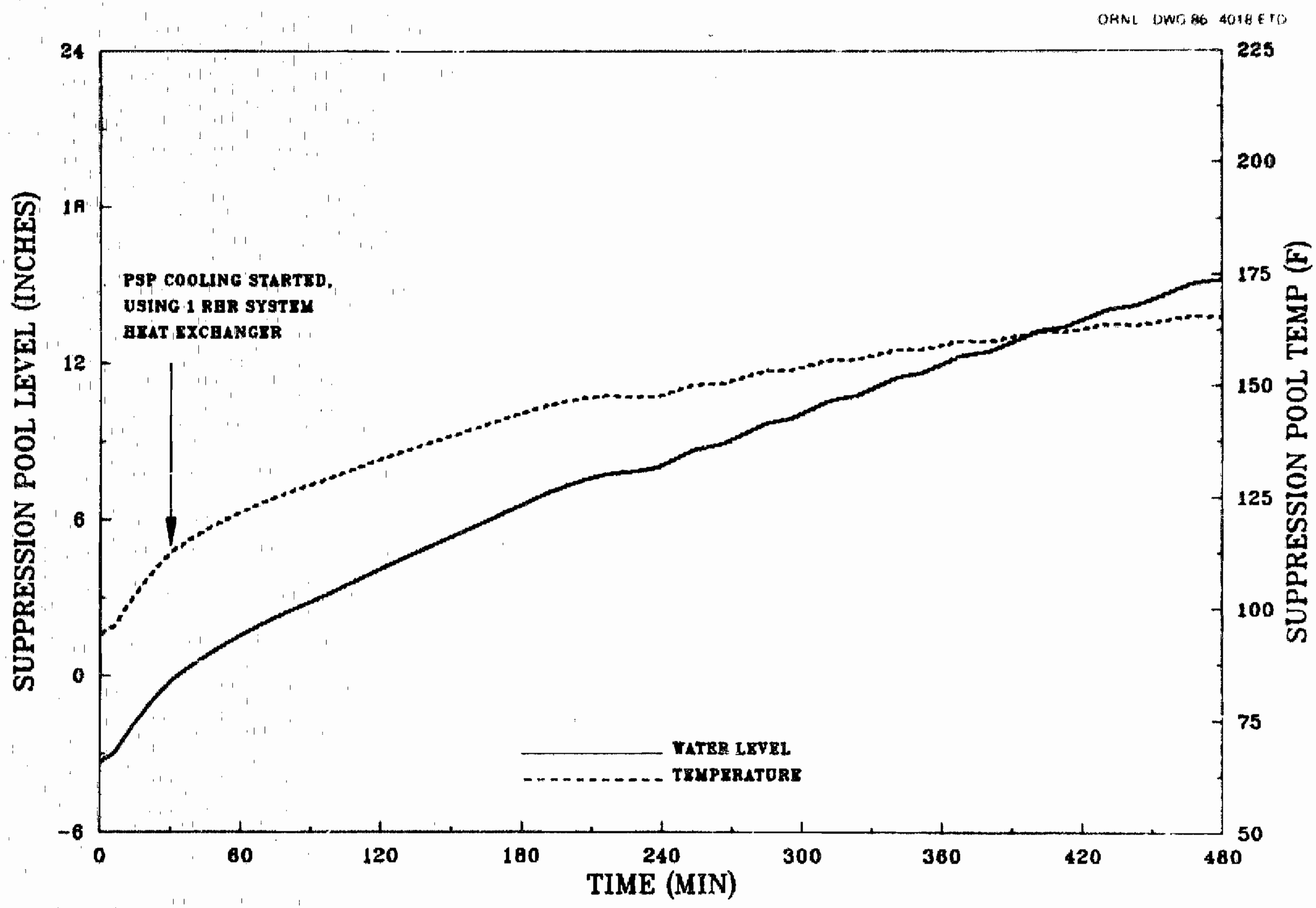


Fig. 5.4. Loss of control air operator action case with reactor vessel maintained at pressure — pressure suppression pool level and temperature.

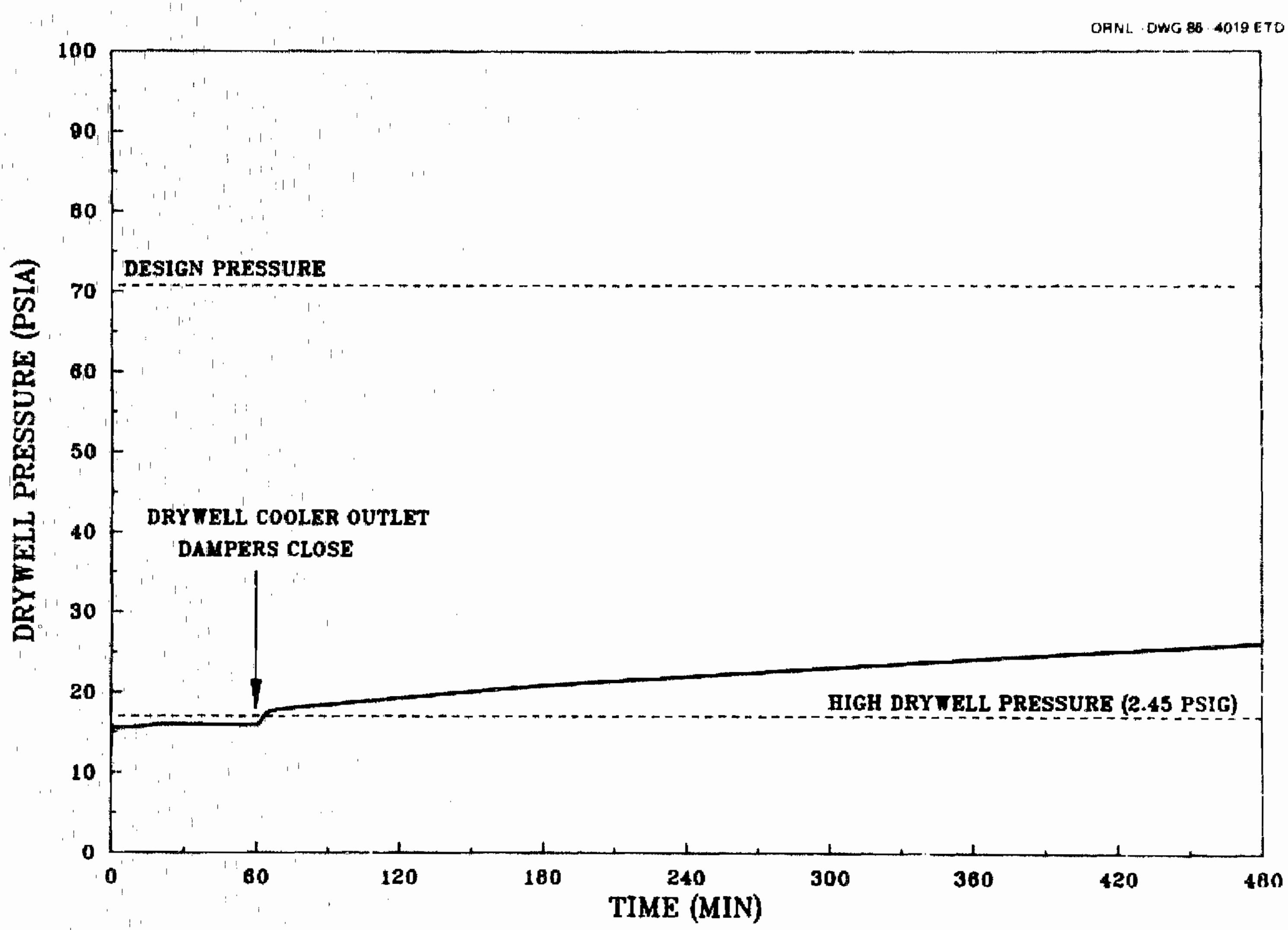


Fig. 5.5. Loss of control air operator action case with reactor vessel maintained at pressure — drywell pressure.

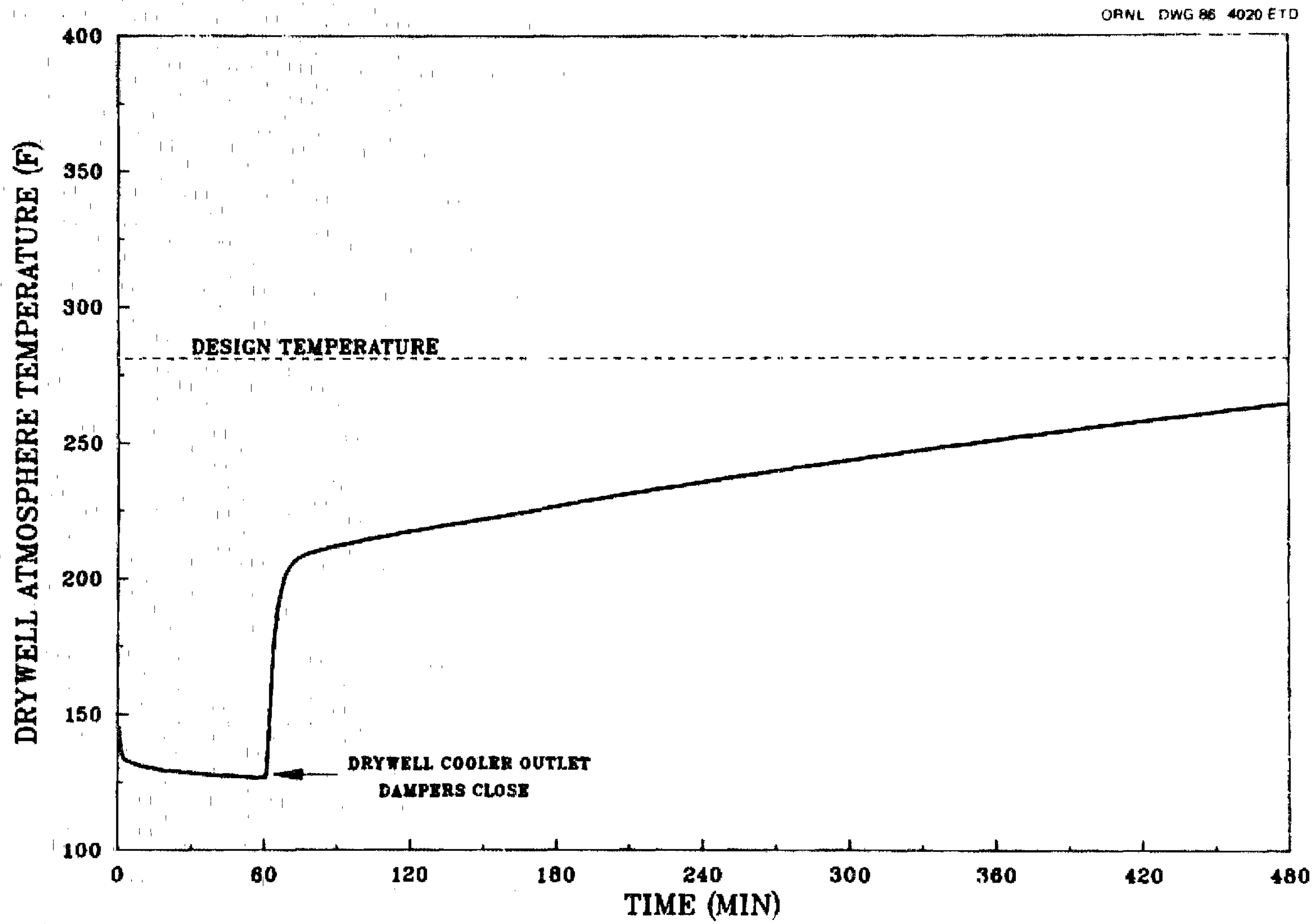


Fig. 5.6. Loss of control air operator action case with reactor vessel maintained at pressure — drywell atmosphere temperature.

air pressure below 60 psig (0.515 MPa), which is followed 2 min later by closure of the outboard main steam isolation valves (MSIVs). During the two minute period between reactor scram and MSIV closure, the main feedwater pumps continue to run, injecting about 10,000 gallons (37.86 m^3) of water into the reactor vessel. Since the steam generation rate is immediately reduced by the scram, this extra water increases the reactor vessel water level (Fig. 5.1) and provides additional time for the operators to enhance the CRD hydraulic system flow before the subsequent water level decrease to the point [413.5 inches (10.50 m) above vessel zero] that would make an emergency depressurization mandatory per the Emergency Procedure Guidelines (EPGs). However, by definition, the reactor vessel remains at pressure throughout this case.

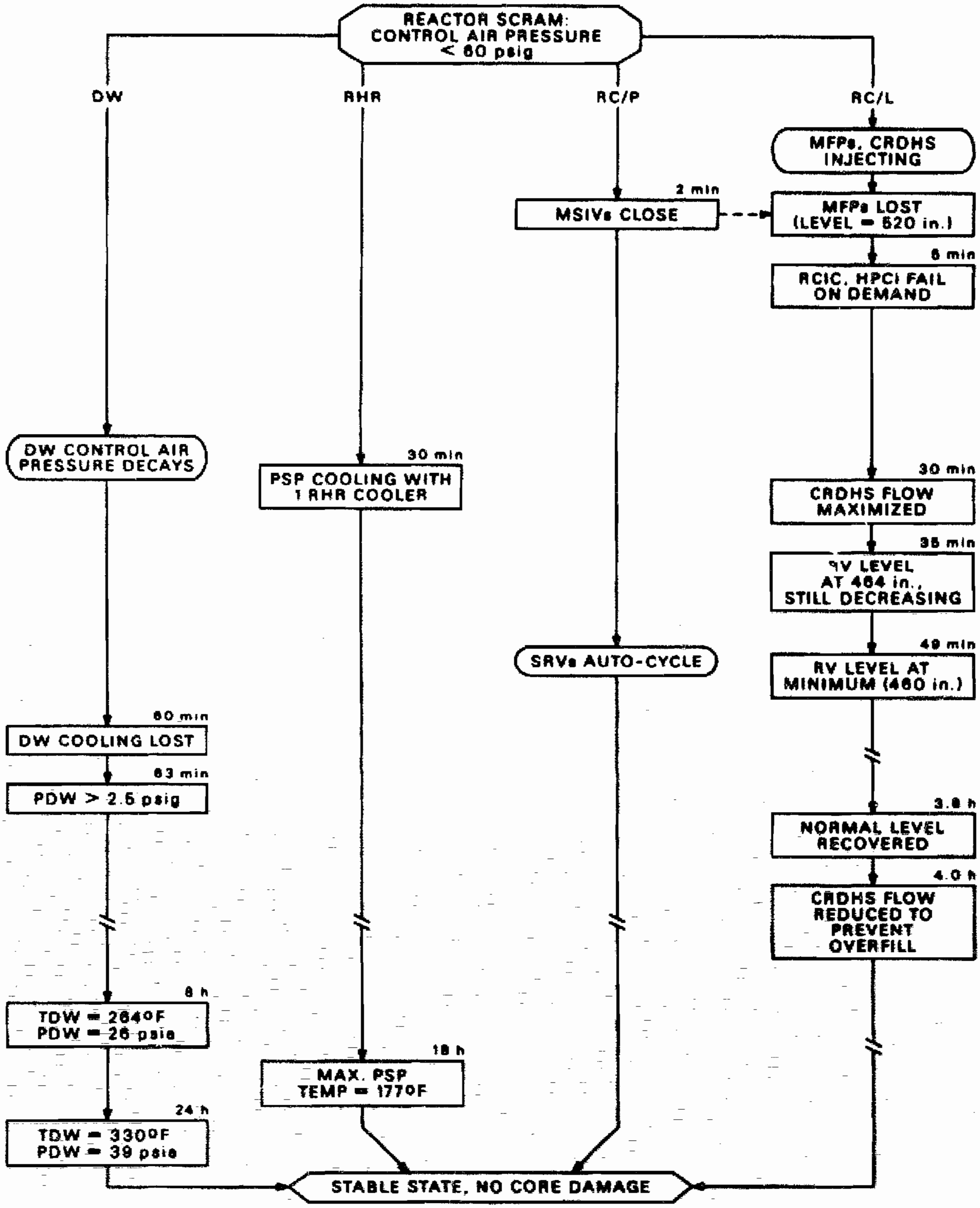
Table 5.1 provides a diagram of the approximate time-sequence of the significant events and illustrates their interdependence. The events are divided into four categories. On the far left, "DW" represents events related to drywell pressure or temperature. At the middle left, "RHR" represents the Residual Heat Removal System, used in the pressure suppression pool cooling mode during this sequence. On the middle right, the column of events for reactor vessel pressure control is labeled "RC/P" and on the far right, "RC/L" heads the column for events that control, or depend upon, reactor vessel water level.

To maintain the reactor vessel at pressure, the operators need only allow the SRVs to actuate automatically,* which results in vessel pressures (Fig. 5.3) between the 1120 psia (7.72 MPa) opening setpoint and the approximately 50 psi (0.345 MPa) lower closing setpoint of the lowest-set bank of SRVs. Automatic SRV actuation does not require the availability of drywell control air to reposition the pilot valves since this is accomplished by the internal pressure of the reactor vessel. Except for a period of a few seconds immediately following scram, when the reactor power has not yet reached decay heat levels, only single SRV operation is required to maintain reactor vessel pressure. Since the individual SRV setpoints are established to a $\pm 1\%$ tolerance, equivalent to a range of about 22 psi (0.15 MPa), the four SRVs in the lowest-set bank are not expected to cycle simultaneously. Rather, the one of these valves with the lowest actual setpoint would cycle repeatedly, as required to relieve the steam produced by decay heat.

Since only the CRD hydraulic system is injecting [at a rate of about 105 gpm (0.00652 m^3/s)], the reactor vessel water level continues to slowly decrease after the reactor scram. The operators first attempt to start the 600 gpm (0.037 m^3/s) RCIC system, and, upon failure, attempt to start the 5000 gpm (0.31 m^3/s) HPCI system. Faced with failure of both systems, the operators reach a decision point. They can take action to enhance the CRD hydraulic system injection rate, or they can depressurize the reactor vessel and supply the necessary additional injection by means of one of the low-pressure injection systems. It is assumed here that the operators decide to conserve the remaining drywell

*Under normal post-scram conditions, the operators are trained to manually cycle the SRVs but OI 32, "Loss of Drywell Control Air Pressure," instructs the operator to minimize manual SRV actuations "to conserve accumulated air" (see Section 6.2).

Table 5.1. Sequence of events for loss of control air with operator action — reactor vessel maintained at pressure.



control air system pressure by avoiding reactor vessel depressurization and take action to enhance the CRD hydraulic system injection.

The easiest actions would be attempted first. Since the control room indication of the rate of CRD hydraulic system injection would be pegged high at the "100 gpm" point, the effect of the operator's actions would have to be verified by observation of the resulting trend in reactor vessel water level. Ten minutes after scram, the control room operators start the spare CRD hydraulic system pump. The increase in flow would be insufficient and downcomer water level (Fig. 5.1) continues to decrease.

After dispatching an assistant operator to the reactor building to open the hand-operated valve 85-551 and to close the electrical supply breakers for motor-operated valve 85-50, the control room operator is able to initiate flow through the pump test bypass line, which injects CRD hydraulic system flow directly into the reactor vessel via a feedwater line. This action increases the total injected flow to about 145 gpm (0.009 m^3/s), and would be expected to occur about 20 min after the reactor scram. This increased injection rate is still insufficient to maintain the reactor vessel water level.

As reactor vessel water level continues to decrease, the assistant operator is again sent to the reactor building, this time to open the 85-527 throttling valve at the CRD hydraulic system pump discharge. The throttling valve is assumed to be opened 30 min after the scram, which results in an increase in injection rate to 290 gpm (0.018 m^3/s). Although this injection rate is sufficient to prevent eventual core uncovery, the reactor vessel water level temporarily continues to decrease, very slowly, and 19 min later (at 49 min after scram), reaches its minimum of 460 in. (11.68 m) above vessel zero. This is some 94 in. (2.39 m) above the height of the top of the active fuel (TAF) in the core. Subsequently, the reactor vessel water level slowly increases.

Four hours later, the reactor vessel water inventory is restored to its normal level of 561 in. (~14.25 m) and eventually, the control room operators would have to take action to reduce the CRD hydraulic system injection rate to prevent reactor vessel overfill.

The operators initiate the pressure suppression pool cooling mode of the RHR system at 30 min after the scram, and even with the assumption that only one of the four available RHR heat exchangers is utilized, the maximum suppression pool temperature is controlled to below 180°F (355 K). This is a low enough temperature to preclude inadequate net positive suction head problems for the RHR pumps even if the containment were not pressurized (Fig. 5.5) to about 27 psia (0.186 MPa). The average drywell atmosphere temperature (Fig. 5.6) is 264°F (402 K) at the end of the 8 h calculation. After another 3 h, it would reach the 281°F (411.5 K) design temperature of the drywell, at which point the emergency procedure guideline-based procedures require manual initiation of the containment spray mode of the RHR system to reduce the atmosphere temperature. This action could be accomplished since the valves required to initiate the sprays are all motor-operated.

The effect of containment sprays has not been simulated in this BWR-LTAS calculation. The drywell atmosphere temperature reaches its design value at the 11-h point, and obviously, use of drywell sprays would reduce the temperature of the drywell atmosphere. However, without sprays the average drywell atmosphere temperature would increase only another 49 Fahrenheit degrees (27 K) during the next 13 h, reaching 330°F (439 K) 24 h after reactor scram.

5.2.2 Effect of stuck-open relief valves

As will be shown in Section 5.3, operator action to depressurize the reactor vessel has only temporary effect during the Loss of Control Air accident sequence. However, if one or more of the reactor vessel safety/relief valves were to stick open during the early part of the sequence, the effect would be to guarantee long-term depressurization of the reactor vessel. This would permit continuous reactor vessel water level control by means of a low-pressure injection system and would therefore be beneficial in preventing the development of the sequence into a Severe Accident.

5.2.3 Effect of failure of the CRD hydraulic system

Failure of the CRD hydraulic system would leave no means of high-pressure injection to the reactor vessel and the only hope for avoidance of a Severe Accident would lie in reactor vessel depressurization.

If the CRD hydraulic system were to fail within 6 h after reactor scram, then there should be sufficient stored air in the individual accumulators associated with the ADS safety/relief valves to permit the operators to depressurize the reactor vessel. Subsequently, the operators could temporarily use a low-pressure injection system to maintain normal reactor vessel water level, but the stored air in the ADS accumulators would be depleted after about 6 h, and the reactor vessel would repressurize.

From the time that the reactor vessel pressure exceeded the shutoff head of the low pressure injection systems, it would require about 2.4 h for the vessel pressure to reach the setpoint [1120 psia (7.72 MPa)] for automatic relief valve actuation. A portion of the stored reactor vessel water inventory would be lost with each subsequent relief valve cycle. Core uncovery would occur about 90 min after the renewal of SRV actuation; this would be about 10 h after scram.

If the CRD hydraulic system failure were to occur more than 6 h after scram, the operators would not be able to depressurize the reactor vessel, since the stored air in the ADS accumulators would have previously dissipated. Boiloff of the reactor vessel water inventory would begin immediately, leading to core uncovery about 90 min after loss of the CRD hydraulic system.

5.2.4 Emergency action levels

The emergency action levels for this case are essentially the same as those for the more-probable case with early reactor vessel depressurization, discussed in subsection 5.3.4.

5.3 Cases in which the Operators Depressurize the Reactor Vessel

5.3.1 Systems function as designed

The second operator action case analyzed in this report differs from the first in that the MSIVs are assumed to drift closed before the scram and the operators are assumed to take action to depressurize the reactor vessel. As in the first case, the accident sequence is initiated by an automatic scram but in this case, the Reactor Protection System signals the scram as the outboard MSIVs, without normal plant control air pressure, drift to less than 10% open. The results of BWR-LTAS calculations for this case are presented in Figs. 5.7 through 5.12. The time-sequence of events and the interrelation of events is diagrammed on Table 5.2.

After MSIV closure, the feedwater turbines are deprived of steam and cease to operate, and without any post-scram injection from the main feedwater pumps and with only the CRD hydraulic system injecting at about 105 gpm (0.0066 m^3/s), the reactor vessel water level (Fig. 5.7) decreases to 413.5 in. (10.50 m) after only 20 min. With the assumption that there is no prospect for restart of HPCI or RCIC, the operators initiate a manual emergency depressurization. As the reactor vessel depressurizes (Fig. 5.9), the operators avoid the vessel flooding problem of the no-operator-action case by turning off most of the low pressure injection pumps. One core spray system pump is allowed to run for refill of the vessel, and is tripped 15 min later to avoid overfill after the normal vessel water level of 561 in (14.25 m) is regained. Intermittent core spray pump operation (Fig. 5.8) is used over the next several hours to maintain level as the reactor vessel remains at low pressure.

Recognizing that the loss of drywell control air pressure will eventually interfere with the capability for remote operation of the SRVs, the operators decide to go into the shutdown cooling mode of the RHR system as soon as possible. (If the core decay heat can be continuously removed, the reactor vessel can be maintained depressurized even if the SRVs are inoperable.) They accomplish this task about 40 min after the reactor scram. However, shutdown cooling is successful for only about 25 min, because a high drywell pressure signal is received 65 min after the scram (Fig. 5.11), and the isolation valves between the reactor vessel and the RHR pump suctions go shut and are interlocked shut.

It is possible that the Primary Containment Isolation System (PCIS) shutdown cooling supply valve interlocks could be defeated by actuating bypass circuitry at the Backup Control Panel located outside the control room. However, the Backup Control Panel is intended to be used in the event of loss of control room habitability and there are no procedures in place to instruct the operators in selective use of the particular backup control panel controls that would have to be actuated to override specific interlocks. The drywell cannot be vented to reduce its internal pressure to below the 2.45 psig (0.118 MPa) setpoint for PCIS

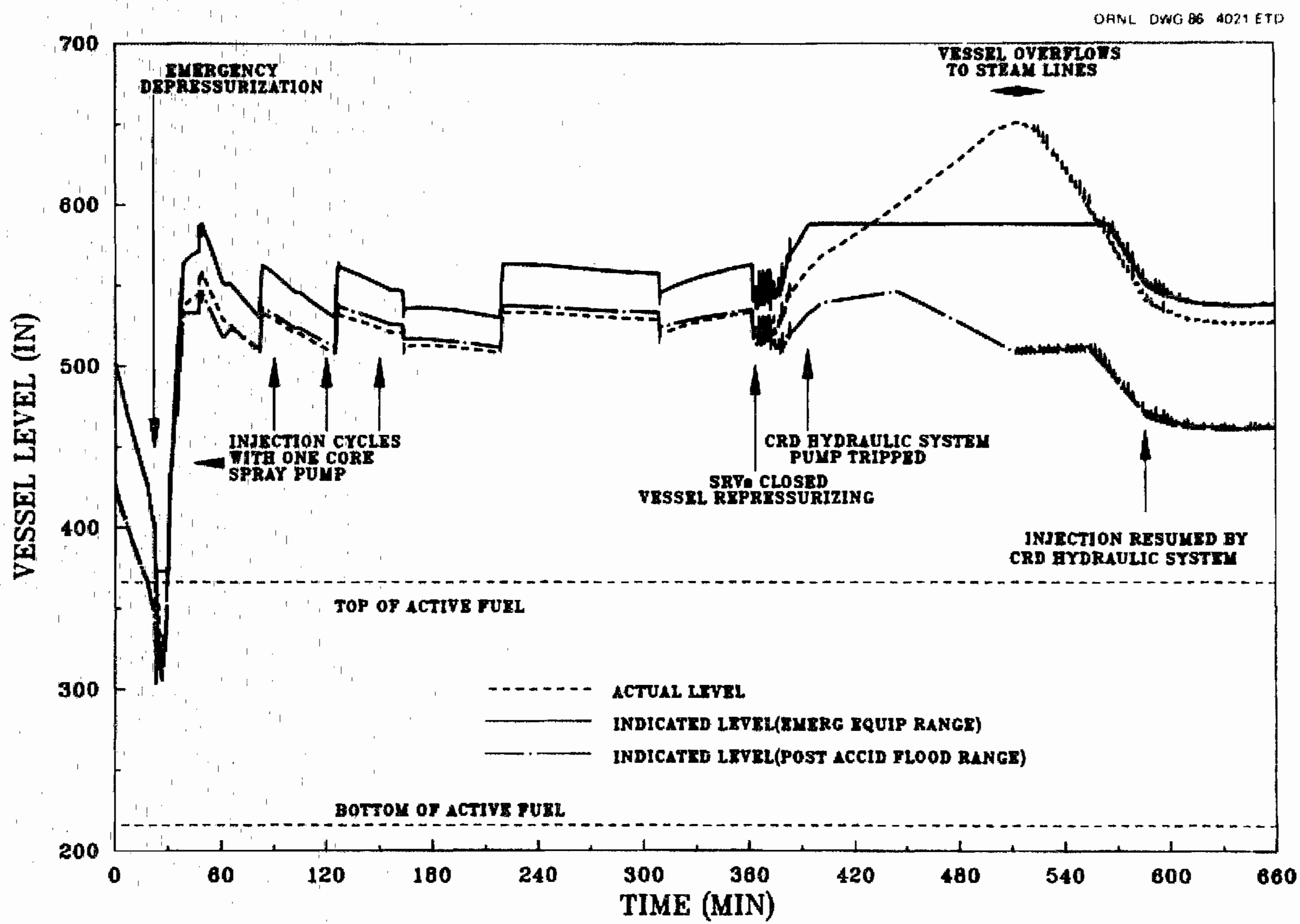


Fig. 5.7. Loss of control air operator action case with depressurization — reactor vessel water level.

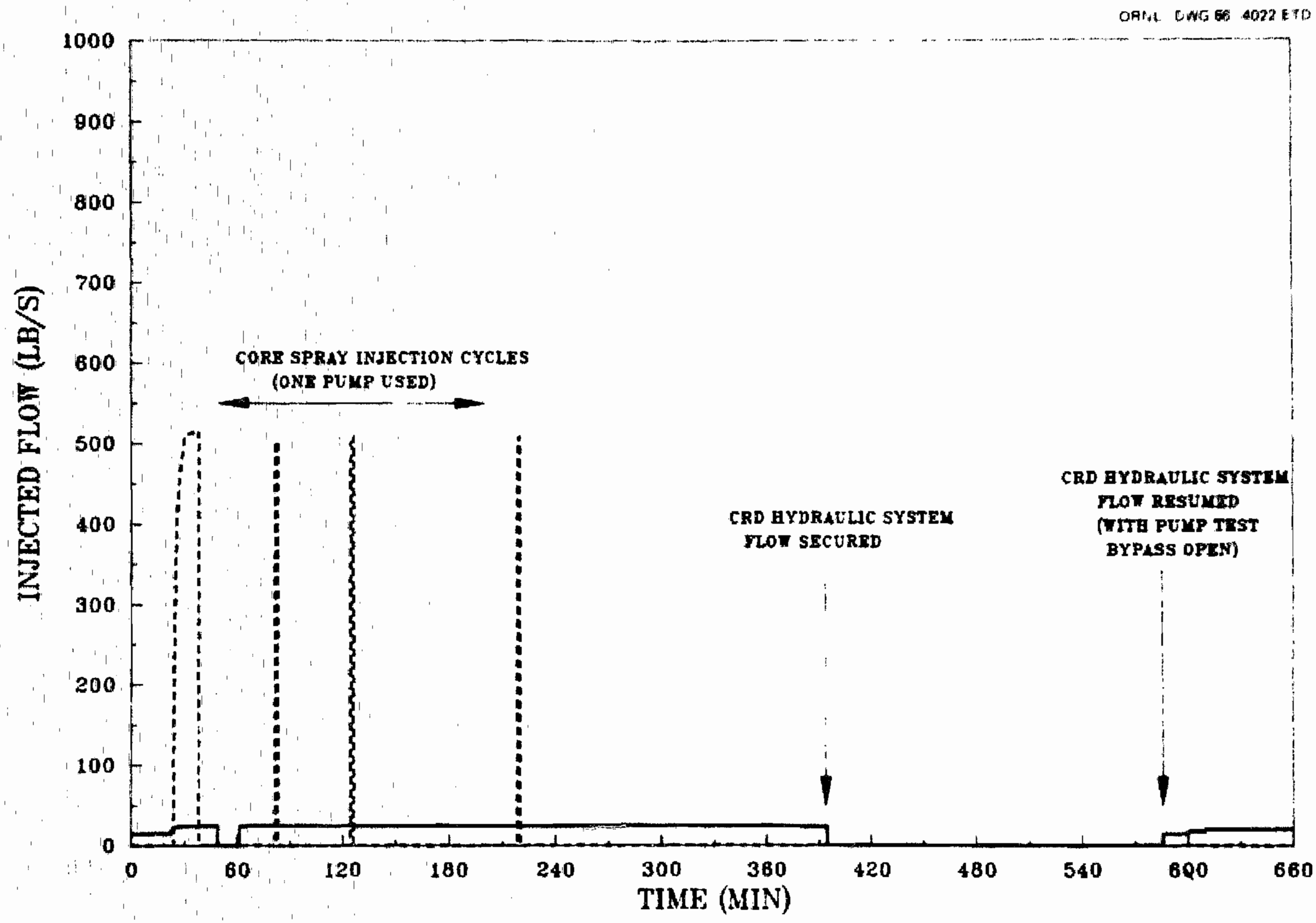


Fig. 5.8. Loss of control air operator action case with depressurization — injected flow.

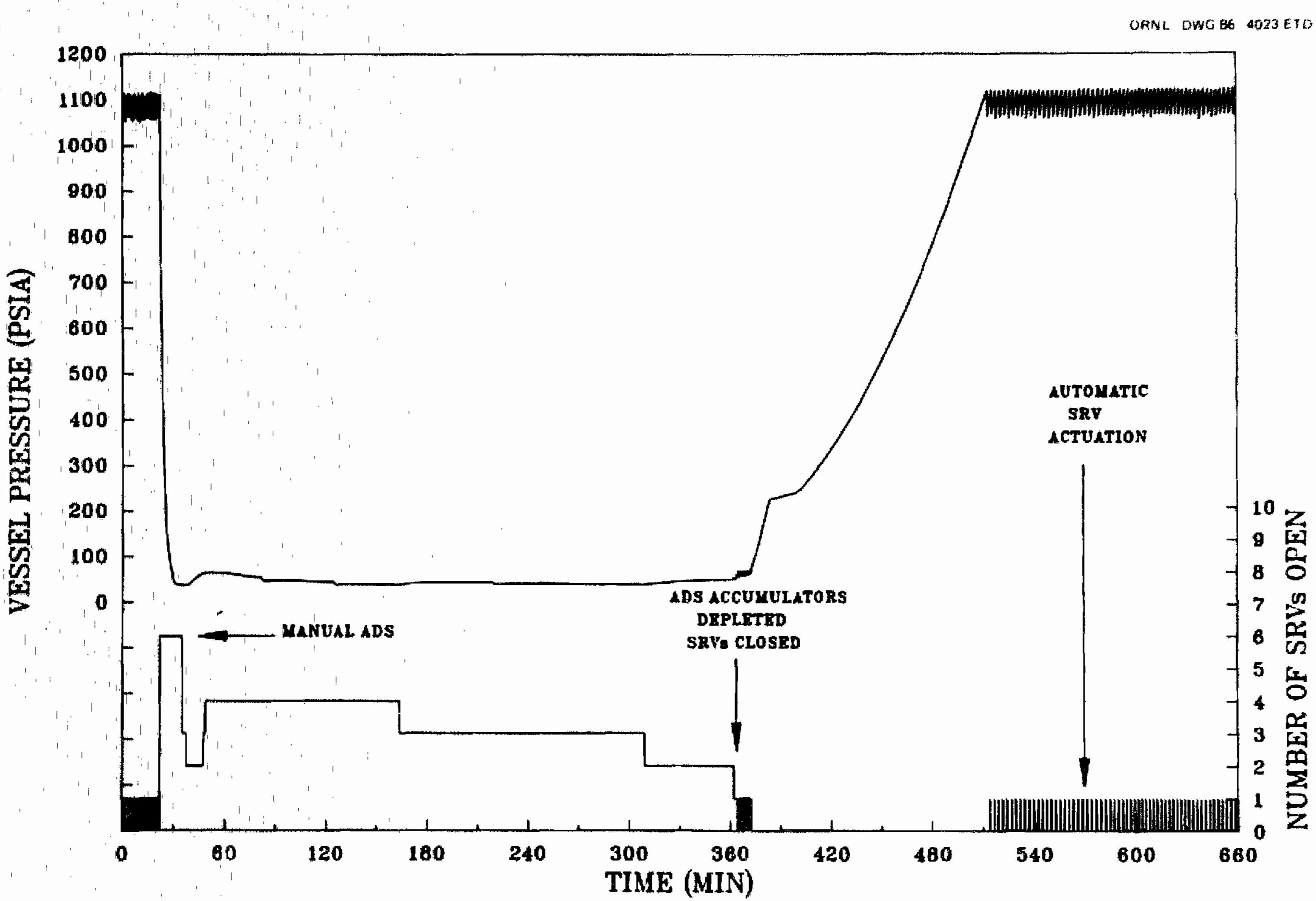


Fig. 5.9. Loss of control air operator action case with depressurization — reactor vessel pressure.

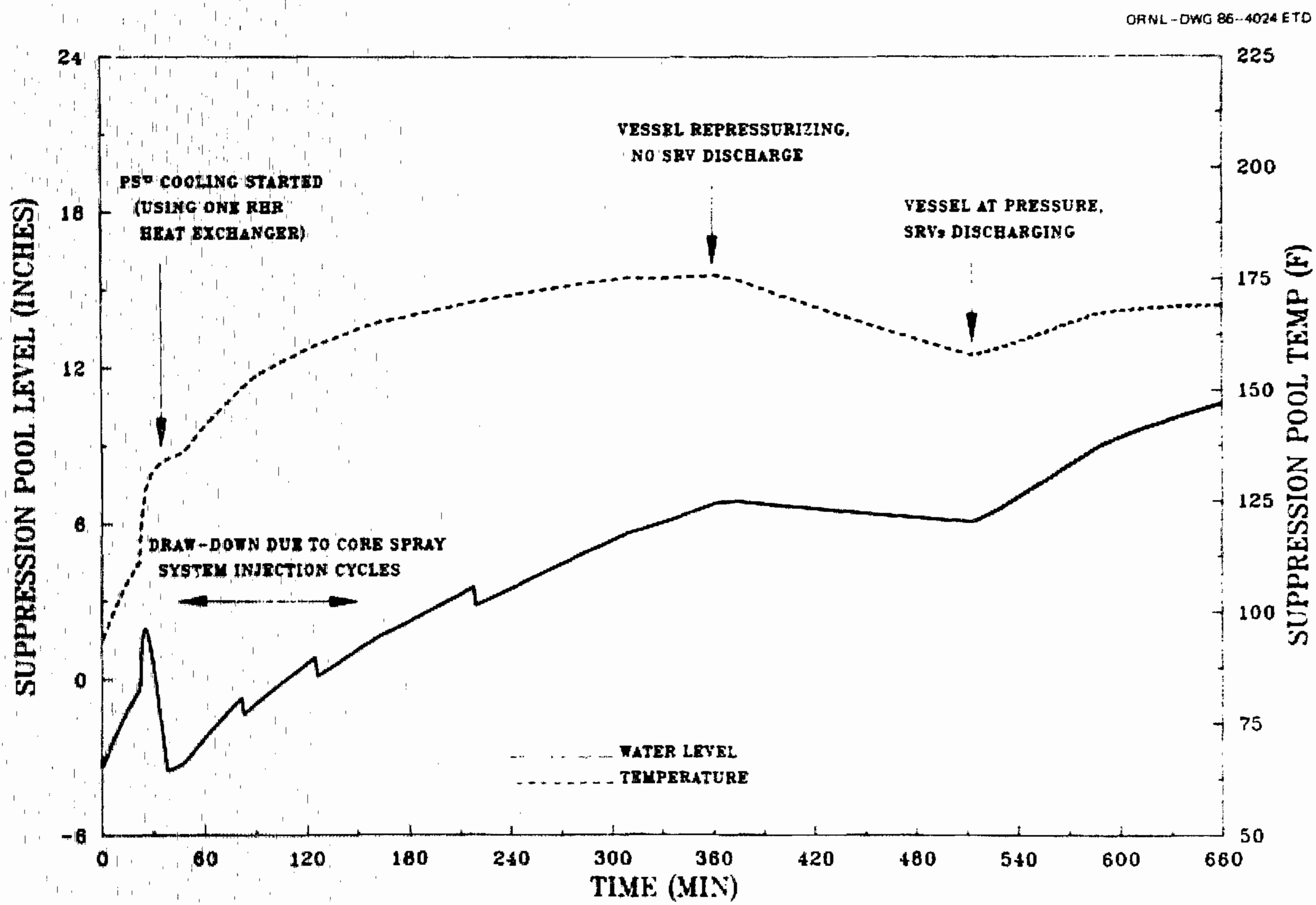


Fig. 5.10. Loss of control air operator action case with depressurization — pressure suppression pool level and temperature.

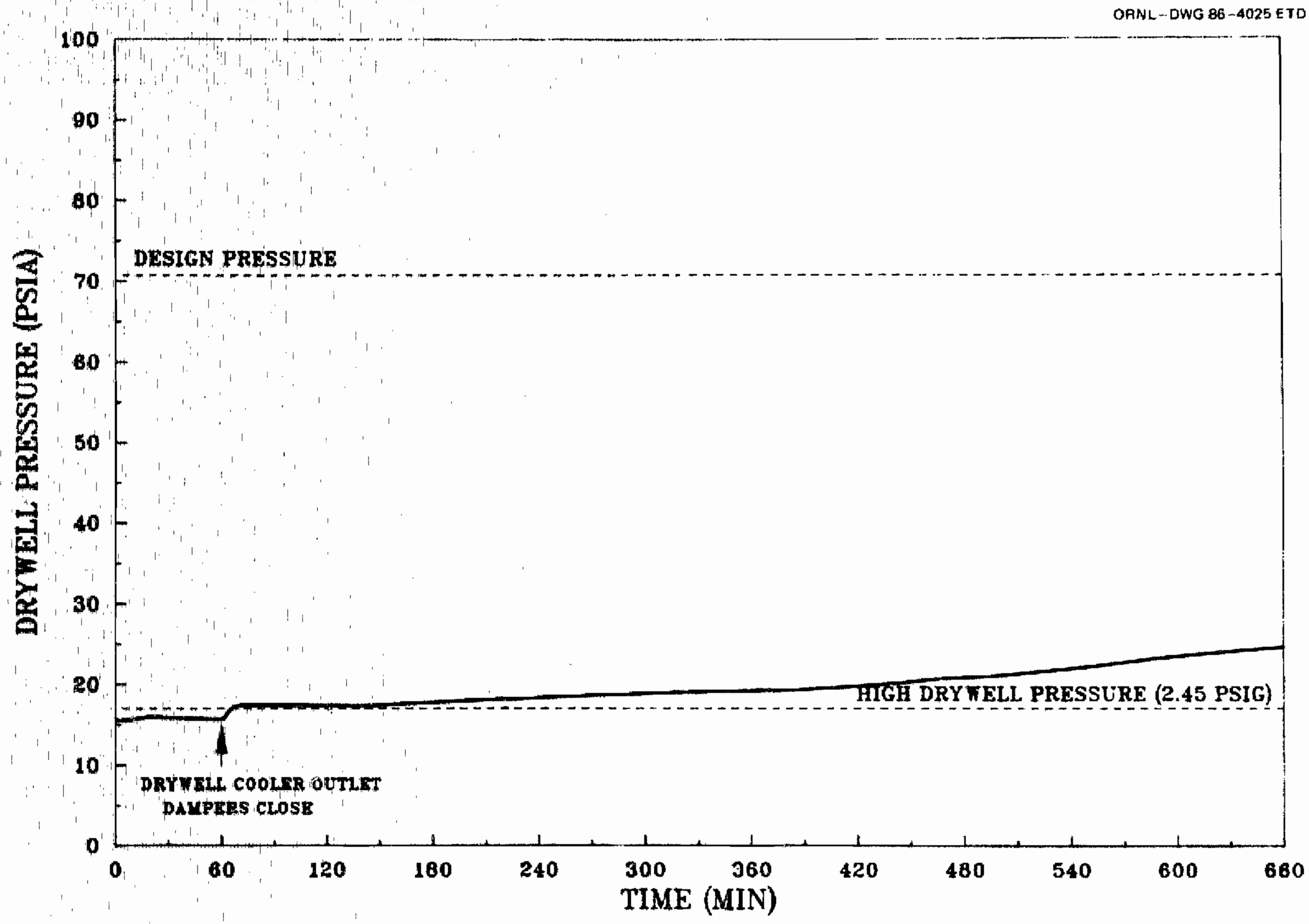


Fig. 5.11. Loss of control air operator action case with depressurization — drywell pressure.

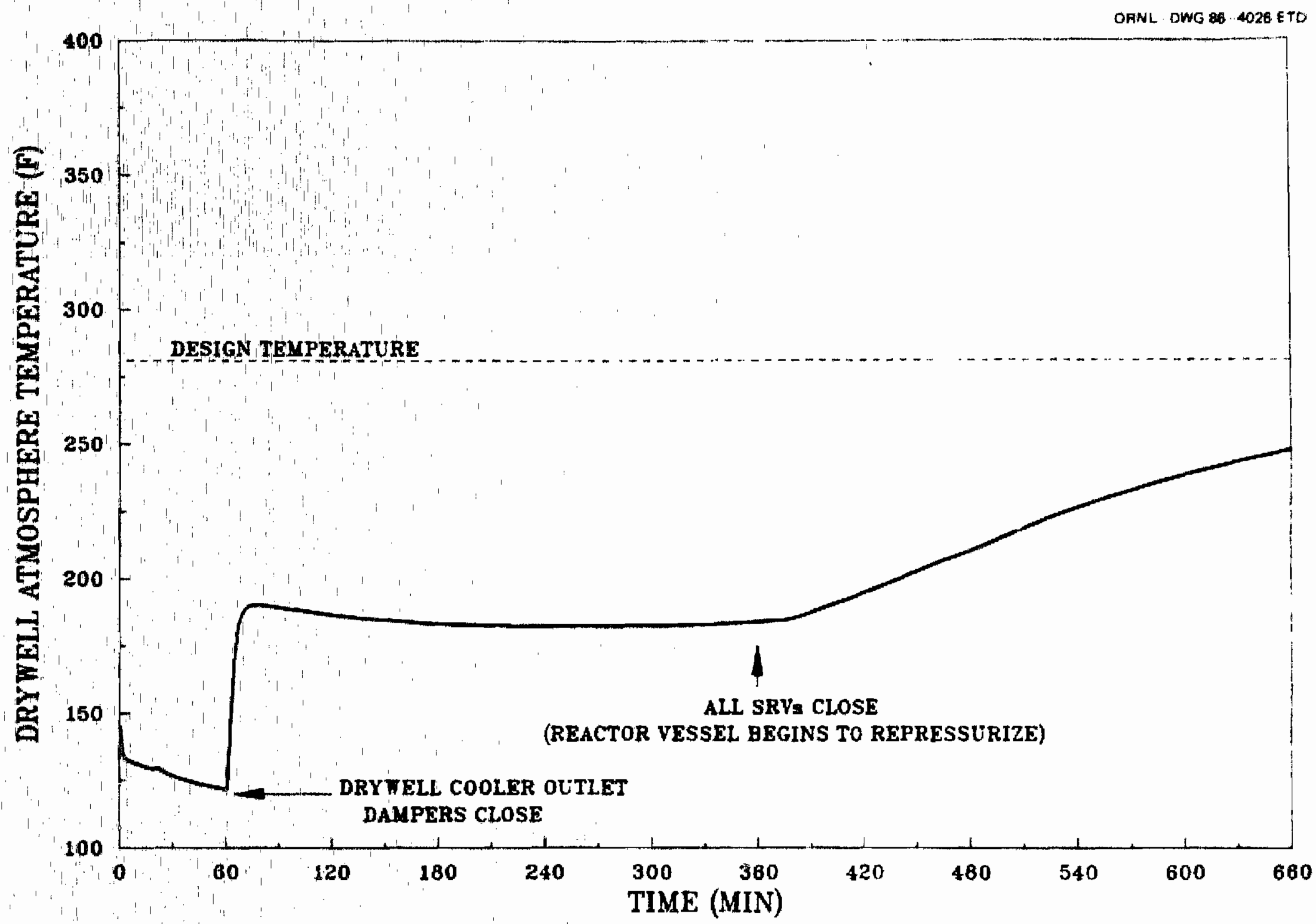


Fig. 5.12. Loss of control air operator action case with depressurization — drywell atmosphere temperature.

Table 5.2. Sequence of events for loss of control air with operator action — reactor vessel depressurized.

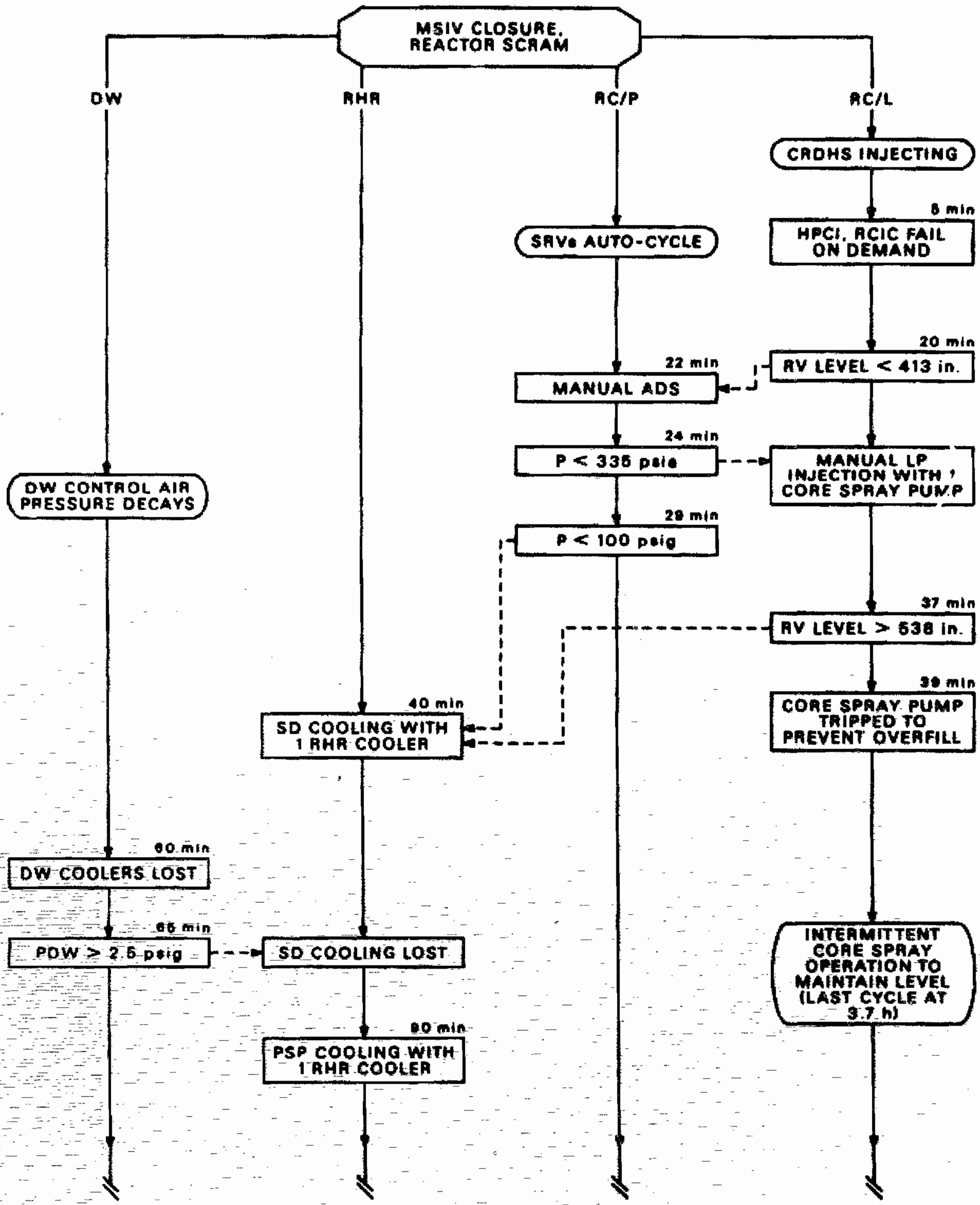
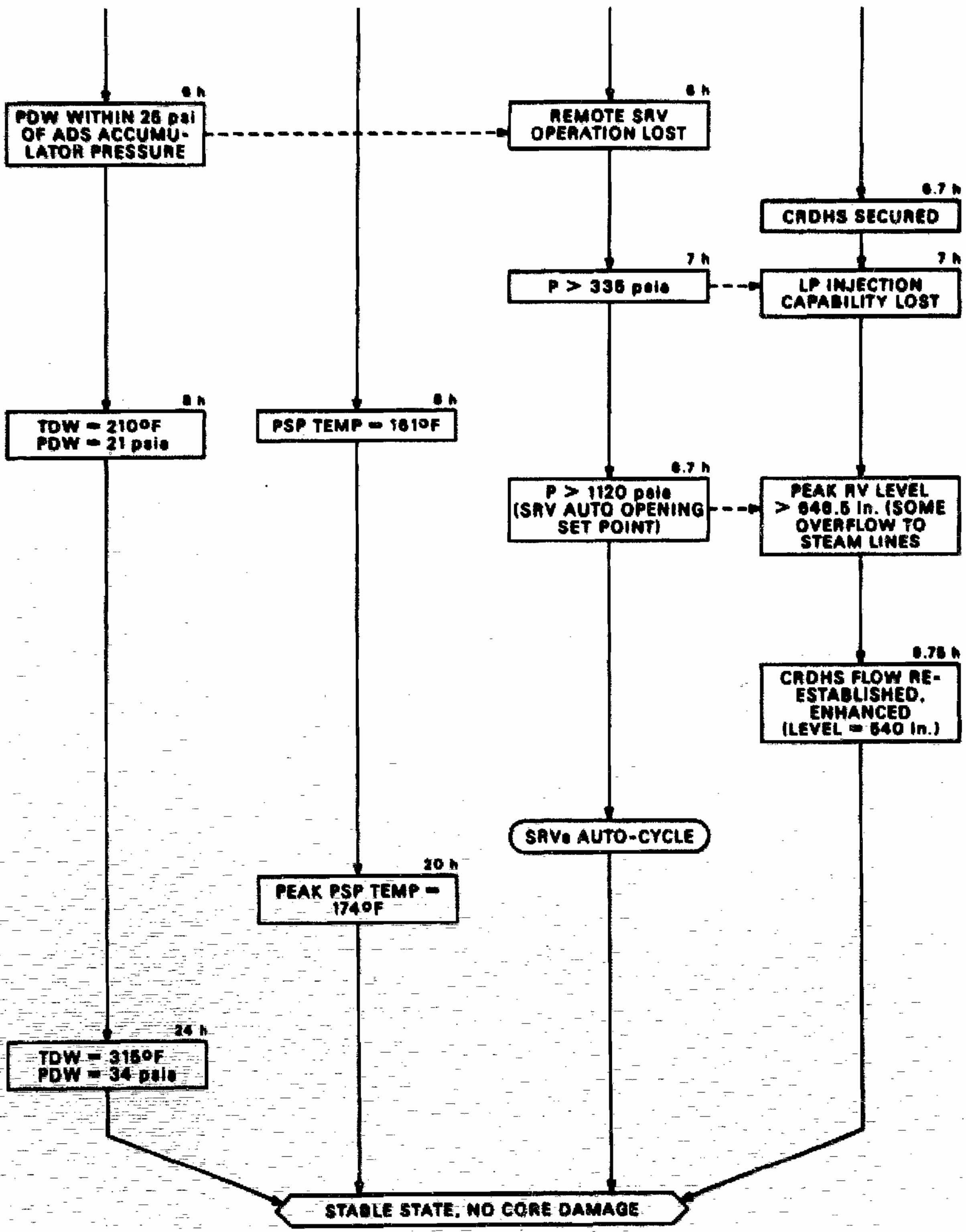


Table 5.2 (continued)



protection logic because the necessary containment ventilation system valves cannot be opened without the availability of plant control air.

After termination of shutdown cooling, the reactor vessel remains depressurized until there is no longer sufficient air pressure in the ADS accumulators to hold one or more SRVs open. About 6 h after reactor scram, the last open SRV closes and the reactor vessel begins repressurizing. During the period that the decay-heat generated steam is pressurizing the reactor vessel, there is virtually no inventory loss from the vessel, and the liquid coolant swells as it is heated by the core decay heat.

The CRD hydraulic system, if permitted to continue to operate, would increase the reactor vessel water inventory during the period of repressurization and thereby contribute to the rate of level increase. It is assumed here that the operators trip the CRD hydraulic system pumps just before the indicated reactor vessel water level goes off-scale high, which would occur at 588 in. (14.93 m).

The BWR reactor vessel contains a large volume of water. Even after trip of the CRD hydraulic system pumps, the reactor vessel water level continues to increase as a result of decay heating and density decrease. The water level reaches 646.5 in. (16.42 m), the level of the lower lip of the main steam line nozzles, and calculations indicate that 334 ft^3 (9.45 m^3) of water would be spilled over into the four steam lines. This overflow would fill the lower horizontal and vertical piping runs immediately upstream of each closed inboard MSIV, but would not result in the introduction of water to the SRVs, which are mounted on the upper surface of the upper horizontal steam line piping runs.

About 2.4 h after the capability for remote, air-operated, SRV actuation is lost, the reactor vessel would have repressurized to the 1105 psig (7.720 MPa) setpoint for automatic, steam-operated, actuation of the lowest-set group of SRVs. Subsequently, the reactor vessel would remain at pressure, with intermittent operation of the lowest-set SRV. Inventory loss through the cycling SRV would, in the absence of recovery of HPCI or RCIC, necessitate the resumption of CRD hydraulic system flow, enhanced to about 145 gpm (0.0092 m^3/s), in order to avoid eventual core uncovery. Failure to resume CRD hydraulic system flow would result in core uncovery within about 90 min after automatic SRV operation resumed, which corresponds to about 10 h after the sequence-initiating reactor scram.

Operation of the RHR system in the pressure suppression pool cooling mode is assumed to begin 65 min after the reactor scram. Even with only one of the four RHR system heat exchangers applied to pool cooling, the peak suppression pool temperature (Fig. 5.10) would not exceed 180°F (355 K). The drywell atmosphere bulk temperature (Fig. 5.12) remains below the 281°F (411.5 K) design temperature of the drywell for more than 12 h. After 24 h, the average drywell temperature, in the absence of any application of drywell sprays, would reach 315°F (431 K).

5.3.2 Effect of stuck-open SRVs

As in other cases, the effect of one or more stuck-open relief valves would actually be beneficial, permitting the reactor vessel to remain depressurized after the capability for remote, air-operated, SRV actuation is lost. With permanent reactor vessel depressurization, the core could be cooled indefinitely by means of the low-pressure injection systems.

5.3.3 Effect of failure of the CRD hydraulic system

Reactor vessel water level can be maintained by operator control of one core spray system pump during the first 6 h after scram because the reactor vessel is kept depressurized by the six open SRVs associated with the ADS system. During the period between 6 h and 8.4 h after scram, while the reactor vessel is repressurizing, there is no need for additional injection since the SRVs are closed and there is no inventory loss from the vessel.

Failure of the CRD hydraulic system after 8.4 h would lead to a boil-off of the remaining reactor vessel water inventory above the core and core uncovery about 90 min later. Therefore, even with failure of the CRD hydraulic system, core uncovery would not occur earlier than 10 h after the reactor scram.

5.3.4 Emergency action levels and timing

This discussion is based upon a review of the BWR-LTAS accident sequence calculations, the Browns Ferry Radiological Emergency Plan, and the corresponding Implementing Procedures Document.

Total failure of the Plant Control Air System might be caused by explosion or fire, events that in themselves would cause declaration of an UNUSUAL EVENT at the plant. If the cause of loss of plant control air pressure were less spectacular, then reactor scram would occur first, with declaration of an UNUSUAL EVENT due to "conditions warranting increased awareness of plant operating staff" occurring soon thereafter.

Failure of the Unit 1 HPCI and RCIC systems to start upon demand and failure of the immediate operator actions taken in attempts to restore these systems would warrant escalation of the plant status to ALERT. This would occur no later than 20 min after the scram.

At about 60 min after the scram, the drywell cooler outlet dampers would fail closed as a result of the decreasing drywell control air pressure. Without cooling, the temperature and pressure of the relatively small drywell atmosphere would rapidly increase, causing pressure and temperature alarms. This would result in declaration of a GENERAL EMERGENCY on the basis of "major internal or external events which could cause massive common damage to plant systems." Allowing for reluctance and time for a general reassessment of the plant status, GENERAL EMERGENCY should be declared no later than 3 h after the reactor scram.

The status of GENERAL EMERGENCY would remain in effect at least until drywell control air pressure could be regained at Units 1 and 2 and all three units could be brought to cold shutdown.

6. DISCUSSION OF UNCERTAINTIES

Most of the uncertainties discussed in this chapter affect only the details of accident sequence development. Nevertheless, uncertainties that concern the fundamental operability or flow capability of the control rod drive (CRD) hydraulic system have the potential to affect the major conclusions of the study.

6.1 Uncertainties in the Calculation Model

The BWR-LTAS code has been used to determine accident sequence events prior to the onset of severe fuel damage in this and all previous ORNL SASA program studies. Comparisons between BWR-LTAS results and those of other computer codes applied to the same event sequences have always been satisfactory (see Chapter 9 of Ref. 6.1).

The most important model uncertainty for the loss of control air study concerns the calculation of the rate of CRD hydraulic system injection into the reactor vessel. The model employed is fairly simple (Ref. 6.1), but is capable of accurately predicting the flow into the reactor vessel if supplied with accurate input data concerning the operation of the system. The piping arrangement of the CRD hydraulic system and the possible valve lineups that determine its flow capacity are described in Chapter 3 of Ref. 6.2.

The most important input data for the CRD hydraulic system is information concerning the head vs. capacity curve for the system pumps. The current input data is taken from the pump manufacturer's design curve, supplied with the pump more than 10 years ago when Browns Ferry Unit 1 was under construction. If pump performance has degraded over the years, then the ability of the CRD hydraulic system to perform as a standby high-pressure injection system would also be reduced. During normal reactor operation, each pump supplies 60 gpm (0.0038 m^3/s) control blade cooling flow, 20 gpm (0.0013 m^3/s) flow back to suction, and 8 gpm (0.0005 m^3/s) of cooling flow for the recirculation pump seals. The flow into the reactor vessel is increased automatically following scram, as discussed in Ref. 6.3, but there is no requirement to periodically confirm the pump capability for higher flows.

Other code input also affects the calculated flow. If the actual hydraulic resistance of the pump test bypass flow path is significantly greater (an order of magnitude or more) than the resistance provided in code input, the calculated flow into the reactor vessel would be significantly lower. Since the numerical value of the flow resistance of the 2.5 in. (0.0635 m) diameter pump test bypass piping, fittings and valves can be estimated with reasonable accuracy, the operability of the valves themselves is the dominating uncertainty. The arrangement of these valves is indicated in Fig. 6.1.

If the motor-operated 85-50 valve in the piping that connects the CRD hydraulic system to the reactor vessel via feedwater line B will not open, then there would be no flow through the pump test bypass path. The remaining flow into the reactor vessel (through the control rod

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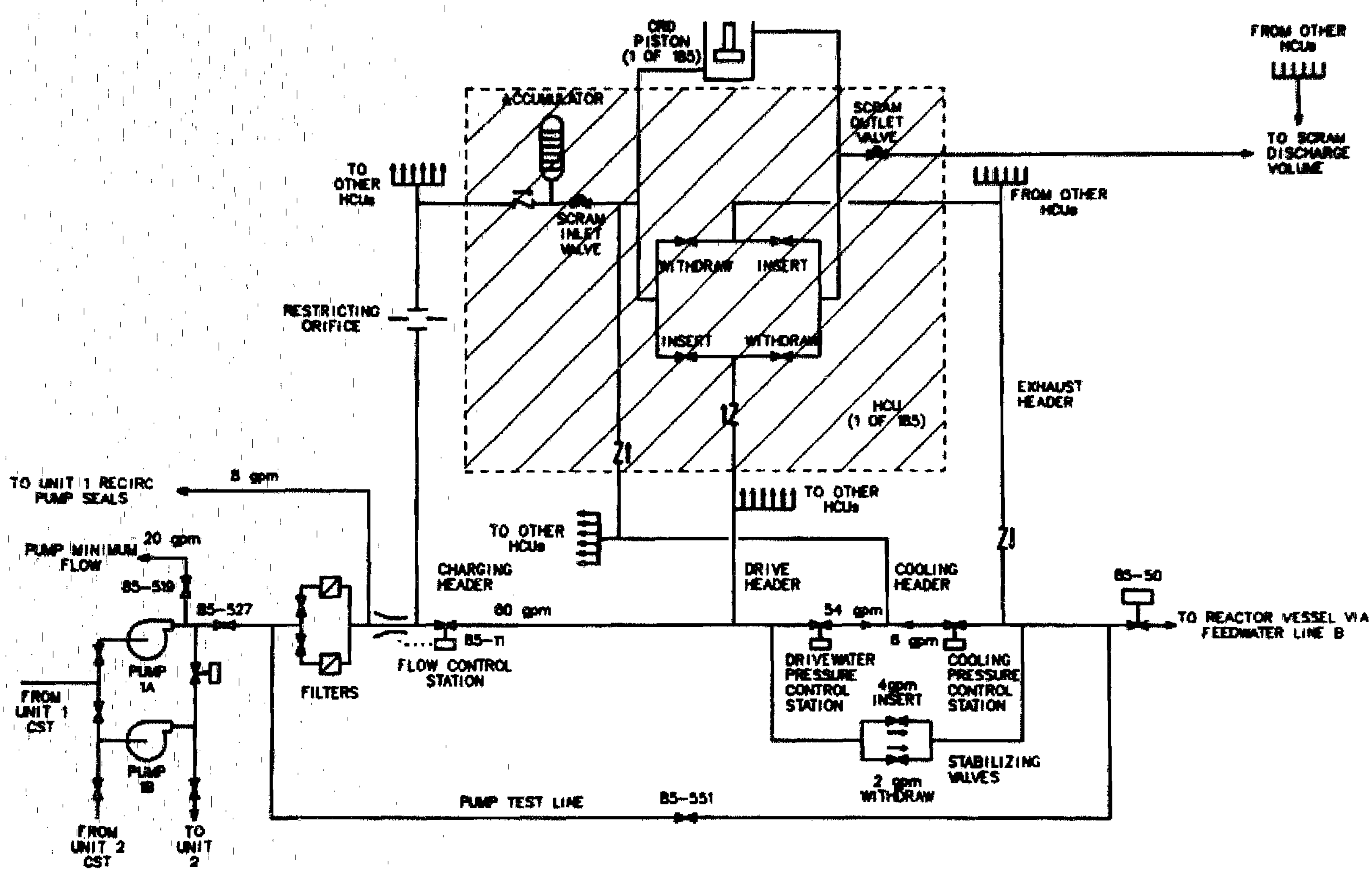

Fig. 6.1. Schematic diagram of the control rod drive hydraulic systems.

drive mechanism assemblies) with the reactor vessel pressurized cannot exceed about 225 gpm (0.0142 m^3/s), which would be achieved with both CRD hydraulic system pumps running and the pump discharge throttle valve 85-527 and the flow control valve 85-11 wide open.

The circuit breakers for the 85-50 valve motor operator are maintained in the open state during normal reactor operation, to preclude inadvertent opening of this valve from the Control Room. Furthermore, there is no requirement that the operability of components in the pump test bypass path be periodically verified. For these reasons, valve operability contributes to the uncertainty of the performance of the CRD hydraulic system as a backup to the steam turbine-driven high-pressure injection systems.

6.2 Uncertainties with Regard to Operator Actions

The operator actions assumed for the analyses presented in Chapter 5 are only those specified by procedure or of such a nature that it is reasonable to suppose that they would occur to any of the well-trained operators involved in the operation of the Browns Ferry plant. As discussed in subsection 5.3.4, the declaration and escalation of emergency action levels from UNUSUAL EVENT to ALERT to GENERAL EMERGENCY would assure the presence of designated emergency response staff members. These personnel would be capable of formulating and evaluating the non-standard maneuvers necessary for recovery from the assumed equipment casualties.

Consider, for example, the need to enhance the CRD hydraulic system injection to the reactor vessel. For the case in which the reactor vessel remains pressurized (Sect. 5.2), it has been assumed that the operators would be able to effect the necessary flow enhancement within the first 30 min following the reactor scram. There is, however, some uncertainty that the on-shift personnel would actually be able to accomplish this. On the other hand, if the operators exercise the option of depressurizing the reactor vessel and utilizing one of the low-pressure injection systems for level control (Sect. 5.3), then the need to enhance the CRD hydraulic system flow does not arise until drywell control air pressure has been lost and the reactor vessel has repressurized, some 9 h after the reactor scram. By this time, a multitude of experienced operating and engineering personnel would, per the station Radiological Emergency Plan, be at hand to assist in the flow enhancement task.

One possible operator action that, if taken, would exacerbate the plant response during the Loss of Control Air accident sequence is exercise of the safety/relief valves (SRVs) in the remote-manual mode more often than necessary. Each manual SRV actuation expends a quantity of air, either from the valve's individual accumulator if one of the six valves associated with the Automatic Depressurization System (ADS) is operated, or from the drywell control air receivers if a non-ADS valve is operated. However, it is unlikely that the operators would waste the available stored air in such a fashion. The Browns Ferry Operating Instruction that addresses Loss of Drywell Control Air Pressure (OI 32, Sect. V.C and V.D) directs the operators to minimize SRV actuations "to conserve accumulated air."

6.3 Uncertainties in Assumed Timing of Equipment Failures

Perhaps the largest uncertainty involved in this accident sequence analysis is the mode of failure of the Plant Control Air System. Depending on the nature of the system failure, the resulting rate of decay of the stored air pressure might be either slow or rapid. Nevertheless, an inability to resolve this uncertainty is not of concern because once the system has failed, no other event of significance will occur until the system pressure has decayed from its normal operating range of 85-110 psig (0.689-0.862 MPa) to the 60 psig (0.517 MPa) setpoint for reactor scram on low plant control air pressure. It has been conservatively assumed in this analysis that the operators do not take action to reduce reactor power or to manually scram the reactor during the period of pressure decay, either because there is insufficient time or because there is a simple failure to act.

Another uncertainty in equipment response involves the timing and mode of failure of the drywell atmosphere coolers. The analyses of this report are based on the assumption that the drywell cooler blower outlet dampers, whose position is adjusted by drywell control air pressure under normal operating conditions, would fail closed 1 h after loss of the Drywell Control Air System. This assumption is based upon actual plant tests in which the time required for the trapped air to bleed from the damper solenoid valve back through a check valve into a depressurized drywell control air header was measured for three dampers, with a minimum time of 1 h (Ref. 6.4).

Loss of effectiveness of the drywell coolers leads to a rapid increase in drywell atmosphere temperature and a concomitant increase in primary containment pressure. A high drywell pressure signal at 2.45 psig (0.118 MPa) triggers many automatic reactor protection and containment isolation responses, including closure of the valves in Primary Containment Isolation System (PCIS) Groups 2, 6, and 8. This has particular impact on the progression of the Loss of Control Air accident sequence case discussed in Section 5.3, in which the operators take action to depressurize the reactor vessel and go into the shutdown cooling mode of operation of the RHR system as quickly as possible, because the shutdown cooling supply isolation valves are included in PCIS Group 2.

In actuality, the period of time between closure of the drywell air compressor suction valves and closure of the drywell cooler outlet dampers might be much greater than 1 h, depending on the rate of decay of the pressure in the drywell control air system supply headers. Therefore, the approach taken here is believed to be conservative.

The equipment failure most crucial to this effort to establish the timing of events in the Loss of Control Air accident sequence is the failure of the remote-manual mode of SRV actuation. The air pressure available for remote-manual operation of the SRVs must exceed drywell pressure by 25 psi (0.172 MPa) if a valve is to be opened or if an already-open valve is to be held open. As previously discussed, the six valves associated with the ADS system have individual accumulators protected by check valves from the general drywell control air headers.

The analysis described in Section A.2 of Appendix A provides a basis for assertion that the SRVs associated with the ADS system would remain available for remote-manual actuation for at least 6 h after the reactor scram. The calculation is based upon the maximum allowable ADS accumulator leak rate of 10 psi/h (0.069 MPa/h). The Browns Ferry Technical Specifications require that the leak rate of each ADS accumulator be measured once per operating cycle. The leakage measurement method is to suddenly remove air pressure upstream of the check valves that isolate each ADS accumulator from the drywell control air header, then to record the accumulator pressure as it slowly decays over the next several hours. If the measured leakage exceeds the allowable rate, repairs are required.

The prescribed method for surveillance that the ADS accumulators will perform their intended mission is unquestionably valid for cases in which drywell control air pressure is suddenly lost. Nevertheless, the subject of this chapter is uncertainties, and it can be argued that the sudden removal of drywell control air pressure upstream of the accumulator isolation check valves during tests results in much better seating of the check valves than would occur if the drywell control air pressure were bled down very slowly. As a matter of fact, the Loss of Control Air accident sequence examined in this study involves a very slow rate of loss of drywell control air pressure.

Nevertheless, even if the ADS accumulators should be totally ineffective as a result of isolation check valve leakage, it is estimated that a safety/relief valve could be held open for at least 3 h after reactor scram, and this would be sufficient to maintain the reactor vessel depressurized. It should be recalled that the drywell control air system receivers provide a large source of stored air for use after compressor failure; the overall system would be expected to lose pressure at the rate of 26 psi/h (0.179 MPa/h) as explained in Section 2.2.

References for Chapter 6

6.1 R. M. Harrington and L. C. Fuller, "BWR-LTAS: A Boiling Water Reactor Long-Term Accident Simulation Code," NUREG/CR-3764, ORNL/TM-9163, February 1985.

6.2 R. M. Harrington and L. J. Ott, "The Effect of Small-Capacity, High-Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One," NUREG/CR-3179, September 1983.

6.3 R. M. Harrington, et al., "SBLOCA Outside Containment at Browns Ferry Unit One — Accident Sequence Analysis," NUREG/CR-2672, ORNL/TM-8119/V1, November 1982, Appendix E.

6.4 Personal communication, R. A. Bollinger, TVA, to S. A. Hodge, 1984.

7. IMPLICATIONS OF RESULTS

The purpose of this chapter is to provide a discussion of the present state of readiness at the Browns Ferry Nuclear Plant to cope with a loss of the Plant Control Air System. Loss of plant control air pressure would cause reactor scram and closure of the outboard main steam isolation valves at all three units.

7.1 Control Room Instruments

The control room operators cannot appreciate the rate of injection achieved by the control rod drive (CRD) hydraulic system after reactor scram because the control room panel instrument range extends only to an indicated rate c. 100 gpm (0.0063 m^3/s). In actuality, the injection rate will be almost twice as high with a scram in effect and the reactor vessel depressurized, and still higher if the operator takes action to enhance the flow. Upgrade of the CRD hydraulic system control room indication range should be considered.

7.2 System Design

It is obvious that the susceptibility of the Browns Ferry Nuclear Plant to loss of the Plant Control Air System is due to the dependence of the Drywell Control Air Systems of Units 1 and 2 upon plant control air to hold open the suction valves to the drywell control air compressors. If drywell control air pressure is lost, ultimately the reactor vessel could not be maintained depressurized and the low-pressure injection systems would not be available for use to keep the core covered.

The present design at Unit 3 keeps the Drywell Control Air System of that unit independent from the Plant Control Air System by providing that the air compressor suction valves are operated by drywell control air instead of plant control air. Consideration should be given to extending this design improvement to Units 1 and 2.

The threat to reactor safety caused by loss of drywell control air pressure obtains from the consequent inability to operate the safety/relief valves and maintain the reactor vessel depressurized. The TVA has voluntarily committed to the NRC to provide a safety-grade, long-term depressurization capability to the safety/relief valves associated with the Automatic Depressurization system by installing supply lines from the nitrogen supply trains of the Containment Atmosphere Dilution (CAD) system. This improvement should reduce the probability to an insignificant level that this severe accident sequence, or any other sequence involving loss of the Drywell Control Air System, will lead to an inability to depressurize the reactor vessel.

7.3 Operator Preparedness

Control room operator training should be expanded to include an explanation of the Control Rod Drive Hydraulic System resources available in an emergency. The flow from this system can be augmented to the extent necessary to avoid a severe accident if the proper steps are taken to start a second CRD hydraulic system pump, open the discharge throttle valve fully, and so forth. The potential of this system is discussed in detail in NUREG/CR-3179.

The motor-operated RHR system shutdown cooling valves from the reactor vessel will automatically shut upon low reactor vessel water level or high drywell pressure [at 2.5 psig (0.115 MPa)]. The feasibility of overriding the high drywell pressure interlock should be evaluated, and instructions for execution of this mitigation strategy should be included in emergency procedures.

Appendix A. MODIFICATIONS TO THE BWR-LTAS CODE FOR THIS STUDY

Several new capabilities were added for the Loss of Control Air study. Analysis of the no-operator-action case (Chapter 4) required the capability to model the discharge of water through the SRVs. The depressurized operator action case (Sect. 5.3) required a model for heat removal from the primary coolant via the shutdown cooling mode of the RHR system. The programming necessary to effect the failure of remote-manual operation of the SRVs was necessary for all cases with depressurization. Although not specifically needed for this study, an expanded simulation of the primary containment heat sinks was employed. Each of these items is discussed below.

A.1 Modeling of Water Discharge Through Safety Relief Valves

Equations were added to accurately calculate the flow of water through the SRVs in the event of flooding of the main steam lines. Additionally, minor modifications were necessary to interface the calculated rate of water discharge with the rest of the existing simulation.

The flow of water through an SRV is calculated by the expression

$$B = C_v / [\rho / (\rho_0 \ \Delta P)]^{1/2}$$

where,

B = the bulk (volumetric) flow of liquid through the open SRV,
C_v = discharge coefficient for flow of water through an SRV,
ρ = the density of water at the SRV inlet,
ΔP = the pressure difference across the SRV, and
ρ_0 = the density of water at 60°F (15.5°C).

This expression is valid for the discharge of subcooled water provided that the valve discharge coefficient, C_v, is known. Fortunately, experimental data recorded under actual water discharge conditions are available (Ref. A.1) and have been used to validate the predicted flow. The assumption of subcooling is reasonable because a vessel overfill incident is most likely to occur when the injection rate of cool water is greatly in excess of the capacity of decay heating to bring to saturation and boil the injected water.

The model is programmed to begin water discharge when the vessel is flooded to above the level of the vessel steam outlet nozzles and sufficient water has overflowed from the vessel to entirely fill the main steam lines inside containment (note: the inboard MSIVs are closed). Steam discharge is allowed to resume when and if the vessel water level subsides to below the level of the steam outlet nozzles. Logic was also added to permanently fail the steam turbine-driven HPCI and RCIC systems after the onset of water discharge.

The rising water level that floods the main steam lines cuts off the steam-filled region at the top of the reactor vessel from communication with the SRVs. The steam region continues to receive steam produced in the core (if any) and to be depleted by condensation on the walls of the upper reactor vessel and on the water surface.

Calculations indicate that the steam region shrinks rapidly in an overfill event because of condensation onto the surface of the subcooled water region. However, a flag is included in the model to halt this condensation process before the steam region is entirely consumed. As discussed in Section 3.2 of Ref. A.2, the method used to calculate the reactor vessel pressure depends upon the existence of a finite volume of steam in the steam region in order to solve for the pressure in the reactor vessel.

A.2 Safety Relief Valve Dependence on Control Air Pressure

The purpose of this subsection is to provide the basis for the assumption of a 6 h period of availability for remote-manual actuation of the ADS SRVs after the loss of the drywell control air compressors. The no-operator-action cases of Chapter 4 and the second operator-action case of Chapter 5 both utilize this assumption. In the discussion below, operation of the two-stage Target Rock SRV is outlined briefly, and then the calculation of the 6-h availability period is presented.

The opening of a two-stage Target Rock SRV is a two step process that is initiated by the opening of the pilot stage of the valve. The repositioning of the pilot valve opens a path through which the above-piston volume of the main disc, previously at upstream (reactor vessel) pressure, is vented to the downstream (suppression pool) side. The resulting differential pressure across the main disc piston overcomes the spring and pressure forces that tend to keep the main disc seated and opens the main stage, allowing steam to flow from the reactor vessel through the main stage to the wetwell. The reactor vessel pressure must be at least 50 psi (0.345 MPa) above the wetwell pressure in order for the main stage to open.

The pilot stage can be pushed off its seat and repositioned by the reactor vessel steam pressure, or it can be pulled off its seat by the action of the control air diaphragm on the pilot stem, or it can be repositioned by a combination of control-air-generated force and reactor vessel steam pressure. For the lowest-set bank of four SRVs, a reactor vessel pressure of 1105 psi (7.62 MPa) above suppression pool pressure will automatically open the pilot stage (and hence the main stage) without any additional force from the control air diaphragm. A net control air pressure of 25 psi (0.172 MPa) above drywell pressure is required to open the pilot stage if the reactor vessel is completely depressurized. The control air pressure necessary for remote-manual opening varies linearly between these two extremes for intermediate reactor vessel pressures. The control air pressure necessary to hold the pilot stage open after its initial unseating is comparable to that required for the initial opening.

For the calculations presented below, it is assumed that the reactor vessel has been depressurized to 100 psia (0.689 MPa), so that a control air pressure of 23 psi (0.159 MPa) above drywell pressure is required for the ADS SRVs to remain open. For the no-operator-action case of Chapter 4, the SRVs are signaled to open and remain open by actuation of the ADS about 1 h after the reactor scram; for the second operator-action case of Chapter 5 the SRVs are signaled to open and remain open soon after the reactor scram by operator actuation of the remote-manual SRV opening switches in the main control room.

For the ADS SRVs, adequate control air pressure is assured by individual accumulators attached to each valve's air supply line. When drywell control air pressure falls below the accumulator pressure, check valves seat and hold captive the air within the accumulators. During normal operation, the accumulators are kept charged as the drywell air compressors cycle on and off between 101.5 and 114.5 psia (0.696 and 0.789 MPa) (see Sect. 2.2). The ADS accumulators are more leaktight than the rest of the system, so their pressure will remain above about 109.5 psia (0.755 MPa) during normal operation. During a long period without drywell control air pressure, the accumulators would lose air pressure by two mechanisms: leakage and valve actuation. The leakage may be taken to be 10 psi/h (69 kPa/h) on the basis of the surveillance and maintenance requirements of the Browns Ferry Technical Specifications. This leak rate is assumed to remain constant during the first 2 h and then to decrease in proportion to the decreasing accumulator pressure.

The volume of air expended for each SRV actuation is determined primarily by the volume of the control air tubing and the volume of air internal to the SRV actuator mechanisms. Air within each accumulator expands by the same fraction for each actuation; hence, pressure should decrease by a characteristic fraction as a result of each remote-manual actuation. Although without precise geometric information, we were able to infer a fractional pressure decrease of 0.12 per actuation from the basic design criterion that was used to determine the required stored air volume of the ADS accumulators:

> Stored air in each accumulator must be sufficient for five actuations, the first and second actuations with a drywell pressure of 35 psig and the remaining three actuations with a drywell pressure of 0 psig.

The 6 h availability time for remote actuation of the ADS SRVs may be calculated from the 0.12 pressure reduction fraction, the 10 psi/h (69 kPa/h) leak rate, and from the following facts about the sequence:

(1) initial ADS accumulator pressure is 109.5 psia (0.755 MPa);
(2) the ADS SRVs are opened after about 1 h and left open;
(3) the reactor pressure is nominally at 100 psia (0.69 MPa) between 1 and 6 h; and

(4) drywell pressure will increase to about 20 psia (138 kPa) by 6 h, so the SRVs will close when accumulator pressure decreases to below 43 psia (296 kPa).

The calculation proceeds in stages, as follows:

P = 109.5 psia (0.755 MPa) at time zero,
P = 99.5 psia (0.686 MPa) after 1 h (but before actuation),
P = 87.6 psia (0.60 MPa) after 1 h (but after actuation),
P = 77.6 psia (0.54 MPa) after 2 h,
P = 68.6 psia (0.47 MPa) after 3 h,
P = 60.6 psia (0.42 MPa) after 4 h,
P = 52.6 psia (0.36 MPa) after 5 h,
P = 45.6 psia (0.31 MPa) after 6 h, and
P = 43 psia (0.30 MPa) after 6.37 h,

where P = the air pressure inside the ADS accumulators. For the BWR-LTAS calculations of this report this figure was rounded to 6 h.

A.3 Shutdown Cooling

The shutdown cooling mode of the RHR system plays a small part in the depressurized loss of control air sequence of Sect. 5.3. The performance of an RHR system heat exchanger in the shutdown cooling mode should be very similar to the performance that could be expected in the pool cooling mode. In either case, the RHR Service Water flow would be the same. The flow of primary coolant (or of suppression pool water for the pool cooling mode) would, if anything, be greater for shutdown cooling than it would be for the pool cooling mode. Therefore it was decided that the heat exchanger formulation developed previously for pressure suppression pool cooling would be applicable.

The following expression for the heat transferred in one RHR system heat exchanger was used to represent shutdown cooling:

$$Q_{sdc} = E\, W_{sw}\, c_p\, (T_{rc} - T_{sw}) \ .$$

where,

Q_{sdc} = heat transferred per heat exchanger from the reactor coolant to the river water,
E = heat exchanger efficiency factor = 0.375,
W_{sw} = service water flow per heat exchanger,
c_p = specific heat of water,
T_{rc} = reactor coolant temperature at heat exchanger inlet, and
T_{sw} = service water temperature at heat exchanger inlet.

A.4 Containment Heat Sinks

Several previously unsimulated heat sinks were added prior to performing the calculations for this report, including the drywell atmosphere miscellaneous metallic heat sinks, and the wetwell atmosphere miscellaneous metallic heat sinks. A single average temperature is calculated for each heat sink. The heat exchange between each of the heat sinks and the atmosphere that surrounds it is treated exactly the same as the heat exchange between the steel drywell liner and the drywell atmosphere (or between the steel wall of the suppression pool torus and the wetwell atmosphere).

Metallic heat sinks can have a significant effect on the heatup rate of the atmosphere because of their characteristic low internal heat transfer resistance and because their heat capacity is large compared to the heat capacity of the atmosphere.

The BWR-LTAS input description of the drywell miscellaneous metallic heat sinks is a mass of 240,000 lbs (109,000 kg) with a total surface area of 8617 ft^2 (800 m^2) and the input for the wetwell heat sinks is a mass of 629,000 lbs (285,909 kg) with a surface area of 22542 ft^2 (2094 m^2). These values are based on information in the TVA report "Integrated Leak Rate Test of the Reactor Containment Building," page 105, reporting miscellaneous steel volumes in the drywell and wetwell of, respectively, 493.7 ft^3 (13.98 m^3) and 2583 ft^3 (73.13 m^3). The estimate was made that approximately half of the wetwell miscellaneous metal mass would be above the water level of the suppression pool and therefore exposed to the wetwell atmosphere. The input for the surface areas is calculated on the assumption that the typical miscellaneous metallic heat sink can be represented in slab geometry, with thickness of 1.375 in. (3.49 cm), and exposed to the atmosphere on both sides.

References for Appendix A

A.1 "Analysis of Generic BWR Safety/Relief Valve Operability Test Results," NEDE-24988-P, Class III, October 1981.

A.2 R. M. Harrington and L. C. Fuller, "BWR-LTAS: A Boiling Water Reactor Long-Term Accident Simulation Code," NUREG/CR-3764, ORNL/TM-9663, February 1985.

Appendix B. MAXIMUM PRESSURE SUPPRESSION POOL TEMPERATURE WITH ONLY ONE RESIDUAL HEAT REMOVAL SYSTEM HEAT EXCHANGER IN THE POOL COOLING MODE

The Browns Ferry Unit 1 RHR system comprises four heat exchangers and four pumps, one pump associated with each heat exchanger. In many accident sequences, some number of the full complement of heat exchangers must be operated in the pool cooling mode in order to maintain a safe suppression pool temperature during the hours following a reactor scram. The question of how many of the four heat exchangers must be available for suppression pool cooling is of obvious safety significance. A peak suppression pool temperature of 200°F (367 K) or below is generally considered safe for the condensation of SRV discharge within the water of the pool. Adequate net positive suction head (NPSH) can safely be assumed at this temperature, especially considering the back-pressure expected in the primary containment during a long accident sequence with the MSIVs closed.

Two BWR-LTAS runs were made to find the peak suppression pool temperature in the event that only one RHR heat exchanger is available for pool cooling. Both runs start after reactor scram from full power and closure of the MSIVs. The suppression pool temperature at the beginning of each run (30 s after the scram from full power) was taken to be 94°F (308 K), which is consistent with a pre-scram pool temperature of 90°F (305.6 K).

For Run #1, the reactor vessel remains at power with the SRVs intermittently opening to discharge the decay heat-produced steam to the suppression pool; vessel injection is provided from the Condensate Storage Tank (CST) by the RCIC system. Run #2 starts identically, but at 10 min, four SRVs are opened to depressurize the reactor vessel, and the reactor vessel remains at low pressure [~100 psia (0.69 MPa)] for the rest of the run. After depressurization, vessel injection is provided from the suppression pool by intermittent operation of a core spray pump.

A peak suppression pool temperature of 182°F (357 K) is predicted by Run #1 (12 h after scram). The peak of 185°F (358.3 K) in Run #2 occurs 15 h after the scram. The higher suppression pool temperature for Run #2 is due to the extra energy transferred to the pool by the reactor vessel depressurization, and to the lower suppression pool mass maintained throughout the run. Pool mass is constantly increasing in Run #1 because vessel injection is from the CST in an open cycle, instead of being recycled from the suppression pool. These two BWR-LTAS runs, taken together, suggest that one RHR heat exchanger is sufficient, but higher pool temperatures might be possible. For example, what if the reactor vessel is depressurized after 10 h instead of after 10 min?

A closed form analytical solution (described in Appendix C) was developed to aid in the parameterization to find the maximum possible suppression pool temperature. Besides saving computer time, the analytical solution has the advantage of eliminating any question about the accuracy of the BWR-LTAS numerical solution. The long term pool temperature response can be simplified into a problem amenable to analytical solution because of the relatively simple nature of the problem.

Except during the relatively brief periods during and immediately following reactor scram and reactor vessel depressurization, the reactor vessel is in a quasi-steady state, discharging to the suppression pool the amount of steam dictated by the slowly decreasing decay heat level. The suppression pool temperature slowly increases as it becomes the recipient of the integrated decay heat production; heat removal from the pool is directly proportional to the increase of suppression pool temperature above the temperature of the river water running on the tube-side of the RHR system heat exchangers.

Significant input numbers for the analytical model include the following:

1. $Q_{pool\ cooling} = 234\ (T_{psp} - 90)$, where $Q_{pool\ cooling}$ is the heat (in Btu/s) removed by one RHR system heat exchanger;
2. suppression pool temperature increase due to depressurization of the reactor vessel is 21°F (11.7 K);
3. suppression pool temperature = 94°F (308 K) at 30 s after reactor scram [consistent with 90°F (305.6 K) pre-scram temperature];
4. service water temperature = 90°F (305.6 K);
5. the single running RHR pump adds 1.5 MW of heat to the suppression pool; and
6. a constant value of 0.37 MW was used to approximate the heat losses from the pool due to radiant and convective heat transfer from the outer surface of the torus.

The results of the calculations utilizing the analytical model are summarized in Table B.1. The peak suppression pool temperature is 201°F

Table B.1. Peak suppression pool temperature as a function of reactor vessel depressurization time

Reactor vessel depressurization time (h)	Peak suppression pool temperature [°F (K)]	Time of peak pool temperature (h)
0.333	184.5 (358.1)	13
1	184.9 (358.3)	13
2	185.6 (358.7)	12.5
3	186.4 (359.1)	12.25
4	187.3 (359.6)	12
5	188.3 (369.2)	11.5
6	189.6 (360.9)	11
8	192.7 (362.6)	9.5
10	196.7 (364.8)	10
12	199.3 (366.3)	12
14	200.5 (366.9)	14
16	201 (367.2)	16
20	200.5 (366.9)	20

(367.2 K), and is the result of delaying 16 h to depressurize the reactor vessel. The 184.5 (358.1 K) peak predicted for the 20 min depressurization time can be regarded as a confirmation of the 185°F (358.3 K) peak temperature predicted by the BWR-LTAS run with depressurization at 10 min after the scram. Overall, the conclusion that one RHR heat exchanger can provide adequate pool cooling is supported by the calculated results.

Appendix C. DEVELOPMENT OF ANALYTICAL MODEL FOR PRESSURE SUPPRESSION POOL (PSP) HEATUP

The analytical model developed in the following paragraphs applies under the following circumstances:

(1) Both the reactor vessel and the PSP are in a quasi-steady state mode with respect to mass inventories. The flow of water from the PSP to the reactor vessel is assumed to be exactly that required to counterbalance the steaming from the reactor vessel to the PSP; therefore PSP mass is constant during pool heatup.
(2) The PSP is receiving decay heat generated steam discharged from the reactor vessel (RV) between 0 and 24 h after scram from full power (note: the decay heat correlation will be inaccurate for times longer than 24 h).
(3) Depressurization is allowed, but it must be rapid (the model assumes instantaneous).
(4) Pool cooling is given by $Q_c = E_{cmin} (T_p - T_{sw})$ where T_p = PSP temperature, T_{sw} = service water temperature, and E_{cmin} is the cooler efficiency times the smaller of the flow-specific heat product of the hot (PSP) or cold (service water) sides of the cooler(s).

The energy and mass balances for the PSP are:

$$\frac{d}{dt}(h_p M_p) = W_s h_s - Q_c - W_f h_p$$

$$\frac{d}{dt}(M_p) = W_s - W_f$$

where,

M_p = total pool mass
h_s = steam enthalpy
W_s = SRV steam discharge
W_f = flow taken from PSP for RV feed
h_p = enthalpy of pool water.

If we substitute the mass balance into the energy balance and assume that the specific heat of water is constant at 1.0, these equations reduce to:

$$Mp \frac{dT_p}{dt} = W_s (h_s - h_p) - Q_c$$

If the reactor vessel is at steady-state, with injection from the PSP, then we can state that

$$Q_{dh} = W_s\ (h_s - h_p)$$

where

Q_{dh} = decay heat generated in the RV.

so the equation for heatup of the PSP becomes:

$$M_p \frac{dT_p}{dt} = Q_{dh} - E_{cmin}\ (T_p - T_{sw})\ .$$

For later convenience in the solution, the pool temperature is referenced to service water temperature

$$\frac{dT_p}{dt} = \frac{d}{dt}\ (T_p - T_{sw}) = \frac{dT_p^*}{dt} = [Q_{dh} - E_{cmin}\ (T_p^*)]/M_p$$

where $T_p^* = T_p - T_{sw}$.

Since we have assumed M_p = constant, we can immediately state two easy solutions to special cases of this equation:

$$T_p^* = T_{p0}^* + \frac{1}{M_p} \int_o^t Q_{dh}\ dt$$ ..... the case with no PSP cooling

$$T_p^* = T_{p0}^* \exp\ [-t(E_{cmin}/M_p)]$$ ..... The case with PSP cooling but no steam discharge (i.e. no heating)

In general, however, it is necessary to solve the inhomogenious, first order differential equation for T_p^*. The ANS standard expression for decay heat (relative to initial power level), not including actinides or activation products, is: $(P/Po) = 0.13/t^{0.283}$, valid for times after scram between 150 s and $4(10)^6$ s. Since the author could not find an analytical solution involving this expression, it was necessary to recast the decay heat in terms of polynomials and sums of simple exponentials:

$$(P/P_o) = C_0 + C_1 t + \sum_{j=1}^{11} E_j \exp(-\lambda_j t)$$

The coefficients E_j and λ_j in the exponential sum were taken from Table V.8-1 of the RETRAN manual and c_0 and c_1 were calculated by means of a least squares fit that minimized the difference between the total decay heat including actinides and activation products as specified by the ANS standard for the first 24 h following scram.

The equation for pool heatup, as demonstrated above, is now of the form

$$\frac{d}{dt} - a \quad y = c_0 + c_1 \ t + \sum_{j=1}^{11} E_j \ e^{-\lambda_j t} \ ,$$

and the handbook solution for this equation is

$$y = A_o e^{at} - (c_0 + c_1 \ t + c_1/a)/a - \sum_{j=1}^{11} \frac{E_j \ \exp(-\lambda_j t)}{(\lambda_j + a)}$$

where,

$y = T_p^*$
A_o = constant of integration (see below)
$a = - E_{cmin}/M_p$

and the previously defined constants c_0, c_1, E_j have been multiplied by the total initial thermal power. To find the constant of integration, A_o, it is only necessary that pool temperature be known at some initial time, t_o:

$$A_o e^{at_o} = y_o + (c_0 + c_1 \ t_0 + c_1/a)/a + \sum_{j=1}^{11} \frac{E_j \ \exp(-\lambda_j t_o)}{(\lambda_j + a)}$$

The short computer code utilized to evaluate the expressions for pool temperature response works in three segments: pool heatup prior to the start of PSP cooling, pool heatup after start of PSP cooling, but before depressurization, and pool heatup after depressurization (pool cooling assumed running). The first segment is just the simple integration of decay heat, as noted previously. The second segment utilizes the handbook solution developed above, with initial conditions taken at the end of the first segment. Depressurization is handled as a discontinuity. The predetermined PSP temperature increase due to depressurization is added to the temperature calculated at the end of the second segment, just prior to depressurization. The resulting temperature becomes the initial condition for evaluation of the constant of integration for the third segment.

Appendix D. KEEPING THE BWR CORE COVERED IN NON-LOCA ACCIDENT SITUATIONS

The reactor vessel injection rate required to replace the water mass converted to steam by decay heat is shown in Fig. D.1. As expected, the curve representing the required injection rate falls off rapidly during the first hour following scram and resembles the decay heat curve. The relation between the required injection flow and the time after scram was calculated assuming constant reactor vessel pressure, an initial power of 100%, decay heat according to the 1979 ANS standard with actinide decay, a coolant injection temperature of 90°F (305 K), and that the steam leaving the reactor vessel via the safety/relief valves is in the dry saturated state.

As noted above, injection according to Fig. D.1 would result in a constant reactor vessel water level [which would be the water level existing at the time of scram, about 561 in. (14.25 m) above vessel zero]. It is also important to an understanding of BWR accident sequences to know the single continuous rate of injection that would prevent core uncovery. Information concerning this calculation for Browns Ferry is provided in a previous ORNL SASA program report (Ref. D.1). Assuming scram from 100% power, a continuous injection rate of 225 gpm (0.014 m^3/s) would result in a continuous level decrease for about 100 min, with the water level just above the top of the core at the end of this period. Subsequently, the water level would slowly but monotonically increase.

The capacities of the available ECCS injection system pumps are each much greater than 225 gpm (0.014 m^3/s). As indicated on Fig. D.2, each of the four available RHR system pumps has a capacity of 10,000 gpm (0.622 m^3/s), the steam turbine-driven HPCI system pump is capable of 5000 gpm (0.311 m^3/s), and each of the four core spray system pumps has a capacity of 3125 gpm (0.194 m^3/s). The steam turbine-driven RCIC system pump, which is not part of the ECCS protection design, also has a capacity [600 gpm (0.037 m^3/s)] that is greater than that necessary to prevent core uncovery in accident sequences other than LOCA.

Some of the postulated BWR severe accident sequences other than LOCA involve loss of ability to depressurize the reactor vessel or to maintain it at low pressure if previous depressurization was successful. The low-pressure injection systems (RHR and core spray) cannot be used unless the reactor vessel is depressurized. This leaves the high-pressure injection systems HPCI and RCIC, which have a calculated rate of combined failure-upon-demand of 1.5%. What happens if the reactor vessel is not depressurized and the HPCI and RCIC systems both fail upon demand, and cannot subsequently be restored? There remains a system that can be utilized to preclude significant core damage.

The last-ditch injection system is the control rod drive (CRD) hydraulic system that injects cooling water for the control blades during normal reactor operation. As explained in Ref. D.1, this system employs electric motor-driven pumps that inject 60 gpm (0.004 m^3/s) through flow-limiting orifices normally, but these orifices are automatically bypassed after scram. Thus, post-scram injection by this

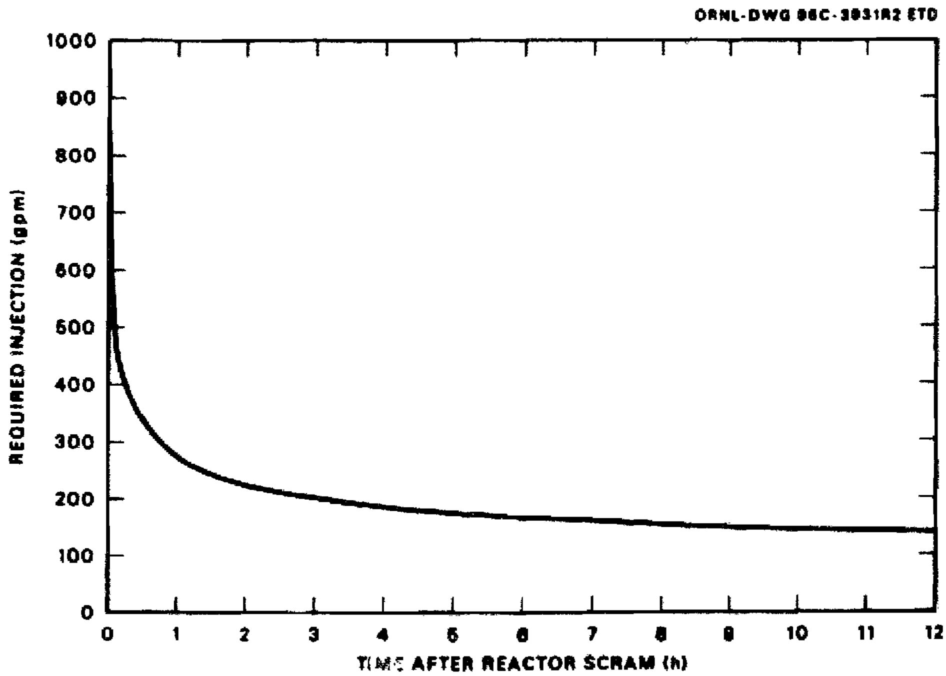

Fig. D.1. The required rate of reactor vessel injection to maintain a constant reactor vessel water level after scram at a Browns Ferry unit.

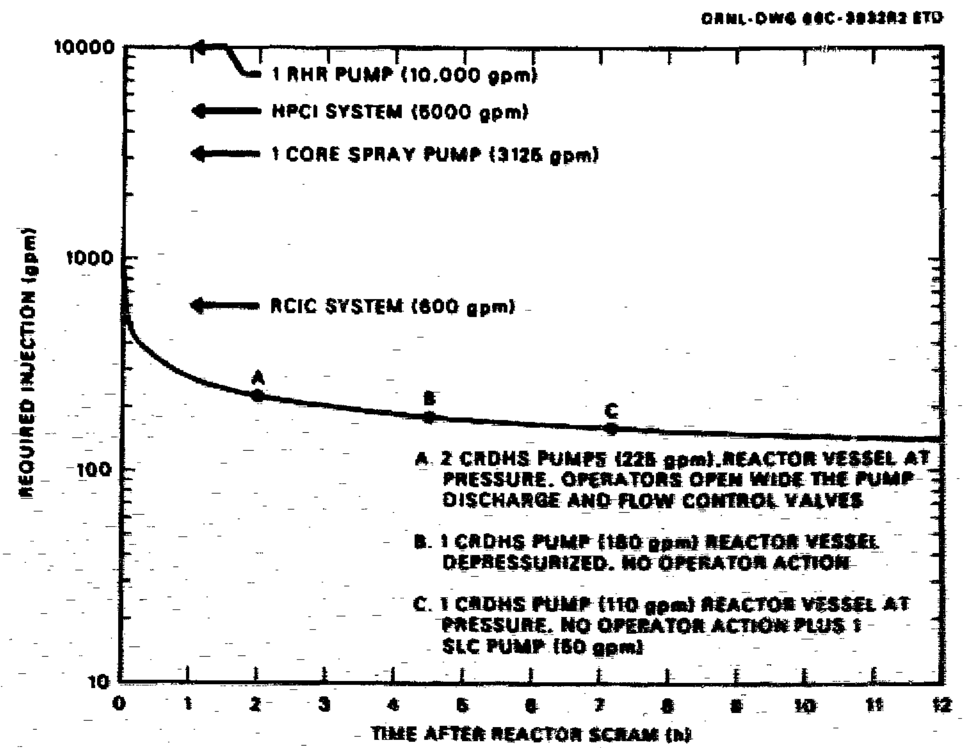

Fig. D.2. Reactor vessel injection system capacities compared to the injection rate required to maintain a constant reactor vessel water level after scram at Browns Ferry.

system automatically rises to about 112 gpm (0.007 m^3/s) with the reactor vessel remaining pressurized and about 180 gpm (0.011 m^3/s) if the vessel is depressurized. In addition, there is the potential for the operators to take action to increase the rate of injection by this system. The effects of three specific cases of operator action are displayed on Fig. D.2.

It is important to remember that the CRD hydraulic system is not a safety system and that its potential for injection under accident situations is plant-specific. This system can be effective at Browns Ferry, as proven by its important role during the actual accident sequence of the fire in March, 1985.

References for Appendix D

D.1 R. M. Harrington and L. J. Ott, "The Effect of Small-Capacity, High-Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One," NUREG/CR-3179, September 1983, Chapter 4.

Appendix E: LIST OF ACRONYMS

ADS	Automatic Depressurization System
ANS	American Nuclear Society
BWR	Boiling Water Reactor
CAD	Containment Atmosphere Dilution
CRD	Control Rod Drive
CS	Core Spray
CST	Condensate Storage Tank
ECCS	Emergency Core Cooling System
EECW	Emergency Equipment Cooling Water
EPG	Emergency Procedure Guidelines
FSAR	Final Safety Analysis Report
HCU	Hydraulic Containment Unit
HPCI	High Pressure Coolant Injection
MSIV	Main Steam Isolation Valve
NPSH	Net Positive Suction Head
ORNL	Oak Ridge National Laboratory
PCIS	Primary Containment Isolation System
PCV	Pressurized Containment Valves
PS	Pressure Switches
PSP	Pressure Suppression Pool
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RBCCW	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RCW	Raw Cooling Water
RHR	Residual Heat Removal
RPS	Reactor Protection System
RSW	Reactor Service Water
RV	Reactor Vessel
RWCU	Reactor Water Cleanup
SASA	Severe Accident Sequence Analysis
SDIV	Scram Discharge Instrument Volume
SDV	Scram Discharge Volume
SLC	Standby Liquid Control
SRV	Safety Relief Valve
TAF	Top of the Active Fuel
TIP	Transversing Interior Probe
TVA	Tennessee Valley Authority

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This study describes the predicted response of the Browns Ferry Nuclear Plant to a postulated complete failure of plant control air. The failure of plant control air cascades to include the loss of drywell control air at Units 1 and 2. Nevertheless, this is a benign accident unless compounded by simultaneous failures in the turbine-driven high pressure injection systems. Accident sequence calculations are presented for Loss of Control Air sequences with assumed failure upon demand of the Reactor Core Isolation Cooling (RCIC) and the High Pressure Coolant Injection (HPCI) systems at Unit 1. Sequences with and without operator action are considered. Results show that the operators can prevent core uncovery if they take action to utilize the Control Rod Drive Hydraulic System as a backup high pressure injection system.

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