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BWR 4/Mark I Accident Sequences Assessment

D. D. Yue
T. E. Cole

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

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CONTENTS

FOREWORD

The Severe Accident Sequence Analysis (SASA) program was established by the Nuclear Regulatory Commission in October 1980 for the purpose of studying potential nuclear power plant accidents beyond the design basis. Under the auspices of the program, boiling water reactor (BWR) studies have been conducted at Oak Ridge National Laboratory (ORNL) using Browns Ferry Unit One as the model plant with assistance and full cooperation from the plant owners and operators, the Tennessee Valley Authority.

The primary analytical tool for the events of each severe accident sequence that would occur after the reactor core has been uncovered is the MARCH code, originally developed by Battelle-Columbus Laboratories.* The MARCH code incorporates the principal meltdown computer models used in the Reactor Safety Study† and various improvements and modifications added thereafter. A recent MARCH code assessment,‡ performed primarily from the standpoint of the application of MARCH to pressurized water reactor (PWR) accident analysis, points out that

> "The code's development, its structure, level of detail, etc. reflect the limited goals of early risk assessments. Thus, for example, relatively simple and fast-running models were needed so that the many types and numbers of accident sequences could be investigated. Further, the uncertainties associated with using these simple models were not considered to be of major concern, in light of the large overall uncertainties present in risk assessment."

The MARCH thermal-hydraulic models are particularly crude; core flow is not modeled, and the reactor vessel is modeled only as a two-node cylindrical volume with water at the bottom and steam at the top. Consequently, it is common practice within the SASA program to circumvent the MARCH thermal-hydraulic models to the maximum extent possible. A complete severe accident analysis as conducted in the SASA program at ORNL involves use of a more detailed thermal-hydraulics code with respect to the primary and secondary coolant systems for the events before core uncovery, visits to the TVA Browns Ferry simulator to act out the accident scenario and assess the control room equipment and instrumentation, and discussions with TVA supervisory and plant operating personnel to determine the up-to-date status of plant systems and the probable operator actions.

In January 1981, soon after the SASA program was established, ORNL was requested to perform stand-alone MARCH code calculations for eight BWR accident sequences identified and briefly outlined by the SASA project at

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*R. O. Wooten and H. I. Avci, *March Code Description and User's Manual*, NUREG/CR-1711, Battelle Columbus Laboratories (1980).

†*Reactor Safety Study*, WASH-1400, NUREG-75/014. Washington, DC: U.S. Nuclear Regulatory Commission (1975).

‡J. B. Rivard et al., *Interim Technical Assessment of the MARCH Code*, NUREG/CR-2285, SAND 81-1672, Sandia National Laboratories (1981).

Idaho National Engineering Laboratories (INEL) and to provide a correlation of the results to the NRC Emergency Action Level Guidelines.* This work is the result of this request, in which the main goal has been to obtain practice and experience in the application of MARCH to BWR accident sequence analysis. The work is generally based on the design of the Browns Ferry Nuclear Plant, but no extensive consultation with TVA personnel or any use of the TVA simulator, to help ensure realism, have been employed in these studies.

The results presented in this report are examples of what can be achieved by stand-alone application of the MARCH code to BWR severe accident analysis. While information gleaned from several sources was used to help characterize behavior during the early parts of the transients studied, there is little question that the credibility and accuracy of the results would be improved by the use of a more detailed thermal-hydraulic code for analysis of the events of each sequence before core uncovery, then initiating MARCH at a point in time just before the core is uncovered.

A great deal of experience in the application of MARCH to BWR analysis has been gained, and many necessary improvements to the BWR models incorporated in the code have been implemented as a direct result of studies such as these. Nevertheless, it must be emphasized that the assumptions leading to the results presented here have not been scrutinized for accuracy to the Browns Ferry Nuclear Plant, nor do the results reflect recent improvements in the MARCH code that correct BWR containment modeling errors.

**Draft Emergency Action Level Guidelines for Nuclear Power Plants*, NUREG-0610. Washington, DC: U.S. Nuclear Regulatory Commission (1979).

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SASA Project Manager
Oak Ridge National Laboratory

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The courtesy and cooperation from the Brookhaven National Laboratory and Battelle Columbus Laboratory are gratefully acknowledged; in particular, Dr. Wm. Trevor Pratt, BNL, who made the MARCH 1.4 version available and Dr. Roger O. Wooton, BCL, who provided valuable guidance in using the MARCH code.

Acknowledgment is made to Dr. Mario H. Fontana, the first SASA project manager at ORNL, and Drs. Raymond DeSalvo and Robert T. Curtis, the successive SASA program managers at NRC. Acknowledgment is also made to Dr. S. A. Hodge, SASA project manager at ORNL during the course of this work, and to C. G. Lawson for their careful reviews of this report.

NOMENCLATURE

ADS	automatic depressurization system
ANS	American Nuclear Society
ATWS	anticipated transient without scram
BAF	bottom of active fuel
BOV	bottom of vessel
BWR	boiling water reactor
CRD	control rod drive
CSRS	core spray recirculation system
ECCS	emergency-core-cooling system
ECI	emergency coolant injection
EPA	electrical penetration assembly
FW	feedwater
HPCI	high-pressure coolant injection
IEEE	Institute of Electrical and Electronics Engineers
LOCA	loss-of-coolant accident
LPCRS	low-pressure coolant recirculation system
MARCH	meltdown accident response characteristics
MSIV	main steamline isolation valve
NPSH	net positive suction head
NR	narrow range
PCS	power conversion system
RCIC	reactor core isolation coolant
RCS	reactor coolant system
RHR	residual heat removal
RHRSW	residual heat removal service water
RWCU	reactor water cleanup
SBGTS	standby gas treatment system
SLCS	standby liquid control system
SRV	safety relief valve
TAF	top of active fuel
TIP	traversing in-core probe

UNITS

Pa (psia)	pressure
K (°F)	temperature
kg (lb)	mass
kg/s (lb/m)	flow rate
J (Btu)	energy
m^3 (ft^3)	volume
m^3/s (ft^3/m)	leak rate
cc/s (cm^3/s)	

SUMMARY

This work sets forth the results of MARCH code calculations made for the major events that may occur at a BWR 4/Mark I nuclear power plant following a number of postulated transients. These events are, in turn, correlated to the NRC Emergency Action Level Guidelines. The Browns Ferry Nuclear Plant Unit 1 was used as a model in this study. Under the assumptions used in this study, all accident sequences analyzed would eventually result in core melt and containment breach unless the operator took corrective action.

The accident sequences studied in this work consist of those identified in the Reactor Safety Study as being dominant contributors to public risk at a BWR nuclear plant: (1) TW (anticipated transient followed by loss of decay heat removal, offsite and onsite ac power assumed available, initiating event assumed to be loss of main condenser vacuum); (2) TC (anticipated transient without scram, manual rod insertion and standby liquid control systems assumed unavailable); (3) TQUV [anticipated transient combined with failure of high-pressure coolant injection (HPCI), reactor core isolation coolant, and low-pressure emergency-core-cooling systems (LPECCSs)]; (4) AE (large LOCA with failure of emergency coolant injection); (5) S_1E (small LOCA with failure of HPCI and LPECCS); (6) S_2E (small LOCA with failure of HPCI, RCIC, and LPECCS); (7) S_2I (small LOCA with failure of low-pressure coolant recirculation system); and (8) S_2J (small LOCA with failure of residual heat removal (RHR) service water system for cooling RHR heat exchangers — LPCI mode of RHR system is available for suppression pool cooling).

Calculations for the eight sequences were made with the MARCH computer code modified to include the actinide decay power, effects of steel-water reaction, and a more accurate modeling of the vessel lower head during the core meltdown accident. The incorporation of these modifications in the MARCH code has been shown in the Browns Ferry station blackout sequence study to predict core uncovery to occur sooner by ~18% and the peak containment temperature and pressure to be higher by ~100%. This trend of earlier core uncovery and higher peak containment temperature and pressure following the core meltdown is in general agreement with predictions by the KESS code. As the MARCH code, including the modified version used in this study, contains a number of limitations and deficiencies, the present study is primarily useful in providing a preliminary assessment of the BWR accident sequences studied and a comparative study of containment failures by overtemperature or by overpressure.

Table 1 gives a summary of containment failure time based on MARCH calculations for the eight BWR accident sequences due to failure in electric penetration assemblies by overtemperature as compared to failure by overpressure as used in WASH-1400. Failure in EPAs by overtemperature has been shown to result in a decrease of containment failure time ranging from 28% for sequence TC to 91% for sequence AE.

Table 2 gives a summary of emergency action levels corresponding to the eight BWR accident sequences studied in this work.

Based on results obtained from this study, although containment breach due to failure in EPA seals would occur earlier as compared with predictions by WASH-1400, consequences of containment failure, however,

Table 1. Comparison of containment failure time

Sequence	Containment failure time (min)		Decrease of failure time (%)
	Overpressure[a]	Overtemperature[b] (drywell EPA)	
TW	1018	1389	
TC	961	692	28.0
TQUV	288	193	33.0
AE	183	17	91.0
S_1E	200	40	80.0
S_2E	210	45	79.0
S_2I	1533		
S_2J	1632		

[a]Containment failure by overpressure at 1.22 MPa, assuming no prior failure due to overtemperature.

[b]Drywell electric penetration assembly seal failure at ambient temperature above 533 K.

Table 2. Summary of emergency action levels

Sequence	Initial condition	System condition	Action level
TW	Loss of condenser vacuum (t = 0–30 s)	HPCI and RCIC start (t = 1 min)	Notification of unusual event
		Operator determines RHR not available (t = 20 min)	Declare alert
		Operator determines RHR/PCS not readily repairable (t = 1 h)	Site emergency
		Core melt and containment breach inevitable (t = 1–3 h)	General emergency
TC	Loss of feedwater (t = 0)	Operator recognizes ATWS (t = 2 min)	Site area emergency
		Power level decreases to afterheat and RHR systems in operation (t ≃ 30 min)	Reduction to alert
		Plant brought down to cold shutdown (t = several hours)	Closeout of offsite emergency
TQUV	Loss of feedwater (t = 0)	Failure of HPCI/RCIC on demand and LPECCS inoperable (t = 1–10 min)	General emergency
AE	Large-break LOCA (t = 0)	Core uncovery (t = 1–10 min)	General emergency
S_1E	Small-break LOCA (t = 0)	Failure of HPCI and unavailability of RHR and CS pumps (t = 1–10 min)	General emergency
S_2E	Small-break LOCA (t = 0)	Failure of HPCI, RCIC, RHR, and CS systems (t = 1–10 min)	General emergency
S_2I S_2J	Small-break LOCA (t = 0)	Primary coolant leak rate >50 gpm (t = 1–10 min)	Alert
		Suppression cooling found to be not available (t ≃ 30 min)	Site area emergency
		Long-term heat removal determined not available (t = several hours)	General emergency

would be considerably mitigated. The containment pressure drop following failure of EPA seals would prevent containment structural failure. Furthermore, the amount of fission product releases outside the containment should also be greatly reduced due to deposition of fission products and filtering effect of EPA seals following degraded core accidents.

BWR 4/MARK I ACCIDENT SEQUENCES ASSESSMENT

D. D. Yue T. E. Cole

ABSTRACT

This work uses the MARCH computer code to investigate the major events that may occur at a BWR 4/Mark I nuclear power plant following a number of postulated transients. These events are, in turn, correlated to the Nuclear Regulatory Commission Emergency Action Level Guidelines. The Browns Ferry Nuclear Plant Unit 1 was used as a model in this study. Under the assumptions used in this study, all accident sequences analyzed would eventually result in core-melt and containment breach unless the operator took corrective action. In each sequence, the effect of parameter variations on the accident progression has also been investigated.

Results of this study show that in most core meltdown sequences overtemperature in the drywell electric penetration assembly (EPA) seals would be the dominant failure mode except for sequences TW, S_2I, and S_2J, in which there is a total loss of decay heat removal capability, with resultant higher pressure buildup in the containment. For the latter sequences, overpressurization would be the dominant containment failure mode. With the assumptions concerning EPA seal failure used in this study, both failure modes would result in containment breach much sooner and correspond to a lower containment pressure than those predicted in WASH-1400 for similar sequences. The amount of fission product releases outside the containment, on the other hand, might be greatly reduced due to deposition of fission products and filtering effect of EPA seals following degraded core accidents.

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1. INTRODUCTION

The primary purposes of this work are to examine, through use of the MARCH code, the major events that may occur at a BWR 4/Mark I nuclear power plant following a number of postulated transients and to correlate those events to the Nuclear Regulatory Commission (NRC) Emergency Action Level Guidelines.[1] The Browns Ferry Nuclear Plant Unit 1 was used as a model for this investigation.

The accident sequences to be studied were provided by EG&G, Idaho, Inc., at the request of NRC[2] and are among those identified in the Reactor Safety Study (RSS)[3] as being dominant contributors to public risk at a boiling water reactor (BWR) nuclear plant. Additional information on the sequences, particularly in the early part of the sequences, may be found in NUREG/CR-2100 (Ref. 4). The main emphasis in this work has been on the back-end part of the sequences as provided by the MARCH code.[5]

Note that MARCH code[5] was originally created for the RSS[3] to give an account of the entire course of a postulated core meltdown accident. It was necessary to model various physical phenomena on a somewhat simplistic basis, sacrificing highly detailed modeling but achieving a fast running, integral meltdown code. Within the context of current uncertainties associated with meltdown accident phenomenology and for the purposes of this preliminary study of behavior during postulated accidents, the use of MARCH code constitutes an appropriate analytical approach. As the MARCH code contains a number of limitations and deficiencies and could sometimes yield nonconservative predictions,[6,7] the present study with modified MARCH is primarily useful in providing a preliminary assessment of the BWR accident sequences studied and a comparative study on containment failures by overtemperature or by overpressure.

The sequences studied in this work consist of the following eight BWR accident sequences:

1. TW (anticipated transient followed by loss of decay heat removal),
2. TC (anticipated transient without scram),
3. TQUV (anticipated transient combined with failure of HPCI, RCIC, and LPECCS),
4. AE (large LOCA with failure of ECI),
5. S_1E (small LOCA with failure of HPCI and LPECCS),
6. S_2E (small LOCA with failure of HPCI, RCIC, and LPECCS),
7. S_2I (small LOCA with failure of LPCRS),
8. S_2J (small LOCA with failure of RHRSW for cooling RHR heat exchangers; LPCI mode of RHR system is available for suppression pool cooling).

The overall plant configuration is shown in the simple schematic diagram given in Fig. 1.1. Two recirculation loops provide drive flow to the 20 jet pumps located around the core and thereby provide coolant flow to the core. The two-phase steam-water mixture generated in the core flows upward through the axial steam separators, and the steam continues through the dryers and directly out through the steam lines to the turbine-generator. The condensate flow is then returned through the feedwater heaters by the condensate-feedwater pumps into the vessel.

In the normal mode of operation, the nuclear plant responds to small variations in input demands in a continuous manner under the action of component controllers. Under abnormal conditions resulting from various initiating events, large transient demands are introduced on the nuclear system. These demands are met by the addition of system protective components that will maintain the plant parameters within permissible bounds. Such a structure may be illustrated by a trip logic tree given in Fig. 1.2.

The characteristics and event timing for each sequence have been determined by the MARCH code,[5] which has been modified to include the actinide decay power,[8,9] effects of steel-water reaction,[10] and a more accurate modeling of the vessel lower head during the core meltdown accident.[11] A number of parameter variations have also been investigated to determine effects on the accident progression. The parameter variations studied include pipe break sizes, vessel depressurizing rates, options in the MARCH code, containment failure modes, and operator's mitigating actions.

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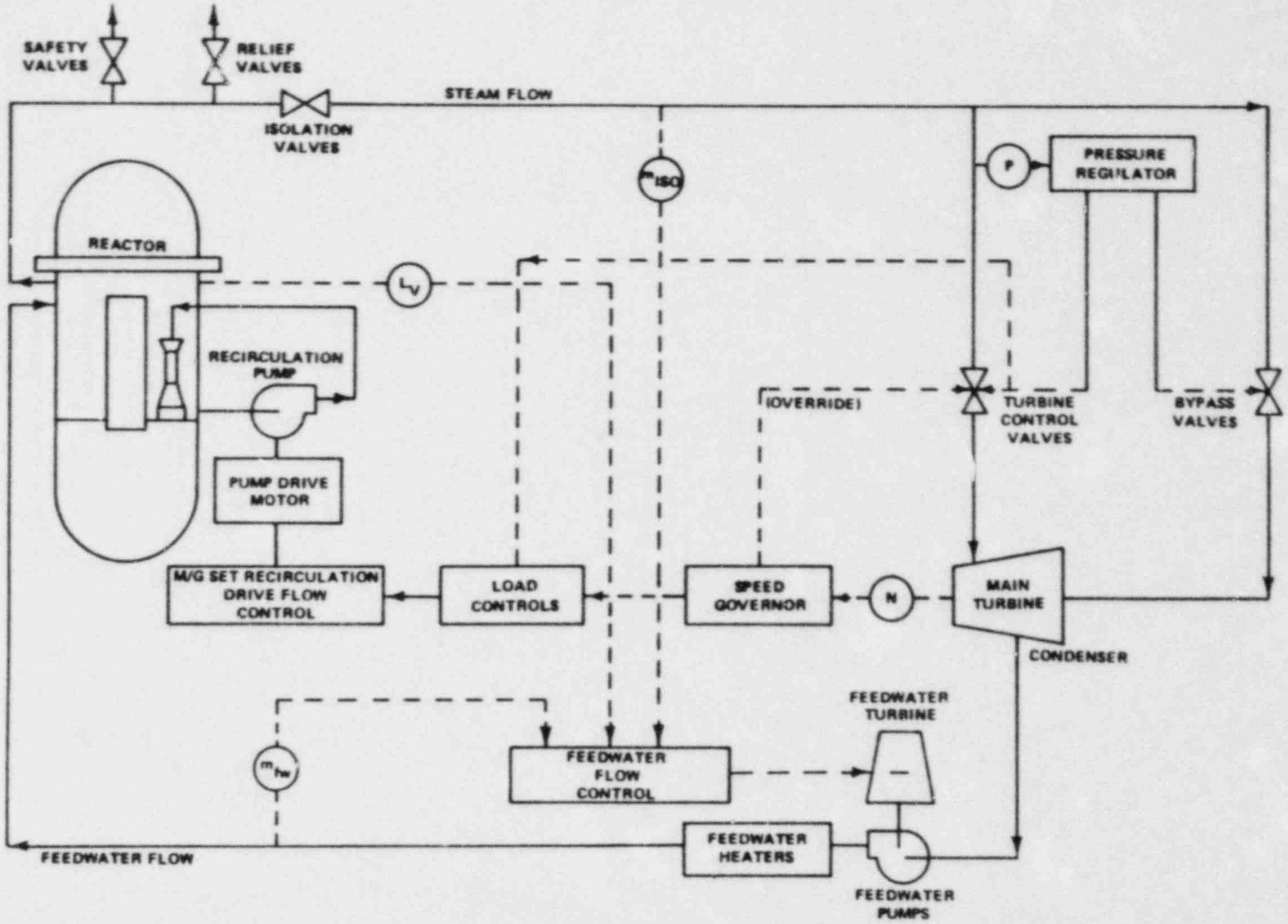

Fig. 1.1. Basic plant configuration.

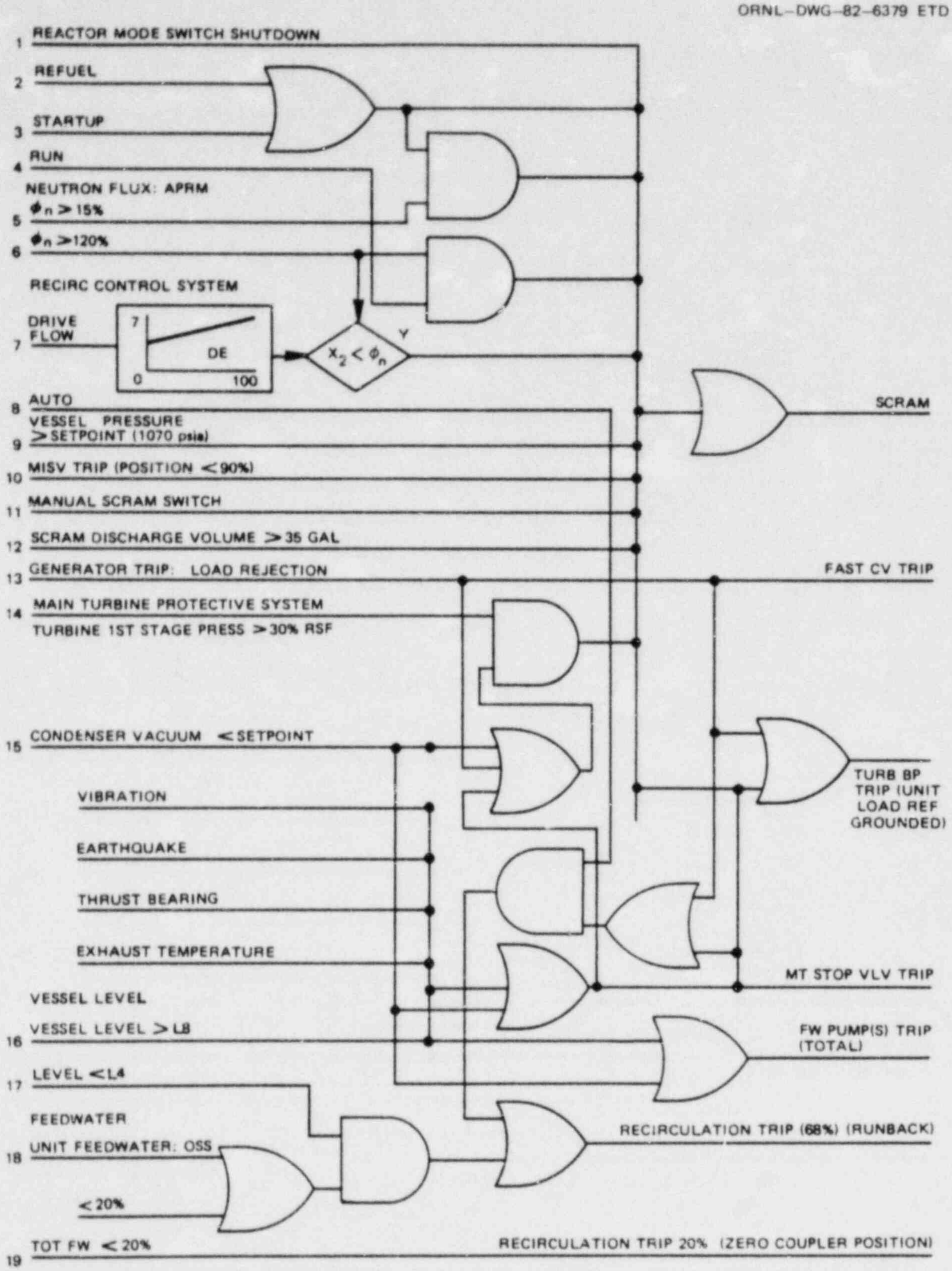

Fig. 1.2. Trip logic.

Following the core-melt accidents, containment could fail either by overtemperature or by overpressurization depending on whether there is a total loss of RHR system. In addition, for sequences in which excessive amounts of superheated steam and noncondensibles are discharged into the suppression pool within a short time, the wetwell could also fail before the drywell due to forces of steam condensation oscillations — the so-called "Wurgassen effect." Containment failure due to steam explosion following the core melt has been found highly unlikely for the Zion plant,[12] and recent experiments at Sandia[13] have shown that corium does not undergo violent explosions upon interaction with water. Therefore, containment failure due to steam explosion is not considered in this work. Furthermore, containment failure as a result of overpressurization by hydrogen burning is also not considered, because the containment atmosphere at Browns Ferry is inert.

It should be noted that MARCH code[5] does not provide detailed core thermal hydraulics or neutronics, nor does it provide system behavior of the balance-of-plant. Descriptions of early events in the sequences are based mostly on NUREG/CR-2100 (Ref. 4). However, NUREG/CR-2100 does not provide timing details for the early events. Timings for those early events such as closure of MSIVs and turbine bypass valves, and tripping of main turbine and feedwater pumps represent best-estimate values based on the FSAR (Ref. 14).

Timings for core melting and fuel relocating following boiloff of vessel water entail a great deal of uncertainties. This is due in part to a lack of fuel relocation modeling in MARCH and in part to a lack of experimental data concerning core meltdown phenomena. For all accident sequences studied in this work, use has been made of meltdown model A, which assumes that all heat in the molten region greater than that required to just keep the core material molten is transferred downward; this approximates the core material movement following the core melt. In a more realistic case, the partially molten core could cause steam blockage, change interfacial areas for cladding oxidation, and change areas and path lengths for heat transfer; these would, in turn, affect the core heatup rates and the Zircaloy cladding oxidation reaction rates. If core deformation significantly increases the interfacial area for oxidation and steam flow to the cladding, the meltdown process would accelerate. On the other hand, a decreased exposure of interfacial area for oxidation and steam blockage would decelerate the meltdown process. Furthermore, a homogenized molten corium is assumed in model A to remain at the fuel melting temperature before the vessel fails. This assumption may prove to be overconservative.

Timings for containment events are based on MARCH calculations, which may prove to be overconservative, because the code is based on the lumped parameter model and does not include spatial effects of temperature and pressure distribution following the accident. Also, the drywell EPA seals are not modeled, and thus no account is taken for the seal behavior as the ambient temperature rises above the design temperature limit.

In each of the eight BWR accident sequences studied in this work, a short description of the initiating events is first given. The chronology for the main sequence is then presented in tabular form. Selected results from MARCH calculations are presented in figures for each accident sequence. In general, these results include the time distribution of core temperature, the fraction of core melted, vessel pressure and water level, and temperature and pressure responses in the containment. Because the water level is calculated using the flow cross section at the core mid-plane, water levels shown in the figures are accurate only between the top and bottom of the core.

Appendix A lists (1) the reactor vessel level and pressure setpoints and their functions and (2) the eight groups of the primary containment isolation system. MARCH code input listings for the eight BWR sequences are given in Appendix B.

2. MARCH COMPUTER CODE

The MARCH computer code[9] has been used to calculate responses of the primary system and containment for the accident sequences. The version used for this work is based on MARCH 1.4 from Brookhaven National Laboratory, which includes effects of steel-water reaction,[10] and a more accurate modeling of the vessel lower head during the core meltdown accident.[11] This version has been further modified to correct an error in the HEAT subroutine and to include the actinide decay power in the ANSQ subroutine. This modified version has been designated as MARCH 1.4B.

In the MARCH 1.4 code, the fission product decay heat source term is based on ANS Standard ANS-5.1 (1973),[15] and the decay of ^{239}U and ^{239}Np are not accounted for. In the modified MARCH code (1.4B) used in this work, the fission product decay power calculations are based on ANS-5.1 (1979)[16] and also include the actinide decay heat source in a BWR as described in Ref. 17.

The actinide heating calculations reported in Ref. 17 were made using the EPRI-CINDER code[18] and included all significant actinides from ^{208}Tl through ^{246}Cm (~50 nuclides). Time-dependent cross sections of major fuel nuclides from ^{232}Th through ^{242}Pu, and four-group neutron fluxes were obtained from EPRI-CELL code[19] calculations. Cross sections for the remaining actinides were processed from ENDF/B into 154 groups and then collapsed to four groups.[20,21] In the calculations of Ref. 17, a depletion of 34,000 MW/t was used for each of the three reactor types investigated; for the BWR case it was indicated that this was equivalent to a depletion time of 45,820 h.

The fission product decay powers, also as reported in Ref. 17, were calculated using the pulse functions from the ANS-5.1 Standard (1979)[16] for an irradiation time of 45,820 h. An upper-bound correction (G_{max}) in the standard was used for fission product absorption effects to obtain a smooth correction with cooling time. Contributions from the actinide decay power are significant since during the cooling period between 0 and 10^5 s, the actinide decay power varies from ~6 to 26% of the fission product power in a BWR for the assumed conditions.[17] However, the result as implemented in March 1.4B produces a decay heat power that significantly differs over much of the time span of interest from either that given by Ref. 17 or that from the ANS-5.1 (1979) model including the actinide decay contribution recommended in the model. Therefore, the reader should note that the decay heat values used in this study may significantly overestimate the decay heat power for a typical BWR.

Results of MARCH 1.4B and 1.4 calculations have been compared for the TW sequence with complete station blackout as the initiating event.[8,9] Results show that the time predicted by MARCH 1.4B for core uncovery is less by about 18% and that the maximum containment wall temperature is increased by more than ~100%. This trend of earlier core uncovery and higher containment wall temperature generally agrees with predictions by the KESS code.[22]

3. CONTAINMENT FAILURE MODES

Following the core-melt, containment failure could occur either in the drywell or in the wetwell. The Reactor Safety Study[3] considers containment overpressurization at about 1.30 MPa (189 psia) to be the dominant failure mode for all severe accident sequences. While this is found to be the dominant failure mode for sequences TW, S_2I, and S_2J, in which the containment residual heat removal systems are assumed to have failed, overtemperature in the drywell electric penetration assemblies (EPAs) has been identified as the dominant failure mode for all other sequences. The pressure at which containment would fail by overpressurization in sequences TW, S_2I, and S_2J is also lower than that used in RSS, since EPA seals would fail at a lower pressure according to the FSAR.[23] In sequences TQUV, TC, AE, S_1E, and S_2E, containment failure is predicted to result from the failure of drywell EPA seals by overtemperature.*

A typical drywell electric penetration assembly canister used in the Browns Ferry nuclear plant is given in Fig. 3.1. According to the FSAR,[23] the EPA seals for Unit 1 are qualified for short-term temperature rating of 436 K (325°F) for 15 min and long-term temperature rating of 411 K (281°F). The pressure ratings for both temperatures were 0.96 MPa (139 psia). At ambient temperatures above 435 K (325°F) longer than 15 min, the EPA seals would not only lose electrical insulation properties but also the sealing integrity. Upon the loss of electrical insulation properties, short circuit could occur, causing further damage to the containment and fires if the ac or dc power were still available. To account for the time constant of the electrical insulation materials undergoing deterioration, a higher temperature of 477 K (400°F) has been used for the containment to develop a leak rate in excess of 10^{-2} std cc/s per EPA, or ~0.5 std cc/s for a total of 50 EPAs in a typical nuclear plant, which has been set as the failure criterion in IEEE-Std-317.[24]

As the drywell temperature increases further, the dielectric material would totally lose electrical insulation properties and sealing integrity until finally an excessive leak rate develops, resulting in containment failure. This is estimated to occur at about 533 K (500°F), allowing for the time constant of the electrical insulation materials to undergo further deterioration. This would correspond to ~1.0 MPa (145 psia) for the TW sequence based on MARCH calculations. This predicted pressure is about 30% lower than that based on failure by overpressure (Reactor Safety Study[3]).

In the wetwell, failure by overpressure due to loss of condensation effectiveness of the pressure suppression pool could also occur. For the ramshead sparger and for straight pipes, condensation instability has been observed during high steam discharge flow rates above the pool temperature limit, that is, T_{bulk} = 344 K (160°F) and T_{local} = 350 K (170°F). Although this pool temperature limit has been raised to T_{bulk} = 361 K (190°F) and T_{local} = 366 K (200°F) for the T-quencher spargers, responses above the T-quencher pool temperature limit have not been verified.[25]

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*It should be noted that the condition for containment failure is a user-input to the MARCH code.

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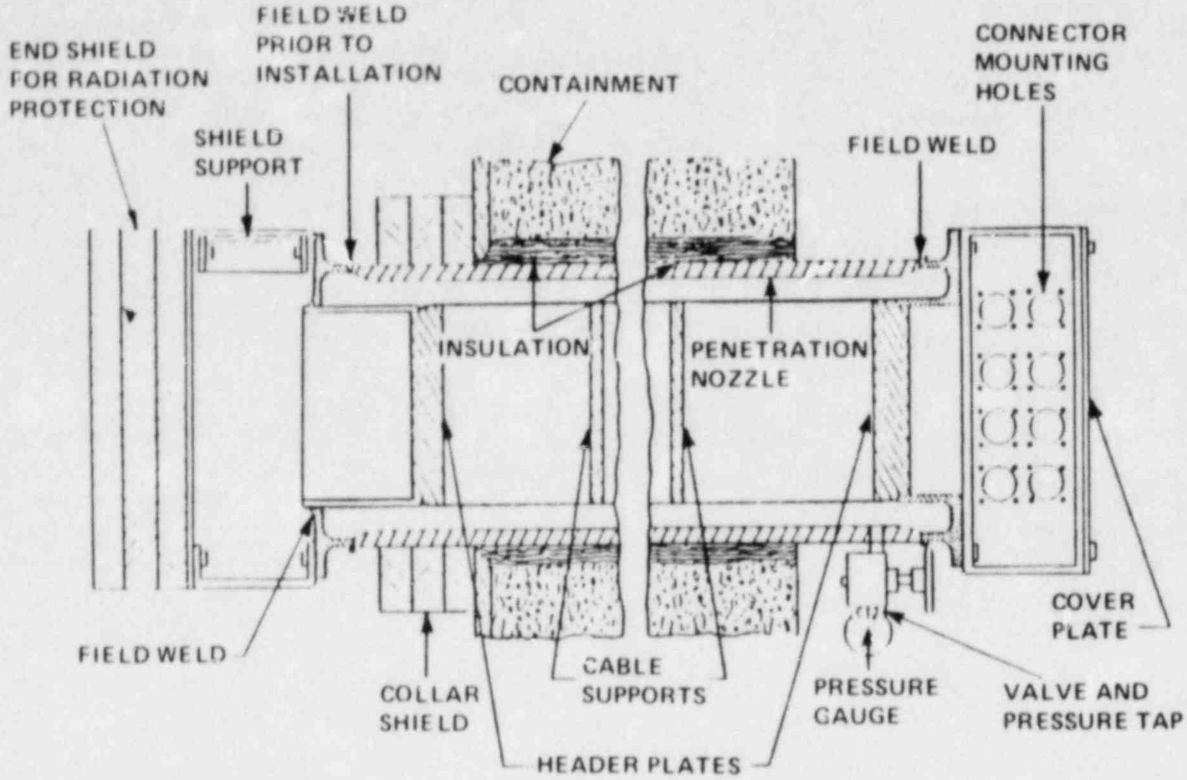

Fig. 3.1. Typical electrical penetration assembly canister.

Indeed, this limit would be exceeded shortly after the accident for the TQUV sequence due to localized SRV steam discharges through the T-quencher spargers and inadequate pool thermal mixing without residual heat removal (RHR) systems operating in the cooling mode.

The difference between the local and average pool temperature might increase from ~0 K at the beginning of the event to as much as ~45 K (113°F) about 100 min later. This indicates that the suppression pool could lose its condensation effectiveness as the local pool temperature exceeds 366 K (200°F) about one-half hour later,* even though the average pool temperature is only 316 K (110°F). At this time, there would still exist a sufficient degree of subcooling for the formation of detached steam bubbles from the T-quencher sparger. These steam bubbles would undergo oscillations as the pressure and temperature change. The resulting pressure loads from condensation oscillations would rapidly increase as the pool loses condensation effectiveness, leading to a possible rupture of the wetwell — the so-called "Wurgassen effect." This is estimated to occur at ~130 min into the sequence.[8] This mode of the wetwell failure, however, would not preclude the subsequent failure of drywell electric penetration assembly seals caused by overtemperature in the drywell following the core melt.

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*Assuming that the pool remains at atmospheric pressure.

4. ASSESSMENT OF ACCIDENT SEQUENCES

This section contains the assessment of the aforementioned eight BWR accident sequences identified as being dominant contributors to public risk in the Reactor Safety Study.[3] In each sequence, it is assumed that the accident would progress by the natural course following an initiating event and that operator actions are limited to vessel depressurization by opening the SRVs, or manual control of the HPCI, RCIC, or FW systems in providing makeup to the vessel.

Main emphases have been placed on the back-end events as stated previously. Except for a few minor differences, descriptions of the early events are mostly based on NUREG/CR-2100[4] in which the Peach Bottom plant has been used as the reference BWR plant. Timings for those early events, such as closure of MSIVs and turbine bypass valves and tripping of main turbine and feedwater pumps, represent best estimate values based on the FSAR.[14]

Some simplifying assumptions have also been made when limitations of MARCH code were encountered. An overflow condition resulted when the desired decay power level was used in the TC sequence. It was necessary to modify the input values to complete the run.

4.1 TW Sequence

The TW sequence is initiated by a transient event followed by a total loss of decay heat removal. Figure 4.1 presents the BWR event tree developed in the Reactor Safety Study[3] for the transient event initiator. As can be seen from Fig. 4.1, the TW sequence involves the subsequent failure to remove decay heat from the suppression pool using the Residual Heat Removal (RHR) system and also involves the failure of the Power Conversion System (PCS) to remove decay heat via the main condenser.

Several initiating events could lead to this sequence. For the purpose of this analysis, it is assumed that the main condenser vacuum is rapidly lost;[4] this could result from loss of condenser coolant flow, though other causes could be postulated. Loss of vacuum (normally >0.085 MPa, or 25 in. Hg.) leads to scram at about 0.078 MPa (23 in. Hg), main turbine trip, ~0.074 MPa (22 in. Hg), with consequent recirculation pump trip.* The turbine bypass valves would be blocked closed and a feedwater turbine trip initiated at ~0.024 MPa (7 in. Hg). The recirculation pump trip results in a rapid flow coastdown as the pump is decoupled from its variable speed source for this trip.

The resulting transients of reactor coolant pressure and level are assumed to lead to isolation of the reactor system at Level 2 as a result of "shrinkage" due to scram combined with a reduction of void fraction due to pressure increase. Rapid coastdown of coolant flow will tend to offset some of the reduction in void fraction, but generally an isolation trip would occur in this type of BWR. As the vessel water level drops below

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*Recirculation pump trip is initiated by fast closure of turbine control valves or stop valves with first-stage turbine pressure above 30%.

ORNL–DWG 81–8287 ETD

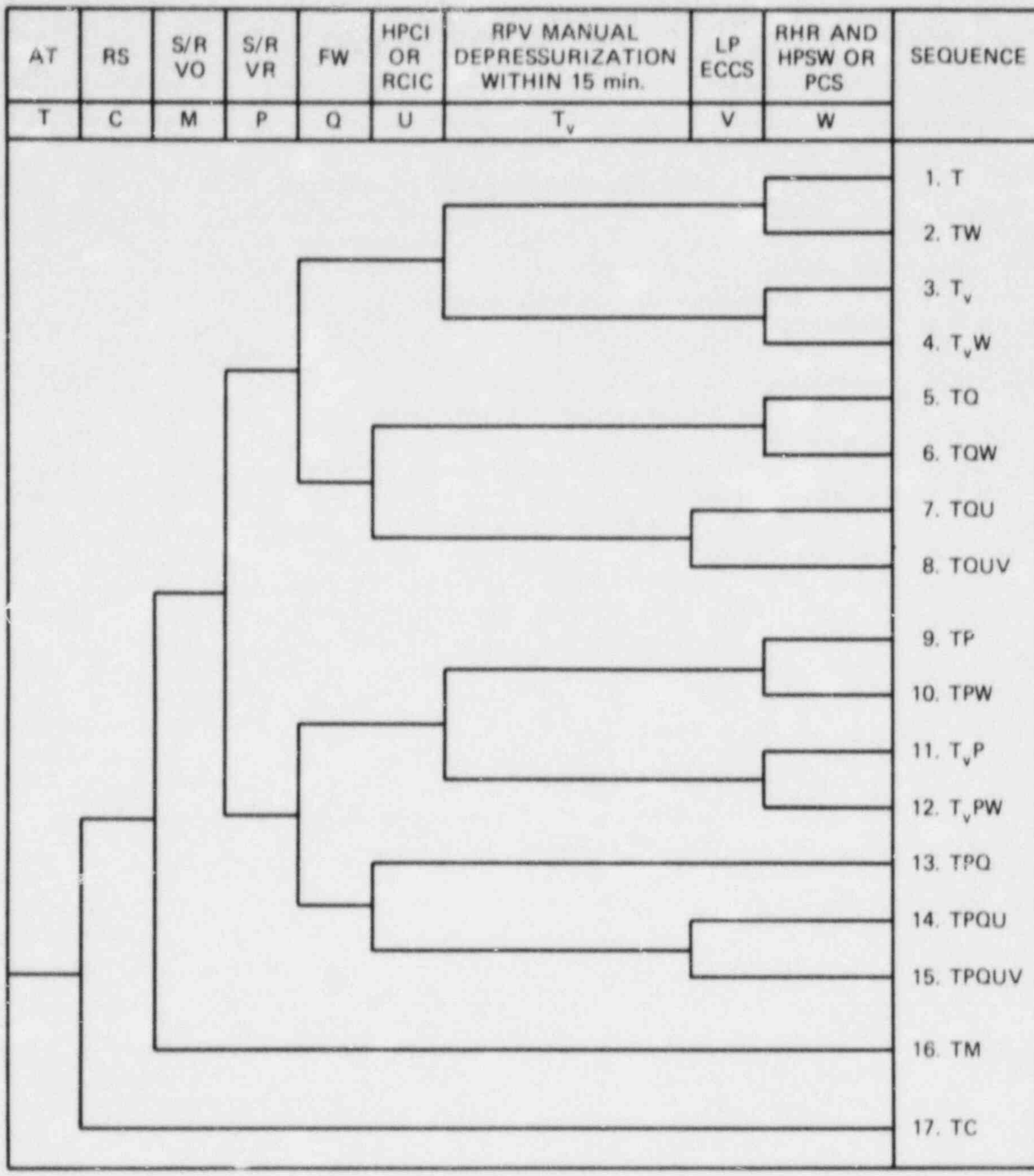

Fig. 4.1. BWR event tree.

Level 2, the MSIVs start to close and the HPCI and RCIC systems are actuated. About 4 s later, the MSIVs are fully closed and, about 30 s later, the makeup water from HPCI and RCIC systems begins to enter the vessel. At the same time, the excess pressure could be relieved through the SRV steam discharges into the suppression pool.

Having provided coolant makeup, the HPCI system would be secured at about 30 min into the transient. Thereafter, makeup water would be supplied by the RCIC system under operator control, while the CRD pumps would continue to supply coolant into the vessel, albeit at a flow rate of only ~3.14 L/s (50 gpm).

It is further assumed that by this time the operator has become aware that the RHR system is not available and that a condition exists in which no heat sink is available. Several alternatives are open to him. For this calculation, it is assumed that the operator depressurizes the vessel at about 15 min to a pressure level between 1.03 and 3.10 MPa (150 and 450 psia) and that the relief valves remain open throughout the transient so that the reactor vessel pressure equalizes with containment pressure. At about 75 min into the transient, coolant makeup would be provided by the core spray system.

With the total loss of decay heat removal from either the PCS or the RHR system, the containment would eventually fail by overpressurization. This is assumed to occur at about 1.22 MPa (174.7 psia). While EPA seals could also fail at this containment pressure, it has been conservatively assumed that the wetwell would fail first. Figure 4.2 presents the containment event tree. Based on MARCH 1.4B calculations and the containment failure modes assumed for this study as discussed in Sect. 3, the wetwell would first fail by overpressure at about 17 h into the TW sequence. Thereafter, the ECCS pumps would fail, the core would be uncovered, and melting would start. Subsequently, the vessel bottom head would fail at about 23 h, and corium would attack the concrete basemat. It is noted that the core melt would be considerably delayed if the EPA seals were assumed to fail before the wetwell. Key results and accident progression signatures obtained from MARCH for the TW0 sequence are presented in Figs. 4.3 and 4.4. Key timings of major events are given in Table 4.1.

Two cases, TW0 and TW1, have been investigated for the TW sequence with respect to the use of different options available in the MARCH code for vessel depressurization. In the first case, TW0, vessel depressurization is actuated when the core exit steam temperature, TVNT2, has exceeded a given temperature; this temperature has been selected such that the vessel would depressurize at about 15 min into the event when all the SRVs are opened. In the second case, TW1, vessel depressurization is initiated by the opening of six SRVs at 15 min. Results of the two MARCH calculations are summarized in Table 4.2. From these comparisons, it may be seen that the first option in which all the SRVs are open is to be preferred because it would result in a longer time before the containment breach and core melting.

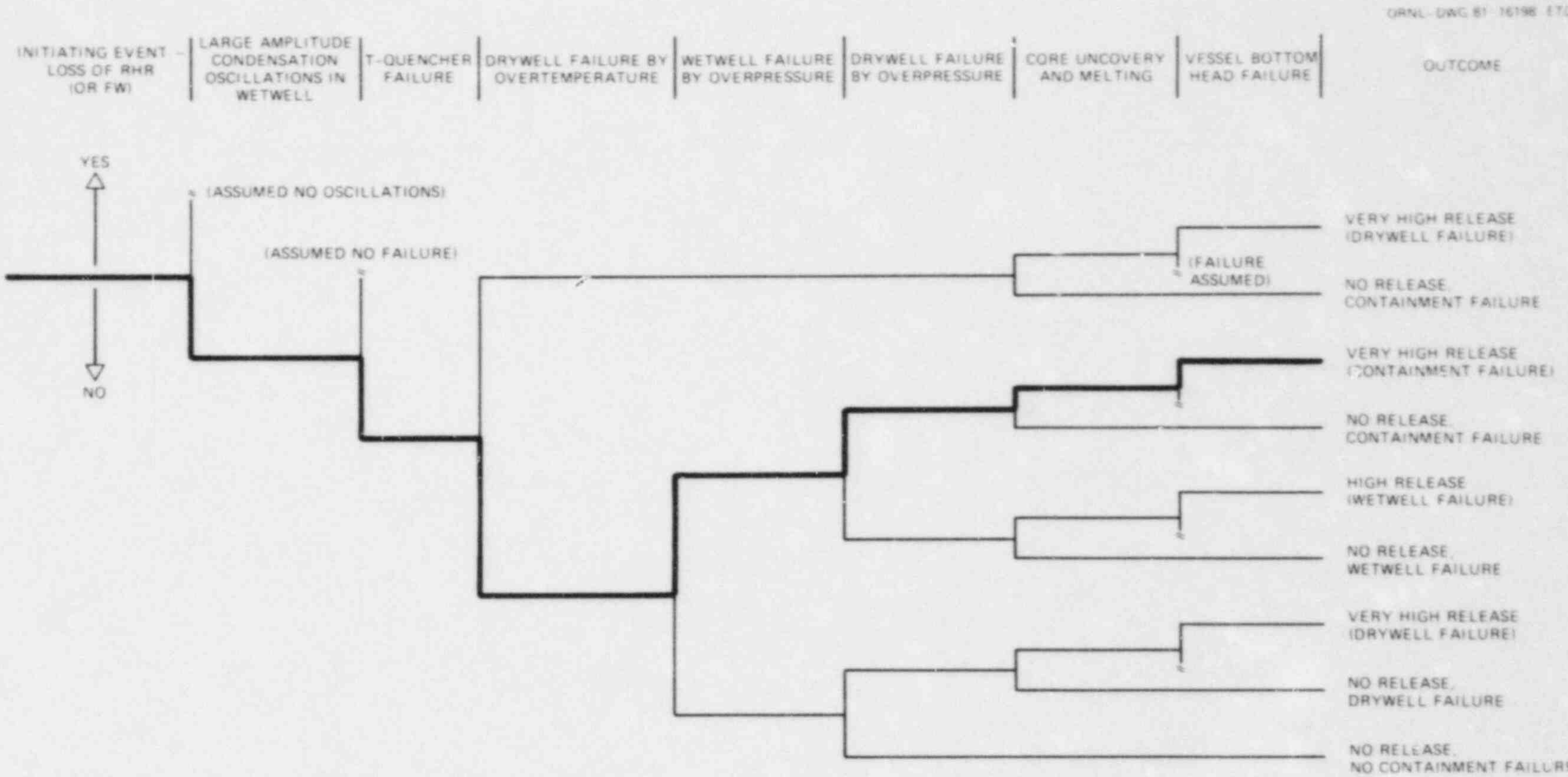

Fig. 4.2. Containment event tree following loss of decay heat removal.

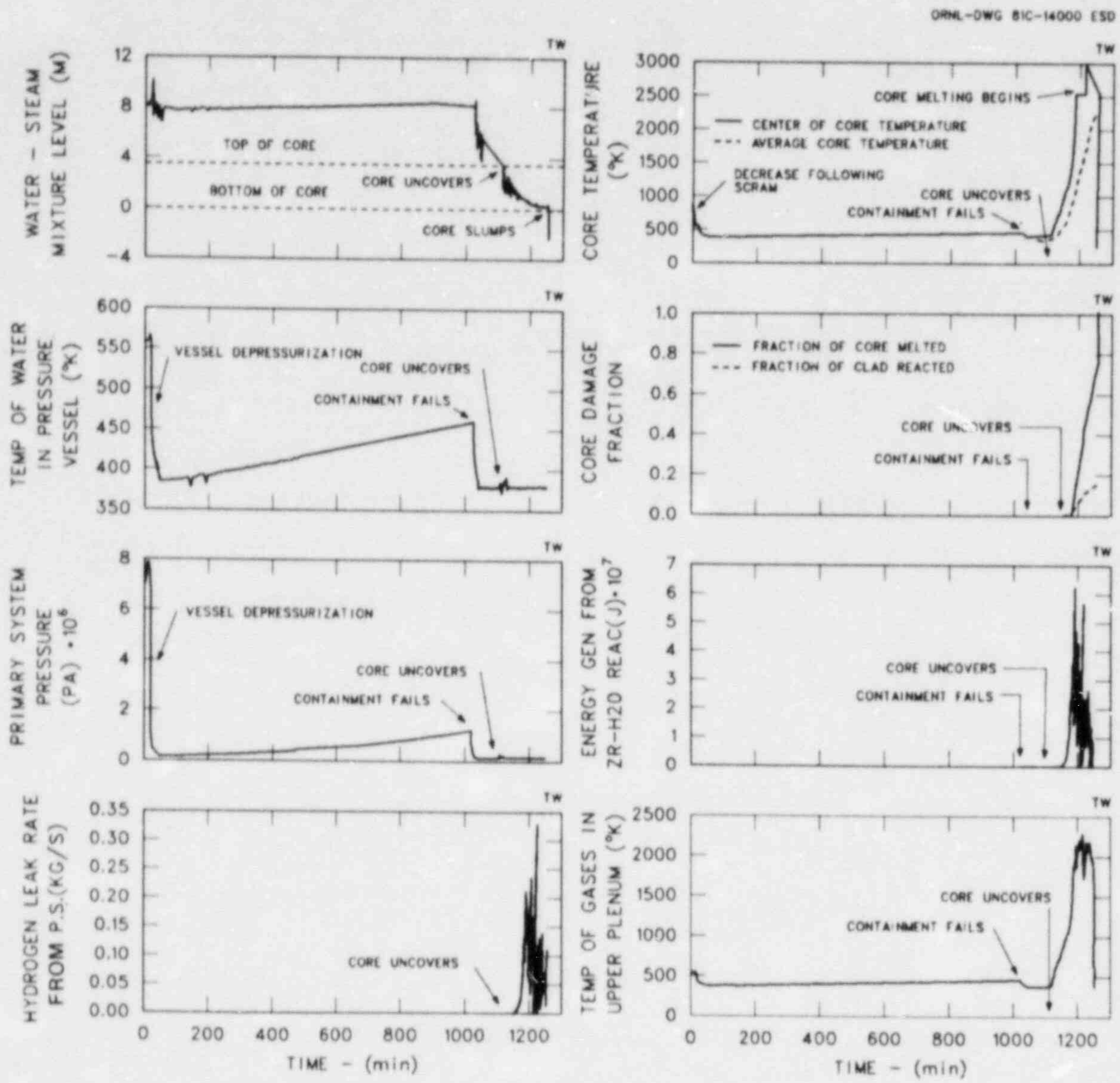

Fig. 4.3. Accident progression signatures for the TW sequence.

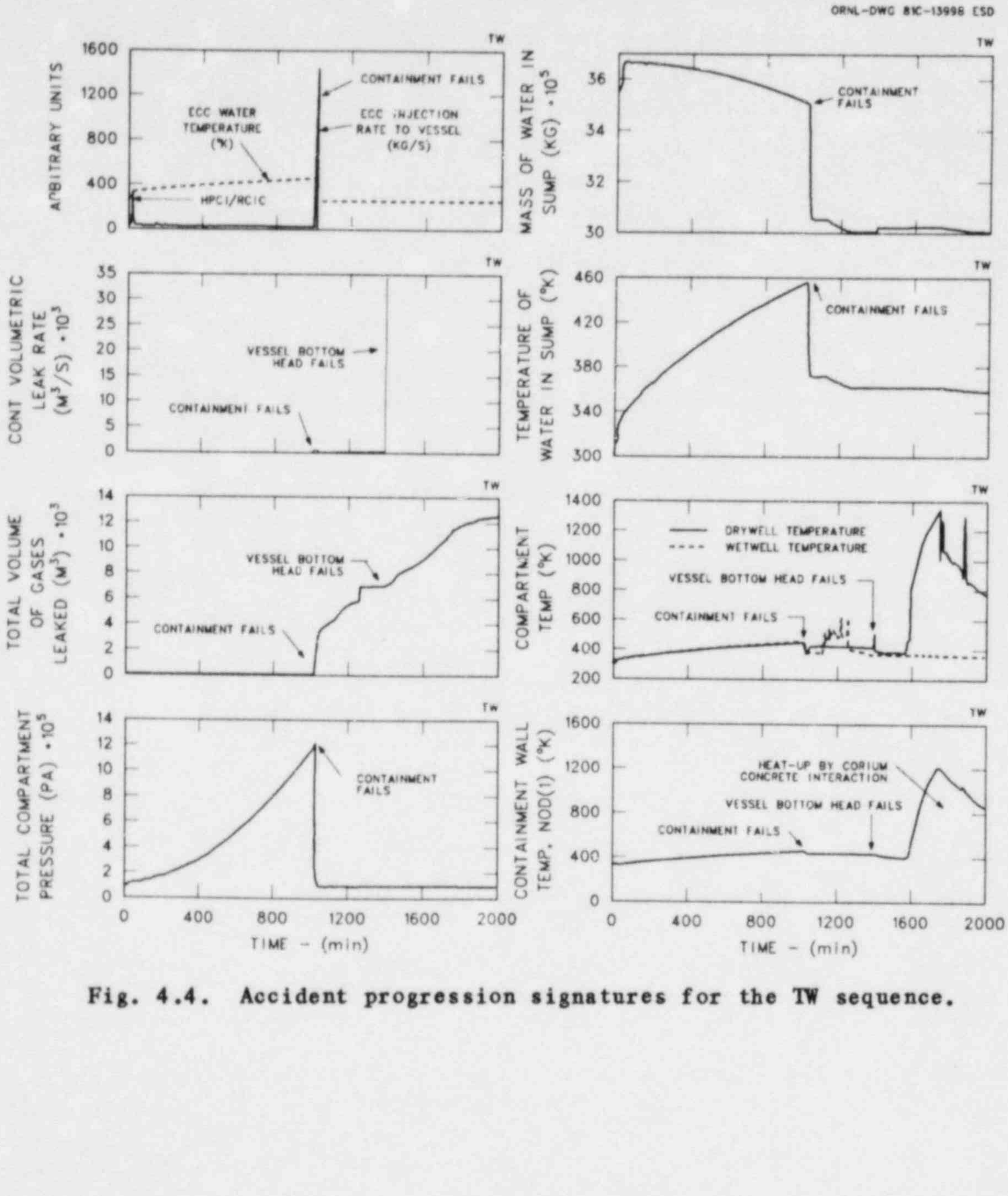

Fig. 4.4. Accident progression signatures for the TW sequence.

Table 4.1. Accident chronology of TW sequence

Time	Event
00:00	The plant is initially operating at 100% power. Initial drywell temperature = 339 K (150°F) Initial wetwell temperature = 308 K (95°F) Initial condenser vacuum range: 0.091 to 0.095 MPa (27 to 28 in. Hg)
00:00	Loss of condenser vacuum occurs following the tripping of condenser circulating water pumps (assuming at ~0.034 MPa or ~10 in. Hg/min).
00:28	Reactor scram is automatically initiated. All primary system isolation valves in groups 2, 3, and 6 are initiated to close at Level 4.
00:30	Main turbine trips off (turbine stop valves are fully closed).
00:31	Turbine bypass valves start to open due to turbine trip and function under pressure control until forced to close due to its condenser vacuum setpoint.
00:31	Recirculation pump and turbine-driven feedwater pumps trip off.
00:32	SRV's are actuated in response to pressure rise resulting from main turbine trip and steam begins to discharge into the pressure suppression pool through T-quenchers.
00:34	Low condenser vacuum initiates turbine bypass valve closure.
00:38	Feedwater flow decreases to zero.
00:45	All SRV's are closed.
01:00	MSIV's closure and HPCI/RCIC systems are actuated at Level 2.
>01:00	SRV's cycle to release decay heat.
15:00	The operator has found that RHR and containment sprays are not available. The operator manually opens SRVs to depressurize vessel.
30:00	The HPCI is secured.
1:15:00	The RCIC is secured. The LPCI is initiated, drawing water from the suppression pool.

Table 4.1 (continued)

Time	Event
16:58:09	The drywell EPA seals have failed at ~1.22 MPa (177.0 psia). The wetwell has also failed structurally due to overpressurization. The drywell and wetwell temperatures have exceeded 455 K (359°F). All ECCS pumps have failed due to insufficient NPSH.
18:23:28	Core uncovers.
19:21:27	Fission products begin to be released into the containment. The drywell and wetwell temperatures are 429 K (313°F) and 368 K (203°F), respectively.
19:35:28	Core melting starts.
20:51:22	Reactor vessel has dried out. The corium slumps to vessel bottom. The debris is starting to melt through the bottom head.
23:09:47	Vessel bottom head fails. The debris, at a temperature over 2082 K (3288°F), is starting to boil water from containment floor and to attack the concrete basement. The drywell temperature has exceeded 533 K (500°F).
27:20:51	The drywell temperature has exceeded 1107 K (1533°F).

Table 4.2. Timing of major events for TW sequence

Sequence	Timing of events (min)				
	Containment failure	Core uncovery	Start of core melt	Core slump	Failure of vessel head
TW0	1018	1103	1175	1251	1390
TW1	956	1041	1112	1185	1320

4.2 TC Sequence

The TC sequence is the ATWS sequence in the BWR. This accident sequence is concerned with a failure to make the reactor subcritical following an initiating event. The BWR transient event tree showing the TC sequence has also been given in Fig. 4.1.

Table 4.3 presents the four cases that have been studied for the TC sequence. A base case, TC0, assumes the Standby Liquid Control System (SLCS) is operational and results in neither core melt nor containment breach. The last two cases in which the power level is assumed to remain at 30% and 100% of the initial power level, however, resulted in computer overflow conditions. For this reason, a 5% power level has been used in the TC1 sequence. It is assumed that for the TC sequence, all four RHR pumps are operating in the suppression pool cooling mode and that all containment coolers and sprays are functioning as designed.

A number of likely transient-initiating events have been identified in the Reactor Safety Study[3] that would lead to the TC sequence. For this work, the loss of all feedwater has been selected to be the initiating event.[2] A loss of feedwater is an operational transient which occurs with a frequency of ~1-2 times per plant year. It may occur as a result of loss-of-offsite and -onsite AC power,[8,9] feedwater pump failures, condensate pump failures, feedwater controller failures, operator errors, or trip on reactor high water level.

Upon a loss of feedwater, vessel water level starts to decrease due to the mismatch between coolant inventory loss in the form of steam and supply of feedwater. The rate of level decrease depends on the initial power level; that is, higher initial power will cause faster level decrease. Because of diminishing injection of feedwater, core inlet flow decreases and temperature increases. This causes slightly more void generation in the core, thereby decreasing the neutron flux. When the plant is in the automatic flow control mode, control systems will function

Table 4.3. TC sequence

Sequence	MARCH input			Computer run result
	TRPS[a] (min)	ANSK[b]	TDK[c] (min)	
TC0	30.0	0.05	31.0	O.K.
TC1	60.0	0.05	1.0E4	O.K.
TC2	60.0	0.30	1.0E4	Overflow
TC3	60.0	1.00	1.0E4	Overflow

[a]TRPS = for time < TRPS the power level is (1.0 − time/TRPS).

[b]ANSK = minimum fractional power level used for time < TDK: ANSK > ANS decay power.

[c]TDK = time at which power level drops to ANS decay power.

to attempt to maintain the core power by increasing the recirculation pump speed (thus, the core flow). When the vessel water level decreases to Level 4, the reactor is scrammed and runback of the recirculation pump is initiated to protect the recirculation pumps from cavitation. For the TC sequence, it is assumed that manual rod insertion also fails after the failure of reactor scram. At this time, all containment isolation valves in Groups 2, 3, and 6 are initiated to close. Meanwhile, the vessel level continues decreasing due to steam flow to the main condenser through the turbine. Eventually, the wide-range sensed vessel level decreases to the Level 2 trip setpoint.

The Level 2 trip closes the MSIVs, trips the recirculation pumps, and initiates HPCI and RCIC. The recirculation pump trip results in a more rapid flow decrease than a loss-of-power transient. Due to the failure of reactor scram and manual rod insertion, the reactor power would increase to a maximum of about 572% of the initial power level shortly after the MSIV closure based on the REDY code.[24] The MARCH code,[5] however, does not account for the coupling between neutronics and thermal hydraulics. To compensate for this power level increase upon the MSIV closure, a longer time to hot shutdown has been used. All the SRVs are actuated to open, releasing vessel pressure through steam discharges into the suppression pool; reactor power is reduced due to increased void content. At about the same time the SRVs are opening, some of the fuel assemblies may have experienced transition boiling; some interaction between the Zircaloy cladding and steam is also predicted to occur, resulting in the generation of some hydrogen gas.

With confirmation from the flux monitoring system and the control rod position indicating system that scram has not taken place, the operator will then activate the SLCS. Assuming it takes the operator 2 min to recognize and verify the ATWS event and there is 1 min of transport time in the SLCS pipelines and the vessel, the reactor shutdown begins at about 3 min into the TC sequence.

With a flow rate of sodium pentaborate at about 3.14 L/s (50 gpm), the reactor will be brought to hot shutdown in ~30 min from the beginning of the event. In about 31 min, the power level will drop to the decay power level. Under normal circumstances, the reactor would be brought to hot shutdown in about 23 min. A longer time is used to compensate for the aforementioned power level increases unaccounted for in the MARCH code.

Based on MARCH 1.4B calculations, the TC0 sequence does not result in core melt or containment breach, although about 0.024% of the cladding has reacted with steam soon after the initiating event. Except during the initial moment, the core temperature has remained at about 565 K (558°F) throughout the event. Key results and accident progression signatures for the TC0 sequence are presented in Figs. 4.5 and 4.6.

For the TC1 sequence in which the SLCS is assumed to be inoperable, the accident does result in core melt and containment breach. Based on MARCH calculations, core melt begins at about 9.2 h and containment failure occurs at about 11.5 h due to overtemperature in the drywell EPA seals. Key results and accident progression signatures for the TC1 sequence are presented in Figs. 4.7 and 4.8.

The accident chronology of the TC0 sequence following the loss of feedwater transient is given in Table 4.4. Comparisons of timing of major events between the TC0 and TC1 sequences are given in Table 4.5.

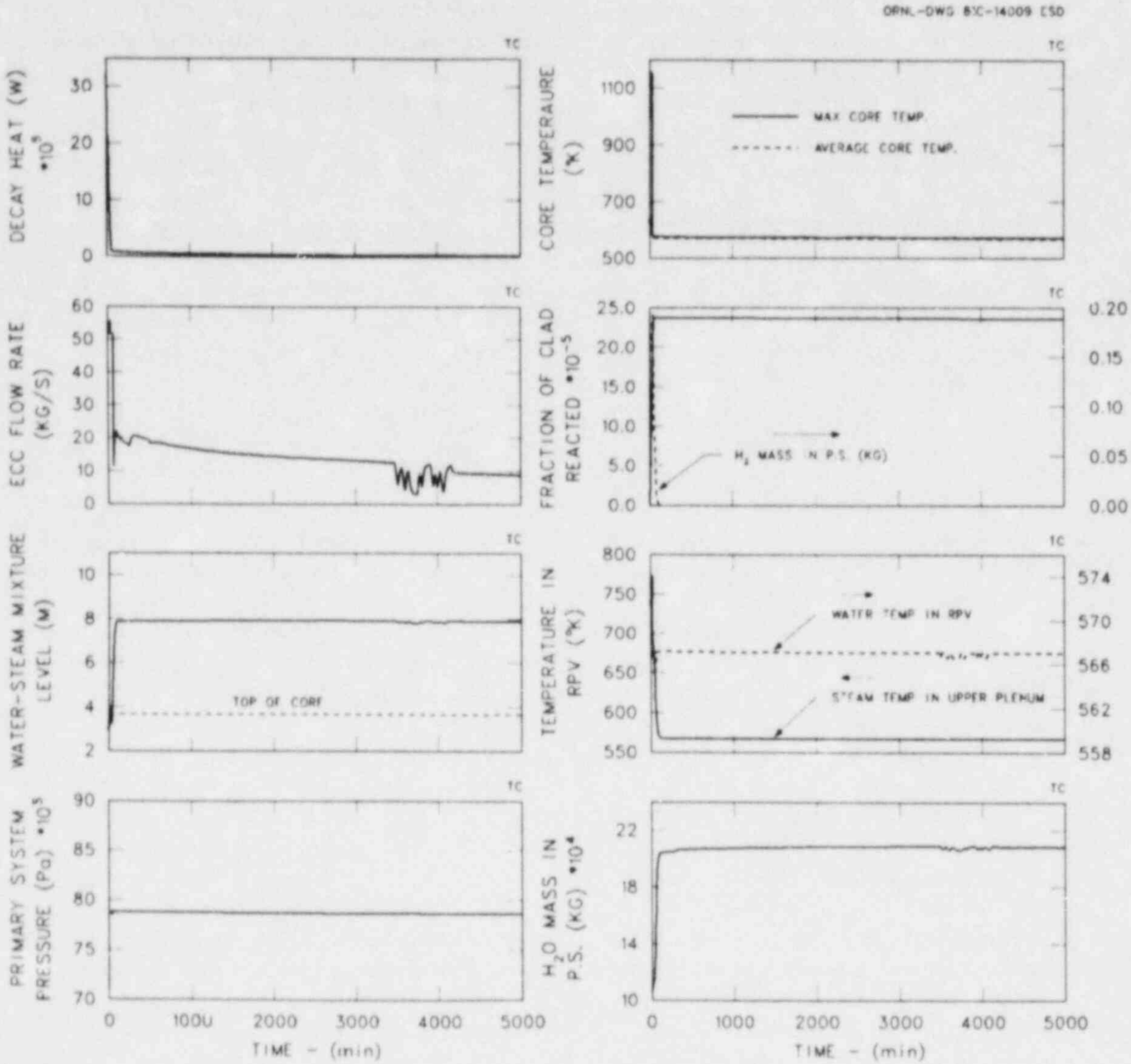

Fig. 4.5. Accident progression signatures for the TC sequence.

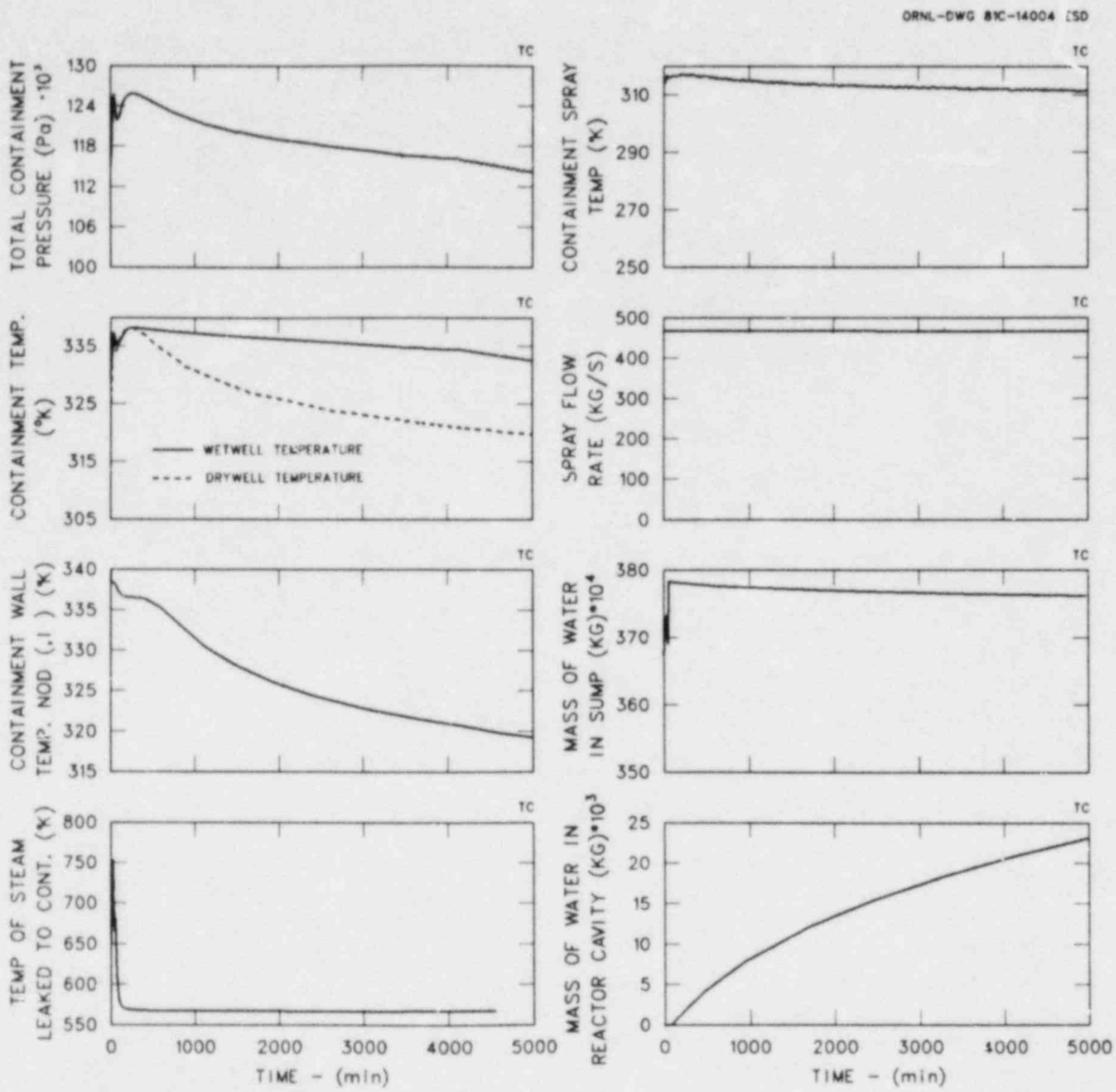

Fig. 4.6. Accident progression signatures for the TC sequence.

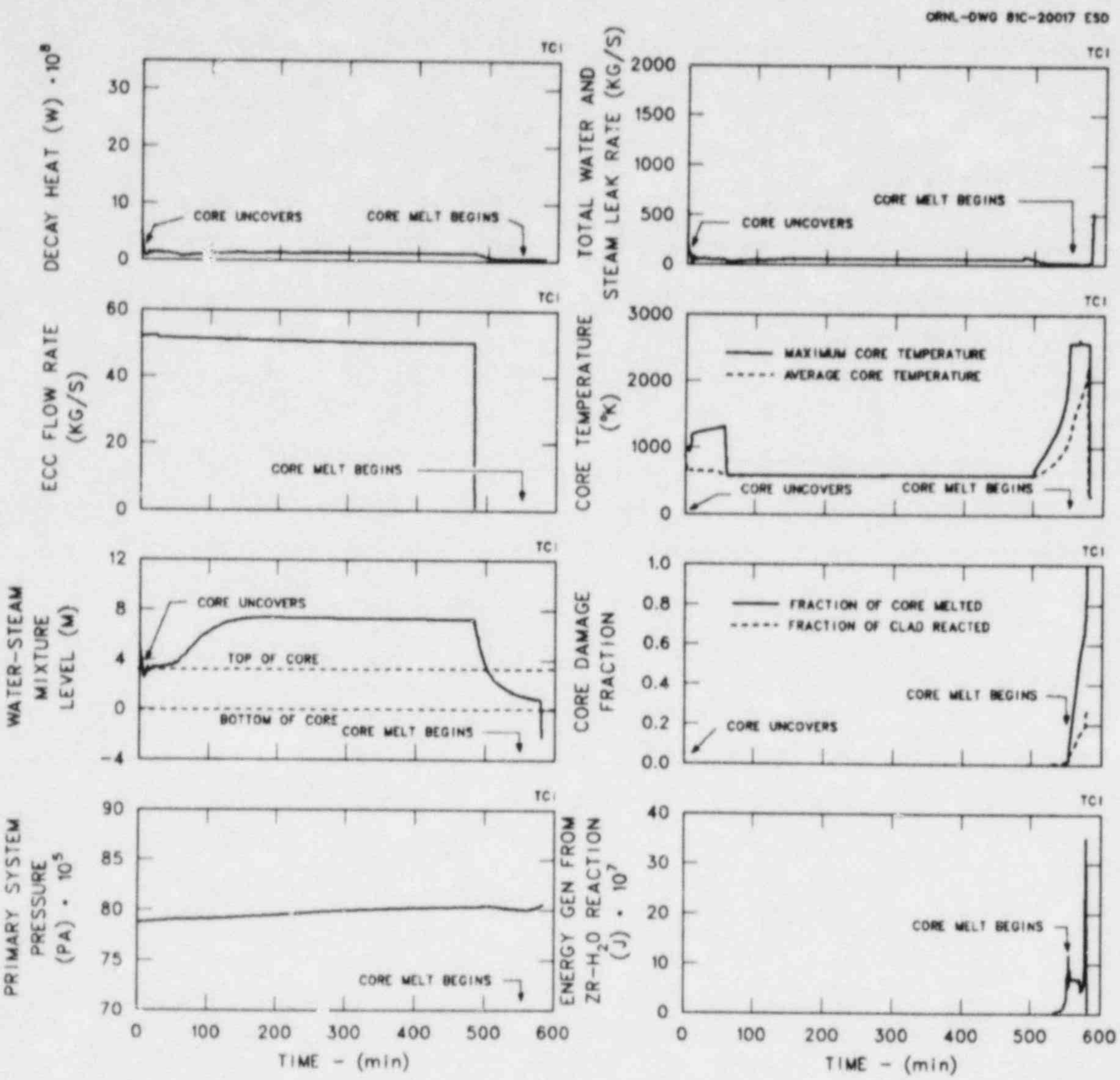

Fig. 4.7. Accident progression signatures for the TC1 sequence.

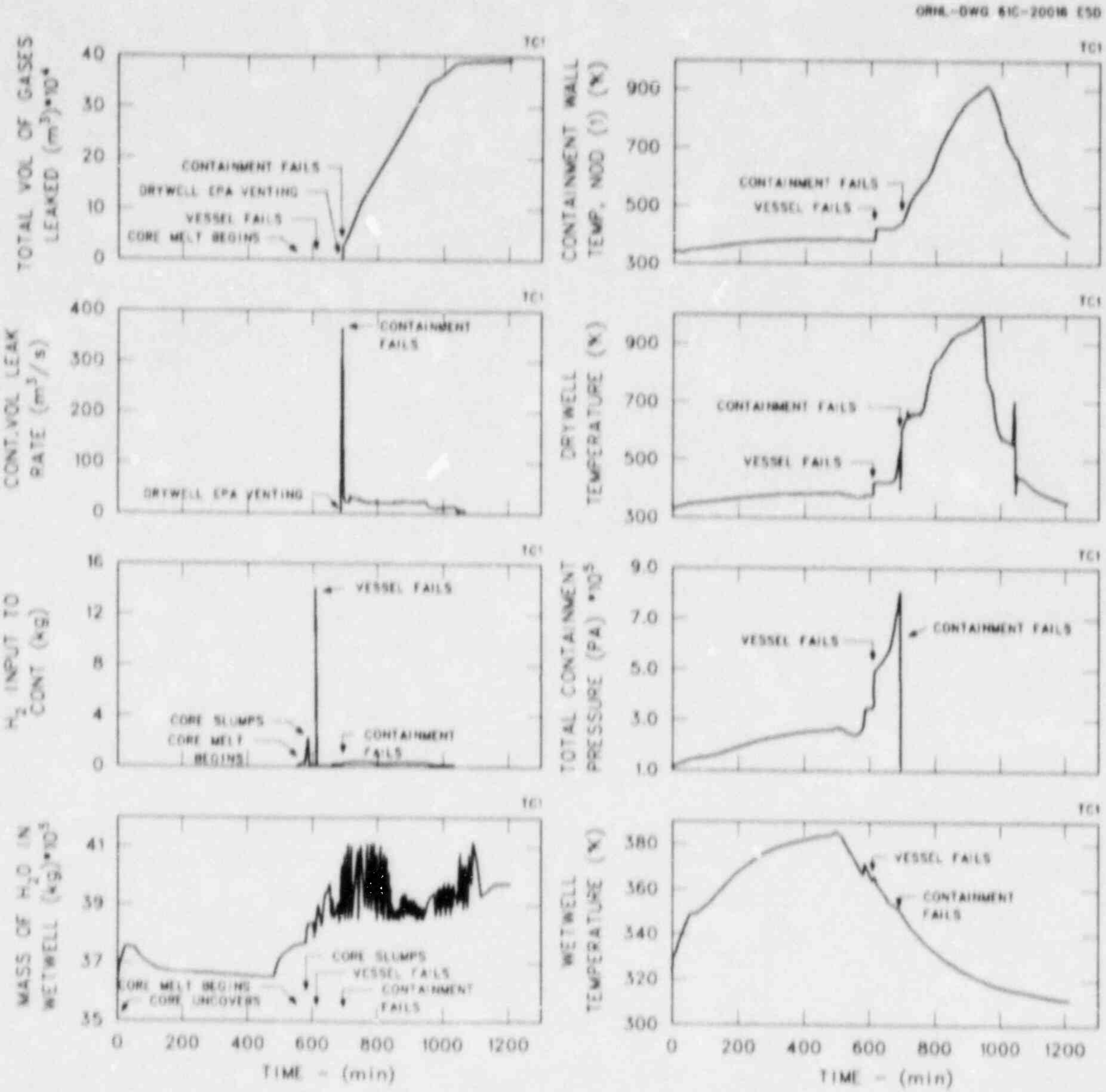

Fig. 4.8. Accident progression signatures for the TC1 sequence.

Table 4.4. Accident chronology for TC sequence

Time	Event
00:00	Trip of all feedwater pumps is initiated.
00:02	Turbine control valves start to close to regulate pressure.
00:03	Narrow-range (NR) sensed water level reached level 4. Reactor scram fails.
00:04	Feedwater flow drops below 20%. Recirculation flow runback to low end of auto-flow control is initiated to bring down the power level.
00:05	Feedwater flow decays to zero.
00:06	All containment isolation valves in Groups 2, 3, and 6 are initiated to close.
00:10	Wide-range sensed water level reaches Level 2. Recirculation pumps are tripped. The MSIVs start to close. HPCI/RCIC systems are initiated.
00:14	MSIVs are fully closed. Vessel pressure begins to rise, resulting in a reduction in void fraction and rapid increase in power.
00:18	Reactor power reaches a maximum of 572% of the initial value.
00:18	SRV's setpoints are actuated. Reactor power begins to decrease rapidly. Some fuel experiences transition boiling. Some Zr-clad and steam reactions take place.
00:19	Manual rod insertion is initiated.
00:23	Vessel pressure peaks at ~8.62 MPa (1250 psia).
00:32	All SRVs are closed.
00:35	SRVs actuate and then close: SRVs continue to cycle to release decay heat.
00:40	The operator has determined from the flux monitoring system that manual rod insertion has failed.
01:17	HPCI/RCIC flow starts entering the vessel.
02:14	SLCS starts.

Table 4.4 (continued)

Time	Event
03:15	Liquid control flow containing sodium pentaborate enters the vessel.
22:48	RHR systems start to function in the cooling mode.
30:15	Reactor is brought to hot shutdown.
31:00	Power level drops to decay power level. HPCI is secured. The operator switches to manual control of the RCIC to maintain water level. The drywell and wetwell temperatures have exceeded 341 K (154°F). The containment pressure is 0.18 MPa (26 psia).
01:43:00	The drywell and wetwell temperatures have cooled down to 335 K (143°F), and containment pressure is about 0.123 MPa (18 psia). Vessel water level has been maintained by ECCS makeup supplies. The four RHR pumps are operating in the cooling mode. Containment coolers are also operating.
41:43:00	The drywell and wetwell temperatures are about 324 K (124°F), and containment pressure is about 0.117 MPa (17 psia). There is no containment failure for this sequence.

Table 4.5. Timing of major events for TC sequence

Sequence	Timing of events (min)					
	Core uncovery	Start of core melt	Core slump	Failure of vessel head	Drywell EPA venting	Drywell failure
TC0	5	No core melt or containment breach				
TC1	5	553	583	611	686	692
TC2	3.6	150	Overflow	Overflow	Overflow	Overflow
TC3	4.9	118	Overflow	Overflow	Overflow	Overflow

4.3 TQUV Sequence

The TQUV sequence is concerned with failure to provide any ECCS makeup following an initiating event. The BWR event tree showing the TQUV sequence has been given in Fig. 4.1. A loss of all feedwater* has been chosen as the initiating event.[2] Because vessel depressurization has been shown to cause core uncovery and melting to occur sooner,[8,9] no vessel depressurization is used for this sequence.

Upon a loss of feedwater, vessel water level starts to decrease because of a mismatch between the coolant inventory loss in the form of steam and the supply of feedwater. As the vessel water level decreases to Level 4, the reactor is scrammed and runback of the recirculation pump is initiated. At this point, the control rods are automatically inserted into the core, terminating full-power operation.

Because there is no ECCS makeup flow, the vessel water level continues to decrease due to boiloff from stored heat and fission product decay. At the level 2 setpoint, the recirculation pumps are tripped and the MSIVs start to close. This isolates the reactor from the power conversion system. Soon afterwards, the vessel pressure reaches the SRV setpoints and excess vessel pressure is relieved by SRV steam discharges into the suppression pool.

Based on MARCH 1.4B calculations, with no HPCI, RCIC, LPCI mode of RHR, or core spray, the core would uncover at about 33 min and core melt would start at about 70 min. After the vessel bottom head has failed, the drywell EPA seals would start venting at about 3.25 h and fail shortly afterwards. The containment event tree after the core-melt is shown in Fig. 4.9. Timing of major events for the TQUV sequence is given in Table 4.6. Key results and accident progression signatures are presented in Figs. 4.10 and 4.11.

4.4 AE Sequence

The AE sequence is concerned with failure to provide sufficient emergency coolant injection (ECI) following a large-break LOCA. The event tree developed in the Reactor Safety Study[3] for the large LOCA initiator is presented in Fig. 4.12. The containment event tree for the AE sequence after core-melt has been given in Fig. 4.9.

The size of the pipe break defined in the Reactor Safety Study[3] ranges from the equivalent of a 0.1524-m-diam (6-in.) break† up to a double-ended rupture of the recirculation pipe of a 0.65-m (25.7-in.) equivalent diameter break‡ inside the primary containment. Table 4.7 gives the five cases studied for the AE sequence.

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*Control rod drive cooling water into the reactor vessel has been neglected in study of this sequence.

†Corresponding to an equivalent hole in the primary cooling system.

‡Corresponding to twice the equivalent hole in the primary cooling system.

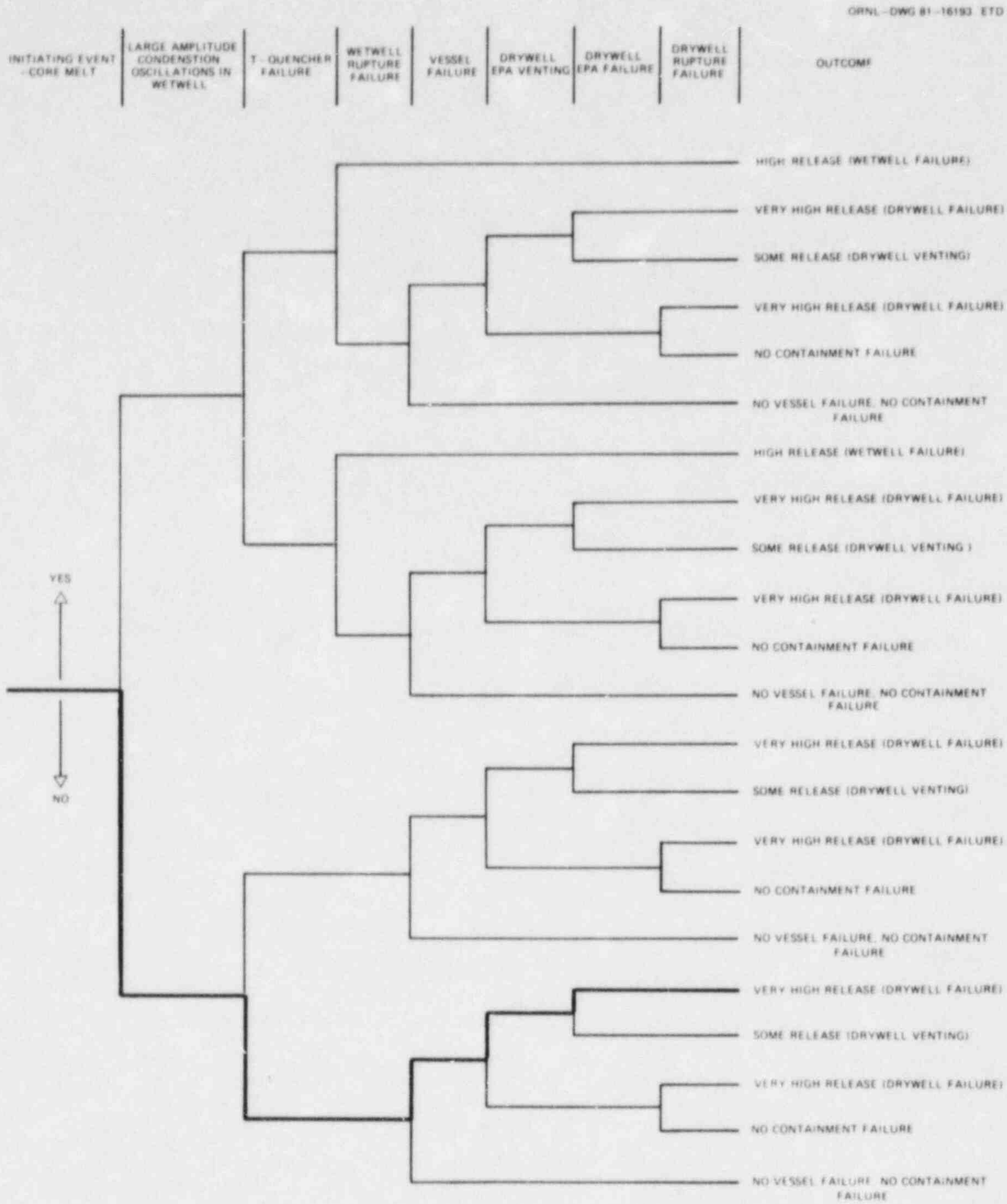

Fig. 4.9. Containment event tree following core melt.

Table 4.6. Accident chronology for TQUV sequence

Time	Event
00:00	Trip of all feedwater pumps is initiated.
00:01	Reactor power starts to drop.
00:02	Turbine control valves start to close to regulate pressure.
00:03	Narrow-range (NR) sensed water level reached Level 4.
00:04	Feedwater flow drops below 20%. Recirculation flow runback to low end of auto flow control is initiated.
00:05	Feedwater flow decays to zero.
00:07	NR sensed water level reaches Level 3. Reactor scram is initiated. All primary isolation valves in Groups 2, 3, and 6 are initiated to close. Automatic depressurization permissive.
00:10	Recirculation system cavitation protection interlock initiates recirculation flow runback to minimum pump speed, when the reactor power drops below ~25%.
00:15	Neutron flux drops below 1%.
00:30	Wide-range sensed water level reaches Level 2. Recirculation pumps are tripped. The MSIVs start to close. RCIC and HPCI automatic actuation signals are initiated but are inoperative. Operator attempts to restore the FW, RCIC, and HPCI systems without success and also finds the LPCI inoperative.
00:38	SRV's setpoints are actuated.
00:56	All SRVs are closed. SRVs cycle to release decay heat.
33:00	Core uncovers.
1:10:00	Core melting starts.
1:37:00	Bottom grid fails, and temperature of structures in bottom head is above water temperature.
1:39:00	The corium slumps to vessel bottom.

Table 4.6 (continued)

Time	Event
1:41:00	The debris is starting to melt through the bottom head. The drywell and wetwell temperatures have exceeded 370 K (206°F) and 344 K (160°F), respectively. The local pool temperature surrounding the SRV T-quencher discharging bay has exceeded 422 K (300°F). Steam condensation oscillations could increase in magnitude due to loss of condensation effectiveness, resulting in wetwell rupture failure.
2:09:00	Vessel bottom head fails, resulting in a pressure spike of 0.34 MPa. The debris, at a temperature over 1820 K (2816°F), is starting to attack the concrete floor.
3:10:00	Drywell electric penetration assembly seals start to vent as ambient temperature has exceeded 477 K (400°F).
3:13:00	Drywell electric penetration assembly seals are blown out as temperature has exceeded 533 K (500°F), resulting in containment failures.

Table 4.7. AE sequence

Sequence	ABRK[a] (ft^2)	W(1)[b] (lb/min)	W(2)[c] (lb/min)	EW(1)[d] (Btu/lb)	EW(2)[e] (Btu/lb)	WLH1[f] (gpm)
AE0	7.2048	1.729E06	1.2087E06	521.8	633.35	0
AE1	5.0367	1.2087E06	0.96696E06	370.0	580.0	0
AE2	5.0367	1.2087E06	0.96696E06	370.0	580.0	10000
AE3[g]	0.19635	N/A	N/A	N/A	N/A	10000
AE4[h]	5.0367	1.2087E06	0.96696E06	370.0	580.0	0

[a]ABRK = area of pipe break.

[b]W(1) = mass flow rate at time 0.

[c]W(2) = mass flow rate at 30 s.

[d]EW(1) = specific enthalpy at time 0.

[e]EW(2) = specific enthalpy at 30 s.

[f]WLH1 = LPECCS flow rate.

[g]Small-break LOCA option (ITRAN = 1).

[h]Containment failure by overpressure at 174.7 psia (WASH 1400[3]).

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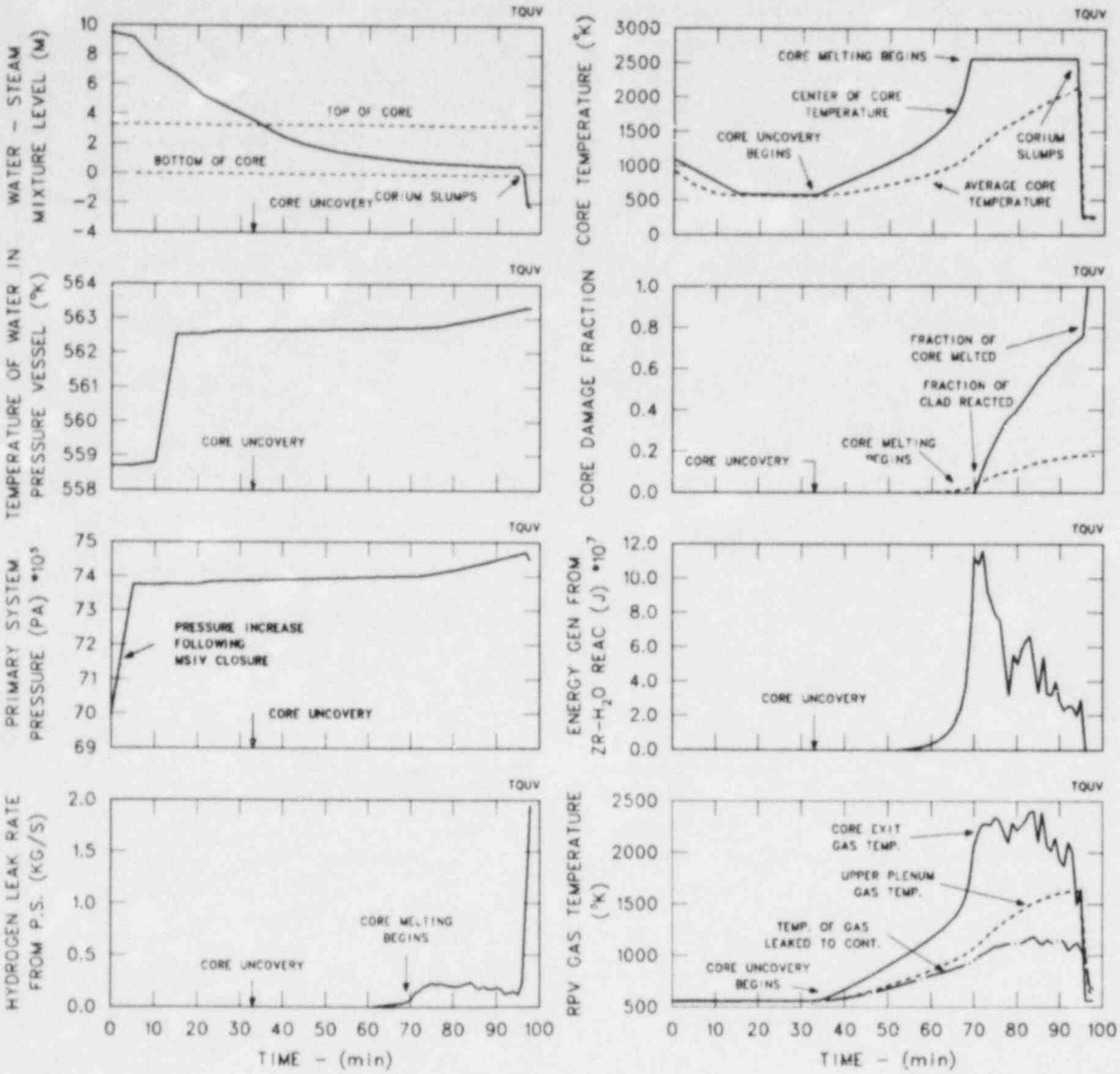

Fig. 4.10. Accident progression signatures for the TQUV sequence.

ORNL-DWG 81-13760 ESD

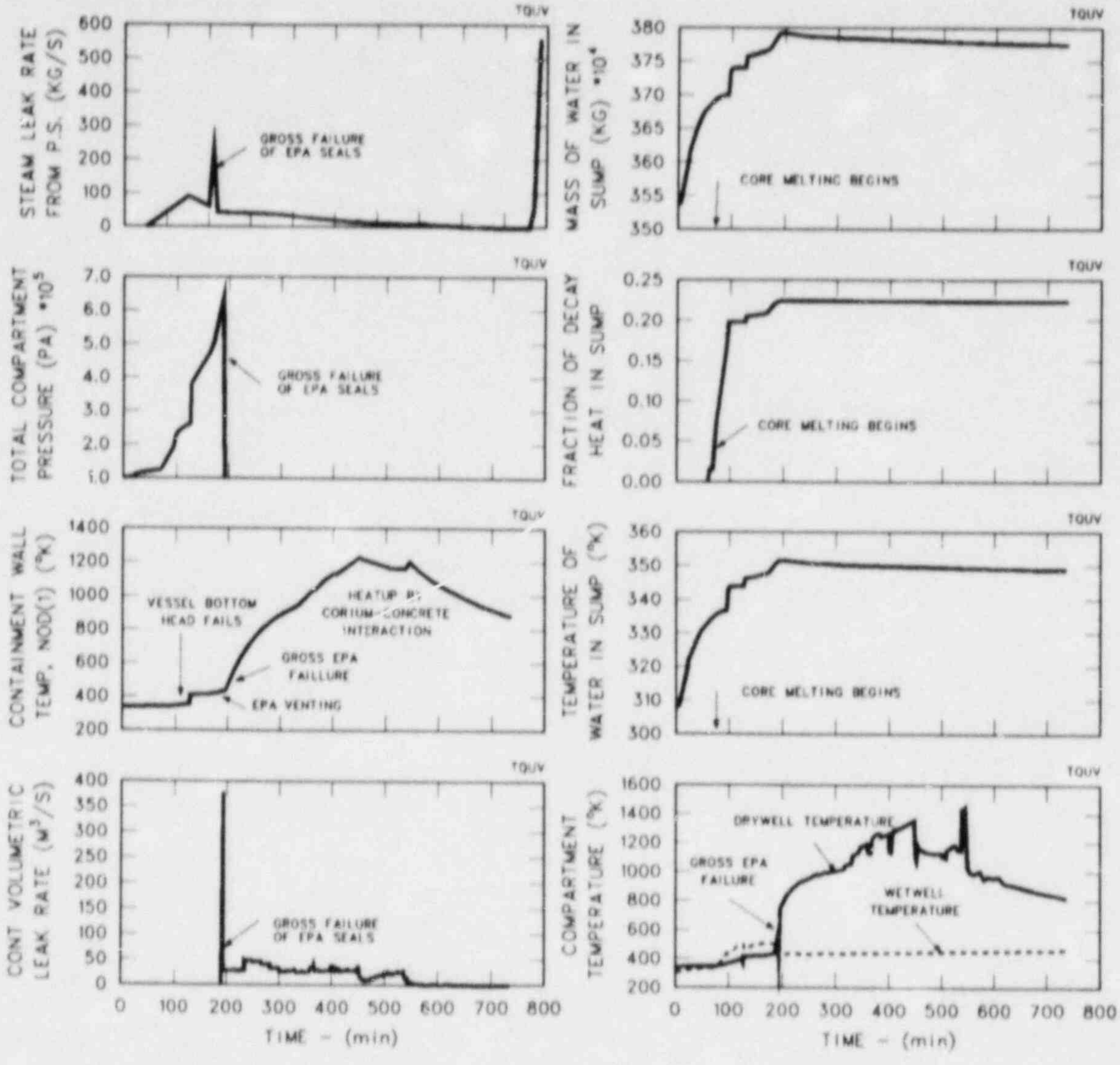

Fig. 4.11. Accident progression signatures for the TQUV sequence.

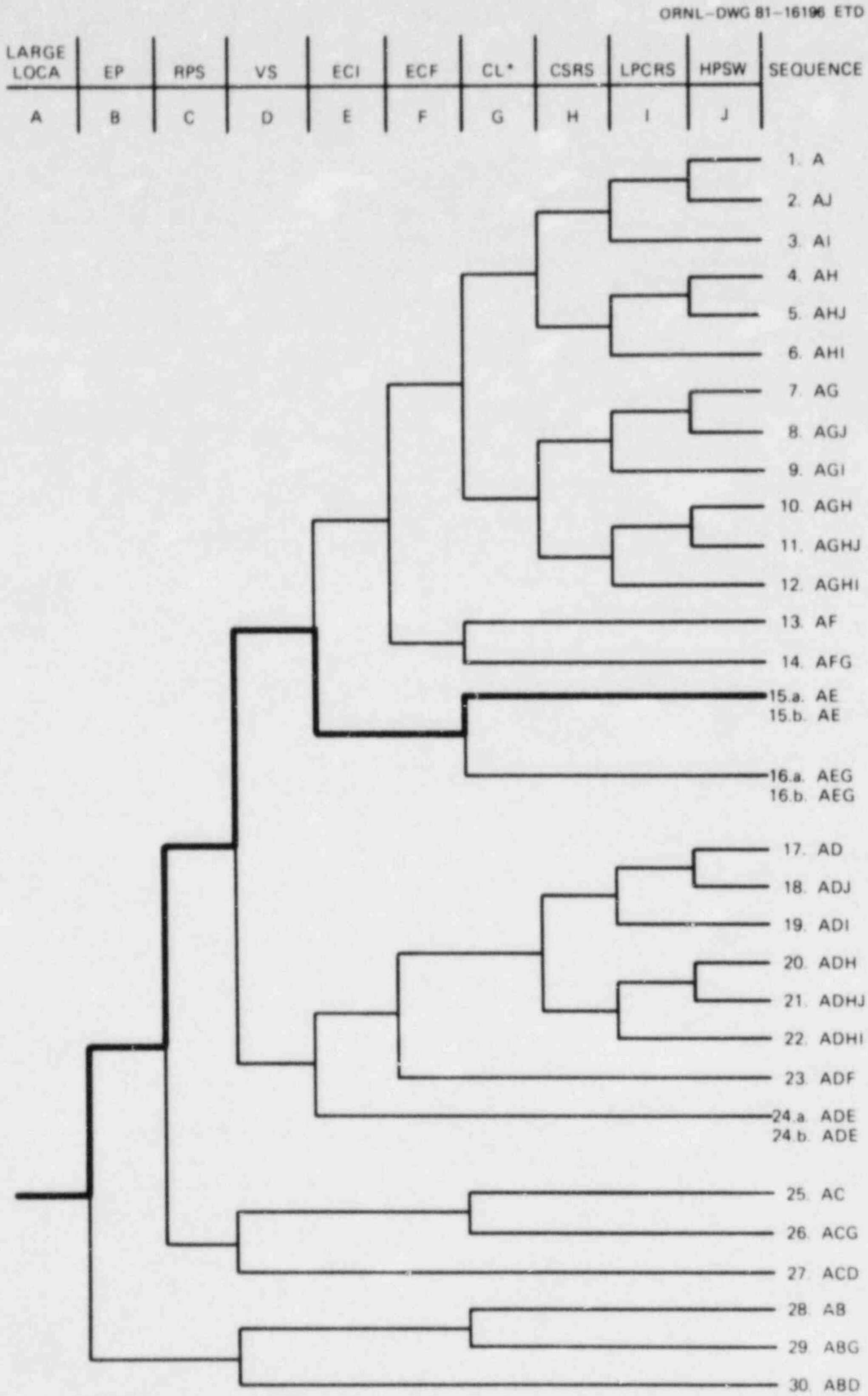

Fig. 4.12. Event tree for large LOCA (AE sequence).

For all AE sequences except AE3, a double-ended rupture of the recirculation pipe of 0.65-m (25.7-in.) equivalent diameter is postulated to occur inside the primary containment. Due to the effect of the jet pump nozzles, the total effective break flow area would be smaller than the actual pipe cross-sectional area. However, this difference in the break flow areas does not significantly affect the final results as compared in AE0 and AE1. In sequence AE0, the actual pipe cross-sectional area is used, whereas effective flow areas are used in all other AE sequences. In sequence AE3, a break size of 0.1524-m (6-in.) equivalent diameter corresponding to a flow area of 0.018 m^2 (0.196 ft^2) is used.

The large LOCA option has been adopted in all AE sequences except for AE3, in which the small LOCA option is used. In addition, while containment failure is assumed to be caused by overtemperature in the drywell EPA seals, failure by overpressure is given in sequence AE4 to provide a comparison with those predicted in the Reactor Safety Study.[3]

Immediately after the postulated large-break LOCA, the core inlet flow rapidly decreases from 100% to ~30% of rated; this is accompanied by a rapid decrease of the vessel pressure and a sudden increase in the core void fraction, which would be sufficient to render the core subcritical. In addition, a scram signal would be initiated almost immediately on high drywell pressure. An additional scram signal would also be initiated at Level 4 when it occurs.

Furthermore, the MSIVs would start to close at low vessel pressure and low water level (Level 2) after the break. All primary containment isolation valves would also start to close. The recirculation pumps would be tripped, and the emergency diesel generators would be signaled to start.

Shortly afterwards, the liquid inventory in the downcomer and the separator region of the vessel would be depleted. At this point, the flow out of the break areas would consist of two-phase flow, resulting in a large increase in the vessel depressurizing rate.

As the vessel water level and pressure continue to decrease following the break, both the LPCIS and the CSIS would be actuated as the level drops below Level 1. The LPCIS and the CSIS are assumed to have failed for sequences AE0, AE1, and AE4. Only one LPECCS pump is assumed to operate in sequences AE2 and AE3, delivering makeup at about 2271 m^3/h (10,000 gpm).

In sequences AE0 and AE1 that have no makeup flows, the core would soon uncover and melting would begin shortly afterwards. No significant difference, however, has been found between sequences AE0 and AE1 due to the difference in assumed break flow area.

Some inconsistent results have been found in comparing between sequences AE2 and AE3, both of which assume the operation of one LP ECCS pump. The AE2 sequence in which the large LOCA option has been used does not result in core melt or containment breach. In contrast, sequence AE3 which involves a smaller break size and in which the small LOCA option has been used results in core melting. Based on MARCH calculations, it has been found for sequence AE3 that about 28% of the core has melted at about 11 h into the accident. This difference is attributed to limitations associated with MARCH initiation stage calculations using the large LOCA option. While the primary system responses following the break for the small LOCA option are calculated independently in the PRIMP subroutine, the primary system responses are calculated in two stages for the large LOCA option. First, the initial blowdown characteristics are provided as input to the subroutine INITIAL. Following the blowdown phase, the primary system responses are calculated in the subroutine BOIL. BOIL input parameters that may change during the initiation stage include HO, PVSL, TCAV, TFEOO, TGOO, TT(I), VOLS, WDED, and WATBH. These parameters after the blowdown phase are best provided by another thermal-hydraulic code, for example, RELAP5/MOD1 or TRAC-BD1. In the absence of computations from such thermal-hydraulic codes, the best-estimate values used as BOIL input parameters after the blowdown phase may result in nonconservative predictions in sequence AE2.

Because a smaller break size in AE3 resulted in core melt, the much larger break LOCA in AE2 should also result in core melt, all other conditions being the same. On this premise, it can be concluded that the use of one LPECCS pump would not keep the core from melting in a double-ended recirculation pipe break LOCA.

Sequence AE4 is identical to AE1 except that containment failure was assumed to be caused by overpressurization as predicted in the Reactor Safety Study.[3] It is noted that containment failure would be delayed from 17 to 183 min if overpressurization was the dominant failure mode rather than EPA seal failure by overtemperature.

The accident time history for the AE0 sequence is presented in Table 4.8. Key results and accident progression signatures based on MARCH calculations for the AE0 sequence are presented in Figs. 4.13 and 4.14. Selected containment responses for the AE4 sequence are presented in Figs. 4.15–4.19. Comparisons of timing of major events for the five AE sequences are given in Table 4.9.

4.5 S_1E Sequence

The S_1E sequence is concerned with a small break in the Reactor Coolant System (RCS). Unlike the AE sequence discussed in Sect. 4.4, the S_1E sequence involves breaks that are not large enough to depressurize the system so that the low-pressure ECCS can be used; the breaks are, however, sufficiently large so that the RCIC and CRD hydraulic supply system cannot adequately replenish the fluid lost through the break. Both the HPCI and LPECCS are assumed to be unavailable for this sequence so that a core-melt accident would result. The event tree developed in the Reactor Safety Study[3] for a small LOCA (S_1) is presented in Fig. 4.20. Containment event tree for the S_1E sequence after core uncovery has been given in Fig. 4.9.

For liquid pipeline breaks, the S_1E sequence encompasses failure with an equivalent break diameter* between about 0.0635 m (2.5 in.) and 0.216 m (8.5 in.). Table 4.10 gives the seven cases studied for the S_1E sequence. These seven cases deal with a spectrum of break sizes ranging from 0.0762-m (3-in.) to 0.1524-m (6-in.) equivalent break diameters, the effect of makeup from feedwater pumps for 15 min, vessel depressurization by one SRV, and different containment failure areas.

*Corresponding to an equivalent hole in the primary cooling system.

Table 4.8. Accident chronology for AE sequence

Time	Event
00:00	A double-ended liquid recirculation line of 0.65-m (25.7-in.) equivalent break diameter, located inside the drywell, breaks at an elevation of 2.54 m (8.33 ft) above bottom of core. Initial flow rate = 1.307 x 10^4 kg/s (1.037 x 10^8 lb/h). Initial specific enthalpy = 1.2134 x 10^6 J/kg (521.80 Btu/lb).
00:00	The reactor goes subcritical due to void formation in the core region.
00:00+	Reactor scrams upon receipt of the high drywell pressure and low water level signals. Control rod motion begins. Drywell temperature has exceeded 366 K (199°F). MSIVs begin to close. All containment isolation valves are initiated to close. Recirculation pumps are tripped and begin to slow down. Emergency diesels are signaled to start. All low-pressure ECCS have failed.
00:03	Blowdown rate is ~1.268 x 10^4 kg/s (~10 lb/h), and specific enthalpy is ~1.240 x 10^6 J/kg (533.23 Btu/lb). Control rods approach full-in position, reactor is subcritical. Drywell and wetwell temperatures have exceeded 385 K (233°F) and 310 K (98°F), respectively.
00:04	MSIVs are fully closed, isolating reactor system. All control rods are fully inserted. Recirculation pump flow has fallen to ~40% of rated. Drywell and wetwell temperatures have exceeded 386 K (233°F) and 311 K (98°F), respectively.
00:06	Blowdown rate is ~1.228 x 10^4 kg/s (9.72 x 10^7 lb/h), specific enthalpy is ~1.266 x 10^6 J/kg (544.4 Btu/lb).
00:30	Blowdown rate has slowed down to ~9.138 x 10^3 kg/s (7.24 x 10^7 lb/h).
01:30	Core uncovers.
04:30	Fission products begin to be released into the containment.
13:30	Core melting starts.
15:30	Drywell EPA seals start venting as the drywell temperature has exceeded 477 K (400°F) at a pressure of ~0.18 MPa (26 psia).
17:06	Drywell EPA seals have failed as the drywell temperature has exceeded 533 K (500°F), resulting in containment failure.
39:33	Core slumps to vessel bottom.
57:33	Vessel bottom head fails. The debris, at a temperature over 2550 K (4130°F) starts to attack the concrete basemat.

ORNL-DWG 81C-20015 ESD

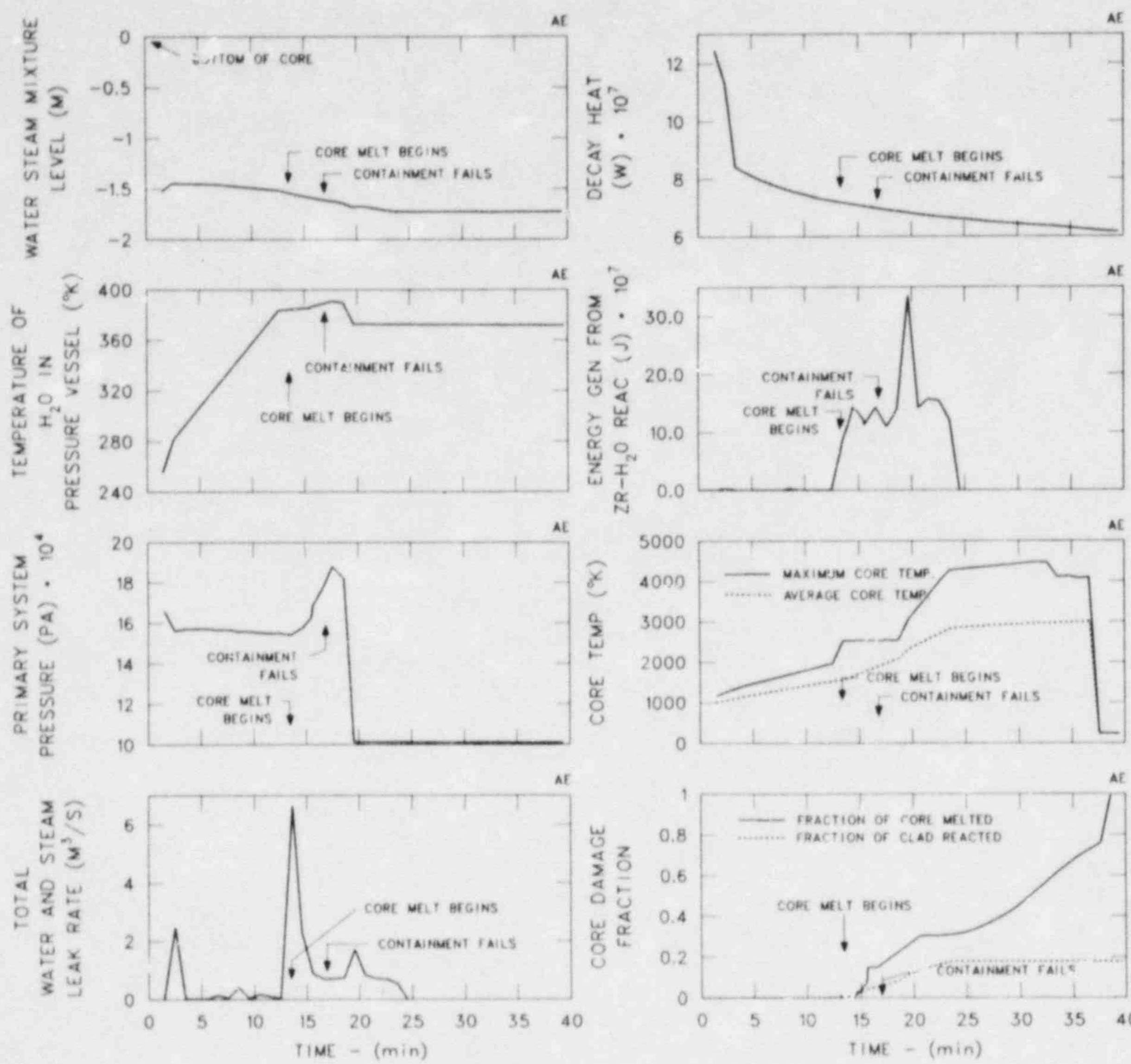

Fig. 4.13. Accident progression signatures for the AE sequence.

ORNL-DWG 81C-20033 ESD

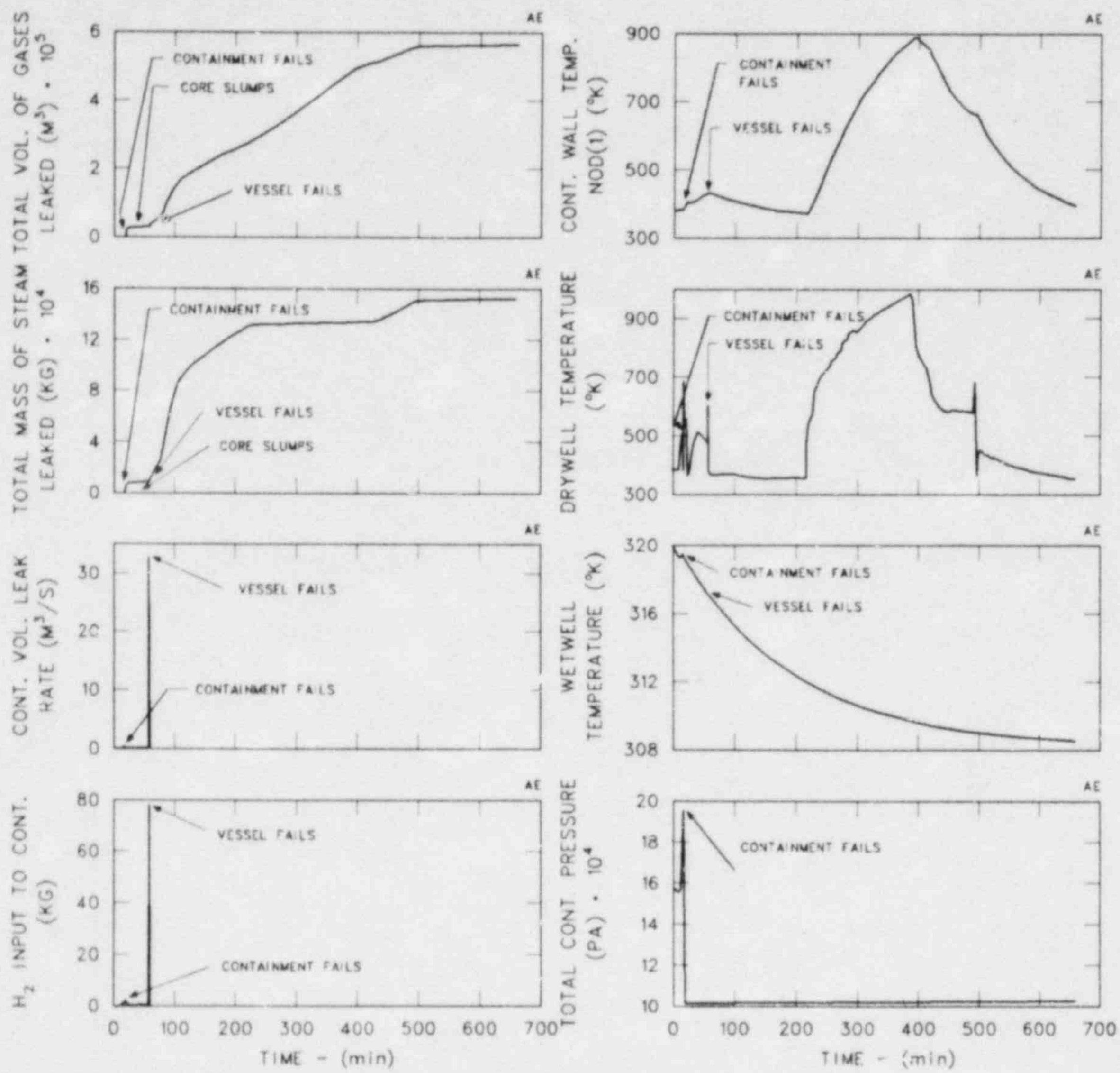

Fig. 4.14. Accident progression signatures for the AE sequence.

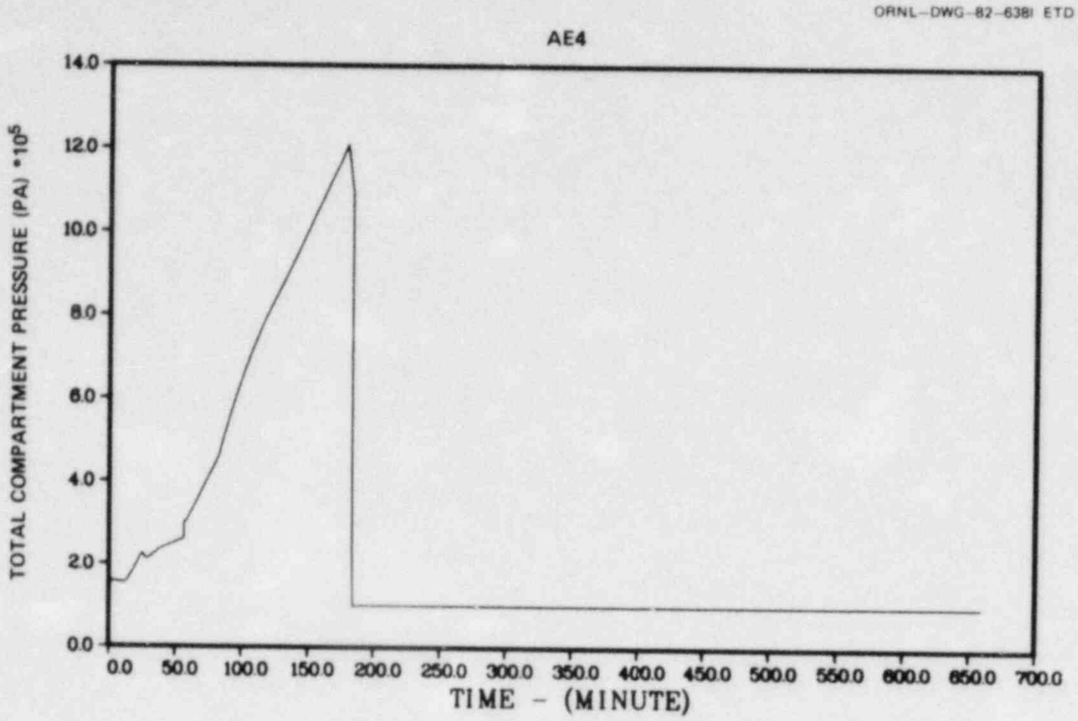

Fig. 4.15. Containment pressure response for AE4.

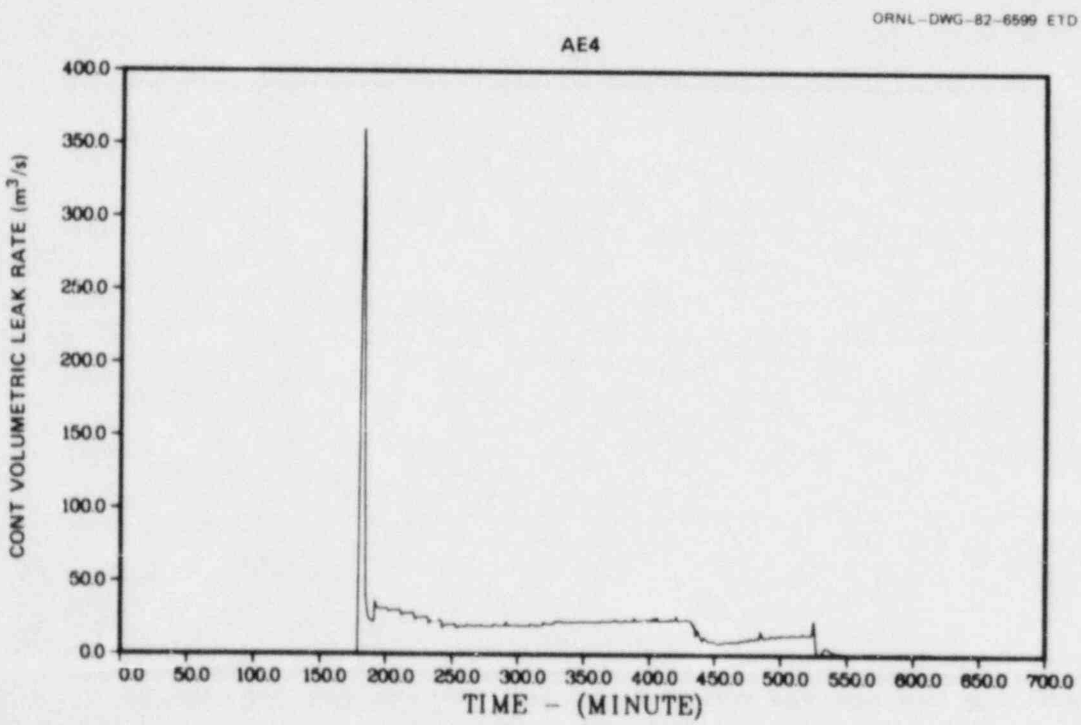

Fig. 4.16. Containment volumetric leak rate for AE4.

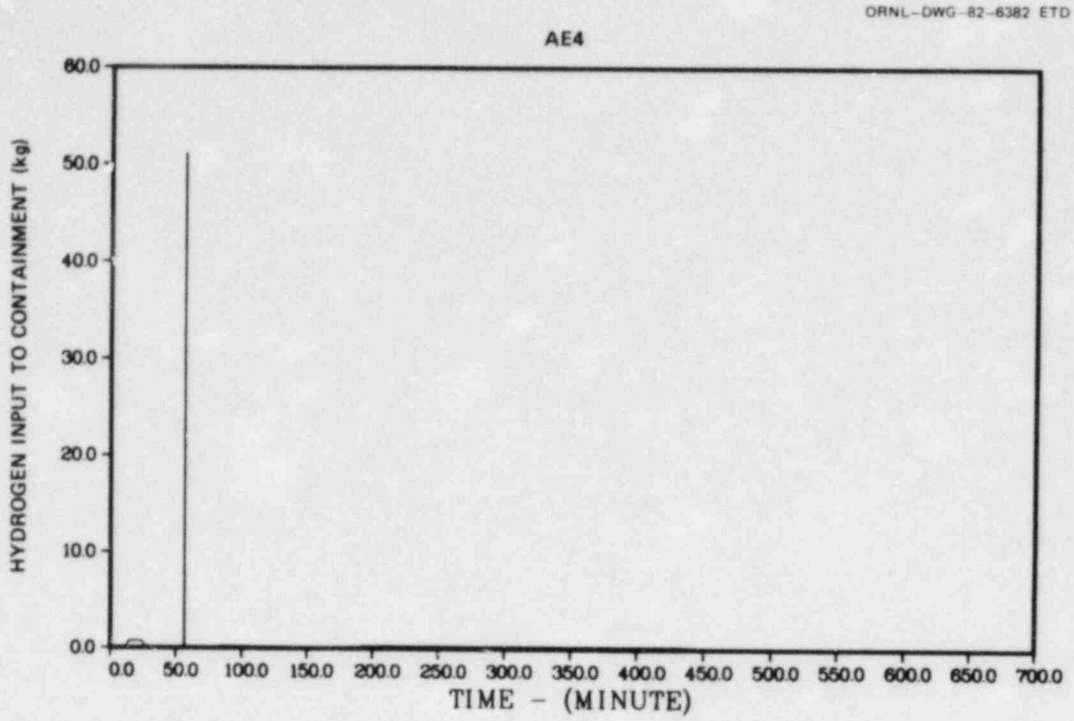

Fig. 4.17. Hydrogen input to containment for AE4.

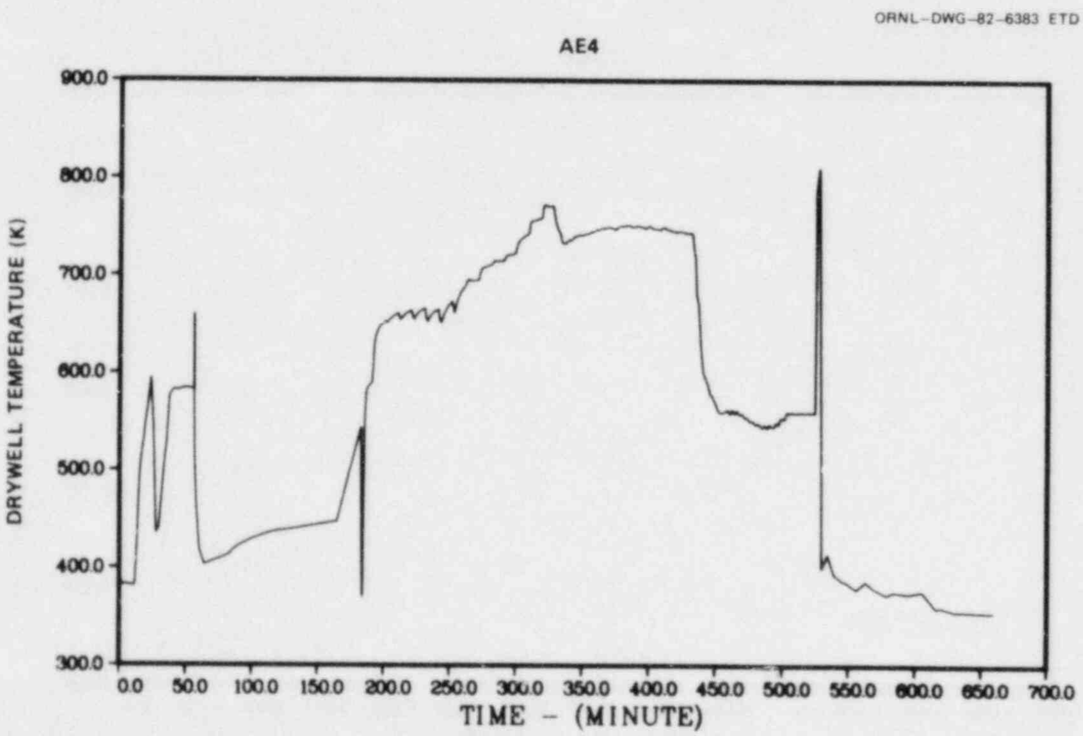

Fig. 4.18. Drywell temperature response for AE4.

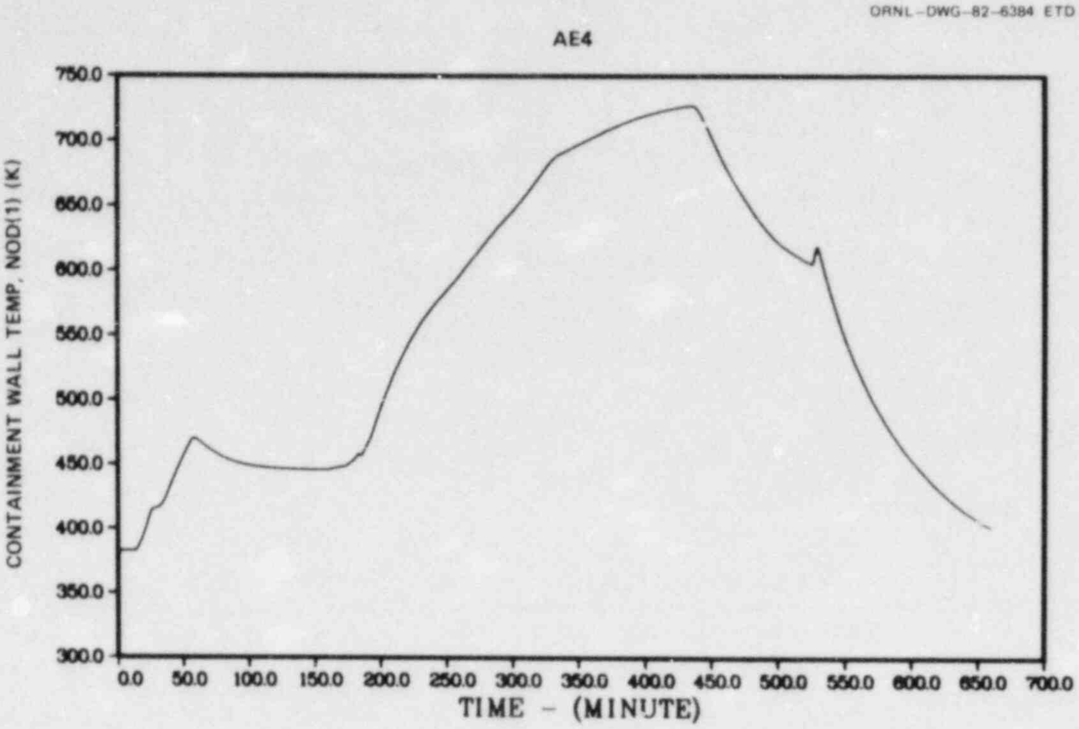

Fig. 4.19. Drywell EPA temperature response for AE4.

Table 4.9. Timing of major events for AE sequence

Sequence	Timing of events (min)					
	Core uncovery	Start of core melt	Core slump	Failure of vessel head	Drywell EPA venting	Drywell failure
AE0	1.5	13.5	39.5	57.5	15.5	17.1
AE1	1.5	13.5	39.5	57.5	15.5	17.5
AE2	1.5	No core melt or containment breach				
AE3	4	43.5	*a*	*a*	*a*	*a*
AE4[b]	1.5	13.5	39.5	57.5	N/A	183

[a]No core slump or containment failure at 11 h.

[b]Containment failure by overpressure at 174.7 psia (WASH-1400[3]).

ORNL-DWG 81-16194 ETD

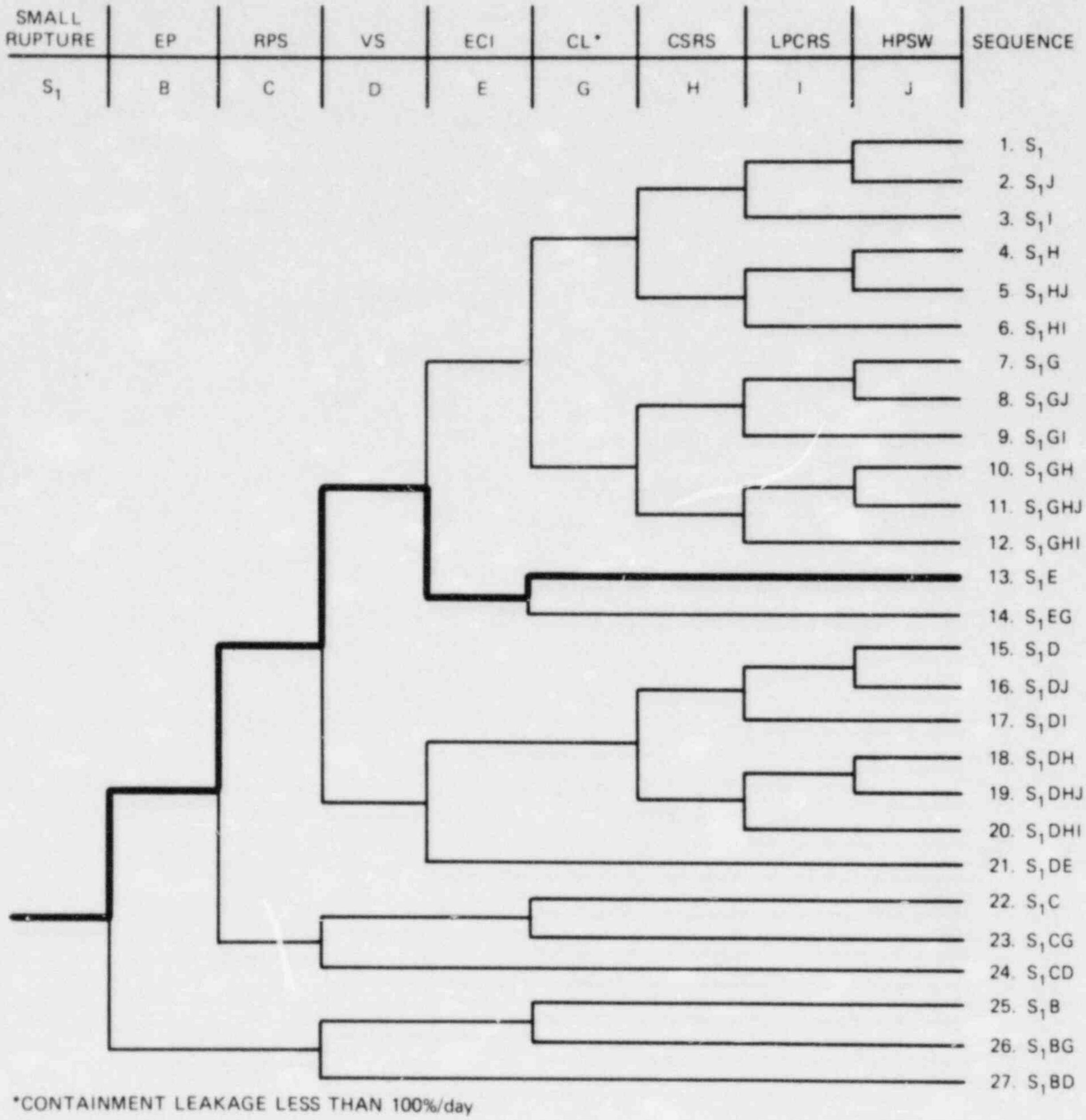

Fig. 4.20. Event tree for the S_1E sequence.

Table 4.10. S_1E sequence

Sequence	ABRK[a] (ft²)	STPLH[b] (min)	AB(1)[c] (ft²)	TB(1)[d] (min)	CSRV[e]	C3(5)[f] (ft²)
S1E0	0.0699	0.0	0.0	1.0E6	3719.06	20.97
S1E1	0.0699	15.0	0.0	1.0E6	3719.06	20.97
S1E2	0.0699	0.0	0.1583	0.0	3432.98	20.97
S1E3	0.19635	0.0	0.0	1.0E6	3719.06	20.97
S1E4	0.13635	0.0	0.0	1.0E6	3719.06	20.97
S1E5	0.0490875	0.0	0.0	1.0E6	3719.06	20.97
S1E6	0.0699	0.0	0.0	1.0E6	3719.06	1.0485

[a] ABRK = pipe break area.

[b] STPLH = stop time for the feedwater pumps.

[c] AB(1) = SRV flow area.

[d] TB(1) = time to actuate SRV.

[e] CSRV = SRV coefficient.

[f] C3(5) = containment failure area.

Immediately after the postulated small LOCA, the vessel pressure and water level would decrease, with a corresponding increase in the drywell pressure. The increase of drywell pressure would initiate a reactor scram and closure of all primary containment isolation valves in Groups 2, 6, and 8. Both the HPCI and the low-pressure ECCS are assumed unavailable for the S_1E sequence.

Before those signals are actuated, the feedwater system would increase its flow to maintain water level within the normal range. However, this makeup flow would soon stop when the water level rises momentarily above Level 8 due to fluid oscillations upon a sudden vessel depressurization.* This would cause the main turbine and feedwater pumps to trip. Thus, feedwater pumps are assumed to be unavailable for the S_1E sequence. The effect of the feedwater makeup flows on the accident progression is, nevertheless, investigated in the S_1E1 sequence, in which the feedwater pumps are assumed to operate for 15 min until stopped either by operator action or upon isolation when the vessel pressure drops below 5.86 MPa (850 psia).

*This level rise is based on MARCH calculations, which might differ from other detailed thermal-hydraulic code calculations.

Because the RCIC and CRD systems are insufficient to replenish the coolant being lost,* the vessel water level would continue decreasing until eventually the core is uncovered and melting starts.

The accident time history for the S_1E0 sequence is presented in Table 4.11 and key results and accident progression signatures are presented in Figs. 4.21 and 4.22.

The effect of vessel depressurization by opening one SRV is examined in the S_1E2 sequence. No significant effect has been found on the accident progression. A variation of break sizes is examined in sequences S_1E3 through S_1E5. Timings for all major events are affected by the break sizes. Finally, the effect of containment failure areas is examined in the S_1E6 sequence. No significant effect has been found on the accident progression due to different containment failure areas based on MARCH calculations.

Selected accident progression signatures for the S_1E1, S_1E3, and S_1E5 sequences are presented in Figs. 4.23–4.28. Comparisons of timing of major events for the seven S_1E sequences are given in Table 4.12.

4.6 S_2E Sequence

The S_2E sequence is defined in the Reactor Safety Study[3] as being initiated by a break small enough that the vessel water level can be maintained by the RCIC operation alone. Thus, in addition to the unavailability of HPCI and LPECCS as in the S_1E sequence, the RCIC is also assumed to be unavailable. The coolant makeup is provided only by the CRD hydraulic system, which is insufficient to maintain the vessel level.† Eventually the core would uncover and melting would start. The event tree developed in the Reactor Safety Study[3] for a small LOCA (S_2) is presented in Fig. 4.29. Containment event tree for the S_2E sequence after core uncovery has been shown in Fig. 4.9.

Two sizes for liquid pipeline breaks of 0.0254-m (1-in.) and 0.0508-m (2-in.) equivalent break diameters† have been investigated in this work for the S_2E sequence. Six cases of the S_2E sequence studied in this work are given in Table 4.13. These six cases have dealt with two break sizes, different duration of the feedwater pump operation, effect of vessel depressurization by one SRV, and different containment failure areas.

Immediately after the postulated small LOCA, the vessel pressure and water level would decrease, with a corresponding increase in the drywell

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*The MARCH pump performance curve option was used for RCIC and CRD pumps; therefore ECC flow as shown in Fig. 4.21 varies from ~6 to 28 kg/s. Actual combined flow for Browns Ferry would be ~4 kg/s for 1 to 1 1/2 min, then constant at ~41.5 kg/s for duration of accident.

†The MARCH pump performance curve option was used for the CRD pump; therefore, the ECC flow as shown in Fig. 4.30 is constant at ~0.5 kg/s due to continued high primary system pressure. The actual flow for Browns Ferry would be constant at about 4 kg/s due to flow control valves.

†Corresponding to an equivalent hole in the primary cooling system.

Table 4.11. Accident chronology for S_1E sequence

Time	Event
00:00	A liquid pipeline of 0.09-m (3.58-in.) equivalent break diameter breaks inside the drywell.
00:03	Reactor scrams upon high drywell pressure. All primary containment isolation valves in Groups 2, 6, and 8 are initiated to close. HPCI pumps are not available.
00:04	Main turbines and feedwater pumps are tripped on high vessel water level at Level 8 caused by fluid oscillations due to a sudden pressure drop after the break.
00:30	SRV's setpoints are actuated. SRVs continue to cycle on setpoints to relieve excess vessel pressure.
01:00	RCIC pump is actuated.
01:30	RCIC makeup flow enters the vessel.
07:44	Core uncovers.
35:00	Drywell EPA seals begin to vent as temperature has exceeded 477 K (400°F) at a containment pressure of ~0.164 MPa (24 psia).
40:00	Drywell EPA seals have failed as temperature has exceeded 533 K (500°F) at a pressure of ~0.187 MPa (27 psia).
41:00	Fission products begin to be released into the containment.
44:13	Core melting starts.
01:13:23	Corium slumps to vessel bottom. Debris is starting to melt through the bottom head.
02:07:45	Vessel bottom head fails. Debris, at a temperature over 2076 K (3277°F), is starting to boil water from containment floor and to attack the concrete basemat.

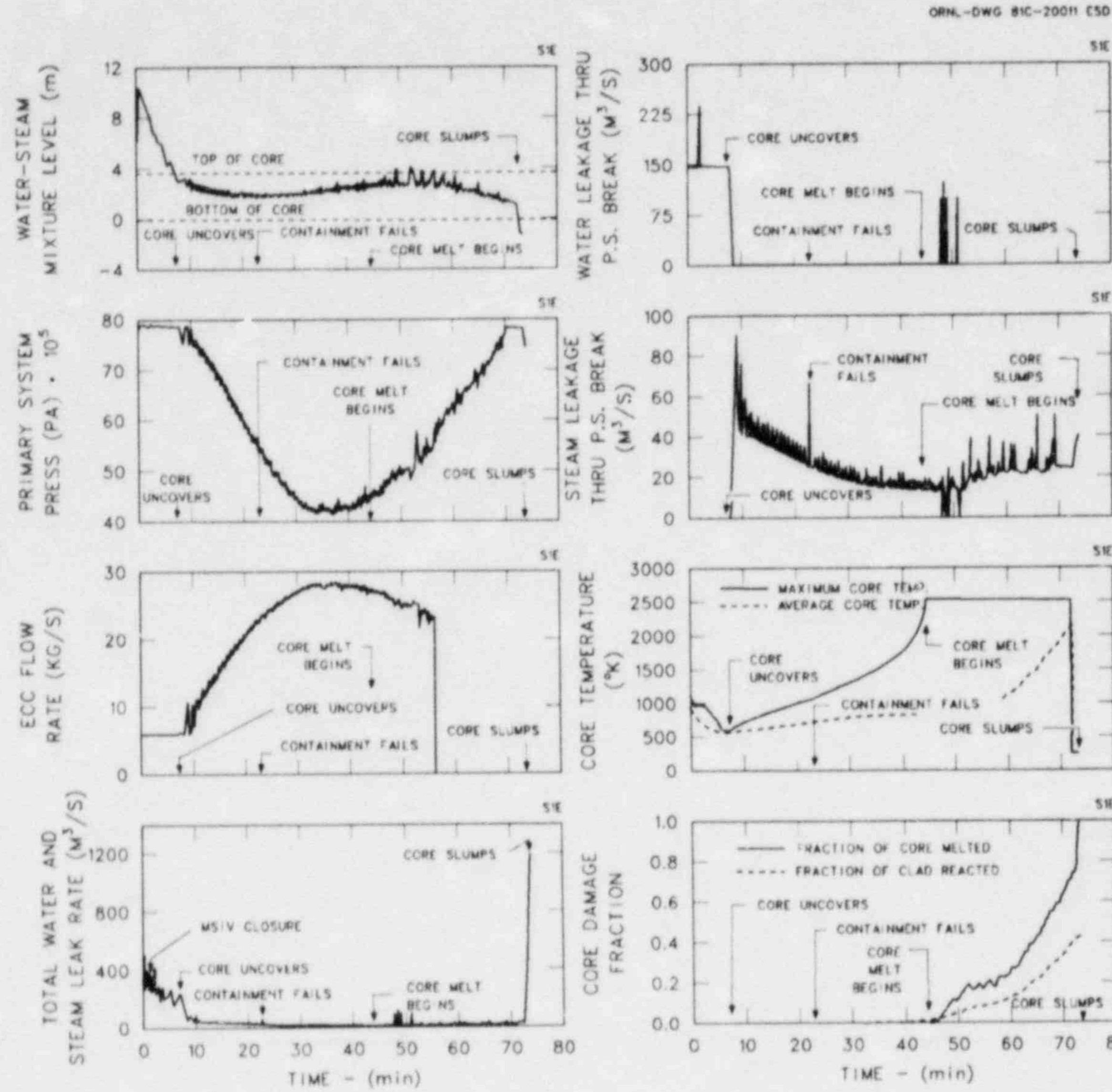

Fig. 4.21. Accident progression signatures for the S1E sequence.

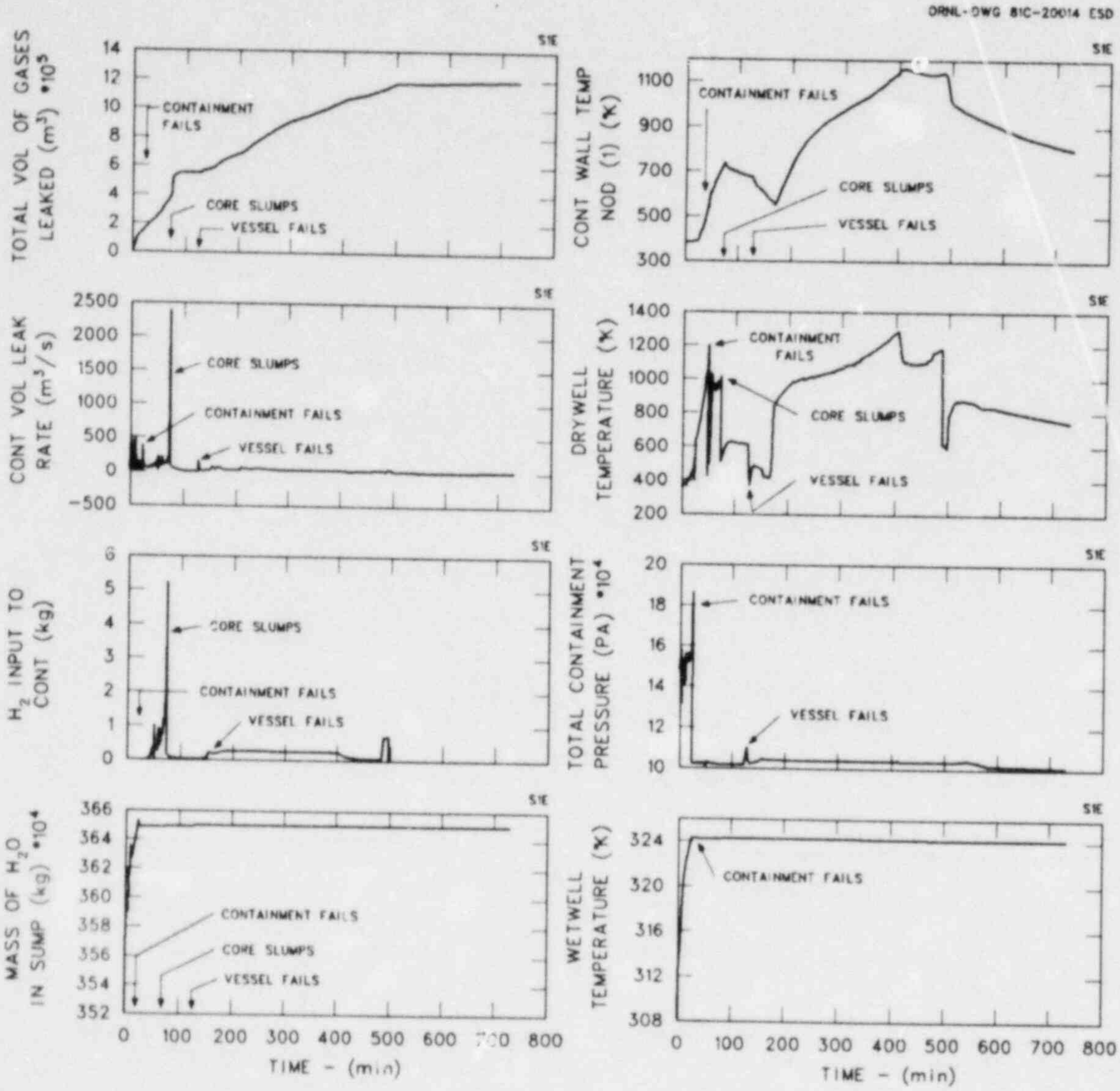

Fig. 4.22. Accident progression signatures for the S1E sequence.

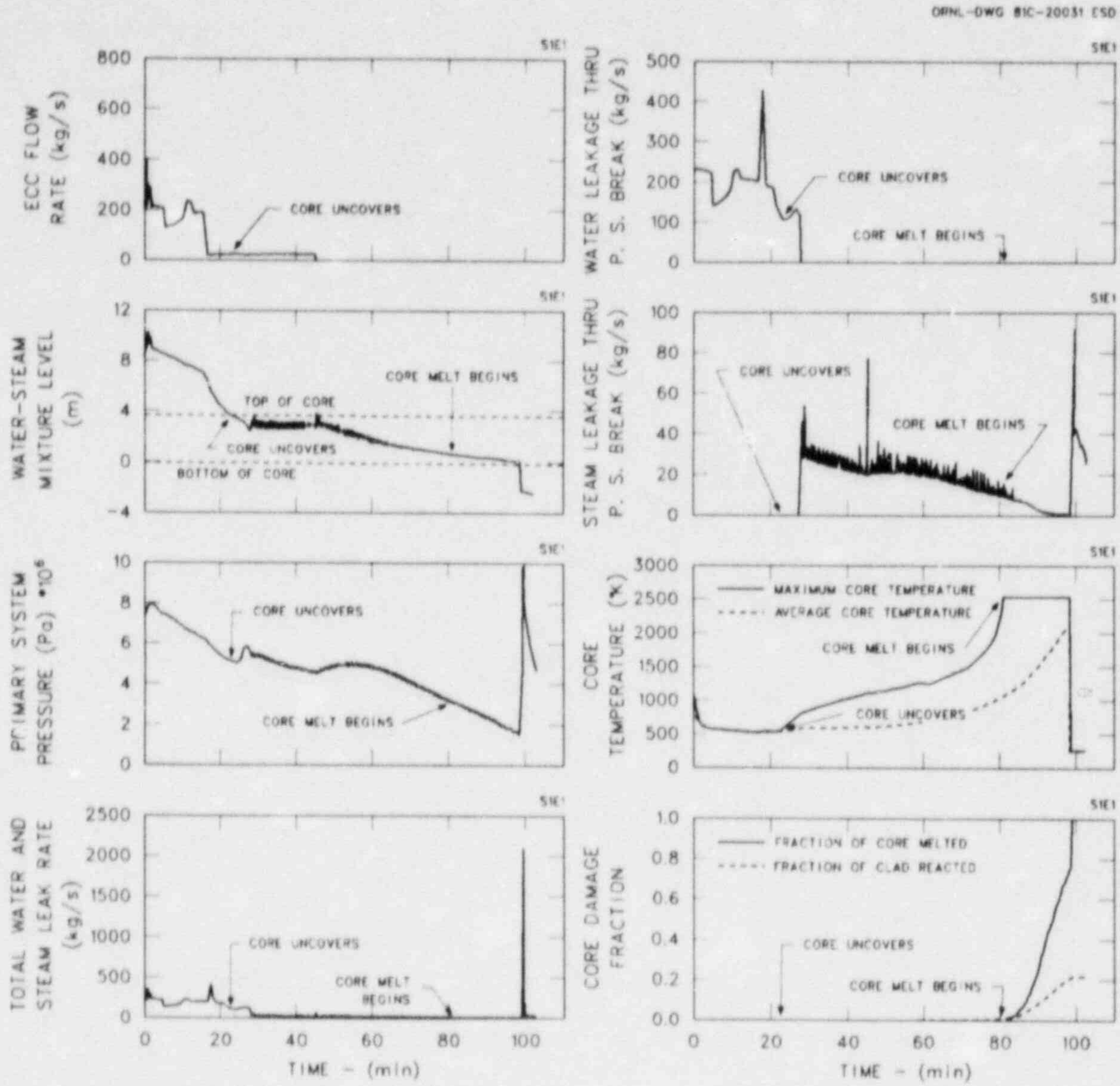

Fig. 4.23. Accident progression signatures for the S1E1 sequence.

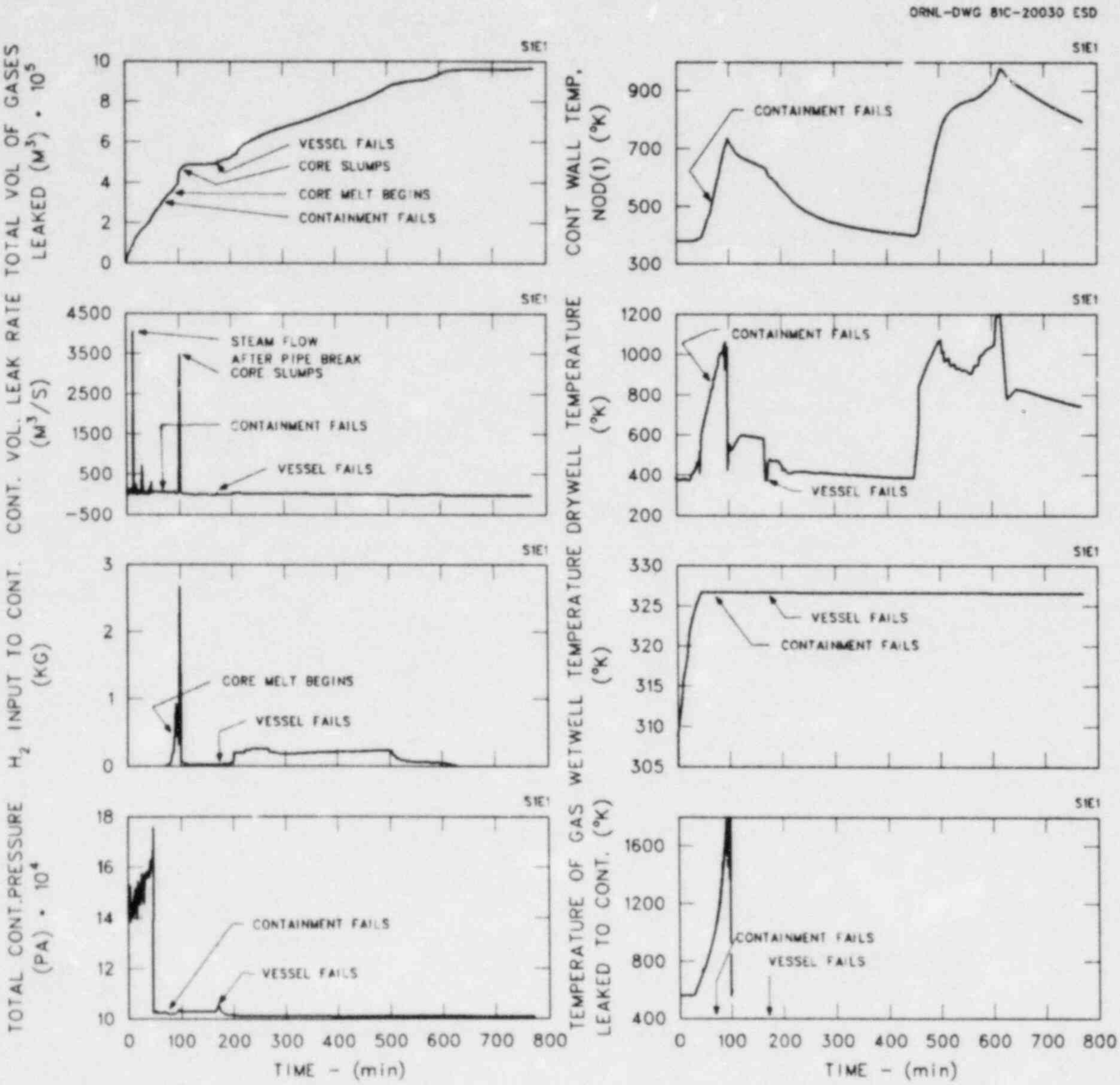

Fig. 4.24. Accident progression signatures for the S1E1 sequence.

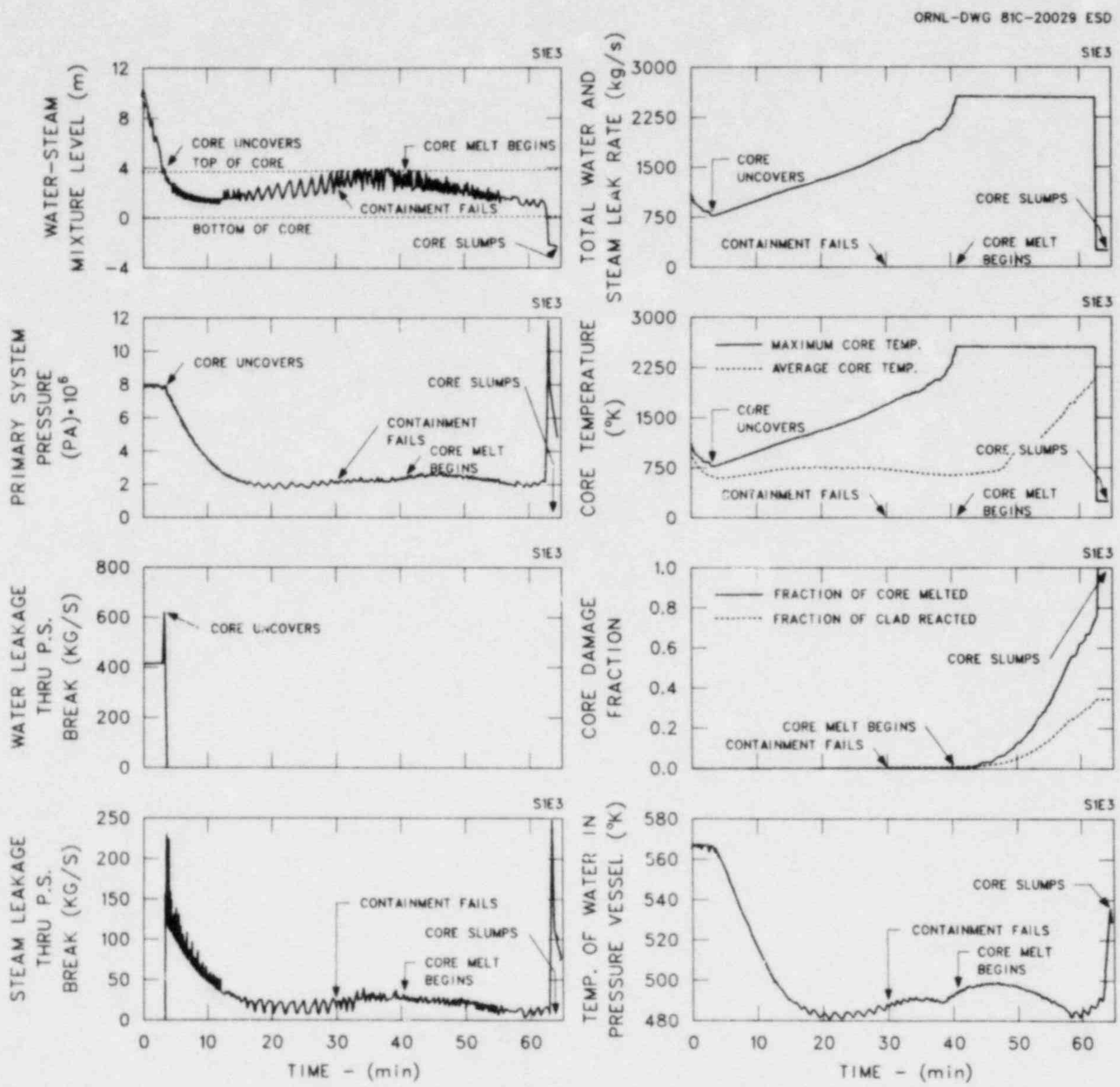

Fig. 4.25. Accident progression signatures for the S1E3 sequence.

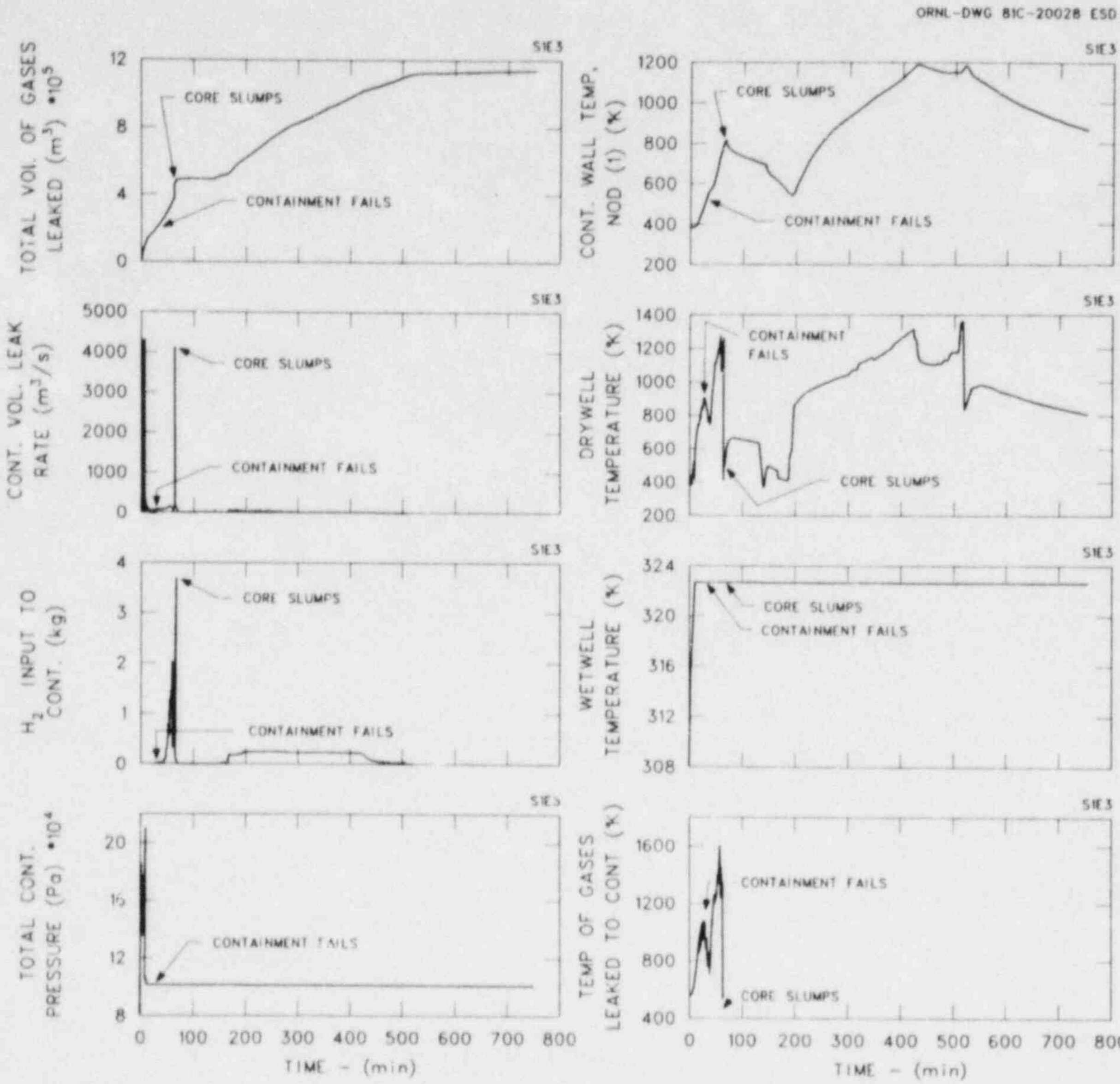

Fig. 4.26. Accident progression signatures for the S1E3 sequence.

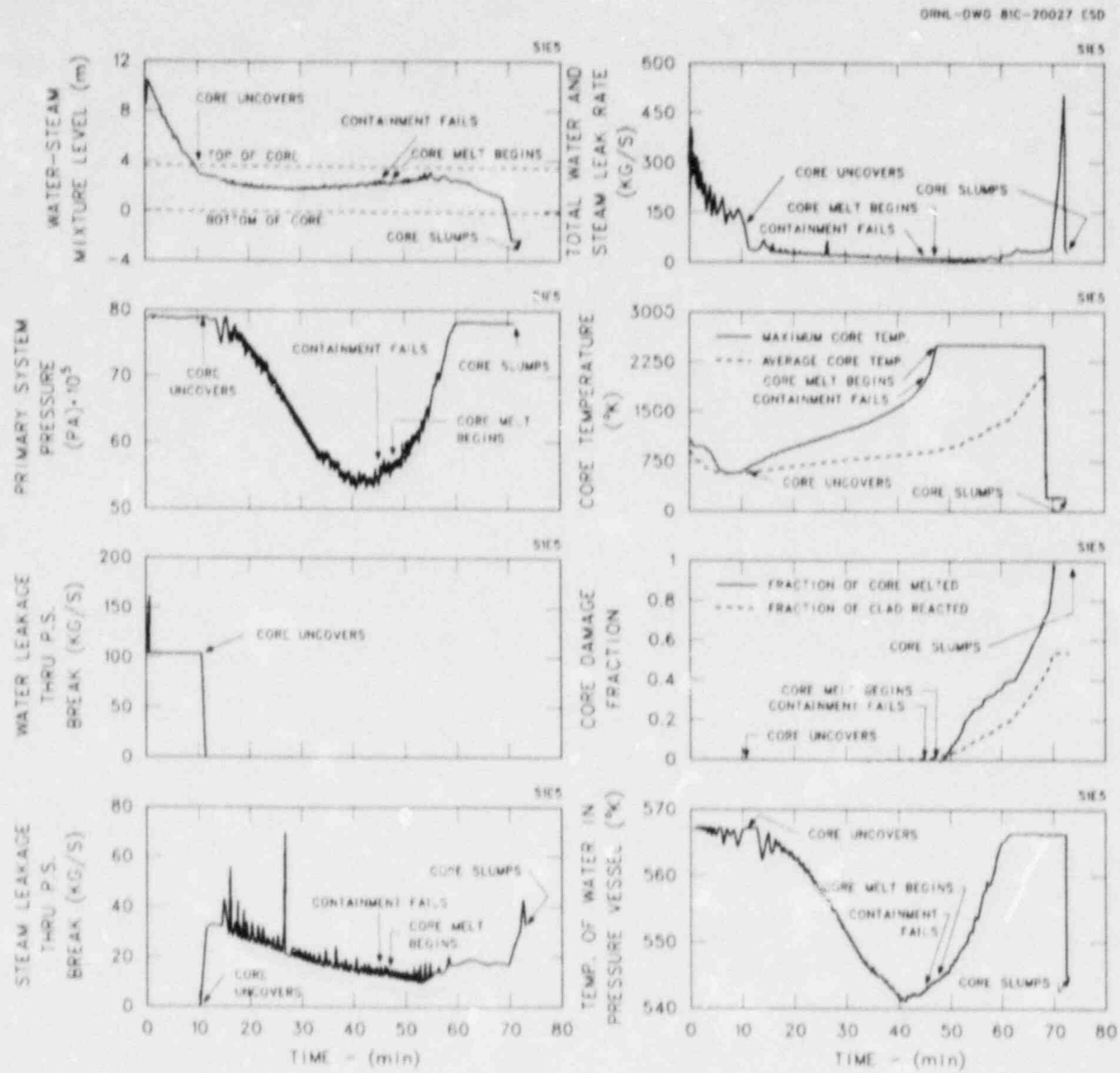

Fig. 4.27. Accident progression signatures for the S1E5 sequence.

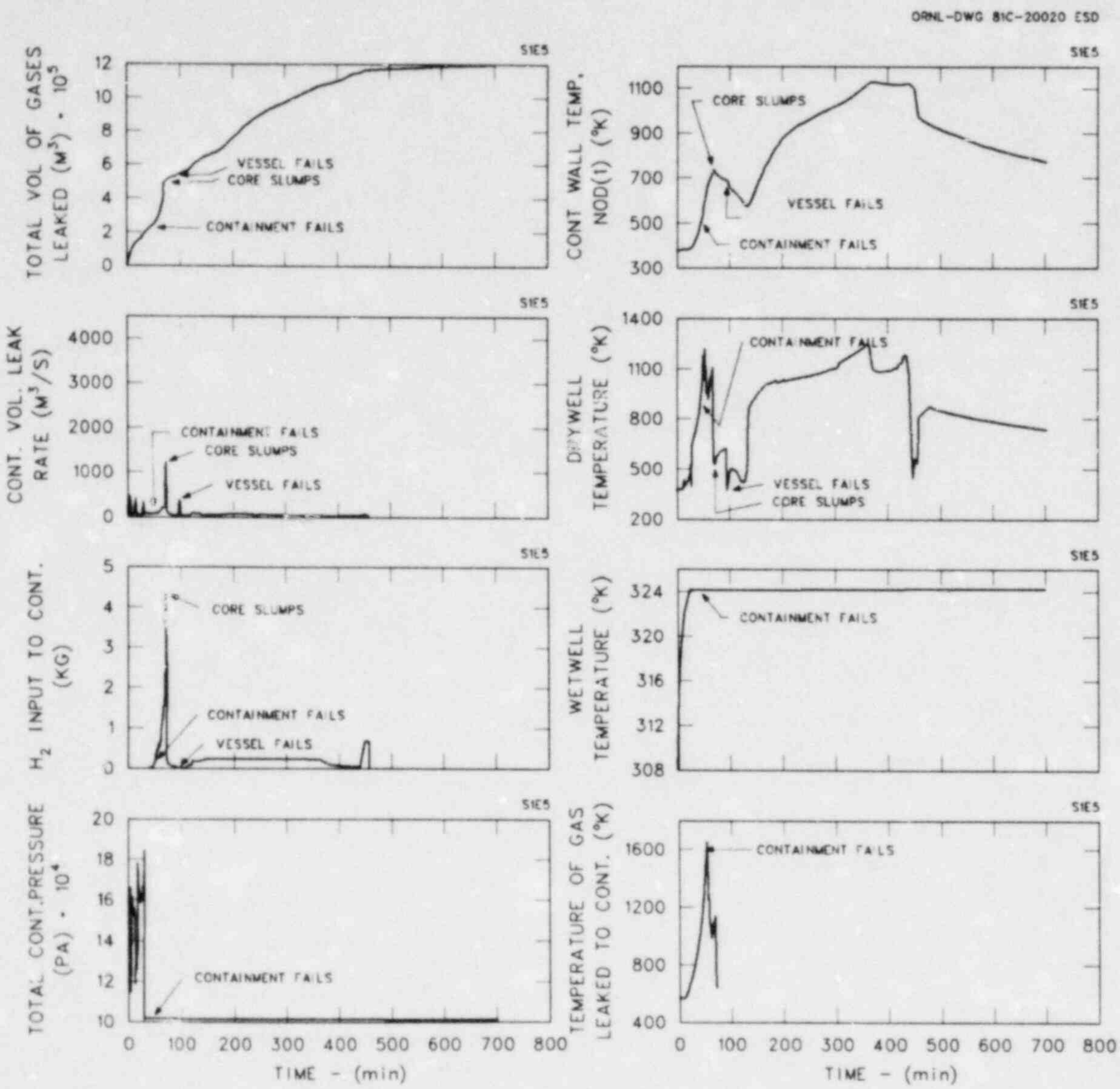

Fig. 4.28. Accident progression signatures for the S1E5 sequence.

Table 4.12. Timing of major events for S_1E sequence

Sequence	Timing of events (min)					
	Core uncovery	Start of core melt	Core slump	Failure of vessel head	Drywell EPA venting[a]	Drywell failure[b]
S1E0	7.74	44.22	73.38	127.75	35	40
S1E1	22.88	80.82	102.55	171.32	65	70
S1E2	7.74	44.22	73.38	127.75	35	40
S1E3	3.21	40.77	64.77	140.82	25	30
S1E4	4.79	37.99	66.17	138.31	25	30
S1E5	10.30	46.91	72.62	96.55	40	45
S1E6	7.76	44.04	73.25	131.59	35	40

[a]Drywell wall temperature reaches 477 K (400°F).

[b]Drywell wall temperature reaches 533 K (500°F).

Table 4.13. S_2E sequence

Sequence	ABRK[a] (ft³)	STPLH[b] (min)	TRWST[c] (°F)	AB(1)[d] (ft³)	TB(1)[e] (min)	CSRV[f]	C3(5)[g] (ft³)
S2E0	0.0218166	0	95	0	1.0E6	3719.06	20.97
S2E1	0.0218166	60	300[h]	0	1.036	3719.06	20.97
S2E2	0.0218166	300	300[h]	0	1.0E6	3719.06	20.97
S2E3	0.0218166	0	95	0.1583	0	3432.98	20.97
S2E4	0.0055	0	95	0	1.0E6	3719.06	20.97
S2E5	0.0218166	0	95	0	1.0E6	3719.06	1.0485

[a]ABRK = pipe break area.

[b]STPLH = stop time for the feedwater pumps.

[c]TRWST = temperature of feedwater makeup.

[d]AB(1) = SRV flow area.

[e]TB(1) = time to actuate SRV.

[f]CSRV = SRV coefficient.

[g]C3(5) = containment failure area.

[h]A lower feedwater temperature of 300°F rather than 420°F had to be used to remain within valid temperature range in the MARCH code.

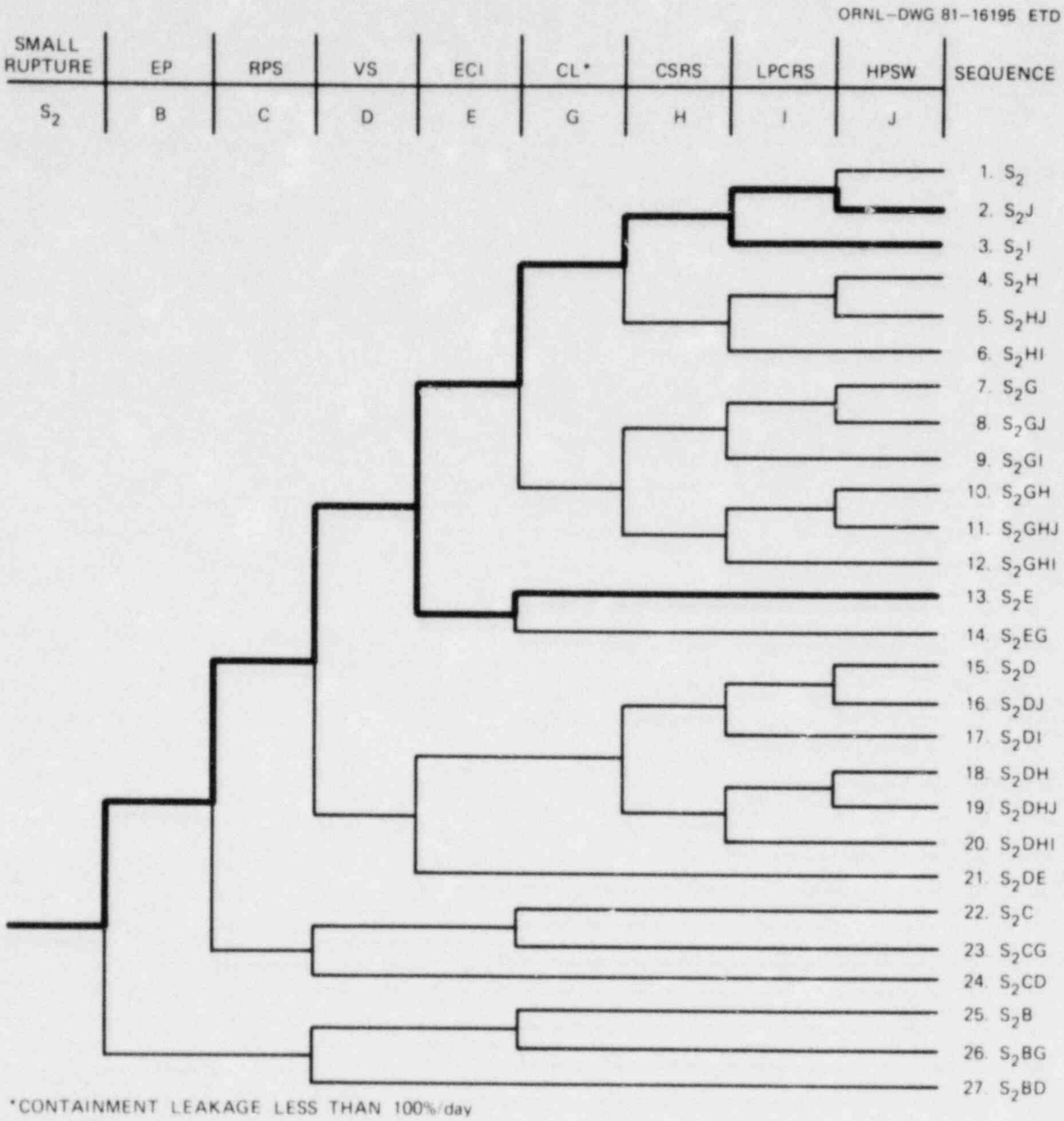

Fig. 4.29. Event tree for the small LOCA (S_2) sequences.

pressure. The rate of vessel depressurization and water level decrease would be proportional to the break size. Similar to the S_1E sequence, the increase of drywell pressure would initiate a reactor scram and closure of all primary containment isolation valves in Groups 2, 6, and 8.

Shortly after the small LOCA, the feedwater flow would increase to maintain the water level within the normal range. However, based on MARCH calculations, this makeup flow would soon stop when the water level rises momentarily above Level 8 due to fluid oscillations upon a sudden vessel depressurization, thereby tripping the feedwater pumps.* Thus, feedwater makeup is assumed unavailable for the S_2E sequence. The effect of the feedwater makeup flows on the accident progression is, however, investigated in sequences S_2E1 and S_2E2, in which the feedwater pumps are assumed to operate for 1 and 5 h, respectively, until stopped either by operator action or by control action at a vessel pressure below 5.86 MPa (850 psia). Based on results for sequences S_2E1 and S_2E2, there would be no core melt if the feedwater makeup were available during the event.

On the other hand, both the vessel depressurization in sequence S_2E3 and reduced containment failure area in sequence S_2E5 have produced no significant effect on the accident progression.

The time history of the accident progression for the S_2E0 sequence is presented in Table 4.14. Key results and accident progression signatures obtained from MARCH for the S_2E0 sequence are presented in Figs. 4.30 and 4.31.

Selected accident progression signatures for the S_2E1 and S_2E4 sequences are presented in Figs. 4.32–4.35. Comparison of timing of major events for the six cases of the S_2E sequence are given in Table 4.15.

4.7 S_2I Sequence

The S_2I sequence is defined in the Reactor Safety Study[3] as being initiated by a break size that is large enough to require either the HPCI or RCIC to maintain the vessel water level. The break is, however, too small to depressurize the system to the point where low-pressure ECCS can be actuated shortly after the break. When the reactor pressure is eventually reduced (due to the break) to the point where the HPCI and RCIC systems can no longer maintain the water level, the low-pressure core spray recirculation system (CSRS) and coolant recirculation system (LPCRS) (i.e., the LPCI mode of the RHR system) would then be actuated. In sequence S_2I, the CSRS is assumed to be available for the makeup supply from the suppression pool, which is, however, not being cooled. Without removal of decay heat, both the temperature and pressure would build up in the suppression pool, which would eventually fail by overpressurization. This would result in flashing of the torus water and cavitation of the ECCS pumps. Shortly thereafter, the core would uncover and melting would start. Similar to the TW sequence, it has been conservatively assumed that the wetwell would fail before the EPA seals. Otherwise, the core melt would be considerably delayed until the increase of suppression pool

*This level rise is based on MARCH calculations, which may differ from other detailed thermal-hydraulic code calculations.

Table 4.14. Accident chronology for S_2E sequence

Time	Event
00:00	A liquid pipeline of 0.0508-m (2-in.) equivalent break diameter breaks inside the drywell.
00:03	Reactor scrams upon high drywell pressure. All containment isolation valves in Groups 2, 6, and 8 are initiated to close.
00:05	Main turbines and feedwater pumps are tripped on high vessel water level at Level 8 caused by fluid oscillations due to a sudden pressure drop after the break.
00:30	SRV's setpoints are actuated. SRVs continue to cycle on setpoints to relieve excess vessel pressure.
14:18	Core uncovers.
40:00	Drywell EPA seals begin to vent as temperature inside the drywell has exceeded 477 K (400°F).
45:00	Drywell EPA seals have failed as the drywell temperature has exceeded 533 K (500°F) at a pressure of ~0.191 MPa (28 psia).
45:22	Core melting starts.
58:10	The corium slumps to vessel bottom.
01:32:11	Vessel bottom head fails. The debris, at a temperature over 1950 K (3050°F), is starting to boil water from containment floor and to attack the concrete basemat.

Table 4.15. Timing of major events for S_2E sequence

Sequence	Timing of events (min)					
	Core uncovery	Start of core melt	Core slump	Failure of vessel head	Drywell EPA venting[a]	Drywell failure[b]
S2E0	14.30	45.36	58.17	92.19	40.0	45.0
S2E1	77.16	121.78	150.56	172.38	115.0	130.0
S2E2	332.47	380.21	406.80	472.07	375.0	380.0
S2E3	13.30	42.36	55.31	78.07	40.0	45.0
S2E4	20.30	51.36	65.36	88.87	50.0	55.0
S2E5	13.30	45.35	58.47	91.13	40.0	45.0

[a]Drywell wall temperature reaches 477 K (400°F).

[b]Drywell wall temperature reaches 533 K (500°F).

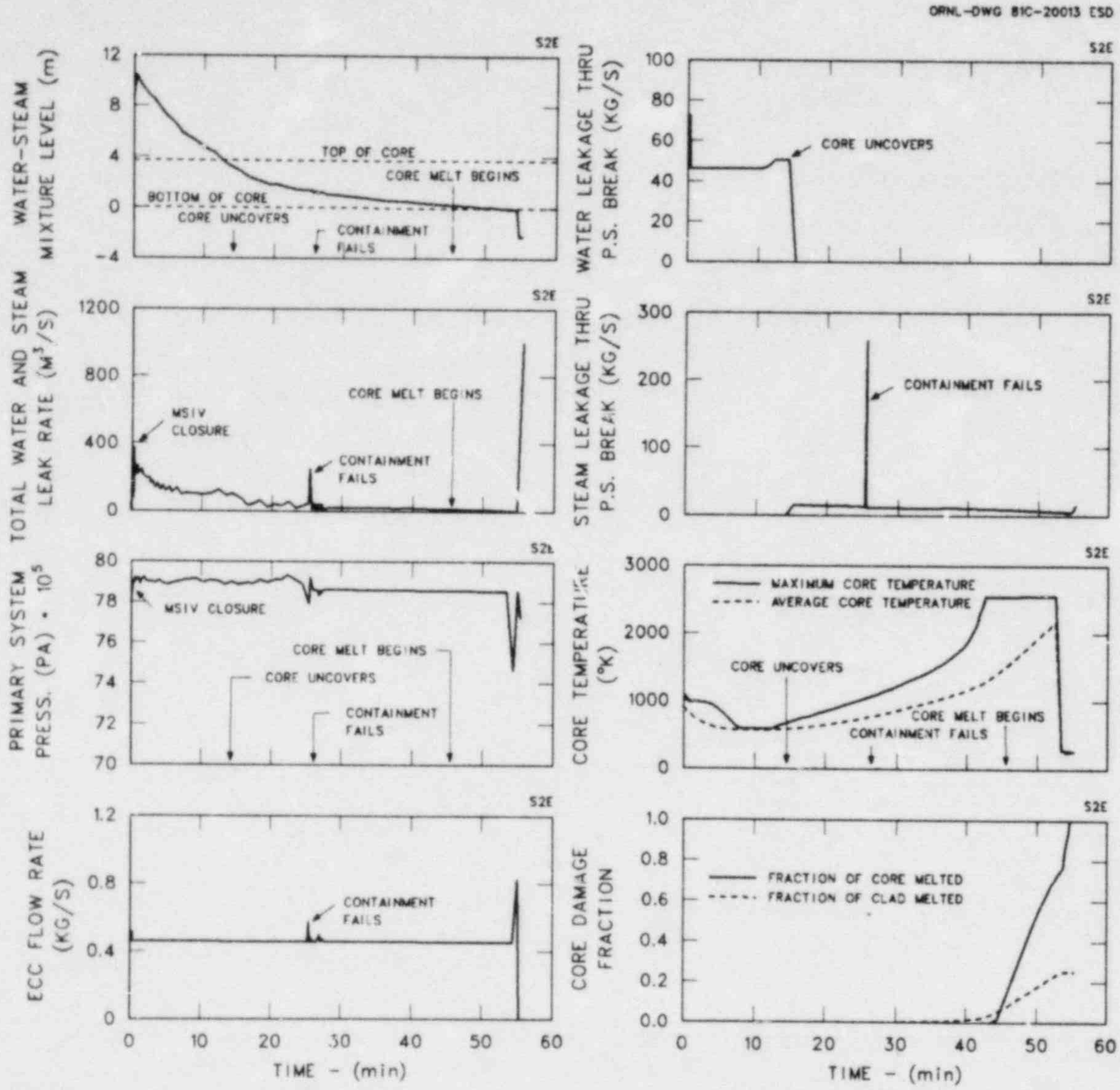

Fig. 4.30. Accident progression signatures for the S2E sequence.

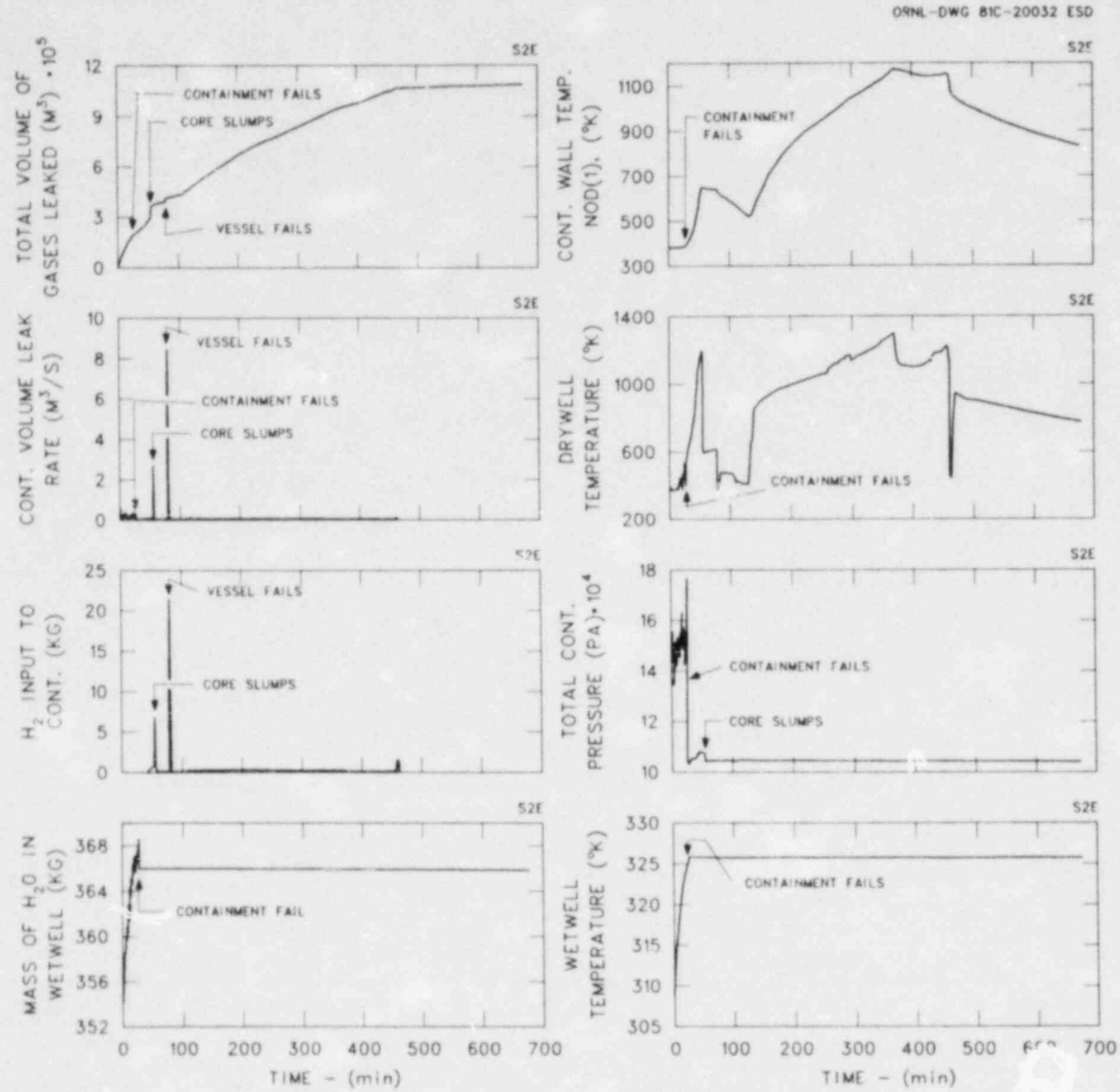

Fig. 4.31. Accident progression signatures for the S2E sequence.

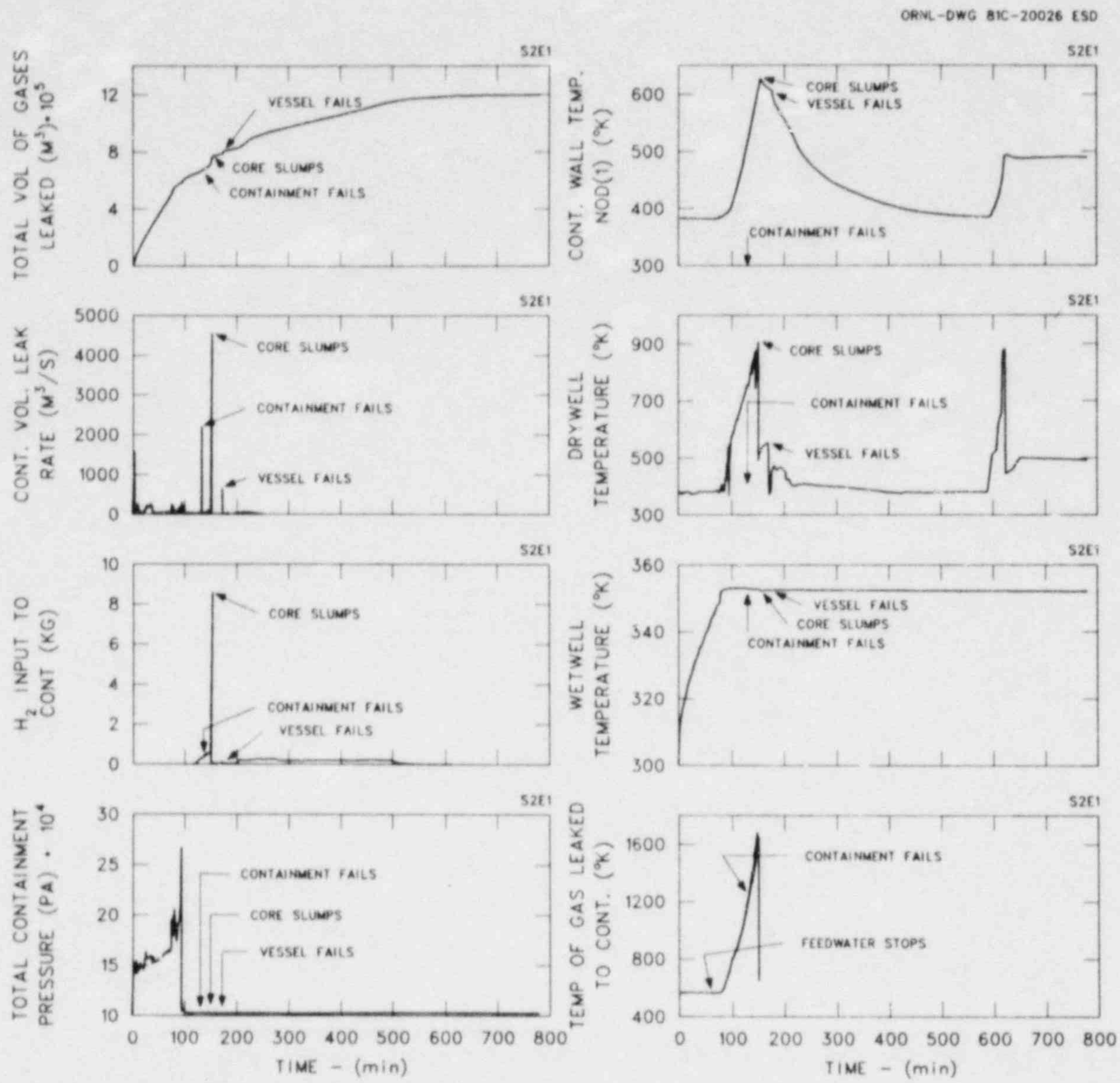

Fig. 4.32. Accident progression signatures for the S2E1 sequence.

ORNL-DWG 81C-20021 ESD

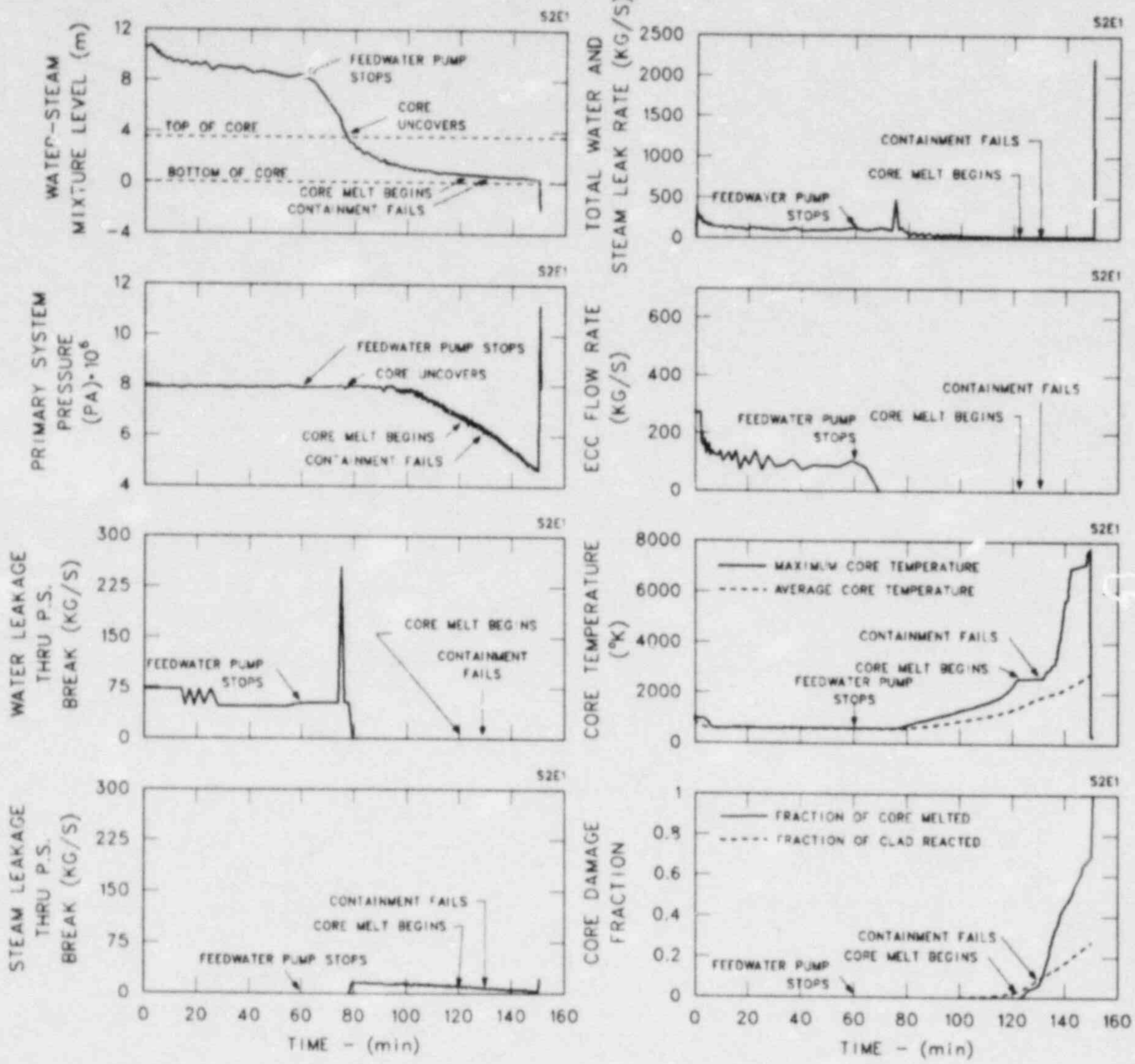

Fig. 4.33. Accident progression signatures for the S2E1 sequence.

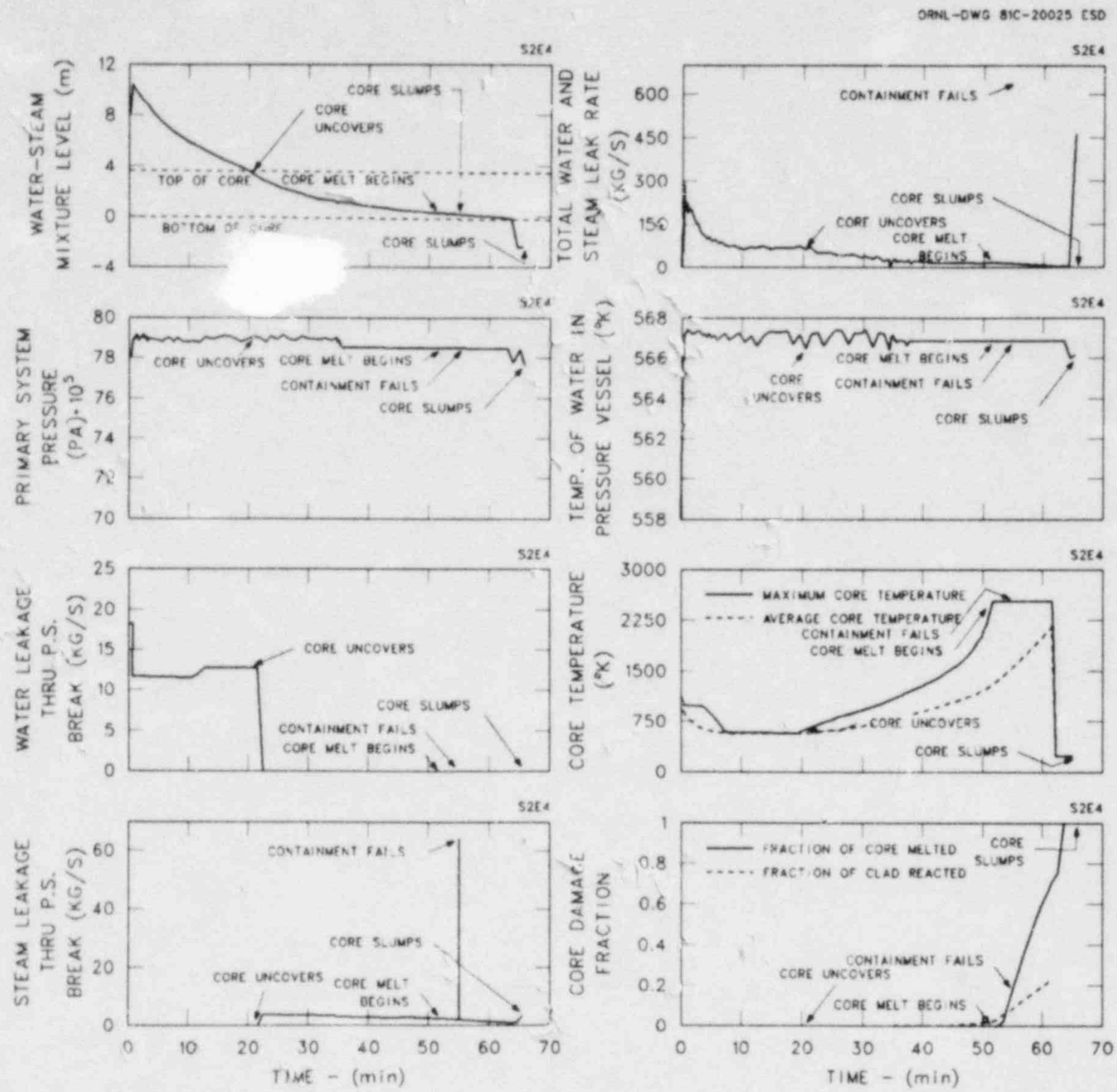

Fig. 4.34. Accident progression signatures for the S2E4 sequence.

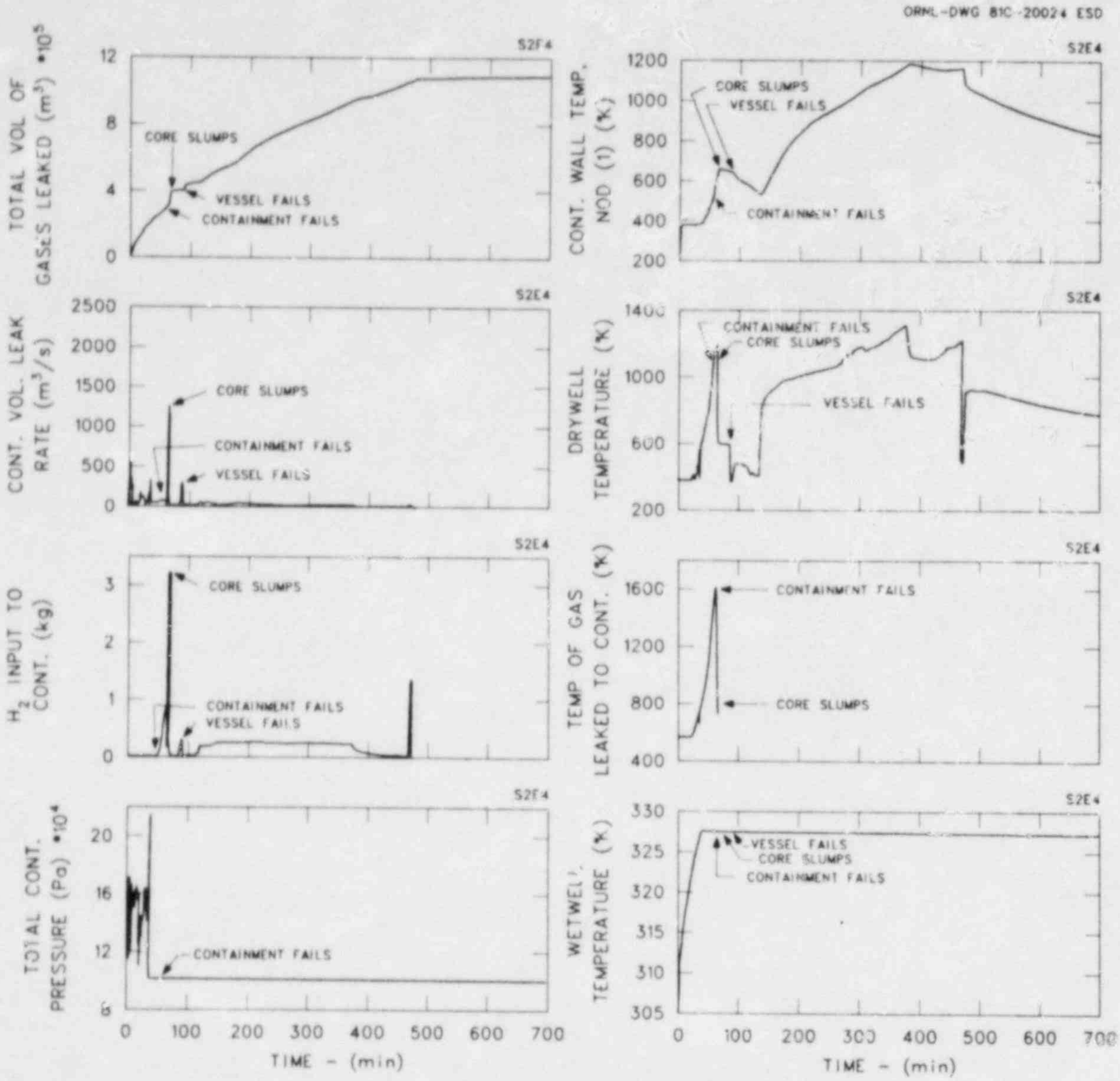

Fig. 4.35. Accident progression signatures for the S2E4 sequence.

temperature eventually causes the ECCS pumps to fail. The event tree developed in the Reactor Safety Study[3] for a small LOCA (S_2) has been given in Fig. 4.29. The containment event tree following the loss of decay heat removal has been given in Fig. 4.2.

Two break sizes of 0.0254-m (1-in.) and 0.0508-m (2-in.) equivalent break diameter* have been investigated for the S_2I sequence. Six cases of S_2I sequence studied in this work are shown in Table 4.16. These six cases have dealt with two break sizes, different durations of the feedwater operation, and effects of vessel depressurization by one and six SRVs.

Table 4.16. S_2I sequence

Sequence	ABRK[a] (ft²)	STP(2)[b] (min)	AB(1)[c] (ft²)	TB(1)[d] (min)	CSRV[e]
S2I0	0.0218166	0	0	1.0E6	3719.06
S2I1	0.0218166	60	0	1.0E6	3719.06
S2I2	0.0218166	300	0	1.0E6	3719.06
S2I3	0.0218166	0	0.1583	0	3432.98
S2I4	0.0218166	0	0.9498	0	2002.57
S2I5	0.0055	0	0	1.0E6	3719.06

[a]ABRK = pipe break area.

[b]STP(2) = stop time for feedwater makeup flow.

[c]AB(1) = SRV flow area.

[d]TB(1) = time to actuate SRV.

[e]CSRV = SRV coefficient.

Immediately after the small LOCA, the vessel pressure and water level would decrease, with a corresponding increase in the drywell pressure. Similar to the S_1E and S_2E sequences, the increase in drywell pressure would initiate a reactor scram, actuate the HPCI pump, and also provide the first of two signals necessary to actuate the ADS. The HPCI makeup begins to enter the vessel within about 30 s of the initiating signal. Later on, the operator would take over manual control of RCIC and HPCI to maintain the vessel water level.

The feedwater makeup flows would have no significant effect on the accident progression for the S_2I sequence as shown in sequences S_2I1 and S_2I2, in which the feedwater flows are assumed available for 1 and 5 h, respectively. For this reason, feedwater makeup is assumed unavailable for the S_2I sequence.

*Corresponding to an equivalent hole in the primary cooling system.

The effect of vessel depressurization by opening one or six SRVs early into the event was examined in sequences S2I3 and S2I4. Except for a momentary core uncovery early in the sequence, no significant changes have been found in the accident progression.

The time history of the accident progression for the S_2I0 sequence is presented in Table 4.17. Key results and accident progression signatures obtained from MARCH for the S_2I0 sequence are presented in Figs. 4.36 and 4.37. Selected accident progression signatures for the S_2I5 sequence are presented in Figs. 4.38 and 4.39 to show the effect of different break size. Comparisons of timing of major events for the six cases of the S_2I sequence are given in Table 4.18.

Table 4.17. Accident chronology for S_2I sequence

Time	Event
00:00	A liquid pipeline of 0.0508-m (2-in.) equivalent break diameter breaks inside the drywell.
00:03	Reactor scrams upon high drywell pressure. All containment isolation valves in Groups 2, 6, and 8 are initiated to close. HPCI system is initiated.
00:30	SRVs setpoints are actuated. SRVs continue to cycle on setpoints to relieve excess vessel pressure.
01:30	Operator takes over manual control of RCIC and HPCI systems to maintain vessel water level.
20:00	Operator has noticed that the suppression pool is not being cooled by RHR systems.
12:00:00	Core spray pumps are actuated.
25:32:52	Drywell EPA seals have failed by overpressure at ~1.22 MPa (177 psia). The wetwell has also failed by overpressure at ~1.22 MPa (177 psia). All ECCS pumps have failed due to insufficient NPSH.
25:51:01	Core uncovers.
26:52:23	Fission products begin to be released into the containment.
27:14:00	Core melting starts.
27:47:36	The corium slumps to vessel bottom.
29:34:53	Vessel bottom head fails. Debris, at a temperature over 1987 K (3117°F), is starting to boil water from containment floor and attack the concrete basemat.

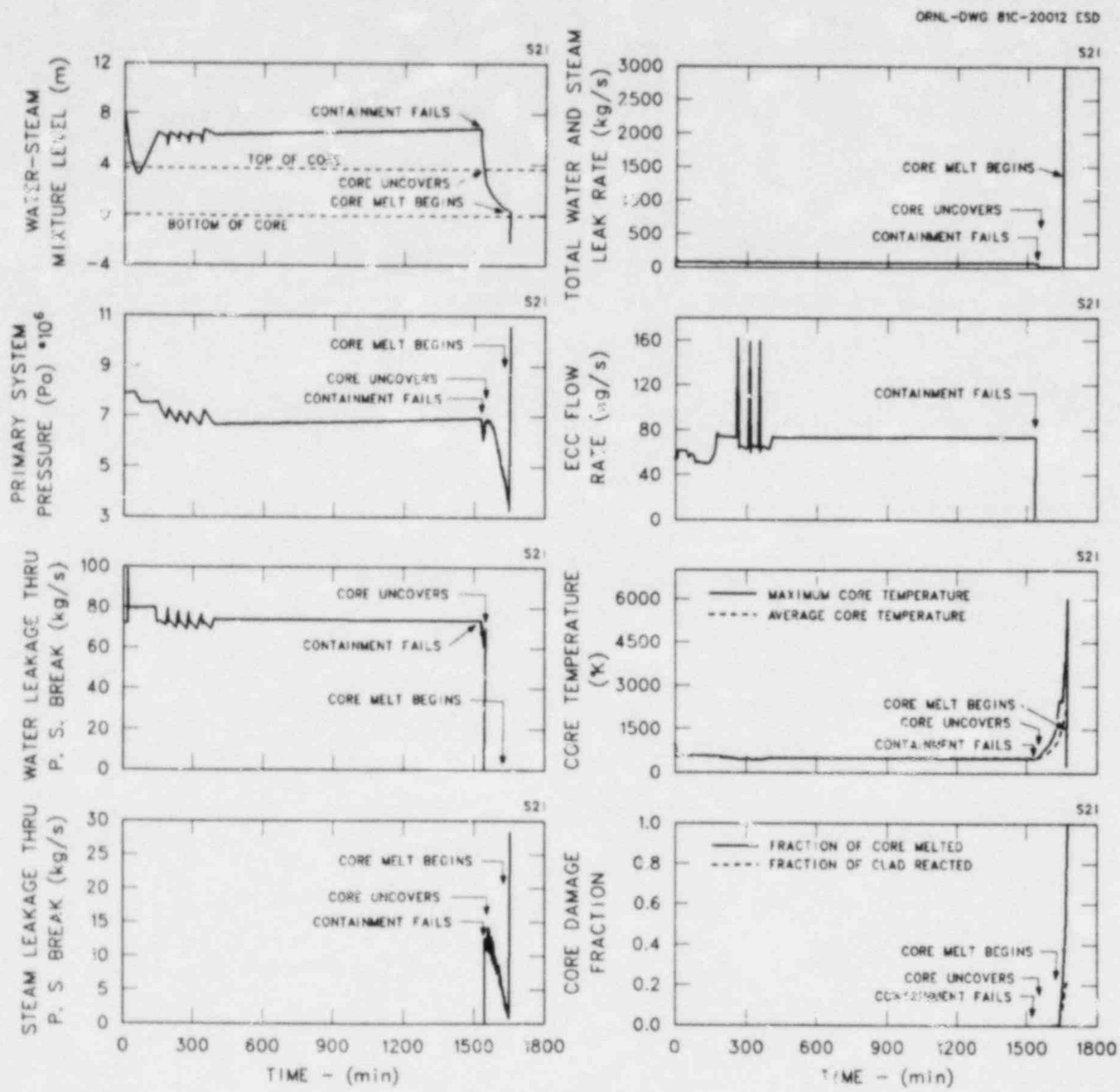

Fig. 4.36. Accident progression signatures for the S2I sequence.

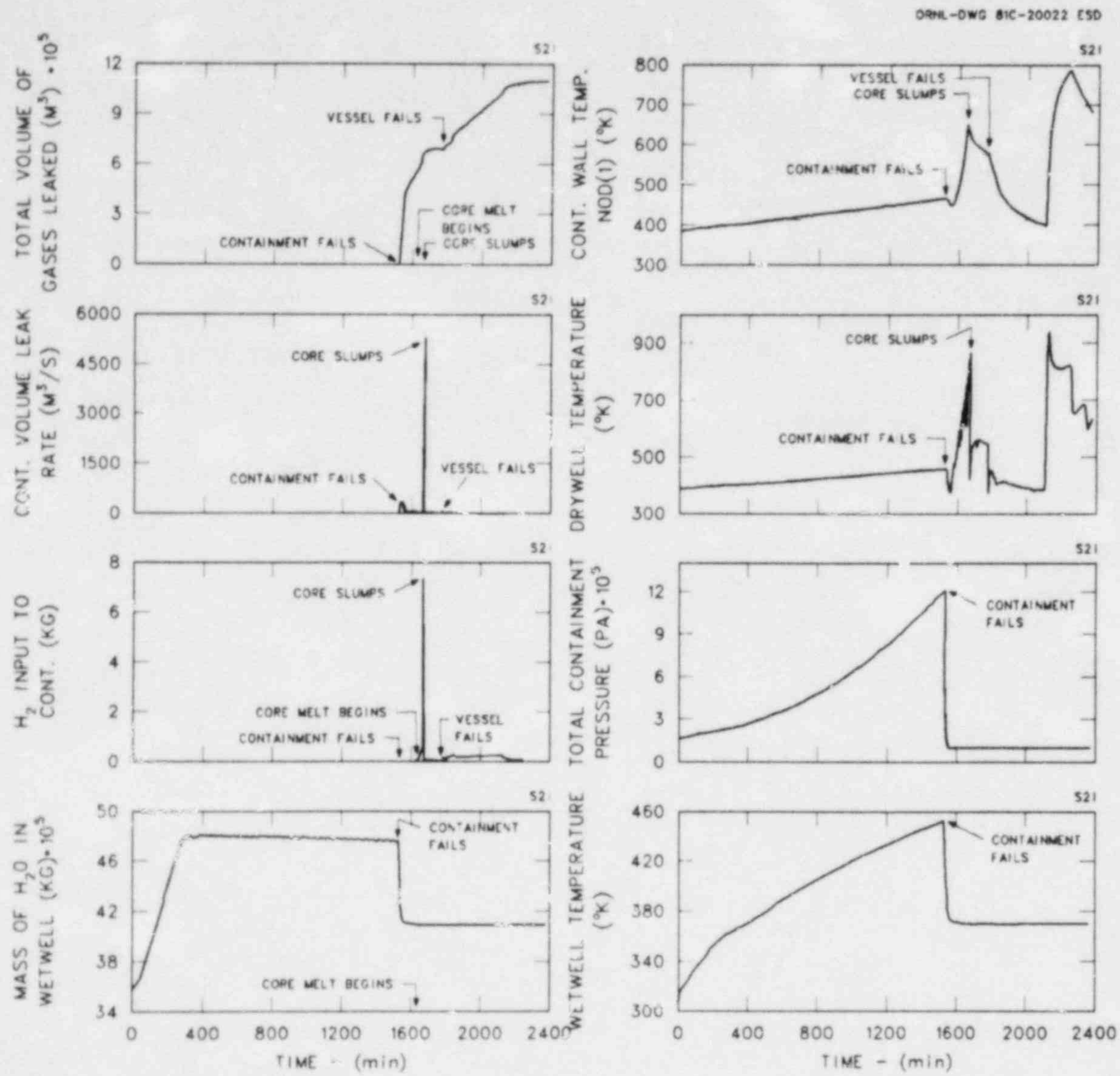

Fig. 4.37. Accident progression signatures for the S2I sequence.

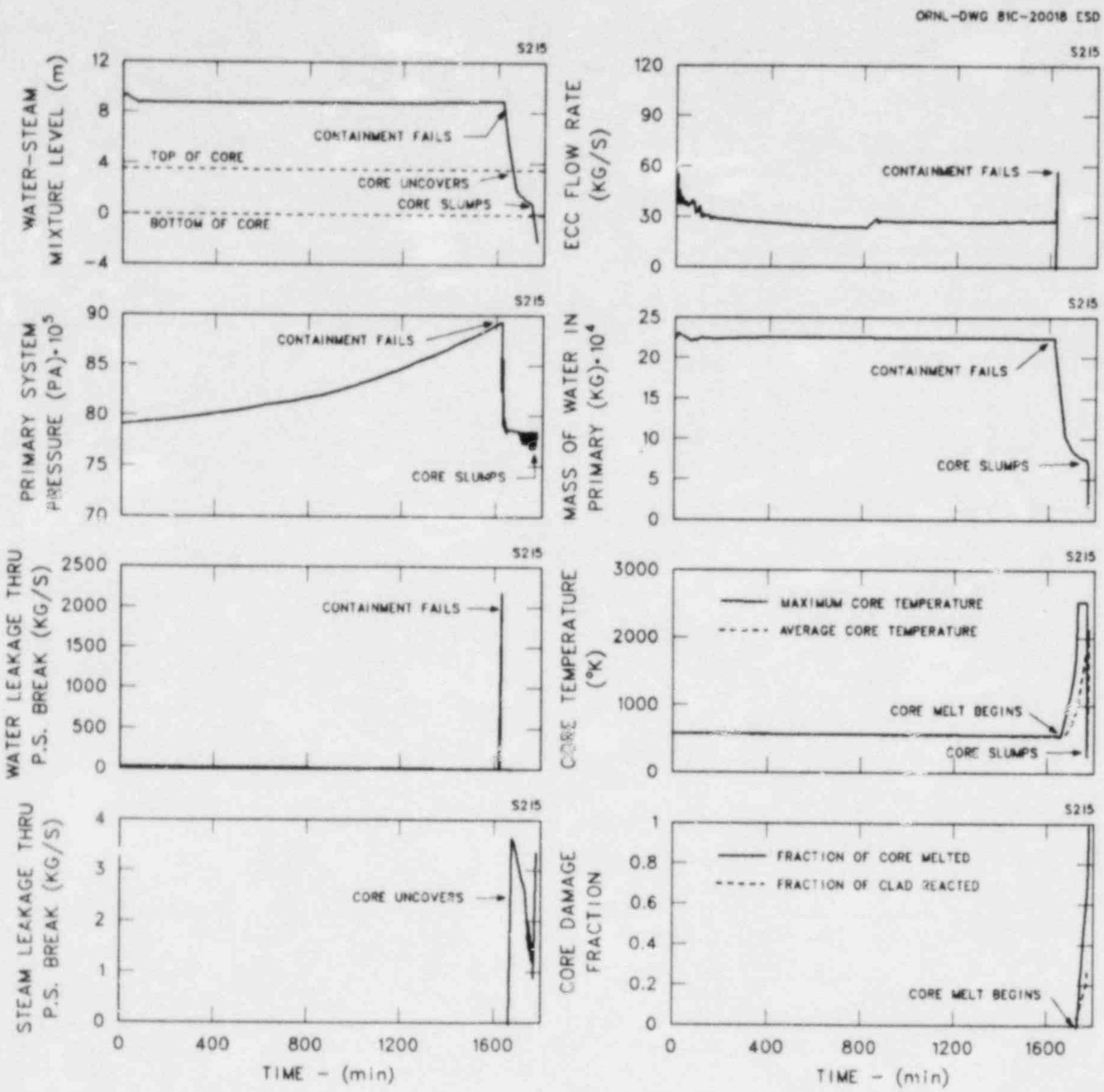

Fig. 4.38. Accident progression signatures for the S2I5 sequence.

ORNL-DWG 81C-20023 ESD

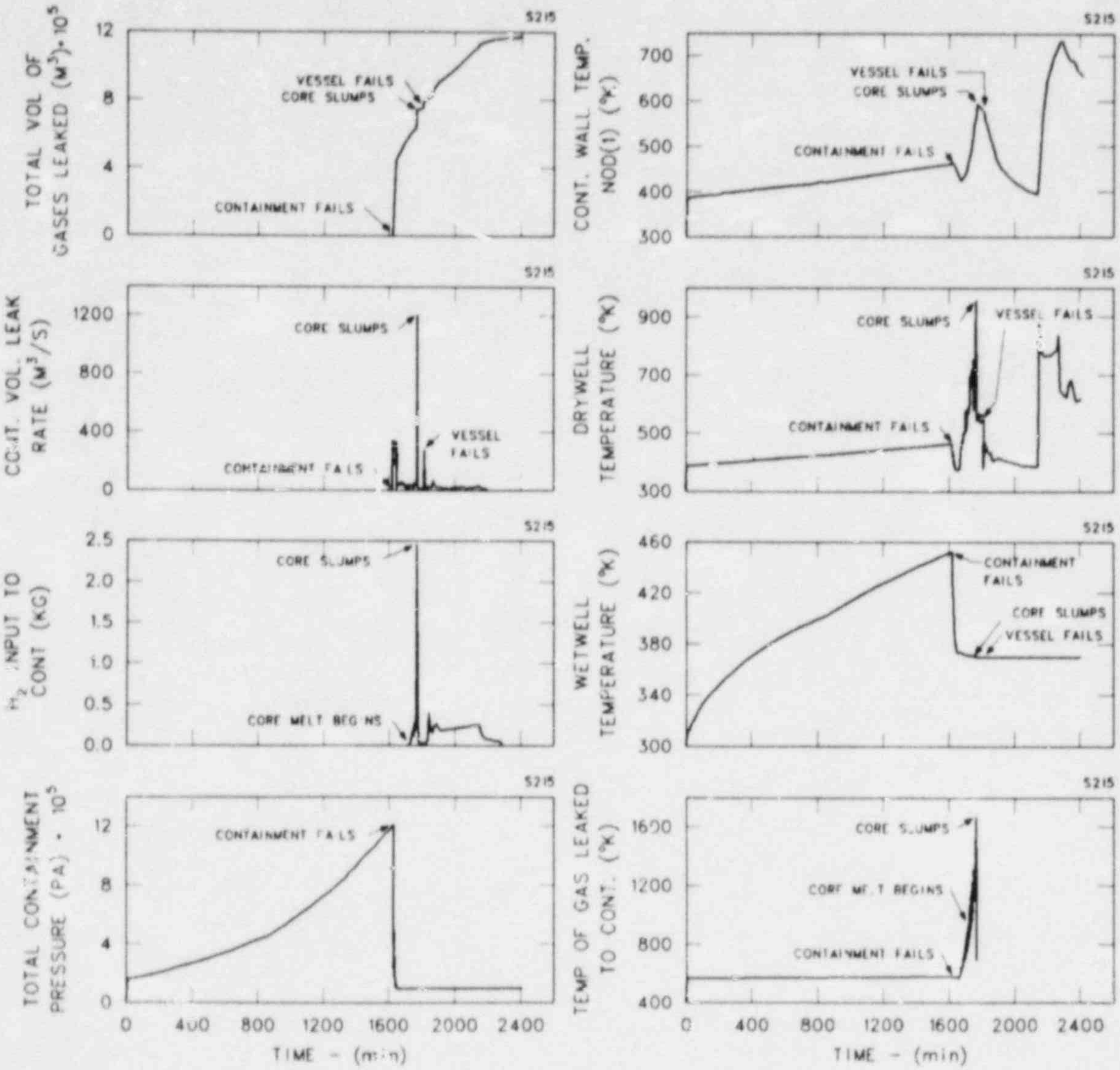

Fig. 4.39. Accident progression signatures for the S2I5 sequence.

Table 4.18. Timing of major events for S_2I sequence

Sequence	Timing of events (min)				
	Core uncovery	Start of core melt	Core slump	Failure of vessel head	Containment failure[a]
S2I0	1551.01	1634.00	1667.60	1774.88	1532.86
S2I1	1532.20	1612.07	1642.11	1766.38	1518.85
S2I2	1534.68	1617.00	1647.19	1769.70	1521.33
S2I3	44.72	1619.05	1649.69	1774.12	1524.44
S2I4	44.59	1618.08	1665.15	1771.41	1523.31
S2I5	1660.18	1727.18	1771.67	1814.56	1622.08

[a]Wetwell failure by overpressurization at ~1.22 MPa (177 psia).

4.8 S_2J Sequence

The S_2J sequence is basically identical to the S_2I sequence except that in this sequence, the loss of decay heat removal is due to the failure of the RHR service water (RHRSW) system, which provides cooling water to the RHR heat exchanger. In this sequence the LPCI mode of the RHR system is available for injection of suppression pool water into the reactor vessel whereas it was not available in the S_2I sequence. The event tree for this sequence has been given in Fig. 4.29; the containment event tree following the loss of decay heat removal has been given in Fig. 4.2.

The initiating event for the S_2J sequence is assumed to be a liquid pipeline break of 0.0254-m (1-in.) equivalent break diameter* inside the drywell. Table 4.19 gives two cases studied for this sequence to show the effect of vessel depressurizing rate on the accident progression. In general, vessel depressurization by opening more SRVs would result in earlier containment breach and starting of core melt.

Similar to the S_2I sequence, the feedwater makeup flow would also have no significant effect on the accident progression for the S_2J sequence because adequate makeup is available from the ECCS systems. Therefore, the feedwater makeup is assumed unavailable for this sequence. It has again been conservatively assumed that the wetwell would fail before the EPA seals. Otherwise, the core melt would be considerably delayed until the increase of suppression pool temperature eventually causes the ECCS pumps to fail.

The time history of the accident progression for the S_2J0 sequence is presented in Table 4.20. Key results and accident progression signatures obtained from MARCH for the S_2I0 sequence are presented in Figs. 4.40 and 4.41. Comparison of timing of major events for the two cases are given in Table 4.21.

*Corresponding to an equivalent hole in the primary cooling system.

Table 4.19. S_2J sequence

Sequence	AB(1)[a] (ft^2)	TB(1)[b] (min)	CSRV[c]
S2J0	0	1.0E6	3719.06
S2J1	0.9498	0	2002.57

[a]AB(1) = SRV flow area.

[b]TB(1) = time to actuate SRV.

[c]CSRV = SRV coefficient.

Table 4.20. Accident chronology for S_2J sequence

Time	Event
00:00	A liquid pipeline of 0.0254-m (1-in.) equivalent break diameter breaks inside the drywell.
00:03	Reactor scrams upon high drywell pressure. All containment isolation valves in Groups 2, 6, and 8 are initiated to close. HPCI system is initiated.
00:40	SRV's setpoints are actuated. SRVs continue to cycle on setpoints to relieve excess vessel pressure.
01:30	Operator takes over manual control of RCIC and HPCI systems to maintain vessel water level.
20:00	Operator has noticed that the suppression pool is not being cooled by RHR systems.
12:00:00	Low-pressure ECCS pumps are actuated.
27:12:05	Drywell EPA seals have failed by overpressure at ~1.22 MPa (177 psia). The wetwell has also failed by overpressure at ~1.22 MPa (177 psia). All ECCS pumps have failed due to insufficient NPSH.
28:20:04	Core uncovers.
29:14:04	Fission products begin to be released into the containment.
29:30:04	Core melting starts.
30:05:04	The corium slumps to vessel bottom.
30:45:31	Vessel bottom head fails. Debris, at a temperature over 1579 K (2383°F), is starting to boil water from containment floor and attack the concrete basemat.

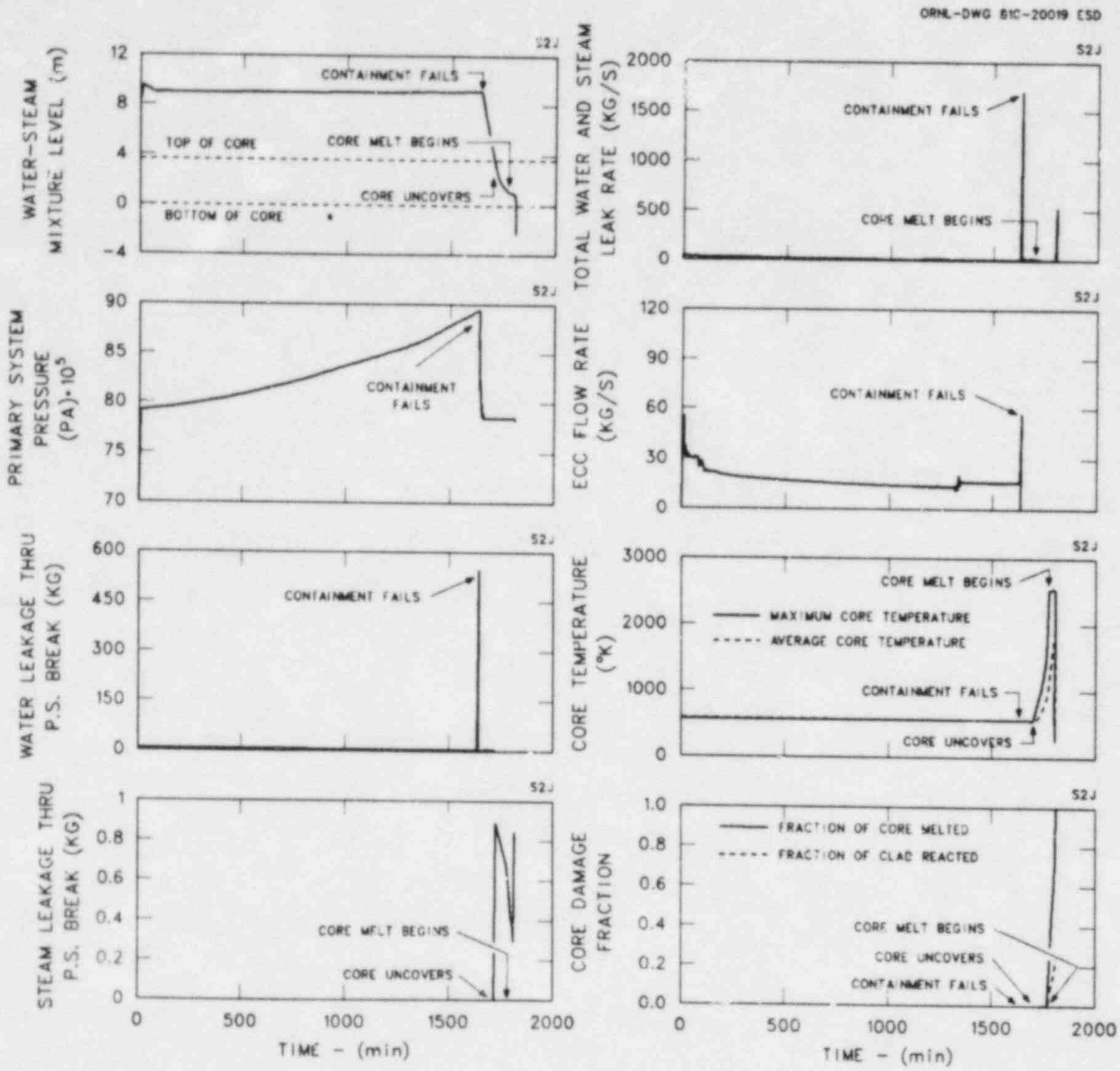

Fig. 4.40. Accident progression signatures for the S2J sequence.

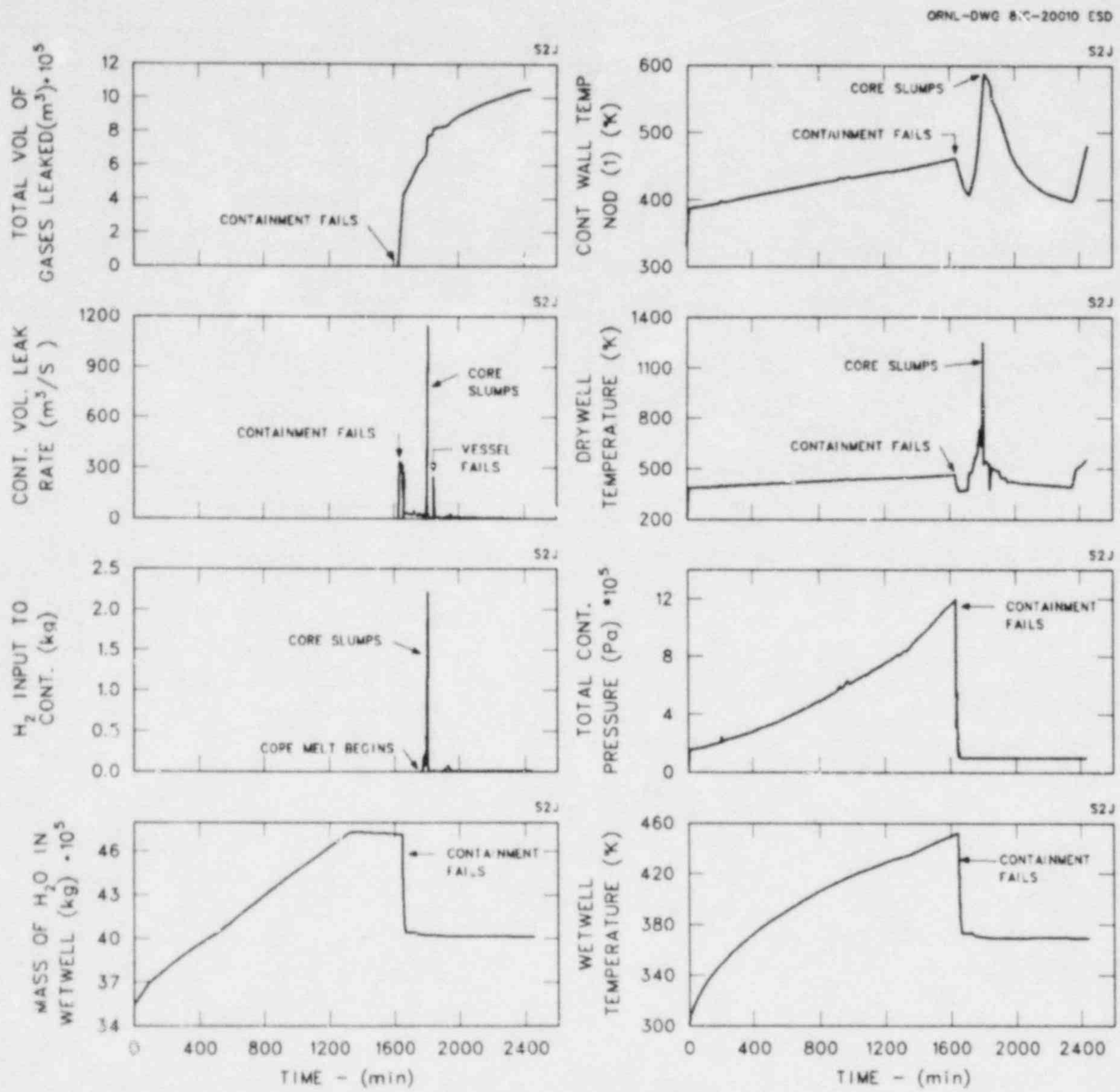

Fig. 4.41. Accident progression signatures for the S2J sequence.

Table 4.21. Timing of major events for S_2J sequence

Sequence	Timing of events (min)				
	Core uncovery	Start of core melt	Core slump	Failure of vessel head	Containment failure[a]
S2I0	1700.07	1770.07	1805.07	1845.51	1632.08
S2I1	1613.05	1681.05	1718.05	1752.48	1542.00

[a]Wetwell failure by overpressurization at ~1.22 MPa (177 psia).

5. CORRELATION OF ACCIDENT SEQUENCES WITH NRC EMERGENCY ACTION GUIDELINES

Each of the eight accident sequences described in Sect. 4 of this report is examined relative to applicability of the Emergency Action Level Guidelines set forth in Appendix 1 of Ref. 1. It is our understanding that these guidelines were prepared as bases for emergency action levels that would be applicable to all reactor accidents, and this examination of applicability is therefore addressed to the following questions:

1. Is a literal interpretation and implementation of these guidelines adequate for each of the sequences addressed in this report?
2. Do the MARCH code results for these sequences indicate the need for improvement in the guidelines; that is, is there any case where action with regard to declaring an emergency level should be taken before it would be required by a literal interpretation of the guidelines?

For the purpose of this examination, we will make the assumption that the operator is familiar with the sequences described earlier in this report. It is also assumed that he is familiar with possible mitigation actions that might change the scenario significantly.

An inherent problem encountered in this examination is that of "foresight." Each of the sequences and associated assumptions were predetermined for the purpose of calculation; therefore, our interpretation of the judgment that an operator/supervisor might exercise in a real situation is biased by foresight. The level and direction of bias probably differs from one case to another, depending on our personal view of how accurately the operator should be able to project the outcome vs what we believe to be a reluctance to escalate the emergency level to that which would have significant impact outside the plant area (e.g., evacuation).

Because the questions set forth at the beginning of this section regard the applicability of the guidelines to the establishment of emergency action levels, we do not address the actions that would be implicit in declaration of a level.

5.1 TW Sequence

The analysis of sequence TW set forth in Sect. 4.1 took as the initiating event a loss of main condenser coolant flow with consequent loss of main condenser vacuum. This event results in loss of ability to reject heat through the power conversion system (PCS). Loss of ability to reject heat via the residual heat removal (RHR) systems is then postulated as a system failure. Specific reasons for the loss of functions are not given, but for both systems it is assumed that in about 1 h the operator has determined that the systems are not repairable in a short time; that is, the heat sink has been lost, and the sequence will run its course.

Table 5.1 presents our assignment of emergency action levels for this accident sequence based on the guidelines given.

Table 5.1. Emergency action level, sequence TW

Event	Approximate time (min)	Action level/comment
Loss of condenser vacuum	0–0.5	Initiating event
HPCI and RCIC start	1	Notification of unusual event
Operator determines RHR not available	20	Declare alert
Operator determines RHR and PCS not readily repairable	60	Site emergency
Operator concludes core melt and containment failure are inevitable	60–180	General emergency

Notification of Unusual Event

Guidance is provided for this level, which is applicable to this sequence. Item 1 under "Example Initiating Conditions — Notification of Unusual Event" states, "Emergency-Core-Cooling System (ECCS) initiated and discharged to vessel." It should be noted that if only RCIC started and discharged to vessel, this guidance item would not apply, as RCIC is not included as part of the ECCS.

Alert

It is assumed that RHR has been found to be unavailable during the course of the procedures required to line up the system for suppression pool cooling. It is clear that determination of RHR unavailability represents "an actual or potential substantial degradation of the level of safety of the plant" as set forth under Alert-Class Description.[1] However, it could also fall under the Site Area Emergency-Class Description, "actual or likely major failures of plant functions needed for protection of the public."

The example initiating conditions for these two levels were examined for possible additional guidance, and if long-term cooling is considered to be a requirement for "plant cold shutdown" (Alert, item 10) or "plant hot shutdown" (Site Area Emergency, item 8) then these items would apply as examples that a change from Unusual Event level would be required.

It seems reasonable to us that at the time RHR is determined to be unavailable, a change from Unusual Event level to Alert level is appropriate and is consistent with the class description for this level. We presume that at the time RHR is found to be unavailable, further investigation would be required to determine how long it would take to restore RHR and/or PCS to operable condition.

Site Area Emergency

The discussion under Alert is also relevant to this level in regard to whether Alert, Site Area Emergency, or both should previously have been declared. Because we previously suggested declaring an Alert on the occasion of determination that RHR was unavailable, we now suggest that it is appropriate to declare a Site Area Emergency on determination that neither the RHR nor PCS is readily repairable. At this time, ~1 h following the initiating event, considerable time remains (~16 h) before containment failure is projected; it appears reasonable for plant personnel to consult with others (NRC) and to reach general agreement prior to declaring a General Emergency. It is also reasonable to assume that efforts would be continued to provide some means of cooling.

General Emergency

Because of the nature of the failure (i.e., loss of heat sink), it would be appropriate to declare a General Emergency at any time following the determination that RHR and PCS are not readily repairable. Specific guidance is given in Item 6.d under "Example Initiating Conditions: General Emergency." However, some latitude for interpretation is provided as a time to core degradation of melt of about 10 h is noted.

General Comment for TW Sequence

It is noted in Appendix 1 of Ref. 1 that, "The example initiating conditions listed after the immediate actions for each class are instrumentation readings (as applicable) which, if exceeded, will initiate the emergency class." For sequence TW, the major events are equipment oriented during the initial phases, and while some dependence on instrumentation and control indicators is necessary in determining what is happening, the basis for emergency class initiation does not appear to depend as much on instrumentation readings as on gross questions of equipment operability and repairability. The latter item, depending on the nature of the difficulty, would probably need to be determined by physical inspection.

In regard to the two questions posed at the beginning of this section, we have the following comments.

1. Is a literal interpretation and implementation of these guidelines adequate for each sequence?

   The guidelines are unambiguous in regard to "Notification of Unusual Event." The guidelines are subject to considerable interpretation in regard to Alert and Site Emergency. For this sequence it does not appear that the latitude for Alert and Site Emergency is significant because the timing is not critical. Specific guidance is provided for declaration of a General Emergency, but some room for interpretation in regard to timing is present.

2. Does the MARCH run for this sequence indicate the need for improvements in the guidelines (i.e., should protective action be taken before it would be required by a literal interpretation of these guidelines)?

The MARCH run for sequence TW does not indicate the need for improvement in the guidelines. However, it is probable that examples more specific to this sequence could be provided if it is considered desirable to allow less latitude for interpretation.

5.2 TC Sequence

The sequence analyzed in Sect. 4.2 is based on loss of feedwater as the initiating event with failure to scram following. It should be noted that TC0 sequence, which incorporates a delayed shutdown via SLCS, does not result in core melt or containment breach. On the other hand, TC1 sequence, in which the SLCS is assumed unavailable, does result in core melt or containment breach. Only the TC0 sequence is treated in this section because it has the higher probability.

Decrease in vessel water level initiates scram, recirculation pump trip, and HPCI and RCIC operation within about 10 s following the initiating event. The operator observes the failure to scram and initiates action to manually insert the rods; manual insertion is also assumed to be unsuccessful. No detailed consideration was made of the steps necessary for the operator to take action and determine that the manual system failed to respond. The SLCS is assumed to be actuated by the operator, and poison starts to enter the core at about 3 min into the sequence.

For the purpose of this investigation, it is assumed that following recirculation pump trip, the power level would decrease to 30% of full power. At the time that SLCS poison starts to have effect, ~3 min, it is assumed that the power level begins a linear decrease, which reaches decay power level at about 30 min. Additional studies are needed to obtain better information regarding the behavior of power vs time for SLCS injection under ATWS conditions, because this is of crucial significance to this sequence.

With the assumptions made, the MARCH calculations indicate that the core does not melt, although some damage may have occurred. The calculations also show that containment failure would not occur. However, as noted previously, more critical analysis is recommended when better information on power vs time becomes available.

Table 5.2 presents our assignment of emergency action levels for the TC sequence. The initial emergency action level is clearly identified in

Table 5.2. Emergency action levels, sequence TC

Event	Approximate time (min)	Action level/comment
Loss of feedwater	0	Initiating event
Operator recognizes ATWS	2	Site area emergency
Power level decreases to after heat and RHR systems in operation	~30	Reduction to alert condition
Plant brought to cold shutdown (with no significant increase in releases)	Depends on details (many hours)	Closeout of offsite emergency

the "Example Initiating Conditions: Site Area Emergency," item 9., "Transient requiring operation of shutdown systems with failure to scram (continued power generation but no core damage immediately evident)." As the calculations (and presumably the instrumentation) indicate a decreasing power level with no break in containment, no basis for escalating the level is apparent.

General Comments for TC Sequence

Some ambiguity exists between assignment of emergency action levels of Site Area and General Emergency. Item 9 under "Example Initiating Conditions: Site Area Emergency" states,

> Transient requiring operation of shutdown systems with failure to scram (continued power generation but no core damage immediately evident).

"Example Initiating Conditions: General Emergency" item 6.a. states,

> Transient (e.g., loss of offsite power) plus failure of requisite core shut down systems (e.g., scram) could lead to core melt in several hours with containment failure likely. More severe consequences if pumps trip does not function.

There appears to be little difference between those examples insofar as providing guidance as to which level should be declared. A more detailed series of analyses with possible operator intervention might reveal a clearer separation or possibly additional guidance as to how the guidelines and/or examples should be stated.

An additional item of note is that very little guidance is provided for reduction in emergency action level. This is probably not as critical as establishment of a level or escalation of level. However, some consideration should be given to this point; hopefully, the frequency of reduction in action levels will be equal to that of declaration or escalation.

5.3 TQUV Sequence

The sequence analyzed in Sect. 4.3 is based on loss of feedwater as the initiating event. Early in the sequence, decrease in vessel water level initiates scram, which is effective. Decrease continues and RCIC and HPCI initiation signals occur on Level 2 at ~30 s; these are ineffective due to system failures, the low-pressure ECCS are also found to be inoperative at this time or shortly thereafter, and therefore the operator does not depressurize.

A review of the sequence analysis was made in regard to the Emergency Action Level Guidelines, and Table 5.3 presents our assignment of levels for this sequence based on the guidelines. It will be noted that a general emergency is declared as the first step following the initiating event. The failure of HPCI and RCIC to function when called upon, coupled with determination that low-pressure ECCS are inoperable, is sufficient

Table 5.3. Emergency action level, sequence TQUV

Event	Approximate time (min)	Action level/comment
Loss of feedwater	0	Initiating event
Failure of HPCI and RCIC on demand and determination that LP ECCS are inoperable	1–10	General emergency

cause for declaring a General Emergency as set forth under the class description for this level. The problem that should be noted and must be faced by the operator is that of the possibility that unavailability of high-pressure and low-pressure injection and cooling systems is caused by conditions that could quickly be remedied. Unfortunately, only about 3 h are available before predicted release of gross quantities of fission products due to core melt, vessel failure, and containment failure; we believe the guidance calls for declaration of a General Emergency condition at the same time, or before, investigations are initiated to determine if the observed failures can be corrected.

We believe the guidance provided is adequate for this sequence. MARCH code calculations for this sequence do not indicate the need for improvement in the guidelines.

5.4 AE Sequence

The initiating event postulated is a large-break LOCA. The break is assumed to be in a liquid recirculation line inside primary containment, and ECI does not reflood the core. The situation is such that the time of declaration of an emergency is dictated more by operator response than by any detailed analyses of data. Increase in drywell temperature and pressure accompanied by rapid loss of vessel water level indicates a major break. So little time is available before the onset of core damage (~5 min) that we believe the appropriate action would be to declare a General Emergency, as cited in Table 5.4.

Table 5.4. Emergency action level, sequence AE

Event	Approximate time (min)	Action level/comment
Large-break LOCA	0	Initiating event
Core uncovery	1–10	General emergency

We believe the guidance provided is adequate. The MARCH code calculation for this sequence does not indicate the need for improvements in the guidelines.

5.5 S_1E Sequence

The initiating event is assumed to be a small-size break of a liquid line inside the primary containment. HPCI fails to provide coolant on demand, and the ADS does not depressurize the vessel due to absence of signals (discharge pressure) indicating availability of either RHR or core spray pumps. RCIC and CRD pumps combined flow is inadequate to maintain coolant inventory and vessel level decreases. Table 5.5 presents our assignment of emergency action levels.

Table 5.5. Emergency action level, sequence S_1E

Event	Approximate time (min)	Action level/comments
Small-break LOCA	0	Initiating event
Failure of HPCI to function and unavailability of RHR and CS pumps	1–10	General emergency

For this situation, the timing is such that declaration of a General Emergency is the only reasonable course of action. Investigation of the cause(s) of the failures in the hope of achieving operability of equipment would be initiated, but emergency steps such as preparation for evacuation must proceed concurrently.

We believe the guidance provided is adequate. The MARCH code calculation for this sequence does not indicate the need for improvements in the guidelines.

5.6 S_2E Sequence

This sequence is similar to S_1E as set forth in Sect. 5.5. The main differences are that the RCIC pump is also assumed to have failed and the leak area is smaller by a little over an order of magnitude. It is assumed that the CRD pump continues to operate but provides insufficient flow to maintain coolant level. Thus, core uncovery and core melt will eventually result. Sequence S_1E analysis indicated core uncovery at about 8 min; S_2E analysis for the conditions assumed results in core uncovery at about 14 min. Although vessel water level is decreasing more slowly for the present case, we can only suggest that this would allow a little more time before a General Emergency should be declared on the basis of early, total failure of ECCS. This is shown in Table 5.6. As noted in Sect. 5, these scenarios or sequences must be considered incomplete from the viewpoint of assigning emergency action levels because consideration of possible operator actions have not been included as part of this study.

For this case, we see no reason to modify the Emergency Action Guidelines.

Table 5.6. Emergency action level, sequence S_2E

Event	Approximate time (min)	Action level/comment
Small-break LOCA	0	Initiating event
Failure of HPCI, RCIC, RHR, and CS systems	1–10	General emergency

5.7 S_2I Sequence

The initiating event is a small-break LOCA. It is also assumed that feedwater flow is lost because no significant difference was found in investigation of alternative MARCH calculation. It is further assumed that the RHR system fails to pump water through the RHR heat exchangers, and thus suppression pool cooling is not available. There are probably a number of actions that would be tried by the operator to provide cooling and thus modify the sequence, but no consideration is given to such action.

Table 5.7 presents our assignment of Emergency Action Levels, and the following comments address the guidance for declaration of the emergency action levels.

Table 5.7. Emergency action level, sequence S_2I and S_2J

Event	Approximate time (min)	Action level/comment
Small-break LOCA	0	Initiating event
Primary coolant leak rate >50 gpm	1–10	Alert
Operator determines suppression pool cooling is not available	~30	Site area emergency
Operator determines no long-term heat removal can be provided	Several hours	General emergency

Alert

Item 5 under "Example Initiating Conditions: Alert" states, "Primary coolant leak rate greater than 50 gpm." This provides adequate guidance for establishing this level based on recognition of the leak through increases in drywell temperature and pressure. Lack of any specific guidance for establishing a higher level would also indicate this as appropriate.

Site Area Emergency

No specific guidance is provided relating to discovery of unavailability of suppression pool cooling. However, it appears that the emergency action level should be escalated to this level as provided for in the example list as No. 17. "Other plant conditions exist that warrant activation of emergency centers and monitoring teams or a precautionary notification to the public near the site." The timing is not considered to be critical, and, as no specific guidance is given, it would appear to be more or less at the discretion of the operator/supervisor. Our view is that the escalation should be made as soon as it is determined that suppression pool cooling cannot be promptly put in operation using the equipment intended for the purpose. Some guidance may be obtained from General Emergency example 6.d, where a time of 10 h to core melt is indicated as appropriate for declaration of general emergency under loss of necessary decay heat removal systems. Some clarification appears appropriate for this case.

General Emergency

Guidance to establishment of this level is quite specific as set forth in example 6.d. Again, the question of timing of declaration is left a little vague, but about 10 h before predicted core melt is clearly indicated and could be considered a minimum. It is our opinion that the operating staff would be attempting to rectify the situation from the time the difficulty is identified and would therefore defer any declaration of General Emergency as long as possible, with the hopes of providing cooling and thereby preventing core melt and allowing a reduction in level rather than an escalation.

5.8 S_2J Sequence

The S_2J sequence analyzed in Sect. 4.8 is nearly the same as the S_2I sequence of Sect. 4.7. The break size is about a factor of 4 smaller in area, and it is assumed that flow to the secondary side of the RHR heat exchangers is lost rather than primary flow. The timing of events is not greatly different; core uncovery is predicted at about 26 h for S_2I and 28 h for S_2J. It might be anticipated that a little longer time would be required to recognize that the initial leak rate is >50 gpm, but as was the case for the S_2I, the timing is not considered to be critical in the early stages of the incident. The critical item relates to recognition of loss of RHRSW and subsequent actions to attempt to restore cooling or substitute some other method to remove heat.

Our assignment of emergency action levels is the same as for S_2I and is included in Table 5.7. Comments made for S_2I sequence also apply to S_2J, with exception of reference to loss of ability of the RHR system to pump water through the RHR heat exchangers, for which statements the loss of ability to provide coolant to the secondary side of the RHR heat exchangers should be substituted.

6. RESULTS AND CONCLUSIONS

Eight BWR accident sequences identified in the Reactor Safety Study[3] as dominant contributors to public risk have been investigated in this work. Based on the scenarios given in NUREG/CR-2100 (Ref. 4) and the assumptions incorporated in this study, all eight sequences would eventually result in core melt and containment breach without operator's mitigating actions. Furthermore, the accident progression in each sequence has been correlated to the NRC Emergency Action Level Guidelines.[1]

The primary analytical tool used in this study was the MARCH computer code. The MARCH code, including the modified version used in this study (MARCH 1.4B), contains a number of limitations and deficiencies, particularly when used for BWR accident calculations. While this is a reflection of the code's development for use in early risk assessments in which the uncertainties were not considered to be of major concern, current applications in more detailed severe accident sequence analyses indicate a need for improvements in the modeling, level of detail, and structure, with particular attention to BWR applications. It should also be noted, as set forth in Chap. 2, that the decay heat power, as incorporated in MARCH 1.4B, may significantly overestimate the decay heat for a typical BWR. For these reasons, the present study is primarily useful in providing preliminary assessments of the BWR accident sequences studied and comparing containment failures by overtemperature or by overpressure.

In each sequence, the effect of parameter variations has also been investigated on the accident progression. These parameter variations provide alternate accident sequences dealing with different pipe break sizes, vessel depressurizing rates, options in the MARCH code, containment failure modes, and operator's mitigating actions, such as the use of SLCS in the TC sequence and feedwater makeup flows in small LOCAs. In the TC sequence, it has been shown that no core melt or containment breach would result if SLCS were functioning as designed.

In the small LOCA sequences, the feedwater pumps would be tripped because of high vessel water level based on MARCH calculations. These trips, however, may be at variance with other more detailed thermal-hydraulic codes. The effect of feedwater makeup flows by operator's actions on the accident progression has also been investigated.

Except for sequences TW, S_2I, and S_2J, which assume loss of decay heat removal, overtemperature in the drywell EPA seals has been identified as the dominant containment failure mode following the accident. This failure mode would cause the containment to fail much sooner and at a much lower containment pressure than predicted in the Reactor Safety Study.[3] For sequences TW, S_2I, and S_2J in which there is no decay heat removal, on the other hand, EPA seals failure by overpressurization has been found to be the dominant containment failure mode. Containment failure by overpressurization is assumed to occur at about 1.22 MPa (177 psia).

Although EPA seal failures would cause an earlier containment breach as compared with predictions by the Reactor Safety Study,[3] consequences of containment breach by this failure mode would be considerably mitigated.

The containment pressure drop following EPA seal failures would prevent any further containment structural failure either in the drywell or in the wetwell. For sequences TW, S_2I, and S_2J, in which there is a total loss of residual heat removal systems, the core melt might be considerably delayed because ECCS pumps would still be operational if the wetwell has not been ruptured by overpressurization. For this study, however, it has been conservatively assumed that the wetwell rupture occurred before the EPA seal failure. The amount of fission product releases outside the containment, on the other hand, would be greatly reduced because of deposition of fission products and filtering effect of EPA seals following degraded core accidents.

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Appendix A

SETPOINTS AND FUNCTIONS

Table A.1. Reactor vessel level setpoints and functions

Level	Setpoints [m (in.)]	Functions
8	14.78 (582)	Turbine trips for HPCI, RCIC, main and reactor feed pump turbines
NORMAL	Normal operating level 14.25 (561)	
4	13.69 (539)	Reactor scram, primary containment isolation, start standby gas treatment system, and recirculation pump runback
2	12.10 (476.5)	Initiate HPCI, RCIC systems, MSIV closure, recirculation pump trip
1	9.77 (384.5)	Initiate core spray, LPCI and ADS
TAF	Top of active fuel 9.14 (360)	
BAF	Bottom of active fuel 5.49 (216)	
BOV	Bottom of vessel	

Table A.2. Reactor vessel pressure setpoints and functions

Pressure setpoints (psig)	Functions
1120	Trip recirculation pump
1055	Reactor scram and MSIV closure, and condenser vacuum scram by-pass
1040	High-pressure alarm
920–1010	Normal reactor pressure
850	Trip feedwater pumps
450	Core spray, RHR (LPCI) initiation and valve interlocks
230	RHR initiation (recirculation valve closure)

Table A.3. Primary containment isolation system groups

E.1 Group 1

Main steam isolation valves
Main steam drain isolation valves
Recirculation loop sample isolation valves

E.2 Group 2

Drywell equipment drain discharge isolation valves
Drywell floor drain discharge islolation valve
Torus drain valve
RHRS shutdown cooling supply isolation valves
Reactor head spray isolation valves
RHR flush and drain vent to torus
RHRS-LPCI to reactor valve

E.3 Group 3

Reactor water cleanup supply valve
Reactor water cleanup return valve

E.4 Group 4

HPCI isolation valves

E.5 Group 5

RCIC isolation valves

E.6 Group 6

Containment N_2 purge inlet isolation valves
Drywell and torus main intake and exhaust isolation valves
Drywell and torus exhaust valve bypass to SBGTS
Main exhaust to SBGTS

E.7 Group 7

RCIC steam line drain
RCIC condensate pump drain
HPCI hotwell pump discharge isolation valves
HPCI steamline drain

E.8 Group 8

TIP withdraw command and isolation valve

Appendix B

MARCH CODE INPUT LISTINGS

B.1 TW Sequence

```
 BROWNS FERRY SEQUENCE TW
 $NLMAR
  ITRAN=1,
  IBRK=0,
  ISPRA=1,
  IECC=2,
  IBURN=0,
  IPDTL=7,
  IPLOT=3,
  IU=3,
  VOLC=2.78E05,
  DTINIT=0.01,
  TAP=2.62E06,
 $END
 $NLINTL
 $END
STEEL     CONCRETE
DRYWELL1  DRYWELL2  CONC SHELLMISC STEELMISC CONC.
C
 $NLSLAB
  NMAT=2,
  NSLAB=3,
  NOD=1,4,13,
  DEN(1)=486.924,157.481,
  HC(1)=.1137,.23817,
  TC(1)=25.001,.80024,
  IVL=1,1,2,
  IVR=1,1,2,
  NNO1=3,9,4,
  MAT1=1,2,1,
  MAT2=1,2,1,
  SAREA=18684.,5358.,15982.,
  X(1)=0.,.01,.02083, X(4)=0.,.01,.03,.07,.15,.31,.63,1.27,2.5,
  X(13)=0.,.01,.03,.0625,
  TEMP=12*150.,4*95.,
 $END
 $NLECC
  PUHIO=0.001,
  UHIO=10.0,
  FACMO=0.001,
  ACMO=10.0,
  TMRH=1.0,
  PHH=1150.,
  WHH1=-5000.0,
  TMSIS=10.0,
  PSIS=1150.,
  WSIS1=-600.0,
  TMLH=75.0,
  PLH=450.0,
  WLH1=-40000.0,
  NP=1,
  TM(1)=0.0,
  P(1)=1150.,
  WEC(1)=-50.0,
  STPHH=30.0,
  STPSIS=75.0,
  RWSTM=3.11E06,
  ECCRC=0.90,
  CSPRC=1.0,
  DTSUB=-100.0,
  TRWST=95.0,
 $END
```

```
$NLECX
$END
$NLCSX
$END
$NLCOOL
 JCOOL=1,
 CQR=1.62E06,
 CWPR=6000.0,
 CTPR=150.0,
 CWSR=1.49E05,
 CTSR=95.0,
 TCOOL=0.0,
 NCOOL=2,
 PCOOL=1.80,
 POFF=0.20,
$END
$NLMACE
 NCUB=2,
 NRPV1=2,
 NRPV2=1,
 NRPV3=2,
 ICECUB=-1,
 DTPNT=250.0,
 IDRY=-1,
 IWET=2,
 WPOOL=7.801E06,
 TPOOL=95.0,
 VDRY=3.839E03,
 VTORUS=257700.0,
 WVMAX=5.146E05,
 NSMP=-2,
 NSMP2=2,
 WVMAKS=5.146E05,
 NCAV=1,
 VCAV=4789.1,
 VFLR=15.0,
 IVENT=-21,
 TVNT1=-0.2,
 TVNT2=560.5,
 AVBRK=292.0,
 CVBRK=4.04,
 VC(1)=159000.0,257700.0,
 AREA(1)=1.6399E03,1.098E04,
 HUM(1)=0.2,1.0,
 TEMPO(1)=150.,95.,
 N=7-
 NS(1)=1,1,1,3,3,2,2,
 NC(1)=1,1,1,1,1,1,2,
 NT(1)=1,2,3,-7,-7,-7,-7,
 C1(1)=1.0E6,1.0E6,0.0,400.0,500.0,174.7,174.7,
 C2(1)=0.0,1.333E5,7.59106,5.9297,.583,0.583,0.583,
 C3(1)=95.0,0.0,1192.5,.00694,20.97,20.97,20.97,
 C4(1)=400.0,
 KT(1,2)=1,
 KT(2,1)=1,
$END
$NLBOIL
 NNT=37436,
 NR=35908,
 NDZ=50,
 ISTR=3,
 ISG=0,
 MELMOD=-1,
 IMWA=2,
```

```
ISTM=1,
IHR=1,
ISAT=1,
WDED=3.50E04,
TPUMP1=75.0,
TPUMP2=101.0,
QPUMP1=5.50E07,
QPUMP2=2.75E07,
TMUP1=0.0583,
TMUP2=0.0833,
WMUP1=30735.34,
WMUP2=6147.07,
QZERO=1.1242E10,
H=12,
HO=28.,
DC=15.59,
ACOR=104.833,
ATOT=287.898,
WATBH=97000.,
D=.04692,
DF=.04058,
DH=0.056,
CLAD=.005594,
XOO=8.33E-06,
RHOCU=68.783,
TGOO=546.,
PSET=1120.0,
CSRV=3719.06,
FZCR=-.5,
DPART=0.0208333,
FZMCR=0.05,
FZOCR=0.08,
FZOS1=0.1,
WFE2=7992.,
TFEOO=546.,
FULSG=0.0,
PVSL=1020.0,
TCAV=1210.,
YBRK=45.0,
DTPNTP=250.0,
DTPN=-50.0,
VOLP=2.459E04,
VOLS=9.638E03,
WCST=3.11E06,
F(1)=0.1,0.25,0.47,0.65,0.84,0.96,1.13,1.27,1.295,1.27,1.24,1.21,1.195,
F(14)=1.15,1.11,1.08,1.05,1.03,1.02,1.016,1.017,1.05,1.06,1.061,1.062,
F(26)=1.061,1.06,1.059,1.059,1.06,1.07,1.075,1.095,1.11,1.12,1.185,1.215,
F(38)=1.25,1.26,1.24,1.21,1.15,1.09,1.0,0.87,0.76,0.6,0.41,0.21,0.1,
PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,0.9665,0.408,
VF(1)=10*0.1,
TT=6*546.,
CM=1824.,7992.,8760.,2460.,5550.,24900.,
AH=740.,263.,9225.,400.,7000.,700.,
DD=1.,1.,1.,.17,.02,.546,
AR=150.,263.,165.,0.,-10.,-20.,
$END
$NLHEAD
 WZRC=140397.0,
 WFEC=30447.73,
 WUO2=361837.0,
 WGRID=66750.,
 WHEAD=175927.08,
 DBH=20.915,
 THICK=0.52198,
 COND=8.0005,
```

```
   E1=.8.
   E2=.5.
  $END
  $NLHOT
   IHOT=100.
   DP=0.25.
   FLRMC=3360..
  $END
  $NLINTR
   CAYC=0.01524.
   CPC=1.30.
   DENSC=2.375.
   TIC=308.16.
   FC1=0.441.
   FC2=0.108.
   FC3=0.357.
   FC4=0.027.
   RBR=0.135.
   R0=322.6.
   R=6000.0.
   HIM=0.2.
   HIO=0.09.
   WALL=1000..
  $END
 #
```

B.2 TC Sequence

```
  BROWNS FERRY SEQUENCE TC
 $NLMAR
  ITRAN=2.
  IBRK=0.
  ICHRK=1.
  ISPRA=1.
  IECC=2.
  IBURN=0.
  IPOTL[illegible].
  IPLOT=3.
  IU=3.
  VOLC=2.78E05.
  TAP=2.62E06.
 $END
 $NLINTL
 $END
STEEL      CONCRETE
DRYWELL1   DRYWELL2   CONC SHELLMISC STEELMISC CONC.
C
 $NLSLAB
  NMAT=2.
  NSLAB=3.
  NOD=1.4.13.
  DEN(1)=486.924.157.481.
  HC(1)=.1137..23817.
  TC(1)=25.001..80024.
  IVL=1.1.2.
  IVR=1.1.2.
  NNO1=3.9.4.
  MAT1=1.2.1.
  MAT2=1.2.1.
  SAREA=18684..5358..15982..
  X(1)=0...01..02083. X(4)=0...01..03..07..15..31..63.1.27.2.5.
  X(13)=0...01..03..0625.
  TEMP=12*150..4*95..
 $END
```

```
$NLECC
 PUHIO=0.001,
 UHIO=10.0,
 PACMO=0.001,
 ACMO=10.0,
 PHH=1150.0,
 WHH1=-5650.0,
 PSIS=1150.0,
 STPHH=481.0,
 RWSTM=3.05E06,
 ECCRC=0.95,
 CSPRC=1.0,
 DTSUB=-100.0,
 TRWST=95.0,
$END
$NLECX
 EQR=2.80E08,
 EWPR=5.96E05,
 EWSR=1.328E06,
 ETP1R=165.0,
 ETS1R=95.0,
$END
$NLCSX
 SQR=1.847E08,
 SWPR=61090.0,
 SWSR=65833.0,
 STP1R=165.0,
 STS1R=95.0,
$END
$NLCOOL
 JCOOL=1,
 CQR=1.62E06,
 CWPR=6000.0,
 CTPR=150.0,
 CWSR=1.49E05,
 CTSR=95.0,
 TCOOL=0.0,
 NCOOL=2,
 PCOOL=1.80,
 POFF=0.20,
$END
$NLMACE
 NCUB=2,
 NRPV1=2,
 NRPV2=1,
 ICECUB=-1,
 DTPNT=1000.0,
 IDRY=-1,
 IWET=2,
 WPOOL=7.801E06,
 TPOOL=95.0,
 VDRY=3.839E03,
 VTORUS=257700.0,
 WVMAX=4.63E05,
 NSMP=-2,
 NSMP2=2,
 WVMAKS=4.0E05,
 NCAV=1,
 VCAV=4789.1,
 VFLR=15.0,
 AVBRK=292.0,
 CVBRK=4.04,
 VC(1)=159000.0,257700.0,
 AREA(1)=1.6399E03,1.098E04,
```

```
 HUM(1)=0.2,1.0,
 TEMPO(1)=150.,95.,
 N=6,
 NS(1)=1,1,3,3,2,2,
 NC(1)=2,1,1,1,1,2,
 NT(1)=1,3,-7,-7,-7,-7,
 C1(1)=0.0,1.0E06,400.0,500.0,174.7,174.7,
 C2(1)=7450.0,7.59106,5.9297,0.583,0.583,0.583,
 C3(1)=95.0,1192.5,0.00694,20.97,20.97,20.97,
 C4(1)=400.0,
 KT(1,2)=1,
 KT(2,1)=1,
$END
$NLBOIL
 NNT=37436,
 NR=35908,
 ISTR=3,
 ISG=0,
 MELMOD=-1,
 IMWA=3,
 IHR=1,
 ISAT=1,
 KRPS=1,
 TRPS=60.0,
 ANSK=0.05,
 TDK=1.0E04,
 YT=28.5,
 YB=8.0,
 DTK=1.0,
 WDED=3.50E04,
 TPUMP1=75.0,
 TPUMP2=101.0,
 QPUMP1=5.50E07,
 QPUMP2=2.75E07,
 TMUP1=1.0,
 TMUP2=5.0,
 WMUP1=30735.34,
 WMUP2=6147.07,
 QZERO=1.1242E10,
 H=12.5,
 HO=28.5,
 DC=15.59,
 ACOR=104.833,
 ATOT=287.898,
 WATBH=97000.,
 D=.04692,
 DF=.04058,
 DH=0.056,
 CLAD=.005594,
 XOO=8.33E-06,
 RHOCU=68.783,
 TGOO=546.,
 PSET=1120.0,
 CSRV=3719.06,
 FDCR=-.5,
 DPART=0.0208333,
 DUO2=0.04058,
 FZMCR=0.05,
 FZOCR=0.08,
 FZOS1=0.1,
 WFE2=7992.,
 TFEOC[illegible]6.,
 FULSG[illegible].0,
 PVSL=1020.0,
```

```
 TCAV=1210.,
 ABRK=0.0,
 YBRK=45.0,
 DTPNTH=1000.0,
 DTPN=-500.0,
 VOLP=2.459E04,
 VOLS=9.638E03,
 WCST=3.05E06,
 F(1)=0.1,0.25,0.47,0.65,0.84,0.96,1.13,1.27,1.295,1.27,1.24,1.21,1.185,
 F(14)=1.15,1.11,1.08,1.05,1.03,1.02,1.016,1.017,1.05,1.06,1.061,1.062,
 F(26)=1.061,1.06,1.059,1.059,1.06,1.07,1.075,1.095,1.11,1.12,1.185,1.215,
 F(38)=1.25,1.26,1.24,1.21,1.15,1.09,1.0,0.87,0.76,0.6,0.41,0.21,0.1,
 PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,0.9665,0.408,
 VF(1)=10*0.1,
 TT=6*546.,
 DD=1.,1.,1.,.17,.02,.546,
 AR=150.,263.,165.,0.,-10.,-20.,
$END
$NLHEAD
 WZRC=140397.0,
 WFEC=30447.73,
 WUO2=361837.0,
 WGRID=66750.,
 WHEAD=175927.08,
 DBH=20.915,
 THICK=0.52198,
 COND=8.0005,
 E1=.8,
 E2=.5,
$END
$NLHOT
 IHOT=100,
 DP=0.25,
 FLRMC=3360.,
$END
$NLINTR
 CAYC=0.01524,
 CPC=1.30,
 DENSC=2.375,
 TIC=308.16,
 FC1=0.441,
 FC2=0.108,
 FC3=0.357,
 FC4=0.027,
 RBR=0.135,
 R0=322.6,
 R=6000.0,
 HIM=0.2,
 HIO=0.09,
 WALL=1000.,
$END
*
```

B.3 TQUV Sequence

```
 BROWNS FERRY SEQUENCE TQUV
$NLMAR
 ITRAN=1,
 IBRK=0,
 ISPRA=1,
 IECC=0,
 IBURN=0,
 NINTER=100,
```

```
  IPDTL=7,
  IPLOT=3,
  IU=3,
  VOLC=416700.0,
  DTINIT=0.01,
  TAP=2.62E06,
 $END
 $NLINTL
 $END
STEEL      CONCRETE
DRYWELL1   DRYWELL2  CONC SHELLMISC STEELMISC CONC.
C
 $NLSLAB
  NMAT=2,
  NSLAB=3,
  NOD=1,4,13,
  DEN(1)=486.924,157.481,
  HC(1)=.1137,.23817,
  TC(1)=25.001,.80024,
  IVL=1,1,2,
  IVR=1,1,2,
  NNO1=3,9,4,
  MAT1=1,2,1,
  MAT2=1,2,1,
  SAREA=18684.,5358.,15982.,
  X(1)=0.,.01,.02083, X(4)=0.,.01,.03,.07,.15,.31,.63,1.27,2.5,
  X(13)=0.,.01,.03,.0625,
  TEMP=12*150.,4*95.,
 $END
 $NLECC
  PUHIO=0.001,
  UHIO=3.11E04,
  PACMO=0.001,
  ACMO=3.11E04,
  PHH=1120.,
  WHH1=0.0,
  PSIS=1120.,
  WSIS1=0.0,
  PLH=1120.,
  WLH1=0.0,
  STPHH=240.,
  RWSTM=3.11E06,
  ECCRC=0.64,
  CSPRC=1.0,
  DTSUB=-100.,
  WTCAV=100.,
  TRWST=95.0,
 $END
 $NLECX
 $END
 $NLCSX
 $END
 $NLCOOL
 $END
 $NLMACE
  NCUB=2,
  NRPV1=2,
  NRPV2=1,
  ICECUB=-1,
  DTPNT=20.0,
  IDRY=-1,
  IWET=2,
  WPOOL=7.801E06,
  TPOOL=95.0,
  VDRY=3.839E03,
```

```
 VTORUS=257700.0,
 WVMAX=5.146E05,
 NSMP=-2,
 NSMP2=2,
 NCAV=-1,
 VCAV=4789.1,
 VFLR=15.0,
 AVBRK=292.0,
 CVBRK=4.04,
 VC(1)=159000.0,257700.0,
 AREA(1)=1.6399E03,1.098E04,
 HUM(1)=0.2,1.0,
 TEMPO(1)=150.,95.,
 N=10,
 NS(1)=1,1,1,3,3,2,2,2,2,2,
 NC(1)=1,1,1,1,1,1,1,2,2,2,
 NT(1)=1,2,3,-7,-7,-7,-7,-7,-7,-7,
 C1(1)=1.E6,1.E6,0.,400.,500.,139.7,189.7,139.7,189.7,159.7,
 C2(1)=0.,1.333E5,7.59106,5.9297,.583,5.9297,.583,5.9297,.583,.583,
 C3(1)=95.,0.,1192.5,.00694,20.97,.00694,20.97,6.94E-4,.0833,6.94E-3,
 C4(1)=0.0,0.0,0.0,14.7,0.0,14.7,0.0,14.7,14.7,14.7,
 KT(1,2)=1,
 KT(2,1)=1,
 STPECC=0.0,
$END
$NLBOIL
 NNT=37436,
 NR=35908,
 NDZ=50,
 ISTR=3,
 ISG=0,
 IMWA=1,
 IHR=1,
 WDED=3.50E04,
 QZERO=1.1242E10,
 H=12,
 HO=28.,
 DC=15.59,
 ACOR=104.833,
 ATOT=287.898,
 WATBH=97000.,
 D=.04692,
 DF=.04058,
 DH=0.056,
 CLAD=.005594,
 XOO=8.33E-06,
 RHOCU=68.783,
 TGOO=546.,
 PSET=1050.0,
 CSRV=3380.,
 FDCR=-.5,
 DPART=0.0208333,
 FZMCR=0.05,
 FZOCR=0.08,
 FZOS1=0.1,
 WFE2=7992.,
 TFEOO=546.,
 FULSG=0.0,
 PVSL=1000.,
 TCAV=1210.,
 ABRK=0.0,
 YBRK=45.0,
 DTPNTB=5.0,
 DTPN=-5.,
 VOLP=2.459E04,
```

```
 VOLS=9.638E03,
 WCST=3.11E06,
 F(1)=0.1,0.25,0.47,0.65,0.84,0.96,1.13,1.27,1.295,1.27,1.24,1.21,1.195,
 F(14)=1.15,1.11,1.08,1.05,1.03,1.02,1.016,1.017,1.05,1.06,1.061,1.062,
 F(26)=1.061,1.06,1.059,1.059,1.06,1.07,1.075,1.095,1.11,1.12,1.185,1.215,
 F(38)=1.25,1.26,1.24,1.21,1.15,1.09,1.0,0.87,0.76,0.6,0.41,0.21,0.1,
 PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,0.9665,0.408,
 VF(1)=10*0.1,
 TT=6*546.,
 CM=1824.,7992.,8760.,2460.,5550.,24900.,
 AH=740.,263.,9225.,400.,7000.,700.,
 DD=1.,1.,1.,.17,.02,.546,
 AR=150.,263.,165.,0.,-10.,-20.,
$END
$NLHEAD
 WZRC=140397.0,
 WFEC=30447.73,
 WUO2=361837.0,
 WGRID=66750.,
 WHEAD=175927.08,
 DBH=20.915,
 THICK=0.52198,
 COND=8.0005,
 E1=.8,
 E2=.5,
$END
$NLHOT
 IHOT=100,
 DP=0.25,
 FLRMC=3360.,
$END
$NLINTR
 CAYC=0.01524,
 CPC=1.30,
 DENSC=2.375,
 TIC=308.16,
 FC1=0.441,
 FC2=0.108,
 FC3=0.357,
 FC4=0.027,
 RBR=0.135,
 R0=322.6,
 R=6000.0,
 HIM=0.2,
 HIO=0.09,
 WALL=1000.,
$END
#
```

B.4 AE Sequence

```
 BROWNS FERRY SEQUENCE AE (D = 25.70 IN)
$NLMAR
 ITRAN=0,
 ICBRK=1,
 IECC=2,
 NPAIR=2,
 IBURN=0,
 IPDTL=7,
 IPLOT=3,
 IU=3,
```

```
 VOLC=2.78E05,
 TAP=2.62E06,
$END
$NLINTL
 T(1)=0.0,0.5,
 W(1)=1.2087E06,0.96696E06,
 EW(1)=370.0,580.0,
$END
STEEL     CONCRETE
DRYWELL1  DRYWELL2  CONC SHELLMISC STEELMISC CONC.
C
$NLSLAB
 NMAT=2,
 NSLAB=3,
 NOD=1,4,13,
 DEN(1)=486.924,157.481,
 HC(1)=.1137,.23817,
 TC(1)=25.001,.80024,
 IVL=1,1,2,
 IVR=1,1,2,
 NNO1=3,9,4,
 MAT1=1,2,1,
 MAT2=1,2,1,
 SAREA=18684.,5358.,15982.,
 X(1)=0.,.01,.02083, X(4)=0.,.01,.03,.07,.15,.31,.63,1.27,2.5,
 X(13)=0.,.01,.03,.0625,
 TEMP=12*150.,4*95.,
$END
$NLECC
 PLH=450.0,
 RWSTM=3.05E06,
 ECCRC=1.00,
 CSPRC=2.0,
 DTSUB=-100.0,
 WTCAV=-100.0,
$END
$NLECX
 EQR=0.70E08,
 EWPR=5.96E05,
 EWSR=1.328E06,
 ETP1R=165.0,
 ETS1R=95.0,
$END
$NLCSX
 SQR=1.847E08,
 SWPR=61090.0,
 SWSR=65833.0,
 STP1R=165.0,
 STS1R=95.0,
$END
$NLCOOL
 JCOOL=1,
 CQR=1.62E06,
 CWPR=6000.0,
 CTPR=150.0,
 CWSR=1.49E05,
 CTSR=95.0,
 TCOOL=0.0,
 NCOOL=2,
 PCOOL=1.80,
 POFF=0.20,
$END
$NLMACE
 NCUB=2,
```

```
 NRPV1=1,
 NRPV2=1,
 ICECUB=-1,
 IDRY=-1,
 IWET=2,
 WPOOL=7.801E06,
 TPOOL=95.0,
 VDRY=3.839E03,
 VTORUS=257700.0,
 WVMAX=5.146E05,
 NSMP=-2,
 NSMP2=2,
 NCAV=1,
 VCAV=4789.1,
 VFLR=15.0,
 AVBRK=292.0,
 CVBRK=4.04,
 VC(1)=159000.0,257700.0,
 AREA(1)=1.6399E03,1.098E04,
 HUM(1)=0.2,1.0,
 TEMPO(1)=150.,95.,
 N=7,
 NS(1)=1,1,1,3,3,2,2,
 NC(1)=2,1,1,1,1,1,2,
 NT(1)=1,2,3,-7,-7,-7,-7,
 C1(1)=0.000,1.0E6,0.0,400.0,500.0,174.7,174.7,
 C2(1)=7450.0,4.67E06,7.59106,5.9297,0.583,0.583,0.583,
 C3(1)=95.0,0.0,1192.5,.00694,1.0485,20.97,20.97,
 C4(1)=400.0,
 KT(1,2)=1,
 KT(2,1)=1,
$END
$NLBOIL
 NNT=37436,
 NR=35908,
 ISTR=3,
 IMWA=3,
 ISTM=0,
 ISG=0,
 ISAT=1,
 TPM=1.74,
 WDED=3.50E04,
 TPUMP1=75.0,
 TPUMP2=101.0,
 QPUMP1=5.50E07,
 QPUMP2=2.75E07,
 TMUP1=1.0,
 TMUP2=5.0,
 WMUP1=30735.34,
 WMUP2=6147.07,
 QZERO=1.1242E10,
 H=12.0,
 HO=-5.0,
 DC=15.59,
 ACOR=104.833,
 ATOT=287.898,
 WATBH=97000.,
 D=.04692,
 DF=.04058,
 DH=0.056,
 CLAD=.005594,
 XOO=8.33E-06,
 RHOCU=68.783,
 TGOO=227.75,
 PSET=1120.0,
 CSRV=0.1215,
```

```
 FDCR=-.5,
 DPART=0.0208333,
 DUO2=0.04058,
 FZMCR=0.05,
 FZOCR=0.08,
 FZOS1=0.1,
 WFE2=7992.,
 TFE00=546.,
 FULSG=0.0,
 PSG=8.555E-02,
 PVSL=19.969,
 TCAV=1210.0,
 YLEG=8.33,
 ABRK=5.0367237,
 YBRK=8.33,
 VOLP=2.459E04,
 VOLS=9.638E03.
 WCST=3.05E06,
 F(1)=0.1,0.25,0.47,0,65,0.84,0.96,1.13,1.27,1.295,1.27,1.24,1.21,1.195,
 F(14)=1.15,1.11,1.08,1.05,1.03,1.02,1.016,1.017,1.05,1.06,1.061,1.062,
 F(26)=1.061,1.06,1.059,1.059,1.06,1.07,1.075,1.095,1.11,1.12,1.185,1.215,
 F(38)=1.25,1.26,1.24,1.21,1.15,1.09,1.0,0.87,0.76,0.6,0.41,0.21,0.1,
 PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,0.9665,0.408,
 VF(1)=10*0.1,
 TT=6*546.,
 CM=1824.,7992.,8760.,2460.,5550.,24900.,
 AH=740.,263.,9225.,400.,7000.,700.,
 DD=1.,1.,1.,.17,.02,.546,
 AR=150.,263.,165.,0.,-10.,-20.,
$END
$NLHEAD
 WZRC=140397.0,
 WFEC=30447.73,
 WUO2=361837.0,
 WGRID=66750.,
 WHEAD=175927.08,
 DBH=20.915,
 TVSL=540.0,
 THICK=0.52198,
 COND=8.0005,
 E1=.8,
 E2=.5,
$END
$NLHOT
 DP=0.25,
 FLRMC=12000.,
 WTR=2.44E05,
 TPOOLH=225.94,
$END
$NLINTR
 CAYC=0.01524,
 CPC=1.30,
 DENSC=2.375,
 TIC=308.16,
 FC1=0.441,
 FC2=0.108,
 FC3=0.357,
 FC4=0.027,
 RBR=0.135,
 R0=322.6,
 R=6000.0,
 HIM=0.2,
 HI0=0.09,
 WALL=1000.,
$END
```

B.5 S_1E Sequence

```
 BROWNS FERRY SEQUENCE S1E (D=3.58 IN)
 $NLMAR
  ITRAN=1,
  IBRK=1,
  ISPRA=1,
  IECC=2,
  NINTER=300,
  IBURN=0,
  IPDTL=7,
  IPLOT=3,
  IU=3,
  VOLC=2.78E05,
  DTINIT=0.01,
  TAP=2.62E06,
 $END
 $NLINTL
 $END
STEEL      CONCRETE
DRYWELL1   DRYWELL2   CONC SHELLMISC STEELMISC CONC.
C
 $NLSLAB
  NMAT=2,
  NSLAB=3,
  NOD=1,4,13,
  DEN(1)=486.924,157.481,
  HC(1)=.1137,.23817,
  TC(1)=25.001,.80024,
  IVL=1,1,2,
  IVR=1,1,2,
  NNO1=3,9,4,
  MAT1=1,2,1,
  MAT2=1,2,1,
  SAREA=18684.,5358.,15982.,
  X(1)=0.,.01,.02083, X(4)=0.,.01,.03,.07,.15,.31,.63,1.27,2.5,
  X(13)=0.,.01,.03,.0625,
  TEMP=12*150.,4*95.,
 $END
 $NLECC
  PUHIO=0.001,
  UHIO=10.0,
  PACMO=0.001,
  ACMO=10.0,
  PHH=1150.0,
  WHH1=-600.0,
  PSIS=1150.0,
  WSIS1=-50.0,
  PLH=1150.0,
  ECCRC=0.95,
  RWSTM=3.05E06,
  CSPRC=1.0,
  DTSUB=-100.0,
  TRWST=95.0,
 $END
 $NLECX
  EQR=2.80E08,
  EWPR=5.96E05,
  EWSR=1.328E06,
  ETP1R=165.0,
  ETS1R=95.0,
 $END
 $NLCSX
  SQR=1.847E08,
  SWPR=61090.0,
```

```
 SWSR=65833.0,
 STP1R=165.0,
 STS1R=95.0,
$END
$NLCOOL
 JCOOL=1,
 CQR=1.62E06,
 CWPR=6000.0,
 CTPR=150.0,
 CWSR=1.49E05,
 CTSR=95.0,
 TCOOL=0.0,
 NCOOL=2,
 PCOOL=1.80,
 POFF=0.20,
$END
$NLMACE
 NCUB=2,
 NRPV1=1,
 NRPV2=-1,
 NRPV3=2,
 ICECUB=-1,
 DTPNT=500.0,
 IDRY=-1,
 IWET=2,
 WPOOL=7.801E06,
 TPOOL=95.0,
 VDRY=3.839E03,
 VTORUS=257700.0,
 WVMAX=5.146E05,
 NSMP=-2,
 NSMP2=2,
 WVMAKS=5.146E05,
 NCAV=1,
 VCAV=4789.1,
 VFLR=15.0,
 IVENT=-12,
 TVNT1=-0.2,
 TVNT2=30000.0,
 AVBRK=292.0,
 CVBRK=4.04,
 VC(1)=159000.0,257700.0,
 AREA(1)=1.6399E03,1.098E04,
 HUM(1)=0.2,1.0,
 TEMPO(1)=150.,95.,
 N=7,
 NS(1)=1,1,1,3,3,2,2,
 NC(1)=2,1,1,1,1,1,2,
 NT(1)=1,2,3,-7,-7,-7,-7,
 C1(1)=0.000,1.0E6,0.0,400.0,500.0,174.7,174.7,
 C2(1)=7450.0,4.67E06,7.59106,5.9297,0.583,0.583,0.583,
 C3(1)=95.0,0.0,1192.5,.00694,20.97,20.97,20.97,
 C4(1)=400.0,
 KT(1,2)=1,
 KT(2,1)=1,
$END
$NLBOIL
 NNT=37436,
 NR=35908,
 ISTR=3,
 ISG=0,
 IMWA=3,
 IHR=1,
 ISAT=1,
```

```
 TPM=1.74,
 WDED=3.50E04,
 QZERO=1.1242E10,
 H=12.0,
 HO=28.,
 DC=15.59,
 ACOR=104.833,
 ATOT=287.898,
 WATBH=97000.,
 D=.04692,
 DF=.04058,
 DH=0.056,
 CLAD=.005594,
 XGO=8.33E-06,
 RHOCU=68.783,
 TGOO=546.0,
 PSET=1120.0,
 CSRV=3719.06,
 FDCR=-.5,
 DPART=0.0208333,
 DUO2=0.04058,
 FZMCR=0.05,
 FZOCR=0.08,
 FZOS1=0.1,
 WFE2=7992.,
 TFEOO=546.,
 FULSG=0.0,
 PVSL=1020.0,
 TCAV=1210.0,
 YLEG=8.33,
 ABRK=0.0699,
 YBRK=8.33,
 VOLP=2.459E04,
 VOLS=9.638E03,
 WCST=3.05E06,
 F(1)=0.1,0.25,0.47,0.65,0.84,0.96,1.13,1.27,1.295,1.27,1.24,1.21,1.195,
 F(14)=1.15,1.11,1.08,1.05,1.03,1.02,1.016,1.017,1.05,1.06,1.061,1.062,
 F(26)=1.061,1.06,1.059,1.059,1.06,1.07,1.075,1.095,1.11,1.12,1.185,1.215,
 F(38)=1.25,1.26,1.24,1.21,1.15,1.09,1.0,0.87,0.76,0.6,0.41,0.21,0.1,
 PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,0.9665,0.408,
 VF(1)=10*0.1,
 TT=6*546.,
 CM=1824.,7992.,8760.,2460.,5550.,24900.,
 AH=740.,263.,9225.,400.,7000.,700.,
 DD=1.,1.,1.,.17,.02,.546,
 AR=150.,263.,165.,0.,-10.,-20.,
$END
$NLHEAD
 WZRC=140397.0,
 WFEC=30447.73,
 WUO2=361837.0,
 WGRID=66750.,
 WHEAD=175927.08,
 DBH=20.915,
 THICK=0.52198,
 COND=8.0005,
 E1=.8,
 E2=.5,
$END
$NLHOT
 IHOT=100,
 DP=0.25,
 FLRMC=3360.,
$END
```

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

B.6 S_2E Sequence

```
 BROWNS FERRY SEQUENCE S2E
$NLMAR
 ITRAN=1,
 IBRK=1,
 ISPRA=1,
 IECC=2,
 IBURN=0,
 IPOTL=7,
 IPLOT=3,
 IU=3,
 VOLC=2.78E05,
 TAP=2.62E06,
$END
$NLINTL
$END
STEEL     CONCRETE
DRYWELL1  DRYWELL2  CONC SHELLMISC STEELMISC CONC.
C
$NLSLAB
 NMAT=2,
 NSLAB=3,
 NOD=1,4,13,
 DEN(1)=486.924,157.481,
 HC(1)=.1137,.23817,
 TC(1)=25.001,.80024,
 IVL=1,1,2,
 IVR=1,1,2,
 NNO1=3,9,4,
 MAT1=1,2,1,
 MAT2=1,2,1,
 SAREA=18684.,5358.,15982.,
 X(1)=0.,.01,.02083, X(4)=0.,.01,.03,.07,.15,.31,.63,1.27,2.5,
 X(13)=0.,.01,.03,.0625,
 TEMP=12*150.,4*95.,
$END
$NLECC
 PUHIO=0.001,
 UHIO=3.11E04,
 PACMO=0.001,
 ACMO=3.11E04,
 PHH=1150.0,
 WHH1=-50.0,
```

```
 PLH=1150.0,
 WLH1=-31594.784,
 STPLH=0.0,
 RWSTM=3.05E06,
 ECCRC=0.0,
 CSPRC=1.0,
 DTSUB=-100.0,
 TRWST=95.0,
$END
$NLECX
 EQR=2.80E08,
 EWPR=5.96E05,
 EWSR=1.328E06,
 ETP1R=165.0,
 ETS1R=95.0,
$END
$NLCSX
 SQR=1.847E08,
 SWPR=61090.0,
 SWSR=65833.0,
 STP1R=165.0,
 STS1R=95.0,
$END
$NLCOOL
 JCOOL=1,
 CQR=1.62E06,
 CWPR=6000.0,
 CTPR=150.0,
 CWSR=1.49E05,
 CTSR=95.0,
 TCOOL=0.0,
 NCOOL=2,
 PCOOL=1.80,
 POFF=0.20,
$END
$NLMACE
 NCUB=2,
 NRPV1=1,
 NRPV3=2,
 ICECUB=-1,
 DTPNT=100.0,
 IDRY=-1,
 IWET=2,
 WPOOL=7.801E06,
 TPOOL=95.0,
 VDRY=3.839E03,
 VTORUS=257700.0,
 WVMAX=5.146E05,
 NSMP=-2,
 NSMP2=2,
 WVMAKS=5.146E05,
 NCAV=1,
 VCAV=4789.1,
 VFLR=15.0,
 IVENT=-12,
 TVNT1=-0.2,
 TVNT2=30000.0,
 AVBRK=292.0,
 CVBRK=4.04,
 VC(1)=159000.0,257700.0,
 AREA(1)=1.6399E03,1.098E04,
 HUM(1)=0.2,1.0,
 TEMPO(1)=150.,95.,
 N=7,
```

```
 NS(1)=1,1,1,3,3,2,2,
 NC(1)=2,1,1,1,1,1,2,
 NT(1)=1,2,3,-7,-7,-7,-7,
 C1(1)=0.000,1.0E6,0.0,400.0,500.0,174.7,174.7,
 C2(1)=7450.0,4.67E06,7.59106,5.9297,0.583,0.583,0.583,
 C3(1)=95.0,0.0,1192.5,.00694,1.0485,20.97,20.97,
 C4(1)=400.0,
 KT(1,2)=1,
 KT(2,1)=1,
$END
$NLBOIL
 NNT=37436,
 NR=35908,
 ISTR=3,
 ISG=0,
 IMWA=3,
 IHR=1,
 ISAT=1,
 TPM=1.74,
 WDED=3.50E04,
 TPUMP1=75.0,
 TPUMP2=101.0,
 QPUMP1=5.50E07,
 QPUMP2=2.75E07,
 QZERO=1.1242E10,
 H=12.0,
 HO=28.,
 DC=15.59,
 ACOR=104.833,
 ATOT=287.898,
 WATBH=97000.,
 D=.04692,
 DF=.04058,
 DH=0.056,
 CLAD=.005594,
 XOO=8.33E-06,
 RHOCU=68.783,
 TGOO=546.0,
 PSET=1120.0,
 CSRV=3719.06,
 FDCR=-.5,
 DPART=0.0208333,
 DUO2=0.04058,
 FZMCR=0.05,
 FZOCR=0.08,
 FZOS1=0.1,
 WFE2=7992.,
 TFEOO=546.,
 FULSG=0.0,
 PVSL=1020.0,
 TCAV=1210.0,
 ABRK=0.0218166,
 YBRK=8.33,
 VOLP=2.459E04,
 VOLS=9.638E03,
 WCST=3.05E06,
 F(1)=0.1,0.25,0.47,0.65,0.84,0.96,1.13,1.27,1.295,1.27,1.24,1.21,1.195,
 F(14)=1.15,1.11,1.08,1.05,1.03,1.02,1.016,1.017,1.05,1.06,1.061,1.062,
 F(26)=1.061,1.06,1.^59,1.059,1.06,1.07,1.075,1.095,1.11,1.12,1.185,1.215,
 F(38)=1.25,1.26,1.24,1.21,1.15,1.09,1.0,0.87,0.76,0.6,0.41,0.21,0.1,
 PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,0.9665,0.408,
 VF(1)=10*0.1,
 TT=6*546.,
 CM=1824.,7992.,8760.,2460.,5550.,24900.,
 AH=740.,263.,9225.,400.,7000.,700.,
 DD=1.,1.,1.,.17,.02,.546,
```

```
 AR=150.,263.,165.,0.,-10.,-20.,
$END
$NLHEAD
 WZRC=140397.0,
 WFEC=30447.73,
 WUO2=361837.0,
 WGRID=66750.,
 WHEAD=175927.08,
 DBH=20.915,
 THICK=0.52198,
 COND=8.0005,
 E1=.8,
 E2=.5,
$END
$NLHOT
 IHOT=100,
 DP=0.25,
 FLRMC=3360.,
$END
$NLINTR
 CAYC=0.01524,
 CPC=1.30,
 DENSC=2.375,
 TIC=308.16,
 FC1=0.441,
 FC2=0.108,
 FC3=0.357,
 FC4=0.027,
 RBR=0.135,
 R0=322.6,
 R=6000.0,
 HIM=0.2,
 HIO=0.09,
 WALL=1000.,
$END
#
```

B.7 S_2I Sequence

```
  BROWNS FERRY SEQUENCE S2I
 $NLMAR
  ITRAN=1,
  IBRK=1,
  ISPRA=1,
  IECC=2,
  IBURN=0,
  IPDTL=7,
  IPLOT=3,
  IU=3,
  VOLC=2.78E05,
  TAP=2.62E06,
 $END
 $NLINTL
 $END
STEEL      CONCRETE
DRYWELL1   DRYWELL2  CONC SHELLMISC STEELMISC CONC.
C
 $NLSLAB
  NMAT=2,
  NSLAB=3,
  NOD=1,4,13,
  DEN(1)=486.924,157.481,
  HC(1)=.1137,.23817,
```

```
 TC(1)=25.001,.80024,
 IVL=1,1,2,
 IVR=1,1,2,
 NNO1=3,9,4,
 MAT1=1,2,1,
 MAT2=1,2,1,
 SAREA=18684.,5358.,15982.,
 X(1)=0.,.01,.02083, X(4)=0.,.01,.03,.07,.15,.31,.63,1.27,2.5,
 X(13)=0.,.01,.03,.0625,
 TEMP=12*150.,4*95.,
$END
$NLECC
 PUHIO=0.001,
 UHIO=10.0,
 PACMO=0.001,
 ACMO=10.0,
 PHH=1160.0,
 WHH1=-5000.0,
 TMSIS=10.0,
 PSIS=1160.0,
 WSIS1=-600.0,
 PLH=1160.0,
 WLH1=-50.0,
 NP=2,
 TM(1)=450.0,
 TM(2)=0.0,
 P(1)=450.0,
 P(2)=1150.0,
 WEC(1)=12500.0,
 WEC(2)=31594.784,
 STP(2)=0.0,
 RWSTM=3.05E06,
 CSPRC=2.0,
 DTSUB=-100.0,
 TRWST=95.0,
$END
$NLECX
$END
$NLCSX
$END
$NLCOOL
 JCOOL=1,
 CQR=1.62E06,
 CWPR=6000.0,
 CTPR=150.0,
 CWSR=1.49E05,
 CTSR=95.0,
 TCOOL=0.0,
 NCOOL=2,
 PCOOL=1.80,
 POFF=0.20,
$END
$NLMACE
 NCUB=2,
 NRPV1=1,
 NRPV2=1,
 ICECUB=-1,
 DTPNT=500.0,
 IDRY=-1,
 IWET=2,
 WPOOL=7.801E06,
 TPOOL=95.0,
 VDRY=3.839E03,
 VTORUS=257700.0,
```

```
 WVMAX=5.146E05,
 NSMP=-2,
 NSMP2=2,
 WVMAKS=5.146E05,
 NCAV=1,
 VCAV=4789.1,
 VFLR=15.0,
 AVBRK=292.0,
 CVBRK=4.04,
 VC(1)=159000.0,257700.0,
 AREA(1)=1.6399E03,1.098E04,
 HUM(1)=0.2,1.0,
 TEMPO(1)=150.,95.,
 N=7,
 NS(1)=1,1,1,3,3,2,2,
 NC(1)=1,1,1,1,1,1,2,
 NT(1)=1,2,3,-7,-7,-7,-7,
 C1(1)=1.0E6,1.0E6,0.0,400.0,500.0,174.7,174.7,
 C2(1)=2.0E04,4.67E06,7.59106,5.9297,0.583,0.583,0.583,
 C3(1)=95.0,0.0,1192.5,.00694,20.97,20.97,20.97,
 C4(1)=400.0,
 KT(1,2)=1,
 KT(2,1)=1,
$END
$NLBOIL
 NNT=37436,
 NR=35908,
 ISTR=3,
 ISG=0,
 IMWA=3,
 IHR=1,
 ISAT=1,
 WDED=3.50E04,
 TPUMP1=75.0,
 TPUMP2=101.0,
 QPUMP1=5.50E07,
 QPUMP2=2.75E07,
 QZERO=1.1242E10,
 H=12.0,
 HO=28.,
 DC=15.59,
 ACOR=104.833,
 ATOT=287.898,
 WATBH=97000.,
 D=.04692,
 DF=.04058,
 DH=0.056,
 CLAD=.005594,
 XOO=8.33E-06,
 RHOCU=68.783,
 TGOO=546.0,
 PSET=1120.0,
 CSRV=3719.06,
 FDCR=-.5,
 DPART=0.0208333,
 DUO2=0.04058,
 FZMCR=0.05,
 FZOCR=0.08,
 FZOS1=0.1,
 WFE2=7992.,
 TFEOO=546.,
 FULSG=0.0,
 PVSL=1020.0,
 TCAV=575.0,
```

```
 ABRK=0.0055,
 YBRK=8.33,
 DTPNTH=500.0,
 DTPN=-50.0,
 VOLP=2.459E04,
 VOLS=9.638E03,
 WCST=3.05E06,
 F(1)=0.1,0.25,0.47,0.65,0.84,0.96,1.13,1.27,1.295,1.27,1.24,1.21,1.195,
 F(14)=1.15,1.11,1.08,1.05,1.03,1.02,1.016,1.017,1.05,1.06,1.061,1.062,
 F(26)=1.061,1.06,1.059,1.059,1.06,1.07,1.075,1.095,1.11,1.12,1.185,1.215,
 F(38)=1.25,1.26,1.24,1.21,1.15,1.09,1.0,0.87,0.76,0.6,0.41,0.21,0.1,
 PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,0.9665,0.408,
 VF(1)=10*0.1,
 TT=6*546.,
 CM=1824.,7992.,8760.,2460.,5550.,24960.,
 AH=740.,263.,9225.,400.,7000.,700.,
 DD=1.,1.,1.,.17,.02,.546,
 AR=150.,263.,165.,0.,-10.,-20.,
$END
$NLHEAD
 WZRC=140397.0,
 WFEC=30447.73,
 WUO2=361837.0,
 WGRID=66750.,
 WHEAD=175927.08,
 DBH=20.915,
 THICK=0.52198,
 COND=8.0005,
 E1=.8,
 E2=.5,
$END
$NLHOT
 IHOT=100,
 DP=0.25,
 FLRMC=3360.,
$END
$NLINTR
 CAYC=0.01524,
 CPC=1.30,
 DENSC=2.375,
 TIC=308.16,
 FC1=0.441,
 FC2=0.108,
 FC3=0.357,
 FC4=0.027,
 RBR=0.135,
 R0=322.6,
 R=6000.0,
 HIM=0.2,
 HIO=0.09,
 WALL=1000.,
$END
#
```

B.8 S_2J Sequence

```
 BROWNS FERRY SEQUENCE S2J (D=0.5 IN)
$NLMAR
 ITRAN=1,
 IBRK=1,
 ISPRA=1,
 IECC=2,
 IBURN=0,
```

```
  IPOTL=7,
  IPLOT=3,
  IU=3,
  VOLC=2.78E05,
  TAP=2.62E06,
 $END
 $NLINTL
 $END
STEEL     CONCRETE
DRYWELL1  DRYWELL2  CONC SHELLMISC STEELMISC CONC.
C
 $NLSLAB
  NMAT=2,
  NSLAB=3,
  NOD=1,4,13,
  DEN(1)=486.924,157.481,
  HC(1)=.1137,.23817,
  TC(1)=25.001,.80024,
  IVL=1,1,2,
  IVR=1,1,2,
  NNO1=3,9,4,
  MAT1=1,2,1,
  MAT2=1,2,1,
  SAREA=18684.,5358.,15982.,
  X(1)=0.,.01,.02083, X(4)=0.,.01,.03,.07,.15,.31,.63,1.27,2.5,
  X(13)=0.,.01,.03,.0625,
  TEMP=12*150.,4*95.,
 $END
 $NLECC
  PUHIO=0.001,
  UHIO=10.0,
  PACMO=0.001,
  ACMO=10.0,
  PHH=1160.0,
  WHH1=-5000.0,
  TMSIS=10.0,
  PSIS=1160.0,
  WSIS1=-600.0,
  PLH=1160.0,
  WLH1=-50.0,
  NP=2,
  TM(1)=450.0,
  TM(2)=0.0,
  P(1)=450.0,
  P(2)=1150.0,
  WEC(1)=12500.0,
  WEC(2)=31594.784,
  STP(2)=0.0,
  RWSTM=3.05E06,
  CSPRC=2.0,
  DTSUB=-100.0,
  TRWST=95.0,
 $END
 $NLECX
 $END
 $NLCSX
 $END
 $NLCOOL
  JCOOL=1,
  CQR=1.6[illegible]06
  CWPR=6[illegible]00.0,
  CTPR=15[illegible].0,
  CWSR=1.49E05,
  CTSR=95.0,
```

```
 TCOOL=0.0,
 NCOOL=2,
 PCOOL=1.80,
 POFF=0.20,
$END
$NLMACE
 NCUB=2,
 NRPV1=1,
 NRPV2=1,
 NRPV3=2,
 ICECUB=-1,
 DTPNT=500.0,
 IDRY=-1,
 IWET=2,
 WPOOL=7.801E06,
 TPOOL=95.0,
 VDRY=3.839E03,
 VTORUS=257700.0,
 WVMAX=5.146E05,
 NSMP=-2,
 NSMP2=2,
 WVMAKS=5.146E05,
 NCAV=1,
 VCAV=4789.1,
 VFLR=15.0,
 IVENT=-12,
 TVNT1=-0.2,
 TVNT2=30000.0,
 AVBRK=292.0,
 CVBRK=4.04,
 VC(1)=159000.0,257700.0,
 AREA(1)=1.6399E03,1.098E04,
 HUM(1)=0.2,1.0,
 TEMPO(1)=150.,95.,
 N=7,
 NS(1)=1,1,1,3,3,2,2,
 NC(1)=1,1,1,1,1,1,2,
 NT(1)=1,2,3,-7,-7,-7,-7,
 C1(1)=1.0E6,1.0E6,0.0,400.0,500.0,174.7,174.7,
 C2(1)=2.0E04,4.67E06,7.59106,5.9297,0.583,0.583,0.583,
 C3(1)=95.0,0.0,1192.5,.00694,20.97,20.97,20.97,
 C4(1)=400.0,
 KT(1,2)=1,
 KT(2,1)=1,
$END
$NLBOIL
 NNT=37436,
 NR=35908,
 ISTR=3,
 ISG=0,
 IMWA=3,
 IHR=1,
 ISAT=1,
 AB(1)=0.9498,
 TB(1)=0.0,
 WDED=3.50E04,
 TPUMP1=75.0,
 TPUMP2=101.0,
 QPUMP1=5.50E07,
 QPUMP2=2.75E07,
 QZERO=1.1242E10,
 H=12.0,
 HO=28.,
 DC=15.59,
```

```
ACOR=104.833,
ATOT=287.898,
WATBH=97000.,
D=.04692,
DF=.04058,
DH=0.056,
CLAD=.005594,
XOO=8.33E-06,
RHOCU=68.783,
TGOO=546.0,
PSET=1120.0,
CSRV=2002.57,
FDCR=-.5,
DPART=0.0208333,
DUO2=0.04058,
FZMCR=0.05,
FZOCR=0.08,
FZOS1=0.1,
WFE2=7992.,
TFEOO=546.,
FULSG=0.0,
PVSL=1020.0,
TCAV=588.5,
ABRK=0.001364,
YBRK=8.33,
DTPNTB=500.0,
DTPN=-50.0,
VOLP=2.459E04,
VOLS=9.638E03,
WCST=3.05E06,
F(1)=0.1,0.25,0.47,0.65,0.84,0.96,1.13,1.27,1.295,1.27,1.24,1.21,1.195,
F(14)=1.15,1.11,1.08,1.05,1.03,1.02,1.016,1.017,1.05,1.06,1.061,1.062,
F(26)=1.061,1.06,1.059,1.059,1.06,1.07,1.075,1.095,1.11,1.12,1.185,1.215,
F(38)=1.25,1.26,1.24,1.21,1.15,1.09,1.0,0.87,0.76,0.6,0.41,0.21,0.1,
PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,0.9665,0.408,
VF(1)=10*0.1,
TT=6*546.,
CM=1824.,7992.,8760.,2460.,5550.,24900.,
AH=740.,263.,9225.,400.,7000.,700.,
DD=1.,1.,1.,.17,.02,.546,
AR=150.,263.,165.,0.,-10.,-20.,
$END
$NLHEAD
 WZRC=140397.0,
 WFEC=30447.73,
 WUO2=361837.0,
 WGRID=66750.,
 WHEAD=175927.08,
 DBH=20.915,
 THICK=0.52198,
 COND=8.0005,
 E1=.8,
 E2=.5,
$END
$NLHOT
 IHOT=100,
 DP=0.25,
 FLRMC=3360.,
$END
$NLINTR
 CAYC=0.01524,
 CPC=1.30,
 DENSC=2.375,
 TIC=308.16,
```

```
 FC1=0.441,
 FC2=0.108,
 FC3=0.357,
 FC4=0.027,
 RHR=0.135,
 R0=322.6,
 R=6000.0,
 HIM=0.2,
 HIO=0.09,
 WALL=1000.,
$END
#
```

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