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SBLOCA Outside Containment at Browns Ferry Unit One—Accident Sequence Analysis

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Prepared for the U.S. Nuclear Regulatory Commission
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SUMMARY

This is the first of a two-volume study describing the predicted response of Unit 1 at the Browns Ferry Nuclear Plant (BFNP) to a hypothetical break in the scram discharge volume (SDV) piping following a reactor scram that cannot be reset. Under scrammed conditions with the scram outlet valves open a flow path exists between the reactor vessel and the SDV, which is located in the reactor building at the 172-m (565-ft) level outside of the primary containment (drywell). Until the scram is reset, which shuts the scram outlet valves, reactor coolant at reactor vessel pressure is contained within the boundaries of the SDV. Thus, a break in this piping system would release reactor coolant directly into the reactor building atmosphere until the scram is reset.

The boiling water reactor—loss of ac power (BWR-LACP) code, described in Appendix A, has been used for the analysis of the sequence of events before core uncovery. It has not been necessary to define a specific location or size for the SDV piping break for this analysis. Rather, the break was assumed to be large enough such that the leakage into the reactor building is constrained only by the flow restriction afforded by the graphitar seals within the control rod drive (CRD) mechanism assemblies. Under normal conditions, the average seal leakage is about 1.89 x 10^{-4} m^3/s (3 gpm) per mechanism assembly, so that the total leakage from the 185 CRD mechanisms would be about 0.035 m^3/s (550 gpm) initially. The graphitar seals are subject to degradation at high temperature; it has been assumed that after 90 min of exposure to the hot coolant leaking past the seals from the reactor vessel, the seals would begin to erode, with the effective leakage area assumed to increase linearly over a 6-1/2-h period to a final value corresponding to 6.31 x 10^{-4} m^3/s (10 gpm) per assembly at normal operating temperature and pressure. The latter value corresponds to CRD mechanism leakage that has been measured in tests conducted with the seals completely removed.

Thus, if the fluid in the reactor vessel were to remain at normal operating temperature and pressure, the leakage model employed for this study would predict a constant total leakage of 0.035 m^3/s (550 gpm) for the first 90 min following the SDV break; after 90 min, the calculated leakage would increase linearly over the subsequent 6-1/2 h to 0.114 m^3/s (1800 gpm), reaching this value at the 8-h point and remaining constant at this level. However, the leakage model actually employs a time-dependent effective seal leakage area so that the leakage flow is a function of the reactor vessel temperature and pressure as well.

The total leakage into the reactor building is the sum of the flow of hot water from the reactor vessel and the ~0.011 m^3/s (170 gpm) of room-temperature flow into the CRD mechanism assemblies from the CRD hydraulic system via the open scram inlet valves. These two flows are assumed to mix uniformly within the CRD mechanism assemblies before passing out to the SDV piping and from there through the break into the reactor building atmosphere.

Once the control room operator is aware that there is leakage into the reactor building and has correctly diagnosed its cause there are several corrective actions that he might take. The most effective action would be to reset the scram so that the scram outlet valves would close,

isolating the SDV piping and the break from the reactor vessel. However, it has been assumed that the reactor scram was initiated by a reactor protection system (RPS) signal that remains in effect throughout the accident.* Therefore, to reset the scram, the operator would have to override the scram signal by the placement of rubber insulation between the applicable relay contacts in the auxiliary instrument room, which is certainly not a standard procedure.

If the scram is not reset, then the most effective operator action would be to depressurize the reactor vessel as quickly as possible while maintaining vessel level with the high-pressure coolant injection (HPCI) system. Depressurization reduces the driving force for leakage from the reactor vessel and decreases the potential for seal erosion by reducing the temperature of the leaked coolant. Full depressurization would greatly reduce the flow from the reactor vessel so that the leakage from the SDV piping would consist primarily of the pumped flow from the CRD hydraulic system. After depressurization, the CRD hydraulic system flow could be throttled to reduce the total leakage while keeping the leakage mixture subcooled.

Once the operator has recognized the existence of a small break in the SDV system, the Emergency Operating Instruction (EOI) applicable to small-break loss-of-coolant accidents (SBLOCAs) outside of primary containment should become controlling. This EOI does not call for depressurization unless the vessel level cannot be maintained by the high-pressure injection systems, which, in the case of the SDV piping break accident, would consist of the HPCI and the reactor core isolation cooling (RCIC) systems.† Therefore, depressurization would not be required by the EOI for small breaks outside of the primary containment unless the leakage exceeds 0.355 m^3/s (5600 gpm), the combined capacity of the HPCI and RCIC systems.

The loss of primary coolant outside of primary containment to any significant degree is a serious matter that can threaten the viability of safety equipment located in the reactor building. Accordingly, it is recommended that reactor vessel depressurization be required for any non-isolatable break outside of the primary containment.

The question of how long it would take for the operator to comprehend the scope of this accident is central to the analysis of the sequence of events. An abnormal occurrence, that is, a scram from full power, would immediately precede the SDV piping break that constitutes the initiating event for the SBLOCA-outside-containment accident sequence considered here. It would not be difficult for the operator to assume that many of the symptoms of the SDV break were in fact produced by the event causing the scram. For example, if the scram were caused by high main steam line radiation, he might incorrectly assume that the reactor building high

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*The scram cannot be reset as long as the scram signal stays in effect. Note that several scram signals do not have provision for bypass from the control room (see Table E.1).

†The feedpumps are not available because the main steam isolation valves (MSIVs) are shut. The CRD hydraulic pump discharge is being lost out the break.

radiation alarms associated with the SDV piping break were also caused by the high radiation within the main steam lines, which he would know had caused the scram.

Nevertheless, the probability that the operator would understand that he is confronted with an SBLOCA-outside-containment in addition to the event that initiated the scram would be a strongly increasing function of time. Roving patrols would report the presence of steam in the reactor building, the radwaste building control room operator would report a marked increase in flow from the reactor building floor drain sump, and there would be both high-temperature and high-radiation alarms in the control room for the monitored locations in the reactor building.

The operator would probably recognize the existence of an SBLOCA-outside-containment before he ascertained its cause. However, the indications of an unusual after-scram condition in the scram discharge system would be prominent, including a persistent CRD high-temperature alarm with high-temperature readings for all of the mechanisms and an abnormal CRD position indication for all of the control rods, caused by the existence of the leakage path from the SDV. Once the operator did understand that the source of the leakage was a break in the SDV piping, he could take action to isolate the break by initiative to override the scram signal and reset the scram as discussed previously.

It is important to note that much of the safety-related equipment important to accident diagnosis and mitigation, such as pump and valve motor control centers located in the vicinity of the SDV piping, has not yet been qualified for exposure to harsh environments. Even if the scram is reset, it is possible that the scram pilot valves would not close because of moisture accumulation within the RPS fuse cabinets that are located near the hydraulic control units. If the circuits within these cabinets are short-circuited, the scram valves would not close because power would not be available to the scram solenoids as a result of the blown fuses associated with the individual hydraulic control units; these cabinets are not environmentally qualified for exposure to the conditions that would be produced by the SDV piping break.*

Without operator action, the situation would develop into a severe accident with core damage and subsequent release of fission products to the atmosphere. Analysis of the sequence of events for this postulated accident without operator action also reveals some important considerations that have application to other accident situations as well.

Without operator action, the HPCI system would automatically cycle to maintain reactor vessel water level between the limits of 12.1 m (476.5 in.) and 14.8 m (582 in.) above the bottom of the reactor vessel.† However, each time the HPCI system is actuated to restore vessel level, the reactor vessel pressure significantly decreases because of the steam flow to the HPCI turbine and the quenching effect produced by the introduction of cold water into the vessel. With the decreasing decay heat and the

*From comments to the draft of this report prepared by J. Rubin of the NRC Office for Analysis and Evaluation of Operational Data (AEOD).

†Reactor vessel level during normal operation is 14.25 m (561 in.). The HPCI system injects water into the reactor vessel from the condensate storage tank (CST).

increasing SDV break size caused by the seal erosion assumed in this study, the average reactor vessel pressure during the level-restoring cycles of the HPCI system would continually decrease.

Although the turbine-driven feed pumps would cease operation upon the high reactor vessel water level caused by continued feed pump operation just after the scram,* the condensate pumps (CPs) and the condensate booster pumps (CBPs) are electric-motor driven and would remain operating with the potential to inject water into the reactor vessel through the idle centrifugal feed pumps. The shutoff head of the CBPs is about 2.86 MPa (415 psia), and there would be no flow into the reactor vessel until the vessel pressure fell below this point. In the interim, the CPs and CBPs would be protected by a recirculation line that permits the limited pump flow to pass back into the main condenser hotwell. The concomitant heatup of the recirculated water has been modeled in this analysis.

When the reactor vessel pressure does drop below the shutoff head of the three CBPs, which have a combined injection capability of 1.892 m^3/s (30,000 gpm), the reactor vessel rapidly fills with cold water, which spills over into the main steam lines. Without operator action, the filling of the main steam lines and, consequently, the steam supply lines to the RCIC and HPCI turbine precludes any further use of these high-pressure injection systems.

With the MSIVs shut, main condenser vacuum could not be maintained, and makeup water to the main condenser hotwells from the CST would be limited to that induced by gravity feed. Because the pumped flow into the reactor vessel would greatly exceed the hotwell makeup rate under these conditions, the hotwell would eventually be pumped dry and the CPs and the CBPs would trip on low suction pressure. After this occurrence, no coolant makeup would be available to the reactor vessel† and a boiloff would occur, exposing the top of the core about 7.4 h after the inception of the accident.

This study shows a clear need for consideration of the vessel overflooding that might occur following injection from the CBPs or, in other accident sequences, the residual heat removal (RHR) and the core spray (CS) pumps. All of these pumping systems have enormous pumping capabilities that can rapidly fill the reactor vessel under accident conditions with the MSIVs shut. However, none of these systems has a high-level shutoff to protect against flooding of the HPCI and RCIC steam supply lines, so that their actuation, unless the operator is very quick, will preclude further operation of the high-pressure injection systems if the reactor vessel repressurizes later in the accident sequence.

The MARCH code has been used for the analysis of the events following core uncovery. The existing versions of MARCH employ thermal-hydraulic

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*The reactor vessel water level increases rapidly after the scram because of continued operation of the three main feed pumps. The feed pumps are automatically tripped on high vessel level and the MSIVs subsequently close on low vessel level.

†The low-pressure Emergency Core Cooling (ECCS) systems are ineffective under these conditions because when vessel level has dropped to initiation setpoints, the pressure has increased to a value above the shutoff head of these systems.

models that are too crude to permit determination of the plant response to postulated operator actions or automatic HPCI system operation before core uncovery. Accordingly, the MARCH code analysis was initiated just before core uncovery with initial conditions provided by the results of the BWR-LACP code.

The MARCH results predict that the water level would be beneath the fuel bundle flow inlets (about 1 ft below the bottom of the core) at about 7.6 h after the inception of the accident, with core melting beginning about 50 min later. The reactor vessel pressure steadily decreases after the fuel bundle flow inlets are uncovered, because the leakage from the vessel is through these openings and the leakage medium shifts from water to steam when they are uncovered. When the differential pressure between the vessel and the wetwell has decreased to a value corresponding to the effective shutoff head (pump shutoff head minus the elevation head between the suppression pool surface and the injection point into the primary system) of the low-pressure emergency-core-cooling systems (LPECCS), flow into the reactor vessel begins.

The effects of two LPECCS configurations have been analyzed in this study. The LPECCS system design specifications provide that the low-pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) system should have a slightly (5 psi) higher effective shutoff head than the Core Spray (CS) system, enabling injection to begin when the reactor vessel pressure has decreased to within 295 psi of the pressure in the wetwell, from which the LPECCS pumps take suction. The meltdown model provides that the melted core slumps down into the bottom of the reactor vessel, and this results in a predicted failure of the reactor vessel bottom head at ~10.9 h after the inception of the accident.

After failure of the reactor vessel bottom head, the core falls onto the concrete floor of the drywell and a corium-concrete reaction begins. There is question at this point as to whether the leakage path to the reactor building atmosphere via the SDV piping break would remain open. Because the failure of the bottom head would cause the CRD mechanism assemblies to fall into the drywell with the molten core, the CRD assemblies most likely would melt, breaking off the connections to the 3/4-in. lines that lead from each assembly to the SDV piping. Assuming that this is the case, there would be 185 open 3/4-in. lines leading from the drywell through the scram outlet valves to the SDV piping. This is equivalent to a break area in the drywell of 0.053 m^2 (0.568 ft^2).

With the assumption that the leakage path to the SDV piping does remain open with an increased leakage area of 0.053 m^3 (0.568 ft^2), there is continuous leakage from the drywell atmosphere into the reactor building after reactor vessel bottom head failure. The drywell coolers are assumed to fail when the drywell atmosphere temperature reaches 149°C (300°F), and there is an assumed subsequent additional breaching of the primary containment at about 11.8 h after the inception of the accident because of failure of the drywell electrical penetration assembly modules by overtemperature. The detailed thermohydraulic parameters needed for the analysis of fission product release from the fuel rods and the subsequent transport to the environment were taken from the MARCH results for this sequence, in which the LPECCS is assumed to function as designed and the injection is primarily core bottom flooding via the LPCI mode of the RHR system.

The second LPECCS configuration analyzed was determined from the head curves for the installed RHR and CS pumps; these curves were made available to this study by the Tennessee Valley Authority (TVA). These test results show that the as-built systems have a significantly higher effective shutoff head than that specified by the system design. As-built, the core spray pumps have an effective shutoff head about 11 psi higher than that of the RHR pumps, enabling core spray to begin when the reactor vessel pressure has decreased to within 342 psi of the pressure in the wetwell.

With the as-built configuration, the core slump and reactor vessel failure are delayed about 0.5 h over the as-designed case, and vessel injection is primarily via the core spray system. MARCH predicts reactor vessel lower head failure at 11.5 h after the inception of the accident with failure of the primary containment drywell via overtemperature-induced degradation of the drywell electrical penetration assembly seals about 40 min later. However, it must be noted that the MARCH analysis for this as-built sequence cannot be accurate because MARCH does not model the spray effects of BWR core spray systems, which have the predominant injection role in the as-built LPECCS configuration. The differences in evert timing between the two sequences are due only to the higher injection initiation pressures user-input for the as-built case. The MARCH code models both forms of injection as bottom flooding.

A third MARCH analysis was performed for the assumption that there is no LPECCS injection to the vessel at all. With no injection, the failure of the reactor vessel bottom head is advanced only about 40 min over the base as-designed case with LPECCS. It must be recognized that the obviously small effect of the LPECCS systems in the no-operator-action accident sequences studied here is because the reactor vessel is never fully depressurized. In the sequences with LPECCS injection, the vessel pressure drops just low enough to allow a small amount of injection flow; the resulting steam generation repressurizes the vessel limiting or eliminating further LPECCS flow.

A fourth MARCH analysis was performed for the case without ECCS injection; it is assumed that the leakage path to the SDV piping is shut off when the reactor vessel lower head fails, that is, that the falling molten core acts to effectively weld the 3/4-in. lines from each CRD mechanism shut. In this case, there is no leakage from the drywell into the reactor building until the drywell electrical penetration assembly seals fail by overtemperature, predicted to occur at about 11.3 h after the inception of the accident. The drywell coolers are again assumed to fail when the temperature of the drywell increases to 149°C (300°F). These MARCH code results were most unsatisfactory, indicating that the drywell boundary would fail by overtemperature earlier if there were no leakage pathway to the surrounding atmosphere than it would if some venting of the hot gases generated by the core-concrete reaction occurred. As discussed in Sect. 6, it is believed that this fallacy is caused by the peculiar method of time-step control utilized in MARCH; the comparison of accident sequences with shut vs open CRD withdrawal lines after vessel bottom head failure has not been further pursued.

An analysis of the efficacy of direct drywell flooding as an accident mitigation technique was performed as a corollary to this study. It was

found that the presence of water on the outer surface of the reactor vessel would be effective in preventing a molten and collapsed core from melting through the reactor vessel bottom head. However, if action to begin direct drywell flooding were delayed until the time of core melting, practical considerations would prevent the drywell flooding from being completed with the existing equipment in time to accomplish its purpose.

Additional analyses were performed to determine the effect of the assumed CRD mechanism seal erosion on the course of the accident. Without the assumed seal erosion, the failure of reactor vessel injection capability would not occur until about 16 h after the inception of the accident. In this case, the failure would be caused by flooding of the reactor building basement to a level sufficient to partially submerge the ECCS pumps and motors located there.

An estimate of the magnitude and timing of the release of the noble gas, cesium, and iodine-based fission products to the environment is provided in Volume 2 of this study for the no-operator-action accident sequence with assumed seal erosion.

SBLOCA OUTSIDE CONTAINMENT AT BROWNS FERRY UNIT ONE — ACCIDENT SEQUENCE ANALYSIS

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ABSTRACT

This study describes the predicted response of Unit 1 at the Browns Ferry Nuclear Plant to a postulated small-break loss-of-coolant accident outside of the primary containment. The break has been assumed to occur in the scram discharge volume piping immediately following a reactor scram that cannot be reset. The events before core uncovery are discussed for both the worst-case accident sequence without operator action and for the more likely sequences with operator action. Without operator action, the events after core uncovery would include core meltdown and subsequent containment failure, and this event sequence has been determined through use of the MARCH code. An estimate of the magnitude and timing of the concomitant release of the noble gas, cesium, and iodine-based fission products to the environment is provided in Volume 2 of this report.

1. INTRODUCTION

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

Each unit comprises a boiling water reactor (BWR) steam supply system designed by the General Electric Company for a power output of 3440 MW(t) [1152 MW(e)]; the maximum power authorized by the operating license is 3293 MW(t), or 1067 net MW(e). The primary containments are of the Mark I pressure suppression pool type, and the three units share a secondary containment of the controlled leakage, elevated release design; each unit occupies a separate reactor building located underneath the common refueling floor.

This report presents the results of an analysis of a postulated small-break loss-of-coolant accident (SBLOCA) outside of the primary containment at Unit 1 of the BFNP. The potential for the particular SBLOCA analyzed here, which involves a break in the scram discharge volume (SDV) piping, was first identified by the Nuclear Regulatory Commission (NRC) Office for Analysis and Evaluation of Operational Data (AEOD) in conjunction with

their analysis of the June 28, 1980, partial failure to scram at Browns Ferry Unit 3 (Ref. 1). Section 2 provides a description of the SBLOCA initiating event and a discussion of the motivation for consideration of the follow-on accident sequence.

An important consideration in this accident sequence is the possibility that the operator will not recognize the nature of the break for a significant period of time. Accordingly, a description of the accident sequence without operator action is provided in Sect. 3. The instrumentation available in the control room for use in operator diagnosis of the accident signatures and for subsequent recovery from the accident is described in Sect. 4. Section 5 then contains a discussion of the accident sequence of events as mitigated by operator action.

Unless stated otherwise at the appropriate locations in this report, it has been assumed that the equipment necessary to the accident mitigation efforts would continue to function in the accident environment created by the piping break. This is an important assumption because much of the safety-related equipment essential to accident diagnosis and mitigation has not yet been environmentally qualified. Accordingly, substantial uncertainty surrounds the assumption of the operability and capability of a wide variety of systems. An attempt has been made to address this uncertainty in Sects. 4 and 5.

The study of this SBLOCA accident sequence includes consideration of possible severe-accident phases. A severe accident, by definition, is an accident that in the absence of effective corrective action proceeds through core uncovery, core meltdown, and the release of fission products from the fuel. The severe-accident events that would follow the no-operator-action sequence described in Sect. 3 were analyzed by application of the MARCH code and are described in Sect. 6.

The conclusions of this SBLOCA-outside-containment accident sequence analysis and the implications of the results are discussed in Sect. 7. This includes an evaluation of the available instrumentation, the level of operator training, the existing emergency procedures, and the overall system design from the standpoint of the requirements for mitigation of this accident.

Appendix A contains a listing of the computer program BWR-LACP developed at Oak Ridge National Laboratory (ORNL) to model operator actions and the associated primary system and reactor building response during the period prior to core uncovery. The MARCH code* was used for calculation of the severe-accident events following core uncovery but required significant modification for application to BWR SBLOCA-outside-containment analysis. A discussion of the desired MARCH code modifications for application to BWR analyses is provided in Appendix B, and the modifications actually made for this study are described in Appendix C. An input listing for the MARCH base case no-operator-action sequence computation is provided in Appendix F.

During discussions in preparation for this study, Browns Ferry operating personnel indicated that they believed direct drywell flooding might be an effective measure of last resort to reduce the consequences of a

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*MARCH version 1.1 from Battelle Columbus Laboratories.

severe accident by preventing an already molten core from melting through the reactor vessel bottom head. Because there are no existing emergency procedures for or studies of the efficacy of such action, an analysis of the efficacy of direct drywell flooding as an accident mitigation technique was performed as an adjunct to this work and is provided as Appendix D.

The operation and design of the scram discharge volume and associated piping is discussed in Appendix E. A knowledge of the characteristics of this system is essential to an understanding of the SBLOCA-outside-containment that is analyzed in this study, and readers unfamiliar with the BWR scram system design should review Appendix E before proceeding to the remainder of this report.

An important finding attendant to the no-operator-action sequence analysis (discussed in Sect. 3) is that the computed series of reactor vessel pressure fluctuations leads to vessel filling by the condensate booster pumps (CBPs) at about 4.6 h after the initiation of the accident. Because this event has a significant impact on the accident sequence, it was deemed desirable that these results of the BWR-LACP code be confirmed by a more sophisticated code. The Severe Accident Sequence Analysis (SASA) team at Idaho National Engineering Laboratory (INEL) performed the confirmation calculations using RELAP V Mod 1; these calculations are discussed in Appendix G.

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

The magnitude and timing of the release of the fission product noble gases and the various forms of iodine to the atmosphere during the severe-accident events following core uncovery are discussed in the second volume of this report. This discussion includes consideration of the escape of fission products from fuel as a function of temperature, specification of the various chemical forms, the modeling of precursor/daughter exchange, and determination of the fission product release pathways. These results are incorporated into a vehicle for the calculation of the transport of the individual fission products from the reactor core, through the reactor vessel, through the primary and secondary containment, control volume by control volume to the atmosphere.

The General Electric Company evaluation of BWR scram system pipe breaks[2] and the subsequent USNRC generic evaluation[3] of this issue were reviewed and utilized in the preparation of this report. The primary sources of plant-specific information were the BFNP Final Safety Analysis Report (FSAR), the USNRC BWR Systems Manual, the BFNP Hot License Training Program Operator Training Manuals, the BFNP Unit 1 Technical Specifications, the BFNP Emergency Operating Instructions, and various other specific drawings, documents, and manuals obtained from the TVA. Additional information was gathered during one visit to the BFNP for inspection of the reactor building and drywell interiors and during several visits to the TVA Power Operations Training Center at Soddy-Daisy, Tennessee, for examination of the control room layout and determination of the instrument responses to a simulation of events similar to those that would occur during a SBLOCA/OC.

The setpoints for automatic equipment response used in this study are the current established safety limits. In many cases, these differ slightly from the actual setpoints used for instrument adjustment at the

BFNP, because the instrument adjustment setpoints are established to provide margin for known instrument error.

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

References

1. U.S. Nuclear Regulatory Commission, *Safety Concerns Associated with Pipe Breaks in the BWR Scram System*, Draft Report NUREG-0785 (April 1981).

2. General Electric Company, *GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks*, NEDO-24342 (April 1981).

3. U.S. Nuclear Regulatory Commission, *Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping*, NUREG-0803 (August 1981).

2. DESCRIPTION OF INITIATING EVENT

The initiating event for the SBLOCA sequence analyzed in this study is a break in the SDV piping outside of the primary containment at Unit 1 of the BFNP. The break is assumed to occur following a scram that cannot be reset so that the break is in effect nonisolable.

The SDV and associated piping are described in Appendix E.* Briefly, the SDV is a receiver volume for the water displaced from the 185 control rod drive (CRD) mechanisms through the scram outlet valves during a scram. During normal reactor operation this volume is vented, drained, and empty, but when a scram occurs, the vent and drain valves shut and the SDV is partially filled with water displaced from above the drive pistons during the control rod scram strokes. Shortly thereafter, the remainder of the SDV fills because of leakage past the CRD seals and, consequently, pressurizes to full reactor vessel pressure. When the scram is reset, the scram outlet valves shut to isolate the SDV from the reactor vessel, the vent and drain valves open, and the SDV empties.

Reset is possible only if the conditions that caused the scram have cleared and is accomplished by operating switches in the control room. If the scram is not reset, the SDV and the associated piping downstream of the scram outlet valves remain exposed to full reactor pressure. Although bypass is provided for some scrams, most scrams cannot be reset until the conditions causing the scram have cleared. If a break occurs in the SDV or associated piping downstream of the scram outlet valves, then the control room operator cannot isolate the break from the reactor vessel unless the scram can be reset, thereby shutting the scram outlet valves.†

As discussed previously, the SDV would be pressurized as a normal consequence of the scram, and it is assumed that a large break in this piping system occurs subsequent to the pressure increase. It is not necessary to the purpose of this study to specify an exact break location or cause, but it is assumed that the break is of sufficient magnitude that the leakage rate from the reactor vessel is controlled by the seals within the CRD.‡ The leakage is into the reactor building atmosphere above the hydraulic control units located as shown in in Appendix E, Fig. E.5.

These assumptions were chosen to permit a worst-case analysis of the effects of a SDV break, which by itself is a very unlikely event.[1,2] The leakage is assumed to be limited only by the CRD mechanism seals to permit analysis of the SDV break accident sequence with the maximum possible leakage rates.

Considering the low probabilities associated with these assumptions, which must be multiplied by the very low probability of a SDV break in the

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*See the discussion in Sect. E.4 and Table E.1.

†As discussed in Sect. 5, the break could be isolated by shutting 185 manual valves in the reactor building if environmental conditions permitted, or by overriding the interlocks in the auxiliary instrument room which prevent the scram reset.

‡The function and location of the CRD mechanism seals is discussed in Appendix E, Sect. E.1.

first place, an SBLOCA outside containment in the manner described here is an extremely unlikely event. Nevertheless, great advantage can be gained by studying its consequences, because many of the lessons learned will be shown to have application to other, more likely accident sequences. Experience has shown that weaknesses in the current plant defenses can be identified by postulating an accident scenario and then carefully studying its consequences, given the existing equipment configuration and operator training.

References

1. U.S. Nuclear Regulatory Commission, *Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping*, NUREG-0803 (August 1981).

2. TVA response to D. G. Eisenhuts' letter to all BWR Licensees April 10, 1981 (A27 810415 014); Subject: Safety Concerns Associated with Pipe Breaks in the BWR Scram System.

3. ACCIDENT SEQUENCE WITHOUT OPERATOR ACTION

3.1 Introduction

This section describes plant response to an assumed SDV break without operator action during the approximately 8-h period before core uncovery. Mitigating actions that the operators might take during this period to prevent the SDV break occurrence from progressing into a severe accident are discussed in Sect. 5.

The results described in the following paragraphs were obtained using the BWR-LACP code (Ref. 1). Modifications made to BWR-LACP especially for the SDV break sequence analysis are described in Appendix A. An analysis of primary coolant system thermal-hydraulic response during the SDV break no-operator-action sequence was also performed by Idaho National Engineering Laboratory using the RELAP-5 code. The purpose of this supplementary analysis was to perform a more detailed (i.e., compared with BWR-LACP) calculation of primary coolant system thermal-hydraulics to verify the overall sequence of events predicted by the BWR-LACP code. The RELAP-5 results (presented in Appendix G) are in substantial agreement with the BWR-LACP results presented in this section (Figs. 3.1–3.7).

As discussed in Sect. 2, the break is assumed to be large enough so that the leakage flow (of water from the lower plenum of the reactor vessel) is controlled by the restriction of the graphitar seals within the control rod drive (CRD) mechanisms. Reference 2 reports test data showing: (1) that initial seal-controlled leakage would be 0.03 m^3/s (550 gpm) at reactor vessel normal operating temperature and pressure,* (2) that even if the CRD mechanism seals were completely removed, the leakage would not exceed 0.11 m^3/s (1800 gpm), and (3) that seal degradation was not observed during 90 min of hot leakage under test conditions.† For the BWR-LACP calculations reported in this section, the initial leakage flow is modeled as critical flow through a 6.51×10^{-4} m^2 (0.007-ft^2) equivalent break area. It is assumed that seal erosion occurs after 90 min; this is simulated by gradually increasing the equivalent break area to 3.29 times the initial area. The increase in equivalent break area begins 90 min after the SDV break and is complete 6.5 h later.‡

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*With 185 CRD mechanisms, this is closely equivalent to an average seal leakage of 3 gpm per mechanism.

†The seals and bushings on the CRD drive piston and stop piston are constructed of graphitar 14, which experiences loss of strength at high temperatures. The extent of seal degradation after 90 min of exposure to hot leakage has not been determined.

‡If the fluid in the reactor vessel were to remain at normal operating temperature and pressure, the model would hold the total leakage to 550 gpm for 90 min, then ramp this leakage up to 1800 gpm over the following 6.5 h. In actuality, the reactor vessel temperature and pressure are time dependent.

3.2 Summary and Conclusions

Without the benefit of operator intervention, the SDV break, which is assumed to occur 30 s after the initiating reactor trip, causes core uncovery after about 8 h. Table 3.1 summarizes major events during this 8-h period. Immediately after the scram, the reactor vessel water level increases rapidly because of continued operation of the three main feed pumps. The feed pumps are automatically tripped when the reactor vessel water level reaches 14.78 m (582 in.). Subsequently, the main steam isolation valves (MSIVs) automatically close when the reactor vessel water level has decreased to 12.10 m (476.5 in.). Detailed results are presented in Sects. 3.3 and 3.4. The reason for the core uncovery is loss of automatically controlled (or self-initiated) reactor vessel injection capability. Loss of injection capability is not caused by the environmental effects [i.e., on emergency core cooling system (ECCS) pumps] of the leak, but by uncontrolled vessel injection by the main feedwater system. This effect, not foreseen at the beginning of the study, was first noted by ORNL investigators at the TVA Browns Ferry simulator and later confirmed by the BWR-LACP results.

After about 4 h, the SDV leakage depressurizes the reactor vessel to the level at which the continuously running condensate pumps (CPs) and CBPs* can inject water into the vessel. Uncontrolled injection by these pumps fills the reactor vessel, depletes the store of water in the main condenser hotwell, and floods the steam supply lines leading to the reactor core isolation (RCIC) and high-pressure coolant injection (HPCI) systems. With the HPCI system inoperable and the hotwell empty, the residual heat removal (RHR) and core spray (CS) systems would initiate on low vessel level, but, without operator action, these pumps cannot pump water into the reactor vessel. This is because the automatic initiation of RHR and CS on low vessel level would not occur until after the pressure within the reactor vessel had been restored to a level above the shut-off head of the pumps. The automatic depressurization system (ADS) does not actuate in this case because (with the break outside primary containment) there is no coincident high drywell pressure signal.

Calculations similar to those summarized were made with the assumption of no CRD seal erosion (constant equivalent break area). Results indicate that the failure of reactor vessel injection capability would not occur until ~16 h after the SDV break initiation and that the cause of the failure in this case would be flooding of the basement to a level sufficient to fail the HPCI, RCIC, RHR, and CS pumps located therein by submergence. Detailed results for this case have not been included in this report.

For the baseline SDV break no-operator-action sequence discussed above, the assumption is made that offsite power is available continuously. This is reasonable since a reactor trip and subsequent SDV break would not be expected to cause a loss-of-offsite power (LOSP). However, even a brief LOSP would significantly change the no-operator-action sequence. The power supply breakers of the CPs and CBPs are automatically

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*These pumps are driven by electric motors and, after a scram and MSIV closure, can pump through the idle turbine-driven feedwater pumps into the reactor vessel. The CBPs have a shutoff head of about 415 psia.

tripped upon LOSP. Even if power is promptly restored, these pumps will not restart without operator action to reset the power supply breakers. Therefore, after LOSP the CPs and CBPs would not flood the vessel and HPCI steam lines and, thus, would not cause early failure of HPCI injection and core uncovery before 8 h. Instead, HPCI injection would maintain vessel level until HPCI failure due to submergence, which would take longer than 8 h. As shown on Fig. 3.13 (from Sect. 3.4.3), the basement flooding depth is only 0.62 m (2.04 ft) at the time of core uncovery for the baseline no-operator-action sequence.

3.3 Results - Reactor Vessel and Primary Containment

Calculational results for reactor vessel and primary containment variables are shown on Figs. 3.1–3.8. Each system variable is discussed in the following paragraphs.

3.3.1 Reactor vessel pressure

Figure 3.1 shows the reactor vessel pressure. During the first several minutes, reactor pressure cycles between 7.72 MPa (1120 psia) and 7.38 MPa (1070 psia) on automatic actuation of a single safety relief valve (SRV). After about 200 s, the HPCI and RCIC systems initiate and begin drawing steam from the reactor vessel.* Figure 3.2 shows steam flow from the reactor vessel. Pressure decreases rapidly during HPCI system operation because of the steam flowing to the HPCI turbine and because of the quenching effect of the large amount of cool water being pumped into the reactor vessel. Each time the HPCI turbine is tripped (on high vessel level), reactor vessel pressure first decreases at a much lower rate and then begins to increase when boiling is reestablished in the core. When pressure has increased to 1120 psia, there is another period of SRV cycling until HPCI once again actuates on low reactor vessel level. Each time HPCI actuates, a lower minimum pressure is reached and the time required for pressure to recover to the SRV actuation pressure is longer. After the fifth HPCI cycle, pressure does not recover to the SRV actuation setpoint.

The long-term trend of decreasing reactor vessel pressure is a result of the SDV break and is accelerated because the break area has been assumed to be increasing as a result of CRD seal erosion. Relatively cool water is introduced to the vessel by the intermittent operation of the HPCI system and is heated to nearly saturation before flowing out the break (Fig. 3.3), thereby removing some of the core decay heat without steam production. As decay heat decreases, the core steam production decreases and the reactor vessel pressure eventually falls below that required for HPCI turbine operation.

Between about 234 and 306 min, pressure is prevented from going below 2.86 MPa (415 psia) by the action of the CPs and CBPs operating near their combined shut-off head. During the last half of this 32-min period, the

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*The HPCI system automatically initiates at a reactor vessel level of 476.5 in. and stops when the level has been restored to 582 in. The RCIC system, however, would make only one such cycle.

reactor vessel has been completely filled and no steam is being produced. When the hotwell is emptied, the CPs and CBPs lose suction, trip, and are no longer available to pressurize the reactor vessel. Accordingly, the pressure drops rapidly (about 300 psi in several minutes) until saturation pressure is reached in the reactor vessel.

After 320 min, the injected water has been heated to the saturation temperature, core steam production resumes, and reactor vessel pressure increases. Without injection, the vessel water level steadily decreases. There is a period of automatic SRV actuation shortly before core uncovery.

3.3.2 Reactor vessel level

Figure 3.4 shows schematically the sources of injection utilized during the no-operator-action sequence. Vessel level, injection flow rate, and total injected coolant volume are shown on Figs. 3.5–3.7. During the first 4 h following onset of the SDV leak, the HPCI system functions automatically to maintain vessel water level between the 12.1-m (476-in.) HPCI initiation setpoint and the 14.8-m (582-in.) HPCI high level trip setpoint. The effective flow area of the break is increasing, but the leakage rate is well below the pumping capacity of the HPCI system.

The HPCI room environment is not distressed sufficiently to fail the HPCI pump or turbine (see Sect. 3.4). Flooding in the HPCI room would not cause failure until a level of about 1.22 m (4 ft) is reached, and there is no possibility of direct impingement of water from the SDV break onto the HPCI pump or turbine. The supply of water in the condensate storage tank (CST) is initially 1371 m^3 (362,000 gal), and if this were depleted, HPCI pump suction would be automatically shifted to the suppression pool. Under these conditions, the HPCI should be capable of maintaining adequate vessel water level for about 16 h after onset of the SDV break; however, premature HPCI failure would be caused by uncontrolled and excessive vessel injection from the main feedwater system.

During the first 4 h, the electric motor driven CPs and CBPs are running continuously. The steam-driven main feedwater pumps are idle (because the MSIV's are closed), but flow can be forced through them by the CPs and CBPs. Figure 3.4 shows schematically the relationship between pumps in the main feedwater system. As long as vessel pressure stays above the combined shutoff head of the CPs and CBPs [about 2.86 MPa (415 psia)], there is no flow into the vessel. Damage to the pumps is prevented by an automatic recirculation valve that allows some water to flow from the hotwell, through the pumps, and back to the hotwell. There is very little effective cooling of the condensate during this sequence; therefore, during recirculation, the pumping power is dissipated within the condensate, causing it to heat up at about 7°C/h (13°F/h).

The first time vessel pressure is below the CBP shut-off head (about 195 min after onset of the break), the CPs and CBPs begin injecting water into the vessel, but vessel pressure recovers and the flow is shutoff. The second time, vessel pressure does not recover and the flow of condensate into the reactor vessel increases to several thousand gallons per minute. Vessel water level increases rapidly; pressure does not increase proportionately because steam is condensed from the diminishing reactor vessel steam space. When the vessel and the main steam lines (up to the MSIVs) are essentially full of water, vessel pressure increases slightly and the

flow continues at a reduced rate as necessary to compensate for leakage from the vessel. The presence of water in the steam lines would preclude HPCI system operation after this point. Without operator action to drain the flooded HPCI steam lines, an automatic HPCI start would admit water into the turbine at high velocity; this would probably fail the HPCI turbine, either directly by damaging the rotor or inlet valve, or indirectly by causing a break in the HPCI steam line, which would lead to closure of the HPCI steam line isolation valves.

After about 60 min, the CPs and CBPs have emptied the hotwell [assumed to hold an initial volume of 371 m^3 (98,000 gal) — slightly more than the volume at low level alarm], lost suction, tripped, and can no longer inject water into the vessel. During the period of CP and CBP operation, makeup water is automatically gravity-fed from the CST to the hotwell until the level in the CST has dropped to the inlet of the standpipe, which serves to reserve the last 511 m^3 (135,000 gal) of water for the HPCI and RCIC systems.

Vessel pressure falls rapidly after failure of CP and CBP injection, reaching a minimum of 0.88 MPa (128 psia). This pressure is just above the setpoint [0.79 MPa (115 psia)] for automatic isolation of the HPCI turbine steam supply line, and operator action would be required to restart the HPCI system if isolation occurred. However, at this time the HPCI would be inoperable anyway because of the flooding of the main steam lines and the HPCI steam supply line. From this time on, there is no further water injection into the reactor vessel.

After failure of CP, CBP, and HPCI injection, the vessel level decreases steadily because of the leak. Because no more cold water is being supplied to the vessel, the vessel water temperature increases to saturation and core steam production begins to repressurize the vessel. When the level reaches the 9.77-m (384.5-in.) automatic initiation setpoint of the RHR and CS systems, the associated pumps start but are not able to inject water into the vessel because vessel pressure is by this time above the effective design shut-off head of the pumps [203 MPa (295 psid) for RHR and 1.99 MPa (289 psid) for CS]. The Automatic Depressurization System (ADS) does not initiate to reduce vessel pressure because the initiation logic for this system requires a high drywell pressure signal coincident with low-sensed vessel level.* Vessel level therefore continues to decrease, and the core is uncovered about 7.4 h after onset of the SDV break.

3.3.3 Primary containment

Because the drywell coolers are not lost during the SDV break accident sequence and because the SDV leakage goes to the reactor building instead of the primary containment, there is no significant increase of drywell temperature or pressure. The drywell heat load would be increased

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*There is a proposed modification [Item II.K.3.18 of the Clarification of TMI Action Plan Requirements (NUREG-0737)] to the ADS logic that would result in ADS actuation in this circumstance by allowing a low HPCI flow signal to bypass the high DW pressure signal. This would have the effect of prolonging the period before core uncovery, which would not occur until after failure of the ECCS pumps caused by basement flooding.

because of the continuing passage of hot water through the uninsulated CRD scram outlet piping below the reactor vessel, but this increase would be offset by the decrease of drywell heat load when the main recirculation pumps trip [on low reactor vessel water level at 12.11 m (476.5 in.)].

As suppression pool water temperature (Fig. 3.8) increases during the period before core uncovery, water vapor evaporates from the pool surface, increasing the total pressure of the suppression chamber. Without the suppression chamber-to-drywell compressor (drywell DP compressor), this temperature increase would have the effect of increasing the suppression chamber pressure by about 9.7 kPa (1.4 psi), without causing a corresponding increase in drywell pressure. [Suppression chamber pressure must exceed drywell pressure by 3.45 kPa (0.5 psi) before the vacuum breakers would open.] With the DP compressor operating (as it would be unless off-site power were lost or unless the hot, steamy environment of reactor building elevation-565 floor level caused it to fail), some of the suppression chamber atmosphere is transferred to the drywell as the automatic compressor control maintains drywell pressure >9 kPa (1.1 psi) above suppression chamber pressure.* For this case, the net increase in suppression chamber pressure is 3.45 kPa (0.5 psi), and the net increase in drywell pressure is 2.76 kPa (0.4 psi). This increase in drywell pressure is not enough to exceed the 114-kPa (1.9-psig) setpoint for the "high drywell pressure" initiation signal for ADS actuation, ECCS, and containment isolation system.

Suppression pool water temperature is shown on Fig. 3.8. This figure illustrates that the suppression pool temperature increase is lower than might be expected. During a normal post-scram period of shutdown with the MSIVs shut, essentially all of the core decay heat must be dissipated in the suppression pool. However, with the SDV break, a large fraction of the core decay heat energy is deposited in the reactor building through the leakage path. As shown by Fig. 3.8, pool temperature has increased to only 49°C (120°F) after 6 h (without pool cooling because operator action would be required to line up the RHR to the pool-cooling mode). Without the SDV break, condensation of the steam generated by decay heat would cause the suppression pool temperature to increase to about 88°C (190°F) after 6 h.

Suppression pool level (not shown) increases by about only 0.1 m (4 in.) over the ~7-h period between event initiation and core uncovery because most of the water injected into the reactor vessel ends up on the reactor building floor. The 0.1-m (4-in.) increase in pool level is not enough to initiate the automatic shift of HPCI system suction from CST to suppression pool that would have occurred if pool level had increased by 0.28 m (11 in.) to an indicated level of 0.18 m (+7 in.).

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*Recent Browns Ferry modifications have shortened the drywell to suppression pool downcomers, thereby resulting in a lowering of the minimum drywell-suppression pool pressure difference from 9 kPa (1.3 psi) to 7.6 kPa (1.1 psi).

3.4 Results - Reactor Building

The Browns Ferry secondary containment consists of the three separate but adjacent reactor buildings and the single shared refueling bay. The reactor building is of low-leakage reinforced concrete construction and completely surrounds the reactor vessel and the primary containment. The refueling bay is of insulated sheet metal and steel beam construction. There is normally no direct communication between the reactor building of each unit and the refueling bay above; however, if the reactor building is internally pressurized to more than 1.72×10^{-3} MPa (36 lb/ft^2), the reactor building blow-out panels will open and relieve directly to the refueling bay.

The Unit 1 reactor building is divided into five major (concrete floor) levels. Table 3.2 lists several important equipment items located on each of the five levels. Each floor level consists of a large, mostly open room; there is direct communication between floor levels via open stairways. The communication afforded by the stairways is sufficient to essentially equalize pressure throughout the reactor building, but it is not sufficient to assure complete thermal mixing of the atmosphere at different floor levels.

The BWR-LACP reactor building model divides the reactor building into three regions: the basement, the elevation-565 room that houses the leaking SDV, and a composite region combining the volume of the upper three floors. Conditions are also calculated for the refueling bay, which begins to receive steam and noncondensible gases from the reactor building after blowout of the building relief panels.

The water leaking from the SDV break is a mixture of hot water from the 185 reactor vessel CRD mechanism seals (initially 550 gpm) and cold water pumped from the condensate storage tank by the CRD hydraulic system (170 gpm throughout the sequence).* Because the mixture is hotter than the 100°C (212°F) saturation temperature at atmospheric pressure, a considerable fraction (initially about one-third) flashes to steam at 100°C. The rate of steam release into the reactor building atmosphere caused by flashing from the leak is shown on Fig. 3.9. The remaining 100°C water spills or sprays onto the elevation-565 floor. Some of the spillage drains to the basement via the floor drains, and some flows through the steel grating stairways to the basement. For the calculations reported here, it is assumed that each of the four basement corner rooms (Chap. 4, Fig. 4.3) would receive an equal fraction of the drainage.

With respect to the no-operator-action sequence, the major benefit of the reactor building calculation during the period before core uncovery is the prediction of environmental extremes to which the HPCI system is exposed. As discussed in Sect. 3.3, the HPCI system is assumed to function normally during the first 4 h after the SDV break. The peak reactor building basement environmental conditions during the period are: 100% humidity, 76°C (168°F) average basement temperature, and 0.32-m (1.06-ft) flood level. The HPCI pump and turbine are on pedestals, and a flood level of ~1.22 m (4 ft) would be required for failure. As presented in TVA's response to the USNRC OIE Bulletin 79-01B, most of the HPCI system

*See the discussion in Appendix E, Sect. E.3.

components, with the exception of the turbine, are qualified for 100% humidity at 100°C (212°F) or higher. Although high-temperature/-humidity qualification of the HPCI turbine (mfg. Terry Turbine Co.) is not documented, the Terry turbine manual and the Browns Ferry FSAR specify that it would be operable to 64°C (148°F) in 100% humid air. This report assumes that exposure to the calculated maximum ambient temperature, which is 12°C (20°F) higher than this, would not cause failure of the turbine.

If air temperature in the vicinity of the HPCI steam line space exceeds 93°C (200°F), one of the four sets of temperature switches located along the HPCI steam line will initiate automatic closure of the HPCI steam line isolation valves, thereby failing HPCI injection. This is not likely to happen because a peak average basement temperature of 76°C (168°F) was calculated for this sequence. The HPCI steam supply line passes through the suppression chamber room and the HPCI room that, for this accident sequence, are expected to be somewhat cooler than the corner rooms. (Hot water and steam from the SDV break must pass through the corner rooms before reaching the HPCI or suppression chamber rooms.)

3.4.1 Reactor building pressure

A number of factors act to keep reactor building pressure very close to atmospheric pressure throughout the SDV break sequence. The following in-leakage rates were measured at Browns Ferry[3] for a reactor building and refueling bay pressure 1.72×10^{-3} MPa (0.25-in. water gauge) below atmospheric: 343%/d of refueling bay free volume and 110%/d of reactor building free volume. For lower building interior pressure the in-leakage rates will be larger; for interior pressures exceeding atmospheric pressure, the direction is reversed and in-leakage becomes out-leakage. For the BWR-LACP reactor building model, the building leakage (in or out) was assumed to be proportional to the 1.5 power of the difference between inside and outside pressure to approximate a combination of laminar and turbulent flow.

The building vacuum relief valves are designed to prevent interior pressures more than 1.24×10^{-4} MPa (0.5-in. water gauge) below atmospheric pressure. The reactor building blowout panels prevent interior pressure from exceeding refueling bay pressure by more than 1.73×10^{-3} MPa (7-in. water gauge). The refueling bay blowout panels prevent interior pressure from exceeding atmospheric by more than 2.47×10^{-3} MPa (10-in. water gauge).

Figure 3.10 shows reactor building and refueling bay pressures during the SDV break accident sequence. Pressure is initially at 1/2 in. below atmospheric pressure because the Standby Gas Treatment (SGT) system actuates before the SDV break,* with a total flow of about 13 m^3/s (28,000 cfm). After initiation of the break, the reactor building pressure climbs very rapidly. When pressure exceeds 1.73×10^{-3} MPa (7-in. water gauge) above refueling bay pressure, the reactor building relief panels blow,

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*The SGT system would actuate on low-sensed reactor vessel level at 13.7 m (538 in.) above vessel zero. The level decreases after the feed pumps trip following the scram.

allowing warm, humid air to flow into the refueling bay. The peak reactor building pressure is not sufficiently high to open the reactor building-to-suppression chamber atmosphere building vacuum breakers [0.5-psi (3.45-kPa) pressure difference required]. The refueling bay pressure relief panels do not blow because, as pointed out previously, the refueling bay has more out-leakage to the atmosphere for a given pressure difference.

After about 3 min of positive building pressure, the rate of expansion of the building atmosphere slows and the Standby Gas Treatment (SGT) system returns the reactor building to a slight vacuum and maintains building pressure below atmospheric for the remainder of the sequence before core uncovery (see Sect. 6 for a discussion of events after core uncovery).

Throughout the no-operator-action sequence, the building exhaust is very warm [66°C (150°F) average temperature] and saturated with moisture. The SGT system has a 40-kW air heating bundle in each of the three SGT exhaust trains to heat the incoming building exhaust by about 8°C (15°F). This is to prevent condensation in the charcoal filters, which can effectively retain iodine only when dry. The heaters are actuated whenever flow is maintained through the train and have a protective shutoff at 82°C (180°F). Because the building exhaust exceeds 82°C (180°F) throughout most of the no-operator-action sequence after core uncovery, the heaters will be shutoff, moisture will accumulate in the charcoal, and the iodine adsorption efficiency will be degraded. This fact (of minor significance to the present discussion) will be significant to calculation of fission product transport after the fuel damage phase of the no-operator-action sequence.

3.4.2 Reactor building atmosphere temperature and humidity

Figures 3.11 and 3.12 show reactor building and refueling bay temperatures and humidities during the period before core uncovery. The temperature or humidity reported for each volume is an average value, and some variation would be expected within each volume.

Since the floor level at elevation 565 receives all of the steam released by the leak, it experiences the highest temperature and humidity. The basement volume receives hot steamy air from elevation 565, and hot water from the leak floods the basement floor, transferring heat and moisture to the basement air. However, the ambient air temperature in the basement is limited somewhat by the ECCS room air coolers (two in each of the RHR corner rooms),* which start automatically when the ambient temperature exceeds 35°C (95°F),

Factors that tend to limit reactor building atmosphere temperatures include heat transfer to structures, condensation on structures, and the flow of outside air drawn into the building by the slight vacuum created by the SGT system. Many structures and miscellaneous items of equipment in the reactor building can act as heat sinks, but only the concrete

*There is a room cooler in each of the Core Spray corner rooms, but these coolers do not actuate unless the Core Spray pumps are running.

floors and walls were considered for this analysis. Because the heat transfer coefficient for natural circulation increases with increasing atmosphere-to-wall temperature difference, the heat transfer rate becomes significant only when room temperature is sufficiently elevated; therefore, atmosphere temperature rises rapidly at first, but the rate of increase slows as heat transfer to the heat sink becomes important. Similarly, the rate of condensation of steam from the atmosphere is initially limited by the rate of diffusion of water vapor through relatively dry air but becomes increasingly important as the concentration of steam increases.

Throughout the period of interest, reactor building temperatures fluctuate slowly in response to the fluctuation of the steam production rate of the leak. For example, when the steaming rate decreases, building temperature is decreased by mixing with outdoor air drawn in by the SGT system and by heat transfer to internal structures.

3.4.3 Basement flooding

Figure 3.13 shows basement water level (see also Chap. 4, Fig. 4.3). There are floor drains in each of the six basement rooms (HPCI room, corner rooms, and suppression chamber room). Floor drains in the east side of the basement floor are routed to the floor drain sump in the southeast corner room; floor drains in the west side of the basement floor are routed to the drain sump in the southwest corner room. The two floor drain sumps are connected by a 0.2-m (8-in.) overflow line. The degree of interconnection between rooms ensures that SDV leakage reaching the basement will result in a basement flooding level that is essentially the same in each of the six rooms. The combined capacity of the two floor drain sump pumps is 0.01 m^3/s (150 gpm). The sump pump motors are mounted on top of the sump covers (they are connected by shaft to the submerged impellers) and are assumed to operate until the basement flooding depth reaches 0.3 m (1 ft).

References

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2. L. F. Fidrych, *GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks*, NEDO-24342 (81NED261), Class I (April 1981).

3. Surveillance Instruction 4.7.C: "Secondary Containment," Tennessee Valley Authority, Browns Ferry Nuclear Plant (August 1979).

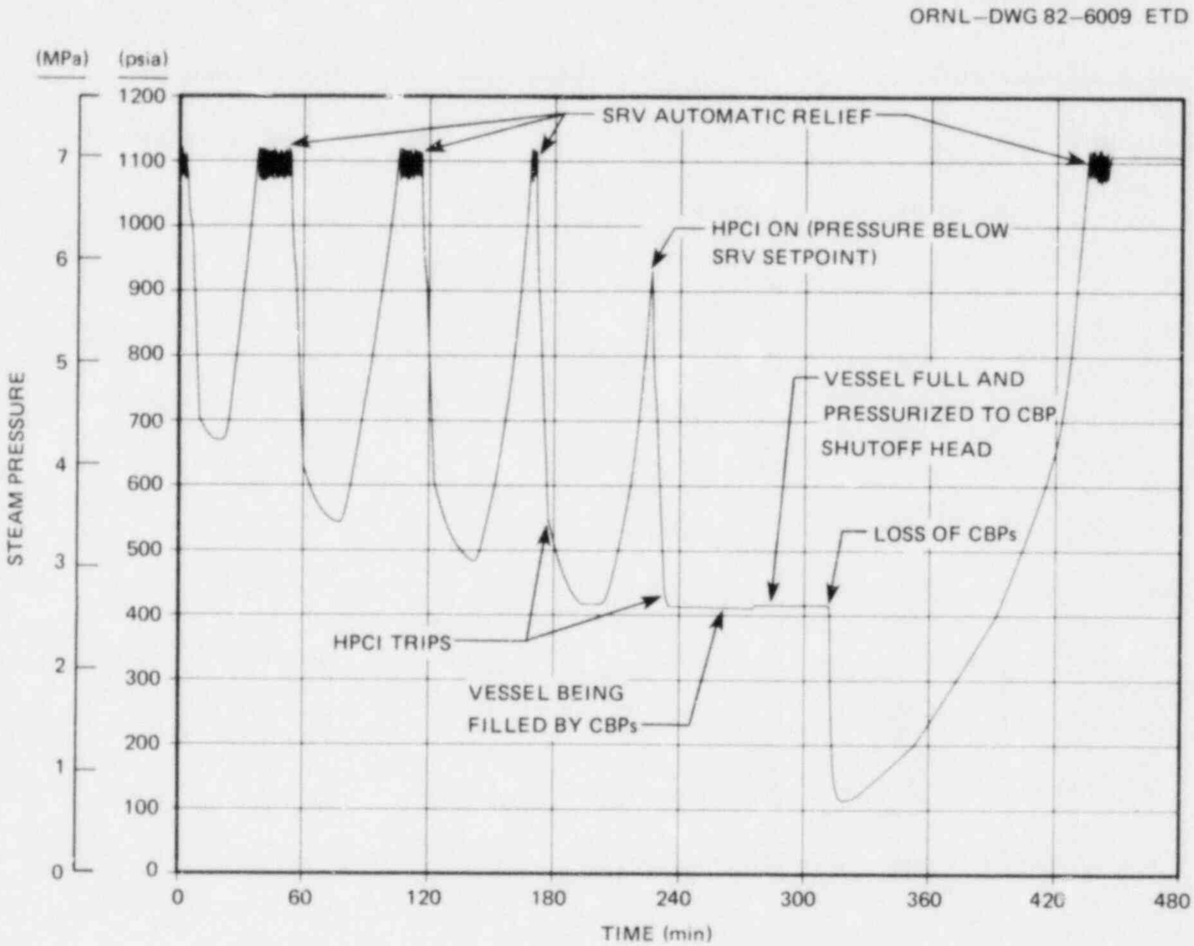

Fig. 3.1. SDV break sequence without operator action — reactor vessel pressure.

ORNL-DWG 82-6010 ETD

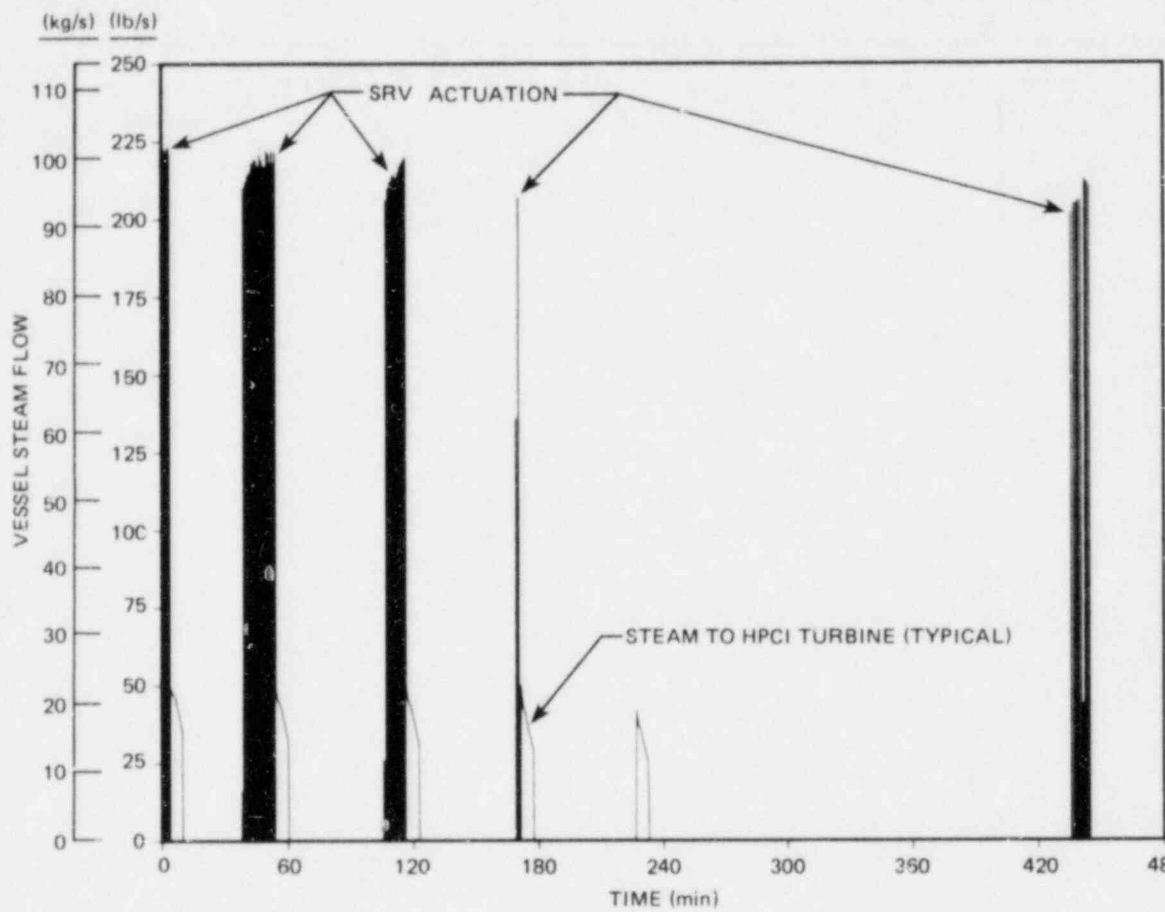

Fig. 3.2. SDV break sequence without operator action — reactor vessel steam flow.

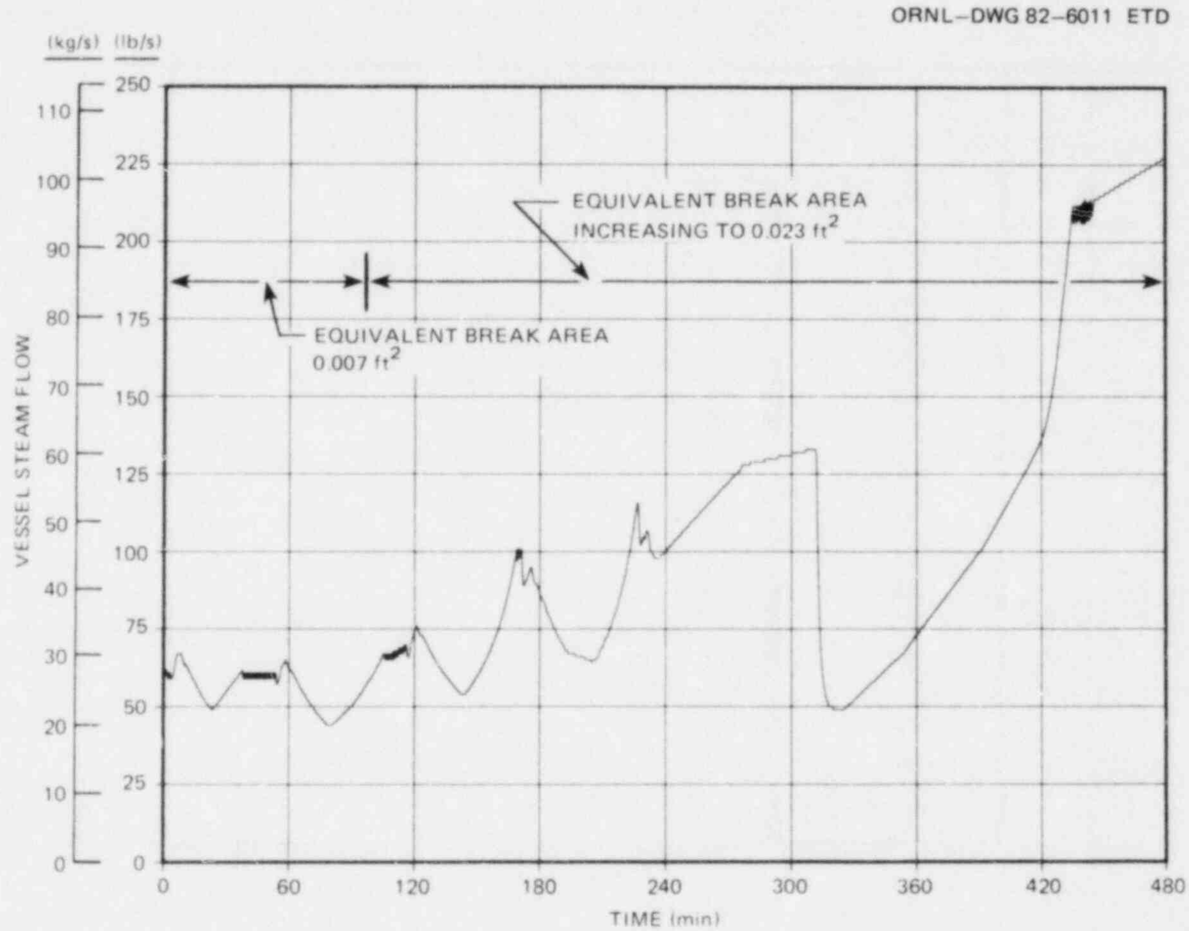

Fig. 3.3. SDV break sequence without operator action — leakage flow from reactor vessel.

ORNL-DWG 82-14975

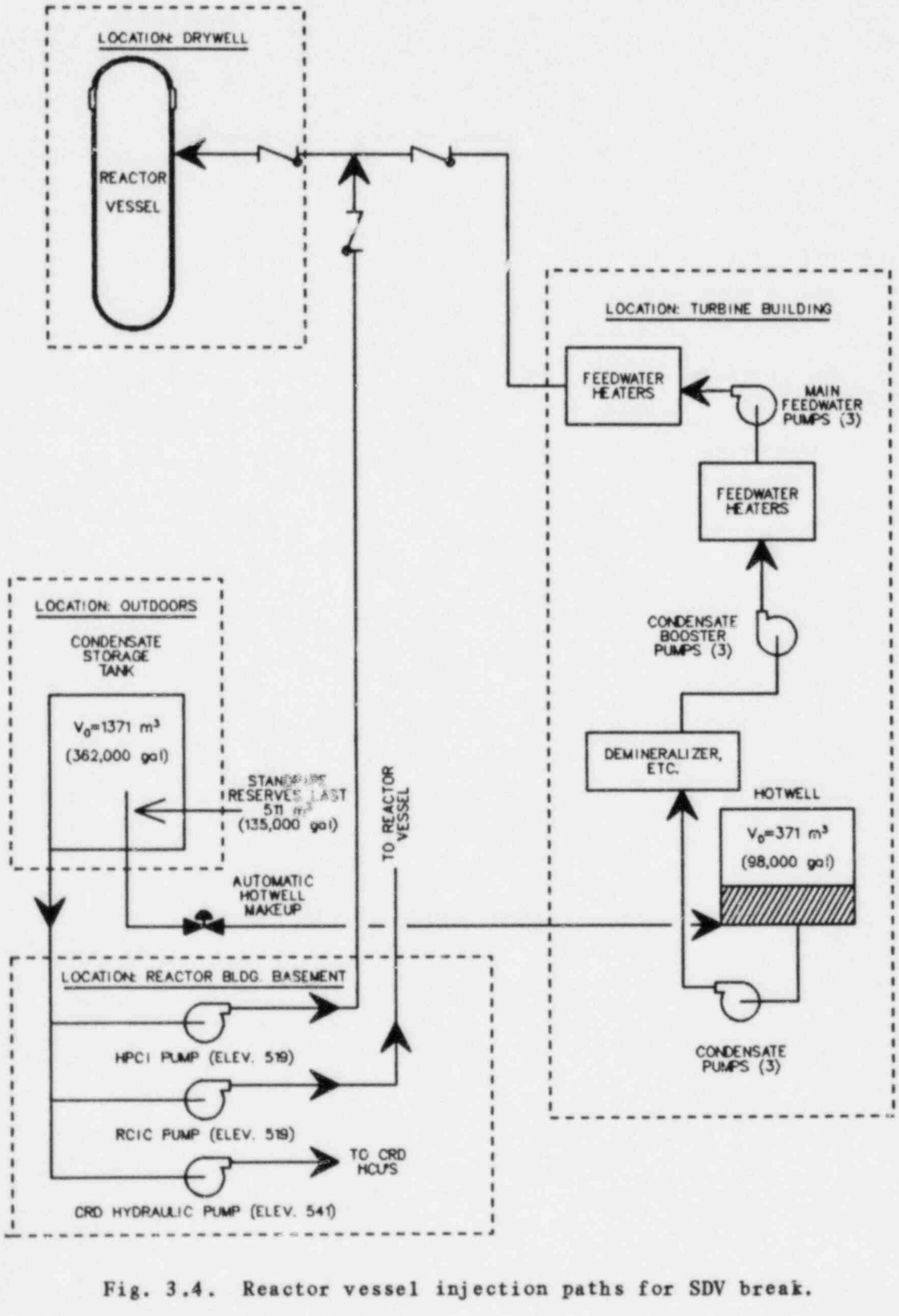

Fig. 3.4. Reactor vessel injection paths for SDV break.

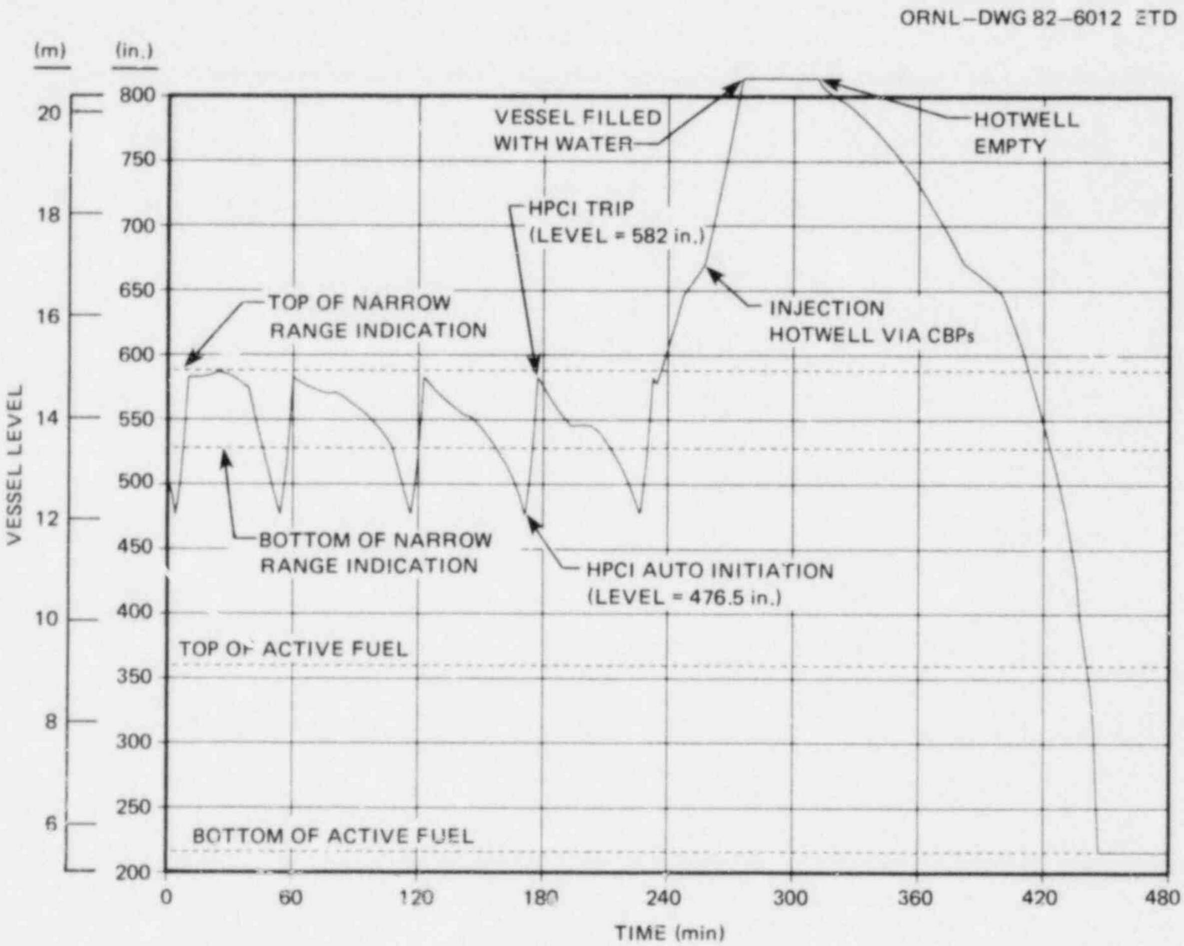

Fig. 3.5. SDV break sequence without operator action — reactor vessel water level.

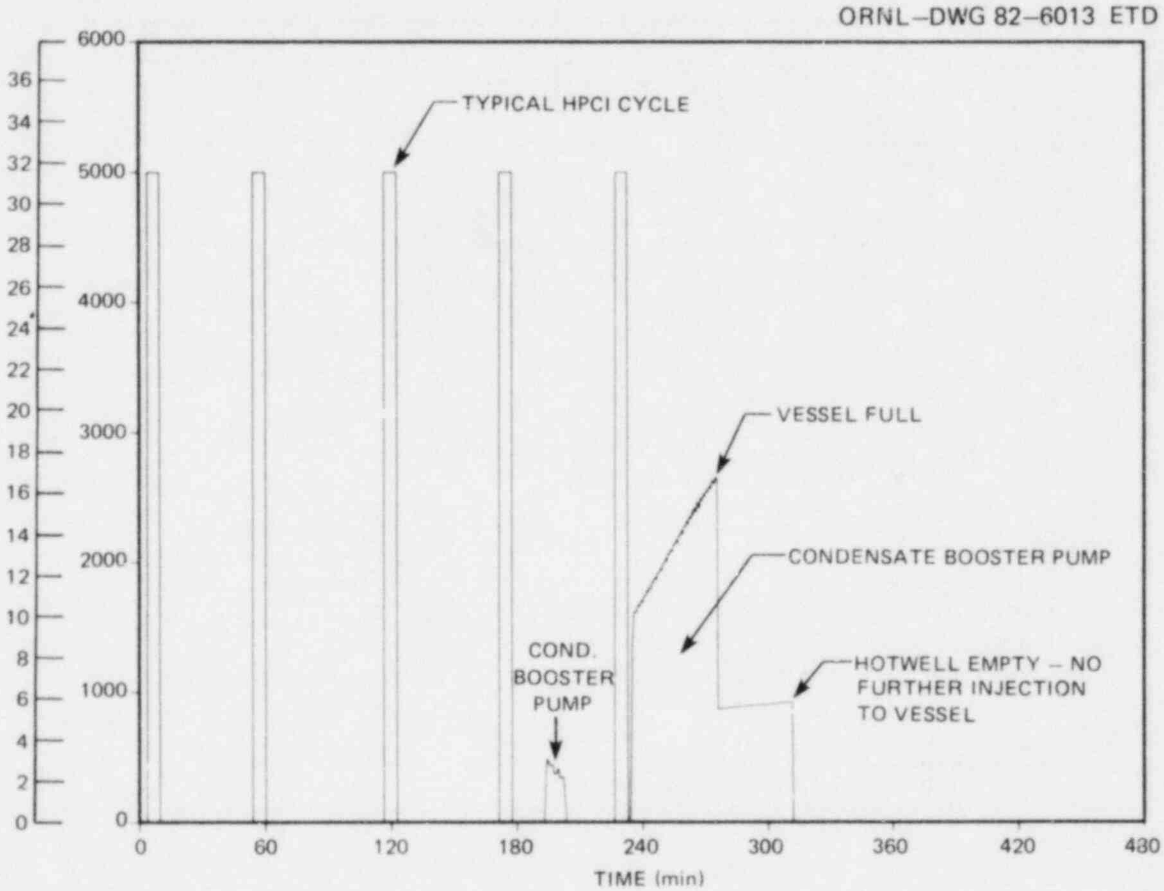

Fig. 3.6. SDV break sequence without operator action — injection flow into reactor vessel.

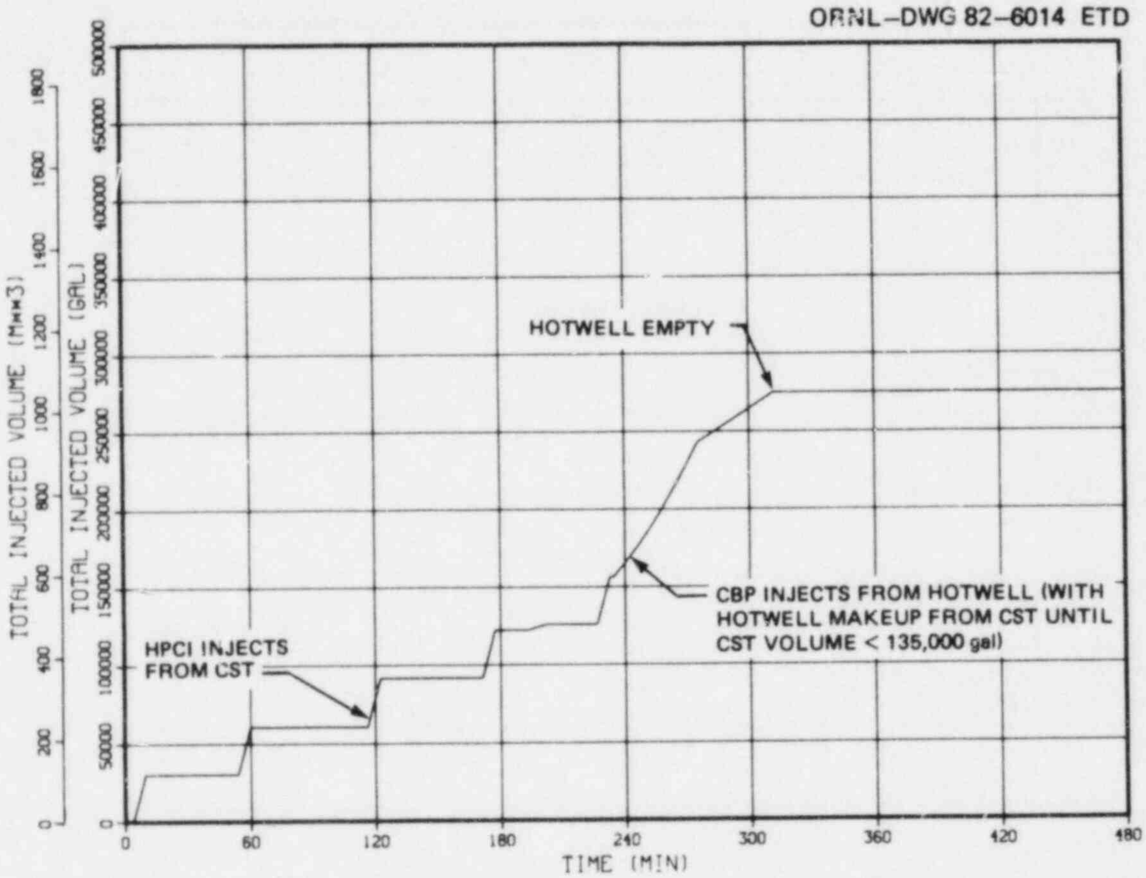

Fig. 3.7. SDV break sequence without operator action — total volume injected into reactor vessel.

ORNL–DWG 82–6015 ETD

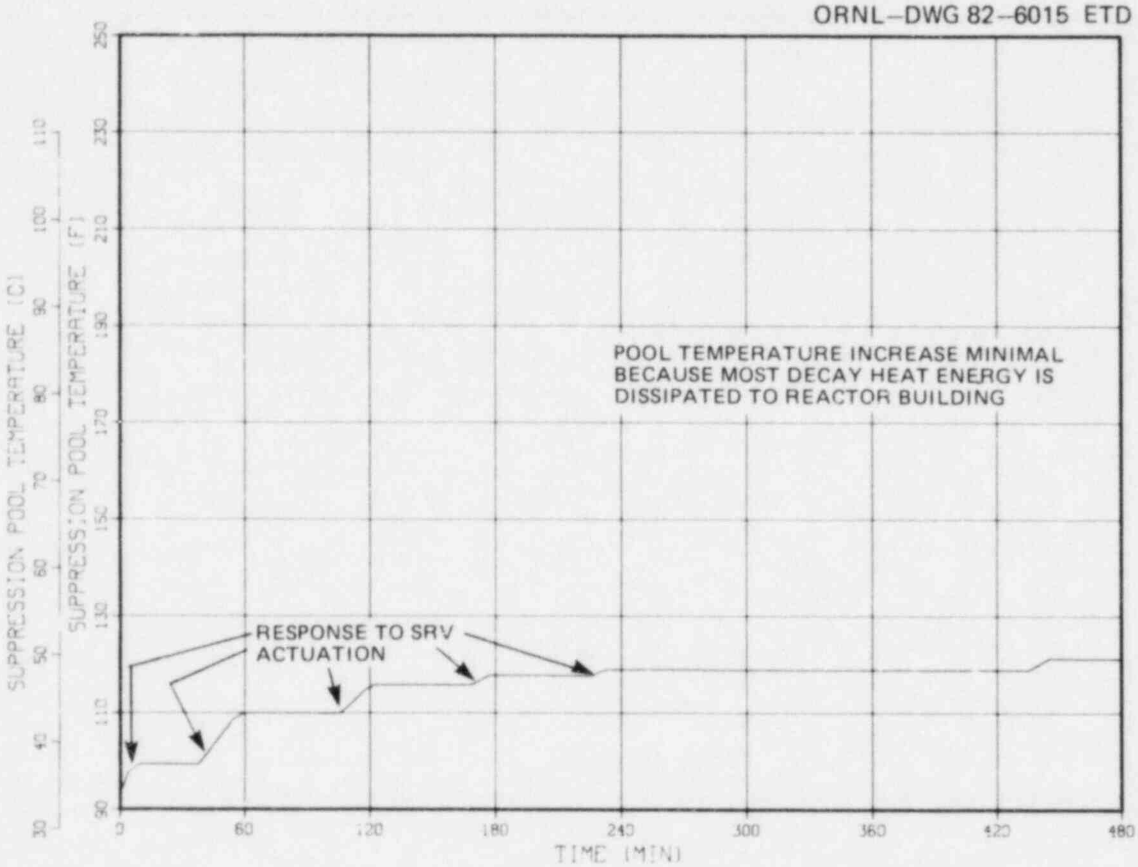

Fig. 3.8. SDV break sequence without operator action — PSP water temperature.

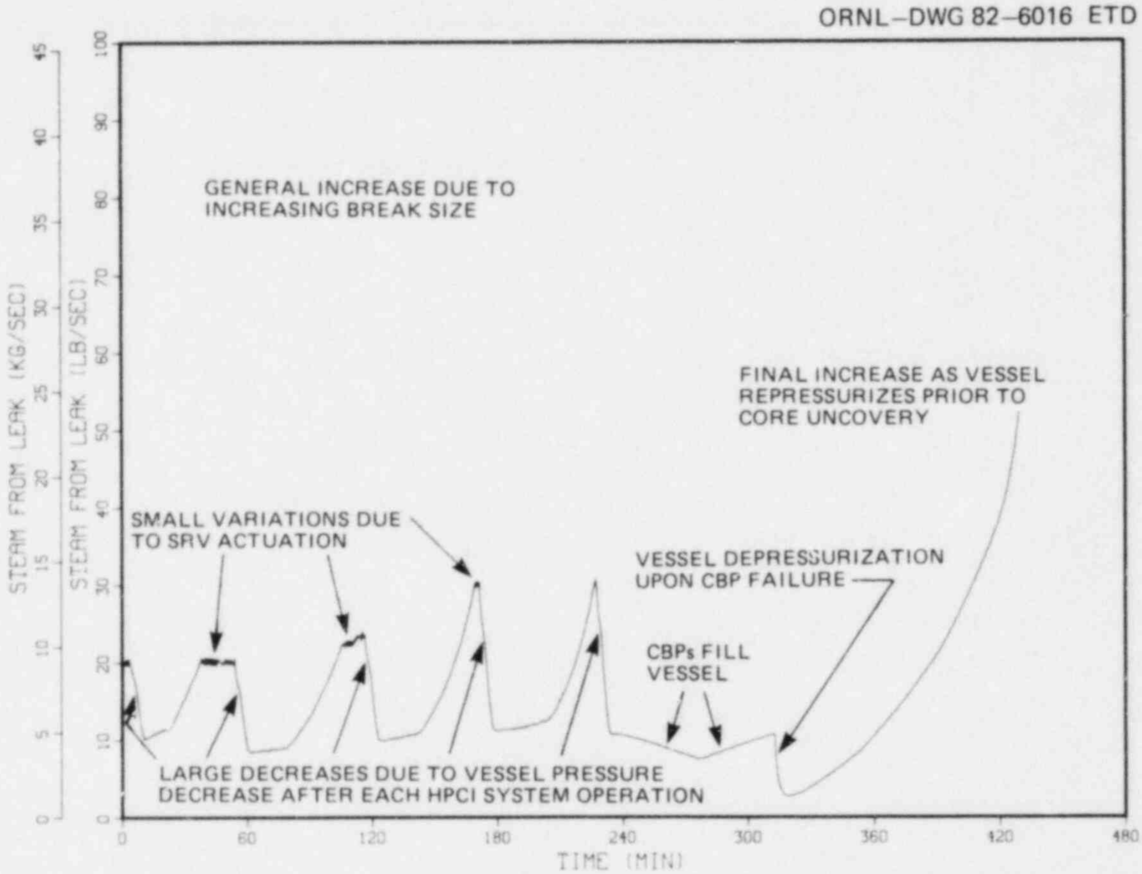

Fig. 3.9. SDV break sequence without operator action — steam release through break to reactor building.

ORNL-DWG 82-6017 ETD

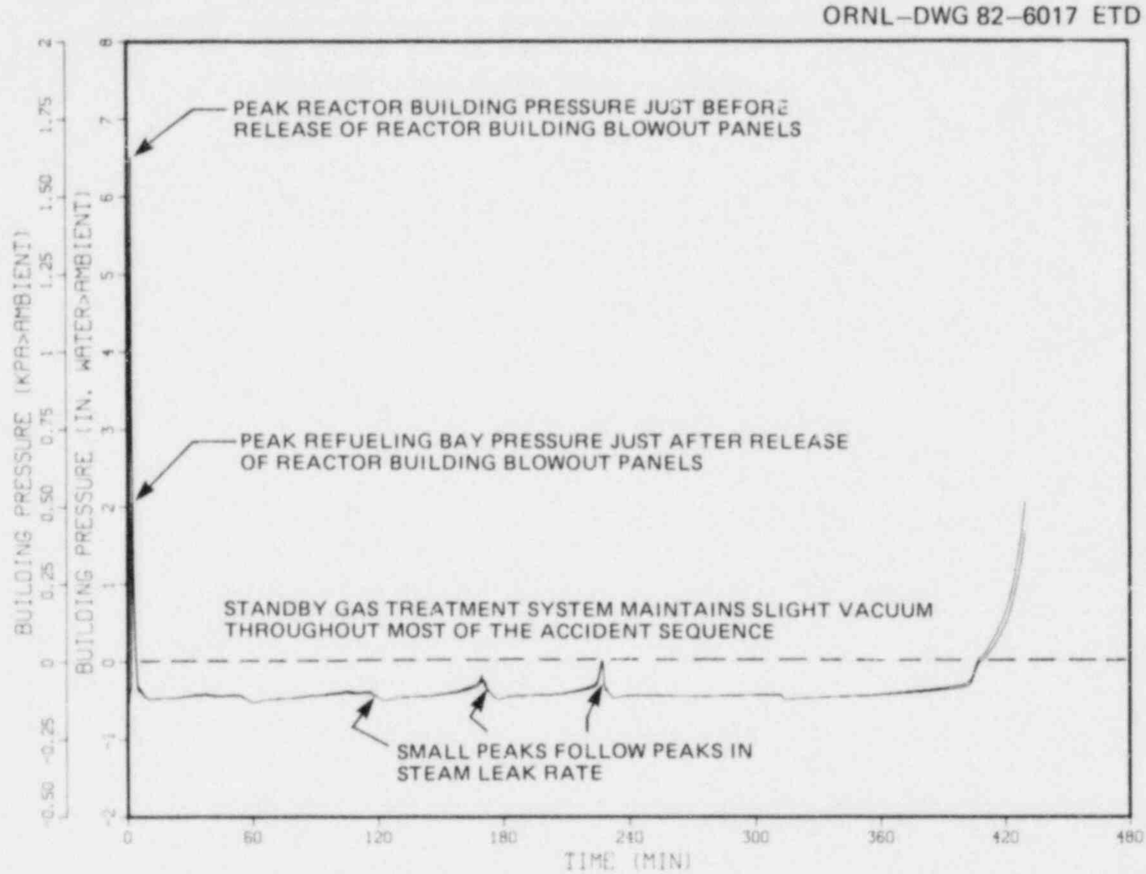

Fig. 3.10. SDV break sequence without operator action — reactor building and refueling floor pressures.

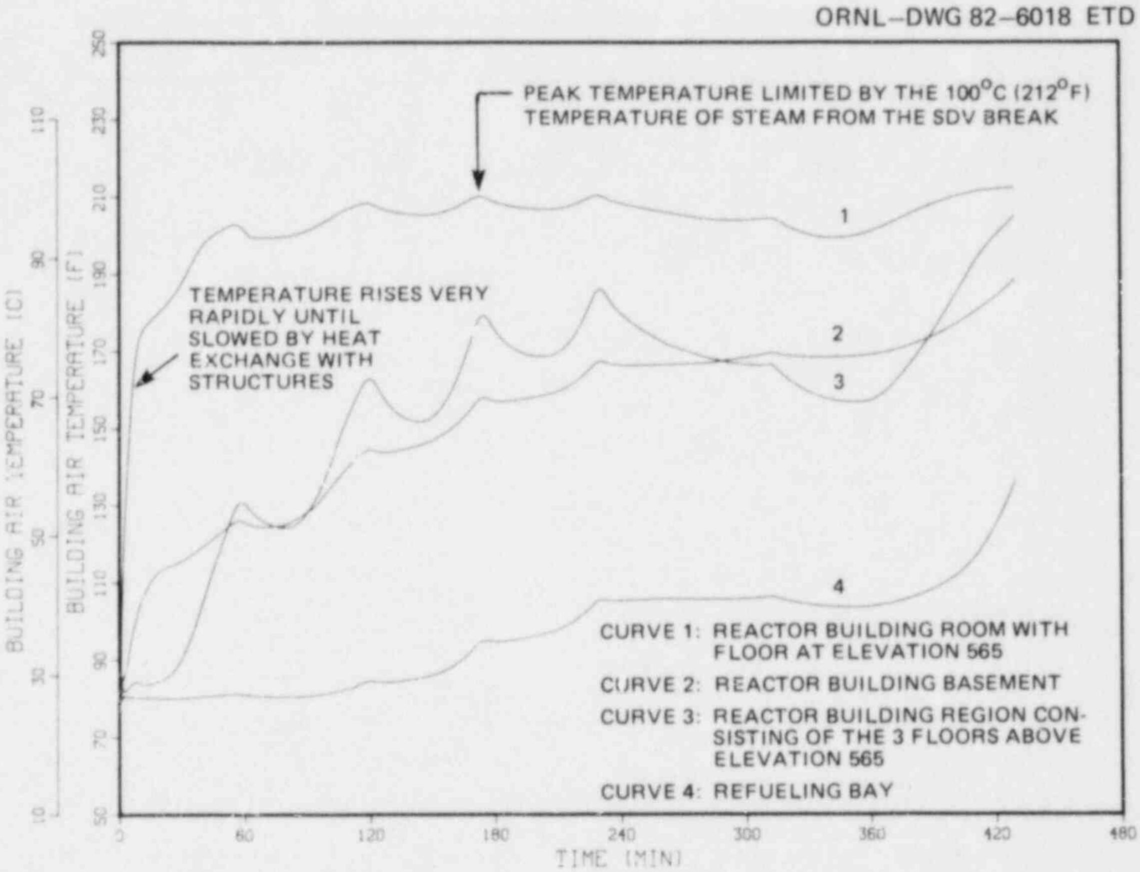

Fig. 3.11. SDV break sequence without operator action — reactor building and refueling bay atmosphere temperatures.

ORNL-DWG 82-6019 ETD

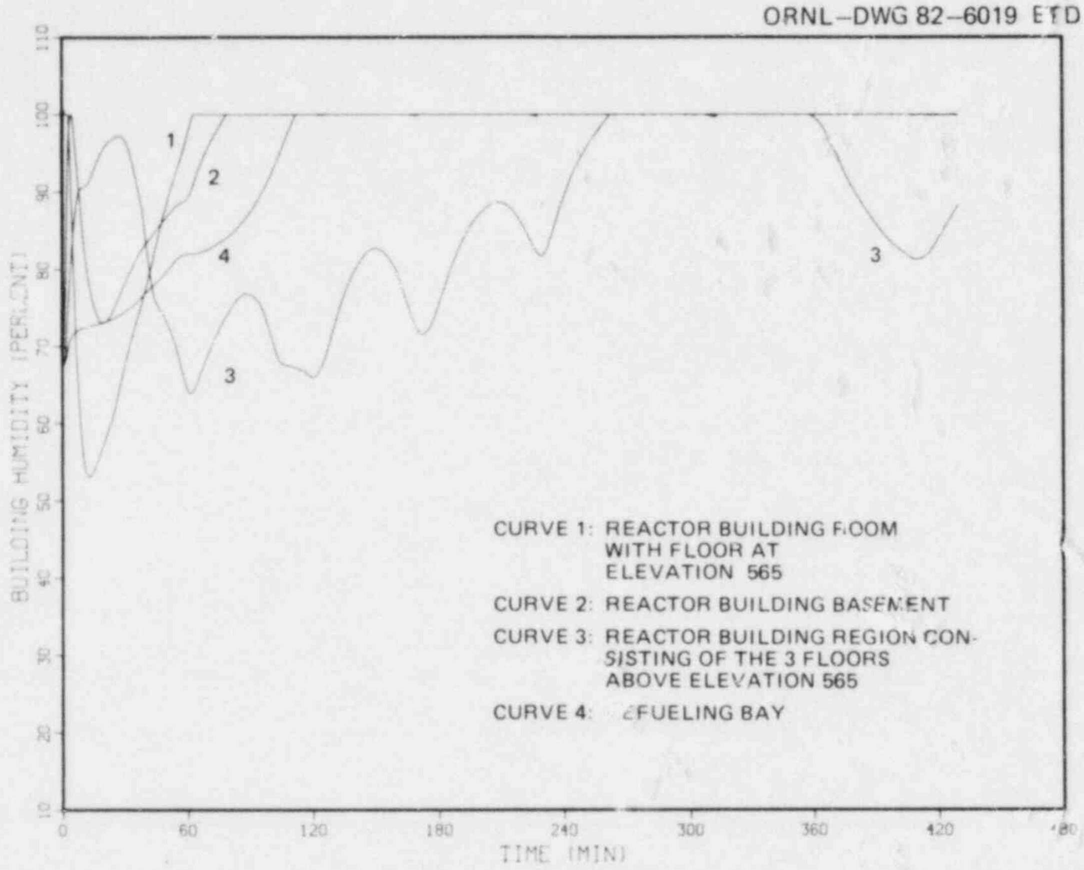

Fig. 3.12. SDV break sequence without operator action — reactor building and refueling bay humidities.

ORNL-DWG 82-15734

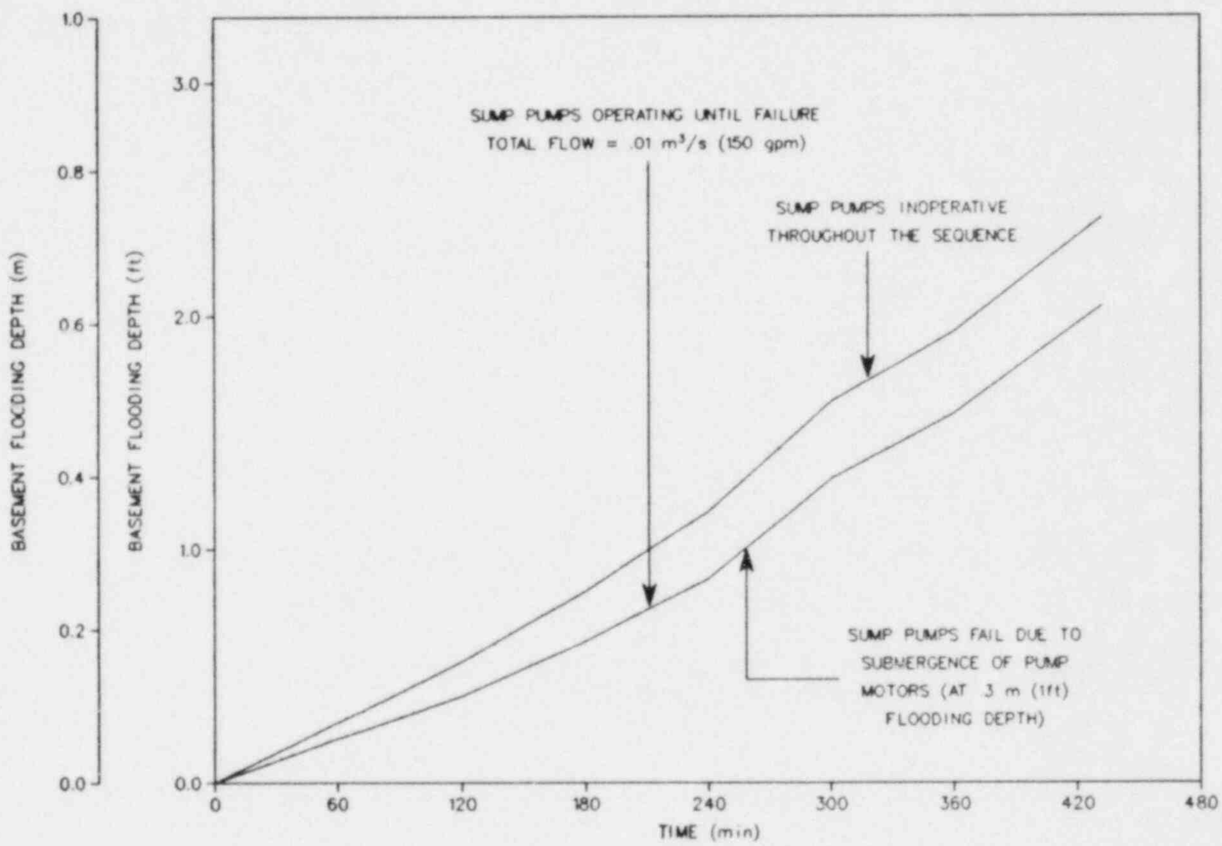

Fig. 3.13. SDV break sequence without operator action — depth of flooding in reactor building basement.

Table 3.1. Major events during first 8 h of no-operator-action sequence

Time (s)	Event
0	Reactor trip
30	SDV break
~60	Isolation of normal building ventilation and start of SGT system on high radiation at reactor building exhaust
200	HPCI and RCIC initiation on low reactor vessel water level; MSIVs close
575	HPCI and RCIC trip on high vessel level (RCIC has no subsequent automatic start)
200–13,940	Vessel level maintained between 12.1 m (476 in.) and 14.8 m (582 in.) by automatic initiation and trip of HPCI system
14,040	Vessel pressure goes below combined head of CP and CBP; condensate from hotwell being pumped into reactor vessel
16,510	Reactor vessel and HPCI/RCIC steam supply lines full of condensate pumped by CP and CBP
18,680	Hotwell empty, CBP tripped on low suction pressure, vessel injection stops
19,080	Reactor vessel pressure near HPCI automatic isolation pressure (isolation would not be automatically reset when pressure increases back above isolation setpoint)
26,400	Core beginning to be uncovered

Table 3.2. Major systems on BFNP#1 floor levels

Floor level [m (ft) above sea level]	Equipment
158 (519) (Basement)	ECCS pumps and HPCI pump and turbine RCIC pump and turbine CRD hydraulic system pump (platform-mounted at elevation 541) Suppression chamber Reactor building floor drain sump Reactor building equipment drain sump
172 (565)	SDV CRD hydraulic control units (185) RHR system heat exchangers Drywell-torus DP compressor
181 (593)	Reactor water cleanup (RWCU) system Closed cooling water heat exchangers RHR system heat exchangers
189 (621.25)	RWCU system Unit 250-V batteries
195 (639)	Standby liquid control system Main reactor vessel recirculation pump motor-generator sets

4. INSTRUMENTATION AVAILABLE FOR OPERATOR DIAGNOSIS AND RECOVERY

4.1 Introduction

Because loss of vital power has not been postulated, all plant instrumentation normally available after a reactor trip would be available for operator use during a Scram Discharge Volume (SDV) break accident. This section describes only plant instrumentation that would be helpful for detection of loss of primary coolant. The following three subsections discuss monitoring of the reactor vessel, primary containment, and secondary containment as applied to the general problem of detecting loss of primary coolant. The last subsection discusses use of CRD instrumentation for possible diagnosis of a leakage condition in the SDV. This section concludes that the existing Browns Ferry control room instrumentation is adequate to allow diagnosis of the leakage condition (although perhaps not the precise location) in a reasonable time. It must be realized, however, that the control room has many more annunciators and indicators (than discussed in this section) that compete for operator attention, and the overall effect can be overwhelming, making the task of detecting the loss of coolant somewhat more difficult than it may seem in the following discussion.

The important instrumentation for each plant system consists of both indicators and annunciators. These are conveniently located on one of the front panels of the U-shaped main control area unless otherwise noted. Indications display the measured value of a parameter. Annunciators warn of abnormal operating conditions as detected directly from a measured parameter or as detected indirectly via actuation of a safety system or other protective function. Upon initiation of an annunciator, an audible alarm (common to a group of annunciators) sounds and the backlighting for a small window, labeled to indicate the specific initiating condition, begins to flash. The audible alarm may be stopped by the operator at any time (even if the initiating condition persists) by depressing the acknowledge button for that set of annuciators. After the first acknowledgement, illumination of the small condition-specific window changes from flashing to solid and continues until the condition clears, and the operator again depresses the acknowledge button. Annunciators for the reactor trip function have a distinctive alarm horn tone, and the trip condition that initiated the scram is accented by solid illumination on a separate "first out" panel.

4.2 Reactor Vessel Instrumentation

The most important reactor vessel primary coolant leak detection instrument signals are those related to vessel coolant inventory. It is feasible to detect small line breaks and to distinguish between steam line breaks and liquid line breaks after the reactor is scrammed by applying

the following diagnostic rules:

1. A small liquid line break will require higher than normal vessel water injection flow but will involve decreased vessel steam production as evidenced by lower than normal vessel pressure or fewer than normal relief valve actuations.
2. A small steam line break may initially require higher than normal vessel water injection, but after vessel pressure stabilizes at a lower pressure the injection requirements will be very nearly normal.

It is unlikely that an operator would be able to discover a small line break using the "greater than normal injection" criterion because of the inherent difficulty of defining normal injection requirement and because of possible measurement inaccuracies.

Table 4.1 gives specific information on reactor vessel indications and annunciations. The important variables are discussed in the following paragraphs.

1. Vessel water level. Automatic controls (or the control room operators) will maintain vessel water level near normal for all but the largest line breaks. For example, the RCIC system could maintain a normal vessel level for a range of small but significant breaks. Therefore, level alone is not necessarily useful for detection of primary coolant leakage.

2. Vessel injection. After a normal reactor trip, water is injected into the reactor vessel to replace water boiled by decay heat or to replace water flashed if there is net loss of vessel pressure. Figure 4.1 shows the vessel injection flow required to remove decay heat as a function of time after reactor trip. If, during a period of shutdown at stable vessel pressure, more injection is required than that indicated in Fig. 4.1, then vessel water leakage would be strongly indicated. If the vessel injection is being supplied by the RCIC system (which is preferred over the HPCI system when high-pressure injection is required with closed MSIVs), as is assumed for the operator-action SDV break sequence in Sect. 5, then injection flow is in the 200- to 700-gpm range and can be monitored accurately. However, if vessel injection is provided by the large-capacity HPCI system or the main feedwater system, an additional injection requirement of several hundred gallons per minute could go unnoticed because the standard dP-type flowmeters for these systems would not be accurate in the bottom 20% of their indication ranges (ranges given by Table 4.1).

As an example of the possible use of Fig. 4.1 to detect an abnormally high injection flow requirement, consider the SDV break operator-action sequence with fast depressurization (Sect. 5); Fig. 5.10 shows vessel pressure and Fig. 5.14 shows injected flow rate. During the first 15 min, an HPCI injection cycle masks the effect of the leak, making RCIC injection flow appear normal. Between 15 and 80 min, the operator is depressurizing the reactor vessel and would therefore expect a high RCIC flow to replace water flashed during the depressurization. Between 80 and 200 min, Fig. 4.1 can be applied because pressure is stable and injection is being supplied by RCIC. During this period, RCIC flow averages 0.021 m^3/s (330 gpm), which is 0.006 m^3/s (100 gpm) above the average injection requirement specified by Fig. 4.1. The correct conclusion is, therefore, that

the average leakage flow between 80 and 200 min is about 0.006 m^3/s (100 gpm). Because plant emergency procedures do not require quantitative determination of excessive injection flow requirements, it is doubtful that such an estimate would be made during an accident.

3. Vessel steam flow. Direct measurement of steam flow after reactor trip will be in most sequences either inaccurate or indirect. If the MSIVs are open, the measurement will be inaccurate because the main steam flowmeters are sized for power operation and thus are inaccurate at low flows. With MSIVs shut (as is the case with the SDV break discussed in this report), steam flowing to the RCIC turbine can be measured with reasonable accuracy. In either case, flow through the SRVs is not measured and would be difficult or impossible to infer from the acoustic valve monitors or the recorded tailpipe temperatures. There is no control room indication of steam flow to the HPCI turbine.

4. Reactor vessel pressure. Operating procedures call for operators to maintain reactor pressure in the 900- to 1100-psi range after reactor trip, unless the decision has been made to depressurize and go to cold shutdown. If pressure cannot be maintained in this range or if the pressure is approximately stable in this range with no (or fewer than normal) relief valve actuations, then primary coolant leakage is strongly indicated.

4.3 Primary Containment

The Browns Ferry primary containment (drywell and communicating suppression chamber) forms an essentially leakproof enclosure around the reactor vessel. The SDV break sequence involves leakage from the reactor vessel into the reactor building outside the primary containment; nevertheless, a brief discussion of primary containment leak detection is provided because the very absence of these signals would provide information concerning the location of the break. Table 4.2 lists the major control room indications and annunciators for the drywell and suppression chamber. Because of the relatively small size and leakproof nature of the Browns Ferry primary containment, even relatively small leaks will cause control room instruments to respond sufficiently to allow effective operator diagnosis.

1. Drywell pressure. This is an early and direct indicator of reactor coolant leakage inside primary containment. A number of protective actions are automatically initiated when drywell pressure exceeds 0.115 MPa (2 psig): reactor trip, HPCI start, Primary Containment Isolation System (PCIS) Groups 2, 6, and 8 actuation, diesel generator start, emergency equipment cooling water (EECW) pump start, standby gas treatment system start, isolation of affected unit (i.e., Unit 1, 2, or 3) normal ventilation system, and isolation and emergency pressurization of the control bay.

2. Drywell atmosphere temperature will be elevated in the event of coolant leakage within the primary containment. Consideration of drywell pressure and temperature alone will not lead to 100% certain diagnosis of all in-containment line breaks because the failure of drywell atmosphere

cooling (i.e., without a line break) can also cause high-drywell temperature (Fig. 3.3 of Ref. 1) and a concomitant ~0.007-MPa (2-psi) elevation of drywell pressure.

3. Drywell sumps. Drywell equipment drain sump and floor drain sump water level, temperature, and sump-pump flow abnormalities are annunciated in the control room. These sump annunciations are very important because they provide early warning for very small leaks (i.e., about 5 gpm for floor drain sump or 20 gpm for the equipment drain sump) that may not significantly elevate drywell pressure or temperature; they also provide positive confirmation for larger leaks that may elevate drywell pressure and temperature slightly.

4. Suppression pool water temperature. The suppression pool is designed to condense the large amounts of steam that can be released to primary containment in the event of an LOCA. Peak primary containment pressure under accident conditions remains below the 56-psig drywell design pressure even in the event of complete severance of the largest primary coolant piping. During a large LOCA, the pool temperature would rise 10 to 20°F in a matter of minutes.

A secondary design goal for the suppression pool is that it should serve as a heat sink for decay heat when the main steam isolation valves are shut. During the first 6 h after reactor trip from full power with the MSIVs closed and the SRVs discharging to the suppression pool, the pool temperature will increase by about 50°C (90°F) (Ref. 1, Fig. 7.14) if no pool cooling is provided. A similar period of shutdown with the SDV break sequence described in this report (Sect. 3) would result in a much lower pool temperature because most of the decay heat energy discharged through the SDV break would be deposited outside the primary containment.

4.4 Secondary Containment

The Browns Ferry secondary containment consists of the three separate but adjacent reactor buildings with a common refueling bay shared by all three units (Fig. 4.2). The reactor building is of low-leakage reinforced concrete construction and completely surrounds the reactor vessel and the primary containment. The refueling bay is of insulated sheet metal and steel beam construction. There is normally no direct communication between the reactor building of each unit and the refueling bay above; however, if the reactor building is internally pressurized to more than 1.72×10^{-3} MPa (36 lb/ft^2), the reactor building blowout panels will open and relieve directly to the refueling bay.

Table 4.3 lists the control room indicators and annunciators relevant to detection of primary coolant leakage within the reactor building. Compared with the detection of leakage within the primary containment, the detection (by the operators using only control room instrumentation) of an unisolated leak within the reactor building is more difficult. This is due to the much larger size of the reactor building and the operation of the ventilation systems. However, an adequate set of indicators and annunciators ensure that the control room operators can detect in a reasonable time an unisolated reactor coolant leak of the magnitude assumed for the SDV break.

1. Reactor building atmosphere temperature is measured at 28 locations within 9 regions or systems. Table 4.4 specifies the location and the primary monitoring function of each of the temperature sensors. Excessive air temperature in any of the monitored regions would be annunciated in the main control room. Primary coolant leaking from the postulated SDV break would leak onto the floor of the reactor building at elevation 565 (Fig. 4.2) and drain to the basement via floor drains and stairways. There is no temperature sensor for the general elevation-565 floor levels, but the large amount of hot water accumulating in the basement would increase the ambient temperature in the reactor building basement and/or on a higher floor level sufficiently to actuate high-temperature annunciators in the control room. High temperature alone is not a sufficient condition for positive diagnosis of primary coolant leakage. Other causes, such as local upsets in normal ventilation flows, could actuate a high-temperature annunciator.

2. Reactor building radiation. Area radiation monitors (ARMs) are distributed throughout the reactor building (Table 4.5). A radiation dose rate of 100 mrem/h in excess of background at any one of the locations listed on Table 4.5 would actuate the reactor building high-radiation annunciator in the control room. The dose rate for each individual area is displayed by meters on the ARM panel in the control room. The occurrence of simultaneous high area radiation and high area temperature within the reactor building would be a strong indication that primary coolant was leaking outside of the primary containment.

For the SDV break sequence, the increase in area radiation exposure would probably be highest at the 565-elevation (east and west CRD-HCU areas listed as 20-21 on Table 4.5), somewhat lower in the reactor building basement rooms, and much lower on the upper floors (for example, in the vicinity of the "M-G set area" sensor on the elevation-639 floor level). This pattern of radiation levels within the reactor building should indicate the existence of a primary coolant leak in the vicinity of the hydraulic control units to the operator.

The reactor building area radiation monitoring system is not considered safety-related and is, therefore, not required to be environmentally qualified for high-temperature or water-spray service. The sensors most severely affected by the SDV break environment would be those located in the vicinity of the leaking SDV header. There are two sensors at opposite sides of the building: one in the HCU-East area and one in the HCU-West area. The wide separation between sensors makes it very unlikely that both would be exposed to direct water/steam impingement from the SDV break; therefore, the Operator-Action analysis presented in Sect. 5 assumes that at least one of these sensors would be available to annunciate in the control room when radiation level exceeds 100 mrem/h.

3. Reactor building flooding. As shown in Fig. 4.3, the reactor building basement (elevation 519) is divided into six rooms: the four corner rooms (which house the low-pressure ECCS and RCIC pumps), the HPCI room, and the central suppression chamber room. Flooding is monitored in each of the six rooms and annunciated in the control room if flood level exceeds 0.06 m (2 in.). Flood levels from SDV break leakage should rise approximately simultaneously in each of the six basement rooms because they are interconnected via the basement floor drain system (see also

Sect. 3.4.3). Basement flooding in combination with high radiation and/ or high area temperatures would make the diagnosis of primary coolant leakage almost inescapable.

4. Reactor building ventilation radiation. Normal ventilation exhaust is monitored, and an annunciator is actuated if the exposure rate exceeds 11 mrem/h above normal background. A high exhaust radiation signal isolates the reactor building from its normal ventilation supply and exhaust and actuates the Standby Gas Treatment system to provide filtered, elevated release of the potentially radioactive exhaust. A low reactor vessel water level signal* also initiates these actions. Depending on the initiating scram, the MSIVs might close in the SDV break accident sequence at the inception of the accident and cause a transient low reactor vessel water level. If this occurred, the normal ventilation system would be isolated and the normal exhaust radiation sensor would not be capable of detecting the increased radiation released to the reactor building after the SDV break.

5. Radwaste building abnormalities. Effluent from the reactor building equipment and floor drain sumps is automatically pumped from the sumps for processing. Liquid waste water from all three BFNP reactor units is processed in the radwaste building, which is equipped with a control room that is continuously manned. During an extended SDV break, the reactor building floor drain sump pump would operate continuously at a flow of about 9.5×10^{-3} m^3/s (150 gpm). This is a much higher rate of collection of this wastewater than the 7.57 m^3 (2000 gal) expected in the radwaste building from each unit during a normal 24-h period. Therefore, an extended SDV break would cause high collection tank water level, which would be annunciated in the radwaste control room. In addition, the reactor coolant leakage from the SDV system would be much more radioactive than reactor building normal floor drain water, which would eventually cause high-radiation alarms that are annunciated both in the radwaste control and the main control rooms.

4.5 Control Rod Drive System

The path of leakage from the reactor vessel to the SDV system goes through the 185 control rod drive hydraulic control units. No installed instrumentation directly measures this leakage. Existing instrumentation is, however, capable of warning operators of the existence of an unusual condition. If the following information were correctly interpreted, the operators could correctly diagnose the SDV break condition.

1. Control rod drive mechanism temperature is measured for each of the 185 control rods. There is a single high-temperature annunciator. Each of the CRD temperatures is indicated on a recorder located on a back panel area in the control room.

The thermocouple for each CRD temperature is located within and near the top of each CRD piston tube (see Appendix E for discussion of CRD system components). Normally, a flow of 1.9×10^{-5} m^3/s (0.3 gpm) of cooling water flows up past the thermocouple and into the reactor vessel via the

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*At 13.7 m (538 in.) above vessel zero.

CRD guide tubes, keeping the indicated temperature below 121°C (250°F). During a normal reactor trip, the upward flow is reversed, while reactor coolant leaks past the CRD mechanism seals and flows into the SDV. When the SDV is filled and pressurized, the cooling water again flows into the CRDs. Thus, a high CRD temperature condition is expected during normal reactor trips, but it should be a transient phenomenon limited to CRDs with higher seal leakage.

After a scram with an SDV break, the inward flow of cooling water would be lost, and reactor coolant would flow continuously downward through the mechanism seals because the SDV rupture, which is assumed to occur 30 s after the reactor trip, would preclude pressurization. Therefore, the high CRD temperature alarm would annunciate as for a normal scram but would not clear. Continued SDV leakage would cause elevated temperature in more and more of the 185 CRD mechanisms. The operator would have to make a special trip to the back panel recorder to discover the number of affected units, but he would be likely to take this action in an extended accident sequence.

2. Rod drive position indication. During a normal reactor scram, the scram outlet valves are opened to create a low pressure on the discharge side of each of the 185 drive pistons. The net inward force drives the control rods into the core and continues to hold them several inches above their normal latched, full-in "00" position. The CRD position indication display illuminates the green full-in indication light on each of the 185 rod drives but presents no position indication because the drives are held above the position detector reed switches. Within about 1 min after the scram, the SDV fills and pressurizes to reactor pressure, thereby equalizing the pressure on both sides of the drive pistons and allowing most of the drives (except perhaps the drives with newly replaced seals) to settle into the normal latched full-in position. At this time, most of the 185 displays would show a "00" position in addition to the green full-in indication.

In the event of a large SDV break, the SDV will not pressurize. Therefore, the rod drive position display would fail to show the "00" position indication that is expected for most of the rod drives shortly after a normal scram. This lack of the usual indication, combined with the unusual number and persistance of CRD high temperatures, could lead to the specific discovery of the SDV leakage condition.

Reference

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

ORNL-DWG 82-14976

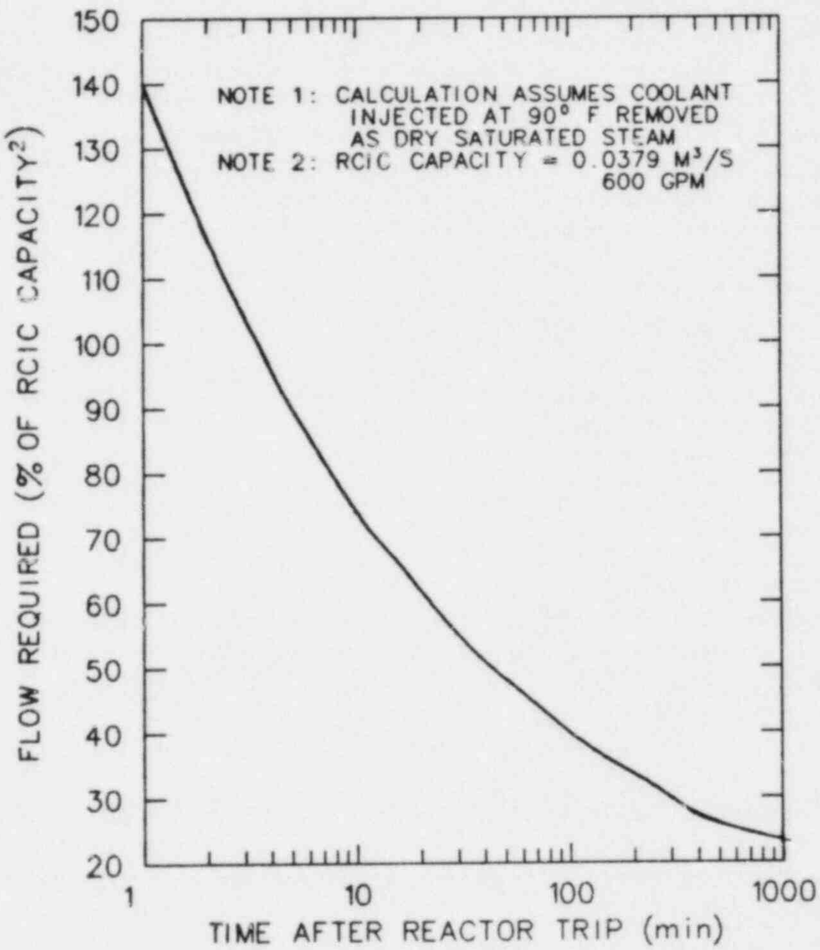


Fig. 4.1. Total injection flow (expressed as fraction of RCIC capacity) required to remove decay heat after reactor trip from full power.

ORNL-DWG 82-6007 ETD

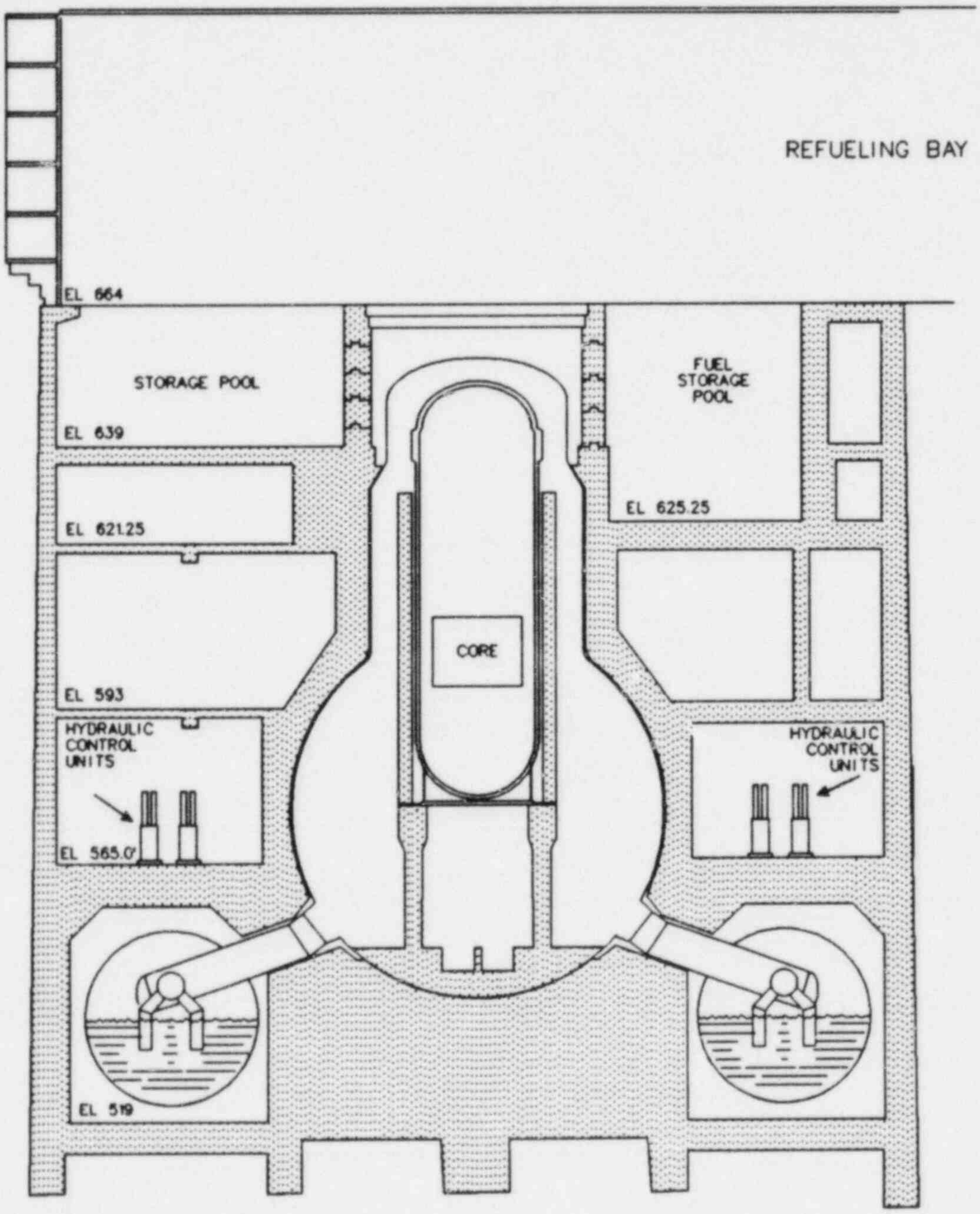
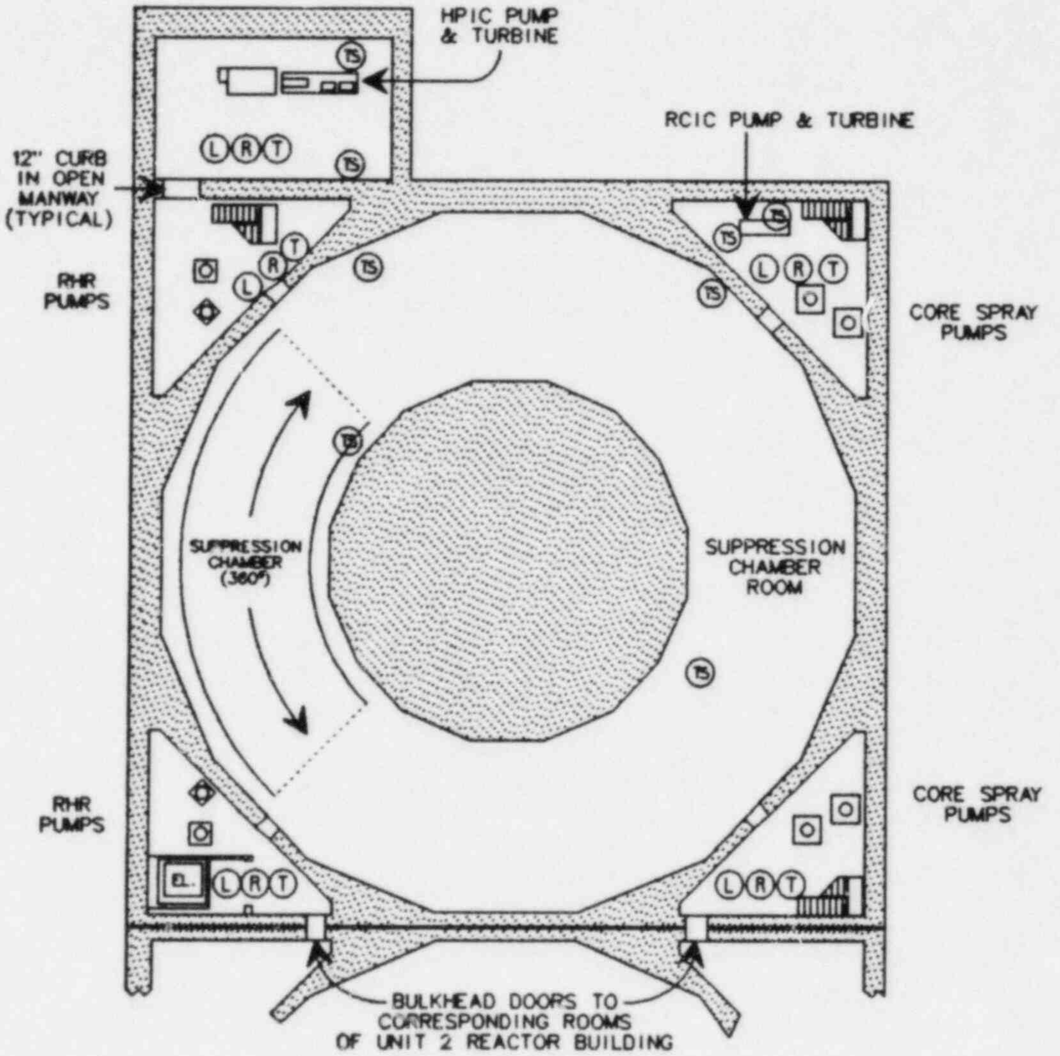


Fig. 4.2. Longitudinal section of BFNP#1 reactor building.

ORNL-DWG 82-14977



L = WATER FLOODING DETECTORS

T = ROOM AIR TEMPERATURE SENSORS

TS = AIR TEMPERATURE SENSORS NEAR RCIC OR HPIC STEAM LINES

R = RADIATION MONITOR

Fig. 4.3. BFNP#1 basement leak detection sensors.

Table 4.1. Reactor vessel parameters for loss-of-coolant detection

Measured parameter	Function[a]	Indication range or alarm setpoint
Reactor vessel		
Water level, m (in.)	I	13.4 (528)–14.9 (588)
	I	9.5 (373)–14.9 (588)
	I	13.4 (528)–23.6 (928)
	I	6.6 (260)–14.9 (588)
	A-high	14.8 (582)
	A-high	14.4 (567)
	A-low	14.1 (555)
	A-low	13.7 (539)
Pressure, MPa (psia)	I	0 (0)–8.28 (1,215)
	A-low	6 (865)
	A-high	7.3 (1,055)
Injection flow, m^3/s (gpm)		
RCIC	I	0–0.04 (700)
HPCI	I	0–0.38 (6,000)
Main FW (loop A or B)	I	0–1.1 (17,500)
CS (System 1 or II)	I	0–0.63 (10,000)
RHR	I	0–0.63 (10,000)
Steam flow – RCIC, kg/s (lb/h)	I	0–6.3 (50,000)
Steam flow – HPCI	I	(No control room indication provided)
Steam flow – main (4 indicators), kg/s (lb/h)	I	0 505 (4,000,000)
Steam flow – RCIC, MPa (psia)	A-high	225% at 1140 (7.86), 210% at 165 (1.14)
Steam flow – HPCI, MPa (psia)	A-high	225% at 1140 (7.86), 210% at 165 (1.14)
Steam flow – main steam (loop A or B), %	A-high	140%

[a] I = Indication.
A = Annunciation.

Table 4.2. Primary containment variables for loss of coolant detection

Measured parameter	Function[a]	Indication range or alarm setpoint
DW		
Pressure, MPa (psia)	I	0–0.55 (80)
MPa (psig)	A-high	0.112 (1.6)
	A-high	0.113 (1.65)
	A-high	0.114 (1.75)
	A-high	0.115 (2.00)
Atmosphere temperature, °C (°F)	I	0–204 (400)
	A-high	63 (145)
Sump temperature,[b] °C (°F)	A-high	54 (130)
Sump level,[b] m^3/s (gpm)	A-abnormal	Leakage in 3.16×10^{-4} (5) range
Sump pump flow[b] (time-averaged), m^3/s (gpm)	A-high	leakage in 3.16×10^{-4} (5) range
DW radiation, mrem	A-high	100 (above background)
PSP		
Pressure, MPa (psia)	I	0–0.55 (80)
MPa (psig)	A-high	0.115 (2)
Water temperature, °C (°F)	I	0–204 (400)
	A-high	35 (95)
Water level, m (in.)	I[c]	−0.64 (−25) +0.64 (+25)
	A-abnormal	Level outside −0.15 (−6) to −0.05 (−2) range

[a]I = Indication.
A = Annunciation.

[b]Provided for both equipment and floor drain sumps except that the alarm setpoint for equipment drain sump is 20 gpm.

[c]Instrument zero is 4.6 m (15.2 ft) above the bottom of the torus. A water level of zero indicates the torus is approximately half full of water.

Table 4.3. Reactor building parameters for loss of coolant detection

Measured parameter	Purpose	Indication range or alarm setpoint
Temperature, °C (°F)		
Area monitoring[a]	I	38–151 (100–400)
RCIC steam line space	A-high	79 (175)
HPCI steam line space	A-high	79 (175)
Main steam line space	A-high	79 (175)
RHR[b] rooms temperature or humidity, °C (°F)	A-high	63 (145)
CS[b] rooms temperature or humidity, °C (°F)	A-high	63 (145)
RWCU system floor drain or area temperature, °C (°F)	A-high	49 (120)
Water flooding detectors, m (in.)		
RHR pump rooms	A-high	0.05 (2)
CS pump room	A-high	0.05 (2)
CS/RCIC pump room	A-high	0.05 (2)
Suppression chamber room[b]	A-high	0.05 (2)
HPCI room	A-high	0.05 (2)
Radiation, mrem/h		
Reactor, turbine building[c] area monitors	I	0.1–1000
Reactor building area monitors	A-high	100 above background
Reactor building normal ventilation exhaust	A-high	100 above background
Refueling zone normal ventilation exhaust	A-high	100 above background

[a]See Table 4.4 for more details on area temperature monitoring.

[b]See Fig. 4.3 for basement elevation.

[c]See Table 4.5 for more details on area radiation monitoring.

Table 4.4. BFNP reactor building area temperature monitoring

System	Sensor location[a]	Elevation[b] above sea level [m (ft)]
RWCU	Demineralizer 1B tank room	199 (653)
	Demineralizer 1A tank room	199 (653)
	In RWCU valve room	189 (621)
	Above RWCU recirculation pump 1A	181 (593)
	Above RWCU recirculation pump 1B	181 (593)
	Above flow control valve 69-2	181 (593)
	Above RWCU heat exchanger	181 (593)
	In front of room exhaust duct for RWCU	181 (593)
RCIC	Above RCIC turbine	158 (519)
	Above RCIC steam supply line	166 (550)
	Above RCIC steam supply line	166 (550)
	Above RCIC steam supply line	166 (550)
HPCI	Above HPCI turbine	158 (519)
	Above HPCI steam supply line	166 (550)
	Above HPCI steam supply line	166 (550)
	Above HPCI steam supply line	166 (550)
RHR	In RHR System I pump room	161 (528)
	In RHR System II pump room	161 (528)
	In RHR System I heat exchanger room	181 (593)
	In RHR System II heat exchanger room	181 (593)
	In RHR valve room	172 (565)
	Above flow control valve 74-76	194 (635)
	Above RHR System II supply line	169 (555)
	Above RHR System I supply line	169 (555)
Main steam	Outside outboard MSIVs	172 (565)
	Above main steam lines in steam vault	172 (565)
	Above main steam lines below bypass valves	182 (596)
	Above main steam control valves	187 (615)

[a]The display panel is located at a back panel in the main control room.

[b]See Fig. 4.2 for illustration of floor level elevations.

Table 4.5. BFNP area radiation monitor locations

Station No.	Sensor and converter location	Range (mrem/h)
1	Reactor building fuel storage pool	0.1–10^{3}
2	Reactor building service floor area	0.1–10^{3}
3	Reactor building new fuel storage	0.1–10^{3}
4	Reactor building M-G set area	0.1–10^{3}
5	Turbine building generator operating floor	0.1–10^{3}
6	Turbine building RFP operating floor	0.1–10^{3}
7	Turbine building turbine operating floor	0.1–10^{3}
8	Main control room	0.1–10^{3}
9	Reactor building cleanup system area	0.1–10^{3}
10	Turbine building feedwater heater area	0.1–10^{3}
11	Turbine building SJAE and SPE area	0.1–10^{3}
12	Turbine building feedwater heater area	0.1–10^{3}
13	Reactor building north cleanup system area	0.1–10^{3}
14	Reactor building south cleanup system area	0.1–10^{3}
15	Turbine building decontamination room	0.1–10^{3}
16	Turbine building hotwell pump area	0.1–10^{3}
17	Turbine building condenser corridor	0.1–10^{3}
18	Turbine building raw cooling water pump	0.1–10^{3}
18	Turbine building condensate demineralizer area	0.1–10^{3}
19	Turbine building outside steam line cavity	0.1–10^{3}
20	Reactor building CRD-HCU west area	0.1–10^{3}
21	Reactor building CRD-HCU east area	0.1–10^{3}
22	Reactor building tip room	10.0–10^{6}
23	Reactor building tip drive area	0.1–10^{3}
24	Reactor building HPCI room	0.1–10^{3}
25	Reactor building RHR west room	0.1–10^{3}
26	Reactor building core spray – RCIC room	0.1–10^{3}
27	Reactor building core spray room	0.1–10^{3}
28	Reactor building RHR east room	0.1–10^{3}
29	Reactor building suppression pool area	0.1–10^{3}
30	Stack room	0.1–10^{3}
31	Radwaste building service water booster pump area	0.1–10^{3}
32	Radwaste building equipment drain sump area	0.1–10^{3}
33	Radwaste building radwaste control room	0.1–10^{3}
34	Radwaste building access corridor	0.1–10^{3}
35	Radwaste building waste packaging area	0.1–10^{3}
36	Radwaste building waste sample tank area	0.1–10^{3}
37	Radwaste building F.D. sample tank area	0.1–10^{3}

5. THE ACCIDENT SEQUENCE WITH OPERATOR ACTION

5.1 Introduction

This section presents the results of BWR-LACP calculations of reactor vessel and reactor building thermal-hydraulic response for SDV break sequences with operator control. As discussed in Sect. 2, the initiating event is assumed to be a reactor trip that cannot be reset. Furthermore, the MSIVs are assumed to be shut throughout the sequence, as they would be if the initiating event were a reactor trip on detected high main steam radiation. This assumption is reasonable for other initiating events as well because of the likelihood of a Group I isolation early in the sequence on high main steam space temperature [at 93°C (200°F)]. During normal power operation, the main steam space (through which the main steam lines pass between the drywell and the turbine building) is cooled by the normal building ventilation, which maintains a flow of air entering the steam space from the reactor building elevation-565 floor level. During the SDV break sequence, the SGT system maintains a similar flow of air, but the temperature of the incoming air is above 88°C (190°F) within 15 min of sequence initiation and therefore cannot effectively cool the main steam space.

Prior to recognition of the existence of a small break in the primary system, the operator would be performing the normal post-scram recovery actions that in this case include the actuation of the RCIC system to provide reactor vessel water level control. One of the first tasks normally performed by the operator following a scram is an attempt to reset the scram from the control room. If the scram condition has cleared, the scram outlet valves close, isolating the SDV break from the reactor coolant system and effectively terminating the accident sequence. If the scram condition has not cleared, then the scram cannot be reset, and the accident sequence should proceed approximately as outlined in this section.

Once the operator has recognized the existence of a small break in the primary system, EOI No. 15, "Breaks/Leaks Outside of Primary Containment," becomes applicable. This EOI does not call for depressurization unless the vessel level cannot be maintained by the high pressure injection systems, which in the case of the SDV piping break accident described here would be limited to the HPCI and RCIC systems.* Therefore, depressurization would not be required by the EOI for small breaks outside of the primary containment unless the leakage exceeds 0.355 m^3/s (5600 gpm), the combined capacity of the HPCI and RCIC systems.

Nevertheless, prompt depressurization for unisolated breaks well below 0.355 m^3/s (5600 gpm) would be a prudent step that the operational staff would be likely to take. Therefore, calculations were performed for two different sequences: one with only the slow depressurization beginning after 15 min and the other with an accelerated depressurization beginning 30 min after the onset of the break. Detailed results for these sequences are presented in Sects. 5.3 and 5.4. Section 5.5 considers a

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*The feed pumps are not available because the MSIVs are shut. The CRD hydraulic pump discharge is lost out of the break.

variety of initiating events (reactor trips) with a brief discussion of the possible effects of each on the SDV break sequence.

Safety-related equipment in the BFNP reactor building is required (by USNRC OIE Bulletin 79-01B) to be qualified to perform its designated function in the event of certain line breaks that can discharge high-energy primary coolant into the reactor building. These breaks are RCIC steam line break, HPCI steam line break, main steam line break, and RWCU system line break. The environmental extremes to which equipment must be qualified are 100% humidity and peak temperatures that depend on location within the building and are associated with a particular line break: a peak of 96°C (204°F) on the elevation-519 floor level because of an HPCI steam line break, a peak of 73°C (163°F) on the elevation-565 floor level because of a main steam line break, and a peak of 101°C (213°F) on the elevation-593 floor level because of an RWCU line break. Most of the equipment essential to recovery from the SDV break is not exposed to temperatures outside the range of these environmental extremes during the sequence with operator action; therefore, this analysis assumes that the safety-related equipment is environmentally qualified as required and that essential systems will function when called upon. Uncertainty is associated with this assumption because the SDV break causes conditions (temperature approaching 93°C and possible water/steam impingement) on the elevation-565 floor level that are more severe than the existing requirements for that location (73°C and no water/steam impingement). For example, it is possible that, even if the scram were reset, the scram pilot valves would not close because of moisture accumulation within the Reactor Protection System fuse cabinets, which are located near the hydraulic control units. If the circuits within these cabinets are short-circuited, the scram valves will not close because power will not be available to the scram solenoids as a result of blown fuses associated with the individual hydraulic control units. This would prevent the operators from using the preferred means of isolating the SDV break from the primary coolant system. There are, of course, other recovery options, including manual closure of the scram outlet isolation valves.

5.2 Summary and Conclusions

A liquid line break of the magnitude assumed for the SDV break [550 gpm (0.03 m^3/s) initial flow rate] would be detected by the control room operator within 30 min after inception of the break. The basis for detection would be annunciations in the control room of high reactor building area radiation and exhaust radiations. Because the BFNP EOI-27, "Reactor Building High Radiation or Ventilation System Failure," requires the control room operator to dispatch a health physics technician to determine the cause of the radiation alarm, diagnosis of the leakage condition would be confirmed by personal observation of the very harsh reactor building environment that exists after the break.

Reactor vessel level, pressure, and injection flow rate signals would probably not lead to detection of the SDV break condition because of the relatively small effective size of the SDV break.

Calculational results for the SDV break operator action sequences show that with fast depressurization, the leak would be stopped within

about 2.5 h after the inception of the break. For the case of slow depressurization, it could take as long as 4.5 h to confirm the location of the break and take corrective action.

The loss of primary coolant outside of primary containment to any significant degree is a serious matter that can threaten the viability of safety equipment located in the reactor building. Accordingly, reactor vessel depressurization is recommended to be required for any nonisolable break outside of the primary containment.

The premature loss of automatic injection capability, which occurs in the no-operator-action sequence (see Sect. 3) does not take place for the operator-action sequences because the operator prevents the uncontrolled injection of water into the reactor vessel by the CPs and CBPs.

5.3 SDV Break Operator-Action Sequence with Slow Depressurization

Results for the slow depressurization case are shown is Figs. 5.1–5.9. Leakage from the SDV break could be eliminated after about 4.5 h. By that time, the atmosphere of the elevation-565 floor level would have cooled sufficiently to allow entry and visual confirmation that the SDV was the leakage source. After visual confirmation, operations personnel would reset the reactor trip by extraordinary means — either by placement of rubber insulation between the applicable relay contacts in the auxiliary instrument room or by manually closing each of the 185 scram outlet isolation valves. If these actions were not taken by the operations staff, then a third recovery possibility would be to reduce reactor vessel pressure all the way to atmospheric pressure, essentially (but not totally) eliminating the leakage from the reactor vessel.

If the operations staff correctly diagnosed the source of the leakage without visual confirmation (i.e., by means of CRD instrumentation as discussed in Sect. 4.5), then recovery measures not requiring personnel access could be performed prior to 4.5 h. Even after the air temperature decreases to below 60°C (140°F), personnel access may be limited by radiation; therefore, the preferred means of recovery are those not requiring entry into the reactor building: (1) resetting the scram or (if this fails) (2) reducing the primary coolant pressure to near atmospheric.

Even though the plant is being cooled down at the normal (nonemergency) rate, the operators would know of the existence of primary coolant leakage to the reactor building within 0.5 h after the beginning of the break. Table 5.1 specifies the time at which various alarms or other leak detection signals would be received in the control room.

Even if the control room operators totally ignore the control room instruments, personnel in the reactor building at the time the break occurred would report the noise, steam, and/or high temperature of the leak (unless prevented from doing so by injury resulting from the leak). The BFNP reactor building, except for the basement and other limited areas, is maintained essentially free of contamination and can be entered without special contamination clothing or other equipment. At any time of the day (during power operation) three or more people are normally present in the

reactor building. Primary ingress and egress routes to the reactor building require passage by the HCUs. Health physics and calibration technicians enter frequently to complete assigned tasks. Custodial personnel inhabit the building as required to complete routine cleaning tasks. The unit auxiliary operator has a desk on the third floor of the reactor building in the RWCU area. Among other duties, he is responsible for recording periodic surveillance of equipment in the reactor and turbine buildings. Any of these personnel in the reactor building would hear, see, or feel the effects of a primary coolant leak of the magnitude considered in this report.

5.3.1 Reactor vessel pressure

Reactor vessel pressure is shown in Fig. 5.1. Before initiating depressurization (beginning at 15 min), the operator uses remote-manual operation of a single SRV, as necessary, to hold the reactor vessel pressure below the 7.72-MPa (1120-psia) automatic actuation setpoint of the SRVs. During depressurization, the operator lowers pressure at a rate such that the saturation temperature corresponding to vessel pressure decreases at the assumed target rate of 53°C/h (95°F/h). At first, the steam flow capacity of one SRV is sufficient to depressurize at the target rate, but when pressure reaches about 0.83 MPa (120 psia), a second SRV must be opened to meet the desired depressurization rate (Fig. 5.2). The same SRV is not used throughout; BFNP procedures require that SRV actuation be alternated among the 13 SRVs to minimize local suppression pool heating.

After about 3 h, the target pressure of 0.45 MPa (65 psia)* is met, and depressurization is halted while the source of the SDV leakage is investigated and eliminated.

5.3.2 Reactor vessel level

Figure 5.4 shows reactor vessel water level during the slow depressurization sequence. The operator maintains the level in the desired range by continuous operation of the RCIC system (Fig. 5.5), supplemented by several intermittent HPCI actuations. As reactor vessel pressure decreases, leakage from the CRDs decreases (Fig. 5.3), but the RCIC system runs at nearly full capacity (at least during the first 90 min) to make up for the additional steam production caused by depressurization.

5.3.3 Reactor building thermal-hydraulic environment

Reactor building response during the early part of the sequence is very similar to that for the no-operator-action sequence (Sect. 3). As the reactor vessel is depressurized, leakage from the vessel decreases. The flashing of steam from escaping coolant (Fig. 5.6) stops after ~3 h, when the temperature of the SDV leakage has decreased to below 100°C (212°F). As explained previously (see Sect. 3.4 and Summary), leakage from the reactor vessel mixes with the $0.111\text{-m}^3/\text{s}$ (170-gpm) flow of room

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*The RCIC turbine can be operated at pressures as low as 65 psia.

temperature water from the CRD hydraulic system before flowing to the SDV and out the break. The SDV leakage continues after 3 h, without flashing of steam, until the scram is reset or the 185 scram outlet isolation valves are closed.

Reactor building pressure (Fig. 5.7) peaks about 80 s after the SDV break, when the reactor building blowout panels open to provide a flow path to the refueling bay. The refueling bay blowout panels do not open because the refueling bay has a higher exfiltration rate for a given positive pressure.

After the initial pressure surge, the SGT system blowers are able to maintain the reactor building under a negative pressure. As the rate of steam flowing from the SDV break is reduced by depressurization, the atmosphere temperatures within the reactor building (Fig. 5.8) begin cooling as the heated air/steam mixture within the building mixes with the cooler outdoor air drawn in by the SGT system blowers.

The SGT system is able to handle the flow of warm [maximum exhaust temperature of 43°C (110°F)], moisture-saturated air because of design features that prevent condensation and moisture accumulation. The features installed in each SGT train include moisture separators, 40-kW air heaters, and heating elements within the charcoal filters that are thermostatically controlled to maintain the ambient temperature of the charcoal at 87°C (125°F) before system operation. The charcoal heaters are deenergized upon SGT initiation.

The atmosphere at the elevation-565 floor level cools very slowly because of the large amount of heat retained within the structures (including the massive concrete walls). After 4.5 h, the elevation-565 air temperature is below 60°C (140°F) — cool enough to allow operations personnel to enter and visually confirm the SDV as the source of the leakage.

Figure 5.9 shows the flood level in the basement. As discussed in Sect. 3.4.3, the basement floor drains are interconnected so that the water level would be approximately uniform throughout the basement floor. The flood level is shown in Fig. 5.9 both with and without the sump pumps operating. In both cases, the water level does not reach the 1-m (3.3-ft) depth that would be necessary to threaten the vital ECCS vessel water injection pumps in the basement. The sump pumps would be expected to ultimately fail by submergence during this accident sequence. Without the sump pumps operating, the rising water level causes control room annunciation of basement flooding after about 40 min. This is about 20 min sooner than for the case with the pumps operating.

5.3.4 Reactor building radiation exposure rate

The tech spec limit for equilibrium primary coolant activity concentration is 0.2 μ Ci/g. The normal BFNP coolant activity is 0.032 μ Ci/g (Ref. 1). Some of the activity carried out the leaking SDV would be released into the elevation-565 atmosphere and would be sensed by the Area Radiation Monitor (ARM) detectors on that floor level. As more activity is released, the radiation dose rate (i.e., whole body dose caused by gamma radiation) increases; when it exceeds 100 mrem/h, the "Reactor Bldg Radiation Hi" annunciator in the main control room would be actuated.

The time required for actuation of the high-radiation alarm is estimated from calculational results reported in Ref. 2 for hypothetical SDV

break scenarios similar to that considered in this report. Reference 2 reports (as Case 6 in Table 4.2) a dose rate of 0.7 rem/h after 0.5 h of SDV leakage at a rate of 0.03 m^3/s (550 gpm); the calculation assumes that the coolant activity is 0.2 μ Ci/g and that the release mixes with 20% of the building free volume. This approximates the dose rate that the ARMs on the elevation 565 would see. At this rate, the 100-mrem/h high radiation alarm would be exceeded after only 4.5 min. If coolant activity were at the normal BFNP equilibrium concentration, the dose rate would exceed 100 mrem/h after 27 min.

5.4 Operator-Action Sequence with Accelerated Depressurization

Figures 5.10–5.18 show the operator-action sequence with an accelerated depressurization beginning 0.5 h after inception of the SDV break. As discussed in Sect. 5.3, the operators would know of the existence of the break within 0.5 h. Depressurization at a faster than normal rate is an obvious way to reduce leakage from an unisolatable leak.

After ~2.5 h, reactor building atmosphere temperature would have cooled enough to allow entry to the elevation-565 floor level and positive confirmation of the SDV as the source of the leakage. After confirmation of the leakage source, the operators would take positive steps to eliminate the leakage flow, as discussed in Sect. 5.3 for the case of slow depressurization. The advantage of the fast depressurization is that recovery can be assured ~2 h sooner.

5.5 SDV Break Sequence with Other Initiating Events

Because the SDV is pressurized only after reactor trip, a reactor scram must precede any SDV break accident. Table E.1 (Appendix E) lists BFNP reactor trips from full power. A reactor scram caused by a high main steam line radiation scram signal could not be reset (as has been assumed in this study) because (1) it has no bypass and (2) it could remain in effect for hours after the initial detected high radiation.

With the exception of the high drywell pressure trip, a large SDV break without operator action would proceed essentially as presented in Sect. 3 for any reactor trip initiating event. The initial part of most SDV break no-operator-action sequences would be different in that most reactor trips do not involve a related closure of the MSIVs. Immediately after reactor trip, the reactor vessel level usually shrinks. The main feedwater automatic control sees the decrease in level and, in response, increases the speed of the three operating main feedwater pumps. The vessel level responds rapidly, and before the controller can reduce flow sufficiently, the reactor vessel level exceeds the setpoint for automatic trip of all three feed pumps.* After all the feed pumps are tripped, the

*Normal operator action would be to trip two of the three operating feed pumps to avoid trip of all feed pumps on high level.

vessel level begins decreasing. When the level reaches 12.11 m (476.5 in.), the MSIVs close, and when the level reaches 12.09 m (476 in.), the HPCI actuates. From this point on, the sequence with no operator action is very similar to that presented in Sect. 3, with the MSIVs closed and reactor vessel level being maintained by the HPCI system.

An SDV break no-operator-action sequence following reactor trip on high drywell pressure would take much longer for the core to be uncovered. The high drywell pressure signal "seals in" and thereby enables the Automatic Depressurization System (ADS) to actuate whenever low reactor vessel water level is reached. This enables the ECCS pumps (residual heat removal and core spray) to inject water into the vessel, preventing core uncovery until the pumps themselves fail because of submergence after about 16 h.

The SDV break operator-action sequence following most of the reactor trips listed in Table E.1 would either be terminated very quickly when the operator successfully resets the reactor trip, or, as a worst case, would proceed approximately as outlined previously in this section. In either case, the core uncovery is avoided by reasonable operator actions. The major difference expected would be that the operators might, for all but the loss of reactor protection system (RPS) power trip, be able to keep the MSIVs open* and provide reactor vessel injection using the main feedwater system instead of the RCIC and/or HPCI systems. This does not affect the timing or ability of the operators to detect the SDV leakage condition because (as discussed in Sect. 5.3) detection of the leak depends mainly on reactor building environmental response factors.

References

1. Letter from L. M. Mills, Tennessee Valley Authority, to H. R. Denton, U.S. Nuclear Regulatory Commission; Subject: TVA response to generic letter 81-34, January 1982, Docket Nos. 50-259, 50-260, and 50-296.

2. U.S. Nuclear Regulatory Commission, *Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping*, NUREG-0803 (undated).

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*MSIVs will close if high temperature in the main steam line space initiates a Group I isolation signal [at 93°C (200°F)].

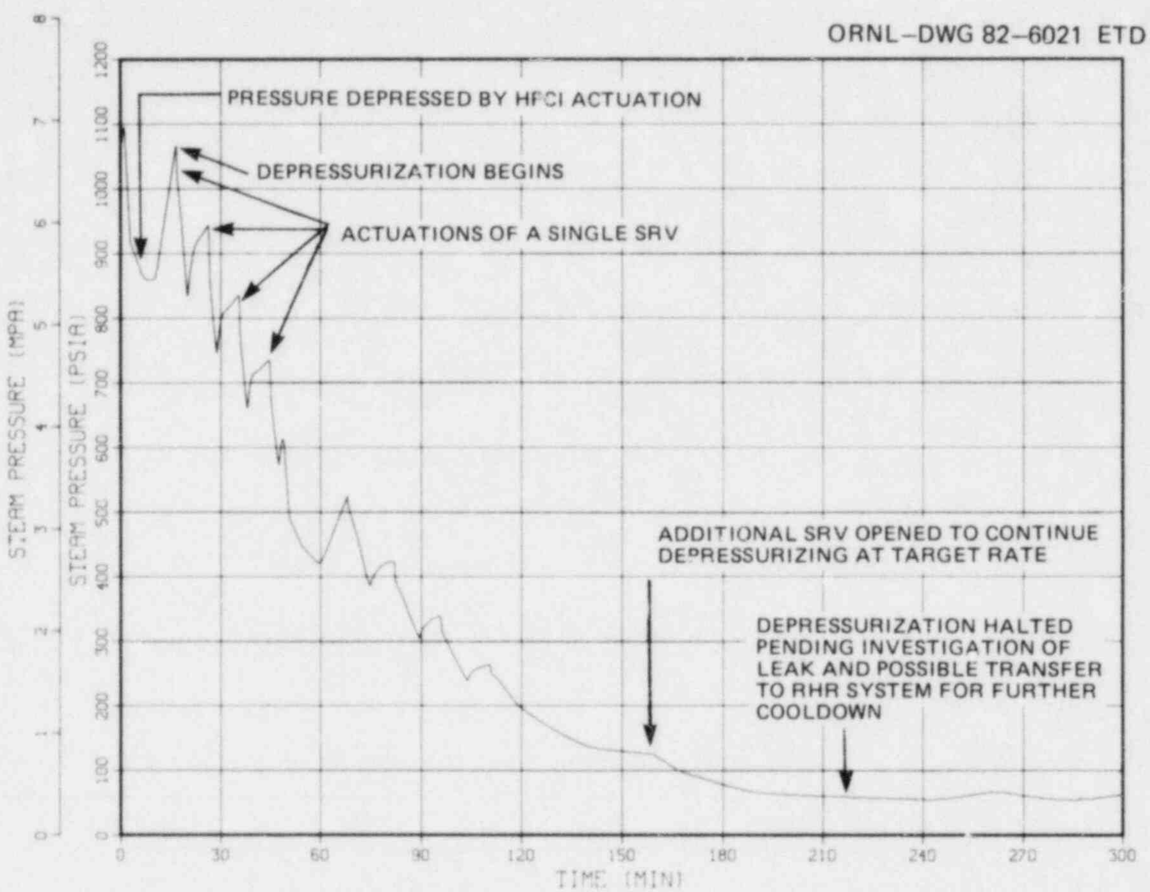

Fig. 5.1. SDV break operator-action sequence with slow depressurization — reactor vessel pressure.

ORNL-DWG 82-6022 ETD

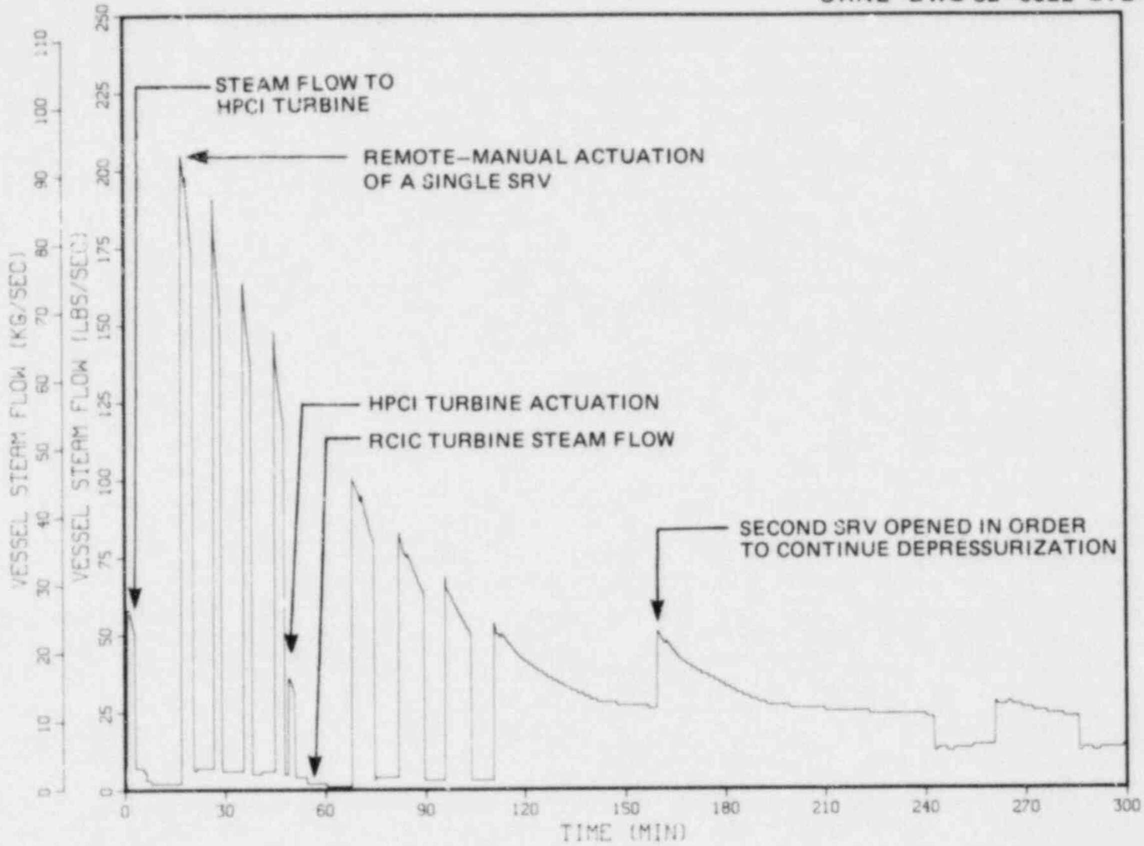

Fig. 5.2. SDV break operator-action sequence with slow depressurization — steam flow from reactor vessel.

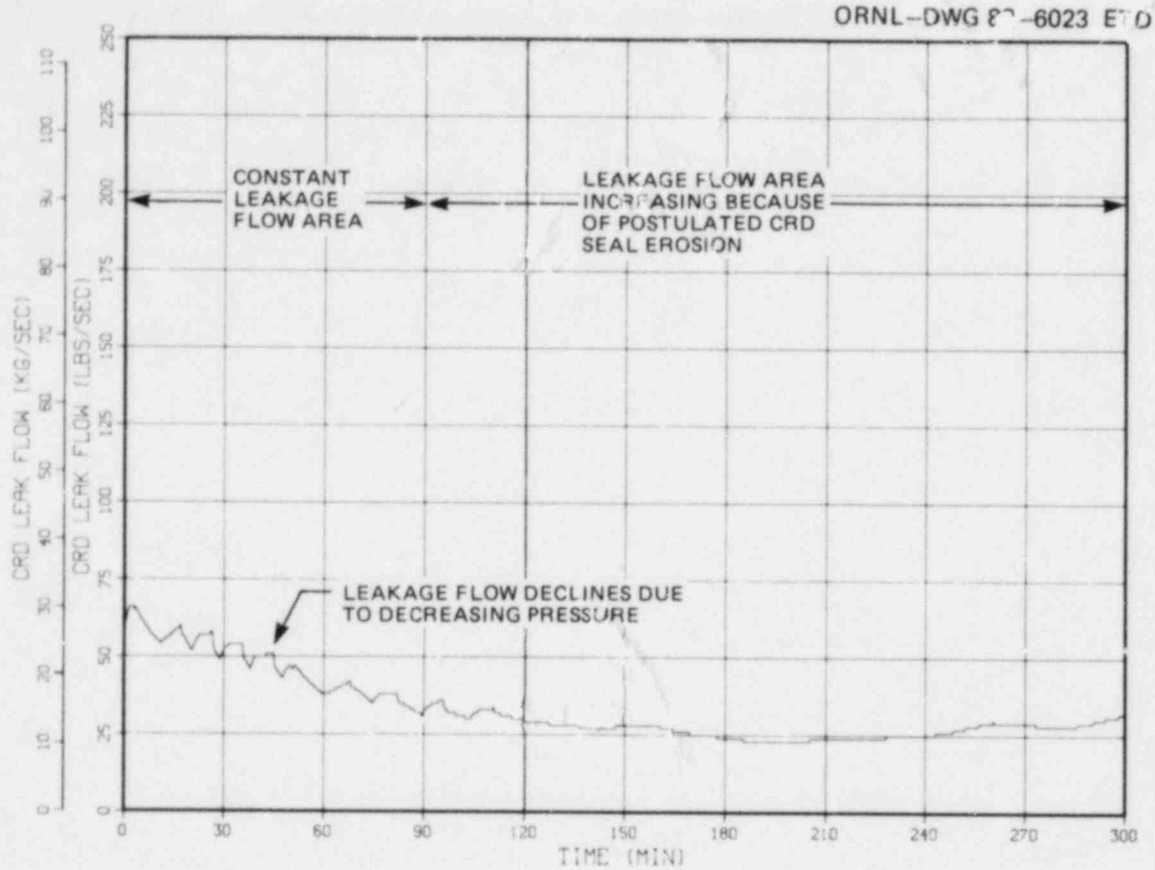

Fig. 5.3. SDV break operator-action sequence with slow depressurization — leakage flow from reactor vessel.

ORNL-DWG 82-6024 ETD

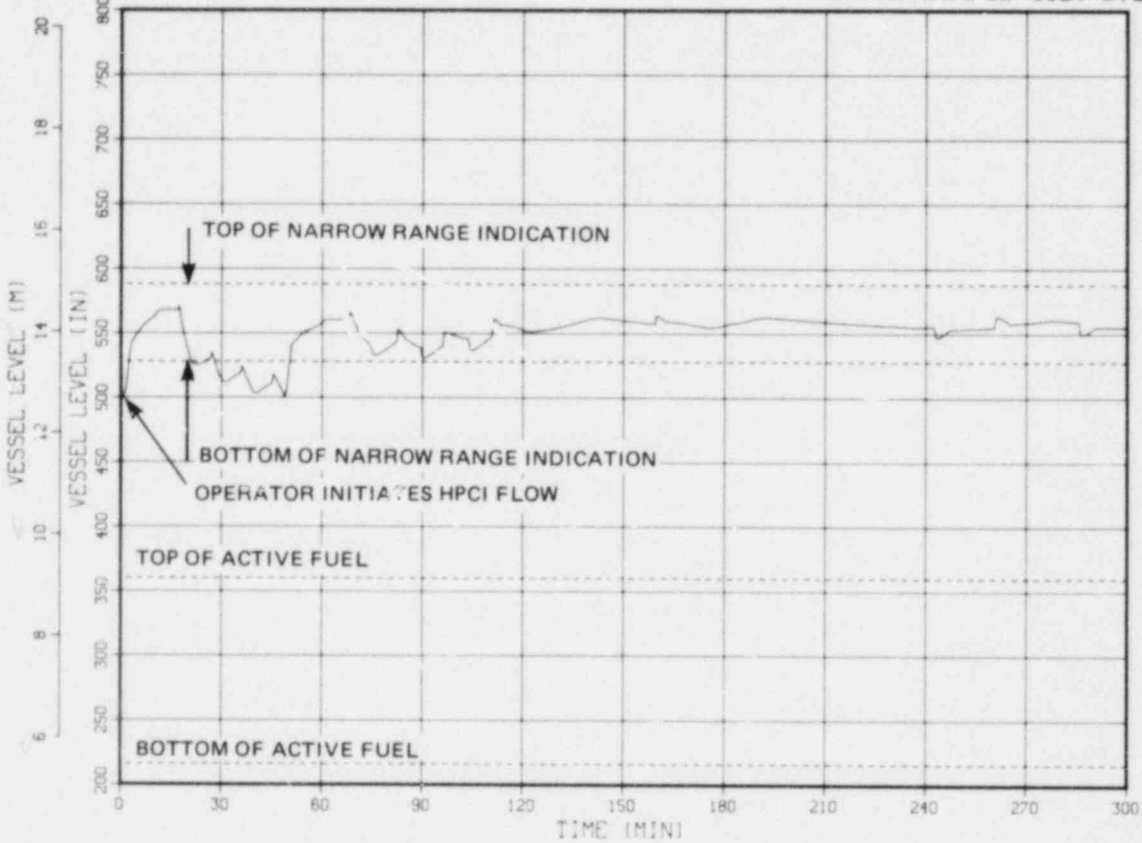

Fig. 5.4. SDV break operator-action sequence with slow depressurization — reactor vessel water level.

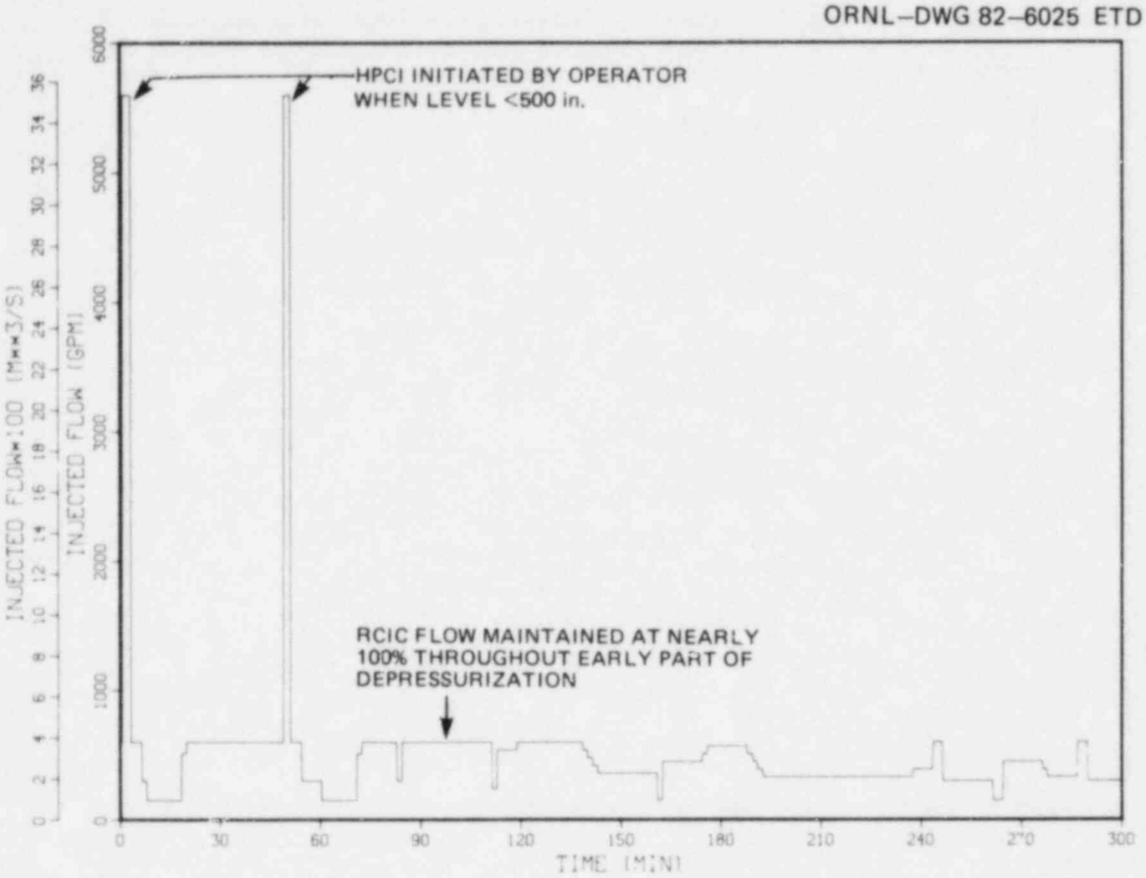

Fig. 5.5. SDV break operator-action sequence with slow depressurization — total reactor vessel injected flow.

ORNL-DWG 82-6026 ETD

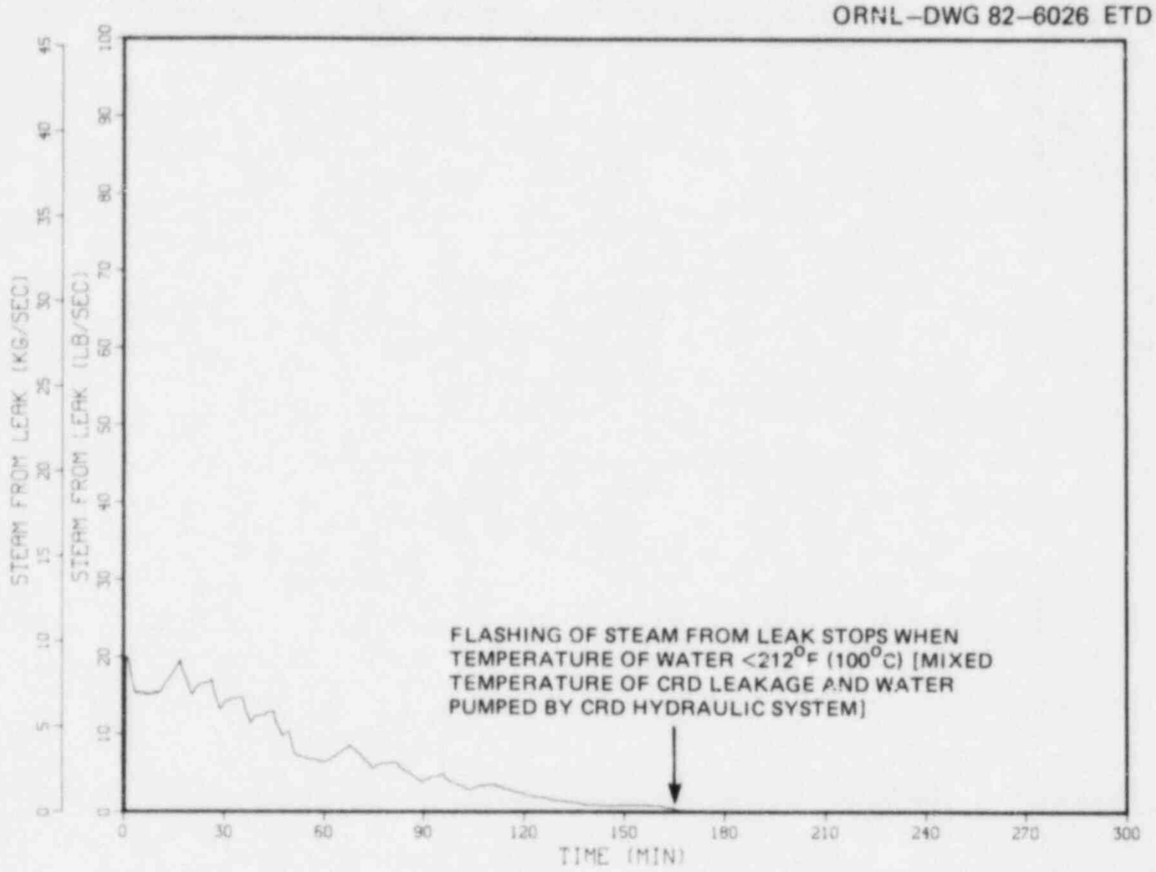

Fig. 5.6. SDV break operator-action sequence with slow depressurization — rate of steam flashing into reactor building.

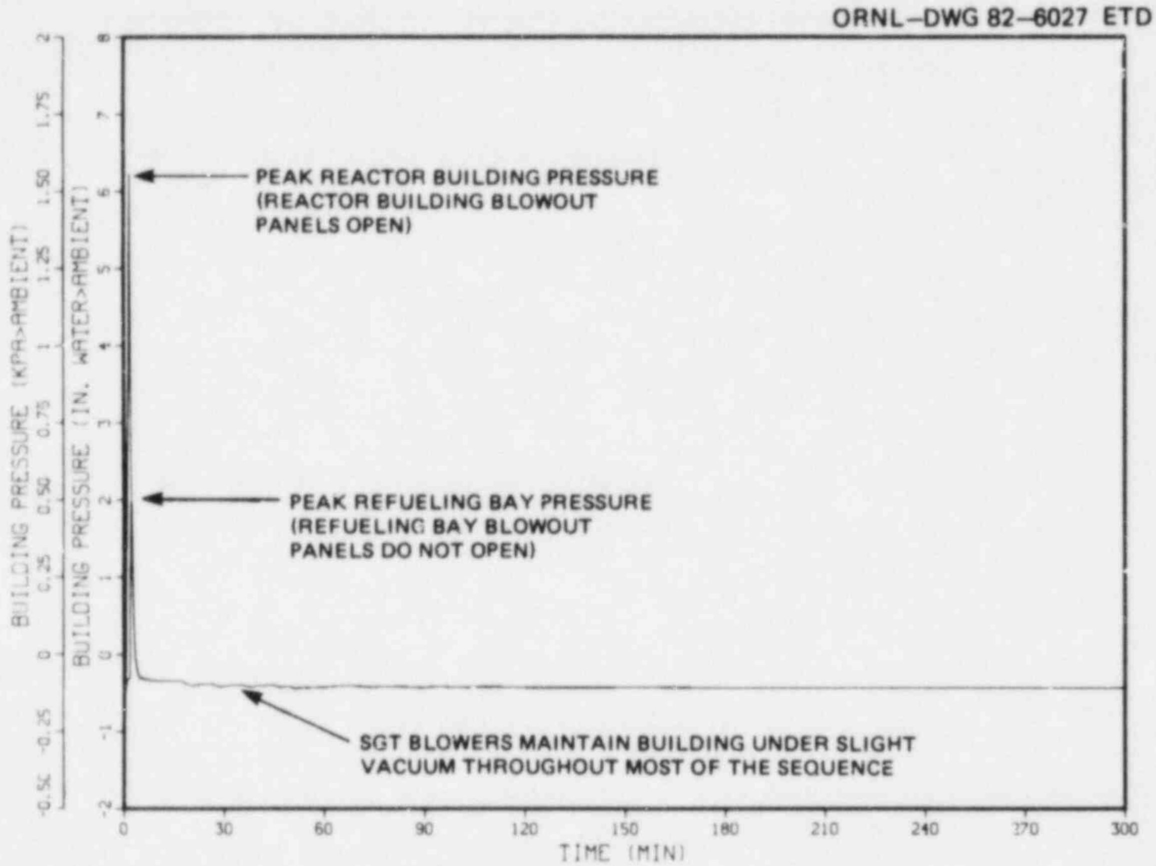

Fig. 5.7. SDV break operator-action sequence with slow depressurization — reactor building pressure.

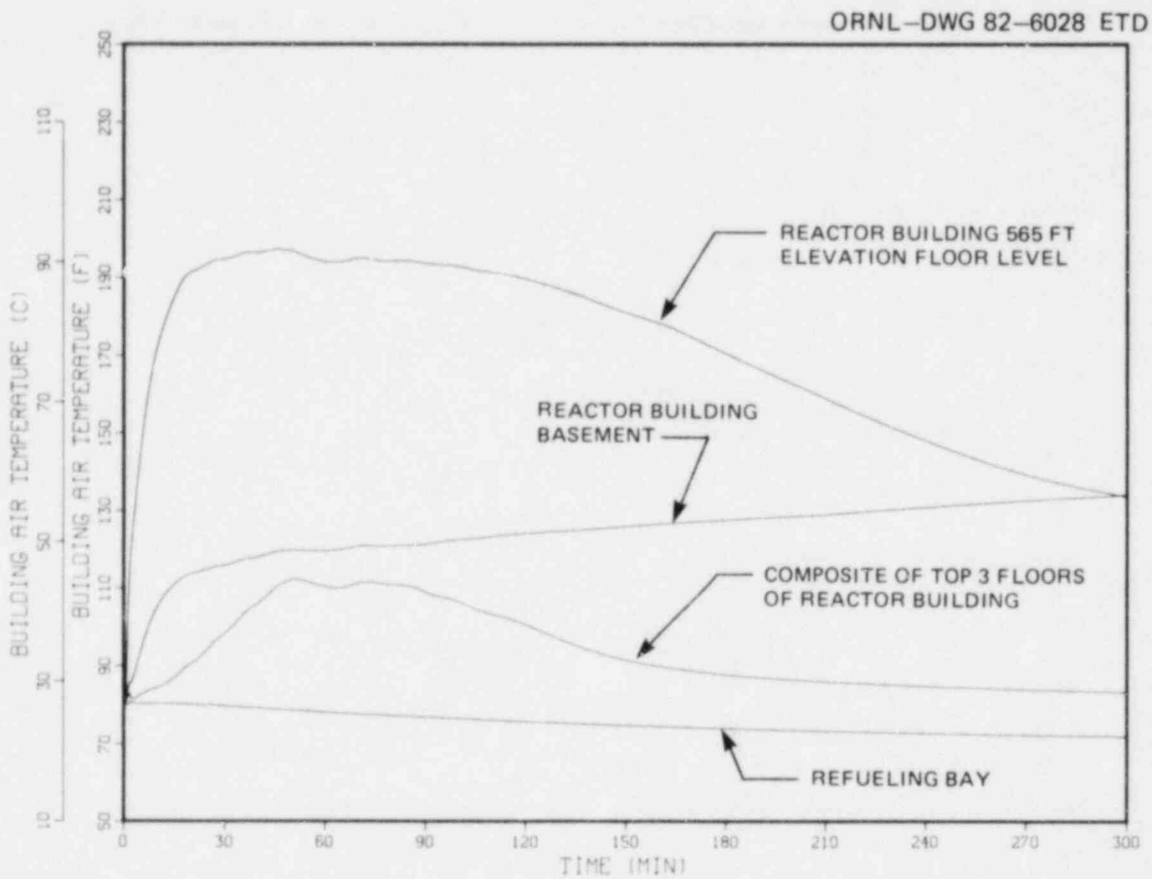

Fig. 5.8. SDV break operator-action sequence with slow depressurization — building atmosphere temperatures.

ORNL-DWG 82-15733

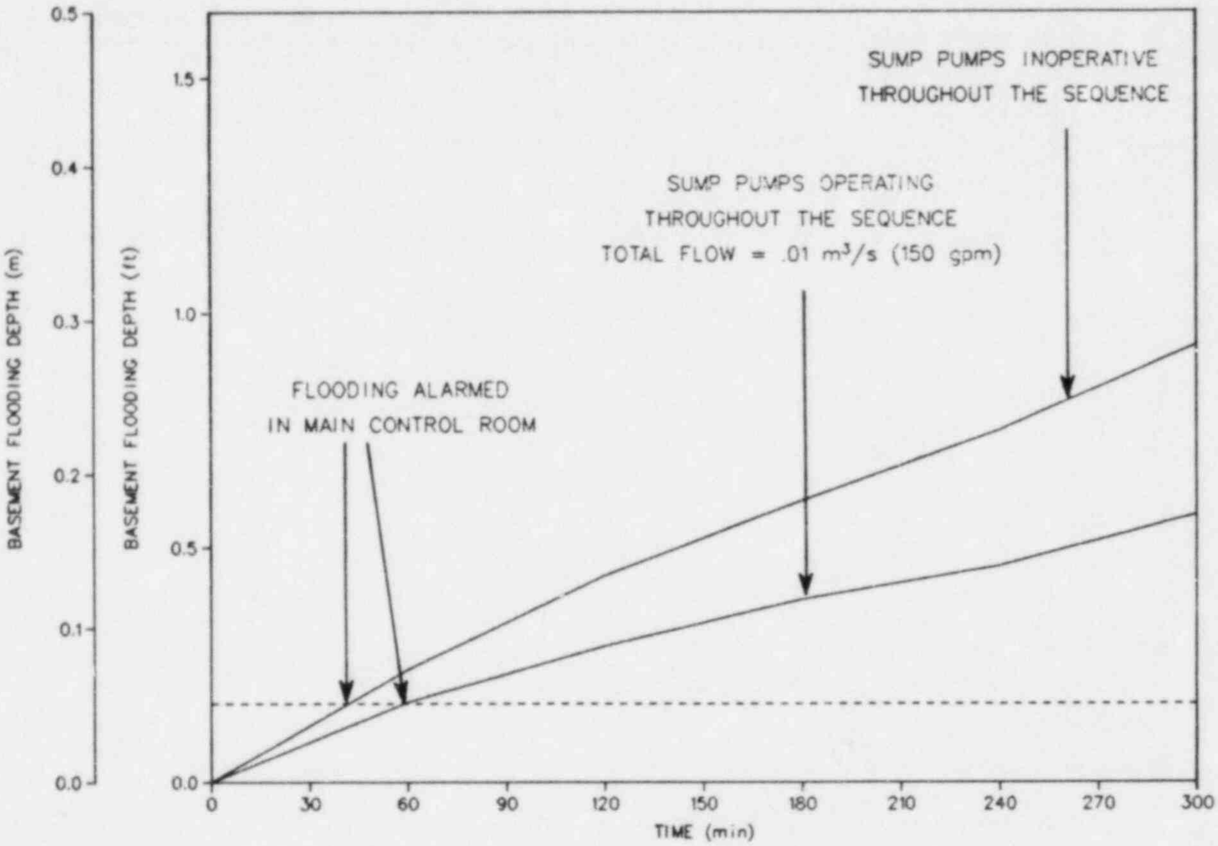

Fig. 5.9. SDV break operator-action sequence with slow depressurization — reactor building basement flooding depth.

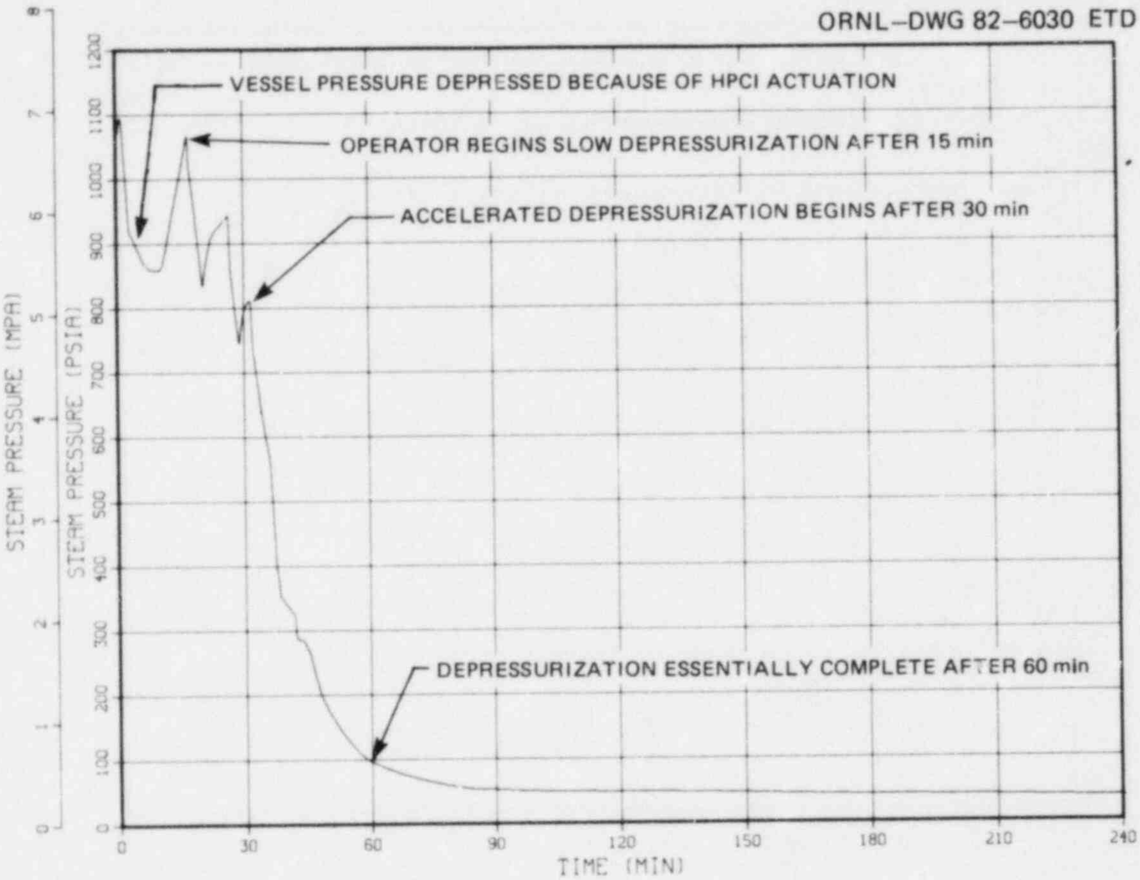

Fig. 5.10. SDV break operator-action sequence with fast depressurization – reactor vessel pressure.

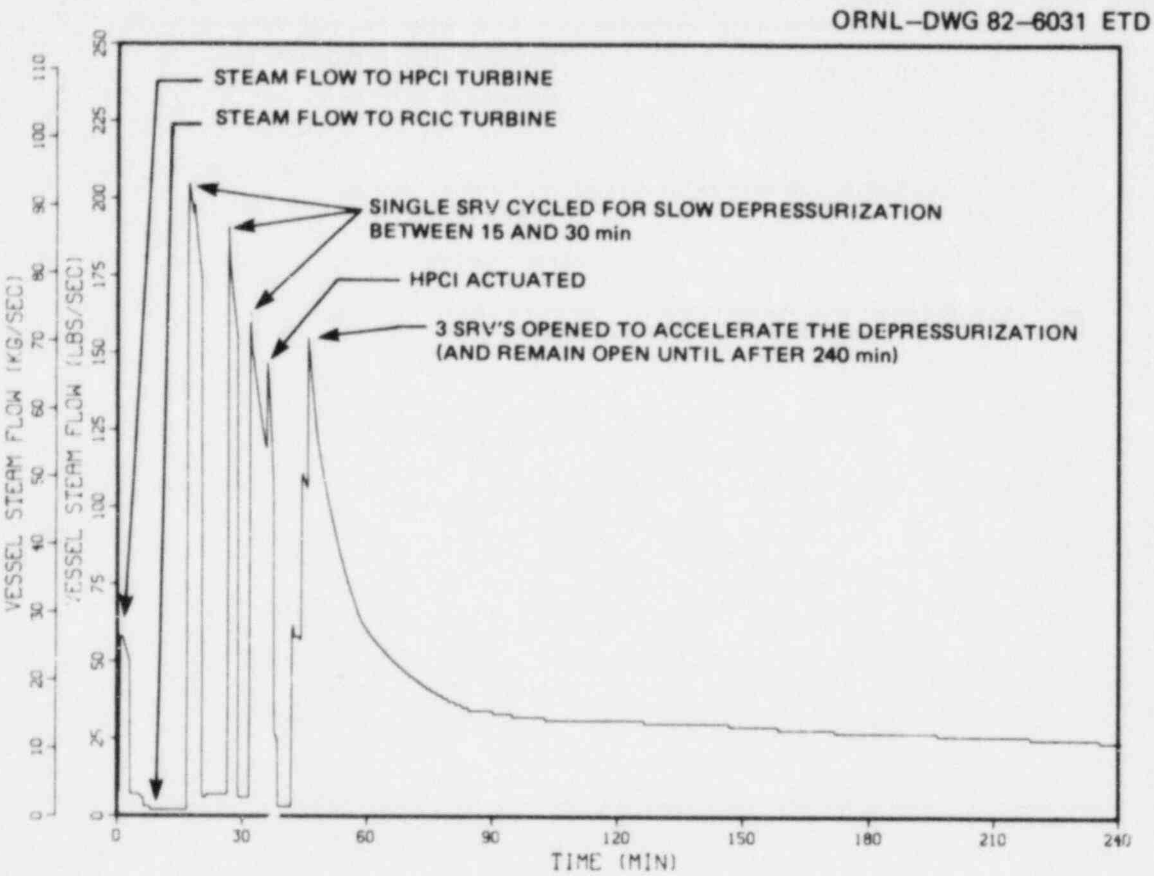

Fig. 5.11. SDV break operator–action sequence with fast depressurization — reactor vessel steam flow.

ORNL–DWG 82–6032 ETD

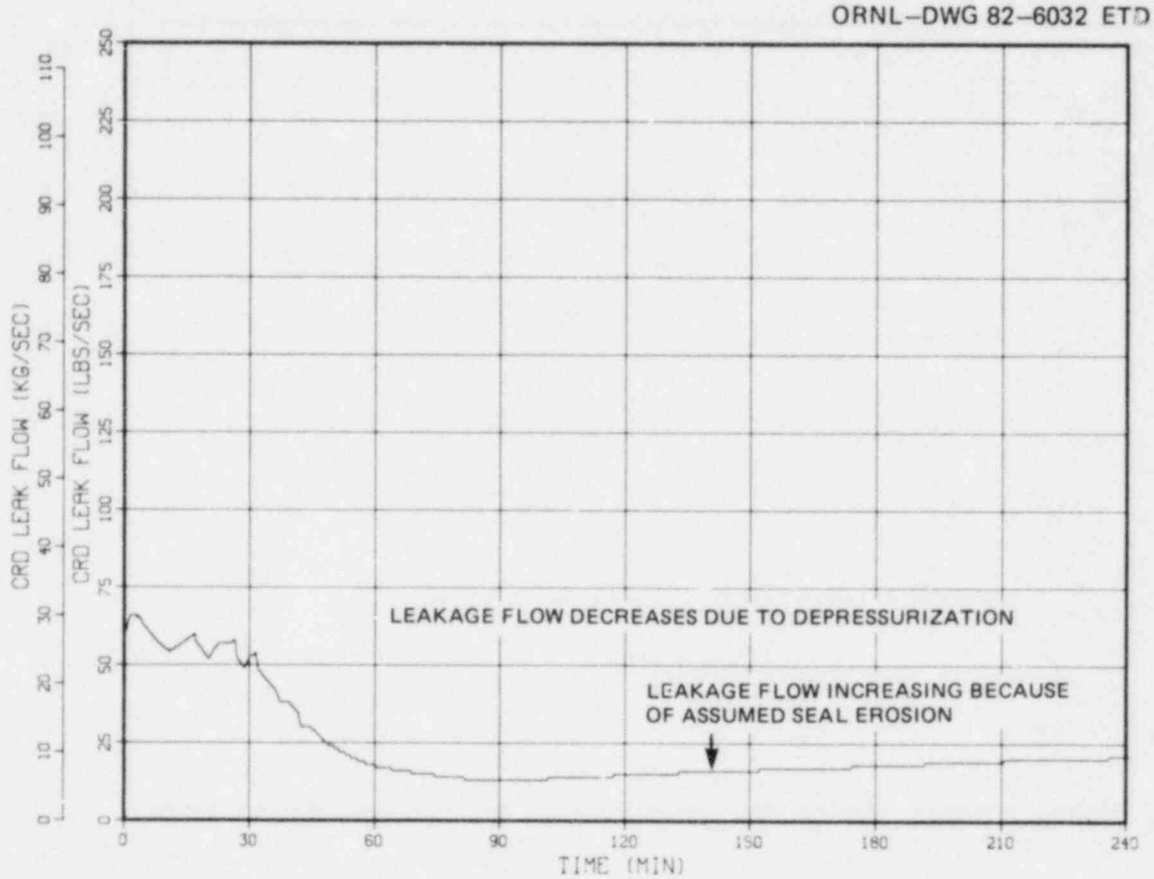

Fig. 5.12. SDV break operator-action sequence with fast depressurization — CRD leakage flow.

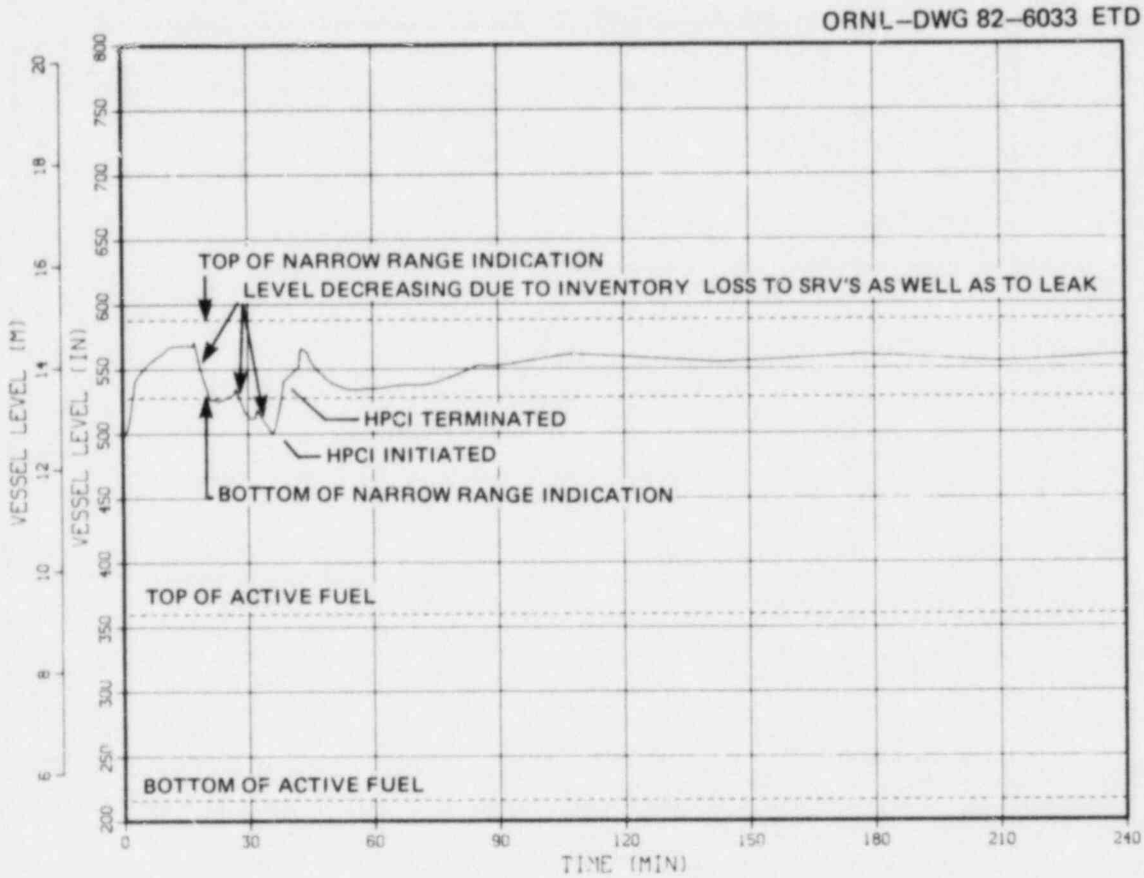

Fig. 5.13. SDV break operator-action sequence with fast depressurization — reactor vessel water level.

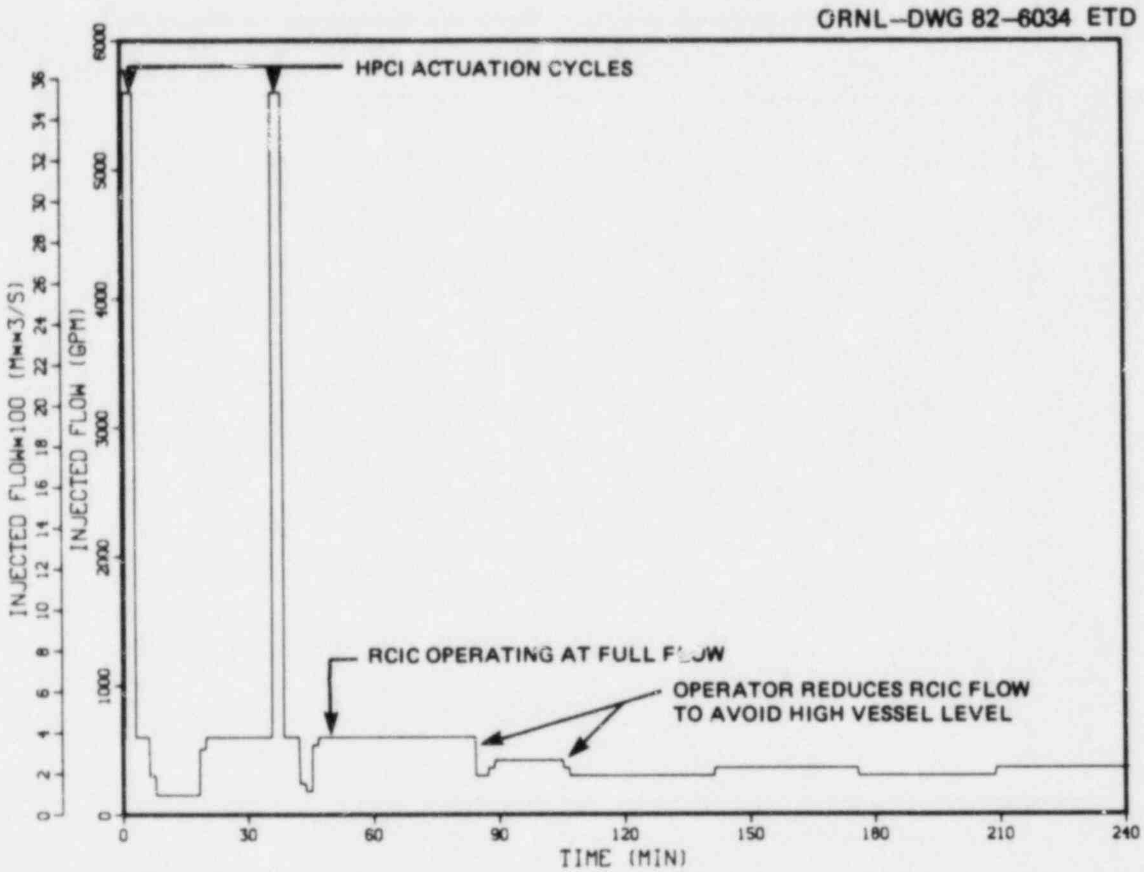

Fig. 5.14. SDV break operator-action sequence with fast depressurization — reactor vessel injected flow.

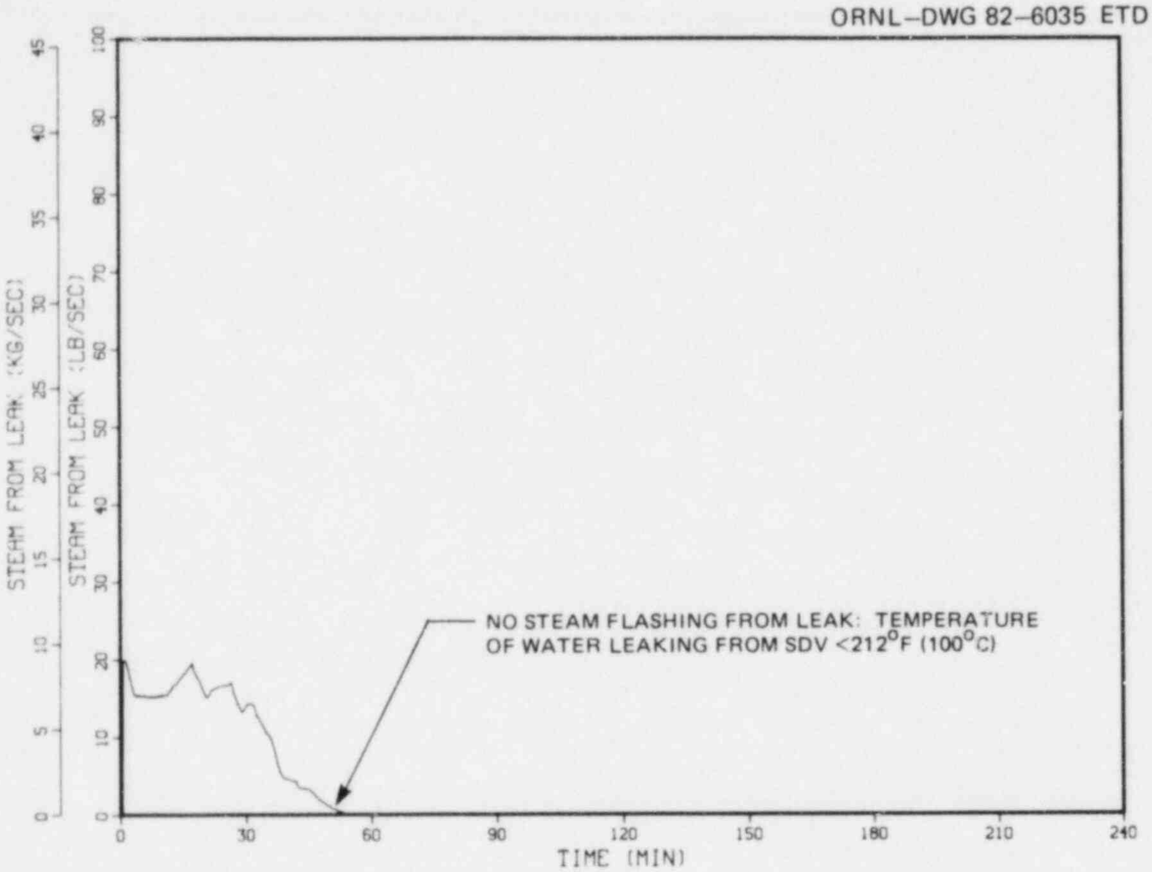

Fig. 5.15. SDV break operator-action sequence with fast depressurization — steam flow from leak.

ORNL-DWG 82-6036 ETD

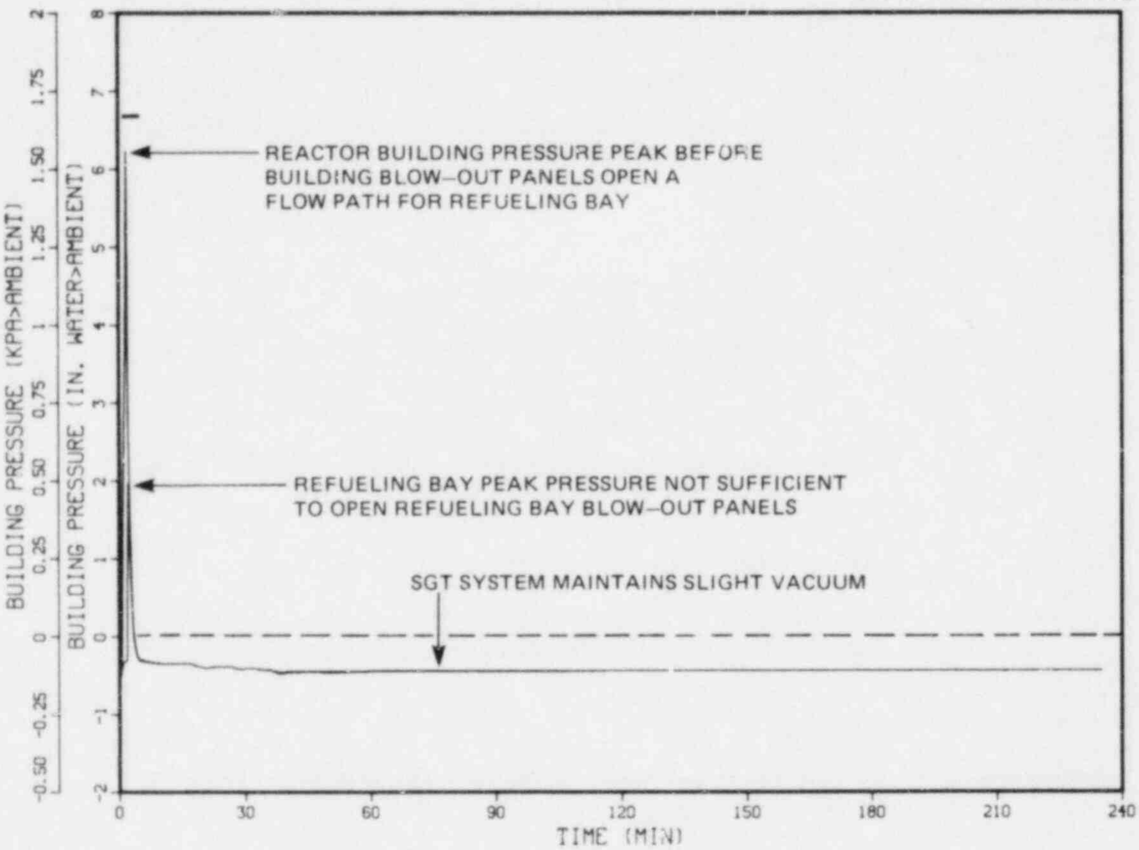

Fig. 5.16. SDV break operator-action sequence with fast depressurization — reactor building pressure.

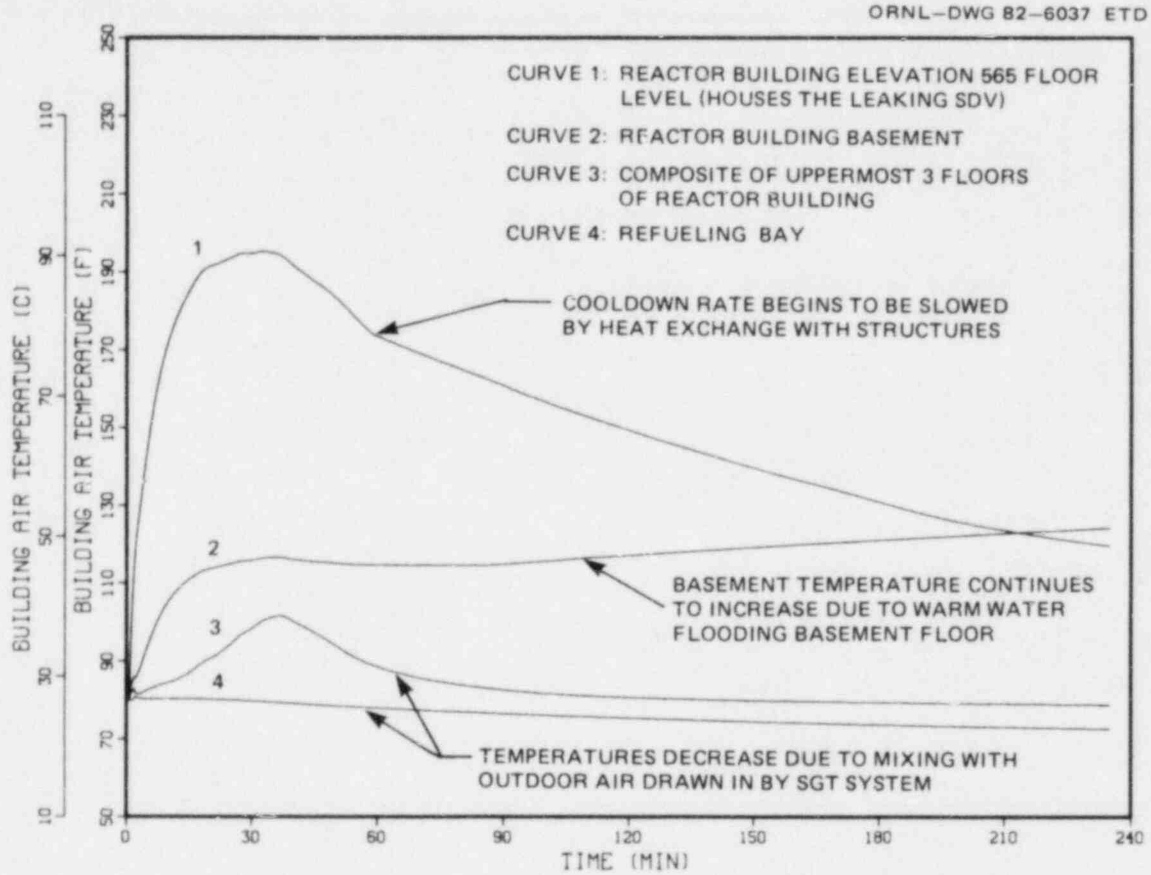

Fig. 5.17. SDV break operator-action sequence with fast depressurization — reactor building atmosphere temperature.

ORNL-DWG 82-15732

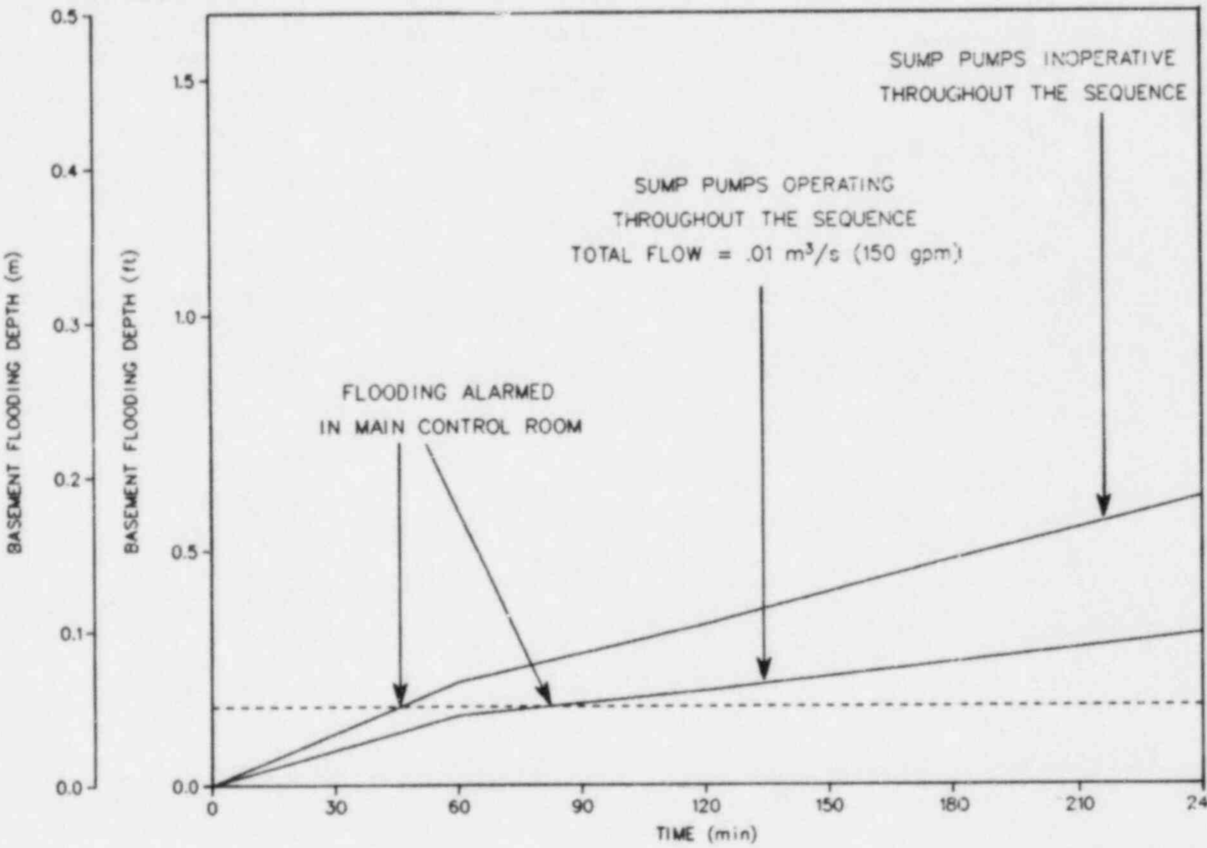

Fig. 5.18. SDV break operator-action sequence with fast depressurization — reactor building basement flooding depth.

Table 5.1. Break detection signals for SDV operator-action sequences

Break detection signal	Time[a] signal received in main control room (min)	Information received in main control room
Reactor zone ventilation exhaust high-radiation alarm (radiation >11 mrem/h)	~1	The following annunciator would actuate: "REACTOR ZONE EXHAUST RADIATION HIGH"
Building high-radiation alarm (radiation >100 mrem/h above background)	27[b]	The following annunciator would actuate: "REACTOR BLDG RADIATION HIGH"
Independent personal observation (of noise, steam, high temperature, flooding)	<30	Personnel in reactor building would report the obvious leakage conditions
Basement flooding alarms [flood level >2 in. (0.051 m)]	60	One or more of the following annunciators would actuate: "RHR PUMP RM FLOOD LEVEL" "CORE SPRAY PUMP ROOM FLOOD LEVEL" "ICS and RCIC PUMP ROOM FLOOD LEVEL"
Basement high-temperature alarms [temperature >130°F (54°C)]	224	One or more of the following annunciators would actuate: "RHR PUMP RM HUMIDITY OR TEMP HIGH" "CORE SPRAY PUMP RM HUMIDITY OR TEMP HIGH" "RCIC TURB ROOM TEMP HIGH HIGH"

[a]Elapsed time after inception of the SDV break.

[b]High-radiation alarm is calculated (Sect. 5.3.4) to occur after 27 min if primary coolant equilibrium activity is normal.

6. EVENTS AFTER CORE UNCOVERY

6.1 Introduction

Events prior to core uncovery in the no-operator-action sequence were discussed in Sect. 3. The purpose of this section is to discuss the sequence and timing of events that occur subsequent to core uncovery for the no-operator-action case.

The analysis presented in this section was performed with an ORNL-modified IBM version of the MARCH 1.1 (Ref. 1) code. The MARCH code was originally developed by Battelle Columbus Labs for risk assessment applications such as the Reactor Safety Study.[2] For such applications, the accident sequences are fixed by initial definitions. While relatively simple bounding analyses might be adequate for risk assessment applications, more sophisticated calculational capabilities clearly are desirable for the subject application. Appendix B discusses problems associated with application of MARCH to BWR accident analysis. Appendix C contains a brief summary of those modifications that were necessary to permit ORNL to complete the analysis discussed in this section. The MARCH input data for the base case analysis that is subsequently used for the fission product transport analysis is contained in Appendix F. The input values shown in Appendix F were derived from plant design information and the results of the BWR-LACP code analysis described in Sect. 3.

6.2 Major Design Considerations Influencing Post-Core-Uncovery Accident Sequences

The accident sequence subsequent to core uncovery in the no-operator-action case is significantly influenced by three distinct design features of the BWR 4/MK I reactor/containment system employed at BFNP. The first two design features are related to the physical flow path followed by material leaking from the reactor vessel and drywell via the break in the scram discharge volume system (Fig. 6.1). During the early stages of the accident, material leaving the reactor vessel must flow through the fuel bundle flow openings [located at the top of each control rod guide tube 0.3 m (~12 in.) below core bottom*], down the length of the guide tubes and CRD mechanism housings, and out to the SDV system via the 3/4-in. CRD withdraw lines, which are located at the base of each CRD housing. Because of the location of the fuel bundle flow openings, direct leakage of reactor vessel water is apparently possible only while the vessel water level is above the height of these flow openings (for intact CRD housings and guide tubes). After the water level has dropped below these openings, the break flow changes from water to steam.

The location of the 185 CRD withdraw lines that connect the control rod drive mechanisms to the scram discharge system is also shown in Fig. 6.1. Because of the physical proximity of these scram withdraw lines to

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*A detailed drawing of one of these openings is shown in Appendix B, Fig. B.7, labeled "flow inlet to fuel bundle."

the reactor vessel bottom head, vessel head failure probably will result in the failure (opening) of a large number of these lines. This is a significant fact because of the effective break flow area available for containment leakage with and without withdraw line failures. Following vessel head failure, the effective break flow area is limited only by the combined flow area of the 185 scram withdraw lines, which is 0.053 m^2 (~0.57 ft^2). The number of scram withdraw lines that would actually fail following reactor vessel head failure is difficult to estimate because of uncertainties in the mode and timing of vessel head failure. It is possible to imagine repair mechanisms by which withdraw lines that had previously failed could effectively reclose themselves via some melt/refreeze phenomena or plugging of the lines because of aerosol particle entrapment.

The third major element that influences the sequence of events in the subject accident is the design of the containment EPAs (Fig. 6.2). The design pressure and temperature of the BFNP containment are 0.386 MPa (gage) (56 psig) and 138°C (281°F) respectively. Containment failure via overpressurization was considered to be the dominant failure mode for an inerted BWR containment in the Reactor Safety Study.[3] Failure was predicted to occur when the containment was stressed to a level between the yield and ultimate tensile stress of the steel liner and inner reinforcement layers. The lower bound for containment failure was assumed to be 1.21 MPa (177 psia).

The containment EPAs are designed to withstand an interior containment temperature of 163°C (325°F) for 900 s (15 min), followed by an indefinite period at 138°C (281°F) (Ref. 3). The design pressure of the EPAs is 0.386 MPa (gage) (56 psig). Based on an analysis of the EPA design criteria, drywell failure would probably occur via EPA elastomer seal degradation at ambient temperatures in excess of 204°C (400°F) (Ref. 4). The combined cross-sectional flow area of all the drywell EPAs in BFNP is 1.95 m^2 (21 ft^2). Accordingly, we have assumed that the EPA seals would begin leaking at 204°C (400°F) with a combined leakage area of 64.5 cm^2 (10 $in.^2$) and fail completely at an ambient temperature of 260°C (500°F).

6.3 ECCS Involvement in Post-Core-Uncovery No-Operator-Action Accident Sequence

As will be discussed in the following sections, the accident progression following core uncovery in this sequence was found to be highly sensitive to the performance characteristics of the low pressure emergency core coolant systems [core spray system and low pressure coolant injection (LPCI) mode of the RHR system]. The high pressure coolant injection system (HPCI) and reactor core isolation cooling system (RCIC)* would not be available for use in the period following core uncovery in this sequence. As described in Sect. 3, the HPCI system would be rendered inoperable because of water overflow from the reactor vessel to the HPCI steam supply lines following condensate booster pump injection at 232 min into the

*Although the RCIC system is not an ECCS, it does have the capability to inject water into the reactor vessel under high vessel pressure conditions.

transient. Without operator action, the RCIC system would automatically begin running when the reactor vessel level first drops to 12.1 m (476.5 in.) at the start of the accident and automatically turn off when the level reaches 12.24 m (482 in.). The RCIC system would remain off thereafter.

Table 6.1 is a summary of the design and as-built performance characteristics of the BFNP#1 performance characteristics. As-built specifications were obtained from pump test data supplied to ORNL by TVA. The core spray (CS) system consists of four electric-motor-driven centrifugal pumps, a spray sparger in the reactor vessel above the core, and the necessary piping to convey water from the suppression pool to the sparger. The core spray pumps automatically begin running upon receipt of a signal indicating either (1) low reactor vessel water level of 9.76 m (384.5 in.), or (2) high drywell pressure of 0.0138 MPa (gage) (2 psig) and low reactor vessel pressure of 3.103 MPa (gage) (450 psig). In the subject sequence, the low water level signal occurs at 438 min into the accident (Sect. 3). However, CS water injection does not begin until the pressure differential between the reactor vessel and the wetwell drops below the effective shutoff head of the core spray pumps (design 289 psid, as-built 342 psid). As used here, the effective shutoff head is defined to be the pump shutoff head minus the elevation head between the suppression pool surface and the injection point into the primary system.

The BFNP#1 LPCI (operational mode of RHR system) consists of four electric-motor-driven centrifugal pumps and the associated piping and valves necessary to convey water from the pressure suppression pool to the reactor recirculation loops. LPCI pump operation is triggered by receipt of the same signals previously described for the CS system. As in the case of the CS, LPCI injection does not begin until the differential pressure between the reactor vessel and wetwell drops below the effective shutoff head of the LPCI pumps.

An analysis of the design and as-built system characteristics in Table 6.1 reveals two major differences:

1. Under design conditions, LPCI system injection should occur before CS system injection as reactor vessel pressure decreases, while under as-built conditions, CS system injection occurs before LPCI injection.
2. The as-built effective pump shutoff head of both the LPCI and CS systems is significantly higher than the associated design shutoff pressures; that is, the actual CS and LPCI pumps will begin to inject water at significantly higher reactor vessel pressures than required by design specifications.

Item 1 is particularly important because an accident condition might be achieved in which steam flashing caused by CS injection can result in a reduced reactor vessel depressurization rate, thereby preventing LPCI injection. The following section will show that differences between the design and as-built LPECCS performance can significantly influence the accident progression rate in the no-operator-action sequence.

6.4 Sequence Definition

Figure 6.3 is a representation of the event-sequence tree following core uncovery in the no-operator-action scram discharge volume break accident. The event timings associated with the sequences shown in Fig. 6.3 are given in Table 6.2. The first branch point in the tree is based on the availability of LPECCSs. In sequences B and B1, it is assumed that no LPECCS is available, while in sequences A and A1, the LPECCSs are assumed to be functional and available. For sequences in which LPECCS is available, we have differentiated between cases in which these systems operate as designed and cases in which these systems operate as built (Table 6.1). Branch point 3 is encountered in all sequences following failure of the reactor vessel bottom head. In sequences A, A1, and B1, all CRD withdraw lines are assumed open upon failure of the bottom head, providing a direct leakage path from the drywell* to the reactor building. In sequence B, these lines are assumed to be closed so that leakage into the reactor building ceases at the time of reactor vessel bottom head failure.

An analysis of the data in Table 6.2 reveals that MARCH predicts sequence B containment failure will occur 34 min later than in sequence B1. This result is surprising because in sequence B, the drywell/wetwell containment is completely isolated, while in sequence B1 the drywell has a functional vent area of 0.053 m^2 (0.57 ft^2) because of failure of the CRD withdraw lines. It is reasonable to presume that drywell venting in sequence B1 would produce a delayed drywell temperature response relative to sequence B and would result in delayed EPA failure. The problem with the MARCH predictions for sequence B appears to be related to the code's choice of the calculational time step employed after vessel head failure in isolated containments. MARCH consistently employs a time step for the sequence B1 calculations, which is orders of magnitude smaller than that employed in sequence B. One impact of this phenomena is that MARCH does not permit the core/concrete reaction to begin in sequence B until 20 min after vessel head failure, while delaying the start of this reaction only 30 s in sequence B1. For these reasons, it is believed that MARCH results for sequence B are incorrect; that is, drywell cooler and EPA failure would actually occur earlier in sequence B than in sequence B1. The results of the sequence B MARCH analysis will not be discussed further, and MARCH analysis of the two event paths labeled "no analysis" on Fig. 6.3 were not performed because of these considerations.

It is our belief that sequence A1 represents the most realistic event path for the no-operator-action accident sequence at BFNP because the actual performance characteristics of the LPECCS pumps are modeled. However, limitations in the MARCH code's ability to model core spray ECC (see Sect. B.8 of Appendix B) severely compromise the credibility of the code's predictions for this sequence. For this reason, we have adopted sequence A as our standard or base case. The results of the sequence A MARCH analysis, which considers the LPECCS as designed, will be utilized

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*Note that the source of the leakage shifts from the reactor vessel to the DW upon reactor vessel bottom head failure in sequences A, A1, and B1.

in the fission product transport analysis to be described in Vol. II of this report. The results of the MARCH analysis of cases A, A1, and B1 are described in the following sections.

6.5 Base Case System Response — Sequence A

Figures 6.4–6.23 are plots of the BFNP post-core-uncovery system response for sequence A, as predicted by the ORNL version of MARCH 1.1. Figure 6.4 is a plot of the decay heat level between the time of core uncovery and core slump. The decay power is relatively constant at 0.8% of full power during this time period.

Figures 6.5–6.7 are plots of the reactor vessel pressure, water temperature, and collapsed water level, respectively, for the time period between core uncovery and core slump. Immediately following core uncovery, reactor vessel pressure (Fig. 6.5) increases to the SRV pressure setpoint and is held at that level by SRV actuation during the remainder of the in-core boil-off phase. Reactor vessel water temperature (Fig. 6.6) remains constant at the saturation temperature corresponding to the SRV actuation pressure during the in-core boil-off phase. As the vessel water level (Fig. 6.7) drops below the bottom of the active core, the steaming rate is reduced, resulting in a slight decrease in vessel water temperatures and pressure. The vessel water level continues to drop, uncovering the fuel bundle flow inlets ~1 min later.

Following flow inlet uncovery, MARCH predicts a brief period of intense flashing, which results in the pressure spike (Fig. 6.5) of 0.32 MPa (~47 psia). This repressurization caused by flashing is an unrealistic phenomena because flashing, which results from vessel depressurization, cannot produce an increase in vessel pressure. This pressure spike, together with numerous similar spikes (Fig. 6.5), are results of errors in the MARCH reactor vessel pool flashing model.

MARCH predicts that LPECCS injection begins at 530 min into the accident (Fig. 6.8) when the primary system pressure has dropped to 2.15 MPa (312 psia) and the reactor vessel water level is 178 in. above vessel zero (26 in. below the level of the fuel bundle flow inlets and 38 in. below the bottom of the active core). Over 45% of the core is molten at this time.

In addition to revealing the time of initiation of ECC injection, Fig. 6.8 indicates that the ECC injection flow is highly oscillatory in nature and that the peak ECC injection rate is slightly less than 529 kg/s (70,000 lb_m/min). The injection rate is ~17% of the maximum combined injection capability of the CS and LPCI systems. MARCH predicts that periodic LPECCS water injection would occur during a 34-min period prior to core slump and that the majority of this flow would be the result of LPCI system injection rather than CS system injection (based on design pump shutoff pressures). Indeed, MARCH predicts that LPCI system injection would occur 39 times during this 34-min period but that CS injection would occur only 6 times during the same period (because of high RPV pressure). Reactor vessel water level is predicted to vary between 215 and 203 in. during the period of ECCS injection. The active fuel region would remain completely uncovered during this period. Based on MARCH predictions for

the accident conditions and LPECCS design characteristics assumed, the LPECCSs are not capable of reflooding the core and preventing a core meltdown accident unless the reactor vessel pressure is reduced by operator action to a point significantly below the LPECCS shutoff head.

The MARCH predictions concerning the impact and performance of the LPECCSs are subject to extremely large uncertainties because of MARCH's inability to accurately model core spray systems (Appendix B). These uncertainties may be less significant in the present case because reactor vessel pressure would remain above the effective CS shutoff head throughout most of the accident. Additional uncertainties are introduced, however, because MARCH employs extremely simplistic core meltdown models. As previously mentioned, in this sequence over 45% of the core is predicted to be molten at the time of LPECCS injection. The present state of knowledge concerning accident phenomenology under these conditions is weak at best.

Figures 6.9 and 6.10 are plots of the water and steam leakage rates from the RPV via the scram discharge volume system break. Water leakage is predicted to peak at 79.3 kg/s (10,490 lb_m/min) just prior to stopping completely at 457 min as the water level drops below the fuel bundle flow inlets. Water leakage is not predicted to resume until 531 min into the accident (74 min later), when LPECCS injection results in water levels above the fuel bundle flow inlets. During the period of LPECCS injection, the peak RPV water leakage rate is predicted to be ~35.8 kg/s (4740 lb_m/min). The maximum steam leakage rate of 22.9 kg/s (3034 lb_m/min) (Fig. 6.10) is predicted to occur when the fuel bundle flow inlets are first uncovered at 457 min into the accident. The oscillatory nature of the decreasing steam leakage (Fig. 6.10) in the period between 457 and 530 min is not believed to be realistic; it appears to be closely coupled to instabilities in MARCH's reactor vessel pool flashing calculations. Steam leakage is predicted to vary between 0 and 1.5 kg/s (202 lb_m/min) in the period between 530 and 564 min (core slump) as the vessel water level oscillates around the level of the fuel bundle flow inlets.

Figure 6.11 is a plot of the hydrogen leakage rate from the reactor vessel during the time preceeding core slump. The leakage rate is predicted to peak at ~0.48 kg/s (63 lb_m/min) at the time the RPV water level first drops below the fuel bundle flow inlets, and to vary between 0 and 0.36 kg/s (48 lb_m/min) throughout the period preceeding LPECC injection. During the LPECC injection phase, hydrogen flow from the reactor pressure vessel is predicted to peak at 0.58 kg/s (77 lb_m/min) with an intense hydrogen flow spike of 12.6 kg/s (1663 lb_m/min) predicted to occur at the time of core slump. Figure 6.12 is a related plot of the in-vessel Zr-H_2O reaction energy generation rate during the period prior to core slump. The reaction is predicted to be highly steam-starved throughout the interval. The oscillatory nature of the plot is caused by oscillations in the MARCH RPV pool flashing calculation, which result from SRV actuation and fuel bundle flow inlet uncovery. The time-dependent fraction of the total fuel cladding reacted is shown in Fig. 6.13. Approximately 34% of the cladding is predicted to be reacted at the time of core slump.

Figures 6.14 and 6.15 are plots of the maximum fuel temperature and the coremelt fraction prior to core slump. Maximum fuel temperatures increase rapidly following core uncovery, rising over 1400°F in the 13-min

period between uncovery of the top and bottom of the core. Core melting (Fig. 6.15) does not begin until 506 min into the accident, when the entire core has been uncovered for ~51 min.

Figures 6.16 and 6.17 are plots of the hydrogen leakage rate and integrated steam leakage into the reactor building throughout the course of the accident. In reality, the small initial safety relief valve flows (labeled "SRV actuation" in Figs. 6.16 and 6.17) would actually discharge into the suppression pool. However, the MARCH user is forced to select a single destination for all break and SRV flows. For this analysis, in which the SRV flow is relatively small, MARCH was instructed to send all break and SRV flow to the reactor building. In actuality, prior to fuel bundle flow inlet uncovery, only saturated water would flow into the reactor building through the scram discharge volume break. An analysis of Fig. 6.17 reveals that when the SRV flow is neglected, ~118,000 kg (260,000 lb_m) of steam would actually be injected into the reactor building during the in-vessel phase of the accident.

Figures 6.18–6.20 are plots of the drywell temperature, pressure, and hydrogen mass throughout the course of the accident. Following vessel head failure, the drywell temperature is predicted to increase rapidly as the molten core debris begins interacting with the concrete floor. The drywell electrical penetration assemblies are predicted to fail by overtemperature at 707 min into the accident when the drywell pressure is ~0.16 MPa (~23 psia). The total drywell hydrogen mass is predicted to peak at 106 kg (~235 lb_m) just prior to EPA failure. The drywell atmospheric temperature response after containment failure is driven by the core-concrete reactions as modeled in the INTER subroutine. INTER is a highly empirical code that was developed as a tool for guiding the direction of core-concrete interaction studies. The authors of the code do not regard it as being sufficiently refined to provide reliable predictions of the course of meltdown accidents in nuclear power plants.[5] MARCH predictions for containment behavior subsequent to initiation of the melt/concrete reaction should, therefore, be viewed only as a rough approximation of actual containment behavior. Unfortunately, these results are very important to the analysis of fission product transport.

The wetwell temperature, pressure, and hydrogen mass are plotted in Figs. 6.21–6.23 respectively. Because of the MARCH break/SRV flow modeling limitations previously discussed, MARCH sends all SRV flow to the reactor building. The result of this limitation is that the suppression pool does not begin to heat up until the reactor vessel bottom head fails at 652 min into the transient. Following vessel bottom head failure, the wetwell temperature and pressure are predicted to rise rapidly, reaching peaks of 62°C (~143°F) and 0.16 MPa (23 psia) just prior to drywell EPA failure. Wetwell hydrogen mass is predicted to peak at 29.9 kg (66 lb_m) just prior to EPA failure.

6.6 System Response — Sequence A1

Figures 6.24–6.34 display the results of the MARCH analysis of sequence A1 ("ECC as-built" case). Prior to the time of initiation of ECC injection at 523 min, the system response is identical to that described

in Sect. 6.5. Section 6.5 describes any features of interest in Figs. 6.24–6.34 that occur prior to 523 min into the accident. In this section, we will discuss only those features of the sequence A1 system response that differ from those of sequence A.

As previously discussed (Sect. 6.3), the ECC as-built performance differs from the design performance criteria in two substantial ways:

1. core spray injection occurs <u>before</u> LPCI in sequence A1 rather than after, and
2. both the CS and LPCI systems can inject water into the reactor vessel at higher RPV pressures than dictated by design criteria.

Because of MARCH's inability to model core spray systems (Appendix B), the results of this sequence A1 analysis are subject to extremely large uncertainties. A significantly more detailed thermohydraulic analysis would be needed before realistic conclusions regarding the SDV break accident mitigation capability of BFNP's LPECCS could be drawn.

Figures 6.24–6.27 are plots of the sequence A1 RPV pressure, water temperature, water level, and ECC injection rate throughout the period preceding core slump. Following initial ECC injection (Fig. 6.27) at 530 min, the system is predicted to rapidly reach an equilibrium state in which the ECC flow is just sufficient to replace the water loss via the SDV break (Fig. 6.28). During this period of ECC injection, the system pressure is predicted to stabilize at 2.45 MPa (gage) (356 psig), water temperature at 221°C (430°F) (~3°F below saturation temperature), water level at 5.54 m [(218 in.) so that 2 in. of active core region is covered] and ECC flow at 32.4 kg/s (4290 lb_m/min). Steam and hydrogen leakage are predicted to cease entirely after ECC injection because of recovery of the fuel bundle flow inlets (Figs. 6.29 and 6.30).

Figure 6.31 is a plot of the time-dependent sequence A1 $Zr-H_2O$ reaction energy. MARCH predicts that the reaction ceases completely following initiation of LPECCS injection. An analysis of the MARCH output reveals that this behavior is caused by steam starvation phenomena. Although MARCH predicts that there is over 362 kg (800 lb_m) of steam within the RPV during this phase of the accident, this existing steam is not allowed to react. MARCH only allows the $Zr-H_2O$ reaction to progress in the presence of moving steam. The reader should recall that MARCH models all ECC injection as if it enters the RPV below the core. The major effects of this are

1. direct production of steam caused by interaction of core spray water with the hot core is not modeled,
2. the temperature of the RPV water is reduced below the saturation temperature corresponding to the existing RPV pressure if the water level is below the bottom of the core, and
3. the water level rises.

In this sequence, MARCH predicts that no steam is available for the $Zr-H_2O$ reaction because the RPV water temperature is predicted to stabilize at ~3°F below saturation (no boiling), the fuel bundle flow inlets are covered with water (no flashing), and the CS water is assumed never to contact the hot core (no vaporization). These MARCH modeling limitations

are particularly unfortunate because an analysis of the MARCH output reveals that for sequence A1, the great majority of this ECC flow would be the result of core spray injection.

The clad reaction and melt fractions for sequence A1 are plotted in Figs. 6.32 and 6.33 respectively. The impact of the reduction in $Zr-H_2O$ reaction energy input can be clearly seen in both figures. Figure 6.34 is a plot of the drywell atmospheric temperature throughout the accident. The drywell is predicted to fail 729 min into the accident via EPA seal degradation when the drywell pressure is only 0.19 MPa (~27 psia).

In summary, MARCH predicts that RPV head and drywell failure occur later in sequence A1 than in sequence A. The delay in the timing of these events appears to be coupled to decreased $Zr-H_2O$ reaction energy generation rather than improved core cooling. MARCH predicts that the integrated pre-core-slump $Zr-H_2O$ reaction energy generation is 2.7×10^{10} J (~2.6×10^7 Btu) less in sequence A1 than in sequence A. Because of restrictions in MARCH's ECC modeling capability, the predictions have limited validity and are not suitable for determination of BFNP LPECCS accident mitigation capabilities for the scram discharge volume break accident. It is unclear whether these results have any validity for cases in which core spray systems are unavailable.

6.7 System Response - Sequence B1

In this section, the BFNP system response during sequence B1 (no LPECC) will be briefly described. Figures 6.35–6.44 display the results of the MARCH analysis of this sequence. Prior to 530 min into the accident, the system response is identical to that of sequence A. Section 6.5 discusses the system response prior to 530 min. The discussion in this section will be limited to the time period after 530 min in this accident sequence.

Figures 6.35–6.37 are plots of the BFNP RPV pressure, water temperature, and water level throughout the period preceding core slump in sequence B1. Unlike sequences A and A1, it is assumed in sequence B1 that low pressure emergency core cooling systems are not available to replace the primary system mass inventory loss via the scram discharge volume break. As a result, the RPV water level is predicted to drop monotonically during the period preceding core slump (Fig. 6.37), and the primary system pressure and water temperature are predicted to behave in a similar fashion (Figs. 6.35 and 6.36).

As the RPV water level drops below the fuel bundle flow inlets, the break flow changes from water to steam (Figs. 6.38 and 6.39) and remains steam throughout the pre-core-slump period. Because the flow inlets remain uncovered, MARCH predicts that the RPV pool periodically flashes throughout this period, thus providing steam to drive the $Zr-H_2O$ reaction (Fig. 6.40). This $Zr-H_2O$ reaction produces significant quantities of hydrogen, which are leaked from the primary system via the SDV break (Fig. 6.41). Approximately 680 kg (1500 lb_m) of H_2 is leaked to the reactor building prior to core slump. If the Standby Gas Treatment system (SBGTS) continued to be operable in this sequence, the possibility of a hydrogen detonation in the reactor building could become significant.

The impact of the $Zr-H_2O$ reaction heating can be seen clearly in Figs. 6.42 and 6.43, which are plots of the time-dependent fraction of fuel cladding reacted and the fraction of the core that is molten during sequence B1. Approximately 38% of the cladding is predicted to be reacted at the time of core slump (549 min).

Figure 6.44 is a plot of the BFNP drywell atmospheric temperature response throughout the course of sequence B1. The drywell coolers are predicted to fail at 644 min into the transient, followed 1 min later by EPA venting and failure.

6.8 Summary

The sequence of events following core uncovery in the scram discharge volume break, no-operator-action accident has been described in this section. The timing of these events is significantly influenced by the location of the control rod guide tube fuel bundle flow inlets, CRD withdraw lines, and the temperature sensitivity of the drywell electrical penetration assemblies.

Analysis of the as-designed and as-built LPECC cases indicate that these systems would not be capable of recovering the core and terminating the accident under the no-operator-action scenario because of excessive reactor vessel pressures. Because of limitations in MARCH's thermal-hydraulic and ECCS modeling capabilities, these results are subject to extreme uncertainty — especially in the ECCS as-built case. The as-designed LPECC case has been selected for use as the base case for the fission product transport calculations to be presented in Vol. II of this report.

Containment failure follows immediately after failure of the drywell coolers in all sequences that were analyzed in this study, but it is unclear that continued operation of these coolers would delay containment failure in any sequence. Further analysis should be performed to address this uncertainty when a more sophisticated core-concrete reaction code such as CORCON is available to replace the MARCH subroutine INTER. The total elapsed time between core uncovery and gross drywell failure caused by EPA seal degradation was determined to be ~4.4 h.

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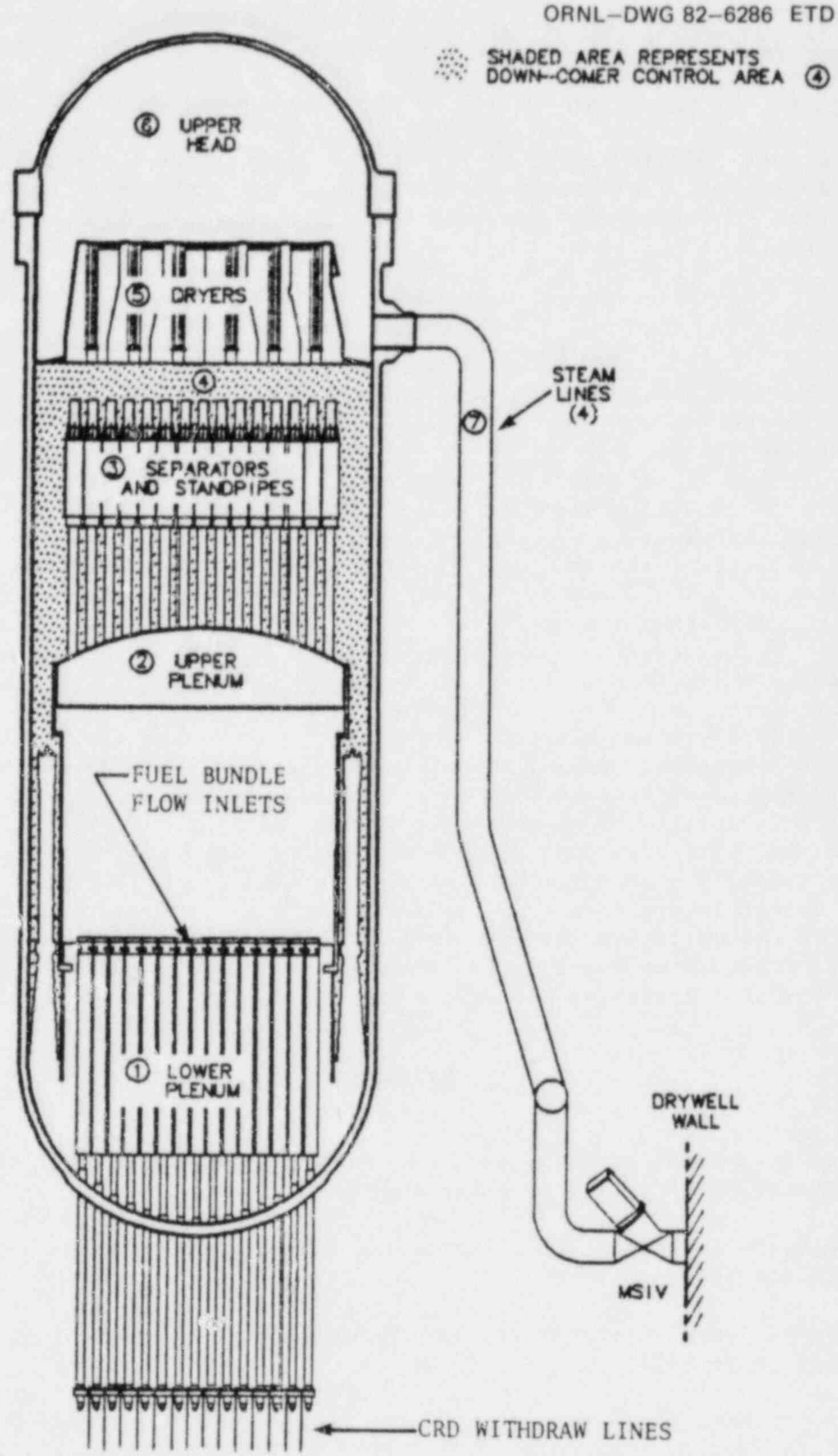

Fig. 6.1. Location of fuel bundle flow inlets and CRD withdraw lines.

ORNL-DWG 82-6287 ETD

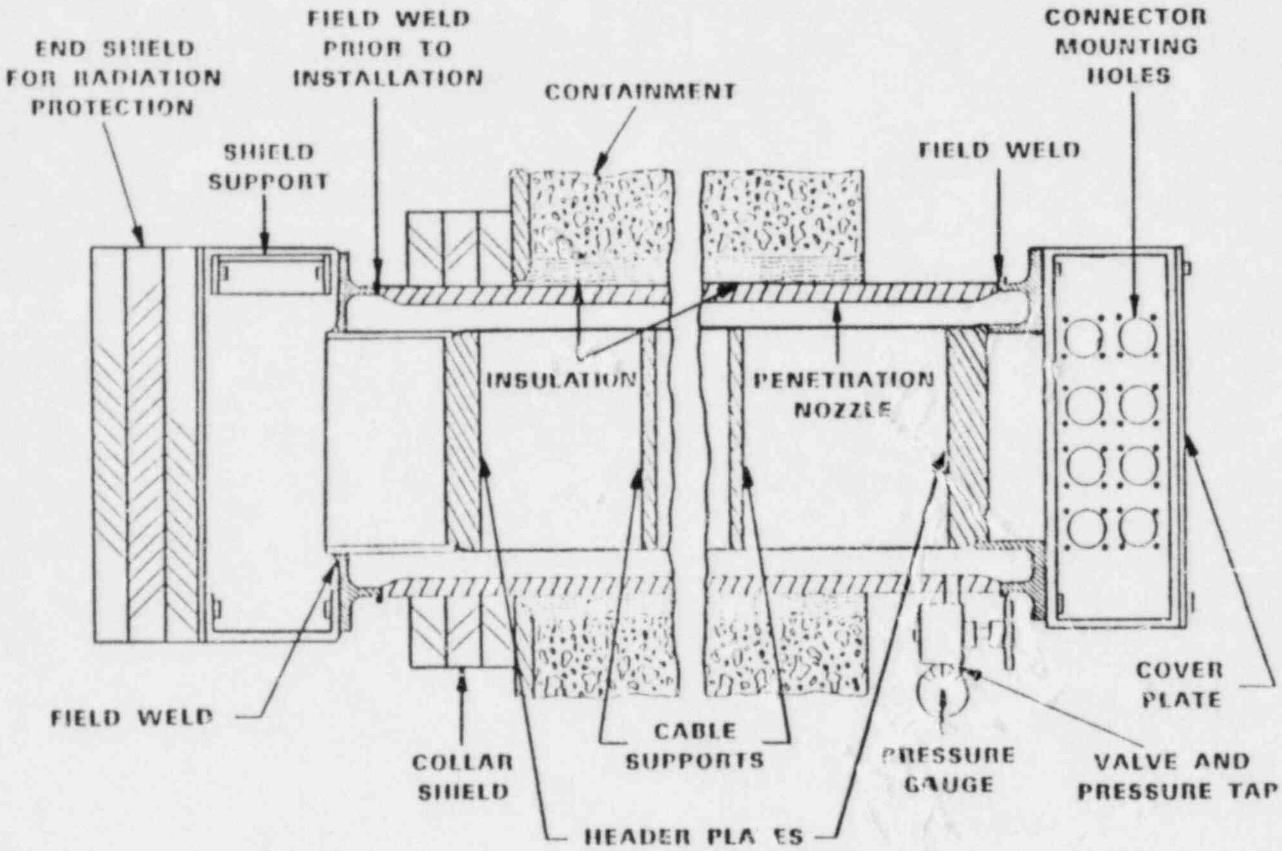

Fig. 6.2. Typical electrical penetration assembly canister.

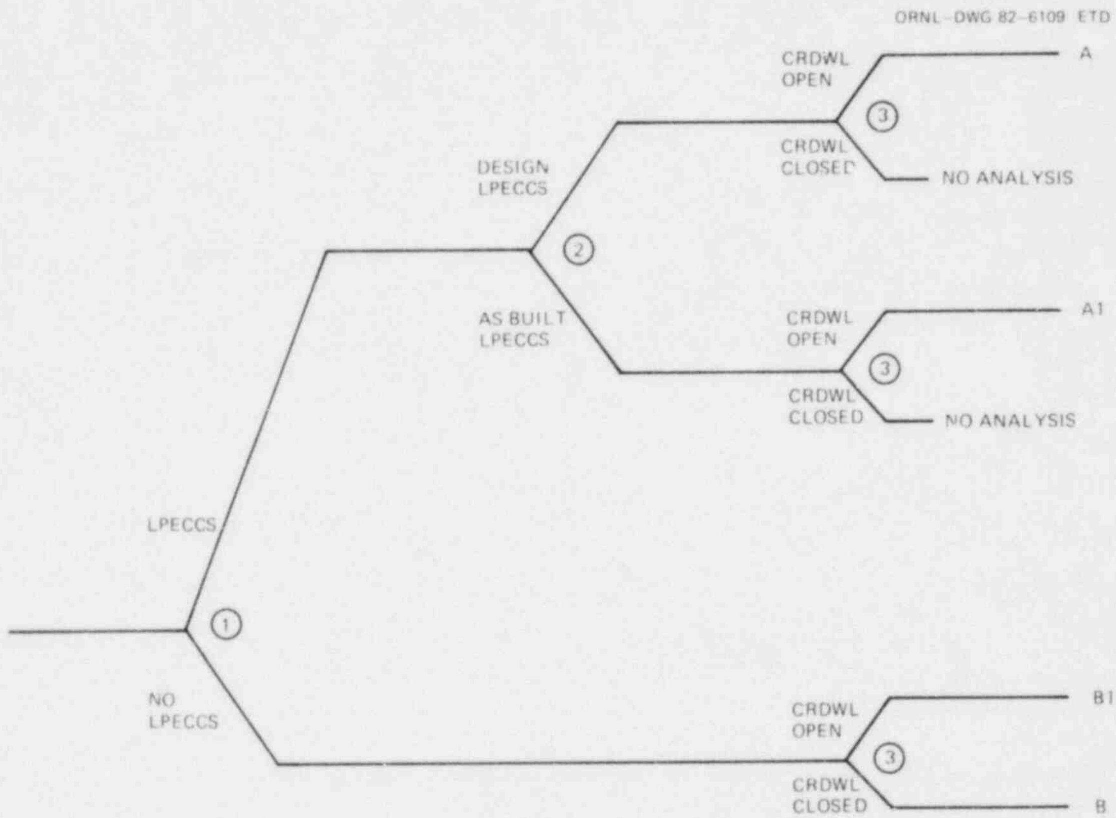

Fig. 6.3. BFNP#1 SDV break, no operator action accident sequences.

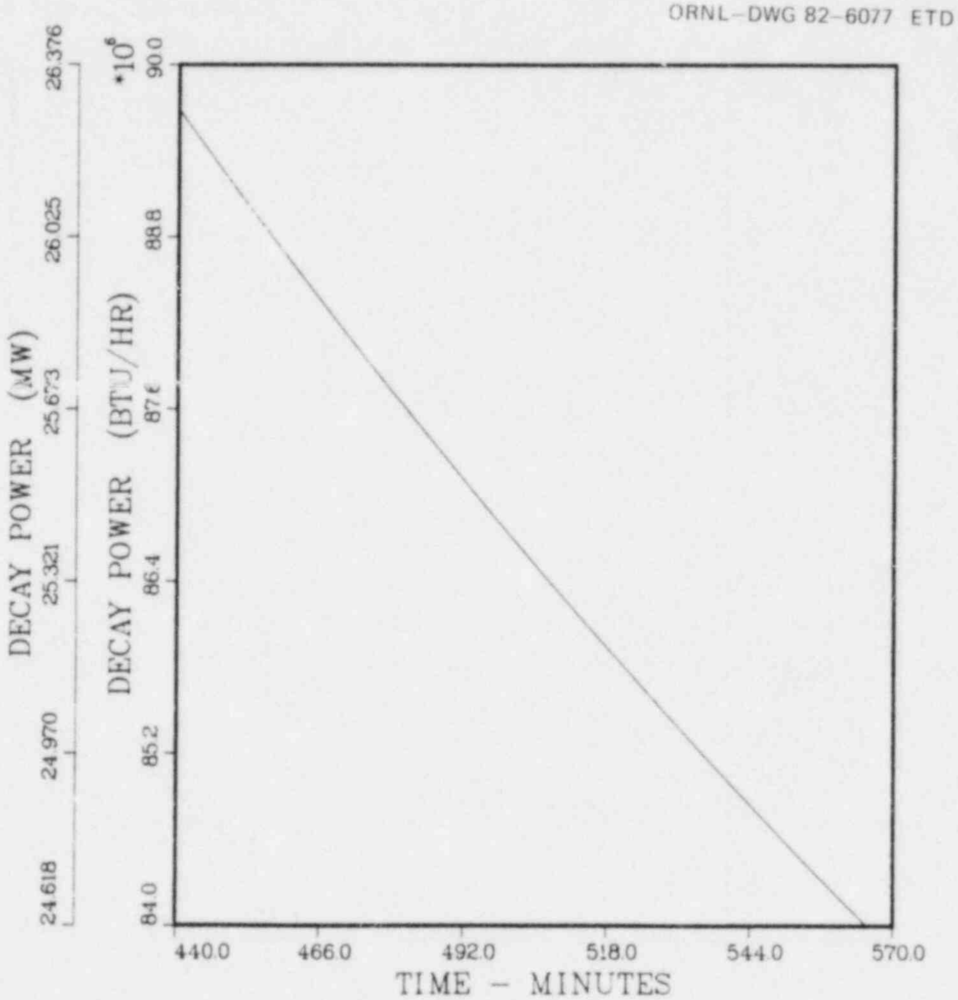

Fig. 6.4. Sequence A decay power.

ORNL-DWG 82-6078 ETD

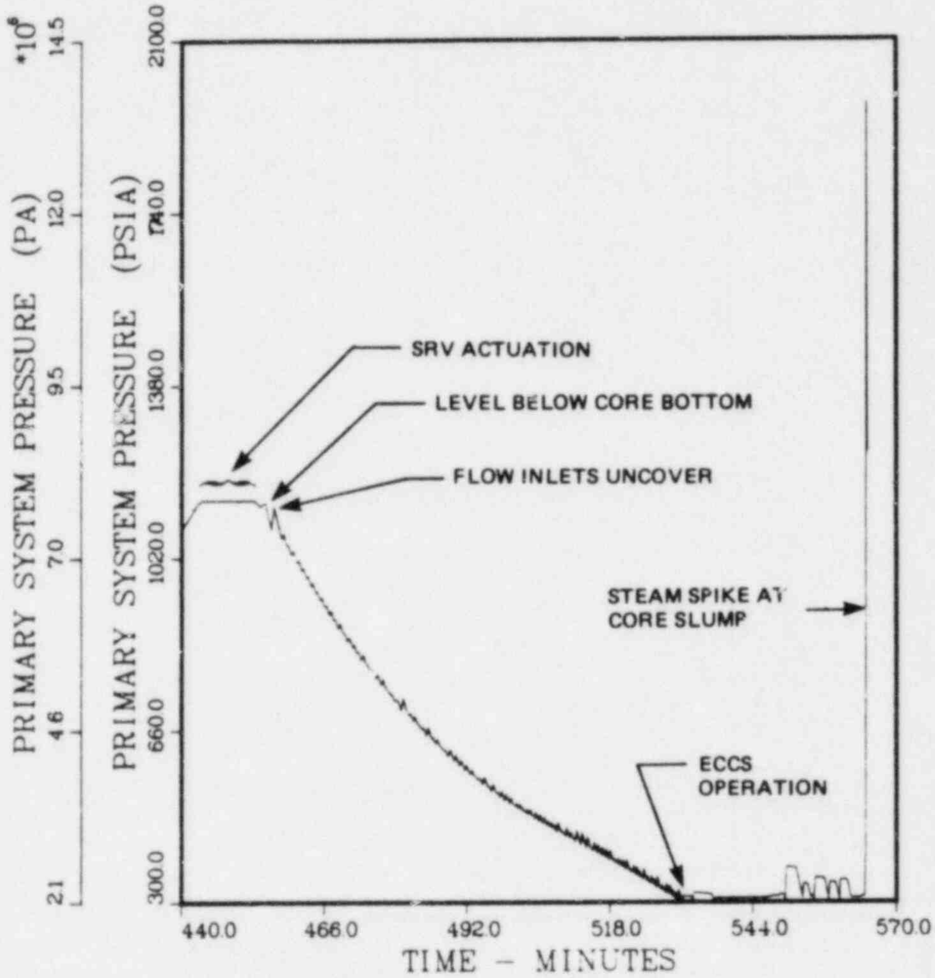

Fig. 6.5. Sequence A primary system pressure.

ORNL-DWG 82-6079 ETD

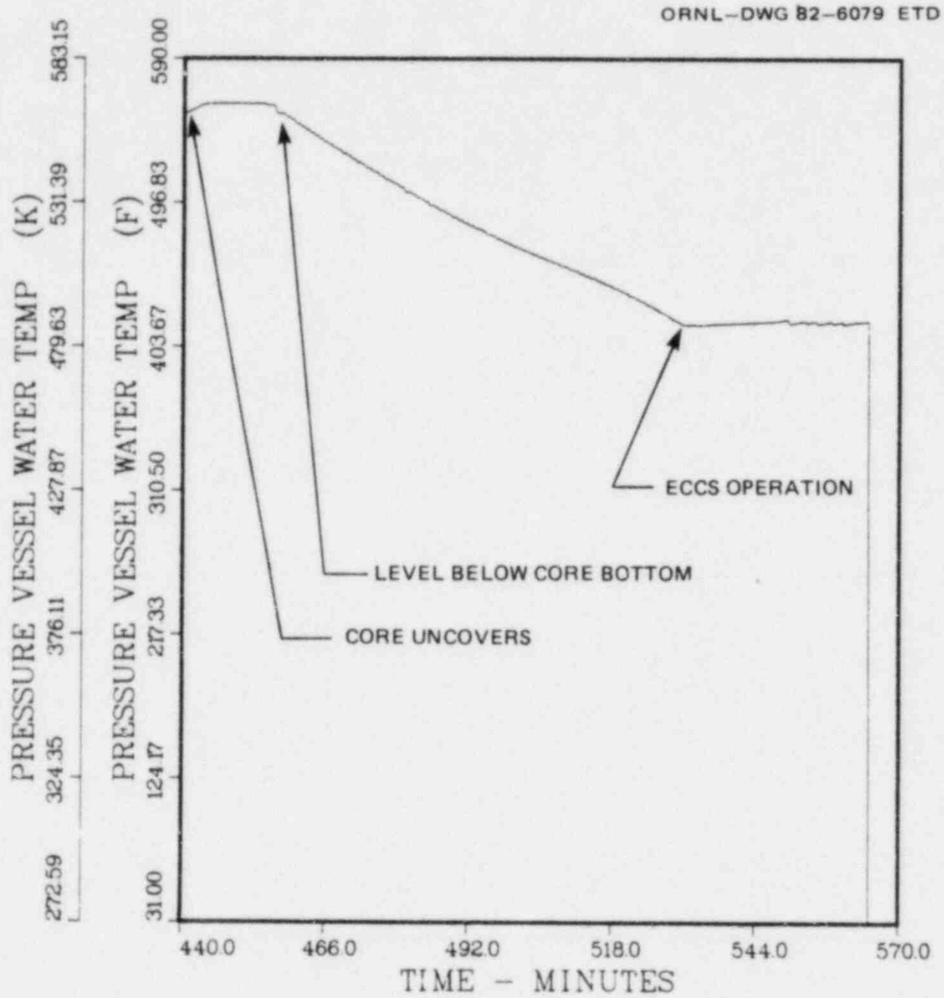

Fig. 6.6. Sequence A RPV water temperature.

ORNL-DWG 82-6080 ETD

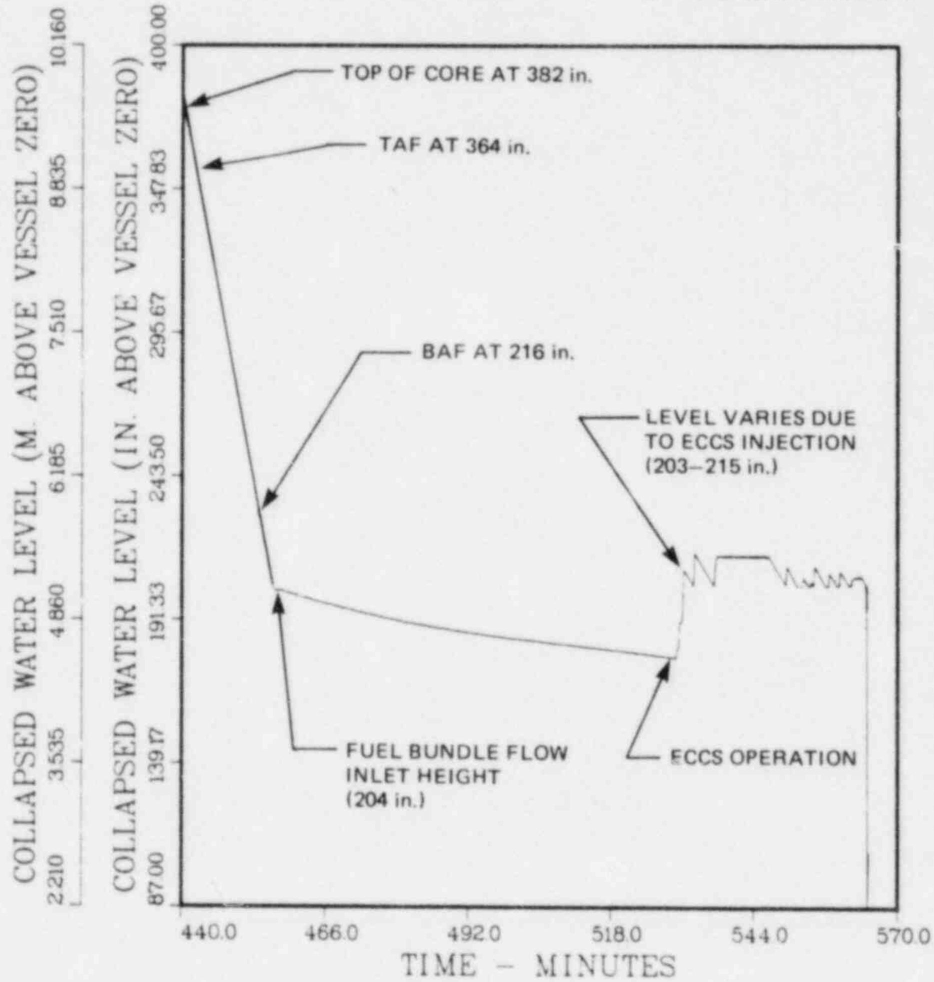

Fig. 6.7. Sequence A collapsed RPV water level.

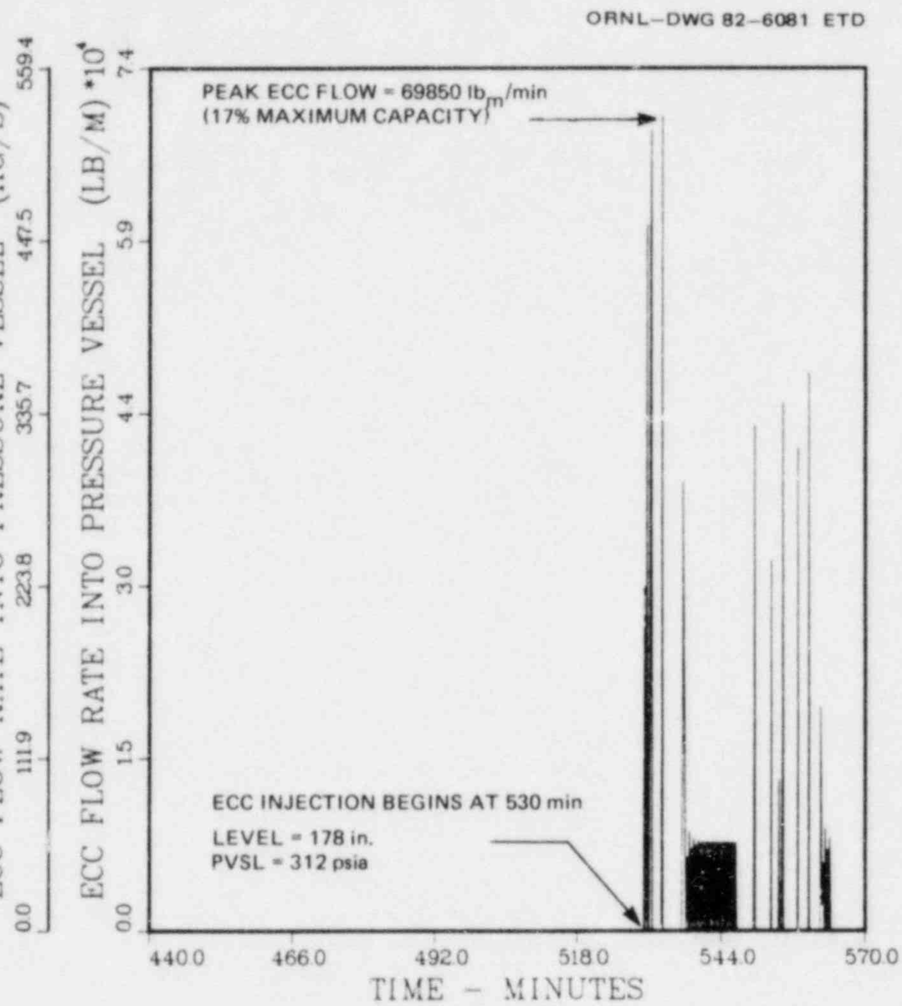

Fig. 6.8. Sequence A total ECC injection rate.

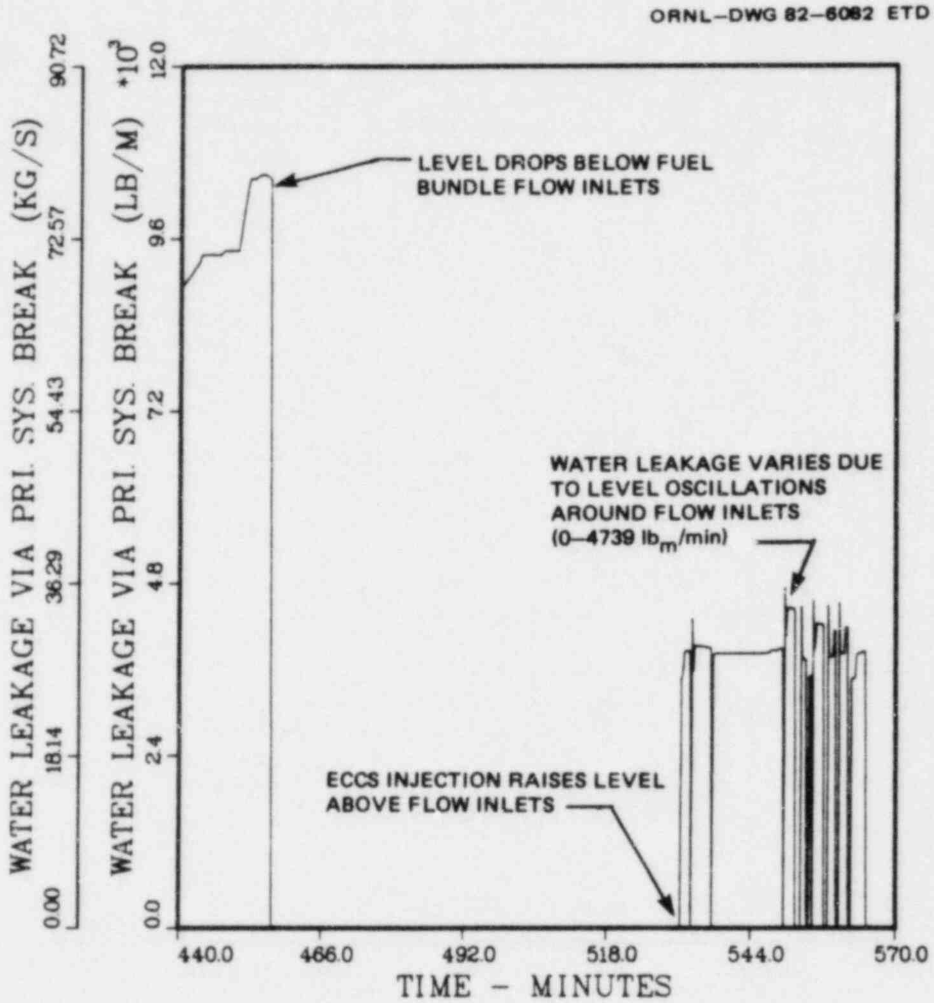

Fig. 6.9. Sequence A water leakage via SDV break.

ORNL-DWG 82-6083 ETD

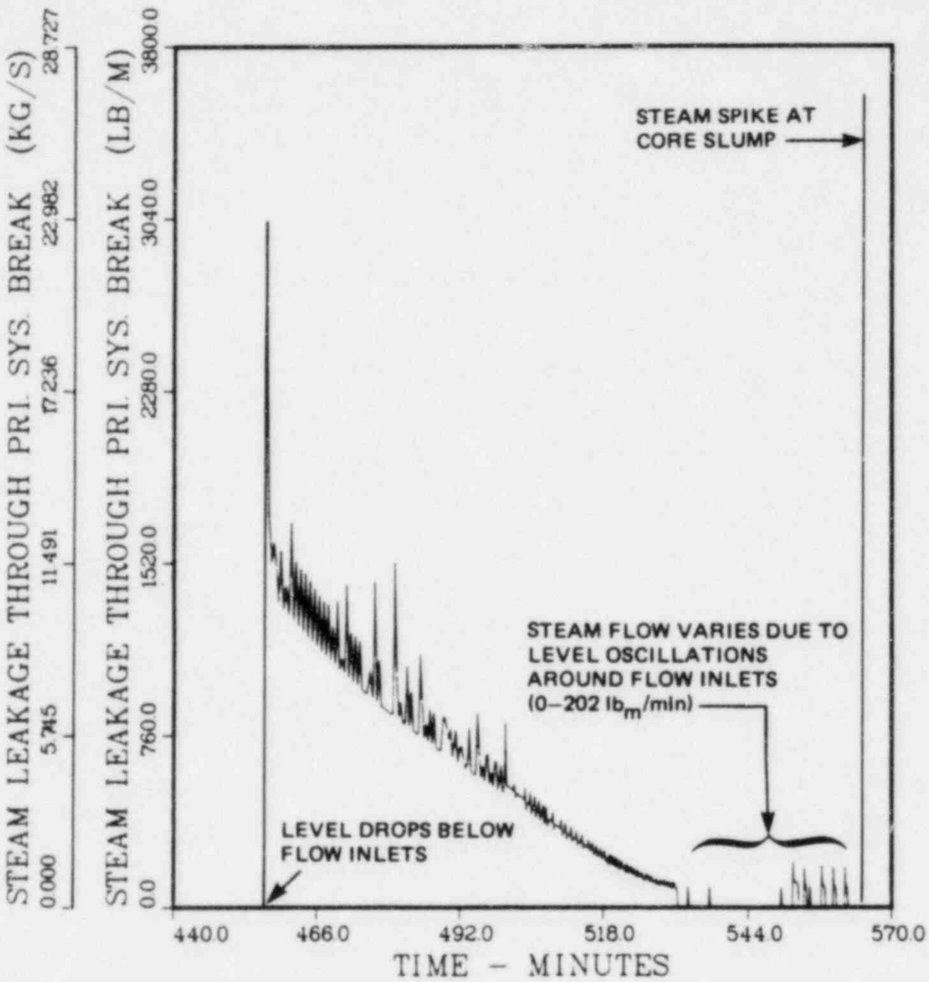

Fig. 6.10. Sequence A steam leakage via SDV break.

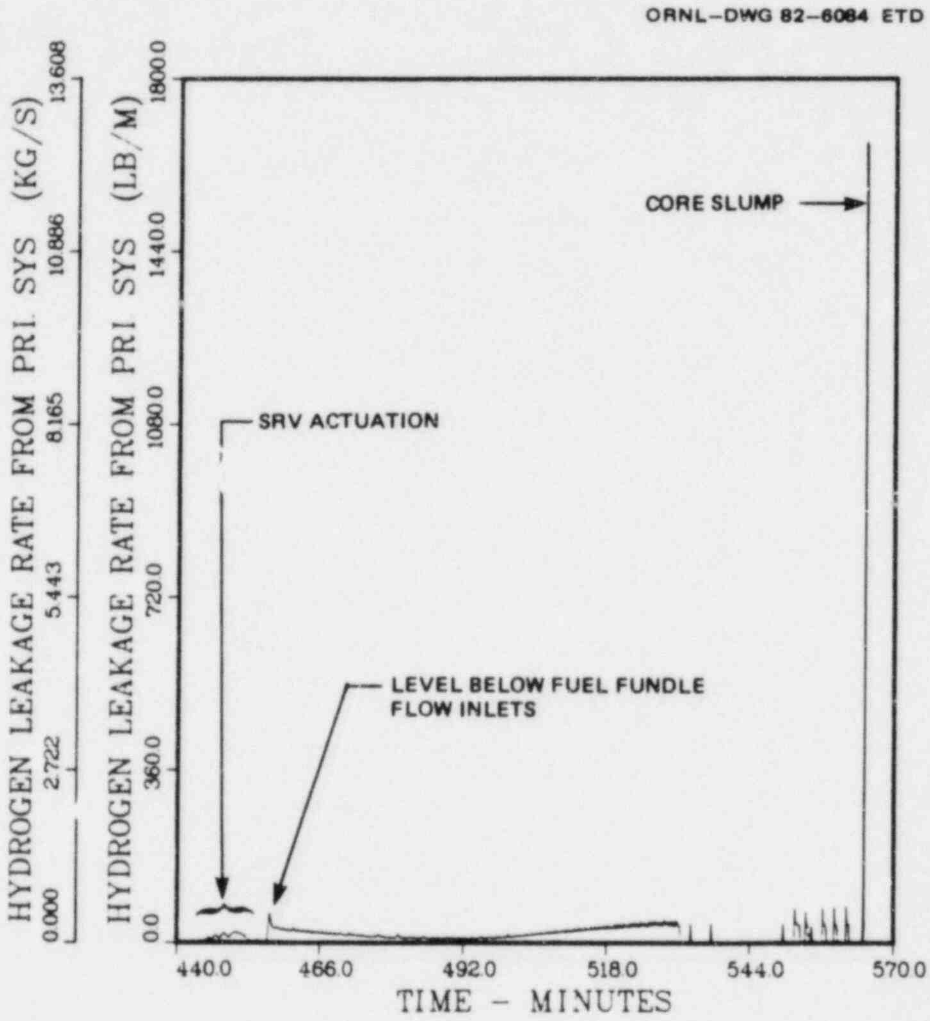

Fig. 6.11. Sequence A hydrogen leakage rate via SDV break and SRVs.

ORNL-DWG 82-6085 ETD

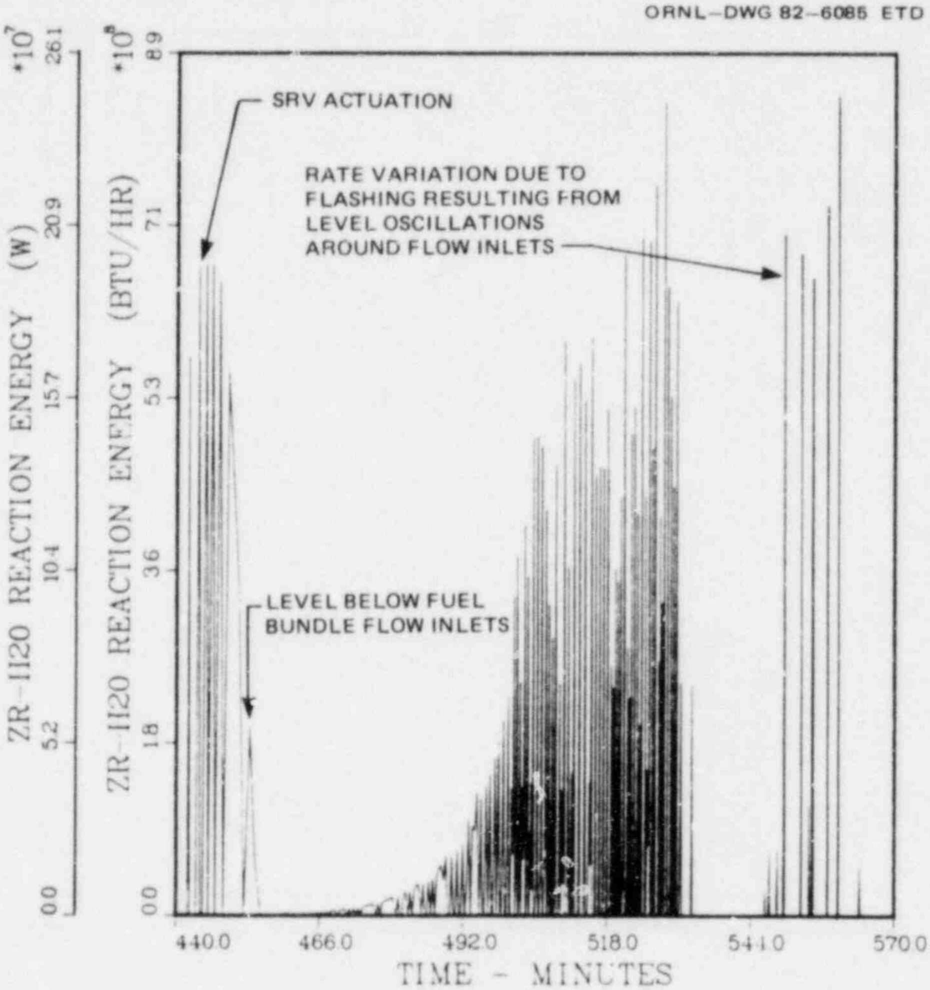

Fig. 6.12. Sequence A $Zr-H_2O$ reaction rate.

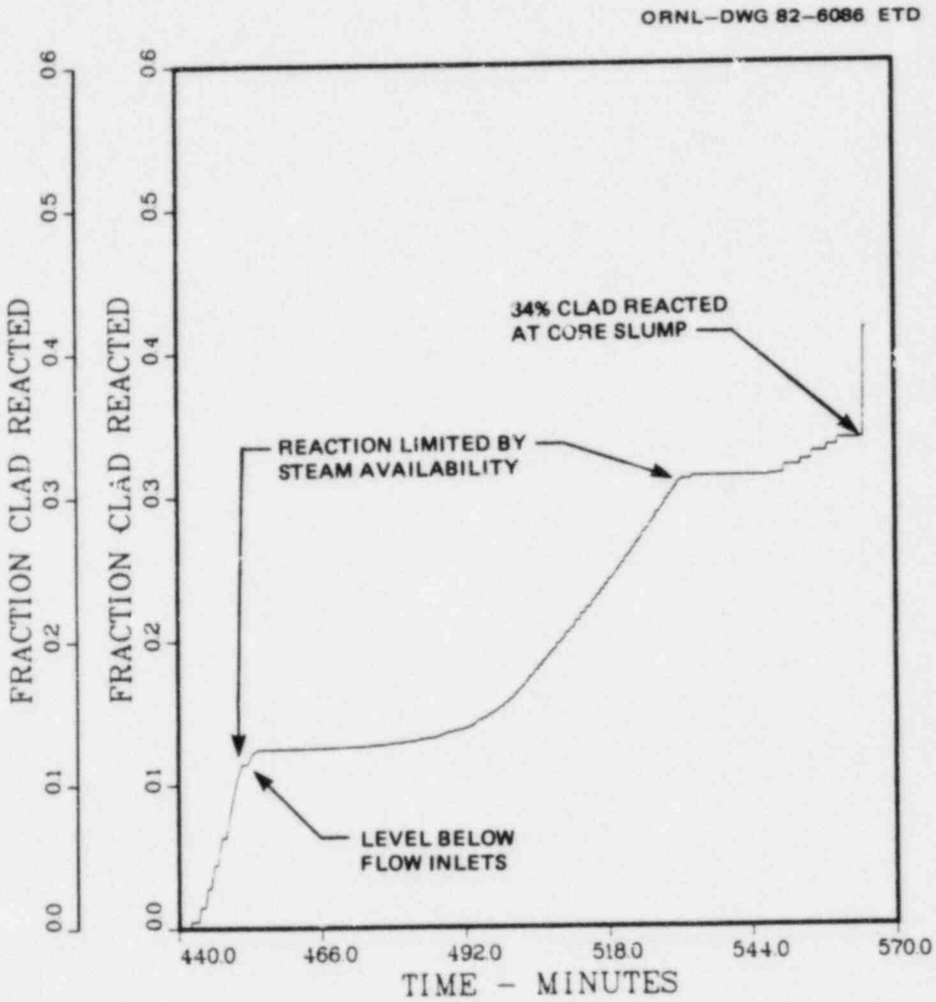

Fig. 6.13. Sequence A fraction of clad reacted.

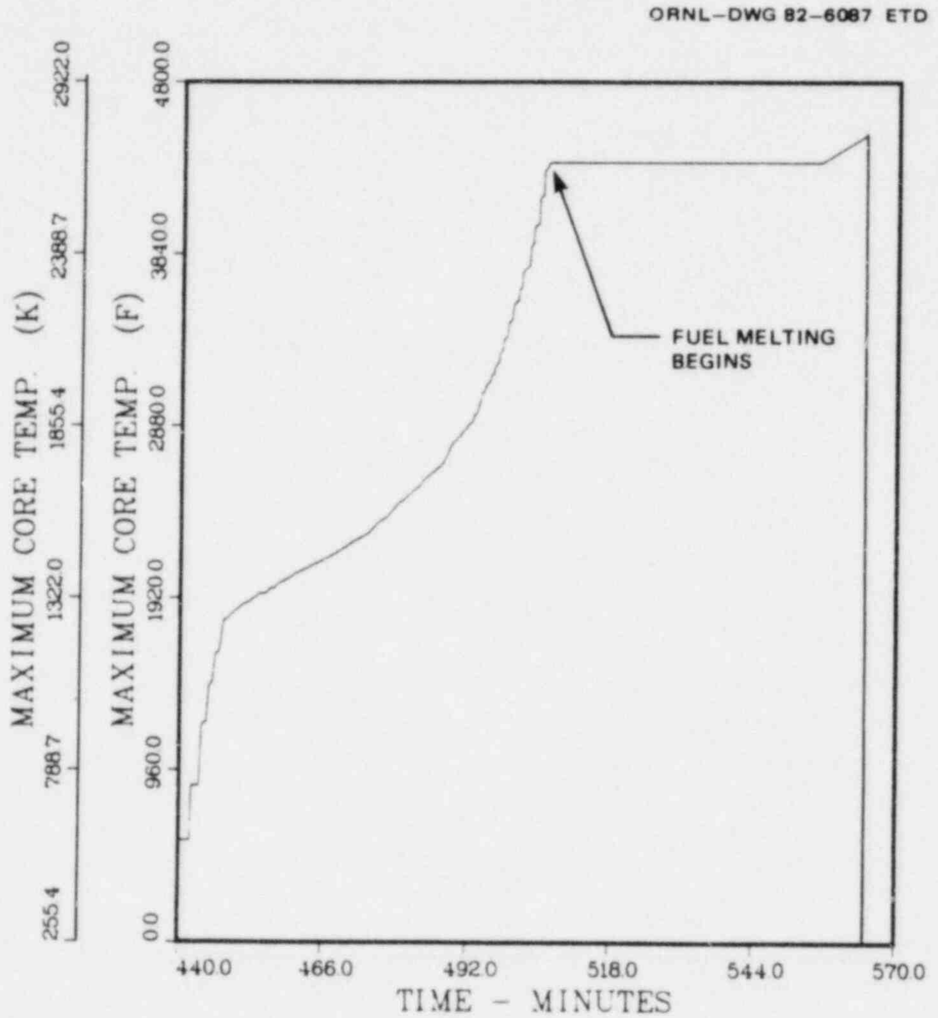

Fig. 6.14. Sequence A maximum core temperature.

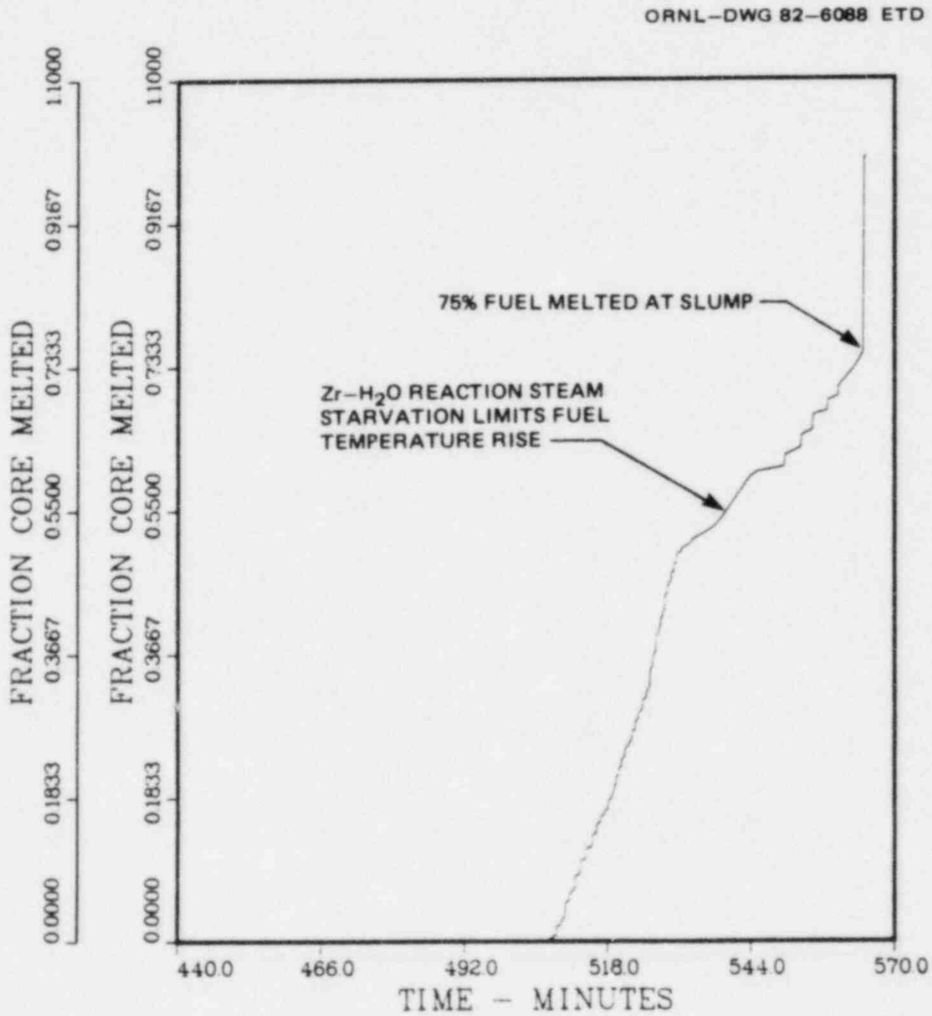

Fig. 6.15. Sequence A fraction of core melted.

ORNL–DWG 82–6089 ETD

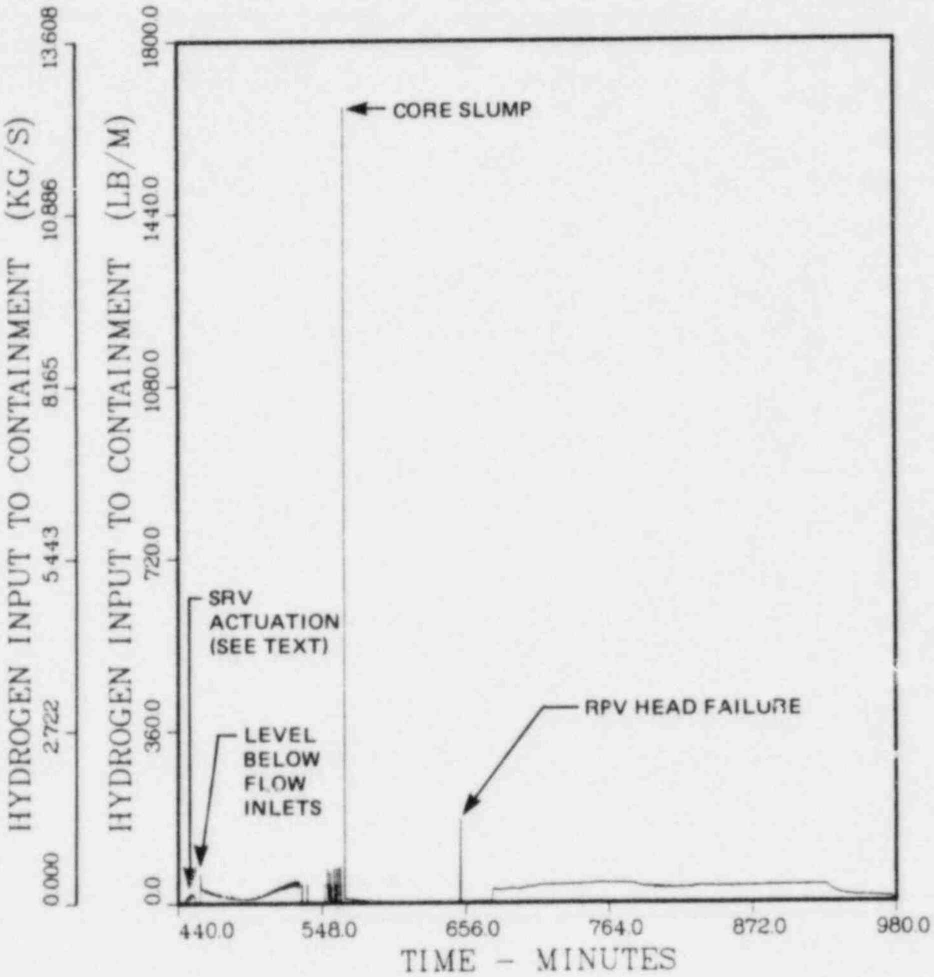

Fig. 6.16. Sequence A hydrogen input to reactor building.

ORNL-DWG 82-6090 ETD

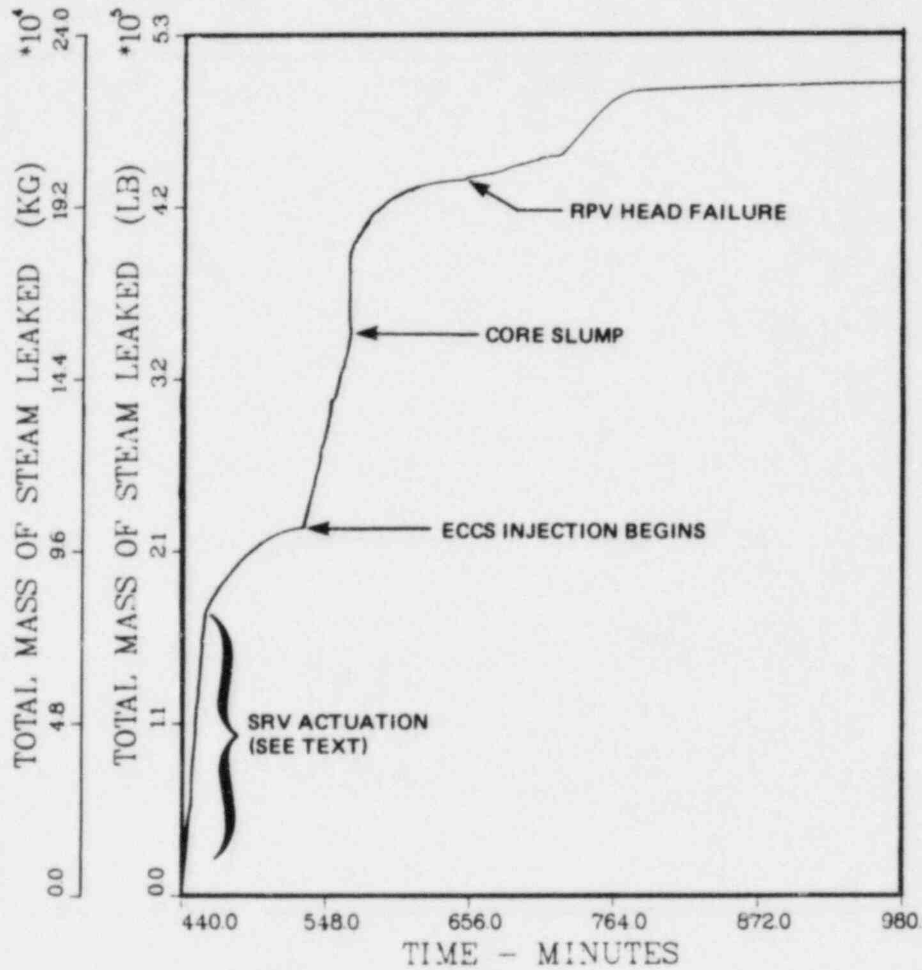

Fig. 6.17. Integrated steam leakage to reactor building.

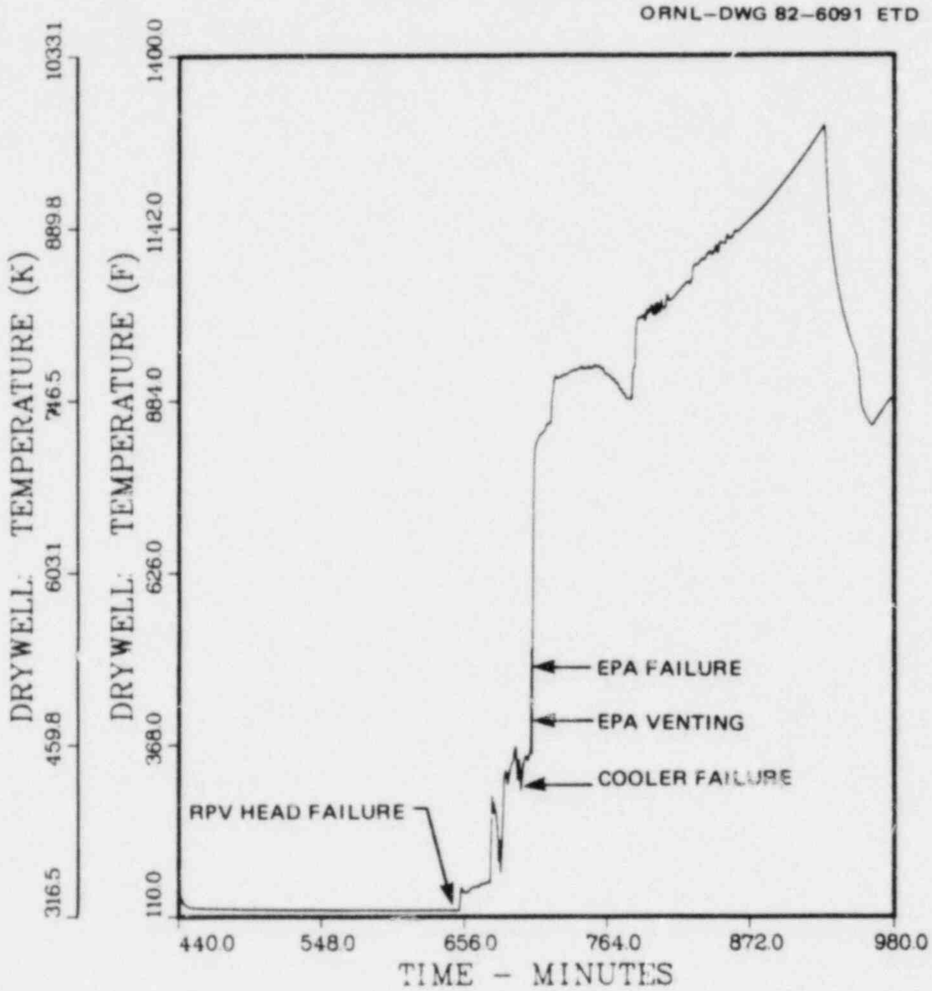

Fig. 6.18. Sequence A drywell temperature.

ORNL-DWG 82-6092 ETD

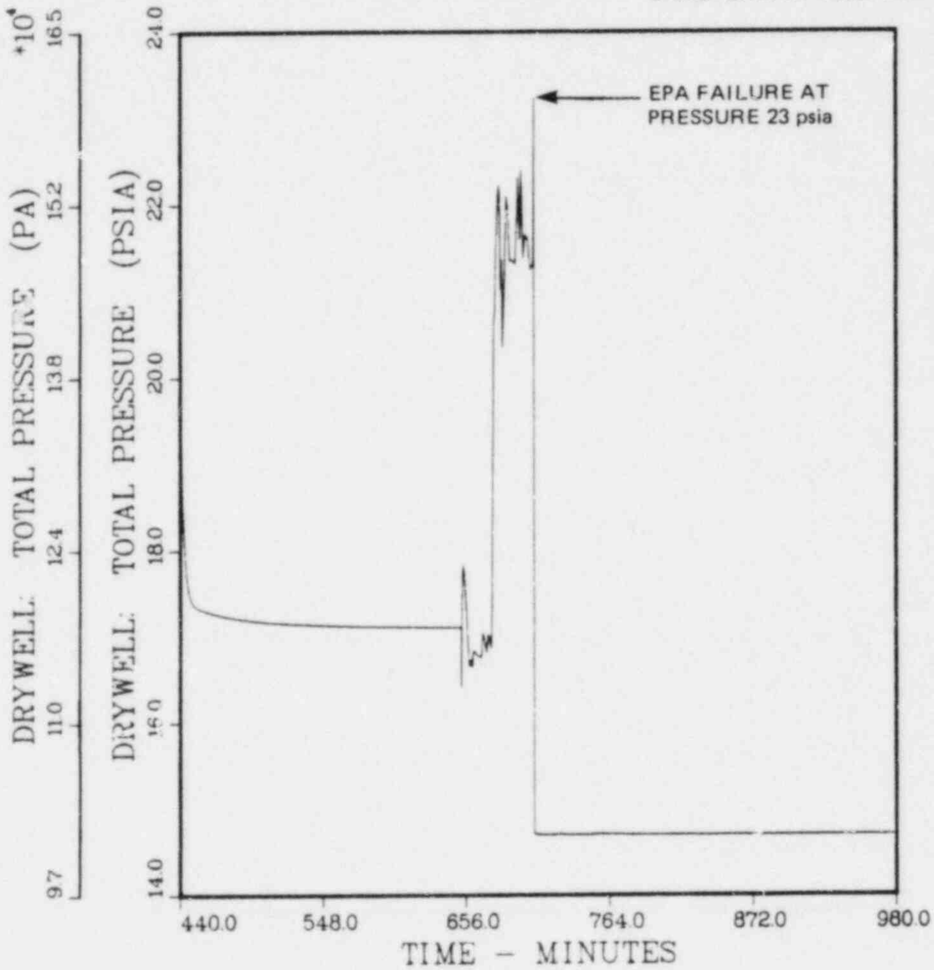

Fig. 6.19. Sequence A drywell pressure.

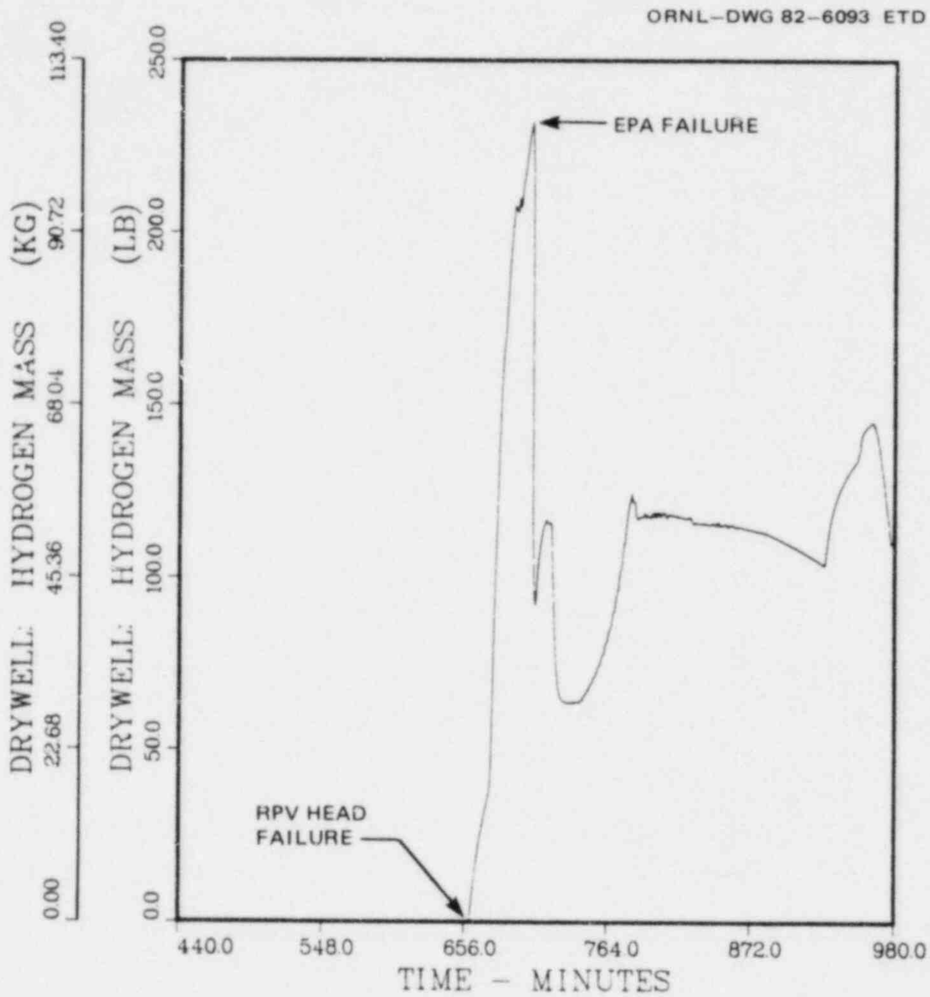

Fig. 6.20. Sequence A drywell hydrogen mass.

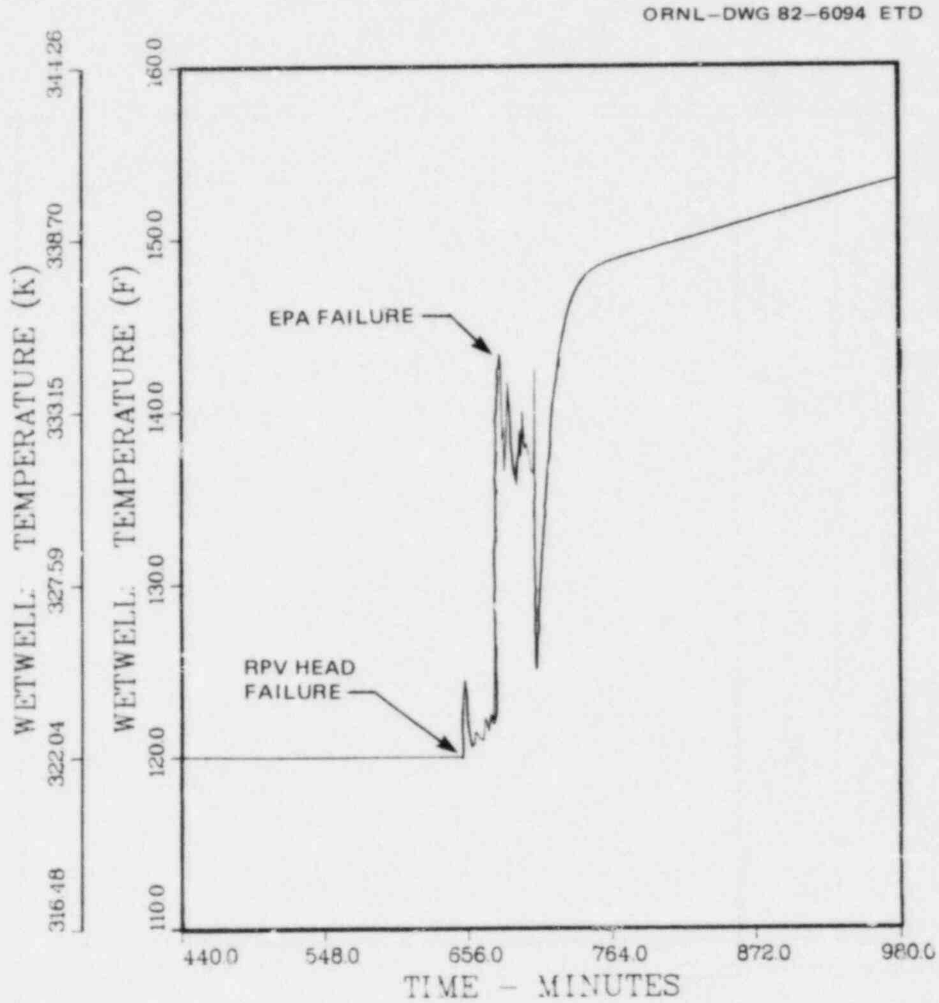

Fig. 6.21. Sequence A wetwell temperature.

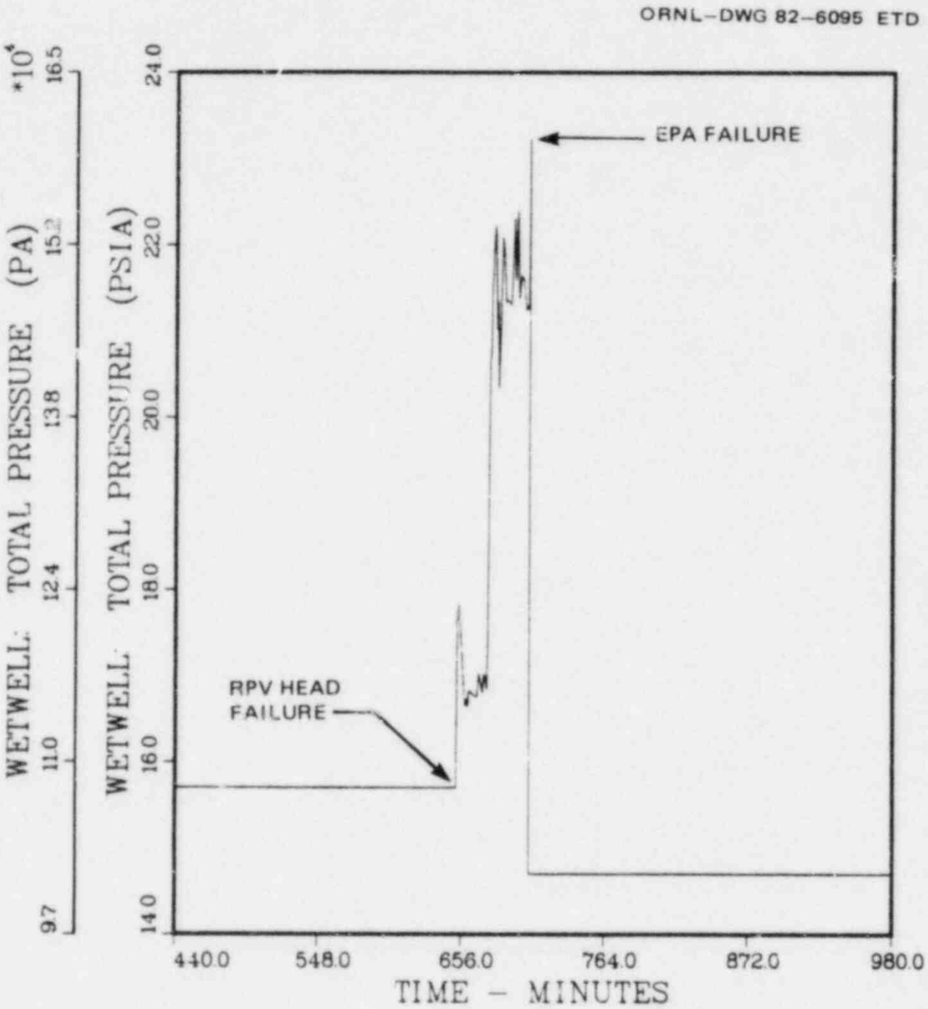

Fig. 6.22. Sequence A wetwell pressure.

ORNL-DWG 82-6096 ETD

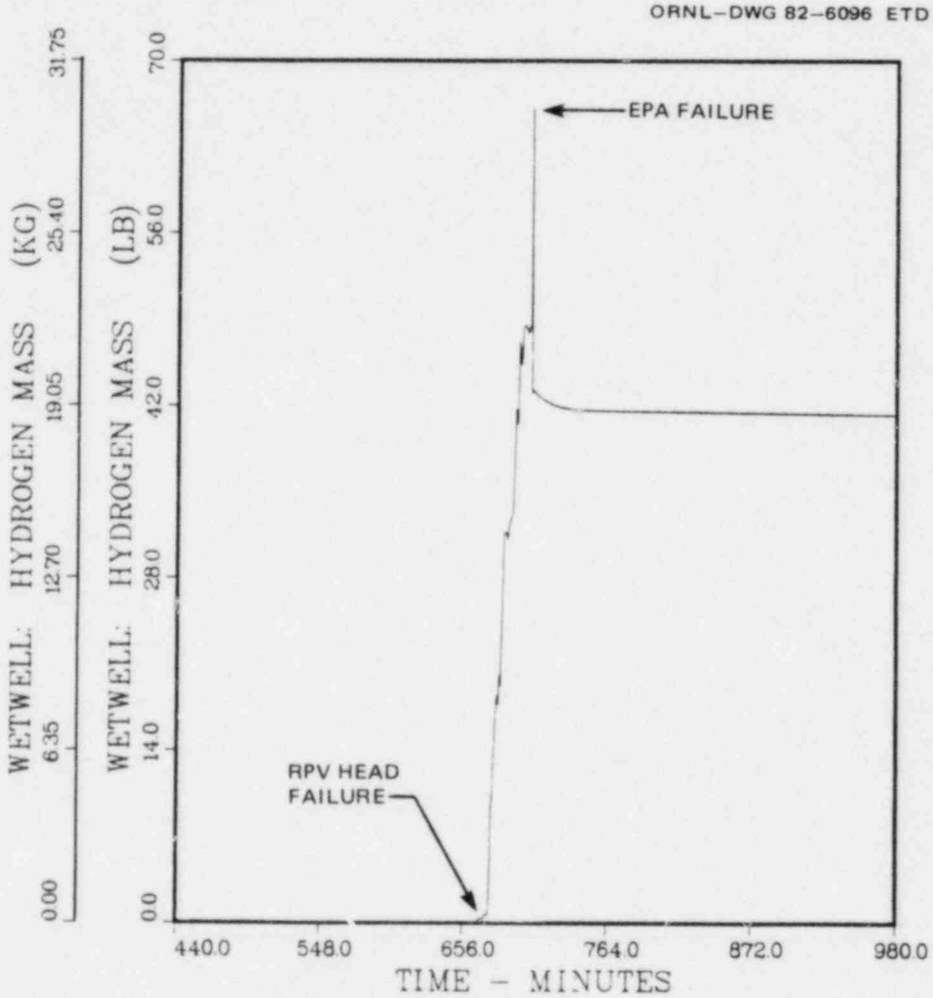

Fig. 6.23. Sequence A wetwell hydrogen mass.

ORNL-DWG 82-6097 ETD

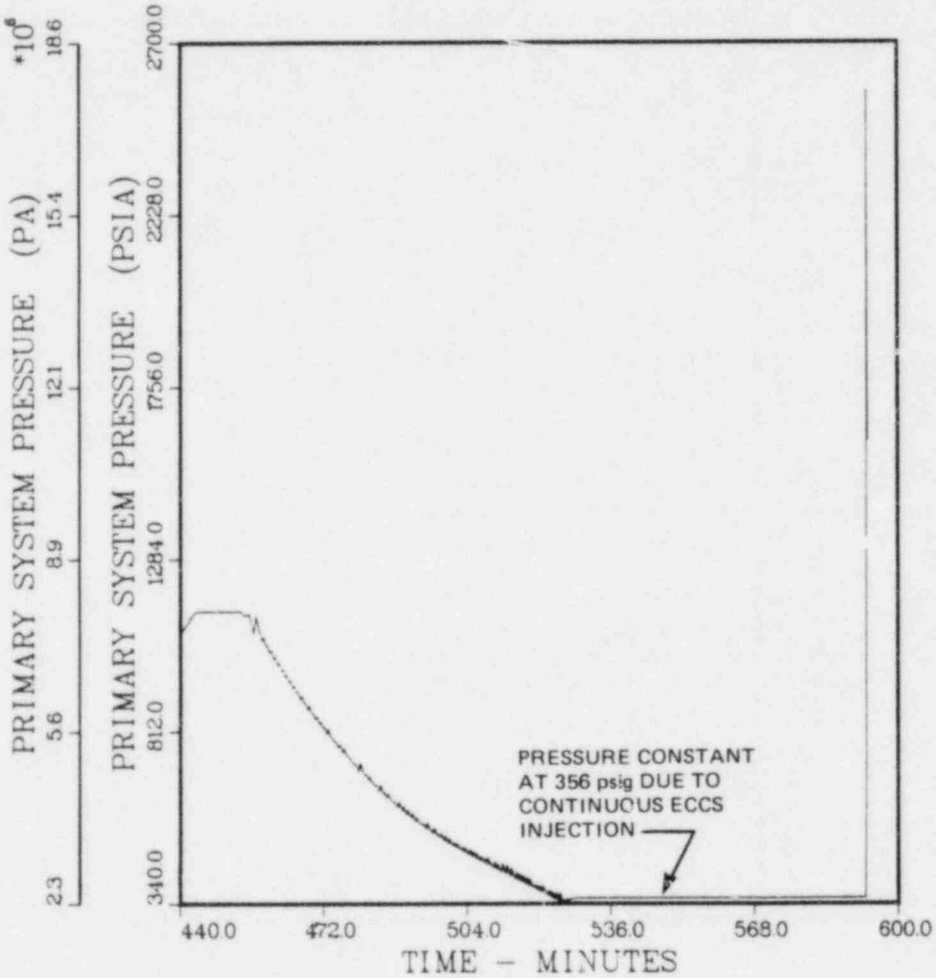

Fig. 6.24. Sequence A1 primary system pressure.

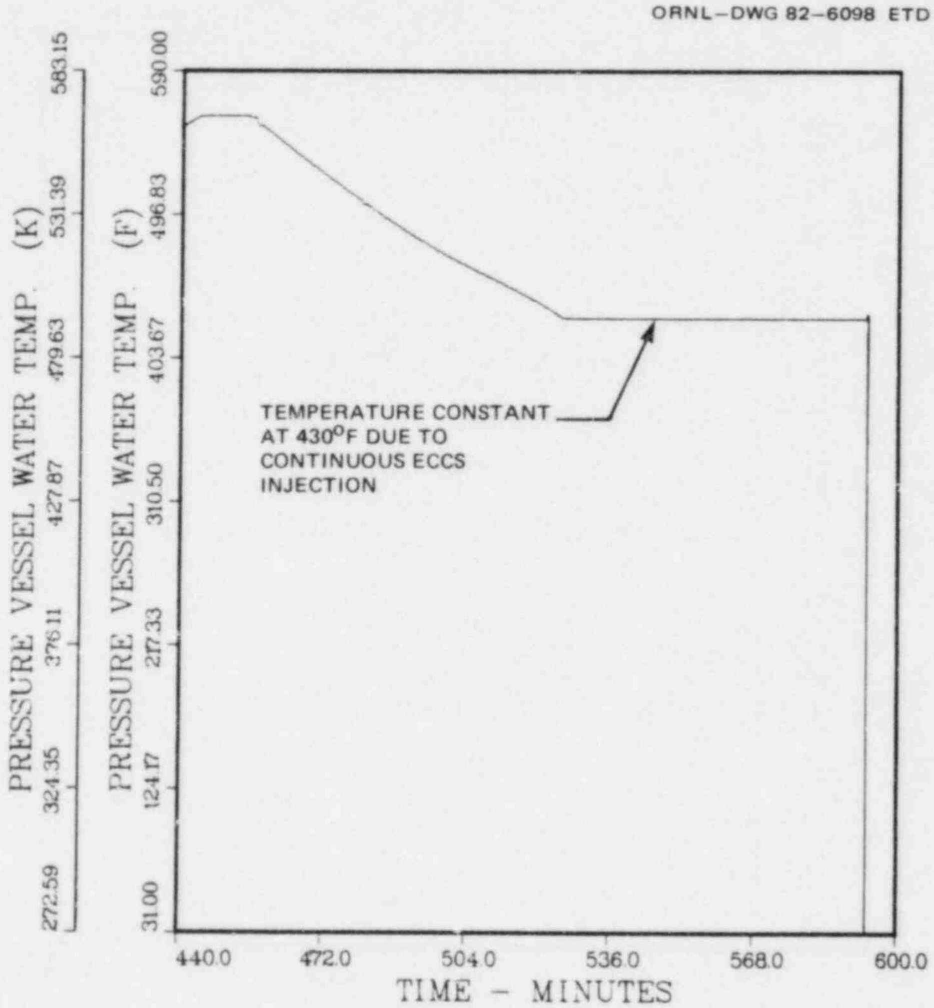

Fig. 6.25. Sequence A1 RPV water temperature.

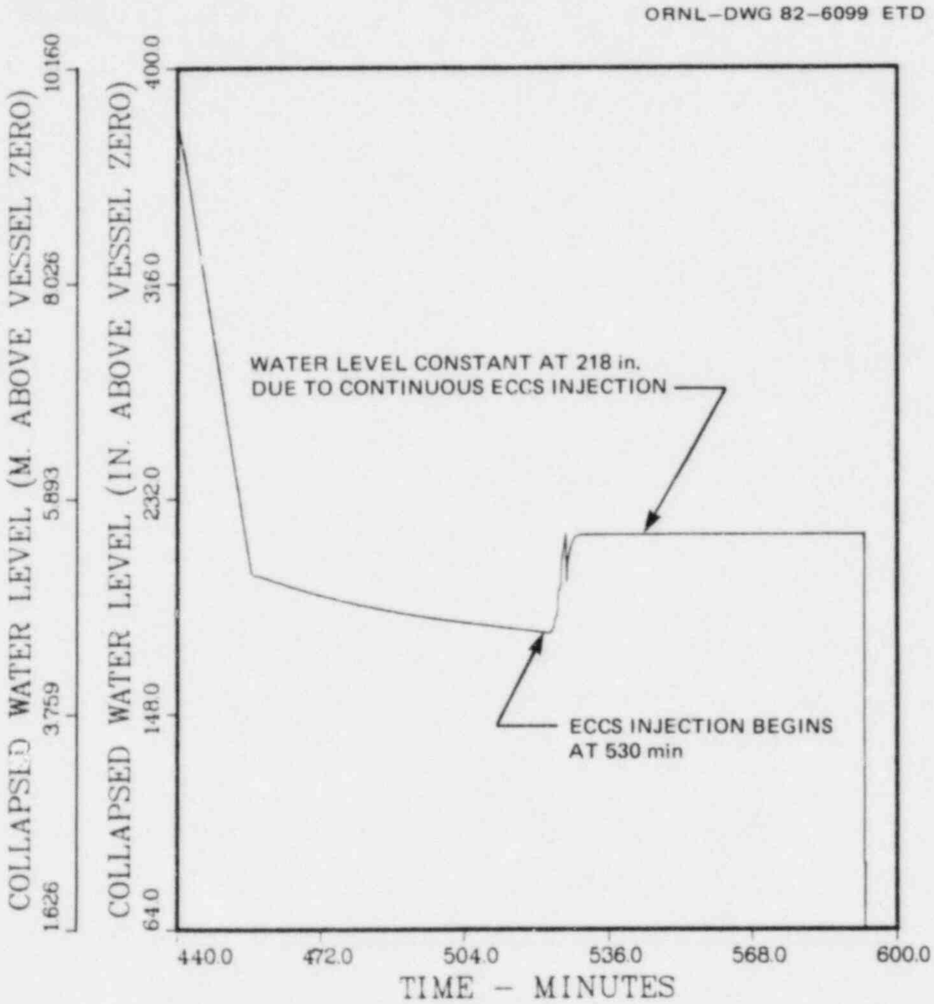

Fig. 6.26. Sequence A1 collapsed RPV water level.

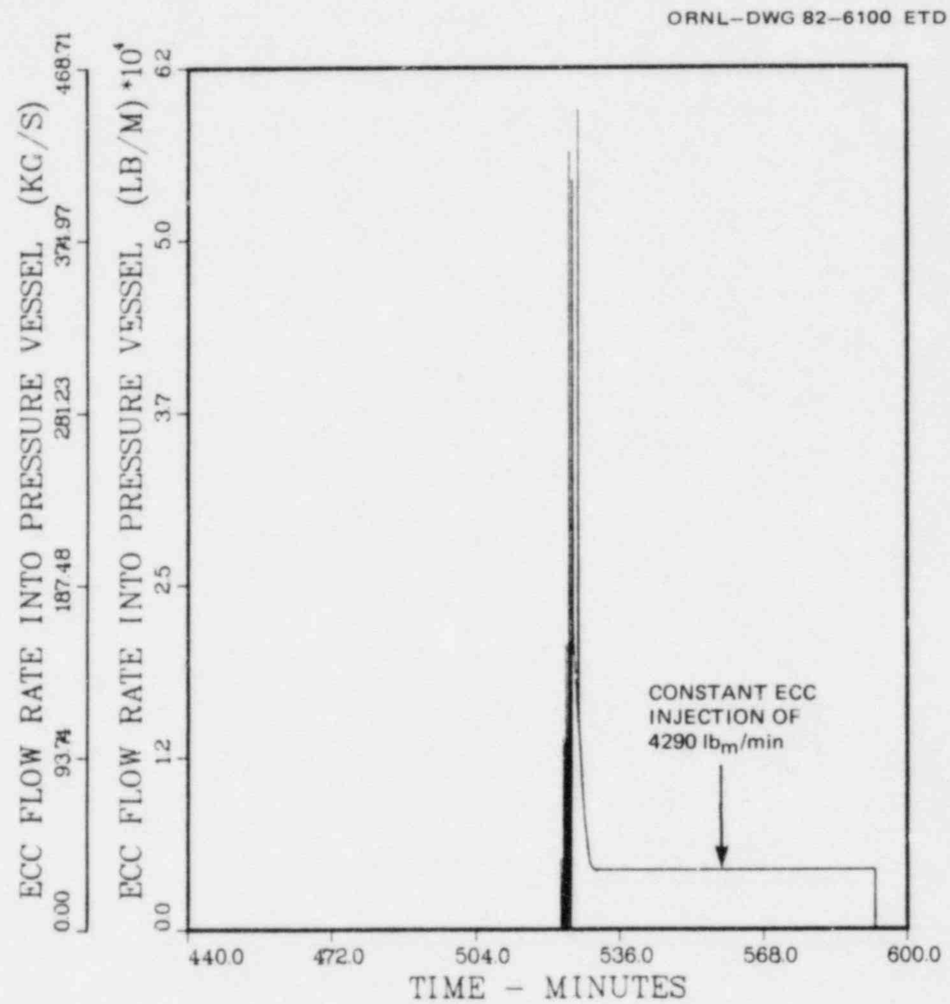

Fig. 6.27. Sequence A1 total ECC injection rate.

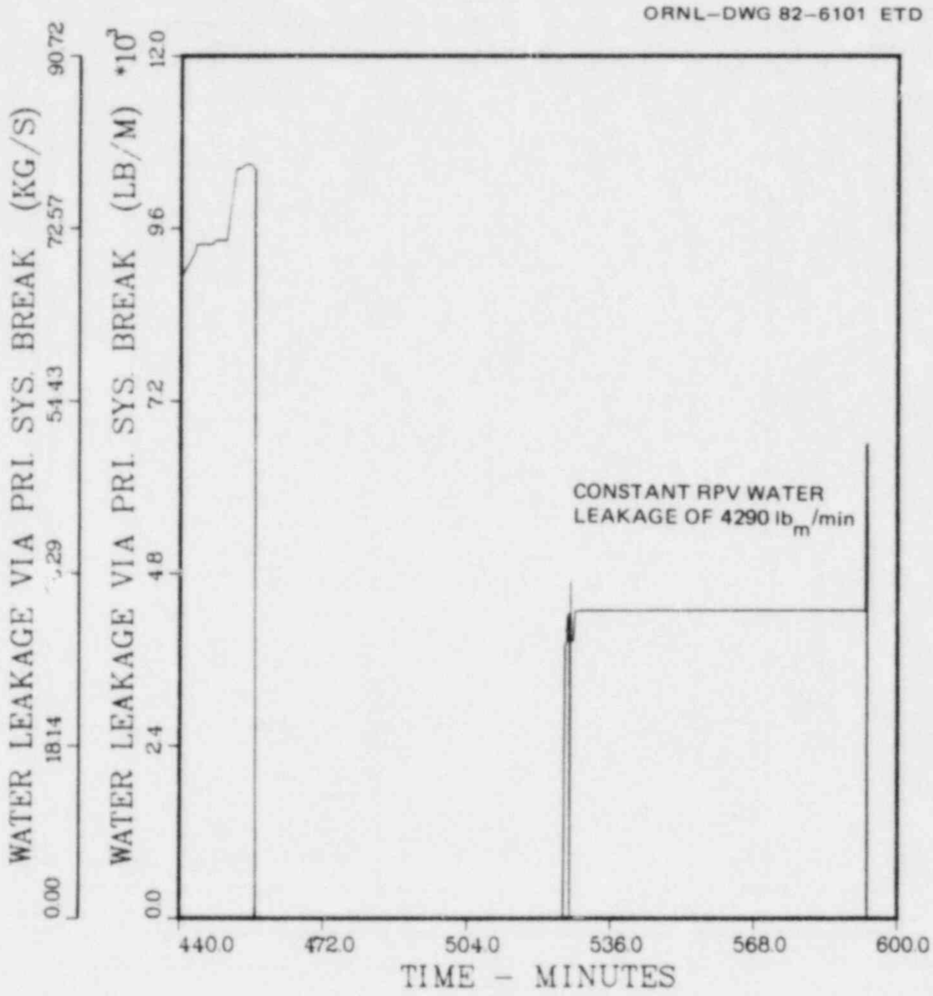

Fig. 6.28. Sequence A1 water leakage via SDV break.

ORNL-DWG 82-5102 ETD

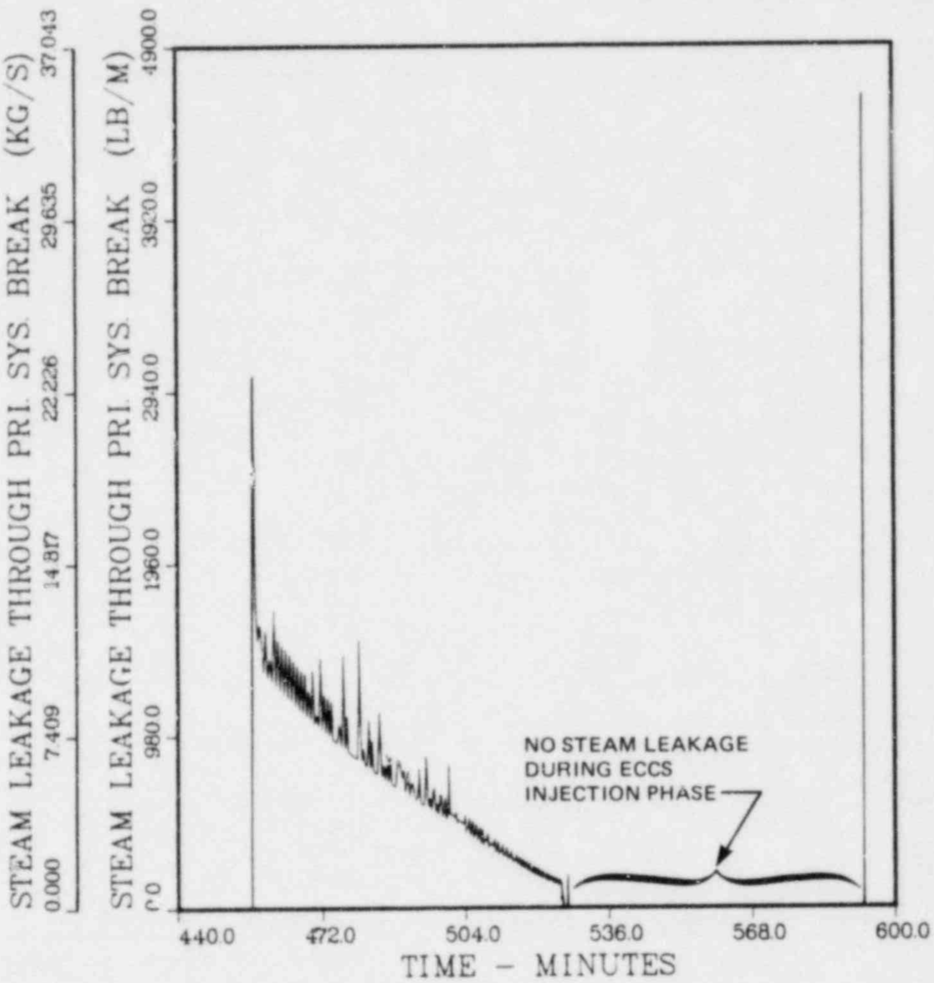

Fig. 6.29. Sequence A1 steam leakage via SDV break.

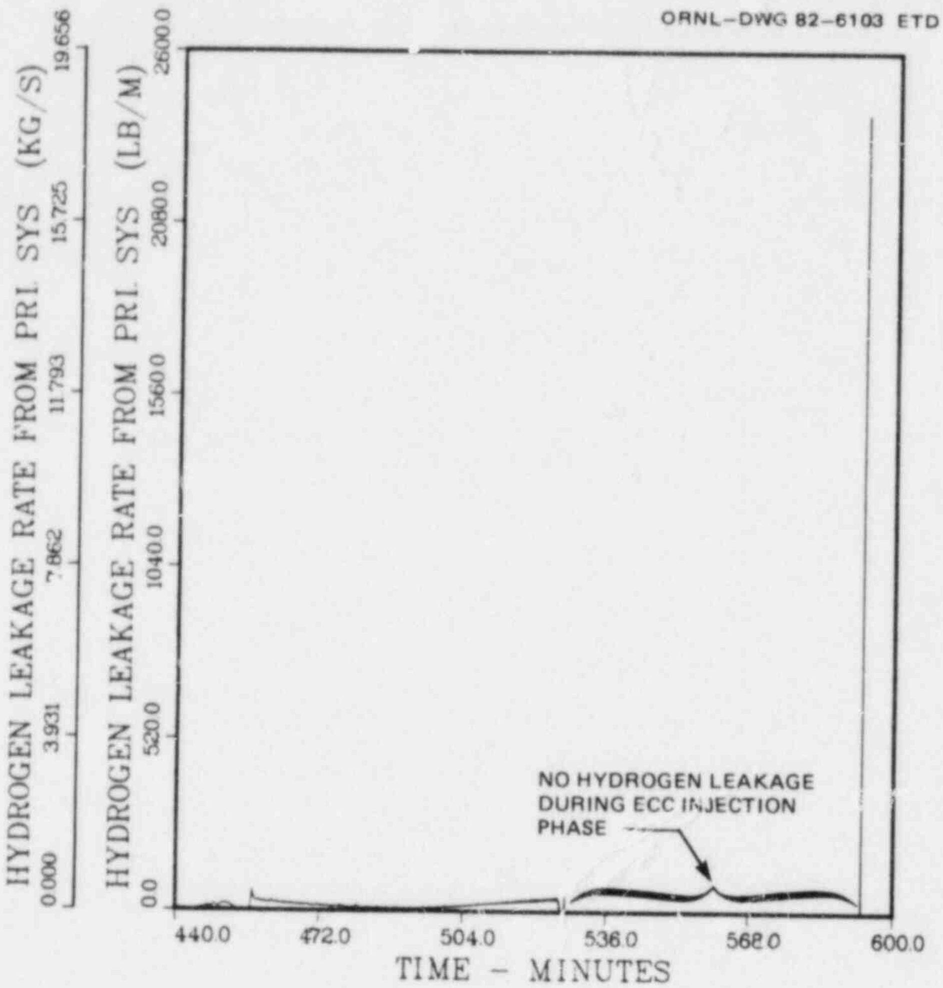

Fig. 6.30. Sequence A1 hydrogen leakage rate via SDV break and SRVs.

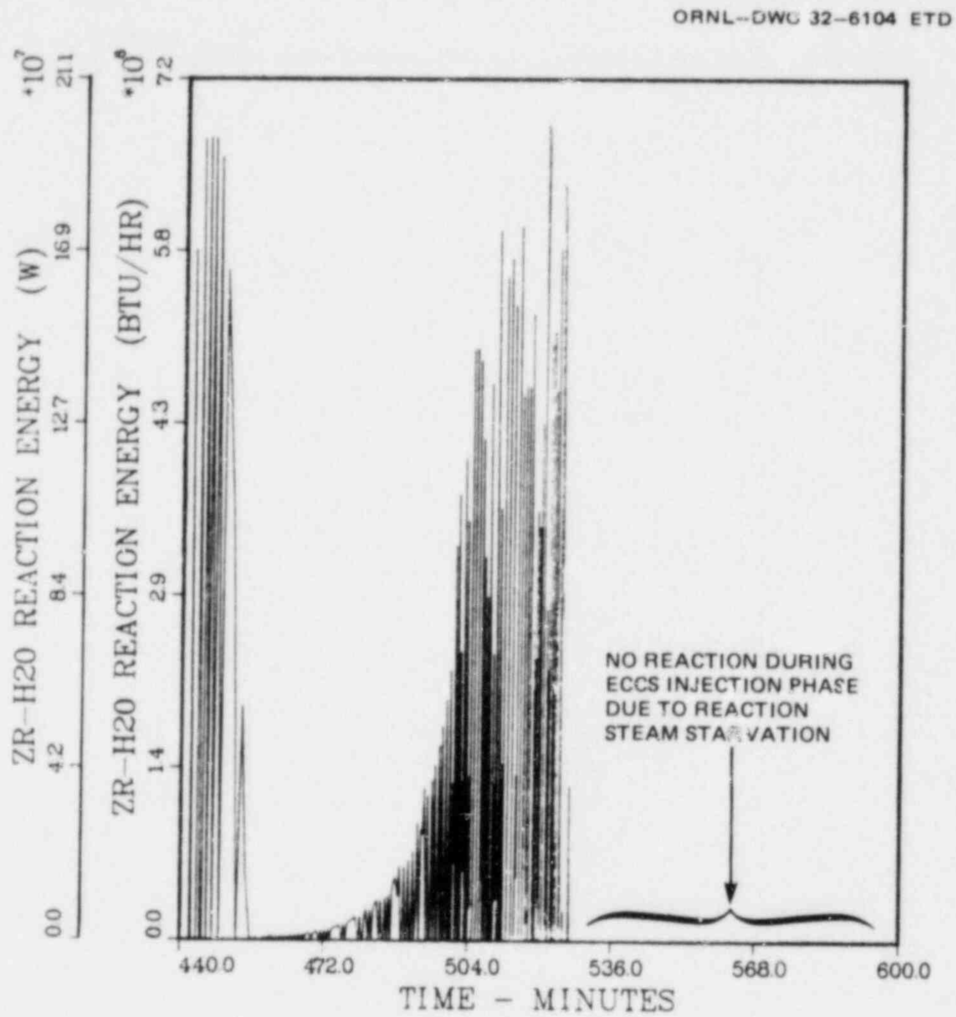

Fig. 6.31. Sequence A1 Zr–H_2O reaction rate.

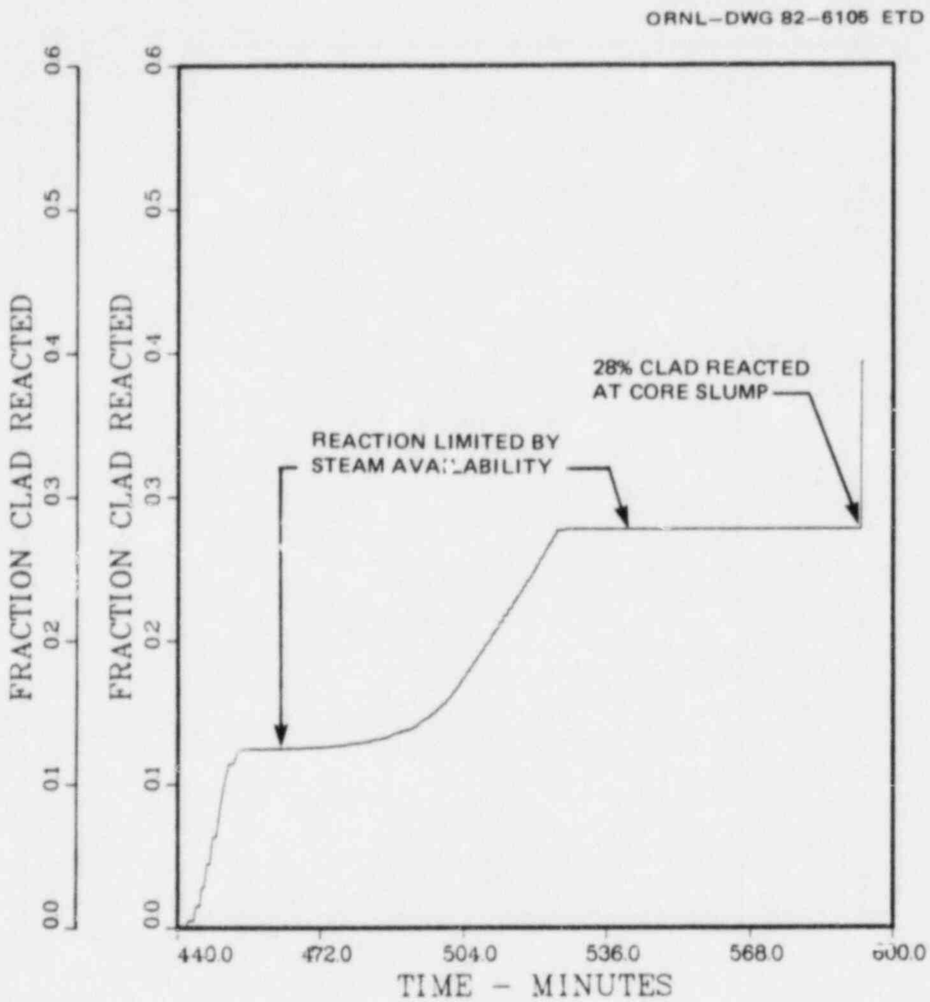

Fig. 6.32. Sequence A1 fraction of clad reacted.

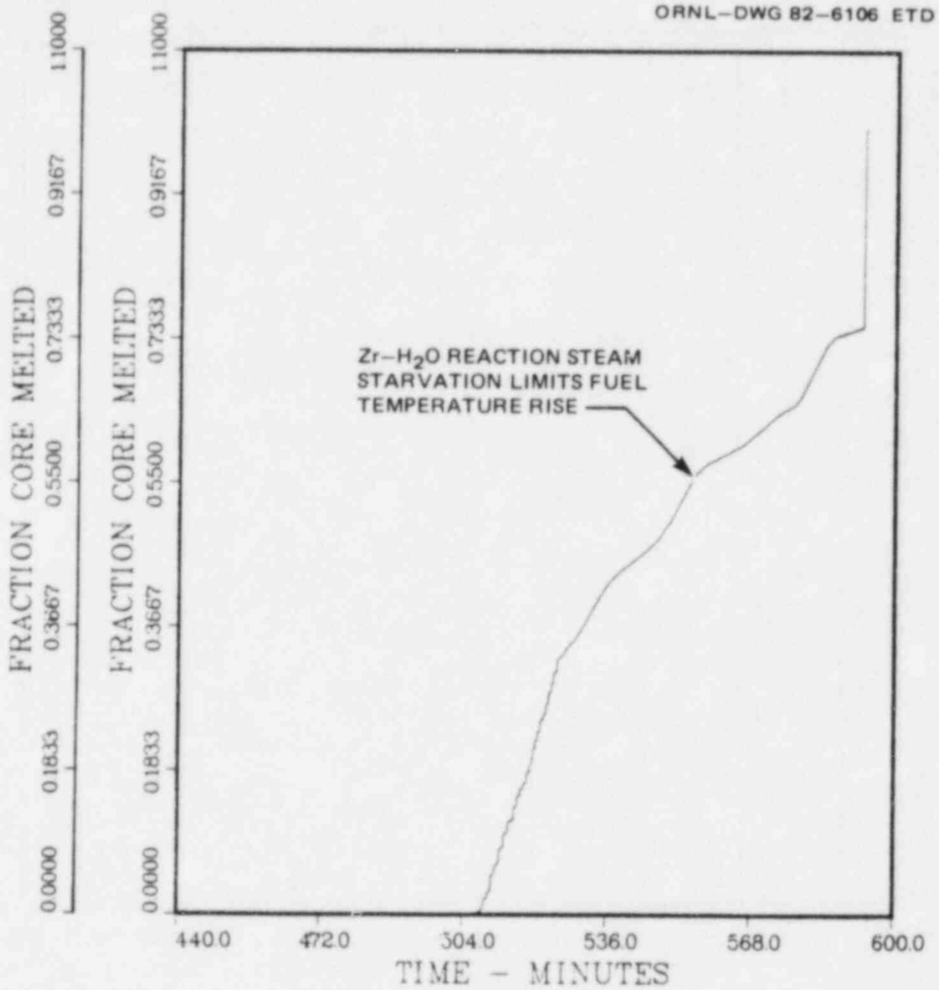

Fig. 6.33. Sequence A1 fraction of core melted.

ORNL-DWG 82-6107 ETD

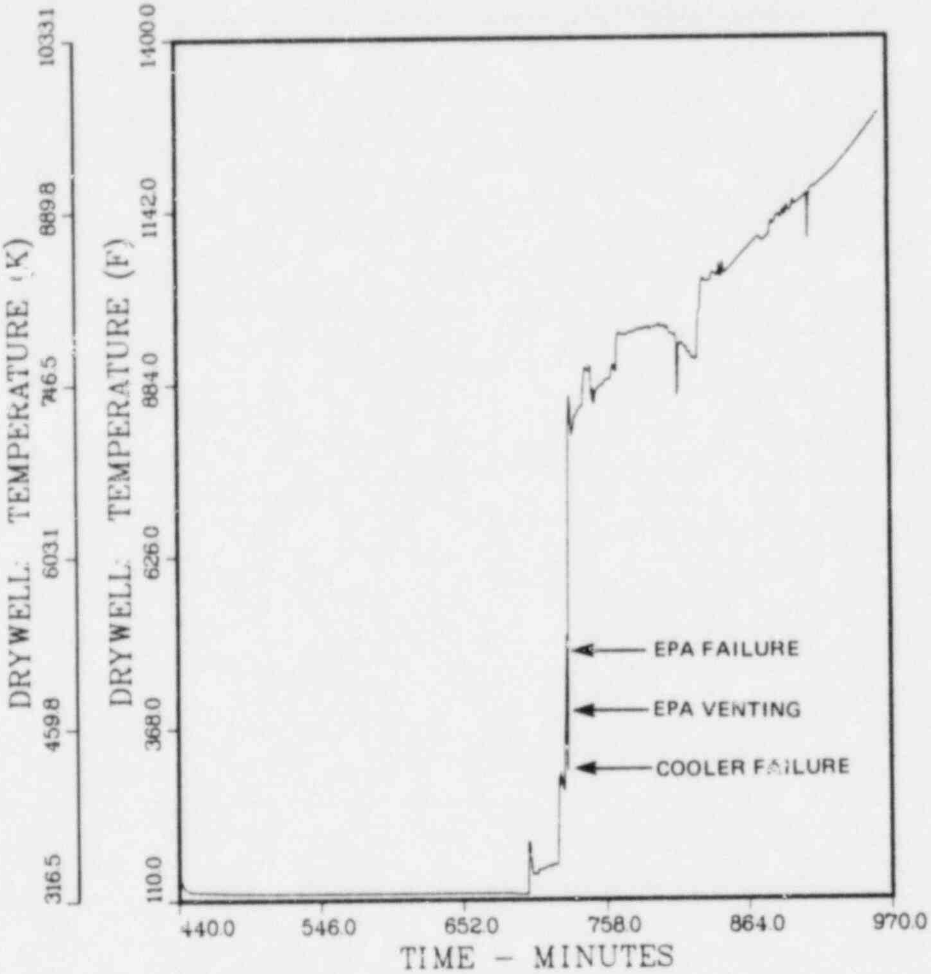

Fig. 6.34. Sequence A1 drywell temperature.

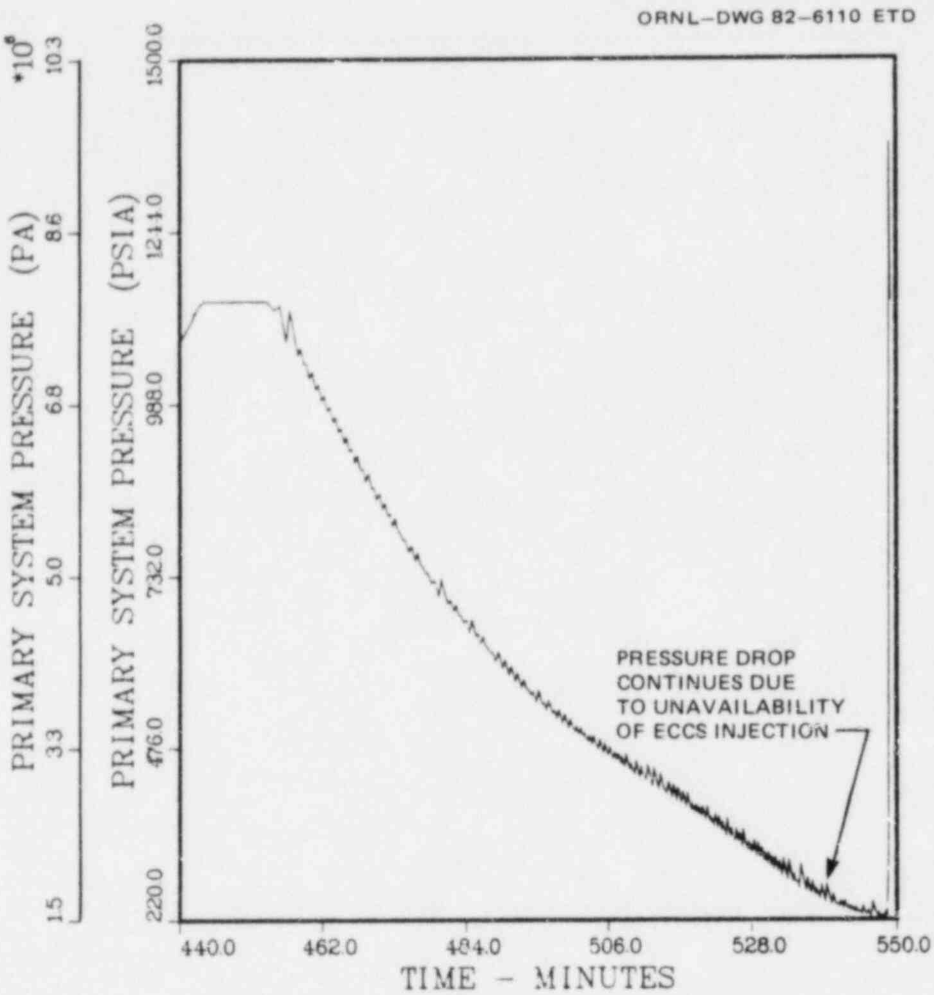

Fig. 6.35. Sequence B1 primary system pressure.

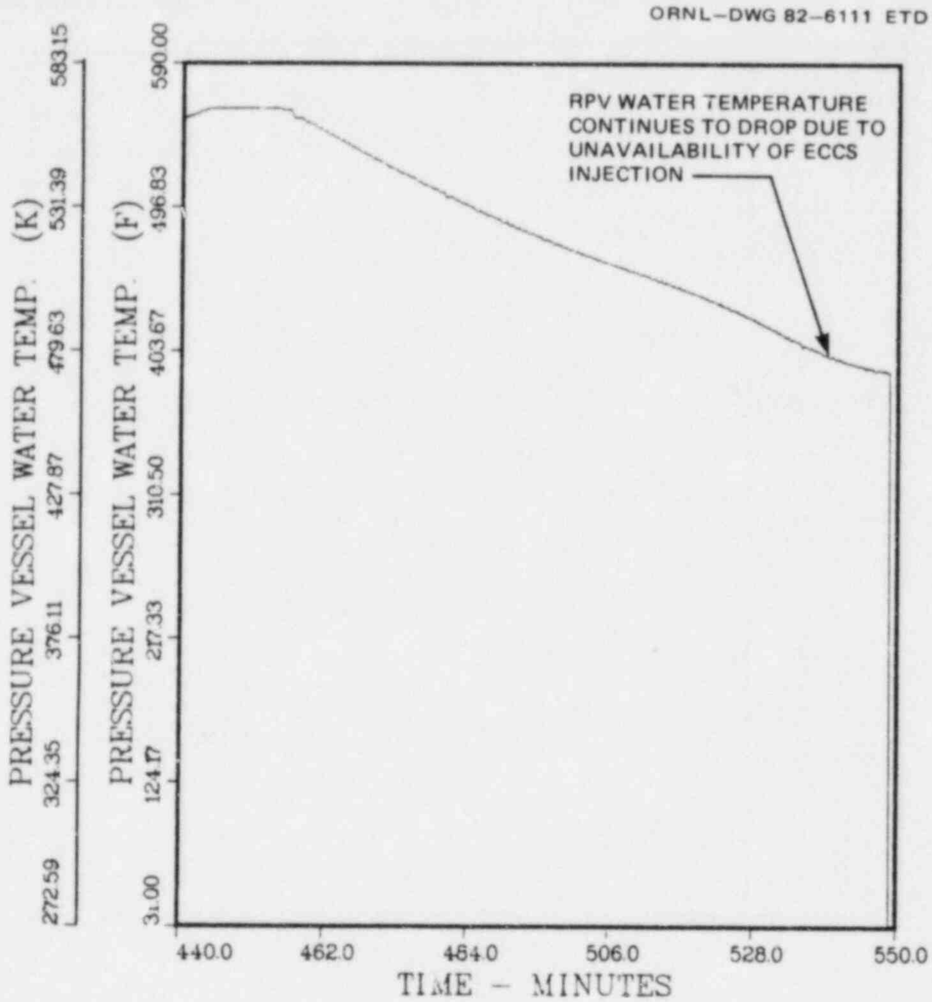

Fig. 6.36. Sequence B1 RPV water temperature.

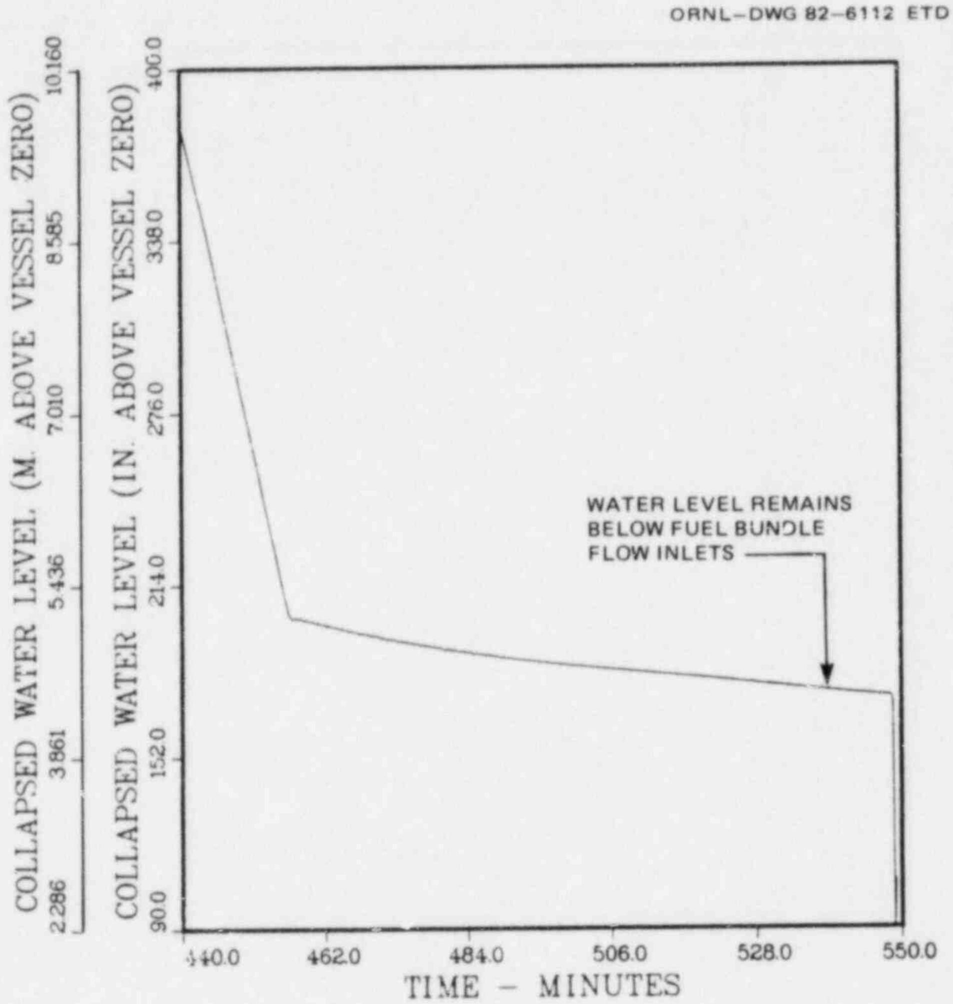

Fig. 6.37. Sequence B1 RPV water level.

ORNL-DWG 82-6113 ETD

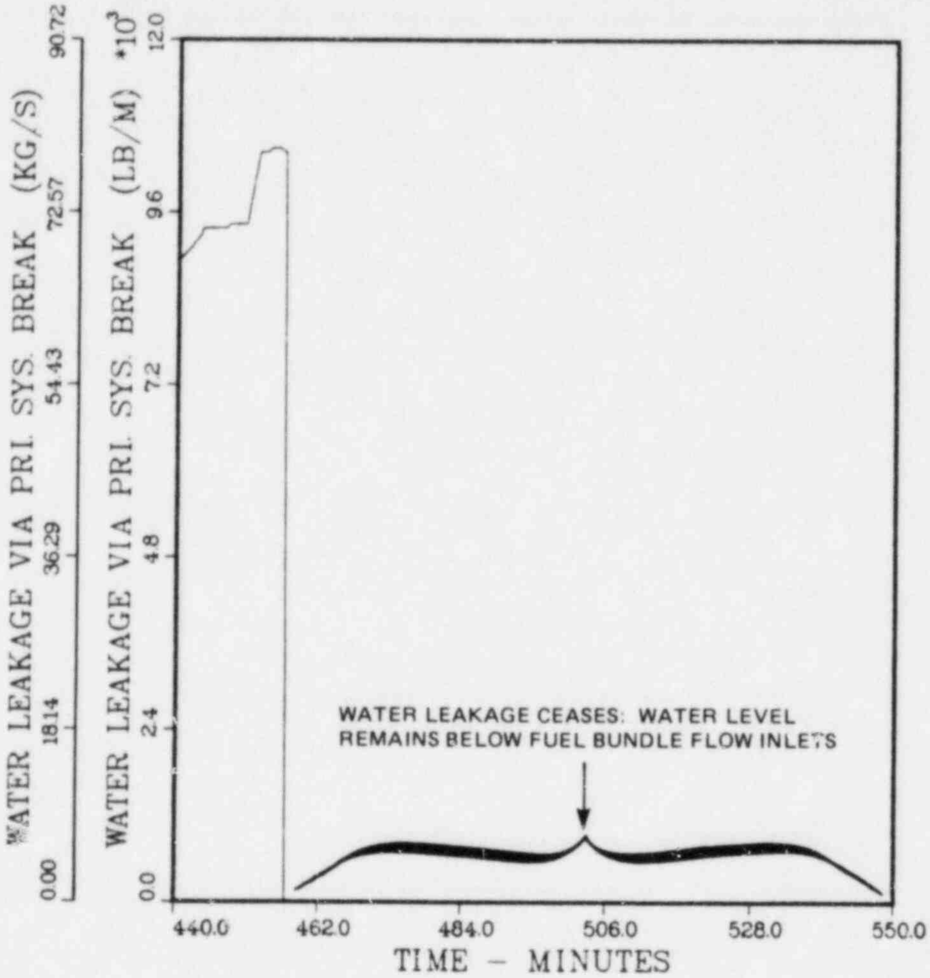

Fig. 6.38. Sequence B1 water leakage via SDV break.

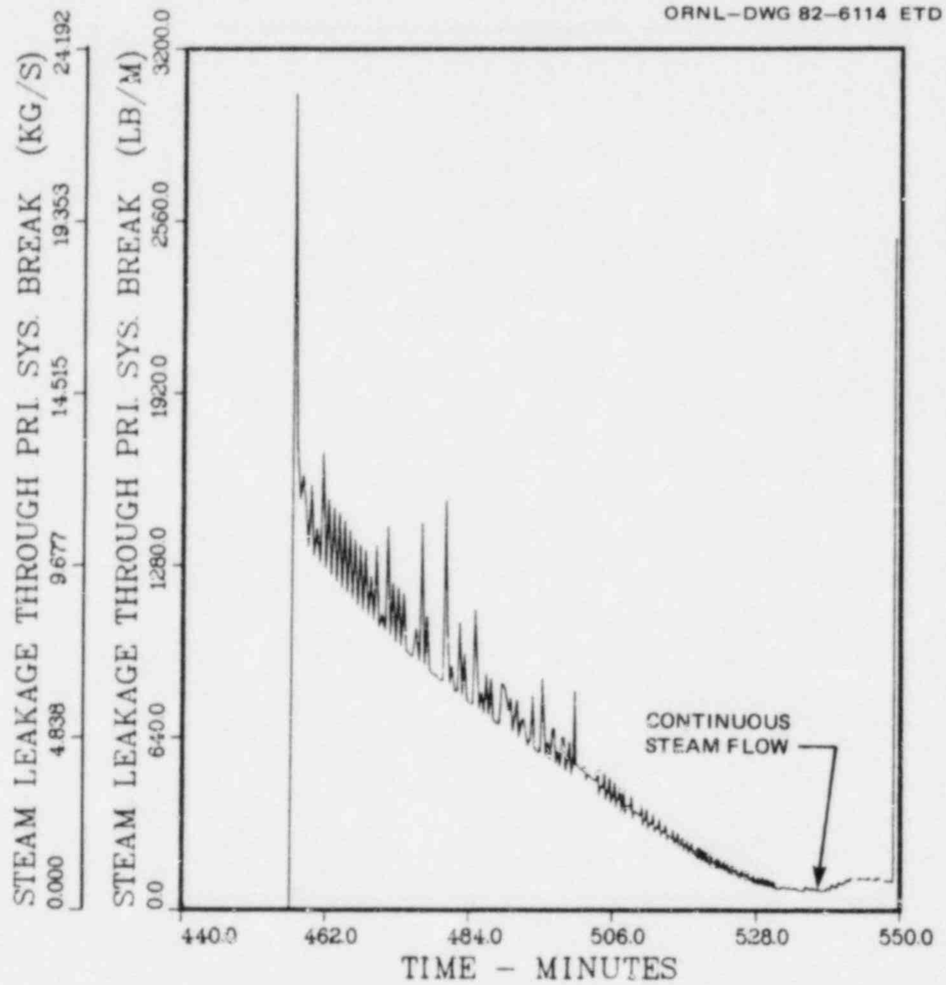

Fig. 6.39. Sequence B1 steam leakage via SDV break.

ORNL-DWG 82-6115 ETD

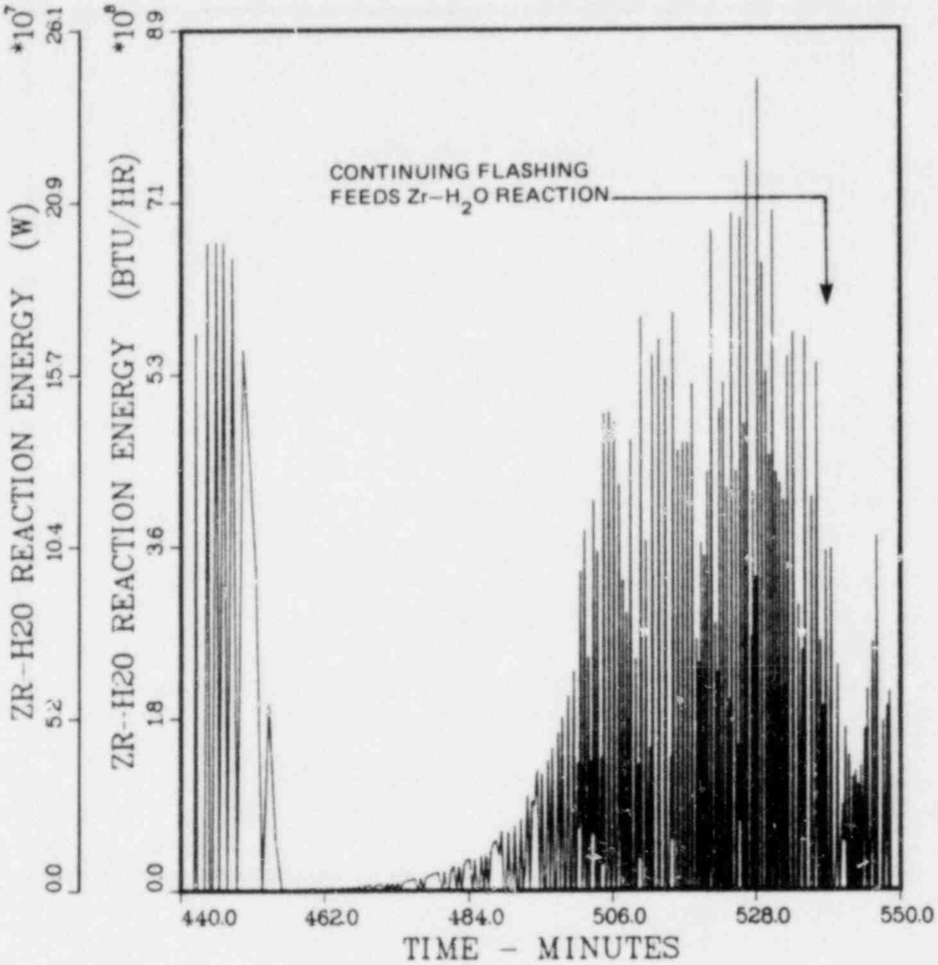

Fig. 6.40. Sequence B1 Zr–H_2O reaction rate.

ORNL-DWG 82-6116 ETD

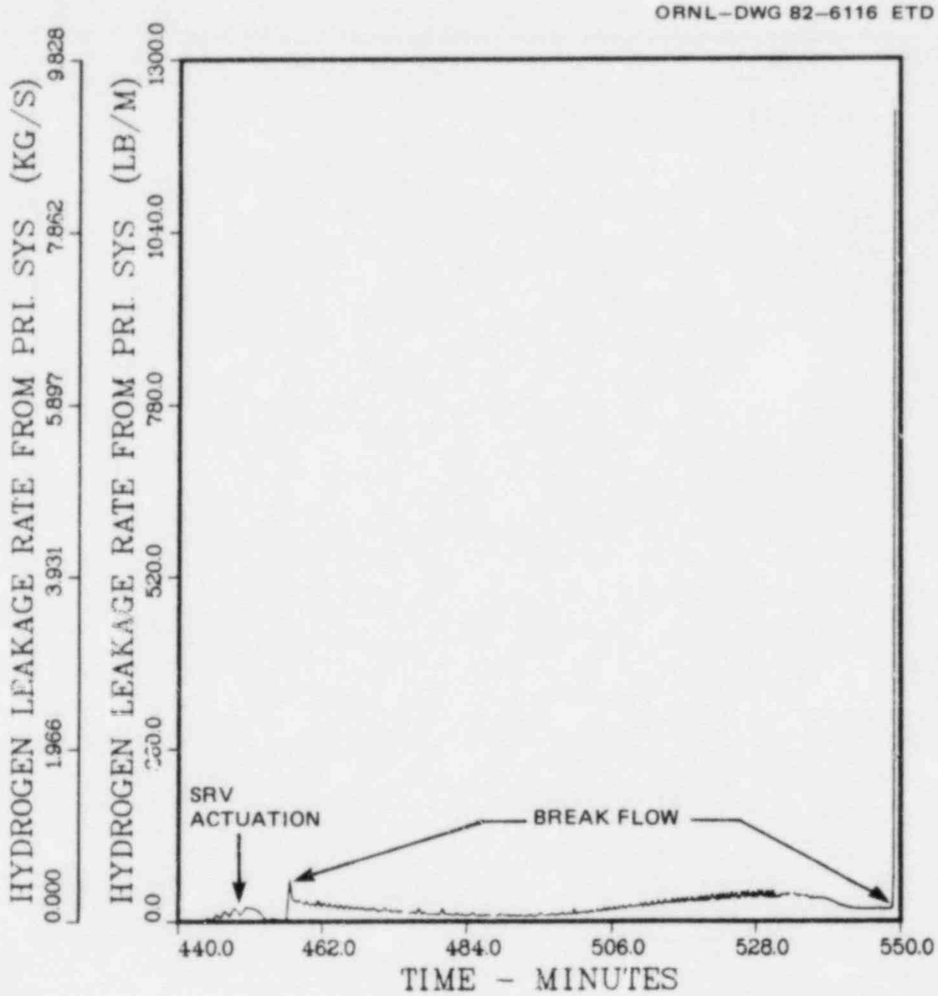

Fig. 6.41. Sequence B1 hydrogen leakage from primary system.

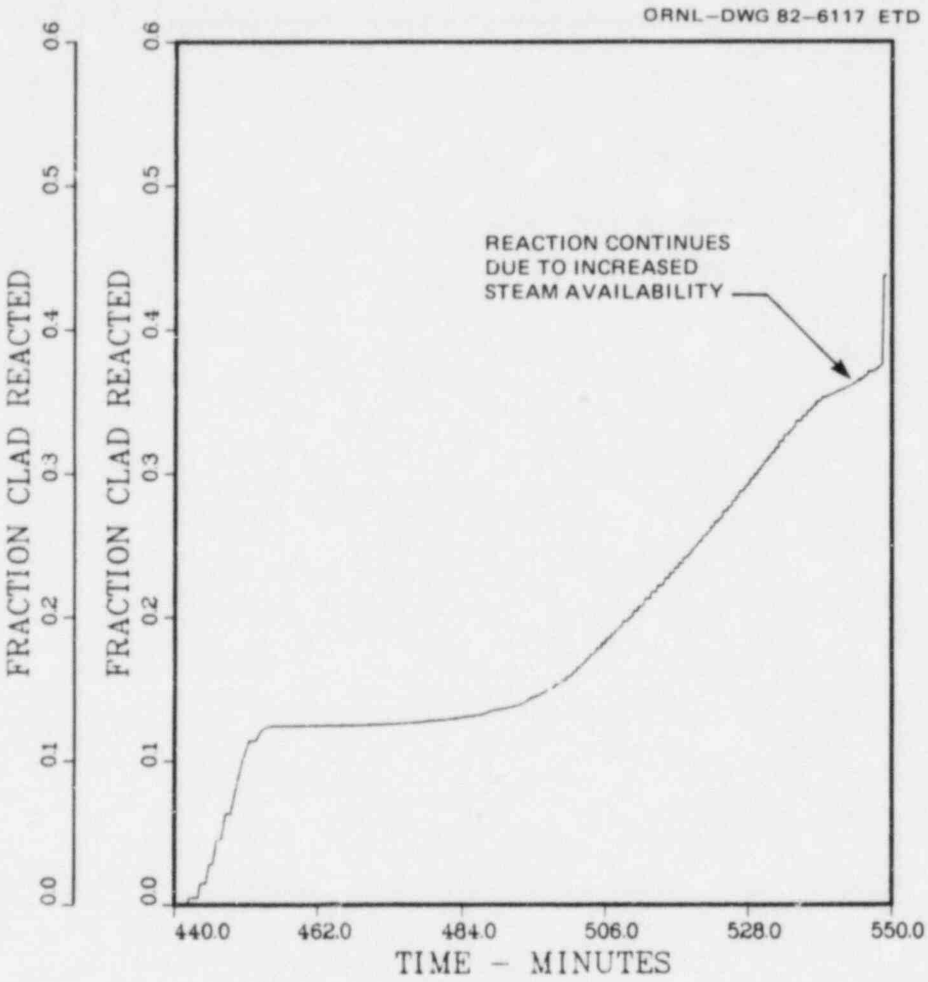

Fig. 6.42. Sequence B1 fraction of clad reacted.

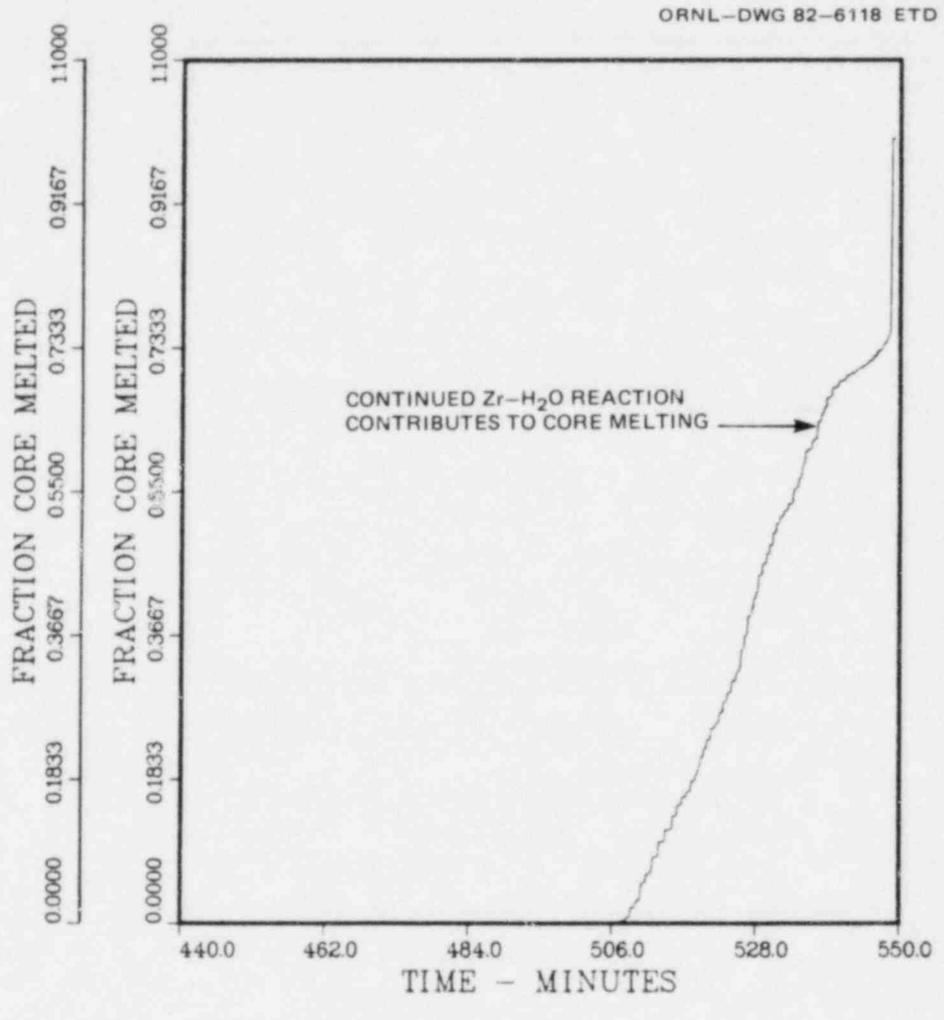

Fig. 6.43. Sequence B1 fraction of core melted.

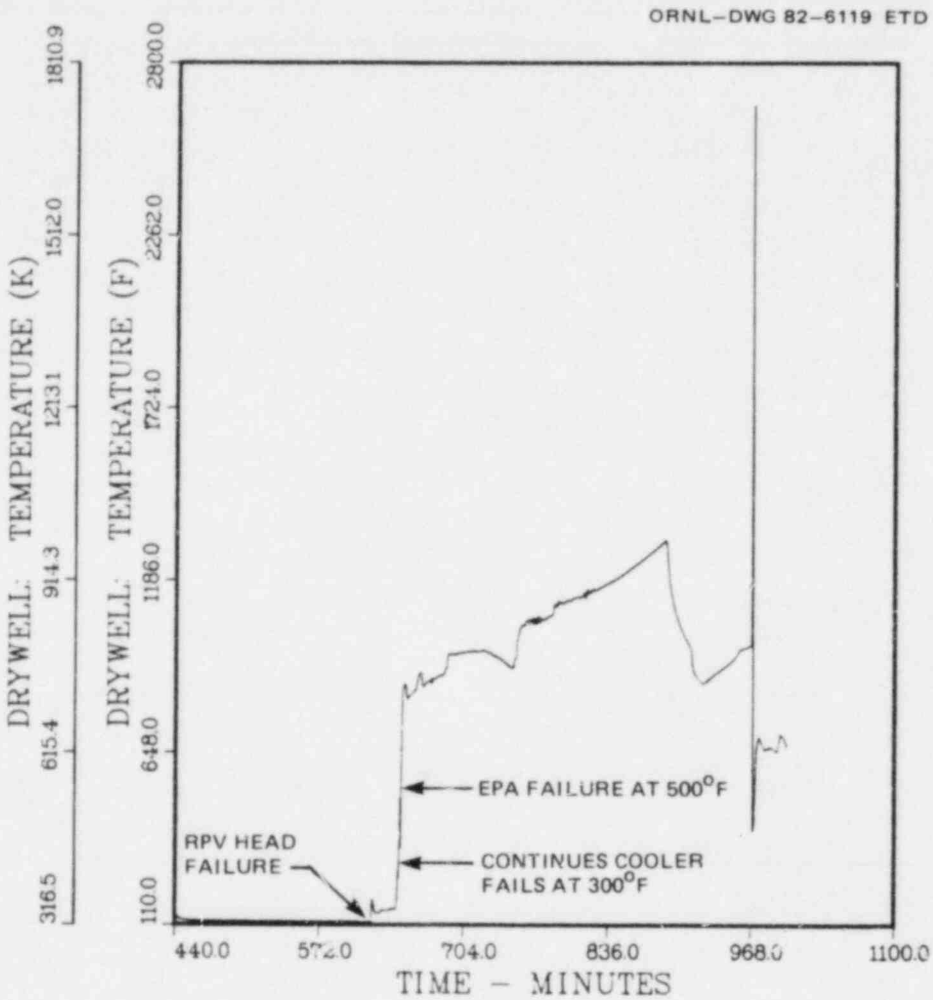

Fig. 6.44. Sequence B1 drywell temperature.

Table 6.1. Design and as-built performance characteristics of BFNP#1 core spray and low pressure coolant injection systems

System	Suction source	Total flow (GPM)	Reactor vessel-to-wetwell pressure differential at which injection begins (psid)	
			Design	As-built
Core spray	Wetwell	12,500 @ 120 psid	289	342
LPCI	Wetwell	40,000 @ 20 psid	295	331

Table 6.2. BFNP#1 scram discharge volume break, no-operator-action accident event timing

Event	Time of occurrence (min)			
	Seq. A	Seq. A1	Seq. B1	Seq. B
Core uncovers	442	442	442	442
Level below fuel bundle flow inlets	456	456	456	456
Coremelt begins	506	506	506	506
ECCS injection begins	530	523		
Core slump	564	593	549	549
Head fails	652	690	610	606
Containment cooler fails	700	727	644	677
EPA venting	707	729	645	680
EPA failure	707	729	646	680

7. IMPLICATIONS OF RESULTS

The purpose of this section is to provide a discussion of the present state of readiness at BFNP to cope with a break in the scram discharge volume (SDV) piping immediately following a reactor scram that cannot be reset. This accident constitutes an SBLOCA outside of containment which occurs in conjunction with a reactor scram. The discussion includes consideration of the available instrumentation, the level of operator training, the existing emergency procedures, and the overall system design.

7.1 Instrumentation

All control room and other plant instrumentation normally available after a reactor scram would be available for operator use during the postulated SDV piping break accident. It should be noted that none of the existing detection and alarm facilities or other monitoring equipment were installed for the purpose of protection against this specific accident. However, as discussed in Sect. 4, several components of the existing instrumention network would be helpful to the operator in his attempts to diagnose the cause of the unusual events pursuant to this accident sequence. The question is whether the existing network is sufficient to ensure that the operator will correctly diagnose the cause within a reasonable amount of time.

It must be recalled that a break in the SDV piping would occur when this piping is pressurized immediately following a reactor scram. Assuming that the reactor scram was caused by an abnormal plant event such as LOSP or loss of condenser vacuum, there would be a natural tendency for the operator to believe that all of the subsequent abnormal responses of his plant monitoring instruments were symptomatic of the plant abnormality that caused the scram. The cause of the scram is clearly indicated to the operator by an illuminated window on a separate panel* reserved for scram alarms at the front of the control room.

It is believed that a high main steam line radiation signal scram would particularly obfuscate the true situation. The high main steam line radiation scram would condition the operator for the receipt of other high-radiation alarms. Indeed, as discussed in the next part of this section, the Emergency Operating Instruction for a high main steam line radiation scram cautions the operator to anticipate high-radiation alarms from the reactor building monitors. Thus, this scram would provide the most severe challenge possible to the question of the adequacy of the existing instrumentation.

The strongest effects of an SDV piping break immediately following a scram from full power would be the release of radioactive saturated steam and water into the reactor building environment and the abnormal effect upon the control rod drive (CRD) mechanisms. As previously discussed, it would not be unreasonable for the operator to assume that the subsequent high reactor building radiation alarms were caused by the fuel failure

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*The "first-out" panel.

associated with a high main steam line radiation scram. Therefore, the presence of reactor building radiation alarms can be discounted as being significant in leading the operator to recognize that a break in the SDV piping has occurred.*

The saturated water leaking from the SDV piping break would fall onto the floor of the reactor building at elevation 565† and drain into the basement via either the floor drains or the open-grated stairwells. The quantity would greatly exceed the capacity of the 150-gpm floor drain sump pumps, and the consequent flooding of the basement rooms would be annunciated in the control room. In addition, the abnormally high flow into the radwaste building from the floor drain sump would almost certainly be noticed by the operator in the continuously manned radwaste building control room.

The saturated steam from the SDV piping break (and the saturated water, to a reduced extent) would cause, one at a time, high-temperature alarms in the control room for most of the 28 monitored locations within the reactor building.

With the evidence of both high temperature and flooding in the reactor building, the operator would certainly be drawn to conclude that there was a significant leakage of a high-temperature fluid into the reactor building. We have estimated in this study that it would take the operator one-half hour to come to this conclusion, again recognizing the obfuscation engendered by the signals accompanying the reactor scram that would have precipitated the SDV piping break.

If it is accepted that the operator would recognize the existence of leakage into the reactor building within one-half hour, then the question arises as to how much longer it would take him to understand that the source of the leakage was the SDV piping. The essential (and only) control room clues would be the abnormal configuration of the control rod position indication and the abnormally high CRD mechanism temperatures.

As discussed in Sect. 4.5, a break in the SDV piping would lead to prolonged high-temperature alarms in the CRD mechanisms, whereas only transient high-temperature alarms are experienced in the usual plant response to a scram. Furthermore, the SDV piping break would continue the excess of pressure on the underside of the control rod drive pistons so that the control rods would be held in an over-travel driven-in position.

There is only one high-temperature alarm in the control room for the CRD mechanisms. The individual temperatures are available on the panels in the rear of the control room, so it would take some effort for the operator to ascertain that <u>all</u> were alarming.

The abnormal control rod position indication would be displayed to the operator at a prominent position central to the control room. It seems probable that he would understand that there was an abnormality in the control rod position indication (over-travel, driven-in) before long.

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*Of course, these alarms would not be discounted if the initiating scram did not indicate the release of radiation from the fuel. However, we are trying the hardest case here.

†See Fig. 4.2.

Thus, a great deal of circumstantial evidence supports the argument that the operator would know that the break was in the SDV piping, and the authors of this study concur in the belief that this recognition would come within one-half hour of the operator's understanding that there was primary coolant leakage into the reactor building. Nevertheless, the primary coolant boundary is extended to the walls of the SDV piping during the period between a reactor scram and the resetting of the scram; there should be a positive control room display of the symptoms of a break in this extended boundary, such as SDV system pressure, in future plant designs.

7.2 Operator Preparedness

The BFNP training simulator does have the capability to model small break loss of coolant accidents (SBLOCAs) outside of containment, and this accident scenario is used in operator training. The available model simulates a break in the 6-inch Reactor Water Cleanup (RWCU) system piping located at the 181-m (593-ft) elevation in the reactor building. The RWCU piping break is assumed to be nonisolatable* with an initial leak rate of 0.126 m^3/s (2000 gpm). This is about four times the initial leak rate for the SDV break accident sequence, and there is no provision at the simulator for varying the break size.

The first control room alarms for the RWCU break sequence as modeled at the simulator are for high radiation in the reactor building. Soon thereafter, high-temperature alarms occur for the vicinity of the RWCU equipment. The correct trainee response is to send a health physics technician to investigate the conditions in the reactor building. After the instructor relays the technician's report, the reactor is depressurized, with reactor vessel level maintained by the CPs.

In summary, operator trainees are taught that the symptoms of a SBLOCA outside containment are high-radiation and -temperature alarms for the reactor building and that the proper action upon confirmation that the break exists and cannot be isolated is to depressurize the reactor vessel. Thus, it is difficult to understand why Emergency Operating Instruction (EOI) No. 15, which covers breaks and leaks outside of primary containment, does not require the operator to depressurize the reactor vessel unless the leakage is of sufficient magnitude that the reactor vessel level cannot be maintained by the high-pressure injection systems. Considering only the HPCI and RCIC systems, this would entail a leak rate in excess of 0.353 m^3/s (5600 gpm). It would certainly be undesirable to continue leakage of primary coolant into the reactor building at rates even significantly less than this. Accordingly, it is recommended that this EOI be modified to require a controlled cooldown after reactor scram for any significant leakage outside containment from a break that cannot be isolated.

For the SDV break sequence studied in this report, it might be assumed that the initiating scram was caused by high-sensed main steam line

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*It is assumed that the inboard RWCU primary containment isolation valve has failed in the open position.

radiation. This eventuality is covered by EOI No. 13, which does require that the operator depressurize the reactor vessel through the primary relief valves at a rate such that the cooldown does not exceed 37.8°C/h (100°F/h). Because the operator would be responding to the requirements of EOI No. 13 before he became aware that there was a SBLOCA outside of containment, it is probable that this depressurization would continue at the authorized cooldown rate. However, at this rate it would take some 4 h to depressurize the reactor vessel. This indicates the need for consideration of a procedure that would unequivocally require a more rapid cooldown and depressurization in the event of a significant break outside of containment that cannot be isolated.

Note that EOI No. 13 cautions the operator to be alert for radiation alarms in both the reactor building and the turbine building as a consequence of a high main steam line radiation situation. This is perfectly proper, but it increases the probability that the high reactor building radiation alarms caused by the postulated SDV piping break would not be recognized as symptoms of a SBLOCA outside of containment by the control room operator.

7.3 System Design

The existing system design provides sufficient instrumentation and equipment to permit plant recovery to a normal shutdown and cooled-down condition with operator action following the SDV piping-break event analyzed in this study. The study did reveal some questionable features of the existing design, which are discussed in the following paragraphs.

It would seem desirable to permit the operator to reset any scram as soon as it has been confirmed that all control rods have been driven into the core. With the existing design, only a few of the various scram signals can be bypassed from the control room, and for the others, the scram cannot be reset until the initiating condition clears. The desirable feature of this design is that it precludes any possibility of the operator attempting a reactor startup while a bona fide scram signal exists. The undesirable feature is that the SDV piping in the reactor building remains pressurized by the primary system until the scram condition clears, which could be many hours in cases such as a scram signal derived from a high-sensed drywell pressure. It is believed that the optimum design would prevent reactor startup until after the scram condition clears but would also permit the operator to align the control rod drive hydraulic system back to a normal configuration* after the control rods have been fully inserted into the core.

As discussed in Sect. 4, high-sensed temperature for any or all of the control rod drive (CRD) mechanisms is alarmed in the control room by a single annunciation. The individual temperatures are indicated on a recorder located in a back-panel area of the control room where they are not

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*That is, with the scram inlet and outlet valves shut, scram accumulators fully charged, and the SDV piping drained and vented. It should be noted that this alignment is necessary if second or third attempts to scram noninserted rods are necessary.

readily accessible to the operator. Because high temperatures are experienced by the CRD mechanisms with higher seal leakage during normal reactor scrams, the control room operator would not be alerted to the fact that all of the CRD mechanisms were in an over-temperature condition unless he took action to check the remote back-panel recorder. An optimum design would provide the operator with a ready indication of the number of CRD mechanisms that were at over-temperature.

Finally, in the accident sequence without operator action discussed in Sect. 3, it was shown that the HPCI and RCIC turbine driven high-pressure injection systems would be rendered inoperable by the reactor vessel flooding induced by the large injection capacity of the condensate booster pumps. This is because there is no installed trip of the condensate booster pumps on high reactor vessel level. Although the Residual Heat Removal (RHR) and core spray pumps were not involved in this accident sequence, they have a combined injection capacity larger than that of the condensate booster pumps and their use would also flood an intact reactor vessel to above the main steam line nozzles unless the operator were very quick.

If the reactor vessel remained depressurized after the actuation of the RHR and core spray pumps, then vessel level could be maintained by these low-pressure injection systems and there would be no need for the HPCI or RCIC systems later. However, many accident sequences, such as the one discussed in Sect. 3, can involve vessel repressurization after an initial depressurization. For this reason, it is recommended that consideration be given to the requirement for high-level trips for the RHR, core spray, and condensate booster pumps.

Appendix A

COMPUTER CODE FOR PERIOD BEFORE CORE UNCOVERY

A.1 Introduction

Temperatures, pressures, and water levels within the reactor vessel and primary containment during the period prior to core uncovery were calculated using the boiling-water reactor-loss of ac power (BWR-LACP) code [developed for the Oak Ridge National Laboratory (ORNL) station blackout investigation]; a listing is provided as an appendix to the station blackout report.[1] To calculate reactor vessel and primary containment response for the scram discharge volume (SDV) break accident sequence, only minor modifications were required: the addition of a calculation of (1) reactor vessel leakage and (2) the flow rate of water injected into the reactor vessel from the condenser hotwell via the condensate and condensate booster pumps.

The reactor vessel leakage is modeled as choked flow of subcooled or saturated water, as illustrated in Lahey and Moody.[2] This model accounts for the increase in mass flow rate of leakage when subcooled water is being discharged. The condensate/condensate booster pump vessel injection flow is calculated from a simple relationship developed by summing the head vs capacity curves of the pumps and subtracting the nonrecoverable pressure drops throughout the condensate/feedwater train [taken to be 1.47 MPa (213 psi) when the plant is operating at 100% load].

Major additions had to be made to the BWR-LACP code to calculate temperatures, pressures, and humidities in the secondary containment. The rest of this appendix is devoted to the secondary containment model.

The secondary containment math model divides the reactor building (Fig. 4.2) into three volumes: (1) the reactor building (from the basement at elevation 519 to the bottom of the refueling floor at elevation 664), (2) the refueling bay (at elevation 664 and above), and (3) the duct work leading from the refueling bay and reactor building to the inlet of the standby gas treatment (SGT) trains. An independent mass and energy balance is performed for each of these three volumes. In addition, a calculation of thermal mixing is performed to estimate conditions within three subvolumes within the reactor building: (1) the elevation-565 floor level (which houses the leaking SDV), (2) the reactor building basement (where the hot water from the SDV leak collects), and (3) a composite subvolume representing reactor building floor levels between elevations 583 and 664.

The most basic assumption of the reactor building modeling is that the temperature and composition of the atmosphere within each volume can be adequately described by a single value that does not depend upon location within the volume. For each volume, a mass balance for each component (i.e., air or water vapor) and an energy balance are updated during the transient after each time step. The temperature of the volume and the partial pressure of each component are then calculated employing the assumption that the gaseous components behave as perfect gases.

The concrete walls of the reactor building have a large heat capacity that significantly influences the building temperature during a long transient such as the SDV break. Because concrete has only a moderate thermal conductivity, the calculation of heat transfer between the atmosphere and wall considers not only the gas-to-wall surface heat transfer coefficient, but also the internal resistance to heat transfer of the concrete. To avoid underestimating the rate of heat transfer into the concrete, the wall is divided into several thicknesses (see Ref. 3 for a complete discussion of the effect of node size on the transient heat-up of slabs).

The temperature of the water collecting in the basement is calculated because this water has a warming and humidifying effect on the basement atmosphere. The model assumes that the water from the SDV leak enters the basement at nearly 100°C (212°F). Heat losses from the water to the basement air and heat losses to the concrete basement floor are accounted for.

A.2 Reactor Building Model Mass Transfer Relations

The flow rate between two points at different pressures is calculated using the following expression, which is valid for incompressible flow and for compressible flow when the pressure drop is a small fraction of the total pressure:

$$W = C^{*}(\rho^{*}\Delta P)^{0.5} ,$$

where

W = mass flow rate,
C = flow coefficient (which is constant for constant friction factor),
ΔP = pressure drop,
ρ = density.

This relation is also used for vacuum breaker or blow-out panel flow by use of logic to modify the flow coefficient.

Infiltration of air into the reactor building (or exfiltration out of the building in the event that building pressure exceeds the outdoor ambient pressure) is calculated as follows:

$$B/B_r = (\Delta P/\Delta P_r)^{0.666} ,$$

where

B = the bulk flow rate into or out of the building,
B_r = the bulk flow rate of air infiltrating into the reactor building measured for pressure difference ΔP_r,
ΔP_r = the absolute value of the pressure drop corresponding to the measured flow rate B_r,
ΔP = the absolute value of the difference between indoor and outdoor pressure.

The 0.666 exponent is used because infiltration flow travels through a variety of small cracks and holes and is therefore a mixture of laminar and turbulent flow. This behavior is similar to that observed for common structures.[4] A value of 1 would apply for the exponent if flow were wholly laminar, and a value of 0.5 would apply for wholly turbulent flow.

The flow exhausted to the atmosphere by the three SGT trains is calculated by reducing the following expression to a second-degree polynomial and solving for the flow:

$$P_i - \Delta P_{sgtc} + \Delta P_{sgtb} - \Delta P_d = Pa ,$$

where

Pa = ambient pressure of outdoor air (input value),
P_i = pressure at the inlet to the SGT air treatment components (calculated by a different segment of the model),
ΔP_{sgtc} = pressure drop of SGT air treatment components (dominated by the filter pressure drops, which are linearly proportional to flow),
ΔP_{sgtb} = pressure increase caused by SGT blower (second-degree polynominal in flow),
ΔP_d = pressure drop of duct work leading from SGT blowers to plant stack (proportional to flow squared).

The amount of steam flowing into the reactor building atmosphere because of flashing of the SDV leakage is given by the following:

$$W_s = W_t \times (h_{sdv} - h_f)/h_{fg} ,$$

where

W_s = mass flow rate of steam flashing from the total SDV leakage W_t,
h_{sdv} = enthalpy of the water leaking from the SDV,
h_f = saturated liquid enthalpy (at building pressure),
h_{fg} = heat of vaporization (at building pressure),

Much of the steam that enters the reactor building condenses on walls and ceilings:

$$W_c = Q_c/h_{fg} ,$$

where

W_c = mass flow rate of condensation,
Q_c = condensation heat transfer rate.

Because a large area is flooded with hot water in the reactor building basement during the SDV break, a calculation of the rate of evaporation is included:

$$W_e = Q_{nc} \times \Delta P_{vp} \times (\text{constant terms}) ,$$

where

W_e = mass flow rate of evaporation,
Q_{nc} = heat transfer from basement water to basement air (independent of evaporation),
ΔP_{vp} = driving potential for evaporation: equal to vapor pressure of basement water minus average water vapor pressure in basement air.

The expression for evaporation follows the approach outlined in Kreith.[5] The driving potential for evaporation is the difference between the vapor pressure of the water and the average vapor pressure in the atmosphere (i.e., no evaporation occurs if the basement air is at 100% humidity). The rate of evaporation is dependent on the convective air currents induced because the water is warmer than the air.

A.3 Reactor Building Model Heat Transfer Relations

The coefficient for convective heat transfer from the building atmosphere to building surfaces such as walls and ceilings is given by the following relation from Kreith:[5]

$$h = \text{constant} \times \Delta T^{0.333} ,$$

where

h = convective heat transfer coefficient,
ΔT = absolute value of the difference between wall surface and average atmosphere temperature.

The coefficient for heat transfer caused by condensation from the building atmosphere to the building surfaces is the same as that used by the MARCH code:[6]

$$h_c = 12.0 - 0.2 \times R \qquad (\text{for } R > 20) ,$$

$$= 66.75/R^{0.707} \qquad (\text{for } R \leq 20) ,$$

where

h_c = condensation heat transfer coefficient,
R = ratio of mass of air to mass of water vapor in reactor building atmosphere.

This coefficient is applied to the difference between wall surface and dewpoint temperatures to calculate the rate of heat transfer to the wall.

A.4 List of Plant Response Code BWR-LACP with Modifications for Application to SBLOCA-Outside-Containment Analysis

(This listing follows the references at the end of this appendix.)

References

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

2. R. T. Lahey, Jr. and F. J. Moody, *The Thermal Hydraulics of a Boiling Water Nuclear Reactor*, American Nuclear Society, p. [illegible] (1977).

3. S. J. Ball, "Approximate Models for Distributed-Parameter Heat Transfer Systems," *ISA Transactions* 3, 38–47 (1964).

4. American Society of Heating, Refrigeration, and Air Conditioning Engineers, *Handbook of Fundamentals* (1977).

5. Frank Kreith, *Principles of Heat Transfer*, International Textbook Company (1966).

6. R. D. Wooton and H. I. Avci, *MARCH Code Description and Users Manual*, BCL/NUREG/CR-1711 (October 1980).

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

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

```
*  ENERGY BALANCE
*  HEAT TRANSF RATES TO METAL AND ATMOSPHERE NEGLEGIBLE
*  WITH RESPECT TO MASS-MIXING ENERGY TRANSFER
    CMHWI=INTGRL(0.,XQI*(FSVI*WSSV*HST+WSTE*FTEI*HSTE)+E1I+E2I)
      E1I=(TLX-32.)*WEQL+(TRX-32.)*WEQR+WML*(TL-TI)+WMR*(TR-TI)
      E2I=FWI*WCSPG*HFGSP+FDI*WDSPW*HD-FSI*WSSPW*(TI-32.)-WEI*1105.
      TLX=TL*COMPAR(LWL,LWI)+TI*COMPAR(LWI,LWL)
      TRX=TR*COMPAR(LWR,LWI)+TI*COMPAR(LWI,LWR)
*
*  CALC OF MIXING FLOWS(THEY DID NOT APPEAR IN THE MASS BALANCE BECAUSE
*  ZERO NET MASS TRANSFER IS ASSUMED
      WML=KMIX*SQRT(ABS(TL-TI))
      WMR=KMIX*SQRT(ABS(TR-TI))
*
*
*  TEMPERATURE CALCULATION ASSUMES CONSTANT SPECIFIC HEAT=1.0 FOR WATER
      TI=32. + ((TWIO-32.)*MWIO + CMHWI)/MWI
*
 ENDMACRO
*
*
     INITIAL
*
*----------INITIALIZATION FOR REACTOR VESSEL CALCULATION---------------
*
*                          CONSTANTS FOR REACTOR MODEL
*
*  ACOP=CORE OUTLET PLENUM FLOW AREA(FT**2)
*  ACOR=CORE FLOW AREA(FT**2)
*  ART=RISER TUBE FLOW AREA(FT**2)
*  CPO=PRE-TRIP CORE POWER(TOTAL) (MWTH)
*  EQXO=FRACTION OF WAY TO SATURATION THAT STEAM CONTACT RAISES
*       INJECTION WATER IF DC LEVEL IS AT LEVEL OF JET PUMP SUCTION
*  FFLASH=FRACTION FLASHED/SEC PER BTU/LB ABOVE SATURATION
*  HCIO=INITIAL CORE INLET ENTHALPY(BTU/LB)
*  HINJIN=ENTHALPY OF INJECTION FLUID(BTU/LB)
*  JETPMP=EXTRA CORE FLOW ADDED BY JET PUMP EFFECT(LB/SEC) AT 20PC SPEED
*  LBOT=HEIGHT OF STM SEP BOTTOM(FT)
*  LDCO=INITIAL DOWNCOMER LEVEL(FT) (HEIGHT ABOVE BOT. OF ACT. FUEL)
*  LHEDER=HEIGHT OF FW HEADER ABOVE BOAF(FT)
*  LOP=AVG. LENGTH OF CORE OUTLET PLENUM(FT)
*  LRT=AVG. LENGTH OF RISER TUBES(FT) (STANDPIPES)
*  PCOR=CORE HEAT TRANSFER PERIMETER(FT)
*  PO=INITIAL REACTOR VESSEL PRESSURE(PSIA)
*  RCICMX=NOMINAL RCIC FULL FLOW(LB/SEC)
*  TAULEN=STABILITY TIME CONSTANT FOR REGION AVERAGE HEAT FLUX CALC(SEC)
*  TCFUEL=TIME CONSTANT TO ACCOUNT FOR RESIDUAL HEAT IN CORE FOR
*   INITIALIZATION CLOSELY FOLLOWING SCRAM
*  TO=TIME AFTER TRIP THAT TRANSIENT INITIATES(SEC)
*  TLEAK=INITIATION TIME FOR SMALL BREAK OUTSIDE OF CONTAINMENT(SEC)
*  TMANOP=TIME DELAY BETWEEN START OF RUN AND OP MANUAL CONTROL
*  VOLP=LOWER PLENUM VOLUME(FT**3)
*  V1DC=VOLUME OF DOWNCOMER BETWEEN BOAF AND PUMP DIFFUSER EXIT(FT**3)
*  WINJO=INITIAL INJECTION FLOW(LB/SEC)
*
*  REFERENCE PARAMETERS FOR NATURAL CIRC LOSS COEFF CALCULATION
*  CPR=RATED CORE POWER(MWTH)
*  HREF=REFERENCE CORE INLET ENTHALPY(BTU/LB)
*  LDCR=REFERENCE DOWNCOMER LEVEL(I.E. NORMAL LEVEL AT STM SEP MIDDLE)
*  PRELR=REFERENCE RELATIVE CORE POWER
*  PRR=REFERENCE REACTOR VESSEL PRESSURE
*  RHOO=DENSITY OF STEAM(LB/FT**3) IN R.V. AT INITIAL PRESSURE=PO
*  WGUESS=GUESS ON FLOW(LB/SEC) FOR ITERATIVE SOLUTION FOR INITIAL FLOW
*  WREF=REFERENCE FLOW(LB/SEC)
```

```
*  XREF=REFERENCE FRACTIONAL QUALITY
*
*  PARAMETERS FOUND IN FUNCTION SUBPROGRAMS
*  VDIFF=JET PUMP VOL BELOW BOAF(FT**3)
*  VFREE=TOTAL STM. VOL. IN R.V.(LESS LOWER PLENUM,CORE,CORE OUTLET PLENUM
*  AND STM. SEPARATORS) AND MAIN STEAMLINES TO ISOLATION VALVES(FT**3)
*  VJET=VOL BETWEEN JP SUCTION AND BOAF(FT**3)
*  VANN=VOL BETWEEN TOP OF CORE OUTLET PLENUM AND JP SUCTION(FT**3)
*  VSSOP=VOL BETWEEN TOP OF CO PLENUM AND BOTTOM OF STM SEP(CU.FT.)
*  WRATED=STM. FLOW THUR 1 SRV (LB/SEC) WHEN PRESSURE=PRATED (PSIA)
*  XKVGJ=EMPIRICLE CONSTANT USED IN CALCULATION OF DRIFT VELOCITY
*  XL1=0=ZERO POINT FOR DC HEIGHT(FT):  COINCODES WITH BOAF
*  XL2=HEIGHT AT JET PUMP SUCTION(FT)
*  XL3=HEIGHT AT TOP OF CORE OUTLET PLENUM(FT)
*  XL4=HEIGHT AT BOTTOM OF STEAM SEPARATORS(FT)
      CONSTANT  ACOR=82., ACOP=234., ART=42.3
      CONSTANT  CPO=3293., EQXO=1., FFLASH=.001
      CONSTANT  LHECSR=23.4, LOP=5.58, LRT=10.0
      CONSTANT LBOT=24.85
      CONSTANT  PCOR=5518., PRATED=1095.
      CONSTANT  TAULEN=.75, TCFUEL=9.5
      CONSTANT  VFREE=13000., VOLP=3350., V1DC=192.,WRATED=223.
*
*  CARDS TO INITIALIZE AT ARBITRARY TIME POINT
      CONSTANT  HCTO=539.00,LDCO=23.67,PO=1070.,TO=30.00,RHOGO=2.420
      CONSTANT VSTO=9897.2
      CONSTANT  TMANOP=30.
*  INITIAL CORE FLOW GUESS DEPENDS ON LDCO AND PRELO
      CONSTANT  WGUESS=8310.
*  REFERENCE PARAMETERS FOR CALCULATION OF NATURAL CIRCULATION
*  FLOW RESISTANCE COEFFICIENT
      CONSTANT  CPREF=3293., PRELR=.32, PRR=1020.,...
                LDCR=27.58, XREF=.133, WREF=9111., HREF=522.
*  RUN CONTROL PARAMETERS
      CONSTANT  RCICMX=82.9875, TLEAK=10.,JETPMP=2800.
*  BCRHY=FLOW FROM CRD HYD. SYS MIXING WITH LEAKAGE FROM CRDS(GPM)
*  BSUMP=RATED FLOW OF REACTOR BLDG FLOOR DRAIN SUMP PUMP(GPM)
*  BHOTMU=MAXIMUM AUTO-MAKEUP(GPM) TO HOTWELL FROM CST
*  VHOTWO=INITIAL WATER VOLUME IN HOTWELL(GALLONS)
*  VHOTMU=HOTWELL LOW VOLUME(GAL) TO START AUTO MAKEUP VIA 4-IN CST LINE
*  VCST=INITIAL CST VOLUME(GAL)
      CONSTANT  BCRHY=170.,BSUMP=50.,BHOTMU=430.
      CONSTANT  VCSTO=3.62E05,VHOTWO=.98E05,VHOTMU=.979E05
*
*                        FUNCTION SUBPROGRAMS
*  CAVOID(XIN,XOUT,WTOTAL,TSAT,RHOF,RHOG)
*  DKFUN(T)=P/PO AS A FUNCTION OF TIME AFTER SHUTDOWN
*  QREGAV(BOTTOM,TOP,KAPPA,PEKAVG)=(AVG HT. FLUX FROM A TO B)/(CORE AVG)
*  RHOTP(RHOF,RHOG,VOID)=2-PHASE DENSITY
*  XLENDC(VOLDC)=DOWNCOMER LEVEL AS FUNCTION OF LIQUID VOLUME
*  VOID(QUALITY,TOTAL FLOW,FLOW AREA,TSAT,RHOF,RHOG)=POINT VOID FRACTION
*  VOLDC(LDC)=DOWNCOMER HEIGHT ABOVE BOT OF ACT. FUEL
*
*     STEAM TABLE CSMP INTERPOLATION FUNCTIONS
*  VSATF(PRESS)=SAT FLUID SPECIFIC VOL(FT**3/LB)
      FUNCTION  VSATF=15.,.0167, 50.,.0173, 100.,.0177, 200.,.0184,...
        400.,.0193, 600.,.0201, 800.,.0209, 1000.,.0216, 1200.,.0223,...
        1400.,.0231
*
*  VSATG(PRESS)=SAT GAS SPECIFIC VOL(FT**3/LB)
      FUNCTION  VSATG=15.,26.3, 50.,8.51, 75.,5.81,100.,4.43,...
        150.,3.01,200.,2.29.400.,1.16, 600.,.77,800.,.569,...
       1000.,.446,1200.,.362,1400.,.302
*
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*   HSATF(PRESSURE)=SAT FLUID ENTHALPY(BTU/LB)
       FUNCTION HSATF=15.,181., 50.,250., 100.,298., 200.,355.,...
         400.,424., 600.,472., 800.,510., 1000.,543., 1200.,572.,...
         1400.,599.
*
*   HSATG(PRESSURE)=SAT GAS ENTHALPY(BTU/LB)
       FUNCTION HSATG=15.,1151., 50.,1174., 100.,1187., 200.,1198.,...
         400.,1205., 600.,1204., 800.,1199., 1000.,1193., 1200.,1185.,...
         1400.,1175.
*
*   TSATM(PRESSURE)=SAT MIXTURE TEMPERATURE(DEG-F)
       FUNCTION TSATM=15.,213., 50.,281., 100.,328., 200.,382.,...
         400.,445., 600.,486., 800.,518., 1000.,545., 1200.,567.,...
         1400.,587.
*
* CONDENSATE BOOSTER PUMP FLOW AS A FUNCTION OF REACTOR VESSEL PRESSURE
       FUNCTION CBPF=0.,4292.,42.,4087.,106.,3716.,...
         217.,2973.,303.,2229.,366.,1486.,404.,743.,418.,0.,1500.,0.
*
*
*   VSC(ENTHALPY)=SUBCOOLED FLUID SPECIFIC VOL(FT**3/LB)
       FUNCTION  VSC=21.4,.016, 121.,.0163, 221.,.0169, 272.,.0174,...
         376.,.0185, 431.,.0193, 488.,.0203, 549.,.0217, 562.,.0221,...
         575.,.0224
*   CALCULATION OF INITIAL SATURATION PROPERTIES
       RHOFO=1./(AFGEN(VSATF,PO))
       HFO=AFGEN(HSATF,PO)
       HGO=AFGEN(HSATG,PO)
       TSATO=FUNGEN(TSATM,2,PO)
*   SUBCOOLED WATER DENSITY
       RHOSCO=1./(AFGEN(VSC,HCIO))
*
*   CALCULATION OF FLOW RESISTANCE COEFFICIENT
*   NEEDS:  REFERENCE CONDITIONS FOR NATURAL CIRCULATION:CPREF,PRELR,PRR,XREF,
*                                                         WREF, HREF, LDCR
*   RETURNS:  LOSS COEFFICIENT:  KLOSSU
*   CALCULATION OF REFERENCE DENSITIES AND SATURATION PROPERTIES
       HFR=AFGEN(HSATF,PRR)
       TSATR=FUNGEN(TSATM,2,PRR)
       RHOFR=1./(AFGEN(VSATF,PRR))
       RHOGR=1./(FUNGEN(VSATG,2,PRR))
       RHOSCR=1./(AFGEN(VSC,HREF))
       QTOTR=CPREF*PRELR*947.8
*   THIS EXP FOR LSC ASSUMES AVG POWER=CORE AVG IN SC REGION
       LSCR=12.*WREF*(HFR-HREF)/QTOTR
       LBR=12.-LSCR
       RSCAVR=RHOFR/2.+RHOSCR/2.
*   FUNCTION SUBPROGRAM CAVOID RETURNS AVERAGE CORE BOILING REGION
*   VOID FRACTION FROM THE GIVEN INPUTS
       CAVR=CAVOID(0.,XREF,WREF,TSATR,RHOFR,RHOGR)
       RHOBR=RHOTP(RHOFR,RHOGR,CAVR)
       ROCORR=RSCAVR*LSCR/12.+RHOBR*LBR/12.
       RHOPR=RHOTP(RHOFR,RHOGR,OPVR)
*   FUNCTION VOID RETURNS POINT VOID FRACTION FROM GIVEN INPUTS
       OPVR=VOID(XREF,WREF,ACOP,TSATR,RHOFR,RHOGR)
       RORTR=RHOTP(RHOFR,RHOGR,RTVR)
       RTVR=VOID(XREF,WREF,ART,TSATR,RHOFR,RHOGR)
       RDP=LDCR*RHOSCR/144.-RORTR*LRT/144.-...
         RHOPR*LOP/144.-ROCORR*12./144.
       KLOSSU=RDP/(WREF*WREF)
*
*   INITIAL RELATIVE POWER
*
*
```

```
      PRELO=ANS(TO)
      QTOTO=PRELO*CPO*947.8
*
*  CALCULATION OF INITIAL TOTAL FLOW BY ITERATIVE SOLUTION, WITH
*  STARTING FLOW GUESS INPUT BY USER
      WCIO=IMPL(WGUESS+JETPMP,.001,WCAL)
      XOU=(QTOTO/WCIO+HCIO-HFO)/(HGO-HFO)
      XO=LIMIT(0.001,1.000,XOU)
*  THIS EXP FOR LSC ASSUMES AVG POWER=CORE AVG IN SC REGION
      LSCO=LIMIT(0.,12.,12.*WCIO*(HFO-HCIO)/QTOTO)
      LBO=12.-LSCO
*  CALCULATION OF INITIAL DENSITIES
      RSCAVO=RHOFO/2.+RHOSCO/2.
      WBOO=WCIO*XO
*  VOID AT BOTTOM AND TOP OF BOILING REGION CALC. BY DFVOID
*  FUNCTION SUBPROGRAM, THEN AVERAGED TO GIVE CORE AVERAGE VOID
      CAVINO=DFVOID(ACOR,WCIO,0.,TSATO,RHOFO,RHOGO)
      CAVEXO=DFVOID(ACOR,WCIO-WBOO,WBOO,TSATO,RHOFO,RHOGO)
      CAVO=(CAVINO+CAVEXO)/2.
      RHOBO=RHOTP(RHOFO,RHOGO,CAVO)
      ROCORO=RSCAVO*LSCO/12.+RHOBO*LBO/12.
      ROPO=RHOTP(RHOFO,RHOGO,OPVO)
      OPVO=DFVOID(ACOP,WCIO-WBOO,WBOO,TSATO,RHOFO,RHOGO)
      RORTO=RHOTP(RHOFO,RHOGO,RTVO)
      RTVO=DFVOID(ART,WCIO-WBOO,WBOO,TSATO,RHOFO,RHOGO)
*  NET GRAVITY PRESS DROP(PSI) AVAILABLE FOR UNRECOV. DROP ACROSS CORE
      LDCX=LIMIT(0.,LDCR,LDCO)
      UNRECO=LDCX*RHOSCO/144. + (12.+LRT+LOP-LDCX)*RHOGO/144.-...
        RORTO*LRT/144.-ROPO*LOP/144.-ROCORO*12./144.
      DPO=LIMIT(.0001,1000.,UNRECO)
      WCAL=SQRT(DPO*RHOBO/(KLOSSU*RHOBR))+JETPMP
*  END ITERATIVE LOOP
*
*
*  DOWNCOMER AND LOWER PLENUM INITIALIZATION
*  VOLUME OF WATER IN DOWNCOMER NODE GIVEN BY FUNCTION
      LDCVZO=12.*LDCO+216.
*  VOLDC( ) AS A FUNCTION OF WATER LEVEL
      MDCO=VOLDC(LDCO)*RHOSCO
      MHDCO=MDCO*HCIO
      MLPO=VOLP*RHOSCO
      MHLPO=MLPO*HCIO
*
      G=4.75*ART*RHOGO
      MTOTO=(LSCO*RSCAVO+LBO*RHOBO)*ACOR+LOP*ACOP*ROPO+LRT*ART*RORTO+G
*
*  PRESSURE CALCULATION INITIALIZATION
      RHOSVR=1./FUNGEN(VSATG,2,PRATED)
      UCSRV=WRATED/SQRT(PRATED*RHOSVR)
      MSTO=(VFREE-VOLDC(LDCO))*RHOGO
      UMSTO=MSTO*(HGO-PO*144./(778.*RHOGO))
*
*
*
*---------INITIALIZATION FOR SUPPRESSION POOL CALCULATION---------------
*
*
*                     CONSTANTS FOR CONTAINMENT MODEL
*
*  ADMET=HEAT TRANS AREA BET DW MET AND DW ATMOS(FT**2)
*  APMET=HEAT TRANSF AREA BET SP MET AND SP ATMOS
*  ASSPW=AREA OF POOL WATER SURFACE(SQ.FT.)
*  BDWC=CU.FT./SEC THRU DRYWELL COOLERS(AT TEMP TDWCE))
*  BDWSPO=CU.FT/SEC/PSI FLOW WHEN DOWNCOMERS CLEARED
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```
*  BSPDWO=CU.FT/SEC/PSI OF FLOW WHEN VALVE OPEN
*  CDMET=MASS*SPEC HEAT OF DW METAL(BTU/DEG-F)
*  CPAIR=MASS*SPEC HEAT OF AIR IN SP CHAMBER ROOM
*  CPMET=MASS*SPEC HEAT OF SP METAL IN CONT WITH GAS
*  DM=AMOUNT OF STM. QUENCHED(UNIFORMLY ARCUND POOL) IF THERE
*  IS ELAPSED TIME BETWEEN NOMINAL START AND INITIALIZATION OF THE RUN
*  DPQR=RANGE(PSI) OVER WHICH QUENCH FRACTION GOES FROM 1 TO 0.
*   WITH INCREASING VAPOR PRESSURE
*  DPQZ=MARGIN(PSI) ABOVE SATURATION FOR CCMPLETE QUENCHING
*  FDI,FSI=FRACTION OF POOL COOLING DISCHARGE AND SUCTION TO POOL NODE I
*  FWE=FLOW BETWEEN S.P. NODES(LB/SEC) PER INCH OF WATER LEVEL DIFFERENCE
*  FWI=FRACTION OF POOL WATER CONTAINED IN POOL NODE I
*  FSVI=FRACTION OF TOTAL SRV FLOW DISCHARGED TO POOL NODE I
*  FTEI=FRACTION OF TURBINE EXHAUST DISCHARGED TO POOL NODE I
*  FLSPG, FLDWG=FRACTION OF ATM. LEAKED TO RX-BLDG. PER SECOND
*  GCH=GAS CONSTANT OF H2
*  GCM=GAS CONSTANT OF M(MISC.)
*  GCN=GAS CONSTANT OF N2
*  GCS=GAS CONSTANT OF H2O VAP(PSI*FT**3/LB*DEG-R)
*  HSRREF=DELTA TO REREFERENCE STM ENTHALPY FROM ASME STEAM TABLES
*  TO PERFECT GAS EXP. THAT HAS H=0.0 AT 0 DEG-R
*  HSTE=ENTH OF TURBINE EXHAUST(BTU/LB)
*  HUMDWO= INITIAL DW GAS HUMIDITY (PERCENT)
*  HUMSPO=INITIAL SP GAS HUMIDITY (PERCENT)
*  KMIX=EMPIRICLE CONSTANT FOR NAT. CIRC MIXING FLOW BET. POOL NODES
*      LBASE=NOMINAL STARTING LEVEL OF POOL(IN. FROM INST. 0.)
*  MHSPGO,MHDWGO=INITIAL MASSES OF H(LBS)
*  MMSPGO,MMDWGO=INITIAL MASSES OF M (LBS)
*  PDCVP=PRESS. DIFF NECESSARY TO CLEAR THE VENT
*  PIPES FOR FLOW FROM DW TO SP
*  PTDWGO= INITIAL TOTAL PRESS OF DW GAS (PSIA)
*  PTSPGO=INITIAL TOTAL PRESS OF SP GAS (PSIA)
*  QRVHLO=REACTOR(+PIPING) HEAT LOSSES (MW) FOR TEMP DIFF=DTRVHL(DEG-F)
*  TAUFDW=AVERAGE RESIDENCE TIME OF FOG IN DW(SEC)
*  TAUFSP=AVERAGE RES. TIME OF FOG IN SP
*  TAUVRV=TIME CONSTANT FOR SP VAC RELIEF VALVE TO LIFT(SEC)
*  TBASE=NOMINAL STARTING TEMP CF POOL(DEG-F)
*  TBII=PROVISION FOR BIASING STARTING TEMP OF POOL NODE I
*  TD=DISCH. TEMP(F) OF POOL COOLING FLOW
*  TDMETO=INITIAL DRYWELL METAL TEMP (F)
*  TDWCE=DRYWELL COOLER EXIT TEMP(DEG-F)
*  TGDWO=INITIAL DW GAS TEMP(DEG-F)
*  TGSPO=INITIAL SP GAS TEMP (DEG-F)
*  TPAIRO=INITIAL SP CHAMBER ROOM TEMP
*  TPMETO=INITIAL TEMP : SP METAL IN CONTACT WITH SP GAS
*  TSTRAT=DELTA BETWEEN BULK POOL TEMP AND TEMP THAT THE
*  T-QUENCHER EFFECTIVELY SEES
*  VGDW=TOTAL FREE VOLUME OF DW (FT**3)
*  VTSP=TOTAL FREE VOLUME OF SP (FT**3)
*  WDLEAK=LEAK RATE(LB/SEC) OF SAT. WATER FROM RV TO DRYWELL
*  WDSPW=DISCHARGE FLOW OF POOL COOLING(LB/SEC)
*  WSSPW=SUCTION FLOW OF POOL COOLING(LB/SEC)
*
*
*  GEOMETRY AND PHYSICAL CHARACTERISTICS
      CONSTANT  ADMET=1.87E04,APMET=1.7E04,ASSPW=10860.
      CONSTANT  BDWSPO=2500.,BSPDWO=2000.
      CONSTANT  CDMET=8.33E04,CPAIR=3200.,CPMET=5.82E04
      CONSTANT  DPQR=2.,DPQZ=2.
      CONSTANT  GCH=5.361,GCM=.2436,GCN=.3829,GCS=.5955
      CONSTANT  HSTE=915.,HSRREF=-854.5,PDCVP=1.75
      CONSTANT  TAUVRV=3.,TAUFDW=30.,TAUFSP=15., TSTRAT=50.
      CONSTANT  VGDW=159000.,VTSP=267600.
*
```

```
*  POOL NODALIZATION
      CONSTANT FD1=1.,FS1=1.,FW1=1.,FSV1=1.,FTE1=1.,FWE=5000.,KMIX=950.
*
*  INITIALIZATION
      CONSTANT  DM=2.9E04,HUMDWO=20.0,HUMSPO=100.,LBASE=-4.00
      CONSTANT  MHSPGO=0.,MHDWGO=0.,MMSPGO=0.,MMDWGO=0.
      CONSTANT  PTDWGO=16.00,PTSPGO=14.50,TBASE=90.00,TBI1=0.
      CONSTANT  TGDWO=145.,TGSPO=90.0,TDMETO=145.,TPMETO=90.0,TPAIRO=90.0
*
*  RUN CONTROL
      CONSTANT  BDWC=2000., TDWCE=90.
      CONSTANT  FLDWG=0.,FLSPG=0.,TD=90.,QRVHLO=1.,DTRVHL=404.
      CONSTANT  WDLEAK=.68,WDSPW=0.
      HD=TD-32.
*
*  INITIALIZATION CALC. FOR DW AND SP MASS AND ENERGY BALANCES
*  ASSUME ZERO INITIAL HYDROGEN AND MISC. GAS
*
*---------FUNCTION TABLES
*  FUNCTION SPLEV : SP LEVEL(INCHES FROM INSTRUMENT ZERO) AS A FUNCTION A
*  FUNCTION OF VOLUME(CU.FT)
     FUNCTION  SPLEV=0.,-182.,20222.,-134.,74646.,-62.,...
           117344.,-14.,130129.,0.,139300.,10.,...
           182563.,58.,242176.,130.,267611.,190.
*
*
*  FUNCTION STFOSV: SATURATION TEMP AS A FUNCTION OF SPECIFIC
*  VOLUME
     FUNCTION STFOSV=2.83,363.5,4.65,324.1,7.65,288.2,12.2,257.6,...
     20.1,228.,31.2,204.,44.7,185.6,76.4,160.5,158.9,129.6,...
          333.6,101.7,641.5,79.6,1235.,59.3
*
*  FUNCTION SPFOT : SATURATION PRESSURE AS A FUNCTION OF TEMPERATURE
     FUNCTION SPFOT=59.3,.25,79.6,.5,101.7,1.0,132.9,2.4,...
              160.5,4.8,185.6,8.5,203.9,12.5,...
              228.,20.,250.34,30.,288.2,56.,324.1,95.,363.5,160.,...
         381.8,200.,417.3,300.,444.6,400.,467.,500.,486.,600.,...
        503.,700.,525.,850.,544.6,1000.,556.,1100.,567.,1200.
*
*
*
*---------PRELIMINARY CALCULATIONS
     SPDWO=NLFGEN(SPFOT,TGDWO)
     SPSPO=NLFGEN(SPFOT,TGSPO)
     PSDWGO=(HUMDWO/100.)*SPDWO
     PSSPGO=(HUMSPO/100.)*SPSPO
*
     PNSPGO=PTSPGO-PSSPGO
     PNDWGO=PTDWGO-PSDWGO
*
*
     MNSPGO=PNSPGO*(VTSP-VWSPO)/(GCN*(TGSPO+460.))
     MNDWGO=PNDWGO*(VGDW)/(GCN*(TGDWO+460.))
*
*  VWSPO=INITIAL TOTAL WATER VOL IN SP
*
     MSSPGO=PSSPGO*(VTSP-VWSPO)/(GCS*(TGSPO+460.))
     MSDWGO=PSDWGO*VGDW/(GCS*(TGDWO+460.))
*
*
     UMSPGO=(TGSPO+460.)*(MNSPGO*(.2475-.1851*GCN)+...
        MSSPGO*(.45-4.89/(TGSPO+460.)-.1851*GCS) )
     UMDWGO=(TGDWO+460.)*(MNDWGO*(.2475-.1851*GCN)+...
        MSDWGO*(.45-4.89/(TGDWO+460.)-.1851*GCS) )
```

```
*
*  DEPTH LBASE SPECIFIED: INCHES FROM INST ZERO(4 IN BELOW CENTER)
*  TBASE IS NOMINAL STARTING TEMP OF POOL
*  TBI(I) PROVIDES FOR BIASING INITIAL TEMPS IN MULTI-NODE MODEL
      VII=2.462*AII
      AII=(LBASE-4.)*SQRT(34580.-LBASE*LBASE+8.*LBASE)+...
          34596.*ARSIN( (LBASE-4.)/186. ) + 54343.
      MWIII=FW1*VII*(63.9-.019*(TBASE+TBI1))
      MII=MWIII
      DTBASE=(1190.-(TBASE-32.))*DM/(MII+DM)
      TIC1=TBASE+DTBASE+TBI1
      VWSPO=(MWIII+FW1*DM)/(63.9-.019*TIC1)
      VGSPO=VTSP-VWSPO
*
*
     DYNAMIC
*------------DYNAMIC PORTION OF REACTOR VESSEL CALCULATION-------------
*
*--------RUN CONTROLS FOR INJECTION CONTROL
      LDCVZ=LDC*12.+216.
      LDDEL=REALPL(LDCVZ0,2.,LDCVZ)
      RCICD=0.
      HPISOL=RST(0.,COMPAR(115.,P),0.)
      HPINIT=COMPAR(476.5,LDDEL)
      HPTRIP=COMPAR(LDDEL,582.)
      HPCID=RST(HPTRIP+HPISOL,HPINIT,0.)*COMPAR(3.9E06,MLEAK)
*---------RUN CONTROLS FOR VESSEL PRESSURE CONTROL
      PSHUT=1070.
      POPEN=1120.
*
*  CONDENSER HOTWELL VOLUME(IN GALLONS)
*  COND BOOSTER PUMP ASSUMED TO TRIP WHEN VHOTW.LT.0.
      VHOTW=INTGRL(VHOTW0,(WHOT-WCBP)*.12)
      WCBP=AFGEN(CBPF,P)*RST(-VHOTW,VHOTW-500.,0.)
*  HOTWELL AUTO-MAKEUP ASSUMED ON AT VOLHOTMU AND OFF AT VOLHOTMU*1.2
      WHOT=.138*BCST1*BHOTMU*RST(VHOTW-1.2*VHOTMU,VHOTMU-VHOTW,0.)*...
          (.53+1.26E-06*VCST)
*
*  CONDENSATE STORAGE TANK VOLUME(GALLONS)
      VCST=INTGRL(VCST0,.12*(-WHPCI-WRCIC-WHOT-WCRHY) )
      BCST1=COMPAR(VCST,135000.)
      BCST2=COMPAR(VCST,0.)
*
*  INJECTION FLOW CONDITIONS
      WINJ=WHPCI+WRCIC+WCBP
      WINTOT=INTGRL(0.,WINJ)
      GP=7.23*WINJ
      GT=(WINTOT*7.481)/62.11
      WHPCI=RCICMX*8.33*HPCID
      WRCIC=RCICMX*RCICD
      HINJIN=( (WRCIC+WHPCI)*(58.+(1.-BCST2)*(TWSPAV-90.))+...
         WCBP*(68.+3.33E-03*TIME))/AMAX1(.01,WINJ)
      WSSPW=(WHPCI+WRCIC)*(1.-BCST2)
*
*  LEAKAGE CONDITIONS: WCRD=LEAK FLOW AND WCRHY=CRD HYDRAULIC SYS
*  FLOW THAT MIXES WITH LEAK FLOW
*  WCRD=LEAKAGE FLOW OUT OF REACTOR VESSEL TO OUTSIDE CONTAINMENT
*  FLOW CALCULATED PER MOODY MODEL(FIG.9-10.A OF LAHEY AND MOODY)
      ALEAK=.007+6.80E-07*(RAMP(5400.)-RAMP(28800.))
      GC=256.*SQRT(P)
      SUBCOR=21.*LIMIT(0.,500.,HF-HLP)
*  GCLIM BASED ON A DISCHARGE COEFFICIENT OF .6
      GCLIM=57.72*SQRT(RHOLP*P)
      WCRD=ALEAK*LIMIT(0.,GCLIM,GC+SUBCOR)
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      WCRHY=BCRHY*.138*BCST2
      HCRDL=(WCRD*HLP+.001*HLP+WCRHY*58.)/(WCRD+.001+WCRHY)
      XCRFL=LIMIT(0.,1.,(HCRDL-180.)/970.)
*
*  MLEAK IS TOTAL WATER MASS COLLECTION IN BASEMENT
      MLEAK=INTGRL(0.,(WCRD+WCRHY)*(1.-.25*XCRFL)-.138*BSUMP)
*
*  CALCULATION OF DYNAMIC SATURATION PROPERTIES
      RHOF=1./(AFGEN(VSATF,P))
      RHOG=1./(FUNGEN(VSATG,2,P))
      HF=AFGEN(HSATF,P)
      HG=AFGEN(HSATG,P)
      TSAT=FUNGEN(TSATM,2,P)
*  INTERMEDIATE CALCULATIONS: AVERAGE HEAT FLUXES
*  'DKFUN' RETURNS DECAY HEAT AS A FUNCTION OF TIME SINCE SCRAM
*  'QREGAV' RETURNS AVERAGE NORMALIZED POWER, GIVEN LOCATION OF TOP
*  AND BOTTOM OF REGION
      T=T0+TIME
      TMIN=T/60.
*
      PREL=ANS(T)
*
      QTOT=PREL*CP0*947.8
      QFCOR=QTOT/(12.*PCOR)
      AVMSC=QREGAV(0.,LSCD)
      AVMB=QREGAV(LSCD,LSCD+LBD)
      QFSC=AVMSC*QFCOR
      QFB=AVMB*QFCOR
      LSCD=REALPL(LSC0,TAULEN,LSC)
      LBD=REALPL(LB0,TAULEN,LB)
*  DOWNCOMER ANNULUS CALCULATION
*  NEEDS: DENSITIES: RHODC,RHOLP
*         FLOWS INTO DOWNCOMER: WINJ, WRECIR, WINJAS
*         INJECTION ENTHALPY:  HINJIN
*         FLOWS OUT OF DOWNCOMER: WCI, WFLDC, WFLLP, WCRD
*         REACTOR VESSEL PRESSURE: P
*  RETURNS: DOWNCOMER HEIGHT(ABOVE BOTTOM OF ACTIVE FUEL) : LDC
*             ENTHALPY INTO LOWER PLENUM: HDC
*
*  INTERMEDIATE CALCULATIONS: MASS RATE OF ASPIRATION FROM STEAM SPACE
*  DUE TO INJECTION FLOW; FLASHING RATE FROM DC MASS; RECIRC FLOW RATE
      EQXX=EQX0*(LHEDER-LDC)/6.
      EQX=LIMIT(0.,1.,EQXX)
      HINJFI=HINJIN+(HF-HINJIN)*EQX
      WINJAS=(HF-HINJIN)*EQX*WINJ/(HG-HF)
      WFLDCX=MDC*FFLASH*(HDC-HF)
      WFLDC=LIMIT(0.,9999.,WFLDCX)
*
*  MASS AND ENERGY BALANCES(B1 IS A LOGIC SIGNAL TO STOP THE
*  INTEGRATION WHEN LDC.LT.0(BELOW BOTTOM OF ACTIVE FUEL)
      CMDC=INTGRL(0.,DMDC+WCOND)
      MDC=MDC0+CMDC
  DMDC=B1*(WRECIR+WINJ+WINJAS-RHODC*(WCI+WFLLP)/RHOLP-WFLDC       )+WEXPC
      CMHDC=INTGRL(0.,DMHDC)
      MHDC=MHDC0+CMHDC
      HSCMAX=QTOT/WCIG+HLP
      HSCEX=AMIN1(HF,HSCMAX)
      TMIX=LIMIT(1.,1000.,(MTOT-MCOR)/WCIG)
      HRECIR=REALPL(HF0,TMIX,HSCEX)
   DMHDC=B1*(HRECIR*WRECIR+WINJ*HINJIN+HG*WINJAS-...
        HDC*(WCI+WFLLP)*RHODC/RHOLP-HG*(WFLDC-WCOND) )
      HDC=MHDC/MDC
      RHODC=1./AFGEN(VSC,HDC)
*
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*  DC LEVEL IS CALCULATED BY FUNCTION SUBPROGRAM XLENDC(VOLUME)
      VDC=MDC/RHODC
      LDC=XLENDC(VDC)
      LDL=LIMIT(0.,LDCR,LDC)
*  LOWER PLENUM CALCULATION
*  NEEDS: SAME AS DOWNCOMER
*  RETURNS:  ENTHALPY AT CORE INLET(HLP)
*  INTERMEDIATE CALCULATIONS:  LOWER PLENUM FLASHING RATE, LP WATER VOL
*  (VOLPW--USED FOR LOGIC CONTROL ONLY SINCE THE LOWER PLENUM IS
*  MODELED AS A CONSTANT VOLUME UNLESS DOWNCOMER IS EMPTY) :
      HLP=MHLP/MLP
      VOLPW=MLP/RHOLP
      RHOLP=1./AFGEN(VSC,HLP)
      XWFLLP=(HLP-HF)*MLP*FFLASH
      WFLLP=LIMIT(0.,9999.,XWFLLP)
*
*  LOGIC CONTROL
*
      X1B1=LDC-.001
      X2B1=VOLPW-1.01*VOLP
      B1N=NOR(X1B1,X2B1)
      B1=NOT(B1N)
*
*  MASS AND ENERGY BALANCES
*
      DMLP=(WCI+WFLLP)*(RHODC/RHOLP)*(B1)-WCI-WFLLP+...
        B1N*(WINJ+WINJAS)
      CMLP=INTGRL(0.,DMLP)
      MLP=MLPO+CMLP
      DMHLP=HDC*(WCI+WFLLP)*RHODC*B1/RHOLP+...
        B1N*(WINJ*HINJIN+HG*WINJAS)-WCI*HLP-WFLLP*HG
      CMHLP=INTGRL(0.,DMHLP)
      MHLP=MHLPO+CMHLP
*  CORE INLET FLOW CALC.(FORCE BALANCE)
*  NEEDS:  DENSITIES: RHODC,RHOLP,RSCAV,RHCB,ROP, RORT
*          REACTOR VESSEL PRESSURE : P
*          FLOW OUT OF BOILING REGION: WBO
*          REGION LENGTHS: LDC,LSC,LB
*  RETURNS: FLOW INTO CORE: WCI
*
     CMTOT=INTGRL(0.    ,(WCI-WRECIR-WTOST+WFLLP-WCRD))
      MTOT=MTOTO+CMTOT
      WBO=(QFB*PCOR*LBD)/(HG-HF)
      WRECIR=LIMIT(0.,10000.,(WCI-WCRD-WTOST)*FRECIR)+WCOLL
      WCOLL=LIMIT(0.,10000.,(LINT-30.)*3000.)
      FRECIR=LIMIT(0.,10.,(LINT-LBOT)/(LDCR-LBOT))
      LINT=LSCD+LB+LOPTP+LRTTP
      WFLOP=0.0
      WFLBR=0.0
      WFLRT=0.0
      MCOREM=12.*ACOR*RHOG
      MOPEM=LOP*ACOP*RHOG
      MRTEM=(LRT+4.75)*ART*RHOG
      WBRAV=WCIG
      RSCAV=(RHOSCX+RHOLP)/2.
      RHOSCX=1./AFGEN(VSC,HSCEX)
      WCIG=REALPL(WCIO,3.,LIMIT(1.E-04,1.E+04,WCI-WCRD) )
*  CALCULATE DENSITIES
      CAVIN=DFVOID(ACOR,WCIG,0.0  ,TSAT,RHOF,RHOG)
      CAVEX=DFVOID(ACOR,WCIG-WBO,WBO            ,TSAT,RHOF,RHOG)
      CAV=(CAVIN+CAVEX)/2.
      RHOB=RHOTP(RHOF,RHOG,CAV)
      OPV=DFVOID(ACOP,WCIG-WBO,WBO            ,TSAT,RHOF,RHOG)
      RHOOP=RHOTP(RHOSCX,RHOG,OPV)
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      RTV=DFVOID(ART,WCIG- WBO,WBO                    ,TSAT,RHOF,RHOG)
      RHORT=RHOTP(RHOSCX,RHOG,RTV)
      BUVIN=DFVOID(ACOR,WCIG,WFLLP,TSAT,RHOF,RHOG)
      BUVEX=DFVOID(ACOR,WCIG-WBO,WTOST,TSAT,RHOF,RHOG)
      BUAVG=(BUVIN+BUVEX)/2.
      ROBUB=RHOTP(RHOF,RHOG,BUAVG)
*
      DELH1=LIMIT(1.,400.,HF-HLP)
      DLSCX=(-2.)*(LSCX*QFSC*PCOR-(WCI-WCRD )*LIMIT(0.,400.,HF-HLP))/...
          (RHOLP*ACOR*DELH1)
      LSCX= INTGRL(LSCO,DLSCX)
      LSC=LIMIT(0.,12.,LSCX)
      MSC=RSCAV*LSCD*ACOR
*
*  FIND THE TWO PHASE LENGTHS AND REGION MASSES
*
      LB=LIMIT(0.,12.-LSCD,LBU)
      LBU=(MTOT-MCOREM+MSCDUM-MOPEM-MRTEM-MSC)/(ACOR*(RHOB-RHOG))
      LCOV=LIMIT(0.,12.,LBCOVU+LSCD)
      LBCOVU=LBU*(RHOB-RHOG)/(ROBUB-RHOG)
      MSCDUM=ACOR*RHOG*LSCD
      MB=LB*RHOB*ACOR
      MCOR=MSC+MB+(12.-LB-LSCD)*RHOG*ACOR
     LOPTP=LIMIT(0.,LOP,(MTOT-MCOR-MRTEM-MOPEM)/(ACOP*(RHOOP-RHOG)))
      MOP=(RHOOP*LOPTP+(LOP-LOPTP)*RHOG)*ACOP
      MRT=MTOT-MOP-MCOR
      LRTTP=(MRT-MRTEM)/(ART*(RHORT-RHOG))
      MPOO=(LB*(1.-CAV)*ACOR+LOPTP*(1.-OPV)*ACOP+...
            LRTTP*(1.-RTV)*ART)*RHOF*(1.-CCMPAR(LSC,11.99))
*
*  TABULATE PRESS DROPS AND CALCULATE WCI
*
      PCOBOT=(MCOR/ACOR+MOP/ACOP+LRTLIM*RHORT+(10.-LRTLIM)*RHOG)/144.
      LRTLIM=LIMIT(0.,10.,LINT-17.52)
      LOSSIN=LDL*RHODC/144.+(LDCR-LDL)*RHOG/144.-PCOBOT
      XDUM=LIMIT(1.E-9,5.,LOSSIN)
      WCI=SQRT(XDUM*RHOB/(KLOSSU*RHOBR)) + WCIJET
      WCIJET=JETPMP*RST(476.5-LDDEL,1.-STEP(1.),0.)
      WCID=REALPL(WCIO,10.,WCI)
*
*  REACTOR VESSEL PRESSURE CALCULATION
*
*  STEAM MASS BALANCE
      CMST=INTGRL(0.,DMST)
      MST=MSTO+CMST
      DMST=WTOST+WEVAP+WFLDC-WEXPC-WCOND-WINJAS-WSTC
      WTOST=LIMIT(0.,1000.,WBO+WFLLP-WPOO)
*  CONDENSATION OF STEAM LIMITED ONLY BY CONVECTION OF WARMER WATER
*  FROM THE INTERFACE DOWN INTO DOWNCOMER. THIS RATE CALCULATED AS
*  50PERCENT OF THE RATE IN EQ.(7-24), KRIETH
*  BUBBLE IS NOT ALLOWED TO TOTALLY COLLAPSE
 WCOND=3.6*COMPAR(VSTD,500.0)*((AMAX1(.0001,HF-HRECIR))**1.333)/(HG-HF)
      WEVAP=0.0
      WEXPC=LIMIT(0.,100.,1.67E-04*MST*(HGST-HST))
      HGST=AFGEN(HSATG,PST)
      WPOO=MPOO*DHF/(HG-HF)
      HFD=REALPL(HFO,2.0,HF)
      DHF=(HF-HFD)/2.
*
      VST=VFREE-VDC
       VSTD=REALPL(VSTO,3.,VST)
       DVST=(VST-VSTD)/3.
      VSVST=VST/MST
*  VST=TOTAL VAPOUR SPACE VOLUME
*  VSVST=SPECIFIC VOLUME OF STEAM
```

```
*
*
*  TOTAL ENERGY BALANCE
      CUMTO=INTGRL(0.,DUMTO)
      UMTO=UMSTO+CUMTO
      DUMTO=WTOST*HG+(WFLDC+WEVAP)*HG-...
      WEXPC*HF-WCOND*HG-WINJAS*HST-WSTC*HST-DVST*P*144./778.
      HST=UMTO/MST+PSTG*VSVST*144./778.
*  STEAM PRESSURE CALCULATED BY FUNCTION SUBPROGRAM PFVH-- STEAM PRESSURE
*  AS A FUNCTION OF SPECIFIC VOLUME AND ENTHALPY
*  PFVH REQUIRES AN INITIAL GUESS OF PRESSURE, PSTG
      PST=PFVH(HST,VSVST,PSTG)
      PSTG=REALPL(PO,2.0,PST)
      P=PST
*
*  SAFETY RELIEF VALVE MODEL--TWO VALVES MODELED--WSSV=TOTAL SRV FLOW
      LOGREL=LOGV1+LOGV2
      LOGV1=REALPL(0.,.62,LOGPV1)
      LOGV2=REALPL(0.,.62,LOGPV2)
      LOGPV1=RST(X1V1,X2V1,0.)
      LOGPV2=RST(X1V2,X2V2,0.)
      X1V1=COMPAR(PSHUT,P)
      X1V2=COMPAR(POPEN-25.,P)
      X2V1=COMPAR(P,POPEN)
      X2V2=COMPAR(P,POPEN+25.)
      WSSV=UCSRV*SQRT(PSTG/VSVST)*LOGREL
*
*  WSTE=STEAM TURB EXHAUST FLOW(RCIC)
*  ENTHALPY IS ASSUMED=915. BUU/LB
      WSTE=(RCICD*(.00613*PST+1.1)+HPCID*(.0387*PST+7.81))
*  WSTC IS TOTAL STEAM FLOW RATE FROM REACTOR VESSEL TO CONTAINMENT
      WSTC=WSTE+WSSV
*
*----------DYNAMIC PORTION OF CONTAINMENT CALCULATION-------------------
*
*  CALLS TC SP WATER MACRO--ONLY ONE NECESSARY FOR THE SINGLE NODE POOL MODEL
      XNQ1 ,TP1 ,WE1 ,MW1 ,LW1 =W(TP1,...
        TP1 ,LW1,LW1 ,TIC1 ,FW1 ,FSV1 ,FTE1 ,FD1 ,FS1 )
      VWSP=MW1/(63.9-.019*TP1)
      MWSP=MW1
      GWSP=7.481*VWSP
*
      TWSPAV=(63.9-MWSP/VWSP)/.019
       LWSPAV=AFGEN(SPLEV,VWSP)
*
*  INTERFACE THE SP WATER TO THE SP GAS
*
      XNQDW=XNQ1
      WSSVNC=WSSV*FSV1*XNQ1
      WSTENQ=WSTE*FTE1*XNQ1
      WTESPW=WE1
*
*
*  INTERFACE VARIABLES : RV TO SUPPRESSION POOL
*  WSSV=STM. FLOW FROM RV THRU RELIEF VALVES(LB/SEC)
*  WSTE=STM  FLOW FROM RV THRU TURBINES(RCIC+HPCI)
*  WHSV=HYDROGEN FLOW FROM RV THRU RELIEF VALVES(LB/SEC)
*  HHM=HYDROGEN ENTHALPY AT MIXED RV TEMP(BTU/LB) (REF TO 0. DEG R)
*  HSTE=ENTHALPY OF TURBINE EXHAUSE(ASME)
*  TM=MIXTURE TEMP OF RV STEAM SPACE
*
*  INTERFACE VARIABLES : RV TO DRYWELL
*  WSDWR=STEAM FLOW, RELEASE TO DRYWELL(LB/SEC)
```

```
*   HSDWR=ENTHALPY OF WSDWR(ASME)
*   WHDWR=HYDROGEN FLOW, RELEASE TO DRYWELL(LB/SEC)
*   HHDWR=ENTHALPY OF WHDWR(REF TO 0.0 DEG R)
*   WMDWR=MISC. FLOW RELEASED TO DRYWELL
*   HMDWR=ENTHALPY OF WMDWR(0.0 DEG R)
*
*   INTERFACE VARIABLES : SP WATER TO SP GAS
*   WSSVNQ=TOTAL NON-QUENCHED RELIEF VALVE FLOW FROM RV TO SP VIA RV'S
*   WSTENQ=TOTAL NON-QUENCHED TURBINE EXHAUST FLOW
*   WCSPG=TOTAL CONSECSATE FLOW, SPG TO SPW
*   WTESPW=TOTAL EVAPORATION RATE, SPW TO SPG
*   HSTSPW=ENTHALPY OF STEAM(REF 0. AT 0. DEG) AT MASS-WEIGHTED SP WATER
*          TEMP, TWSP
*   HCSPG=ENTHALPY OF CONDENSED SP STEAM(ASME)
*
*   MASS BALANCES FOR SP GAS AND DW GAS
*
*   INPUT FROM OTHER CALCULATIONS:
*   VWSP=SP TITAL WATER VOL (FT**3) (VARIABLE)
*   WSSNQ= TOTAL NON-QUENCHED STEAM FLOW (LB/SEC)
*   WHSV= H FLOW FROM THE RV (THROUGH RELIEFS)
*   WSDWR= FLOW RATE OF STEAM RELEASED DIRECT TO DRYWELL
*   WHDWR= FLOW RATE OF HYDROGEN RELEASED TO DRYWELL
*   WMDWR= FLOW RATE OF MISC. RELEASED DIRECT TO DRYWELL
*
*
*   DEFINITIONS
*
*   MSSPG= MASS STM IN SPG
*   MHSPG= MASS H IN SPG
*   MNSPG= MASS N IN SPG
*   MMSPG=MASS M IN SPG
*
*   FLOWS = LB/SEC UNLESS OTHERWISE SPECIFIED
*   WSSVNG= TOTAL NON-QUENCHED STM FROM SV'S ANT TURB, EXHAUST
*   WHSV= TOTAL HYDROGEN FLOW FROM RELIEF VALVES
*   DWSP= DESIGNATES FLOW FROM DW TO SP
*   SPDW= DESIGNATES FLOW FROM SP TO DW
*   SPL= SP GAS LEAKAGE
*   B= BULK FLOW RATE (FT**3/SEC)
*   WCSPG= TOTAL CONDENSATE FLOW FROM SP GAS
*   WCDWG= TOTAL CONDENSATE FLOW FROM DW GAS
*   WCDWC=CONDENSATION IN DRYWELL COOLERS
*
*
*   SP GAS MASS BALANCE:
 MSSPG=INTGRL(MSSPGO,WSSVNQ+WSDWSP+WSTENQ+WTESPW-WSSPDW-WSSPL-WCSPG)
      MHSPG=INTGRL (MHSPGO,WHSV+WHDWSP-WHSPDW-WHSPL)
      MNSPG= INTGRL(MNSPGO,WNDWSP-WNSPL-WNSPDW)
      MMSPG=INTGRL (MMSPGO,WMDWSP-WMSPL-WMSPDW)
*
      BSPL=VGSP*FLSPG
      WSSPL=BSPL*(MSSPG/VGSP)
      WHSPL=BSPL*(MHSPG/VGSP)
      WNSPL=BSPL*(MNSPG/VGSP)
      WMSPL=BSPL*(MMSPG/VGSP)
*   BSPL= BULK LEAKAGE FROM SP GAS
*   FLSPG=FRACTION OF TOTAL SP GAS VOLUMES LEAKED PER SECOND
*   VGSP=VOLUME OF SP GAS, VWSP=SP WATER VOLUME
      VGSP=VTSP-VWSP
*
*   NOTE: 0.5 PSI IS NECESSARY TO OPEN VAC RELIEF VALVE
      BSPDWX=LIMIT(0.,1.E06,BSPDWO*(PTSPG-PTDWG-.5))
      BSPDW=REALPL(0.,TAUVRV,BSPDWX)
```

```
      WSSPDW=BSPDW*(MSSPG/VGSP)
      WHSPDW=BSPDW*(MHSPG/VGSP)
       WNSPDW=BSPDW*(MNSPG/VGSP)
      WMSPDW=BSPDW*(MMSPG/VGSP)
*  BSPDWO= (FT**3/SEC)/PSI OF PRESS DIFFERENCE WHEN
*  VAC RELIEF VALVE IS OPEN
*
      BDWSPX=LIMIT (0.,1.E06,BDWSPO*(PTDWG-PTSPG-PDCVP))
      BDWSP=REALPL(0.,TAUVRV,BDWSPX)
      WSDWSP=BDWSP*(MSDWG/VGDW)*XNQDW
      WHDWSP=BDWSP*(MHDWG/VGDW)
      WNDWSP=BDWSP*(MNDWG/VGDW)
      WMDWSP=BDWSP*(MMDWG/VGDW)
*  BDWSPO=(FT**3/SEC)/PSI WHEN DW PRESS IS GREAT ENOUGH
*  TO CLEAR THE VENT PIPE DOWNCOMERS
*  PDCVP=PRESS DIFF. NECESSARY TO CLEAR THE VENT PIPES
*
      WCSPG=WWCSPG+WVCSPG
      WVCSPG=(MSSPG/TAUFSP)*(1.-100./HUMSP)*COMPAR(HUMSP,100.)
*  WWCSPG=RATE OF COND. ON SPG WALL
*  WVCSPG=RATE OF COND. FROM SPG VOLUME
*
*  DRYWELL MASS BALANCE:
      MSDWG=INTGRL(MSDWGO,WSDWR+WSSPDW-WSDWSP-WSDWL-WCDWG-WCDWC)
      MHDWG=INTGRL(MHDWGO,WHDWR+WHSPDW-WHDWL-WHDWSP)
      MNDWG=INTGRL(MNDWGO,WNSPDW-WNDWSP-WNDWL)
      MMDWG=INTGRL(MMDWGO,WMDWR+WMSPDW-WMDWSP-WMDWL)
*
      WCDWG=WWCDWG+WVCDWG
      WVCDWG=(MSDWG/TAUFDW)*(1.-100./HUMDW)*COMPAR(HUMDW,100.)
*
*  HYDROGEN AND MISC RELEASES SET TO ZERO
*  FFRACT=FRACTION OF HOT LIQUID LEAK THAT FLASHES TO STEAM
      FFRACT=(AFGEN(HSATF,PST)-AFGEN(HSATF,PTDWG))/...
          (AFGEN(HSATG,PTDWG)-AFGEN(HSATF,PTDWG))
      WSDWR=WDLEAK*FFRACT
      HSDWR=AFGEN(HSATG,PTDWG)
      WHDWR=0.
      WMDWR=0.
      HHDWR=0.
      HMDWR=0.
*
* WWCDWG=RATE OF COND. ON DWG WALL
*  WVCDWG=RATE OF VOLUME COND. IN DRYWELL
*
*  ENERGY BALANCES FOR DW AND SP GAS
*
*  INPUT FROM OTHER CALCULATIONS:
*  HMDWR=ENTHALPY OF MISC. RELEASED DIRECT TO DRYWELL
*  HHDWR=ENTHALPY OF HYDROGEN RELEASED DIRECT TO DRYWELL
*  HHRV=ENTHALPY OF HYDROGEN IN REACTOR VESSEL
*  HSRV=ENTHALPY OF STEAM IN REACTOR VESSEL
*  HSDWR=ENTHALPY OF STEAM RELEASED DIRECT TO DRYWELL
*  HSTE=ENTHALPY OF STEAM FROM TURBINE EXHAUST
*  NOTE THAT THE ABOVE 3 ARE REF. TO ZERO AT 32.F WATER
*  (ASME STM TABLES), BUT ARE RE-REF TO ZERO
*  AT ZERO DEG-R FOR CONTAINMENT CALC.
*  QVSPG,QVDWG=VOLUME HEAT SOURCE (FROM F.P.'S)
*  QRVHL=HEAT LOSS (THROUGH INSULATION) FROM RV TO SP GAS
*  QLSPG=HEAT LOSS FROM SP GAS TO SP LINER
*  QLDWG=HEAT LOSS FROM DW GAS TO DW LINER
*
*  DEFINITIONS:
*
```

```
*  TGSP=TEMP OF HOMOGENIZED SP GAS (DEF-F)
*  TGDW=TEMP OF  HOMOGENIZED DW GAS (DEG-F)
*  UMSPG=TOTAL INTERNAL ENERGY OF SP GAS (BTU)
*  UMDWG=TOTAL INTERNAL ENERGY OF DW GAS (BTU)
*  MASS FLOWS DEFINED IN MASS BALANCE SECTION
*
*  SP GAS ENERGY BALANCE:
      CUMSPG=INTGRL(0.0,DUMSPG)
      UMSPG=UMSPGO+CUMSPG
 DUMSPG=EPSSP+EPHSP+EPNSP+EPMSP+QVSPG-QLSPG-MWORK+QSSPG
      EPSSP=WSSVNQ*(HSSVNQ)+WSTENQ*(HSTE+HSRREF)+...
      WSDWSP*(HSTGDW)-(WSSPDW+WSSPL)*HSTGSP-...
      WCSPG*(HFGSP+HSRREF)+WTESPW*HSTSPW
      MWORK=DVDTPG*PTSPG*.1851
      DVDTPG=(VGSP-VGSPD)/5.
      VGSPD=REALPL(VGSPO,5.,VGSP)
*  WSSVNQ=TOTAL SRV FLOW THAT EXITS SURFACE OF POOL
*  WSTENQ=TOTAL NON-QUENCHED STM FROM TURBINE EXHAUST
*  HSRREF=DELTA-H TO REREF. STEAM ENTHALPY
*  HFGDW=SAT FLUID ENTH AT DW TEMP(ASME)
*  HFGSP=SAT FLUID ENTH AT SP TEMP(ASME)
*  HSSVNQ=ENTHALPY OF NON-QUENCHED STEAM ENTERING SP GAS(REF TO 0.
*        AT 0. DEG-R)
      HSSVNQ=HST+HSRREF
      HSTGSP=.45*TGSPR-4.89
      HSTSPW=.45*(TWSPAV+460.)-4.89
      HSTGDW=.45*TGDWR-4.89
      HSDWCE=.45*TDWCER-4.89
*
*
      HFGSP=TGSP-32.
      HFGDW=TGDW-32.
*
      EPHSP=WHDWSP*HHTGDW+WHSV*HHRV-(WHSPDW+WHSPL)*HHTGSP
      HHTGSP=3.466*TGSPR-40.
      HHTGDW=3.466*TGDWR-40.
*
      EPNSP=HNTGDW*WNDWSP-(WNSPDW+WNSPL)*HNTGSP
      HNTGSP=.2475*TGSPR
      HNTGDW=.2475*TGDWR
*
      EPMSP=HMTGDW*WMDWSP-(WMSPDW+WMSPL)*HMTGSP
      HMTGSP=.21*TGSPR-20.8
      HMTGDW=.21 TGDWR-20.8
*
*  DRYWELL ENERGY BALANCE
*  DRYWELL COOLER CALCULATION:  ASSUMES ATMOSPHERE EXIT TEMP =TDWCE AND
*  BULK FLOW = CONSTANT AT COOLER EXIT=BDWC
      TDWCER=TDWCE+460.
      PSDWCE=LIMIT(0.,SPDWCE,PSDWG)
      SPDWCE=NLFGEN(SPFCT,TDWCE)
      WSDWCR=BDWC*PSDWCE/(GCS*TDWCER)
      WNDWC=BDWC*(PTDWG-PSDWCE)/(GCN*TDWCER)
      WSDWCS=WNDWC*MSDWG/MNDWG
      WCDWC=WSDWCS-WSDWCR
      CUMDWG=INTGRL(0.0,DUMDWG)
      UMDWG=CUMDWG+UMDWGO
      DUMDWG=EPSDW+EPHDW+EPNDW+EPMDW+QVDWG+...
      QRVHL-QLDWG
*
      QSSPG=(TWSPAV-TGSP)*ASSPW*5.3E-05*((ABS(TWSPAV-TGSP))**.33)
      QLSPG=QPMWET+QPMDRY
      QVDWG=0.
      QVSPG=0.
```

```
      QRVHL=QRVHLO*948.*(TSAT-TGDW)/DTRVHL
      QLDWG=QDM
*
*
      EPSDW=(HSDWR+HSRREF)*WSDWR+HSTGSP*WSSPDW-HSTGDW*(WSDWSP+WSDWL)-...
      HFGDW*WCDWG+WSDWCR*HSDWCE-WSDWCS*HSTGDW
*
      EPHDW=HHDWR*WHDWR+HHTGSP*WHSPDW-HHTGDW*(WHDWSP+WHDWL)
*
      EPNDW=HNTGSP*WNSPDW-HNTGDW*(WNDWSP+WNDWL)+.2475*WNDWC*(TDWCE-TGDW)
*
      EPMDW=HMDWR*WMDWR+HMTGSP*WMSPDW-HMTGDW*(WMDWSP+WMDWL)
*
*
*  SOLUTION FOR DW AND SP GAS TEMPERATURES
*  ITERATION IS NOT NECESSARY SINCE THE S,N,H,AND M
*  ENTHALPIES ARE ASSUMED LINEAR WITH TEMP:
*  H(N2)=.2475T, H(H2O)=.45T-4.89, H(M)=.21T-20.8, H(H)=3.466T-40.
*
      TGSPR=(UMSPG+4.89*MSSPG+20.8*MMSPG+40.0*MHSPG)/...
          (MNSPG*(.2475-.1851*GCN)+MSSPG*(.45-.1851*GCS)+...
          MMDWG*(.21-.1851*GCM)+MHSPG*(3.466-.1851*GCH) )
      TGDWR=(UMDWG+4.89*MSDWG+20.8*MMDWG+40.*MHDWG)/...
      (MNDWG*(.2475-.1851*GCN)+MSDWG*(.45-.1851*GCS)+...
      MMDWG*(.21-.1851*GCM)+MHDWG*(3.466-.1851*GCH))
      TGSP=TGSPR-460.
      TGDW=TGDWR-460.
*  TOTAL AND PARTIAL PRESSURES(PSIA) CALC. FROM TEMP(F)
      PNSPG=MNSPG*GCN*TGSPR/VGSP
      PHSPG=MHSPG*GCH*TGSPR/VGSP
      PSSPG=MSSPG*GCS*TGSPR/VGSP
      PMSPG=MMSPG*GCM*TGSPR/VGSP
      PTSPG=PNSPG+PHSPG+PSSPG+PMSPG
*
      PNDWG=MNDWG*GCN*TGDWR/VGDW
      PHDWG=MHDWG*GCH*TGDWR/VGDW
      PSDWG=MSDWG*GCS*TGDWR/VGDW
      PMDWG=MMDWG*GCM*TGDWR/VGDW
      PTDWG=PNDWG+PHDWG+PSDWG+PMDWG
*
*  HUMIDITIES AND DEWPOINTS
      HUMSP=100.*PSSPG/NLFGEN(SPFOT,TGSP)
      HUMDW=100*PSDWG/NLFGEN(SPFOT,TGDW)
      TDEWSP=NLFGEN(STFOSV,VGSP/MSSPG)
      TDEWDW=NLFGEN(STFOSV,VGDW/MSDWG)
*
*
      BDWL=VGDW*FLDWG
      WSDWL=BDWL*(MSDWG/VGDW)
      WHDWL=BDWL*(MHDWG/VGDW)
      WNDWL=BDWL*(MNDWG/VGDW)
      WMDWL=BDWL*(MMDWG/VGDW)
*
*  CALCULATION OF SP AND DW WALL METAL TEMP ASSUMPTIONS:
*      1.CONSIDER ONLY METAL SURFACE IN CONTACT WITH GAS
*      2.NO CONDENSING UNLESS METAL TEMP BELOW DEWPOINT
*  STM. CONDENSATION ON WALLS IS AIR-LIMITED AND IS CALCULATED
*  FROM EQUN. III.B.26 OF MARCH MANUAL(NUREG/CR-1711)
      QPM=QPMDRY+QPMWET-QPGAIR
      QPMDRY=(TGSP-TPMET)*5.3E-05*((ABS(TGSP-TPMET))**.3333)*APMET
     QPMWET=COMPAR(TDEWSP,TPMET)*(TDEWSP-TPMET)*APMET*...
          .0185*((MSSPG/MNSPG)**.707)
      TPMET=INTGRL(TPMETO,QPM/CPMET)
      QDM=QDMDRY+QDMWET
```

```
      QDMDRY=(TGDW-TDMET)*5.3E-5*((ABS(TGDW-TDMET))**.3333)*ADMET
      QDMWET=COMPAR(TDEWDW,TDMET)*(TDEWDW-TDMET)*ADMET*...
          .0185*((MSDWG/MNDWG)**.707)
      TDMET=INTGRL(TDMETO,QDM/CDMET)
*
*  CALC OF CONDENSATION RATE ON DW AND SP WALLS. APPROXIMATE VALUE
*  OF 900 BTU/LB IS USED FOR (HG-HF)
      WWCSPG=QPMWET/900.
      WWCDWG=QDMWET/900.
*
*  CALCULATION OF POOL ROOM AIR TEMP
      QPAIR=QPGAIR+QPWAIR
      QPWAIR=5.3E-05*((ABS(TWSPAV-TPAIR))**.3333)*APMET*(TWSPAV-TPAIR)
      QPGAIR=5.3E-5*((ABS(TPMET-TPAIR))**.333)*APMET*(TPMET-TPAIR)
      TPAIR=INTGRL(TPAIRO,QPAIR/CPAIR)
*
*  NOTE THAT THE TERM APMET IS THE SAME IN BOTH EQUATIONS
*  BECAUSE IT IS FOR ONE HALF OF TOTAL METAL SURFACE AREA
*
*  THESE CALCULATIONS ARE TO BE USED FOR AVERALL THERMO
*  CONSERVATION CHECK
      DMSP=INTGRL(0.,WSTC)
      HEATIN=INTGRL(0.,QTOT)
      DMHRV=INTGRL(0.,WINJ*HINJIN-WSTC*HST-WCRD*HLP)
*
      LINTVZ=LINT*12.+216.
 DUM=OUT(TMIN,LDCVZ,P,GP,GT,WCRD,HLP,WSTC,TWSPAV,LWSPAV,PTDWG,TGDW)
* DUM=OUT(TMIN,LDCVZ,P,GP,GT,WSTC,TWSPAV,LWSPAV,PTSPG,PTDWG,TGSP,TGDW)
*  OUT IS A SUBPROGRAM FOR WRITING SPECIFIED VARIABLES
*  ON DISC FOR LATER PLOTTING
      PRINT  LDCVZ,HRECIR,LSC,WCBP,P,WTOST,WSTC,WCID,HLP
      TIMER  DELT=.25, FINTIM=28810., PRDEL=10.
      METHOD RECT
      NOSORT
       CALL DEBUG(1,0.)
      CALL DEBUG(1,3600.)
       CALL DEBUG(1,5400.)
      CALL DEBUG(1,10800.)
      CALL DEBUG(1,17999.)
      CALL DEBUG(1,21600.)
      CALL DEBUG(1,25200.)
      CALL DEBUG(1,28800.)
      CALL DEBUG(1,36000.)
      CALL DEBUG(1,46800.)
      CALL DEBUG(1,50400.)
      CALL DEBUG(1,57600.)

END
```

$$$ TRANSLATION TABLE CONTENTS $$$	CURRENT	MAXIMUM
MACRO AND STATEMENT OUTPUTS	450	1200
STATEMENT INPUT WORK AREA	1388	3800
INTEGRATORS+MEMORY BLOCK OUTPUTS	44 + 0	300
PARAMETERS+FUNCTION GENERATORS	103 + 10	400
STORAGE VARIABLES+INTEGRATOR ARRAYS	0 + 0/2	50
HISTORY AND MEMORY BLOCK NAMES	21	50
MACRO DEFINITIONS AND NESTED MACROS	7	126
MACRO STATEMENT STORAGE	37	200
LITERAL CONSTANT STORAGE	0	100
SORT SECTIONS	2	20
MAXIMUM STATEMENTS IN SECTION	378	987

```
      FUNCTION DFVOID(A,WSATF,WSATG,TSAT,RHOF,RHOG)
      REAL JF,JG,K3
      DATA CO,K3/1.1,2.5/
      WF=AMAX1(WSATF,0.)
      JF=WF/(RHOF*A)
      JG=WSATG/(RHOG*A)
      SURTEN=(817.-TSAT)*(.0052)/767.
      VGJ=K3*((RHOF-RHOG)*SURTEN*1036.84/(RHOF*RHOF))**.25
      DFVOID=JG/(CO*(JF+JG)+VGJ)
      RETURN
      END
      FUNCTION CAVOID(XIN,XOUT,WTOTAL,TSAT,RHOF,RHOG)
      DATA  ACOR, XKVGJ/ 77., 2.5/
C  THIS FUNCTION RETURNS BOILING REGION AVERAGE VOID
C  MODIFIED DRIFT FLUX CORRELATION IS USED(CO=1.0, VGJ=0 AT X=1)
      W=AMAX1(WTOTAL,.01)
      XOMXI=AMAX1((XOUT-XIN),.001)
      XOUT=AMIN1(1.0,XOUT)
      G=W/ACOR
      TS=TSAT
      SURTEN=(817.-TS)*(.0052)/767.
      VGJ=XKVGJ*((RHOF-RHOG)*SURTEN*32.2*32.2/(RHOF*RHOF))**.25
      IF(XOMXI.GT. .01) GO TO 10
      X=(XIN+XOUT)/2.
      C1=RHOG*(1.-X)/RHOF+X
      C2=RHOG*VGJ*(1.-X)/G
      CAVOID=X/(C1+C2)
      GO TO 100
 10   CONTINUE
      A=RHOG/RHOF+RHOG*VGJ/G
      B=1.-A
      XLN=ALOG((A+B*XIN)/(A+B*XOUT))
      CAVOID=1./B+A*XLN/(B*B*XOMXI)
 100  RETURN
      END
      FUNCTION RHOTP(RHOF,RHOG,VOID)
      RHOTP=(1.-VOID)*RHOF+VOID*RHOG
      RETURN
      END
      FUNCTION VOID(X,W,A,TSAT,RHOF,RHOG)
C  THIS FUNCTION RETURNS VOID FRACTION AT ANY POINT
C  MODIFIED DRIFT FLUX CORRELATION USED(CO=1., VGJ=0 AT X=1)
      DATA XKVGJ/2.5/
      X=AMIN1(X,1.0)
      TS=TSAT
      WL=AMAX1(W,.01)
      SURTEN=(817.-TS)*(.0052)/767.
      VGJ=XKVGJ*((RHOF-RHOG)*SURTEN*32.2*32.2/(RHOF*RHOF))**.25
      C1=RHOG*(1.-X)/RHCF+X
      C2=RHOG*VGJ*(1.-X)/(WL/A)
```

```
      VOID=X/(C1+C2)
      RETURN
      END
       FUNCTION ANS(TIM)
C*** ORNL RCUTINE TO CALCULATE DECAY POWER BASED ON
C*** ANSI 5.1 - 1979, INCLUDING DECAY OF ACTINIDES
C*** U239 AND NP239
C
C*** USE ORIGINAL MARCH 1.1 FITS TO FISSION PRODUCT DECAY
C
      TAP=7.88E07
      T1=TIM
      T2=TAP
      T3=TIM+TAP
      IPASS=1
      T=T1
  5   CONTINUE
      IF(T.LE.1.0) GO TO 8
      IF(T.LE.10.0) GO TO 10
      IF(T.LE.150.0) GO TO 20
      IF(T.LE.4.E6) GO TO 30
      GO TO 40
  8   POWER=1.
      GO TO 50
 10   POWER=0.0603/T**0.0639
      GO TC 50
 20   POWER=0.0766/T**0.181
      GO TC 50
 30   POWER=0.1300/T**0.283
      GO TC 50
 40   POWER=0.266/T**0.335
 50   IF(IPASS.EQ.2) GO TO 60
      IPASS=2
      POWER1=POWER
      T=T3
      GO TO 5
 60   FP=POWER1-POWER
C
C*** NOW CALCULATE THE G CORRECTION FACTOR FOR NEUTRON CAPTURE IN
C*** FISSICN PRODUCTS
C
C******CAUTION******
C*** THESE CALCULATED G FACTORS MAY BE TOTC LARGE FOR T2=1.27E8
C*** AND T1=1.E5
      PSI=1.
      G=1.+(((3.24E-6+(5.23E-10*T1))*(T2**0.40))*PSI)
C
C*** NOW CORRECT FOR ACTINIDES U239 AND NP239, BASED ON
C*** EQS. 14 AND 15 FROM ANSI 5.1 - 1979
C
      Q=200.
      EU=.474
      R=.6
      XLAM1=4.91E-4
      EN=.419
      XLAM2=3.41E-6
C
       XPO=AMIN1(XLAM1*T2,10.)
       EXP1TT=EXP(-XPO)
      EXP1T=EXP(-XLAM1*T1)
      FU=EU*R*(1.-EXP1TT)*EXP1T
C
      XPO=AMIN1(XLAM2*T2,10.)
      EXP2TT=EXP(-XPO)
```

```
      EXP2T=EXP(-XLAM2*T1)
      ZLAM1=XLAM1/(XLAM1-XLAM2)
      ZLAM2=XLAM2/(XLAM1-XLAM2)
      TERM1=ZLAM1*(1.-EXP2TT)*EXP2T
      TERM2=ZLAM2*(1.-EXP1TT)*EXP1T
      FN=EN*R*(TERM1-TERM2)
      ACT=(FU+FN)/Q
C
      ANS=FP*G
      ANS=ANS+ACT
      ANS=AMIN1(ANS,1.)
      RETURN
      END
      FUNCTION XLENDC(V)
C  RETURNS HEIGHT OF WATER IN DOWNCOMER AS A FUNCTION OF TOTAL VOLUME
C  ZERO HEIGHT=BOTTOM OF ACTIVE FUEL
C
C  L1=0.0, L2=HEIGHT AT JET PUMP SUCTION INLET, L3=HEIGHT AT TOP OF CORE
C  OUTLET PLENUM, L4=HEIGHT AT BOTTOM OF STM SEPARATORS
C
C  VDIFF=JET PUMP AND DIFFUSER VOLUME BELOW B.O.A.F., VJET=VOL BET JP SUCTION
C  AND BOAF, VANN=DC VOL BET TOP OF CORE OUTLET PLENUM AND JP SUCTION,
C  VSSOP=VOL BET TOP OF OUTLET PLENUM AND BOT OF STM SEP
C
      DATA  VDIFF,VJET,VANN,VSSOP,VUV1,VSL,VUV2
     1 /192.,58.4,1044.,4434.7,1063.,1811.,2031./
      DATA  XL1,XL2,XL3,XL4,XL5,XL6,XL7
     1 /0.,8.,17.5,32.63,35.88,37.88,44.09/
      V1=VDIFF
      V2=V1+VJET
      V3=V2+VANN
      V4=V3+VSSOP
      V5=V4+VUV1
      V6=V5+VSL
      V7=V6+VUV2
      DL1=(V-V1)*(XL2-XL1)/(V2-V1)
      DL2=(V-V2)*(XL3-XL2)/(V3-V2)
      DL3=(V-V3)*(XL4-XL3)/(V4-V3)
      DL4=(V-V4)*(XL5-XL4)/(V5-V4)
      DL5=(V-V5)*(XL6-XL5)/(V6-V5)
      DL6=(V-V6)*(XL7-XL6)/(V7-V6)
      DL1=AMAX1(DL1,0.)
      DL1=AMIN1(DL1,XL2-XL1)
      DL2=AMAX1(DL2,0.)
      DL2=AMIN1(DL2,XL3-XL2)
      DL3=AMAX1(DL3,0.)
      DL3=AMIN1(DL3,XL4-XL3)
      DL4=AMAX1(0.,DL4)
      DL4=AMIN1(DL4,XL5-XL4)
      DL5=AMAX1(0.,DL5)
      DL5=AMIN1(DL5,XL6-XL5)
      DL6=AMAX1(0.,DL6)
      XLENDC=DL1+DL2+DL3+DL4+DL5+DL6
      RETURN
      END
      FUNCTION VOLDC(XLDCO)
      DATA  VDIFF, VJET, VANN, VSSOP/ 192., 58.4, 1044., 2125./
      DATA  XL1, XL2, XL3, XL4/ 0.,8.,17.5, 24.75/
      X=XLDCO
      DV1=((X-XL1)/(XL2-XL1))*VJET
      DV1=AMAX1(0.,DV1)
      DV1=AMIN1(DV1,VJET)
      DV2=((X-XL2)/(XL3-XL2))*VANN
      DV2=AMAX1(0.,DV2)
```

```
      DV2=AMIN1(DV2,VANN)
      DV3=((X-XL3)/(XL4-XL3))*VSSOP
      DV3=AMAX1(0.,DV3)
      VOLDC=DV1+DV2+DV3+VDIFF
 200  RETURN
      END
      FUNCTION PFVH(H,VS,P)
C   PFVH     STEAM PRESSURE FROM ENTHALPY AND SPECIFIC VOLUME
      DO 10 J = 1,10
      TP = TVAPH(P,H)
      VC = VVAP(TP,P)
      IF( ABS(VC-VS)    .LE.0.1D-4) GO TO 20
      P = P*VC/VS
   10 CONTINUE
      WRITE(6,101) VC,VS,TP,P
  101 FORMAT(1H ,' NON-CONVERGE IN PFVH, VC =',F8.5,' VS =',F8.5,
     1 ' T=',F7.2,' P=',F8.2)
   20 CONTINUE
      PFVH=P
      RETURN
      END
      FUNCTION QREGAV(L1,L2)
C
C  THIS SUBPROGRAM CALCALATES RATIO: (AVG FLUX FROM A TO B)/(AVG FLUX OVER
C  CORE)
C  SPECIFIC FOR HEATED LENGTH=12. FT
C
      REAL L1,L2,LINT
      DIMENSION P(13)
      DATA P/.61,1.04,1.16,1.1 ,1.16,1.11,1.09,1.07,
     1 1.05,1.03,0.92,0.72,0.33/
C
C SOLVE FOR FUNCTIONAL VALUES AT TOP AND BOTTOM OF REGION
C
      INT1=INT(L1)
      INT2= INT(L2)
      FRAC1=FLOAT(INT1+1)-L1
      FRAC2=L2-FLOAT(INT2)
          PL1=(1.-FRAC1)*P(INT1+2)+FRAC1*P(INT1+1)
      PL2=(1.-FRAC2)*P(INT2+1)+FRAC2*P(INT2+2)
C
C TEST TO SEE IF MORE THAN 1 NODE BOUNDARY IS INVOLVED
      NNOB=INT2-INT1
      IF(NNOB.GT.1)  GO TO 11
      QREGAV=(PL1+PL2)/2.
      RETURN
C
C IF 2 OR MORE NODE BOUNDARIES INVOLVED,THEN AVERAGE THE INERRIOR AND
C THE END SEGMINTS AND COMBINE THESE INTO A GRAND AVERAGE
 11   DUM=0.0
      NNODES=NNOB-1
      DO 13  J=1,NNODES
            JJ=J+INT1+1
            DUM=DUM+(P(JJ)+P(JJ+1))/2.
 13         CONTINUE
      LINT=FLOAT(NNODES)
      AVINT=DUM/LINT
      AVTOP=(P(INT2+1)+PL2)/2.
      AVBOT=(P(INT1+2)+PL1)/2.
      QREGAV=(AVBOT*FRAC1+AVINT*LINT+AVTOP*FRAC2)/(L2-L1)
      RETURN
      END
      FUNCTION VVAP(TF,PF)                                          VVAP 100
C  VVAP    SPECIFIC VOLUME OF STEAM FROM TEMP AND PRESSURE
```

```
CALCULATES SPECIFIC VOLUME OF STEAM                                  VVAP 102
C      KEENAN + KEYES, 1951, P. 15, EQ. 13                           VVAP 105
       P=PF/14.6959                                                  VVAP 110
       T=(TF+459.69)/1.8                                             VVAP 115
       TAU=1./T                                                      VVAP 120
       TAU2=TAU*TAU                                                  VVAP 125
       TAU3=TAU2*TAU                                                 VVAP 130
       TAU12=TAU3**4                                                 VVAP 135
       EXTAU=-(10.**(80870.*TAU2))                                   VVAP 140
       BO=EXTAU*TAU*2641.62+1.89                                     VVAP 145
       Q=BO*TAU                                                      VVAP 150
       QP=Q*P                                                        VVAP 155
       G1=(-1.6246D5*TAU+82.546)*TAU                                 VVAP 160
       G2=-1.2697D5*TAU2+.21828                                      VVAP 165
       G3=+6.768D64*TAU12*TAU12-3.635D-4                             VVAP 170
C      SIGN OF G3 REVERSED FROM K+K                                  VVAP 175
       B=(((G3*QP**9+G2)*QP*QP+G1)*QP+1.)*BC*.0160185                VVAP 180
       VVAP=.072964908*T/P+B                                         VVAP 185
       RETURN                                                        VVAP 190
       END                                                           VVAP 195
       FUNCTION TVAPH(PI,HI)                                         TVAPH100
C  TVAPH   TEMPERATURE OF STEAM FROM PRESSURE AND ENTHALPY
CALCULATES TEMPERATURE OF STEAM FROM PRESSURE AND ENTHALPY USING HVAP ANTVAPH102
C      LINEAR INTERPOLATION                                          TVAP2105
       P=PI                                                          TVAPH110
       H=HI                                                          TVAPH115
C      CHECK FOR SUPERHEAT(ADDED ON 8-12-71)
       T = TSAT1(P)
       H1 = HVAP(T,P)
       IF(H1.GE.H) GO TO 10
C      START WITH ABOUT 50 DEGREES SUPERHEAT                         TVAPH120
       DELT=40.                                                      TVAPH125
       T=8574./(15.424-ALOG(P))-410.
       H1=HVAP(T,P)                                                  TVAPH135
       IF(H1-H)2,10,1                                                TVAPH140
C    1 USE LESS SUPERHEAT                                            TVAPH145
     1 DELT=-DELT                                                    TVAPH150
     2 T=DELT+T                                                      TVAPH155
     5 HO=H1                                                         TVAPH160
       H1=HVAP(T,P)                                                  TVAPH165
       DELT=(H1-H)*DELT/(HO-H1)                                      TVAPH170
       T=DELT+T                                                      TVAPH175
       IF( ABS(DELT)-.01)10,10,5
    10 TVAPH=T                                                       TVAPH185
       RETURN                                                        TVAPH190
       END                                                           TVAPH195
       FUNCTION TSAT1(P)
C  TSAT1   SATURATION TEMP FROM PRESSURE
CALCULATES SATURATION TEMPERATURE (DEG F) OF WATER, .178PSIA TO CRITICALTSAT 102
C      KEENAN + KEYES, 1951, P.14, EQ.11 + 12                        TSAT 105
       T=500.                                                        TSAT 110
       CORR=3.2                                                      TSAT 115
       FOP=ALOG(3206.37858/P)/2.30258509
       GOTO1                                                         TSAT 125
C    4 TEMP BELOW 140C = 284F                                        TSAT 130
     4 CORR=((1.1702379D-8*TK*TK+5.86826D-3)*TK+3.2437814)           TSAT 135
     1      /(2.1878462D-3*TK+1.)                                    TSAT 136
     1 TS=1165.09/(FOP/CORR+1.)                                      TSAT [illegible]
       IF( ABS(TS-T)-.01)10,10,2
     2 T=TS                                                          TSAT 150
       TK=(1165.09-T)/1.8                                            TSAT 155
       IF(P-52.418)4,4,3                                             TSAT 160
C    3 TEMP ABOVE 140C = 284 F                                       TSAT 165
     3 CORR=(((6.56444D-11*TK+7.515484D-9)*TK*TK+4.14113D-2)*TK      TSAT 170
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```
 1      +3.346313)/(1.+1.3794481D-2*TK)
      GO TO 1
   10 TSAT1=TS-459.69
      RETURN
      END
      FUNCTION HVAP(TF,PF)
C  HVAP    HEAT OF VAPORIZATION FROM TEMP AND PRESSURE
CALCULATES ENTHALPY OF STEAM REFERRED TO WATER AT 32F AND .08854 PSIA
C      KEENAN + KEYES, 1951, PP. 15-16, EQ 13-15
      COMMON /H2OQEN/ TOLD,POLD,TAU,TAU2,B0,Q,Q3,Q12,DB0,F0,F1,G1,F2,G2
     1,F3,G3
      P=PF/14.6959
      T=(TF+459.69)/1.8
      IF(T-TOLD)33,32,33
   32 IF(P-POLD)34,220,34
   33 TOLD=T
   34 POLD=P
      TAU=1./T
      TAU2=TAU*TAU
      TAU3=TAU2*TAU
      TAU12=TAU3**4
      EXTAU=-(10.**(80870.*TAU2))
      B0=EXTAU*TAU*2641.62+1.89
      Q=B0*TAU
      Q2=Q*Q
      Q3=Q2*Q
      Q12=Q3**4
      DB0=(372420.11*TAU2+1.)*EXTAU*2641.62
      F0=DB0*TAU+B0
      DG1=-3.2492D5*TAU+82.546
      G1=(-1.6246D5*TAU+82.546)*TAU
      F1=(G1*F0*2.+Q*DG1)*Q
      DG2=-2.5394D5*TAU
      G2=-1.2697D5*TAU2+.21828
      F2=(G2*F0*4.+Q*DG2)*Q3
      DG3=+1.62432D66*TAU12*TAU12*T
      G3=+6.768D64*TAU12*TAU12-3.635D-4
C     SIGNS OF G3 + DG3 REVERSED FROM K+K      AVOIDS - IN F3
      F3=(G3*F0*13.+Q*DG3)*Q12
  220 DHCT=(((P**9*F3/3.25+F2)*0.5*P*P+F1)*.5*P+F0)*P*4.3557D-2+.2158633
C     DHCT=CHANGE IN H AT CONSTANT T AS PRESSURE GOES FROM ZERO TO PF PL
C     CHANGE IN H AT 32F AS P=.08854 GOES TO P=0
      DHZP=((3.7783D-4*T+1.472)*T+ALOG(T)*47.8365)*0.43-300.420108
C     DHZP=CHANGE IN H AT ZERO PRESSURE AS TEMP. GOES FROM 32F TO TF
C     INTEGRAL(CP0*DT) EVALUATED AT 32F  = 300.420108
      HVAP=DHZP+DHCT+1075.8
C     H OF EVAPORATION AT 32F + .08854PSIA = 1075.8 BTU/LBM
      RETURN
      END
      FUNCTION OUT(T,LDC,P,GALI,GALT,WSSV,TSP,LWSP,PSP,PDW,TGP,TGD)
C  FOR ACTUAL 11 SYSTEM VARIABLES PRINTED SEE MAIN PROGRAM
      OUT=1.0
       INBET=10
       IF(T. GT . 180. ) INBET=30
      IF(T. GT. 360.) INBET=10
      TSEC=T*60.
      ITC=IFIX(TSEC+.01)
      IF(ITC.EQ.IITC) GO TO 200
      IF(ITC/INBET*INBET.NE.ITC) GO TO 200
      WRITE(20,100)     T,LDC,P,GALI,GALT,WSSV,TSP,LWSP,PSP,PDW,TGP,TGD
 100  FORMAT(F6.2,1X,F5.1,1X,2(F5.0,1X),F7.0,1X,F4.0,1X,
     1 F5.1,1X,F4.0,1X,F5.1,1X,F6.2,1X,F5.2,1X,F5.1)
 200  IITC=ITC
      RETURN
      END
```

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

 LABEL  BROWNS FERRY REACTOR BUILDING
 MACRO T,TC,F,H,P,WC=ME(TE,WA,WS,WW,WX,AC,V,TO,TCO,HO,PO,QHX)
* T=MIXED ATMOSPHERE TEMP
* TC=CONCRETE SURFACE TEMP
* ST=STEEL SURFACE TEMP
* F=FOG DENSITY(TOTAL MASS IN DROPLETS/REGION VOLUME)(MW/V)
* H=RELATIVE HUMIDITY
* P=TOTAL PRESSURE
* WC=TOTAL CONDENSATION RATE
* TE=WEIGHTED MIXTURE TEMP OF MASS INFLUX
* WA=TOTAL AIR FLOW ENTERING
* WS=TOTAL STEAM FLOW ENTERING
* WW=MASS FLOW RATE OF WATER INTO NODE IN FOG DROPLETS
* WX=TOTAL MASS FLOW OUT(INCLUDES FOG)
* AC=CONCRETE SURFACE AREA
* AS=STEEL SURFACE AREA
* TS=STEEL THICKNESS
* V=TOTAL FREE VOLUME
* TO=INITIAL VALUE OF ATMOSPHERE TEMP
* TCO=INITIAL VALUE OF CONCRETE SURFACE TEMP
* TSO=INITIAL VALUE OF STEEL TEMP
* HO=INITIAL VALUE OF HUMIDITY
* PO=INITIAL VALUE OF TOTAL PRESSURE
* QHX=NET HEAT EX. BET. ATMOSP. AND MISC.(POS. FOR A COOLER)
*
* INITIALIZATION
*
      PSO=(HO/100.)*NLFGEN(SPFOT,TO)
      MSO=V*PSO/(GCS*(TO+460.))
      MAO=V*(PO-PSO)/(GCA*(TO+460.))
      MHO=(MSO*CVS+MAO*CVA)*(TO+460.)
*
* CONSERVATION RELATIONS
*
*
* MASS
*
      CMS=INTGRL(0.,WS-WC-(MS/MT)*WX)
      CMA=INTGRL(0.,WA-(MA/MT)*WX)
      MS=MSO+CMS
      MA=MAO+CMA
      DMW=WW+COMPAR(H,100.)*(H-100.)*MS/100.-MW/TAUFOG-MW*WX/MT
      MW=INTGRL(0.,DMW)
      F=MW/V
      MT=MS+MA+MW
*
* ENERGY
*
      CMH=INTGRL(0.,ENTHP-Q)
      MH=MHO+CMH
      ENTHP=EP1-EP2-EP3
      EP1=(TE+460.)*(WS*CPS+WA*CPA)+WW*(TE-880.)
   EP2=(T+460.)*WX*(MS*CPS/MT+MA*CPA/MT)+(T-880.)*(MW*WX/MT+MW/TAUFOG)
      EP3=WWC*(T-880.)
* ENTHALPY OF WATER IS 970 BTU/LB LOWER THAN THAT OF STEAM AT 212
* ,THEREFORE HW=T-880
*
      Q=CQDRY+CQWET+QHX
      WWC=(CQWET)/HFG
      WC=WWC+MW/TAUFOG
*
* HEAT TRANSFER TO CONCRETE AND STEEL SURFACES
*
```

```
  TC,CQFDRY,CQFWET=CONCR(T,TCO,PS,PA)
      CQDRY=AC*CQFDRY
      CQWET=AC*CQFWET
*
* CALCULATION OF TEMPERATURE AND PRESSURE
      T=(MH+MW*1340.)/(MS*CVS+MA*CVA+MW)-460.
        PS=MS*GCS*(T+460.)/V
      PA=MA*GCA*(T+460.)/V
      P=PS+PA
      H=100.*PS/NLFGEN(SPFOT,T)
*
 ENDMACRO
 MACRO WEAZ,WESZ=FLO(PZ,PCOM,CZ,HZ,TZ)
*
* WEAZ=FLOW OF AIR FROM NODE Z TO NODE COM
* WESZ=FLOW OF STEAM FROM NODE Z TO NODE CCM
* PZ=TOTAL PRESSURE OF NODE Z
* PCOM=TOTAL PRESSURE OF NODE COM
* CZ=AZ*SQRT(9273.6/KZ)
* HZ,TZ=TEMP,HUMIDITY OF NODE Z
      RHOTZ=RHOSZ+RHOAZ
      PSZ=(HZ/100.)*NLFGEN(SPFOT,TZ)
      RHOSZ=PSZ/(GCS*(TZ+460.))
      RHOAZ=(PZ-PSZ)/(GCA*(TZ+460.))
      WEZ=COMPAR(PZ,PCOM)*CZ*SQRT(RHOTZ*ABS(PZ-PCOM))
      WEAZ=(RHOAZ/RHOTZ)*WEZ
      WESZ=(RHOSZ/RHOTZ)*WEZ
 ENDMACRO
 MACRO TCON,CD,CW=CONCR(TBULK,TCO,PSTM,PAIR)
*
* THIS ROUTINE CALCULATES HEAT SINK TEMPS AS WELL AS HEAT FLUXES DUE
* TO NATURAL CIRC. AND CONDENSATION. THE CONCRETE IS DIVIDED INTO TWO
* SLAB GEOMETRY REGIONS .
*
* ASWR=AIR TO STEAM WEIGHT RATIO
* BATS=BREAK POINT FOR CONDENSATION COEFFICIENT CORRELATION
* DXCON1=THICKNESS OF OUTER CONCRETE NODE(FT)
* DXCON2=THICKNESS OF INNER CONCRETE NODE
* HNAT=NATURAL CIRCULATION HEAT TRANSFER COEFFICIENT(BTU/SEC/SQ.FT/DE
* HCOND=CONDENSATION HEAT TRANSFER COEFF.(BTU/SEC,SQ.FT/DEG-F)
*
* HEAT TRANSFER COEFFICIENT CALCULATIONS
      SVSTM=GCS*(TBULK+460.)/PSTM
      TDEW=NLFGEN(STFOSV,SVSTM)
      ASWR=PAIR*GCS/(PSTM*GCA)
      TNAT=AMAX1(TCON,TDEW)
      DTNAT=AMAX1(.01,ABS(TBULK-TNAT) )
      HNAT=5.28E-05*(DTNAT**.3333)
      BATS=COMPAR(ASWR,20.)
    HCONDX=BATS*(3.33E-03-ASWR*5.56E-05)+(1.-BATS)*(.0185/(ASWR**.707))
      HCOND=LIMIT(5.56E-04,.278,HCONDX)
*
* WALL HEAT FLUX TERMS
      CD=HNAT*(TBULK-TNAT)
      CW=HCOND*AMAX1(0.,TDEW-TCON)
      TCON=TCON1
*
* CONCRETE TEMP CALCULATIONS
      SHC1=DXCON1*RHOCON*CPCON
      SHC2=DXCON2*RHOCON*CPCON
      DTCON1=(CD+CW-FLXKON)/SHC1
      DTCON2=FLXKON/SHC2
      TCON1=INTGRL(TCO,DTCON1)
      TCON2=INTGRL(TCO,DTCON2)
```

```
      FLXKCN=(2.*KCON*(TCON1-TCON2)/(DXCON1+DXCON2))
      TWAVG=(TCON1*DXCON1+TCON2*DXCON2)/(DXCON1+DXCON2)
*
 ENDMACRO
 INITIAL
*
* REACTOR BUILDING PARAMETERS
*
* ACB=TOTAL SURFACE AREA OF REACTOR BLDG HEAT SINK CONCRETE(SQ.FT.)
* ACR=SURFACE AREA(SQ.FT.) OF REFUELING REGION CONCRETE HEAT SINK
* ASB=                                                     STEEL
* ASR=SURFACE AREA OF STEEL HEAT SINK IN REFUELING REGION
*  A0,A1,A2=COEFFICIENTS IN THE SECOND DEGREE POLYNOMIAL FIT OF
*      SGT HEAD VS FLOW CURVE: HEAD(INCHES W.G.)=A0+A1*(KCFM)+..
*      A2*(KCFM**2)
* BULKBR=BULK INFILTRATION FLOW (CFS) INTO REACTOR BLDG AT
*         REFERENCE PRESSURE DIFFERENCE DPINR
* BULKRR=REFUELING REGION INFILTRATION FLOW(CFS) AT REFERENCE
*   DELTA-P(DPINR)
* CBRP=FLOW COEFFICIENT FOR REACTOR BLDG. RELIEF PANELS
* CRRP=SAME FOR FEFUELING REGION
* CRSUCK=FLOW COEFFICIENT FOR SGT DUCTION BETWEEN REFUELING REGION AND
*      SGT FILTER/BLOWER INLET HEADER
* CBSUCK=TOTAL FLOW COEFFICIENT(FT**2) BETWEEN REACTOR BLDG AND SGT FAN
*        SUCTION: A*SQRT(9273.6/K), WHERE K=NUMBER OF VELOCITY HEADS
*          LOST, REFERENCED TO AREA A
* CEXH=FLOW(LOSS) COEFFICIENT FOR ALL DUCTWORK(DOES NOT INCLUDE
*      FILTER BANK) BETWEEN SGT BLOWER EXIT AND EXHAUST STACK OUTLET
* CMXBVR=MAX FLOW COEFFICIENT(FT**2) FOR FULLY OPEN BUILDING VACUUM
*        RELIEF VALVE: A*SQRT(9273.6/K)
* CPA,CVA=AIR SPECIFIC HEATS(BTU/LB*DEG-F)
* CPS,CVS=STEAM SPECIFIC HEATS(BTU/LB*DEG-F)
* DPWFBR=PRESSURE DROP ACROSS FILTER BANK(INCHES W.G.) AT REFERENCE
*      FLOW PER BANK, QFBR
* DPINR=REFERENCE PRESSURE DIFF(PSID) FOR INLEAKAGE FLOW BULKBR
* DPIVR=PRESSURE DIFFERENCE(POA-PB) AT WHICH REACTOR BLDG RELIEF
*       STARTS TO OPEN
* DPBRP=PRESSURE DIFFERENCE(PSID) NECESSAARY TO BLOW THE BUILDING
*       RELIEF PANELS
* DPRRP=SAME FOR REFUELING REGION RELIEF PANELS
* GCA,GCS=PERFECT GAS CONSTANTS FOR AIR AND STEAM(PSIA*CU.FT/LB*DEG-R)
* HBO=INITIAL BLDG HUMIDITY(PERCENT)
* HRO=INITIAL REFUELING REGION HUMIDITY
* HOA=OUTSIDE AIR HUMIDITY(PERCENT)
* PBVR=PROP BAND FOR OPENING BLDG VACUUM RELIEF(PSID)
* PBO=INITIAL BUILDING TOTAL PRESSURE(PSIA)
* PRO=INITIAL REFUELING REGION TOTAL PRESSURE(PSIA)
* POA=OUTSIDE AIR PRESSURE(PSIA)
* QFBR=REFERENCE FILTER BANK FLOW(FT**3/SEC)
* STB=BUILDING STEEL HEAT SINK THICKNESS
* STR=THICKNESS OF STEEL HEAT SINK IN REFUELING REGION
* TAUFOG=AVERAGE RESIDENCE TIME OF FOG DROPLETS IN REACTOR BLDG.
* TBLOR=AIR TEMPERATURE AT WHICH BLOWER HEAD:FLOW CURVE IS SPECIFIED
* TBO=INITIAL BUILDING ATMOSPHERE TEMP(F)
* TCO=INITIAL BUILDING CONCRETE HEAT SINK SURFACE TEMP(F)
* TOA=OUTSIDE AIR TEMP
* TRO=INITIAL REFUELING REGION ATMOSPHERE TEMP(DEG-F)
* TRCO=INITIAL REFUELING REGION CONCRETE TEMP
* TRSO=INITIAL REFUELING REGION STEEL HEAT SINK TEMP
* TSO=INITIAL BUILDING STEEL HEAT SINK SURFACE TEMP(F)
* VTR=FREE VOLUME(CU.FT.) OF REFUELING REGION ATMOSPHERE
* VTB=TOTAL REACTOR BLDG FREE VOLUME
      CONSTANT ACB=1.88E05,ASB=0.0,BULKBR=20.65,CBSUCK=163.
      CONSTANT  CMXBVR=475.
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```
      CONSTANT   CPA=.24,CVA=.17,CPS=.45,CVS=.34
      CONSTANT   DPINR=8.98E-3,DPIVR=.0106,PBVR=8.98E-3
      CONSTANT   GCA=.37,GCS=.586
      CONSTANT   HBO=70.,HOA=70.,PBO=14.682,POA=14.7
      CONSTANT   STB=.02,TAUFOG=300.,TBO=80.,TOA=70.,TCO=80.,TSO=80.
      CONSTANT   VTB=1.5E06
      CONSTANT   HFG=971.
     CONSTANT RHOCON=144.,CPCON=.2,KCON=1.5E-04,DXCON1=.17,DXCON2=.5
      CONSTANT DPRRP=.35,DPBRP=.25,BULKRR=104.
      CONSTANT CBRP=1000.,CRRP=1000.,CRSUCK=167.
      CONSTANT ACR=0.,ASR=1.1E05,STR=.0062,VTR=2.6E06
      CONSTANT TRO=80.,TRCO=80.,TRSO=80.,HRO=70.,PRO=14.682
      CONSTANT RHOSTE=490.,CPSTEE=.11
*
* EXHAUST FLOW CONSTANTS
      CONSTANT AO=16.,A1=.41,A2=-.0813,TBLOR=70.
      CONSTANT   DPWFBR=6.,QFBR=150.,CEXH=400.
      PCON1=62.1*GCA*(TBLOR+460.)/(12.*14.7)
      BO=PCON1*AO
      B1=PCON1*A1*60./1000.
      B2=PCON1*A2*60.*60./(1000.*1000.)
      AFB=DPWFBR*62.1/(QFBR*12.*144.)
      CONSTANT ACX=0.,ASX=0.,STX=1.,VTX=1.E04,PXO=14.691
*  CONSTANTS FOR ELEVATION 565 CALCULATION
      CONSTANT FF65=.2,FC65B=20000.,AC65=40000.,VT65=3.8E05
* CONSTANTS FOR 519 AND 593 ELEVATIONS
      CONSTANT   FF19=.2,FF93=.6,R1993=.33,FF6593=.75
      CONSTANT   AC93=8.63E04,AC19=5.70E04,VT93=6.8E05,VT19=4.5E05
* CONSTANTS FOR THE 519 CONCRETE FLOOR TEMP CALCULATION
      CONSTANT   DXCB1=.17,DXCB2=.33,DXCB3=.66,DXCB4=1.5,ABMW=1.67E04
      CONSTANT   MFILBA=1.04E06,MFILCO=2.01E05,ABMCOR=3248
      ABMNCO=ABMW-ABMCOR
      FXCB=2.*KCON/(DXCB1+DXCB2)
      FXCB2=2.*KCON/(DXCB3+DXCB4)
      FXCB1=2.*KCON/(DXCB2+DXCB3)
      SHCB1=DXCB1*RHOCON*CPCON
      SHCB2=DXCB2*RHOCON*CPCON
      SHCB3=DXCB3*RHOCON*CPCON
      SHCB4=DXCB4*RHOCON*CPCON
      RWC=DXCB1/KCON
      MBMWO=500.
      EBMWO=(TBO-32.)*MBMWO
*
* FUNCTION SPFOT : SATURATION PRESSURE AS A FUNCTION OF TEMPERATURE
      FUNCTION SPFOT=59.3,.25,79.6,.5,101.7,1.,132.9,2.4,...
           160.5,4.8,185.6,8.5,203.9,12.5,...
           228.,20.,250.34,30.,288.2,56.,324.1,95.,363.5,160.,...
           381.8,200.,417.3,300.,444.6,400.,467.,500.,486.,600.
*
*
* FUNCTION STFOSV: SATURATION TEMP AS A FUNCTION OF SPECIFIC VOLUME
*
     FUNCTION STFOSV=2.83,363.5,4.65,324.1,7.65,288.2,12.2,257.6,...
       20.1,228.,31.2,204.,44.7,185.6,76.4,160.5,158.9,129.6,...
         333.6,101.7,641.5,79.6,1235.,59.3
*
 DYNAMIC
*
*TEST MODEL OF WHOLE REACTOR BLDG
*
* BOUNDARY CONDITIONS FOR INITIAL CHECKOUT
      WCRHY=STEP(35.)*23.46
      NOSORT
      CALL STMFLO(TIME,WCRHY,SFLAB,SNF)
```

```
       SORT
       WSFLAB=SFLAB
       WLNF=SNF
       TSFLAB=212.
* TEB=TEMP CF FLOW FROM OTHER NODES(MIXED)
* WESB=TOTAL STEAM FLOW INTO REACTOR BLDG FROM OTHER NODES
* WEAB=TOTAL AIR FLOW INTO REACTOR BLDG FRCM OTHER NODES
* WEWB=TOTAL FOG FLOW INTO REACTOR ABLDG FRCM OTHER NODES
       WESB=WTRB*RHOSR/RHOTR
       WEAB=WTRB*RHOAR/RHOTR
       TEB=TR
       WEWB=WTRB*FR/RHOTR
       WXB=CBSUCK*SQRT(RHOTB*ABS(PB-PSUCK))*COMPAR(PB,PSUCK)
       PAB=PB-PSB
       PSB=(HB/100.)*NLFGEN(SPFOT,TB)
       PSOA=(HOA/100.)*NLFGEN(SPFOT,TOA)
       PAOA=POA-PSOA
*
* RHOTB= TCTAL REACTOR BLDG DENSITY(STEAM+AIR+FOG)
       RHOTB=RHOSB+RHOAB+FB
       RHOSB=PSB/(GCS*(TB+460.))
       RHOAB=PAB/(GCA*(TB+460.))
*
* OA=OUTSIDE AIR
       RHOOA=RHOSOA+RHOAOA
       RHOSCA=PSOA/(GCS*(TOA+460.))
       RHOACA=PAOA/(GCA*(TOA+460.))
* BULKB=BUILDING IN- OR OUT-LEAKAGE(CFS)
       BULKB=BULKBR*((ABS(POA-PB)/DPINR)**.6666)
* WTVR=TOTAL VACUUM RELIEF FLOW
       WTVR=CBVR*SQRT(RHOOA*ABS(POA-PB))
       CBVR=CMXBVR*LIMIT(0.,1.,(POA-PB-DPIVR)/PBVR)
       WABVR=(RHOAOA/RHOCA)*WTVR
       WSBVR=(RHOSOA/RHOOA)*WTVR
       WABIN=BULKB*COMPAR(POA,PB)*RHOAOA
       WSBIN=(WABIN/RHOACA)*RHOSOA
* TSFLAB=TEMP OF STEAM LEAKED FROM RC SYSTEM
* WSFLAB=FLOW OF STEAM FROM RCS TO REACTOR BLDG
       WBEX=BULKB*COMPAR(PB,POA)*RHOTB
       WTXB=WBEX+WXB+WTBR
       WESTB=WSBVR+WSBIN+WESB+WSFLAB
       WEATB=WABVR+WABIN+WEAB
       WEWTB=WEWB
 NUM1=(WESB*CPS+WEAB*CPA+WEWTB)*(TEB+460.)+WSFLAB*CPS*(TSFLAB+460.)+...
  ((WSBVR+WSBIN)*CPS+(WABVR+WABIN)*CPA)*(TCA+460.)
       DENUM1=WESTB*CPS+WEATB*CPA+WEWTB
       TTEB=NUM1/AMAX1(DENUM1,1.E-06)-460.
*
* TTEP=NET MIXED TEEMP OF ALL INCOMING FLOW INTO REACTOR BLDG
* WEATB=TOTAL AIRK FLOW INTO REACTOR BLDG
* WESTB=TOTAL STEAM FLOW INTO REACTOR BLDG
*
* CALL TO ENERGY CONSERVATION MACRO
*
 TB,TC,FB,HB,PB,WTCB=ME(TTEB,WEATB,WESTB,WEWTB,WTXB,ACB,...
       VTB,TBO,TCO,HBO,PBO,0.)
*
* SOME PRESSURE DIFFERENCES IN INCHES W.G.
       DPBSGS=(PB-PSUCK)*27.87
*
* SOME FLOWS IN CFM
* BBSUCK=BULK(VOL) FLOW FROM BLDG TO SGT SUCTION
* BBEXIN=BULK BLDG EX- OR IN- FILTRATION
* BBVR=BULK BLKG VACUUM RELIEF
```

```
* BBSL=BULK RCS TO BLDG STEAM LEAK FLOW
      BBSUCK=WXB*60./RHOTB
      BBVR=(WTVR/RHOOA)*60.
      BBEXIN=60*BULKB
      BBSL=(WSFLAB/RHOFL)*60.
      RHOFL=PB/(GCS*(TSFLAB+460.))
* SOME TOTAL MASSES FOR CONSV. CHECK
      MWB=VTB*FB
       MAB=RHOAB*VTB
      MSB=RHOSB*VTB
      MTB=RHOTB*VTB
      MALOST=INTGRL(0.,WTXB*MAB/MTB)
      MSLOST=INTGRL(0.,WTXB*MSB/MTB)
      MWLOST=INTGRL(0.,WTXB*MWB/MTB+WTCB)
      EALOST=INTGRL(0.,(WTXB*MAB/MTB)*(TB+460.)*CPA)
      ESLOST=INTGRL(0.,(WTXB*MSB/MTB)*(TB+460.)*CPS)
      EWLOST=INTGRL(0.,(WTXB*MWB/MTB+WTCB)*(TB-880.))
*
*  PROGRAMMING FOR REFUELING FLOOR PRESSURE,TEMP, AND FLOW
*
* REACTOR BLDG TO REFUELING REGION FLOW
*
* BBRLOG=REACTOR BLDG RELIEF PANEL LOGIC(I.E. THE PANELS BLOW
*     WHEN PRESSURE DIFFERENTIAL REACHES DPBRP)
      BRPLOG=RST(-99.,PB-PR-DPBRP,0.)
      WTBR=CBRP*BRPLOG*SQRT(RHOTB*ABS(PB-PR))*COMPAR(PB,PR)
      WTRB=CBRP*BRPLOG*SQRT(RHOTB*ABS(PB-PR))*COMPAR(PR,PB)
      WABR=WTBR*RHOAB/RHOTB
      WSBR=WTBR*RHOSB/RHOTB
      WWBR=WTBR*FB/RHOTB
*
* REFUELING REGION IN/EX-FILTRATION AND VACUUM RELIEF
      BULKR=BULKRR*((ABS(POA-PR)/DPINR)**.6666)
      WARIN=BULKR*COMPAR(POA,PR)*RHOAOA
      WSRIN=(WARIN/RHOAOA)*RHOSOA
      WTREX=RHOTR*BULKR*COMPAR(PR,POA)
* VACUUM RELIEF BEGINS AT DPINR AND GOES TO MAX OPENING
*   OVER A PBVR RANGE
      CRVR=CMXBVR*LIMIT(0.,1.,(POA-PR-DPIVR)/PBVR)
      WTRVR=CRVR*SQRT(RHOOA*ABS(POA-PB))
      WARVR=(RHOAOA/RHOOA)*WTRVR
      WSRVR=(RHOSOA/RHOOA)*WTRVR
*
* FLOW TO/FROM SGT SUCTION HEADER
      RSULOG=COMPAR(PR,PSUCK)
      WTXR=CRSUCK*SQRT(RHOTB*ABS(PR-PSUCK))*(1.-RSULOG)
      WAXR=WTXR*RHOAB/RHOTB
      WSXR=WTXR*RHOSB/RHOTB
      WWXR=WTXR*FB/RHOTB
      WTRX=CRSUCK*SQRT(RHOTR*ABS(PR-PSUCK))*RSULOG
*
* FLOW THRU REFUELING REGION BLOW-OUT PANELS
      RRPLOG=RST(-99.,PR-POA-DPRRP,0.)
      WTRR=CRRP*RRPLOG*SQRT(RHOTR*ABS(PR-PCA))
      WTEORP=WTRR*COMPAR(POA,PR)
      WAEORP=WTEORP*RHOAOA/RHOOA
      WSEORP=WTEORP*RHOSOA/RHOOA
*
* TOTAL FLOWS TABULATED
      WEATR=WABR+WARIN+WAEORP+WARVR+WAXR
      WESTR=WSBR+WSRIN+WSEORP+WSRVR+WSXR
      WEWTR=WWBR+WWXR
      WTOUTR=WTREX+WTRX+WTRR+WTRB
*
```

```
* AVG MIXED INLET TEMP CALCULATED
      NUM2=((WABR+WAXR)*CPA+(WSBR+WSXR)*CPS+WEWTR)*(TB+460.)+...
       (TOA+460.)*((WARIN+WARVR+WAEORP)*CPA+(WSRIN+WSEORP+WSRVR)*CPS)
      DENUM2=(WEATR*CPA+WESTR*CPS+WEWTR)
      TTER=NUM2/AMAX1(DENUM2,1.E-06)-460.
*
* CALL TO ME(MASS/ENERGY CONSERVATION MACRC)
  TR,TRC,FR,HR,PR,WTCR=ME(TTER,WEATR,WESTR,WEWTR,WTOUTR,ACR,...
          VTR,TRO,TRCO,HRO,PRO,0.)
*
* TOTAL REFUELING DENSITY CALCULATION
      PSR=(HR/100.)*NLFGEN(SPFOT,TR)
      PAR=PR-PSR
      RHOSR=PSR/(GCS*(TR+460.))
      RHOAR=PAR/(GCA*(TR+460.))
     RHOTR=RHOSR+RHOAR+FR
*
* REFUELING REGION UNITS TRANSFORMS FOR OUTPUT
      DPRSGS=(PR-PSUCK)*27.87
      BRERU=WTBR*60./RHOTB-WTRB*60./RHOTR
      BREXIN=60.*BULKR
      BBRVR=(WTRVR/RHOOA)*60.
      BRSUCK=((WTRX-WTXR)/(RHOTB*(1.-RSULOG)+RHOTR*RSULOG))*60.
      DPBR=(PB-PR)*27.87
      UTOTB=(MAB*CVA+MSB*CVS)*(TB+460.)+MWB*(TB-880.)
*
*
* PROGRAMMING FOR SGT VENT DUCTING NODE
*
* TOTAL INLET AND OUTLET FLOWS TABULATEC
      PSUCK=PX
      WEATX=WTRX*RHOAR/RHOTR+WXB*RHOAB/RHOTB
      WESTX=WTRX*RHOSR/RHOTR+WXB*RHOSB/RHOTB
      WEWTX=WTRX*FR/RHOTR+WXB*FB/RHOTB
      WTOUTX=WEXH*RHOTX/RHOEXH
*
* MIXED INLET TEMP CALCULATED
      NUM3=(CPA*RHOAB+CPS*RHOSB+FB)*WXB*(TB+460.)/RHOTB+...
        (CPA*RHOAR+CPS*RHOSR+FR)*WTRX*(TR+460.)/RHOTR
      DENUM3=WEATX*CPA+WESTX*CPS+WEWTX
      TTEX=NUM3/AMAX1(DENUM3,1.E-06)-460.
*
* CALL TO ME(MASS AND ENERGY CONSERVATION MACRO)
      TX,TXC,FX,HX,PX,WTCX=ME(TTEX,WEATX,WESTX,WEWTX,...
        WTOUTX,ACX,VTX,TBO,TBO,HBO,PXO,0.)
      PSX=(HX/100.)*NLFGEN(SPFOT,TX)
      PAX=PX-PSX
      RHOSX=PSX/(GCS*(TX+460.))
      RHOAX=PAX/(GCA*(TX+460.))
      RHOTX=FX+RHOSX+RHOAX
* FLOW CALCULATION FOR SGT EXHAUST FLOW
*
*
      NTRAIN=3.
      MUMUL=.91*MFAX+.45*MFSX+TX*(1.29E-03*MFAX+1.19E-03*MFSX)
      MFAX=PAX/PX
      MFSX=PSX/PX
      RHOEXH=RHOAX+RHOSX
      CQ=PX-POA+RHOEXH*B0/144.
      BQ=B1/(144.*NTRAIN)-MUMUL*AFB/(RHOEXH*NTRAIN)
      AQ=B2/(144.*RHOEXH*NTRAIN*NTRAIN)-1./(RHOEXH*CEXH*CEXH)
      R1=(-BQ+SQRT(BQ*BQ-4.*AQ*CQ))/(2.*AQ)
      R2=(-BQ-SQRT(BQ*BQ-4.*AQ*CQ))/(2.*AQ)
      WEXH=AMAX1(R1,R2)
```

```
      BSTACK=WEXH*60./RHOEXH
      DPBANK=27.87*MUMUL*AFB*WEXH/(RHOEXH*NTRAIN)
      DPBOD=27.87*WEXH*WEXH/(RHOEXH*CEXH*CEXH)
      DPBLOW=27.87*(RHOEXH*B0/144.+B1*WEXH/(144.*NTRAIN)+...
          B2*WEXH*WEXH/(144.*RHOEXH*NTRAIN*NTRAIN))
      BBEX=WBEX*60./RHOTB
      BBBOP=60.*WTBR/RHOTB
      DPROA=(PR-POA)*27.87
      DPBOA=(PB-POA)*27.87
      TMIN=TIME/60.
    DUM=OUT(TMIN,WSFLAB,H19,H65,H93,HR,DPBOA,...
          DPROA,T19,T65,T93,TR,DEEPC,DEEPNC)
*
* OFF-LINE CALCULATIONS TO ESTIMATE TEMPERATURES AND HUMIDITIES OF
* THE 3 DIFFERENT COMPARTMENTS OF THE REACTOR BUILDING: BASEMENT ROOM
* (BETWEEN 519 AND 565 ELEVATIONS), THE 565 ELEVATION ROOM, AND THE
* COMPOSITE ROOM(ACTUALLY 3 FLOORS) BETWEEN 593 AND 664 ELEVATIONS
*
* FIRST EXPRESSIONS FOR FLOWS BETWEEN 3 ROOMS
*
      BT65B=(P65-PB)*FC65B*COMPAR(P65,PB)
      WT65B=BT65B*RHOT65
      WS6593=WT65B*FF6593*RHOS65/RHOT65
      WA6593=WS6593*RHOA65/RHOS65
      WW6593=WT65B*FF6593*F65/RHOT65
      WS6519=R1993*WS6593
      WA6519=R1993*WA6593
      WW6519=WW6593*R1993
      BTB65=(PB-P65)*FC65B*COMPAR(PB,P65)
      WS9365=BTB65*RHOS93*FF6593
      WS1965=BTB65*RHOS19*(1.-FF6593)
      WA9365=BTB65*RHOA93*FF6593
      WA1965=BTB65*RHOA19*(1.-FF6593)
      WW9365=BTB65*F93*FF6593
      WW1965=BTB65*F19*(1.-FF6593)
*
*
*  PROGRAMMING FOR CALCULATION OF RB ROOM WITH FLOOR AT 565
      WTOU65=WT65B+FF65*(WXB+WTBR+WBEX)
      WEAT65=FF65*(WABVR+WABIN+WEAB)+WA9365+WA1965
      WEST65=FF65*(WSBVR+WSBIN+WESB)+WSFLAB+WS9365+WS1965
      WEWT65=FF65*(WEWB)+WW9365+WW1965
      NUM65=(WESB*CPS+WEAB*CPA+WEWTB)*FF65*(TEB+460.)+...
        WSFLAB*CPS*(TSFLAB+460.)+(TOA+460.)*...
          ((WSBVR+WSBIN)*CPS+(WABVR+WABIN)*CPA)*FF65+...
    (T93+460.)*(WS9365*CPS+WA9365*CPA+WW9365)+...
      (T19+460.)*(WS1965*CPS+WA1965*CPA+WW1965)
      DNUM65=WEST65*CPS+WEAT65*CPA+WEWT65
      TTE65=NUM65/AMAX1(DNUM65,1.E-06) - 460.
    T65,T65C,F65,H65,P65,WTC65=ME(TTE65,WEAT65,...
    WEST65,WEWT65,WTOU65,AC65,VT65,TB0,TC0,HB0,PB0,0.)
      PA65=P65-PS65
      PS65=(H65/100.)*NLFGEN(SPFOT,T65)
      RHOA65=PA65/(GCA*(T65+460.))
      RHOS65=PS65/(GCS*(T65+460.))
      RHOT65=F65+RHOA65+RHOS65
      DP65B=(P65-PB)*27.87
*
* PROGRAMMING FOR 519(BASEMENT) LEVEL
*
      WEAT19=FF19*(WABVR+WABIN+WEAB)+WA6519
      WEST19=FF19*(WSBVR+WSBIN+WESB)+WS6519+WEVAP
      WEWT19=FF19*(WEWB)+WW6519
      NUM19=(WESB*CPS+WEAB*CPA+WEWTB)*FF19*(TEB+460.)+...
```

```
      (TOA+460.)*((WSBVR+WSBIN)*CPS+(WABVR+WABIN)*CPA)*FF19+...
      (T65+460.)*(WS6519*CPS+WA6519*CPA+WW6519)+...
      WEVAP*CPS*(TBMW+460.)
      DNUM19=WEST19*CPS+WEAT19*CPA+WEWT19
      TTE19=NUM19/AMAX1(DNUM19,1.E-06) - 460.
      FEX19=LIMIT(0.,2.5,1.+(P19-PB)*20.)
       WTOU19=RHOT19*BTB65*(1.-FF6593)+FF19*(WXB+WTBR+WBEX)*FEX19
      T19,T19C,F19,H19,P19,WTC19=ME(TTE19,WEAT19,...
   WEST19,WEWT19,WTOU19,AC19,VT19,TB0,TC0,HB0,PB0,QT19AT)
      QT19AT=Q19CLR-Q19AIR
      Q19CLR=9.01*AMAX1(0.,T19-95.)
      PA19=P19-PS19
      PS19=(H19/100.)*NLFGEN(SPFOT,T19)
      RHOA19=PA19/(GCA*(T19+460.))
      RHOS19=PS19/(GCS*(T19+460.))
      RHOT19=F19*RHOA19*RHOS19
      DP19B=27.87*(P19-PB)
*
* BASEMENT WATER
*
      MBMW=INTGRL(MBMW0,WLNF-WEVAP+WTCB-WSMP)
      DEEPBW=MBMW/(62.*ABMW)
      DOC=((MBMW-MFILBA)/(62.*ABMW))*COMPAR(MBMW,MFILBA)
      DELPC=LIMIT(0.,1.,MBMW/(62.*ABMCOR))+DOC
      DELPNC=LIMIT(0.,1.,(MBMW-MFILCO)/(62.*ABMNCO))+DOC
      EBMW=INTGRL(EBMW0,DEBMW)
      DEBMW=180.*WLNF-(TBMW-32.)*WSMP-Q19AIR-QBMC+...
          WTCB*(.5*TC+.5*TB-32.)-WEVAP*1180.
      TBMW=EBMW/MBMW+32.
      VPBW=NLFGEN(SPFOT,TBMW)
*
      WSMP=6.91
*
      HBMA=5.83E-05*(ABS(TBMW-T19))**.333
      Q19AIR=ABMW*HBMA*(TBMW-T19)
      WEVAP=(NMB/DNMB)*COMPAR(VPBW,PS19)
      NMB=ABMW*HBMA*12.3*(VPBW-PS19)*P19
      DNMB=(2.*PA19-VPBW)*(T19+460.)*(RHOS19+RHOA19)
*
      HBMC=1.667E-03*(ABS(TBMW-TBC1))**.25
      QFBMC=(TBMW-TBC1)*HBMC/(1.+RWC*HBMC)
      QBMC=ABMW*QFBMC
      FLX1T2=FXCB*(TBC1-TBC2)
      FLX2T3=FXCB1*(TBC2-TBC3)
      FLX3T4=FXCB2*(TBC3-TBC4)
      TBC1=INTGRL(TB0,(QFBMC-FLX1T2)/SHCB1)
      TBC2=INTGRL(TB0,(FLX1T2-FLX2T3)/SHCB2)
      TBC3=INTGRL(TB0,(FLX2T3-FLX3T4)/SHCB3)
      TBC4=INTGRL(TB0,FLX3T4/SHCB4)
*
*
*
*
* PROGRAMMING FOR 593 LEVEL
*
      WEAT93=FF93*(WABVR+WABIN+WEAB)+WA6593
      WEST93=FF93*(WSBVR+WSBIN+WESB)+WS6593
      WEWT93=FF93*(WEWB)+WW6593
      NUM93=(WESB*CPS+WEAB*CPA+WEWTB)*FF93*(TEB+460.)+...
     (TOA+460)*((WSBVR+WSBIN)*CPS+(WABVR+WABIN)*CPA)*FF93+...
      (T65+460.)*(WS6593*CPS+WA6593*CPA+WW6593)
      DNUM93=WEST93*CPS+WEAT93*CPA+WEWT93
      TTE93=NUM93/AMAX1(DNUM93,1.E-06)-460.
      FEX93=LIMIT(0.,2.5,1.+(P93-PB)*20.)
      WTOU93=FF6593*BTB65*RHOT93+(WXB+WTBR+WBEX)*FF93*FEX93
```

```
 T93,T93C,F93,H93,P93,WTC93=ME(TTE93,WEAT93,...
  WEST93,WEWT93,WTOU93,AC93,VT93,TB0,TC0,HB0,PP0,0.)
   PA93=P93-PS93
   PS93=(H93/100.)*NLFGEN(SPFOT,T93)
   RHOA93=PA93/(GCA*(T93+460.))
   RHOS93=PS93/(GCS*(T93+460.))
   RHOT93=F93+RHOA93+RHOS93
   DP93B=27.87*(P93-PB)
*
*
   PRINT DPBOA,DPROA,T19,T65,T93,DEEPC,F19,H65,H93
     TIMER  DELT=0.20, FINTIM=10., PRDEL=10.
   METHOD RECT
   NOSORT
    CALL DEBUG(1,0.)
    CALL DEBUG(1,500.)
    CALL DEBUG(-1,3600.)
    CALL DEBUG(-1,7200.)
   CALL DEBUG(-1,10800.)
    CALL DEBUG(-1,14100.)
   CALL DEBUG(1,17900.)
   CALL DEBUG(1,21600.)
   CALL DEBUG(1,25900.)
    CALL DEBUG(1,36000.)
    CALL DEBUG(1,43000.)
END
STOP
```

Appendix B

MARCH 1.1 APPLICATION TO BWR SEVERE-ACCIDENT ANALYSIS — AN ASSESSMENT

B.1 Introduction

Although MARCH 1.1 is a remarkably versatile and useful tool for many light-water reactor (LWR) accident analysis applications, its suitability for boiling-water reactor (BWR) accident sequence analysis is significantly compromised by its failure to consider many of the basic design and operating differences that distinguish BWRs from pressurized-water reactors (PWRs). Since receiving the original MARCH 1.1 IBM version from the Tennessee Valley Authority (TVA) in June 1981, Oak Ridge Natonal Laboratory (ORNL) has identified several unique problem areas associated with the application of MARCH to BWR accident analysis. The purpose of this appendix is to discuss the nature and extent of the identified problems. The reader should bear in mind that the contents of this appendix are not the result of a formal MARCH BWR application assessment, but rather the outgrowth of the severe-accident sequence analysis (SASA) MARCH BWR application experience at ORNL. The nature of the accompanying assessment is, therefore, reflective of the type of applications for which the code was utilized (i.e., severe accident sequence analysis). However, many of the problems discussed in this appendix can significantly influence the results of any BWR severe-accident evaluation effort, both for probabilistic risk assessment (PRA) or SASA applications. Because future MARCH BWR application experiences will undoubtedly reveal additional problem areas, the present discussion should be considered as both preliminary and incomplete; Ref. 1 gives further information regarding both general and PWR-related MARCH modeling concerns.

B.2 MARCH BWR Problem Classification System

In the course of ORNL's MARCH application efforts, a broad range of problems associated with modeling of BWR accidents has been identified. The types of problems include missing subsystem and component models, errors in modeling of system interactions and physical phenomena, and desirable improvements in the areas of general code organization and user options. In previous MARCH assessment work, Rivard et al.[1] adopted a problem classification framework based on seven attributes: problem type, importance, knowledge, adequacy of modeling, scrutability, improvement, and phase. While this classification system is both desirable and useful, the extremely limited nature of the present discussion precludes utilization of all seven attributes. For the purposes of this appendix, the following limited classification system is utilized:

Type	G = general, CNT = chemical/nuclear reactions, TR = transport phenomena, SD = structural deformations/failures, and SS = safety systems and equipment.
Importance	Magnitude of potential change in code result if limitation is remedied. Scale of 1 to 3 in increasing importance. Four indicates importance is unknown.
Knowledge	State of knowledge of the underlying physical properties, phenomena, systems, or interactions. Scale of 1 to 3 with decreasing state of knowledge. Zero indicates that attribute is not applicable.
Phase	Accident phase(s) to which the limitation has significant application: 0 - all phases; 1 - precore uncovery; 2 - core uncovery and melt; 3 - in-vessel recovery or accident termination; 4 - core melt in lower head; 5 - melt interaction in containment; 6 - containment response, systems; and 7 - fission product distribution.

Although subjective in nature, the ratings associated with these attributes represent a best-estimate attempt at problem qualification and are useful vehicles for conveying information.

B.3 MARCH BWR Problem Description and Classification

Tables B.1–B.5 contain a brief summary of the BWR-related MARCH modeling problems that have been identified to date. Each problem has been assigned a unique identifier based upon its "type" classification and is accompanied by best-estimate values for the previously described attributes.

A brief analysis of Tables B.1–B.5 will reveal that the majority of the identified problems are of the "Safety Systems and Equipment" (SS) type. Many of these problems are a result of the development history of the code. Most of the BWR-oriented models within MARCH appear to be "back fits" or attempts to modify coding that was previously developed for PWR applications.

Each of the concerns listed in Tables B.1–B.5 will be briefly discussed in the following sections.

B.4 MARCH BWR Modeling Concerns - General

Many "general" MARCH modeling concerns (e.g., those dealing with program structure and documentation) were previously identified in Sandia's assessment effort.[1] Those concerns will not be repeated here. In this category we have listed only two items. The plotting routine, which is supplied with the code, is incompatible with IBM computer systems because of its use of the CDC "extended core" option (G-IB). Secondly, the input

variable PRESS (namelist NLMACE) is incorrectly defined on page 3-56 of the MARCH manual and omitted entirely from the input data listing on page 4-10. PRESS is actually a 10 x 10 array, rather than a single variable as described on page 3-56. As utilized in MARCH (subroutine MACE), PRESS (J,K) is the minimum pressure difference [P(J)-P(K)] that must exist between compartments J and K before they are connected to allow mass and energy to flow from compartment J to compartment K. This variable is particularly useful for BWR modeling, because it allows the user to simulate the vent clearing pressure of the suppression pool downcomers as well as the wetwell to drywell vacuum breakers. Figure B.1 is a simplified schematic of a MK 1 drywell/wetwell containment system. In this example PRESS (1,2) should be set equal to the pressure equivalent of the water column of height h, and PRESS (2,1) would be set equal to the vacuum breaker opening pressure differential.

B.5 MARCH BWR Modeling Concerns — Chemical/Nuclear

Table B.2 contains a listing of modeling concerns related to the MARCH 1.1 treatment of chemical and nuclear reactions. The MARCH 1.1 decay heat subroutine ANSQ is based upon the 1971 version of American Nuclear Society (ANS) Standard 5.1 and does not include the decay heat contributions of heavy elements (such as ^{239}U and ^{239}NP) or the effects of neutron absorption in fission products (CNT-1B). Figure B.2 is a graph of the time-dependent decay heat level from MARCH 1.1 and the decay heat level predicted from the 1979 version of ANS Standard 5.1 with consideration of heavy element decay and fission product neutron absorption. It is clear from a comparison of the two curves that the MARCH 1.1 formulation significantly underpredicts the decay heat level relative to the current standard. Figure B.3 is a plot of the time-integrated difference between the two formulations as a function of transient time. Figure B.4 is a plot of the resulting error in MARCH 1.1 average core temperature for a BWR adiabatic heatup transient and is useful for estimating the impact of this error on MARCH accident analysis results. It is clear that for accident transient times >1000 s, MARCH 1.1 can significantly underpredict core temperatures.

The MARCH 1.1 user is limited to modeling a single fuel type (CNT-2B). This is an unrealistic situation for many reactors such as Browns Ferry Nuclear Plant Unit 1 (BFNP#1), which currently is loaded with three different types of fuel, each having different fuel pin diameter, cladding thickness, active fuel length, and fuel loading. It is difficult to assess the impact that this restriction may have on the validity of MARCH results, but it is clearly an undesirable limitation.

MARCH does not have the capability to model BWR fuel assembly shrouds and associated hardware such as that shown in Fig. B.5 (CNT-3B). The MARCH user must either ignore the shrouds altogether, or artificially lump the mass of shrouds and channel spacers into the cladding of the fuel pin. A typical BWR may contain almost 150,000 lb_m of miscellaneous zircaloy within the core region, compared with a UO_2 mass of ~350,000 lb_m. This situation is very dissimilar to that found in PWRs, which have relatively little miscellaneous metal within the core. Because the analyst's

only option for modeling this material is to add it to the actual fuel cladding inventory, the MARCH fuel pin cladding thickness must be artificially increased by almost 70%. A second undesirable result of this approximation is that the temperature of the zirconium structure within a given MARCH core node is always equal to that of the fuel. These assumptions can obviously have a great impact on the energy release and hydrogen production associated with in-core zirconium oxidation reactions.

B.6 MARCH BWR Modeling Concerns — Transport Phenomena

Table B.3 contains a summary of several concerns relating to MARCH's modeling of heat and mass transfer in BWRs. The MARCH models for heat transfer to BWR upper and lower reactor vessel internal structures are inadequate and in some cases inappropriate (TR-1B). Figure B.6 is a cross-sectional view of a typical BWR reactor vessel.* The major vessel internals are the control rod guide tubes, core shroud head and standpipes, steam separators, and steam dryers. Typical values for mass and heat transfer area of these structures are given in Table B.6. The MARCH user's manual[3] states that the heat transfer coefficient employed for upper internals heat transfer is the maximum of h_1 or h_2 expressed as

$$h_1 = 0.0144\ C_{pm} \frac{G^{0.8}}{d^{0.2}} \frac{\text{Btu}}{\text{h}\cdot\text{ft}^2\cdot{}^\circ\text{F}}\ , \tag{B.1}$$

and

$$h_2 = 0.2\,|T_1 - T_s|^{1/3} \frac{\text{Btu}}{\text{h}\cdot\text{ft}^2\cdot{}^\circ\text{F}}\ , \tag{B.2}$$

where

C_{pm} = specific heat of gas mixture Btu/lb_m°F,
G = mass flux, $\text{lb}_m/\text{ft}^2\cdot\text{h}$,
d = flow equivalent diameter, ft,
T_1 = gas temperature, °F,
T_s = structure temperature, °F.

The source of these correlations is not clear, but it is apparent that Eq. (B.1) is inappropriate for application to structures such as the BWR shroud head because of the equation's dependence on flow diameter d. As employed in the code, G is actually the combined steam and hydrogen break and safety relief valve (SRV) flow calculated by PRIMP. Equation (B.1) is therefore not valid for conditions in which break flow does not exit the reactor vessel via the main steam or SRV lines.

*See also Fig. 6.1.

Heat transfer from the reactor vessel to the drywell atmosphere is not modeled by MARCH (TR-2B). Measurements at the BFNP indicated that heat is transferred from the reactor vessel, and steam lines to the drywell atmosphere at an approximate rate of 586 kW (2×10^6 Btu/h) at standard operating conditions. For station blackout conditions, previous studies[3] have shown that heat transfer via this mechanism is a significant factor influencing the time at which drywell failure occurs.

The MARCH fuel-to-coolant heat transfer coefficient is a constant value derived from a steady-state energy balance using the initial fuel and coolant temperatures input by the user (TR-3B). Changes in this heat transfer coefficient due to flow regime changes are not considered, and rod-to-rod and rod-to-shroud radiative heat transfer are not considered (TR-4B).

Water is erroneously allowed to flow out the SRVs for cases in which the input parameter ABRK equals zero and the vessel water height is greater than YBRK (TR-5B). This is equivalent to setting the height of the SRVs equal to YBRK for cases in which there is no break and would result in erroneous SRV water flows in BWRs.

Finally, the pressure differential employed in subroutine PRIMP for calculation of break flows is incorrect for breaks outside containment (TR-6B). PRIMP always employs the pressure differential between the reactor vessel and the primary containment in its break flow calculations, regardless of the location of the actual break.

MARCH does not check for consistency of the user input values for the primary system steam volume VOLS and the reactor vessel water level HO (TR-7B). The reactor vessel water mass calculation in MARCH is incorrect for BWRs in which the initial collapsed water level HO is above the top of the core (TR-8B). MARCH does not utilize the user-supplied value for the primary system volume VOLP for transients and small-break accidents (TR-9B). Rather, the code calculates a new value in subroutine PRIMP, based upon the calculated value of water mass and the input value of VOLS. Because of the nature of these problems, the analyst must input an incorrect value for either HO or VOLS to force MARCH to calculate the correct value for VOLP.

B.7 MARCH BWR Modeling Concerns — Structural Deformations/Failures

The four concerns listed in Table B.4 represent, quite probably, the four greatest areas of uncertainty associated with application of MARCH to BWR accident analysis. MARCH currently allows the user to select one of three core-melt models. Model A assumes that all heat in excess of that required to sustain the molten condition is transferred downward. Model B assumes that the excess heat is transferred upward. In both Model A and B, the geometry of the molten fuel remains unchanged until a user-specified fraction of the core (FDROP) is molten. The entire core is assumed to collapse when the core fraction FCOL is molten. Model C assumes that melted nodes immediately fall to the bottom of the pressure vessel.

Although Model A is believed to most closely approximate the behavior of the actual core, none of the MARCH models treat the fuel pin and shroud

melt/flow/freeze phenomena in a mechanistic fashion (SD-1B). The MARCH core collapse models are highly unrealistic for BWRs, because of their unique core support system (SD-2B). Figure B.7 is a drawing of the BWR core support system. The weight of each group of four fuel assemblies is independently supported by a single control rod guide tube and drive mechanism housing, which in turn transfers the weight of these four assemblies directly to the reactor vessel bottom head. Because of the design of this support system, it is extremely unlikely that a coherent core collapse would occur. Under these conditions, the use of simplistic core collapse criteria such as FDROP and FCOL is highly unrealistic. A closely associated problem is the fact that MARCH does not allow the molten core to attack the lower reactor vessel head until the entire core is collapsed (SS-3B). Because of the nature of the BWR core, it is quite common for MARCH to predict that the outermost zone of fuel remains intact long after the remainder of the core is molten. This phenomena is probably a realistic representation of an actual core meltdown accident. Because MARCH will not allow the molten core to attack the lower vessel head until all of the core is collapsed, the analyst must use an unrealistically low value of FCOL to allow MARCH to progress beyond the core-melt stage.

Figure B.8 is a detailed drawing of one of the 185, 15.2-cm-diam (6-in.) control rod guide tubes and drive mechanism housings, which penetrate the lower reactor vessel head of a BWR 4. A molten core mass would probably erode through the relatively thin guide tube walls and escape the reactor vessel long before the vessel head would fail by exceeding the tensile stress criterion utilized in MARCH. MARCH does not model vessel head failure via this mode (SD-4B).

B.8 MARCH BWR Modeling Concerns — Safety Systems and Equipment

The majority of the problems associated with application of MARCH to BWRs are related to MARCH's modeling (or lack of modeling) of systems and structures that are unique to BWRs. These concerns are summarized in Table B.5. MARCH does not model the BWR reactor building and its interaction with the primary containment for loss-of-coolant accidents outside containment (LOCAs/OC) (SS-1B). The MARCH manual[2] states that an "auxiliary" building may be modeled by setting the input parameter ICKV equal to −1, but this option does not function for BWRs.

The analyst may model the initial phase (prior to drywell or wetwell failure) of LOCAs/OC by specifying,

NCUB = 3
NRPV1 = 3
NRPV2 = 1
KT(1,2) = KT(2,1) = 1
KT(1,3) = KT(3,1) = KT(2,3) = KT (3,2) = 0,

where compartment one is the DW, two is the WW, and three is the reactor building. This input data specification will have the effect of setting

up a problem in which all Reactor Pressure Vessel (RPV) boil-off and break flow is routed to a reactor building, which is isolated from the DW/WW primary containment. All SRV flow will also be routed to the reactor building, but for cases in which there is little SRV actuation, this approximation may be acceptable. However, at some point in the accident, RPV head failure will occur. Following vessel head failure, the drywell or wetwell will begin to communicate with the reactor building, either via the original primary system break path, or due to drywell/wetwell containment failure. This communication cannot be modeled within MARCH because of the original specification of the KT array elements.

The BWR reactor building (Fig. B.9) is a secondary containment system and, as such, can play an important role in restricting the release of fission products to the environment during some severe accidents. Primary system (and fission product) leakage into the reactor building might occur via direct leaks from the primary system (as in the scram discharge system break) or via DW failure and subsequent venting into the reactor building. The absence of a suitable reactor building model significantly limits the usefulness of MARCH results for application in fission product transport analysis.

MARCH does not allow separate destinations for break and SRV flow (SS-2B). While this is not a concern in many types of PWR accidents, this restriction is completely incorrect for BWRs in which SRV flow is routed to the pressure suppression pool. In the typical MARCH BWR problem, two volumes are utilized to represent the drywell and wetwell, respectively. The analyst is forced to choose a single destination for all primary system boil-off flows via the NRPV1 input parameter in namelist NLMACE. For accidents other than simple boil-off transients without pipe breaks, the analyst is forced into the dilemma of choosing either an incorrect SRV or break flow destination. For accidents involving sustained periods of SRV actuation, this is a particularly critical concern.

A closely related problem involves MARCH's modeling of mass flows into the suppression pool for BWR pipe-break accidents outside containment. The BWR suppression pool incorrectly receives all primary system boil-off for such accidents regardless of the value assigned NRPV1 (SS-3B). This problem is manifested as an increase in suppression pool mass and temperature for LOCAs/OC in which there is no SRV actuation.

MARCH does not have the capability to model BWR Mark II containment systems (SS-4B). Figure B.10 is a schematic representation of a typical BWR MARK II containment structure. In structures of this type, a molten core mass apparently could erode through the containment floor directly beneath the reactor vessel and subsequently drop into the pressure suppression pool. MARCH has no models that allow this sequence of events to occur.

The existing MARCH pressure suppression pool model is extremely simplistic (SS-5B). The pool is treated as a single homogeneous volume with instantaneous dispersion of all incoming mass flows. It is assumed that all entering steam in excess of that needed to keep the wetwell steam partial pressure at the pool saturation pressure is condensed. Localized pool boiling (and associated wetwell pressurization) cannot be modeled.

MARCH does not model inerted containments (SS-6B), which are a common design feature of BWRs. The impact of containment inerting on hydrogen burns cannot, therefore, be evaluated.

The principal operating parameter of interest in BWRs is reactor vessel water level. This water level is a control variable for all BWR emergency core cooling systems (ECCS) and containment sprays and as such has a major influence upon the timing and sequence of events in severe BWR accidents. The existing MARCH vessel water level calculation is correct only while the water level is within the active core region (SS-7B). Variable flow areas of standpipes and separators are not incorporated into the calculation. MARCH does not model ECCS actuation and control based on vessel water level (SS-8B), nor does it model high-pressure ECCS isolation caused by low primary system pressure (SS-9B).

It was previously mentioned that MARCH does not model heat transfer to the drywell atmosphere from the reactor vessel and steam lines. Under these circumstances, the analyst should be aware that it is unrealistic to model drywell coolers (SS-10B). This concern may be more important for accidents such as SBLOCA/OC, in which the drywell response prior to vessel head failure is driven by this heat transfer mechanism. Modeling drywell coolers without simulating the vessel-to-drywell heat transfer can result in ambiguous and erroneous containment responses for accidents of this type.

MARCH models a single ECCS recirculation heat exchanger, which is placed in the suction line of all pump-driven ECCS systems when the fractional amount of water remaining in the "refueling water storage tank" is less than the input variable ECCRC. This is an unrealistic model for BWRs such as the Browns Ferry units in which the residual heat removal (RHR) system [low pressure coolant injection (LPCI) mode] is the only ECCS system that has a heat exchanger (SS-11B).

In many SBLOCAs, normal plant response procedures would include either manual operator-controlled primary system depressurization or automatic depressurization system (ADS) actuation to reduce primary coolant inventory loss and to enable utilization of low-pressure ECCSs (LPECCSs). Emergency operating instructions typically call for either depressurization to a selected system pressure within a specified time or a primary system cooldown/depressurization rate expressed in terms of degrees per hour. Although these actions are among the most important SBLOCA mitigation procedures available to plant operators, MARCH does not currently allow the analyst to directly model this phenomenon (SS-12B). The analyst wishing to model such accidents must employ an alternative procedure involving at least two MARCH runs. In the first run, the analyst must specify a "break" size representing the total SRV flow area to be used and the time at which the break is to occur. This run is analyzed to determine when the primary system reaches the desired pressure. A second MARCH run must then be made with the "break" flow area reset to zero at the proper time as determined from the first run. A related problem is the fact that MARCH does not allow time-dependent primary system set pressures (SS-13B).

All methods of emergency cooling water injection into the reactor vessel must be modeled in MARCH as "ECCS" systems. MARCH 1.1 has the capability to utilize nine pump-driven cooling systems. Three of the nine systems can be modeled as either constant flow or variable (pump head curve) systems, while the remaining six systems can be modeled only as fixed flow systems (SS-14B). The pump head curve utilized for the three primary ECCS systems is based on the pressure differential between the

primary system and the drywell. Table B.7 contains information on the principal primary system water injection systems of concern in BWR accident analysis. It is clear from an analysis of Table B.7 that accurate simulation of some of these systems would necessitate the use of a pump curve based upon the pressure differential between the primary system and the wetwell, atmosphere, or condenser hotwell. MARCH is incapable of accurate simulation of these systems (SS-15B).

Steam that is removed from the reactor vessel to drive the high pressure emergency core cooling system turbines is not accounted for in MARCH's primary system mass loss or suppression pool mass gain calculations (SS-16B). The Browns Ferry HPCI and RCIC systems incorporate steam-turbine-driven pumps with steam extracted from the main steam lines upstream of the main steam isolation valves (MSIVs). The HPCI turbine has a steam demand of between 18.8 and 23.18 kg/s (149,000 and 184,000 lb_m/h), while the RCIC turbine has a steam demand of ~3.53 kg/s (28,000 lb_m/h). These steam demands represent ~20% of the rated flow of a single primary relief valve (105.7 kg/s or 838,900 lb_m/h) and should not be disregarded.

Discussions between ORNL staff and Browns Ferry operating staff have indicated that drywell flooding is often viewed as the ultimate severe-accident mitigation procedure. MARCH does not have the capability to simulate this process (SS-17B).

For primary system breaks outside containment, MARCH containment response calculations subsequent to bottom head failure are incorrect (SS-18B). For cases in which no containment failure events have been input by the user, the code sets the containment failure pressure equal to the drywell pressure at the time of bottom head failure and continues to leak steam and hydrogen from the containment via the primary system leak path. The code apparently does not allow the original containment atmosphere to leak out via this path, because the MACE output parameters TOTNLK, TOTOLK, TOTHLK, TOTMLK, and TOTDLK remain unchanged for this case. The code does not print a containment failure message, and the MACE intercompartment transfer data output does not show any transfers from the reactor containment to the atmosphere. In short, the MACE output for this case is incomplete and contradictory and can easily confuse and mislead the unknowing analyst.

For cases in which containment failure events are input by the analyst, MARCH executes all the events shortly after head failure regardless of whether the event triggers on containment pressure, temperature, or time. The analyst does not, therefore, have control of the timing of containment failure. Because of the nature of these problems, it is extremely difficult to utilize MARCH results for times after head failure in LOCA/OC sequences. Existing MARCH documentation does not address this problem.

MARCH does not correctly model core spray ECC (SS-19B). In effect, MARCH assumes that all ECCS injection occurs at the bottom of the reactor vessel and is instantly mixed with the remaining reactor vessel water. The major impact of ECCS injection under this assumption is to raise the water level and lower the water temperature in the reactor vessel. This is clearly an invalid representation of the CS system, in which water is sprayed directly on the top of the core. With the current MARCH model,

reactor vessel pressure is impacted only by changes in break flow and pool flashing because of oscillations in water level around the break height. During periods in which the core is completely uncovered, MARCH assumes that steam production occurs only because of flashing that is predicted to occur as the reactor vessel water level drops below the height of the break. Because the $Zr-H_2O$ reaction is generally steam starved during these periods, these models result in unrealistic hydrogen production rates and $Zr-H_2O$ reaction distributions for cases when core spray systems are operable.

Finally, the current version of MARCH does not model pump-driven ECCS injection subsequent to head failure. BWR LPECC systems have the capability to inject flow under 0-psid conditions. For accidents in which these systems are operable, injection can continue after failure of the RPV bottom head.

In summary, MARCH version 1.1 is an extremely simplistic code that is not well-suited for BWR accident evaluation applications. The code contains a variety of modeling simplifications, errors, and omissions that severely compromise its usefulness in a wide variety of such applications. For this reason, the code should be utilized only by individuals who are fully aware of the scope of its limitations and of the design of BWR systems.

References

1. J. B. Rivard et al., *Interim Technical Assessment of the MARCH Code*, Sandia National Laboratory/USNRC Report/CR-2285, SAND81-1672 R3 (November 1981).

2. R. O. Wooton and H. I. Avci, *MARCH Code Description and User's Manual*, Battelle Columbus Laboratories/USNRC Report/CR-1711 (October 1980).

3. D. H. Cook et al., *Station Blackout at Browns Ferry Unit One – Accident Sequence Analysis*, ORNL/NUREG/CR-2182/V1 (November 1981).

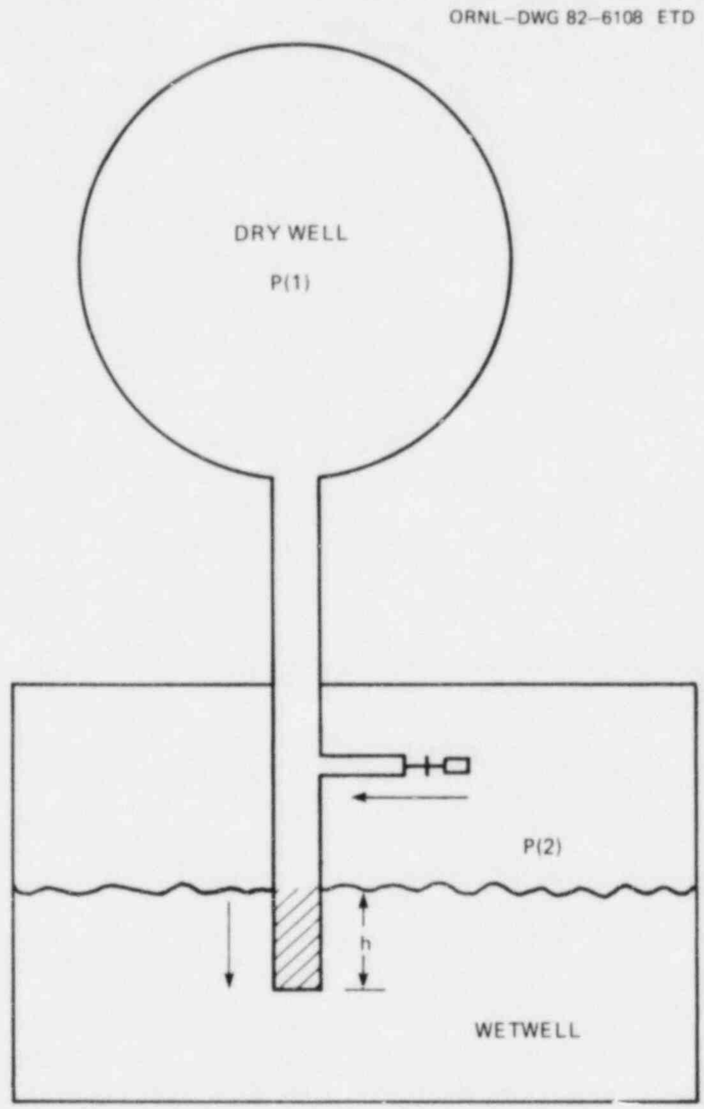

Fig. B.1. MK 1 drywell/wetwell schematic.

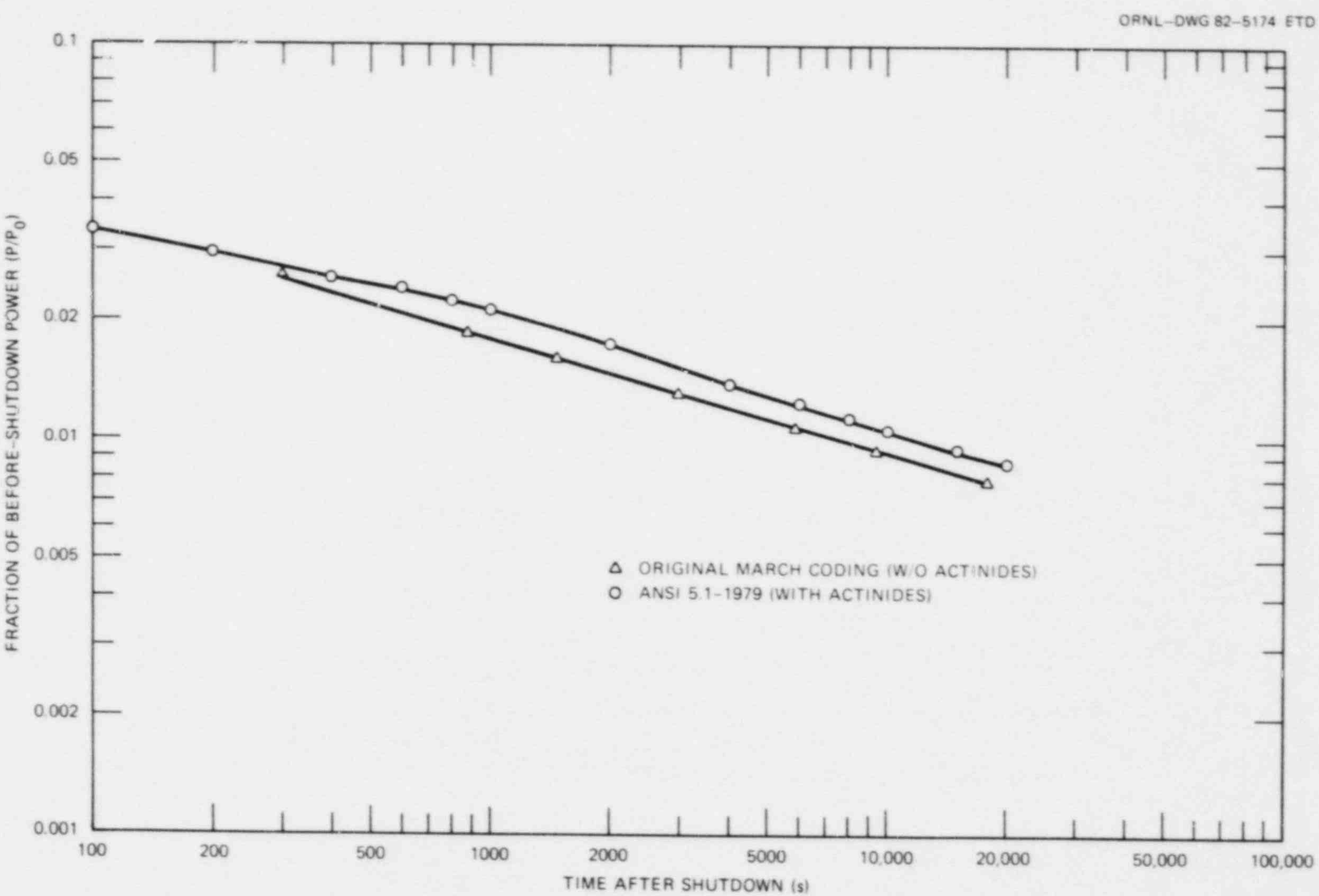

Fig. B.2. Comparison of MARCH ANSQ and ANS 5.1-1979.

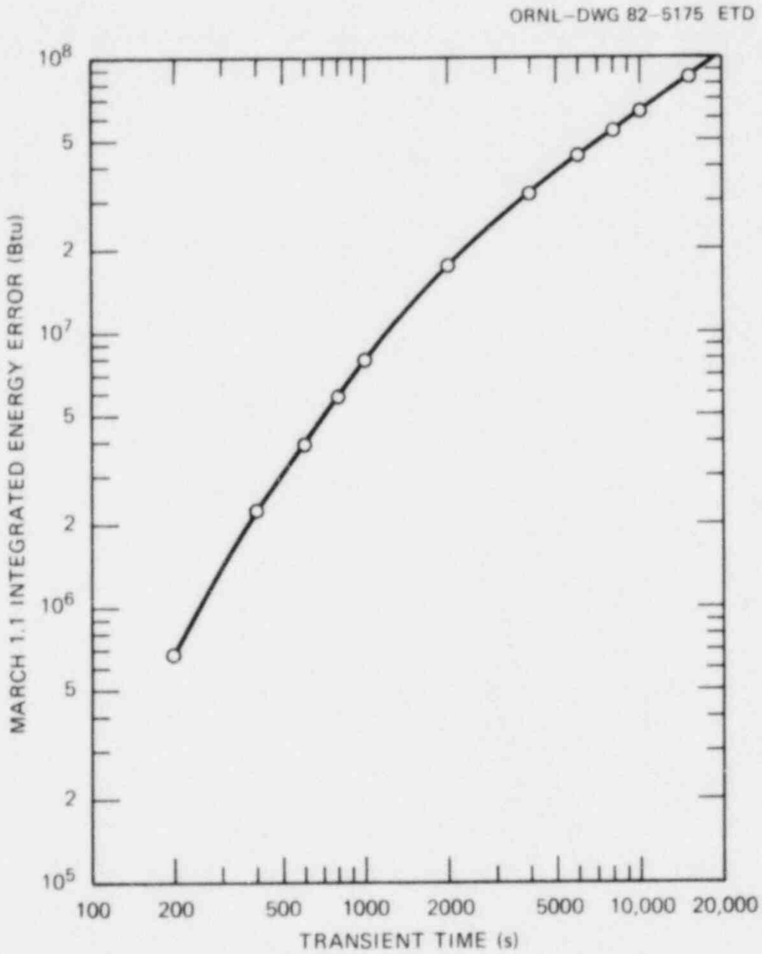

Fig. B.3. MARCH 1.1 integrated decay energy error.

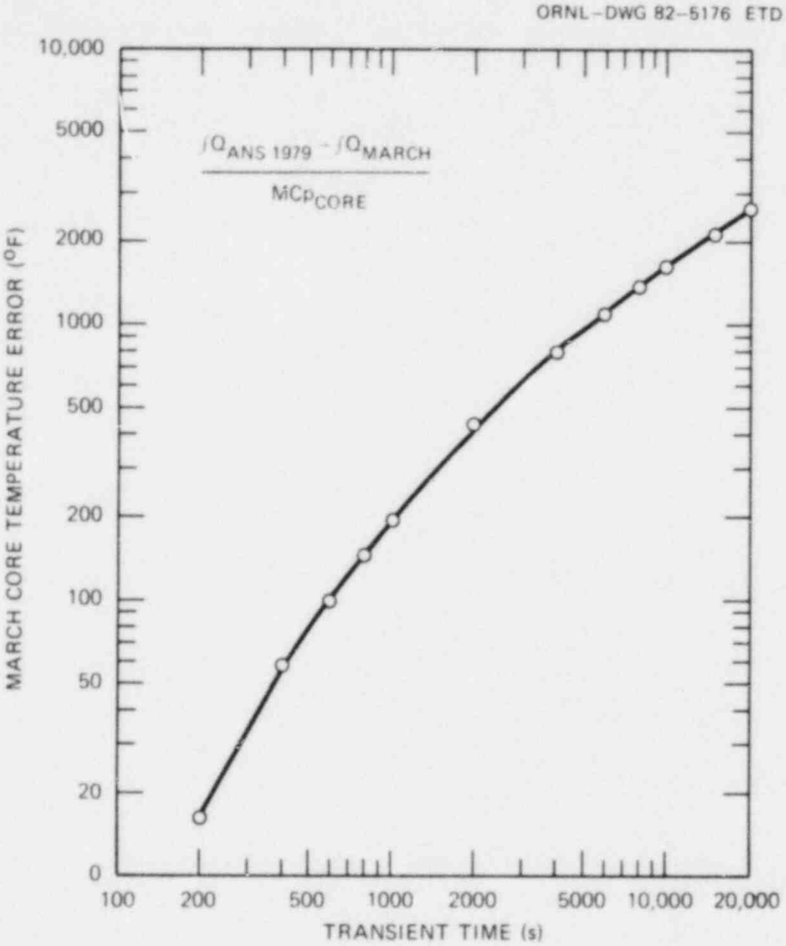

Fig. B.4. MARCH 1.1 core temperature error.

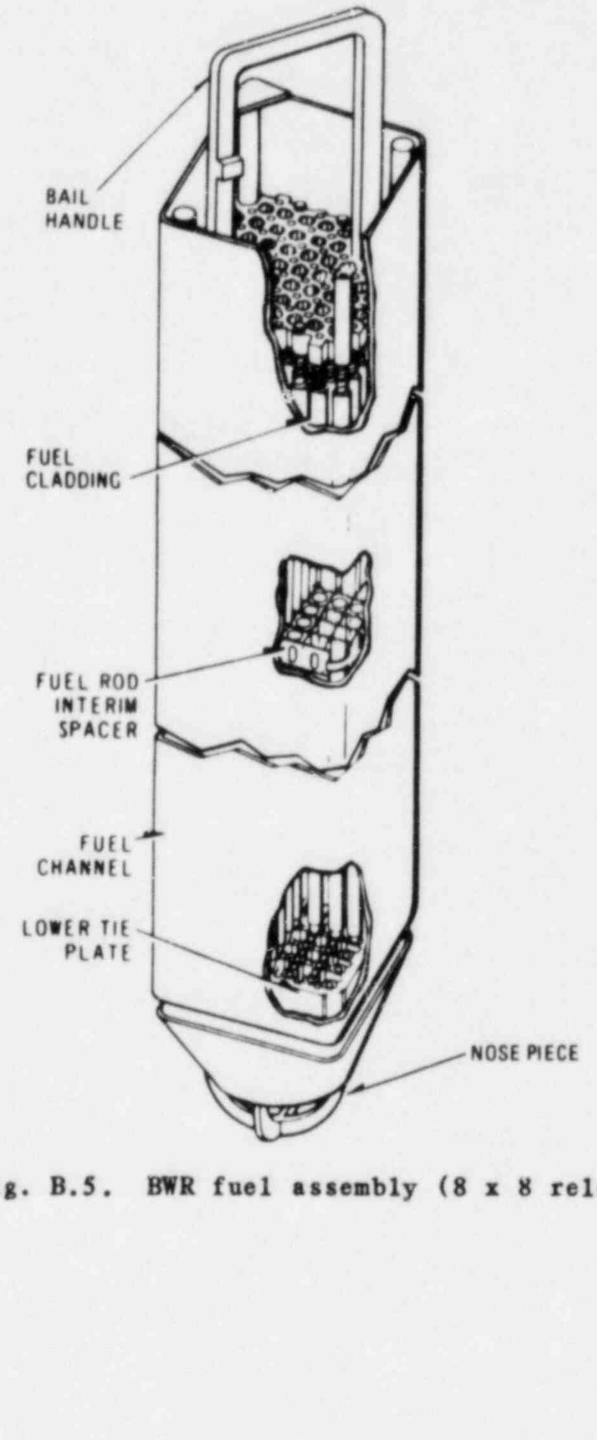

Fig. B.5. BWR fuel assembly (8 x 8 reload).

ORNL-DWG 82-5178 ETD

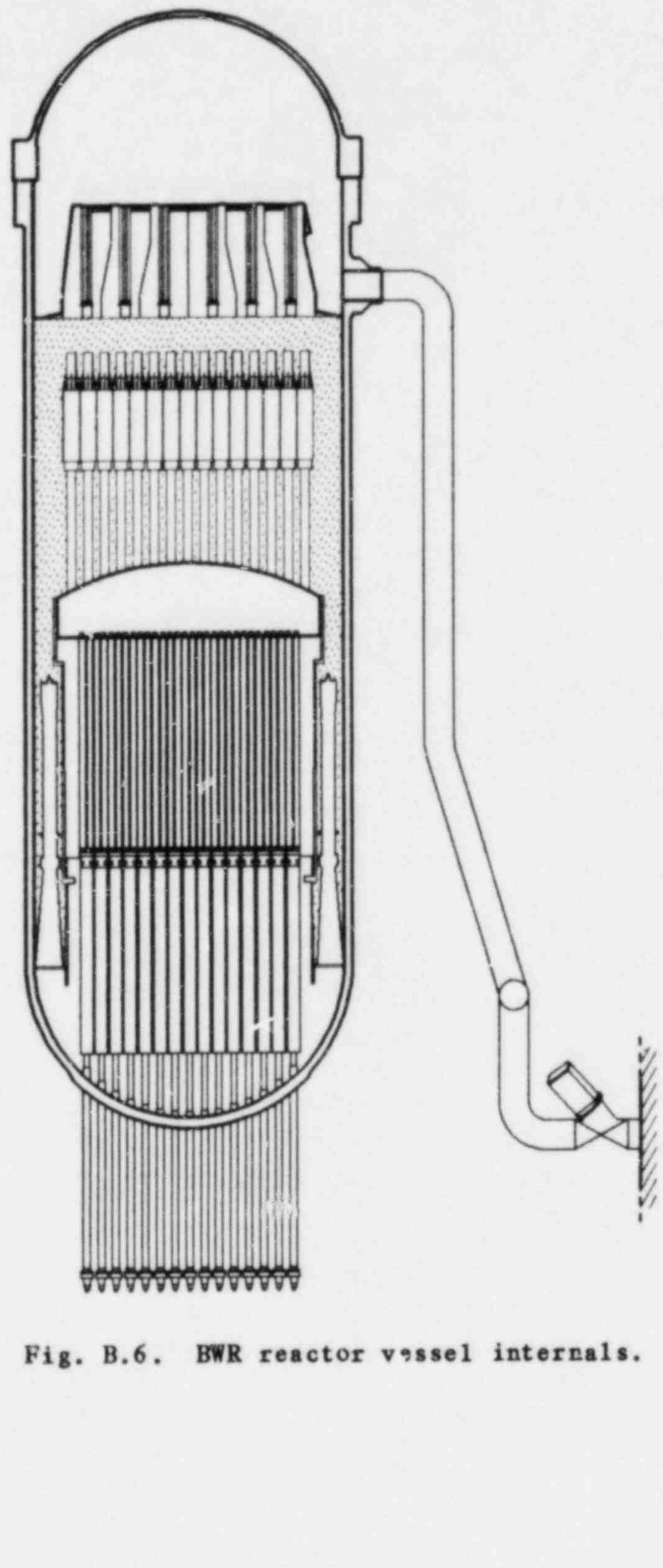

Fig. B.6. BWR reactor vessel internals.

ORNL-DWG 82-5179 ETD

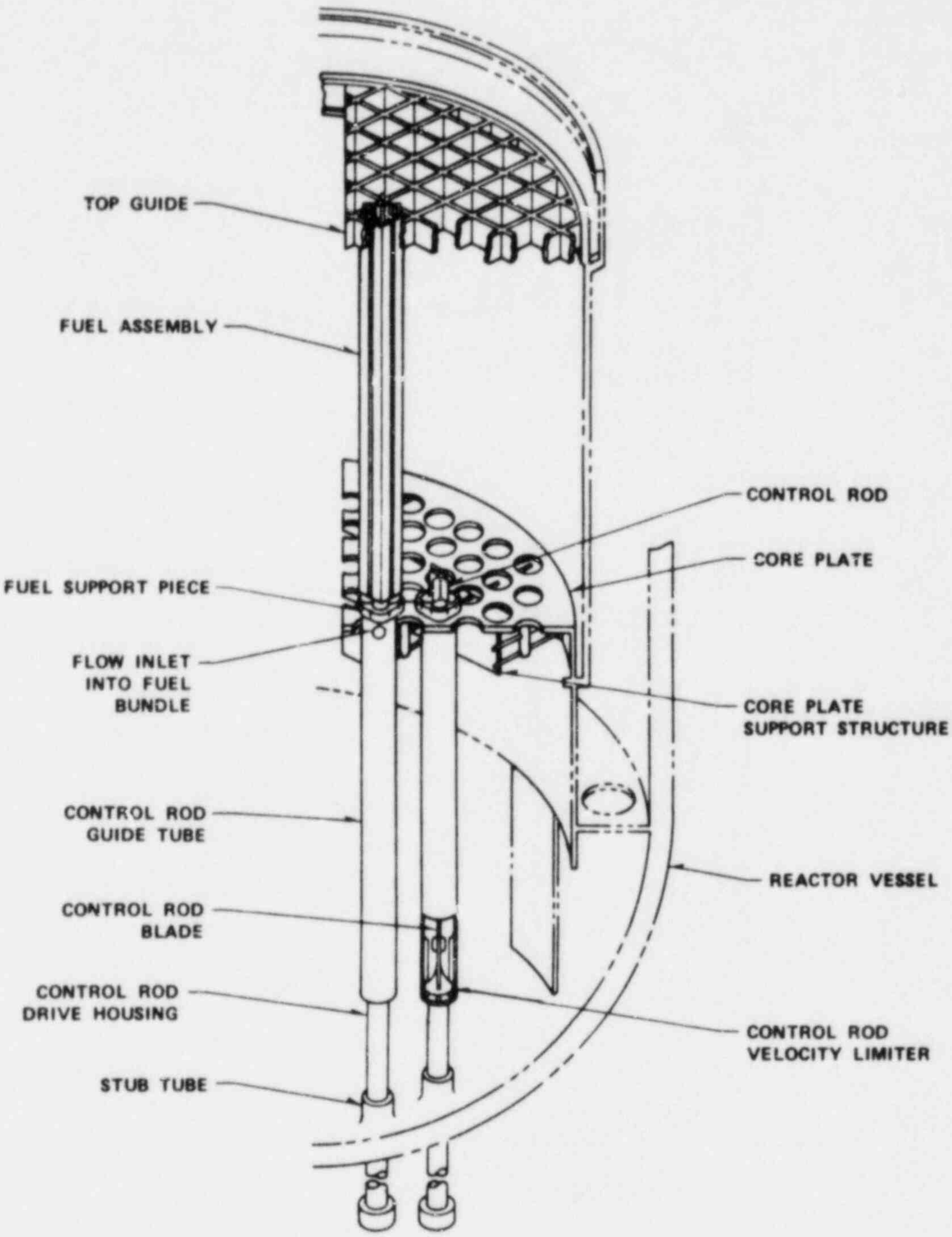

Fig. B.7. BWR fuel assembly support.

ORNL-DWG 82-5180 ETD

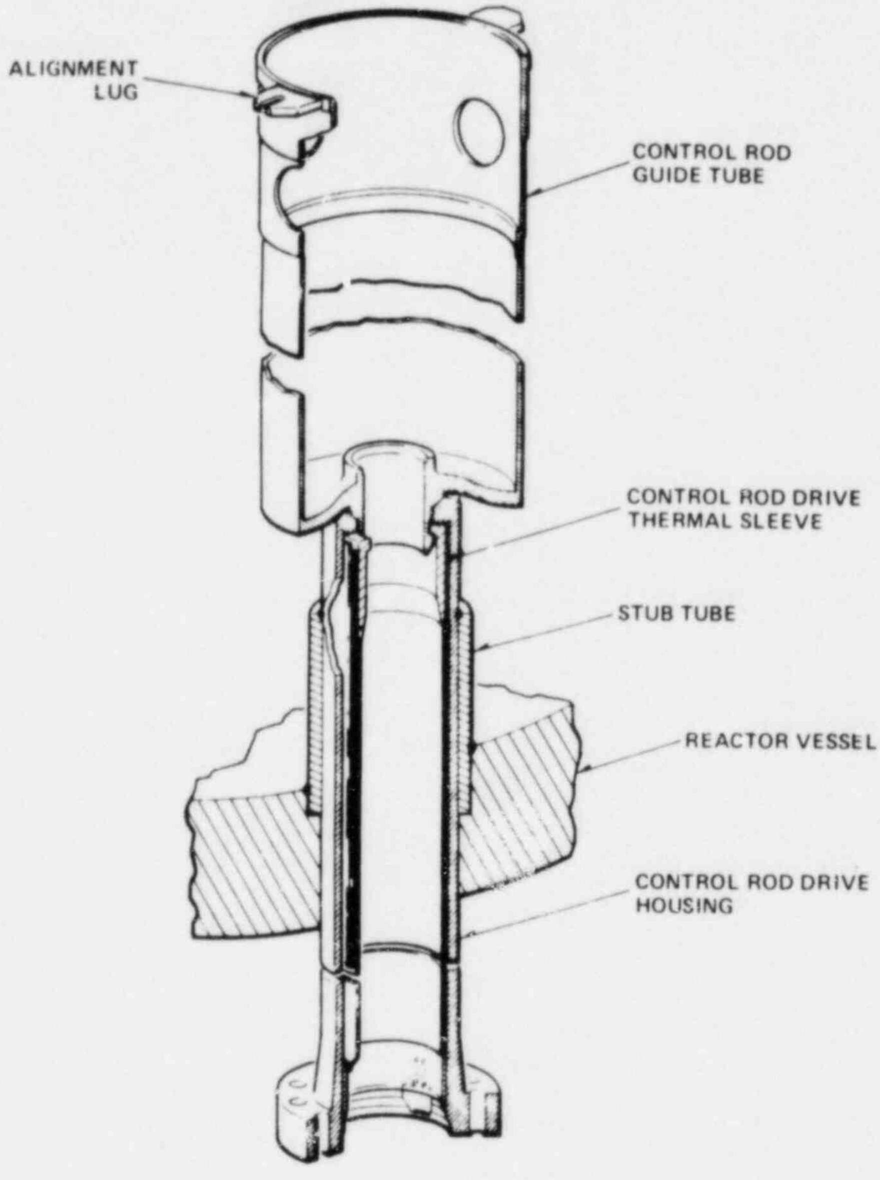

Fig. B.8. CRD housing reactor vessel penetration.

ORNL–DWG 82–5181 ETD

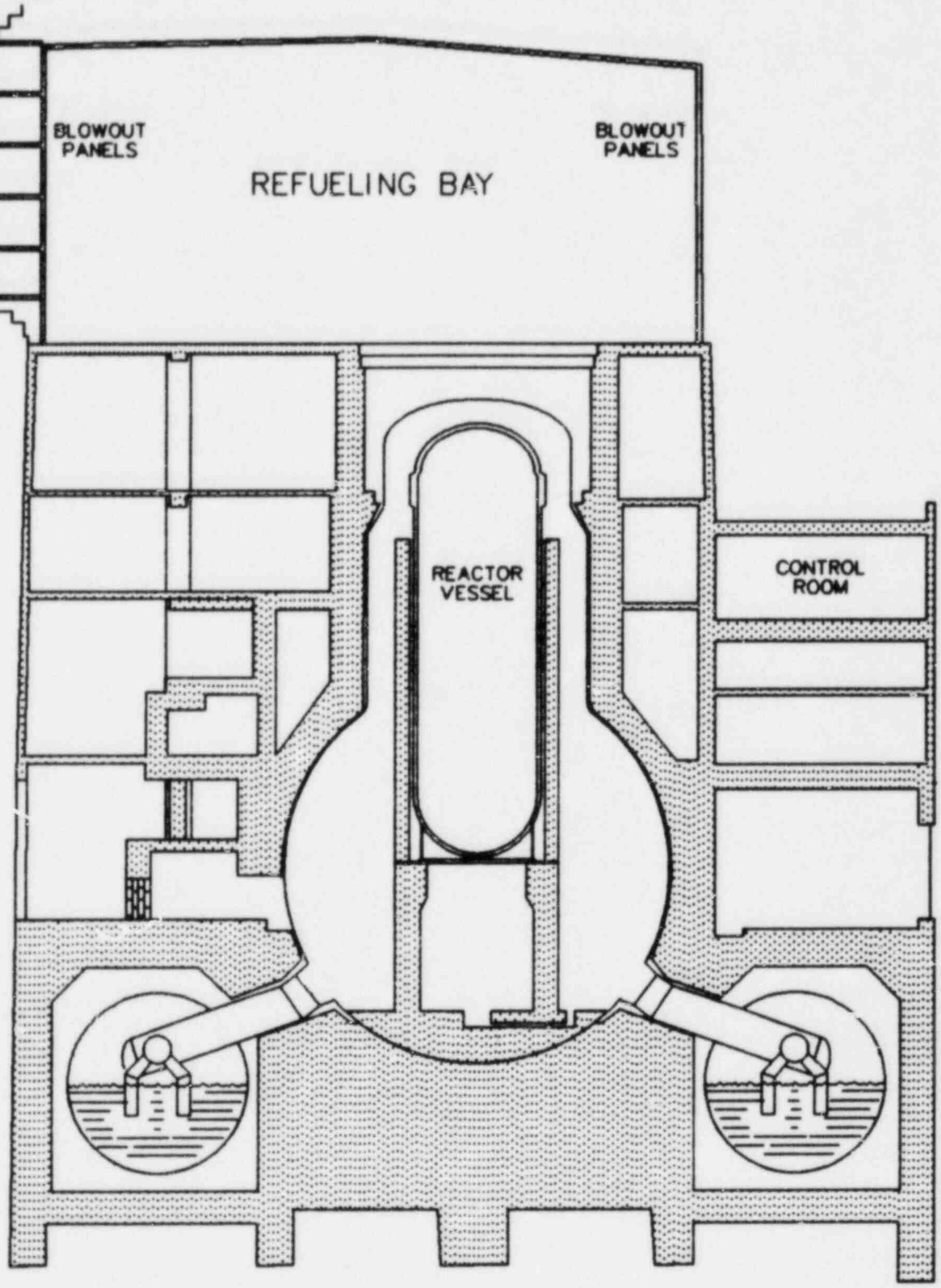

Fig. B.9. Browns Ferry reactor building.

ORNL-DWG 82-5182 ETD

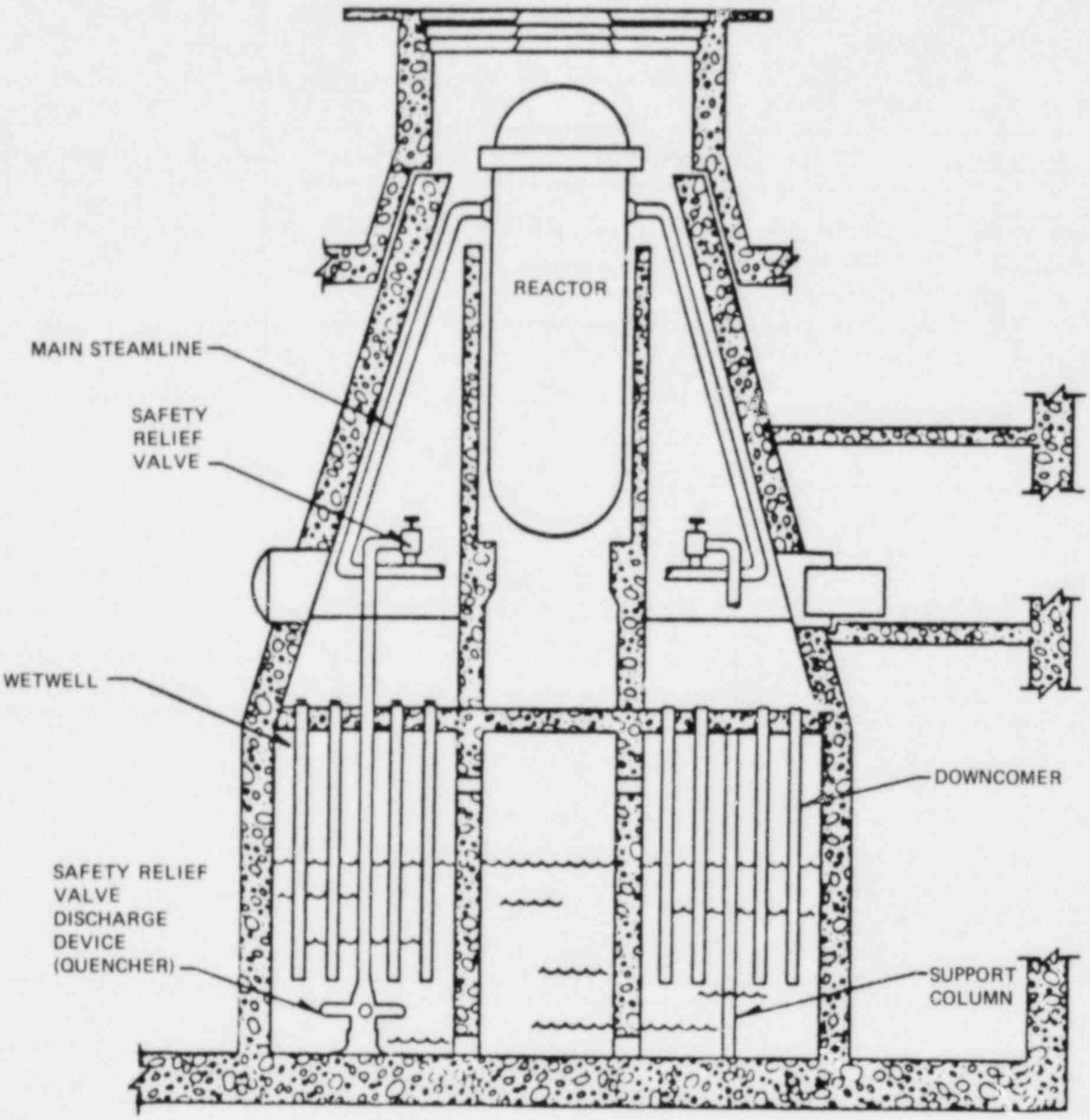

Fig. B.10. BWR Mark II containment.

Table B.1. MARCH modeling concerns — general

Identifier	Description	Phase	Importance	Knowledge
G-1B	MARPLT program is incompatible with IBM computer systems	0	3	0
G-2B	Variable PRESS is defined incorrectly on page 3-56 and omitted completely from input data listing for namelist NLMACE	0	2	1

Table B.2. MARCH modeling concerns — chemical/nuclear reactions

Identifier	Description	Phase	Importance	Knowledge
CNT-1B	Actinide decay and fission product neutron absorption are not included in decay heat calculations	0	3	1
CNT-2B	Only one fuel type is allowed in core	0	2	1
CNT-3B	Fuel assembly shrouds are not modeled. Shroud material must be lumped into fuel pin cladding	1–4	3	1

Table B.3. MARCH modeling concerns — transport phenomena

Identifier	Description	Phase	Importance	Knowledge
TR-1B	Modeling of heat transfer to upper and lower vessel internals is inadequate. Heat transfer coefficient correlations are inappropriate for many situations	1–4	2	1
TR-2B	Heat transfer to drywell atmosphere from reactor vessel, steam lines, etc., is not modeled	1–4	2	1
TR-3B	Fuel to coolant heat transfer coefficient is not dependent upon flow regime. Value is calculated from initial fuel and coolant temperatures	1–3	2	2
TR-4B	Rod-to-rod and rod-to-shroud radiative heat transfer is not considered in core	1–3	2	2
TR-5B	Water is allowed to flow out SRVs	1–3	2	1
TR-6B	Pressure differential employed in break flow calculations is incorrect for breaks outside containment	1–4	2	1
TR-7B	Code does not check for consistency of the input values of HO and VOLS	0	3	1
TR-8B	Calculation of initial primary system water mass is incorrect for initial vessel water levels above top of core	0	3	1
TR-9B	Code does not utilize input value of VOLP for transients and small-break accidents	0	3	1

Table B.4. MARCH modeling concerns — structural deformation/failures

Identifier	Description	Phase	Importance	Knowledge
SD-1B	Fuel pin and shroud melt/slump/freeze phenomena are not mechanically modeled	2–3	2	2
SD-2B	Core collapse model is highly unrealistic for BWRs. Realistic values of FDROP yield questionable results	2–4	3	2
SD-3B	Bottom head melt cannot begin until entire core is collapsed	4	3	2
SD-4B	Failure of bottom head via control rod drive tube penetrations is not modeled	4	3	2

Table B.5. MARCH modeling concerns — safety systems and equipment

Identifier	Description	Phase	Importance	Knowledge
SS-1B	Code does not correctly model BWR reactor building interaction with primary containment for LOCA/OC	6	3	1
SS-2B	Code does not allow separate destinations for break and SRV flow	0	2	1
SS-3B	For BWR small-break accidents outside containment (ICECUB = -1, ITRAN = 1), code discharges break flow into suppression pool	6	3	1
SS-4B	BWR Mark II containments cannot be modeled	6	2	1
SS-5B	Model of suppression pool and drywell/wetwell interaction is too simplistic	0	2	2
SS-6B	Inerted primary containment cannot be modeled	6	2	1
SS-7B	BWR vessel water level calculation is not correct. Variable flow areas of standpipes and separators are not modeled	1–3	3	1
SS-8B	ECCS control on primary system water level is not modeled	1–3	2	1
SS-9B	High pressure ECCS isolation because of low primary system pressure is not modeled	1–3	2	1
SS-10B	Because heat transfer from vessel and steam lines to drywell is not modeled (TR-2B), it is unrealistic to model drywell coolers	6	2	
SS-11B	In BWRs only RHR systems have coolers — therefore a different value for ECCRC is needed for each system	0	2	1
SS-12B	Manual operator depressurization or ADS actuation and subsequent depressurization to a selected pressure cannot be modeled	1–4	2	1
SS-13B	Time-dependent primary system set pressures cannot be modeled	1–4	1	1
SS-14B	Pump curves cannot be utilized for auxiliary ECCS systems	1–4	1	1
SS-15B	Pump curve ΔP is incorrect for some ECCS systems	1–4	2	1
SS-16B	ECCS turbine extraction steam is unaccounted for in both primary mass loss and suppression pool mass gain calculations	0	3	1
SS-17B	Impact of drywell flooding on accident mitigation cannot be studied	4–7	1	2
SS-18B	Containment response calculations are incorrect subsequent to vessel head failure in LOCA/OCs	6	3	1
SS-19B	Core spray ECC systems are not modeled	1–4	4	2
SS-20B	Pump driven ECC system flows are terminated at head failure	5–7	4	2

Table B.6. BWR internal structures

Structure	Mass [kg (lb_m)]	Heat transfer area [m^2 (ft^2)]
Control rod guide tubes	19,958 (44,000)	641 (6,900)
Shroud head and stand-pipes	10,886 (24.000)	27 (290)
Separators	37,194 (82,000)	474 (5,100)
Dryers	32,930 (72,600)	2,945 (31,700)

Table B.7. Principal BWR emergency cooling capabilities

Coolant system	Flow control		Source pressure
	Automatic	ΔP	
HPCI	*		
RCIC	*		
CRD pumps	*		
LPCI		*	Containment
Core spray		*	Containment
Condensate booster pumps		*	Condenser hot well
RHR service water pumps (standby coolant supply system)		*	Atmosphere (river)

Appendix C

MODIFICATIONS TO THE MARCH CODE

Appendix B presented a brief assessment of the applicability of MARCH 1.1 to boiling-water reactor (BWR) accident analysis. When confronted with limitations such as those discussed in Appendix B, the analyst must either restrict the scope of the analysis in a manner that will allow him to avoid identified code limitations or modify the code to eliminate its deficiencies. For the purposes of this analysis we have, where possible, followed the former path of action. For example, by initiating the MARCH analysis at a time just prior to core uncovery, many of the problems associated with MARCH's modeling of safety relief valve (SRV) and emergency core cooling phenomena (TR-5B, TR-8B, SS-2B, and SS-7B through SS-9B) are minimized or avoided completely.

During the course of our severe-accident sequence analysis efforts, it became clear that several of the deficiencies outlined in Appendix B were of such a severe and unavoidable nature that some code modification would be required. Table C.1 contains an abbreviated description of the BWR modifications that were implemented for the present analysis and their relationship to the problem areas discussed in Appendix B.

Table C.1. Oak Ridge National Laboratory MARCH modifications

Model No.	Description	Related problem identifer
ORNL 0006	Inerted containment option	SS-6B
ORNL 0009	Incorporation of ANS 5.1-1979 decay heat standard	CNT-1B
ORNL 0011	Elimination of water flow through the SRVs for BWRs	TR-5B
ORNL 0012	Elimination of break leakage flow path to suppression pool for LOCA/OC	SS-3B
ORNL 0017	Corrected pressure differential employed in break flow calculation for LOCA/OC	TR-6B
ORNL 0018	Corrected containment failure calculations	SS-18B
ORNL 0019	Usage of user-supplied primary system volume VOLP for all accidents	TR-9B
ORNL 0020	User input value for initial reactor water mass above core bottom	TR-8B

Appendix D

DIRECT DRYWELL FLOODING AS AN ACCIDENT MITIGATION TECHNIQUE

D.1 Introduction

The standby coolant supply system at Browns Ferry has been provided to permit the maintenance of a long-term reactor core and primary containment cooling capability, which does not require reactor vessel or primary containment integrity or operability of the residual heat removal (RHR) system associated with a given unit. One portion of the standby coolant supply system can be aligned so that the RHR service water pumps can be used to supply river water directly to the Unit 1 reactor vessel as the vessel pressure approaches 50 psig, and this alignment could be used to keep the core covered under severe-accident conditions with the reactor vessel depressurized. The excess water would be passed to the pressure suppression pool through the remote-manually opened safety relief valves or, in the event of a loss-of-coolant accident, would pass to the drywell from the break.

The standby coolant supply system can also be aligned to permit the RHR service water pumps to flood the drywell and wetwell directly, by means of the primary containment spray headers or via the RHR system test return line to the pressure suppression pool. This alignment for direct flooding of the primary containment might be used in a severe-accident situation in which the reactor vessel could not be depressurized. Browns Ferry operating personnel have indicated that direct drywell flooding should be an effective measure of last resort to reduce the consequences of such a severe accident by preventing an already molten core from melting through the reactor vessel bottom head. However, no calculations have been performed to determine the accident mitigation effects of drywell flooding on a boiling-water reactor system or to determine the time required to flood the drywell. Therefore, no information is available to prepare the operator for difficulties that might be encountered in attempting direct flooding of the drywell.

The purpose of this appendix is to present the results of an investigation to determine the effectiveness and practicality of direct drywell flooding as a mitigating procedure to prevent reactor vessel melt-through at Browns Ferry Nuclear Plant Unit 1 (BFNP#1). Various heat transfer computer codes were used to determine if a flooded drywell would preclude vessel melt-through, and an analysis of the existing equipment was performed to determine if a practical means exists to directly flood the drywell to the required level. These analyses are described in the following sections along with sample calculations and suggested input parameters for the computer codes.

D.2 Steady-State Models

The system configuration of the primary containment for each unit at the BFNP is presented in Fig. D.1. This figure shows the location of the

reactor vessel within the drywell as well as the volumes that would require filling in the event that drywell flooding became a necessity. Direct flooding of the drywell with the standby coolant supply system would be accomplished by the activation of a maximum of two RHR service water pumps, which would pump river water into both the drywell and the wetwell simultaneously. To sufficiently cover the reactor vessel bottom head, water must be pumped into the drywell and wetwell regions until the water level is approximately equal to that shown in Fig. D.1.

Two different computer codes were used to model the effect of a water heat sink on the outer surface of the reactor vessel after the core has slumped against the bottom head. These computer codes were developed by Hagen[1] to perform heat transfer calculations on maritime pressurized-water reactors and employ steady-state models to calculate various temperatures within the reactor vessel bottom head and the core debris. A brief description of these steady-state models and the results of calculations for the BFNP#1 are presented in the following paragraphs.

Figure D.2(a) is a representation of the reactor vessel bottom head containing the completely slumped core and debris. The layers shown represent the reactor vessel bottom head, the molten uranium dioxide (UO_2) fuel, and a layer of molten steel. To model the situation presented in Fig. D.2(a), several simplifying assumptions can be made. The first is that the layers of debris can be approximated by slabs of various thicknesses. The thickness of each slab is assumed to be the maximum thickness of that debris layer. Therefore, the situation shown in Fig. D.2(a) is approximated by the geometry shown in Fig. D.2(b). This assumption is both simplifying and conservative because the calculations now become one dimensional with heat being transferred through a slab of maximum thickness.

The second assumption is that heat is assumed to be transferred from the molten debris surface by radiation to its surroundings only. This is also a conservative assumption because as long as the reactor vessel integrity is maintained, some cooling would occur because of the presence of steam above the debris. It should also be noted that the decay heat source within the molten debris is assumed to be constant throughout these calculations and equal to that at the time of core slumping. This provides additional conservatism to the calculations because in actuality the decay heat would be slowly decreasing with time.

Finally, assumptions must be made about the state of the debris after core slumping has occurred, giving rise to the two different steady-state models utilized in this analysis. Figure D.3 shows the two different debris configurations employed by the steady-state models. Model 1 assumes six layers of debris and corresponds to the situation that would exist for a relatively low decay heat. As shown in Fig. D.3, these layers are (1) the carbon steel reactor vessel, (2) a layer of UO_2 that has solidified because of cooling from the reactor vessel, (3) a molten layer of UO_2 having no convection currents, (4) a convective layer of UO_2, (5) a layer of UO_2 that has solidified because of cooling from above, and (6) a convective layer of molten steel. Note that this final layer of molten steel was formed from the debris of reactor internals, which existed below the core (i.e., core supports and control rod guides), and not from melting of the reactor vessel itself. The situation presented in Model 1 can be modeled using the equations contained in Table D.1. These equations were

derived by Hagen[1] to model the heat transfer through each layer and can be solved simultaneously to obtain the temperature T_1 at the inner surface of the reactor vessel.

The second steady-state model shown in Fig. D.3 is similar to the first one but is applicable to a situation in which the existing decay heat is higher than the decay heat assumed in Model 1. The higher decay heat prevents the formation of the solid UO_2 layer (L5) above the layer of convective UO_2, instead producing a single large layer of convective molten UO_2. For this situation, two simplifying assumptions can be made. First, the temperature of this large layer of molten UO_2 undergoing convection and boiling is assumed not to exceed the boiling point of UO_2. Second, the convective currents in this layer are assumed not to disturb the stagnant molten UO_2 underneath. With these two assumptions, the temperature T_3, shown for Model 2 in Fig. D.3, is the boiling point of UO_2. Therefore, the heat transfer below T_3 can be completely described by the last two equations presented in Table D.1 with T_3 set equal to the boiling point of UO_2.

D.3 Selection of Input Parameters for the Steady-State Analysis

In performing the steady-state analysis, several parameters are required for input to the computer codes. However, there exists some uncertainty concerning the proper values for these parameters. For this reason, several parametric studies were performed to analyze the effects of drywell flooding for a range of accident conditions, which might exist at the BFNP. The parameters of interest are the decay heat, the thermal conductivity, and the reactor vessel outer surface temperature. Of these parameters, decay heat is the most important and has the largest effect upon the calculations. Because the decay heat decreases with the time interval between system scram and core slump, the value of the volumetric heat source (watts per cubic centimeter of UO_2 debris) was varied from 0 to 8 W/cm^3.*

The results of these calculations were strongly dependent on the chosen value for the thermal conductivity of the reactor vessel bottom head. This parameter was varied from 0.2 to 0.6 W/(cm·K), spanning the actual thermal conductivity of the carbon steel bottom head, 0.48 W/(cm·K). This parametric study was necessary because the reactor vessel bottom head is covered by 3 in. of mirror insulation. As shown in Fig. D.4, there is a gap of ~17 in. between the vessel and the insulation surrounding the bottom head. The computer programs used in this study assume a constant temperature for the outer surface of the reactor vessel. Because insulation is not modeled, the assumed thermal conductivity for the reactor vessel should be an effective thermal conductivity representing both the carbon steel and the insulation outside the vessel. The effective steady-state thermal conductivity of the reactor vessel and insulation during normal operation is ~8.34×10^{-4} W/(cm·K) (Ref. 2).

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*The decay heat power would reach 8.0 W/cm^3 at ~1 s after shutdown following infinite operation at 100% power.

On the other hand, if drywell flooding is accomplished the water will penetrate the mirror insulation, effectively removing the insulation from the heat transfer process due to the convective currents within the gap between the bottom head and the insulation. Therefore, the effective thermal conductivity of the reactor vessel bottom head with the drywell flooded would be close to the thermal conductivity of carbon steel alone, ~0.4 W/(cm·K). However, this value is at best a rough estimate and therefore the thermal conductivity was varied from a conservatively high value to a conservatively low value in an attempt to bound this uncertainty.

The final parameter of interest is the vessel outer surface temperature T_0, which corresponds to the temperature of the drywell water adjacent to the reactor vessel bottom head. As with the decay heat and thermal conductivity, the calculations were significantly dependent on the outer surface temperature. This temperature was varied from a value of 311 K (100°F) to a value of 422 K (300°F), which corresponds to ~50 K (90°F) above the boiling point of water at atmospheric pressure.

D.4 Results

Several calculations were performed to determine the effects of the variation in input parameters discussed in Sect. D.3. The results of these calculations are presented in Figs. D.5–D.10. On each of these figures, the inner vessel wall surface temperature T_1 can be read on the abscissa as a function of the debris decay heat on the ordinate for three different assumed vessel wall outer surface temperatures T_0. If the temperature T_1 is below the melting point of steel, it can be concluded that the vessel would not fail at the corresponding decay heat level. A comparison of Figs. D.5–D.7 illustrates the effect of variance in the thermal conductivity over the range from 0.2 to 0.6 W/(cm·K). Figures D.8–D.10 are analogous to Figs. D.5–D.7 with the exception that they represent calculations performed with Model 2. A comparison of Figs. D.5–D.7 with Figs. D.8–D.10, respectively, shows that the calculations for Model 1 yield results very similar to those obtained for Model 2.

If the drywell is not flooded, the effective thermal conductivity for the reactor vessel bottom head may be taken as the thermal conductivity of the surrounding mirror insulation material. This material is designed to have a thermal conductivity close to that of air, and a realistic value is 8.34×10^{-4} W/(cm·K) as stated previously. Figures D.5 and D.8 imply that vessel melting will occur for any situation in which the thermal conductivity is 0.2 W/(cm·K) or less and the decay heat is greater than 1.5 W/cm^3.* Therefore, with an effective thermal conductivity of 8.34×10^{-4} W/(cm·K), vessel melting would be expected to occur.

An analysis of Figs. D.5–D.10 indicates that for realistic values of effective thermal conductivity with the drywell flooded (i.e., ~0.4),

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*From the 1979 American National Standards Institute-5.1 Standard with Actinides, the decay heat power would remain above 1.5 W/cm^3 for ~160 h after shutdown from infinite operation at 100% power. No allowance has been made for the loss of volatile fission products from the core debris.

vessel melting will occur only for very high decay heats of ~4 W/cm³ or greater. However, a debris decay heat power as large as 4 W/cm³ could only exist if core slumping occurred within ~150 s after reactor scram, which is unrealistic. Therefore, it can be concluded that a flooded drywell would prevent vessel melt-through under the assumptions of the steady-state models.

D.5 Practicality of Drywell Flooding at the BFNP

This section of the analysis is concerned with the practicality of drywell flooding as an accident mitigation technique at the BFNP. The first problem that must be considered concerns the pumping capabilities available for drywell flooding.

As shown in Fig. D.1, the wetwell volume would not completely fill because of a trapped airspace in the top of the torus above the wetwell-to-drywell vacuum breakers. The volume taken up by the trapped air is significant in reducing the total free volume that must be filled when attempting to flood the drywell, so care would have to be taken to ensure that inadvertent venting of the wetwell airspace did not occur during the flooding process.

The final volume of the trapped air with the drywell flooded depends upon the initial pressure within the drywell and wetwell at the time the wetwell water level covers the vacuum breakers.* The initial volume of the trapped air is 84,019 ft³, and Table D.2 is a tabulation of the final air volume obtained for various initial pressures. These values were calculated using the ideal gas law and take into account the initial pressure and the final head of water in the drywell above the torus, equivalent to 16.34 psi. Therefore, an initial pressure of 14.7 psia would result in a final wetwell pressure of 31.04 psia. Using the ideal gas law and assuming that the wetwell airspace temperature does not significantly change,

$$P_1V_1 = P_2V_2$$

$$(14.7)(84{,}019.33) = (14.7 + 16.34)V_2$$

$$V_2 = 39{,}790.08 \ .$$

Table D.2 shows that the final trapped air volume is largest when the initial pressure is 14.7 psia (atmospheric pressure) and the drywell is simultaneously vented and flooded. This situation is assumed to be the most probable sequence of events,† and therefore a trapped air volume of

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*The containment pressure at this time would depend upon the specific accident sequence and whether the containment had been vented.

†As will be shown, drywell venting is required if the drywell is to be flooded to the point where the reactor vessel bottom head is covered. It is assumed that the wetwell airspace is not vented.

39,790 ft^3 can be subtracted from the total free volume that requires filling. Under these conditions, the total volume to be filled is 208,165 ft^3. For the other four cases shown in Table D.2, it has been assumed that the drywell is not vented. This results in higher pressures and lower airspace volumes at the completion of flooding.

Figure D.11 shows a schematic of one-half of the Residual Heat Removal (RHR) system configuration for Unit 1 and the portion of the RHR service water (RHRSW) system available for use in drywell flooding.* The RHRSW system valves, which must be opened to initialize Unit 1 drywell flooding from the river, are the standby coolant supply valves 23-57 and 74-101, and the river return valve 23-52 must be shut. This delivers river water from the RHRSW pumps into the RHR system piping downstream of heat exchangers B and D; from here it can enter the drywell spray headers via valves 74-74 and 74-75 and can enter the wetwell via the spray header and valves 74-71 and 74-72 or via the test return line and valves 74-71 and 74-73.

The RHRSW pumps D1 and D2 have a combined pumping capability of ~1200 ft^3/min (9000 gpm) under normal operating conditions (i.e., at 120 psid across the pumps).† If the drywell is not vented during the flooding operation, the resulting high back-pressure would reduce the rate of RHRSW flow into the drywell. However, assuming a pumping capability of 1200 ft^3/min, the time required to flood the BFNP drywell to the level shown on Fig. D.1 would be ~3 h. Because direct drywell flooding is a last resort effort and would not be initiated until the operator was positive that core slumping was imminent, this number is understandably too high and would prohibit the use of drywell flooding as a practical mitigating action.

To realistically meet the requirements for molten core retention in the reactor vessel by direct drywell flooding, the operator must have the ability to sufficiently flood the drywell within a very short time, because he or she would probably not resort to direct drywell flooding until after core melting had begun. The MARCH computer code was used to obtain typical values for the time between the initiation of core melting and the occurrence of core slumping.[3,4] This time was on the order of 30 min. To flood the drywell in 30 min, the required pumping capability would be 6730 ft^3/min, which would require an increase in the current BFNP standby coolant supply pumping capabilities from the RHRSW system by a factor of 6.‡

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*The other half of the RHR system for Unit 1 is similar, but does not include provision for drywell flooding from the RHRSW system.

†The capacity of pump D3 is reserved for the Emergency Equipment Cooling Water System, which provides cooling water for the onsite diesel-generators, and other emergency equipment. None of this emergency equipment can be valved off, so pump D3 is unavailable for drywell flooding.

‡Note that this rate could be achieved with the present design if the RHR pumps were used, taking suction on the condensate storage tanks. However, an amount equivalent to the capacity of 4.2 condensate storage tank volumes would be required.

During the containment flooding process, a large volume of free air is displaced by the incoming water and compressed. The compression of the displaced air would lead to containment pressures that exceed the design standards of the drywell and wetwell structures unless the containment is vented during the flooding process. This is the second problem associated with drywell flooding: it would not be attempted unless the reactor vessel water level were low, but valve interlocks prevent direct drywell flooding unless (1) the core is at least two-thirds covered or a keylock override is actuated and (2) the drywell pressure is at least 1 psig. On the other hand, the criterion for drywell venting is that the drywell pressure must be <2 psig.* Therefore, in most accident situations, drywell flooding cannot be initiated if drywell venting is available and vice versa, unless the drywell is maintained at a pressure >1 psig while being vented through the standby gas treatment (SGT) system and interlocks are overridden by means of installed keylock bypass switches in the control room.†

Table D.3 includes a tabulation of the resulting pressure in the compressed drywell atmosphere after flooding to the level indicated in Fig. D.1 is completed without venting. It shows that pressures of >70 psia would result depending upon the initial drywell pressure at the time the wetwell water level covers the vacuum breakers. In each case, the final pressure was calculated only from consideration of the compression of the initial drywell atmosphere; the partial pressure of steam generated from the water in contact with the vessel surface has not been included. Because the RHRSW pumps, which would be used for drywell flooding, have a total pumping head (shutoff) of 180 psi, in most severe-accident situations the drywell could not be flooded to the required level unless it were vented.

In summary, if direct drywell flooding is to be used as an accident mitigating action after the core has been uncovered, then the current BFNP safety system interlocks would have to be modified or overridden to make provisions for the occurrence of drywell flooding and drywell venting simultaneously. Furthermore, the current RHRSW pumping capabilities at the BFNP would have to be increased to allow direct drywell flooding to be accomplished in a period of 30 min or less. However, as discussed in the next section, drywell flooding may not be a reasonable action per se.

D.6 Discussion of Unresolved Phenomenology

In performing this analysis of drywell flooding, several questions arose that could not be addressed within the scope of this project. The purpose of this section is to present a discussion of these questions.

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*The containment ventilation system valves, which surround the drywell, are automatically shut and held shut by the primary containment and reactor vessel isolation control system if the drywell pressure exceeds 2 psig.

†The drywell can be vented through the SGTS from the SGTS control panel located in back of the main control room panels without the need for operation of keylock bypass switches.

First, it was not determined what effect the presence of the control rod drive (CRD) mechanism penetrations within the stub tubes welded to the bottom head wall would have on the reactor vessel integrity. There are 185 4-in.-diam CRD mechanisms passing through stub tube penetrations on the bottom head of the reactor vessel. The effect of these penetrations on vessel failure in accident situations is not yet known, although two definite possibilities exist. First, the molten UO_2 may simply melt the CRD mechanism material and welds and flow out the bottom head through the CRD stub tube penetrations. This would lead to an effective failure of the bottom head with spraying of the corium through the holes. On the other hand, the molten UO_2 might not melt the CRD welds and simply remain in the bottom head until total melt-through of the vessel wall as considered in this analysis.

Another area of interest concerns steam explosions (i.e., a large production of steam over a short period of time resulting in large pressure increases). Two situations could possibly lead to the occurrence of a steam explosion. First, if a pool of water is present below the core prior to core slumping, then a steam explosion in the reactor vessel is possible, although this possibility is thought to be very small for coherent melting. For steam explosions to occur, the core must slump into the water and disperse into fragments quickly. The presence of the control rod guide tubes and other miscellaneous equipment below the core would inhibit the fast slumping of the core. However, it should be pointed out that a steam explosion, if it occurred, could lead to reactor vessel failure by overpressurization.

The second area that could lead to a steam explosion situation is one in which the drywell is not completely flooded prior to vessel failure. This situation would occur if drywell flooding were initiated by the operator but not completed by the time that vessel failure occurs. In this situation, the molten core would drop from the failed bottom head into a pool of water at the bottom of the drywell. A steam explosion under these conditions could lead to failure of the drywell itself allowing large quantities of fission products to be released into the containment building.

The final area of interest involves the problem of thermal shock. If drywell flooding is not accomplished prior to core slump but the drywell is flooded to such a point that the bottom head becomes covered prior to vessel failure, then this situation is one in which the bottom head has been significantly heated by the molten core and is then covered by the incoming water, which is at a significantly lower temperature. The problem with this situation is that the water coming into contact with the heated vessel may cause a thermal shock that induces failure of the reactor vessel.

D.7 Conclusions

This study entails a two-part analysis investigating both the effects and capabilities of direct drywell flooding at the BFNP. The first part of this study was conducted under the assumption that drywell flooding was possible and presents an analysis of the effects that drywell flooding

would have on preventing vessel melt-through. The study was performed using previously developed heat transfer models[1] and data supplied for the BFNP.[5] From the analysis, it is concluded that drywell flooding sufficient to cover the bottom head with water would preclude vessel melt-through for accident situations in which the core slump is delayed until after the completion of the flooding.

The second part of the analysis was a determination of the feasibility of direct drywell flooding given the existing system configurations at the BFNP. This analysis revealed several problems that would tend to prohibit the use of direct drywell flooding as an operator mitigating action. It was found that with the current design, direct drywell flooding would not be an effective operator action to prevent vessel melt-through if it were initiated after core uncovery.* This conclusion is based on both the necessity for drywell venting in order to flood the drywell to the required level and the inordinate time required to inject this volume of water with the existing pumping capacity.

References

1. R. C. Hagan, *Transient Thermal Analysis of Molten Core-Reactor Vessel Systems*, University Microfilms International, 1980.

2. Staff Report, *Systems Manual - Boiling Water Reactors*, Nuclear Regulatory Commission.

3. R. O. Wooton and H. I. Avci, *MARCH Code Description and User's Manual*, Battelle Columbus Laboratories, 1980.

4. D. H. Cook et al., *Station Blackout at Browns Ferry Unit One - Accident Sequence Analysis*, ORNL/NUREG/CR-2182/V1 (November 1981).

5. *Browns Ferry Nuclear Plant, Final Safety Analysis Report*, Tennessee Valley Authority.

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*It should be recalled that the RHR leads into the reactor vessel as well as through spray headers into the drywell and wetwell. It is assumed here that the reactor vessel pressure during core melt-through would be greater than the shutoff head of the RHR and RHRSW pumps. Otherwise, the water pumped into the containment could instead be pumped directly into the reactor vessel and employed to keep the core covered.

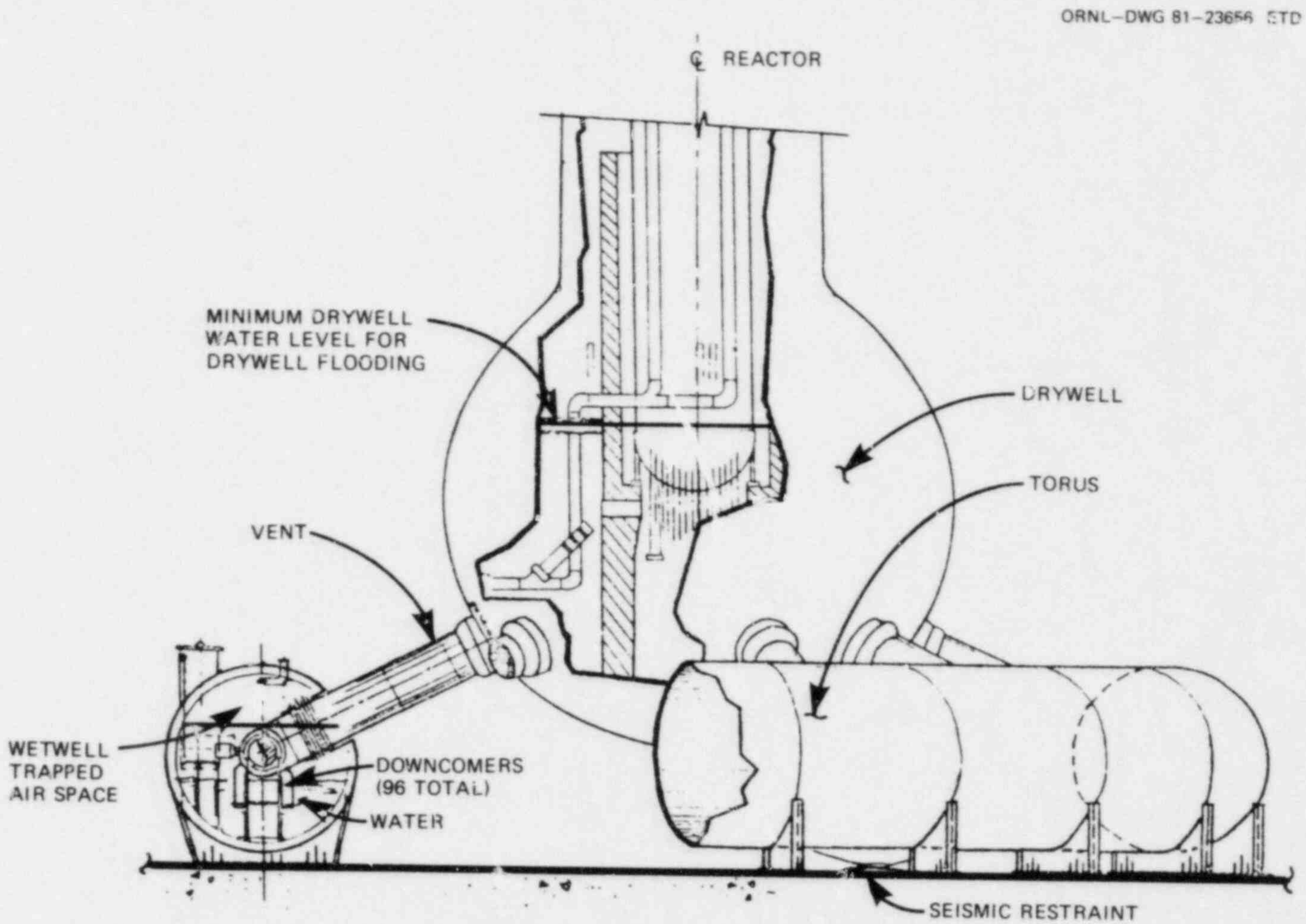

Fig. D.1. Mark I pressure suppression containment system.

ORNL-DWG 81-23650 ETD

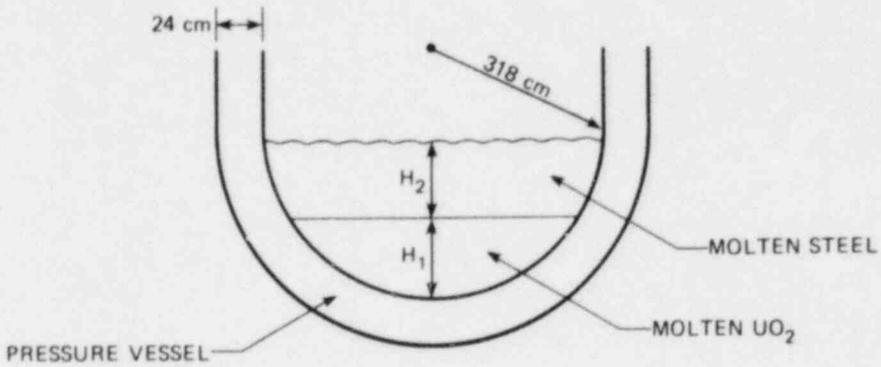

Fig. D.2(a). Bottom reactor head with core debris.

ORNL-DWG 81-23651 ETD

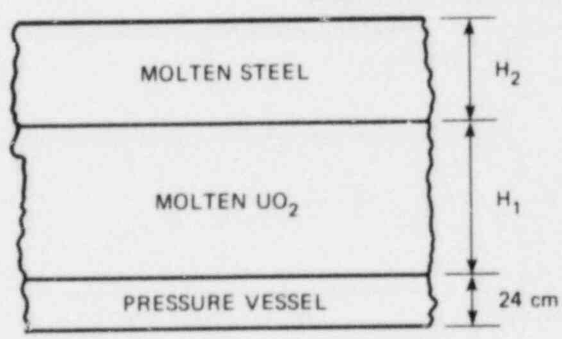

Fig. D.2(b). Computer representation of bottom head and debris.

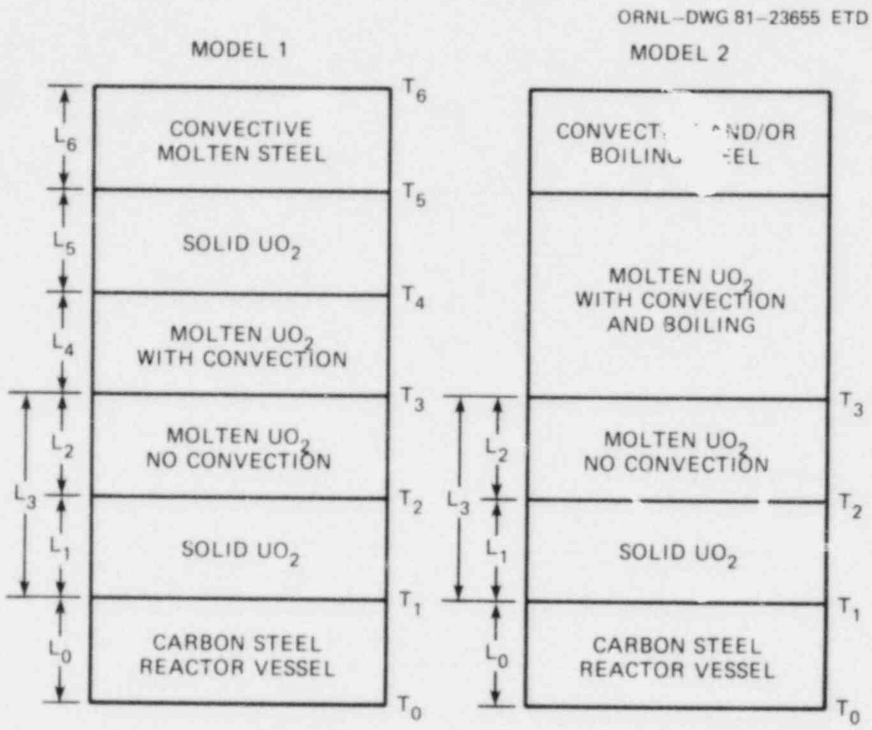

Fig. D.3. Model descriptions.

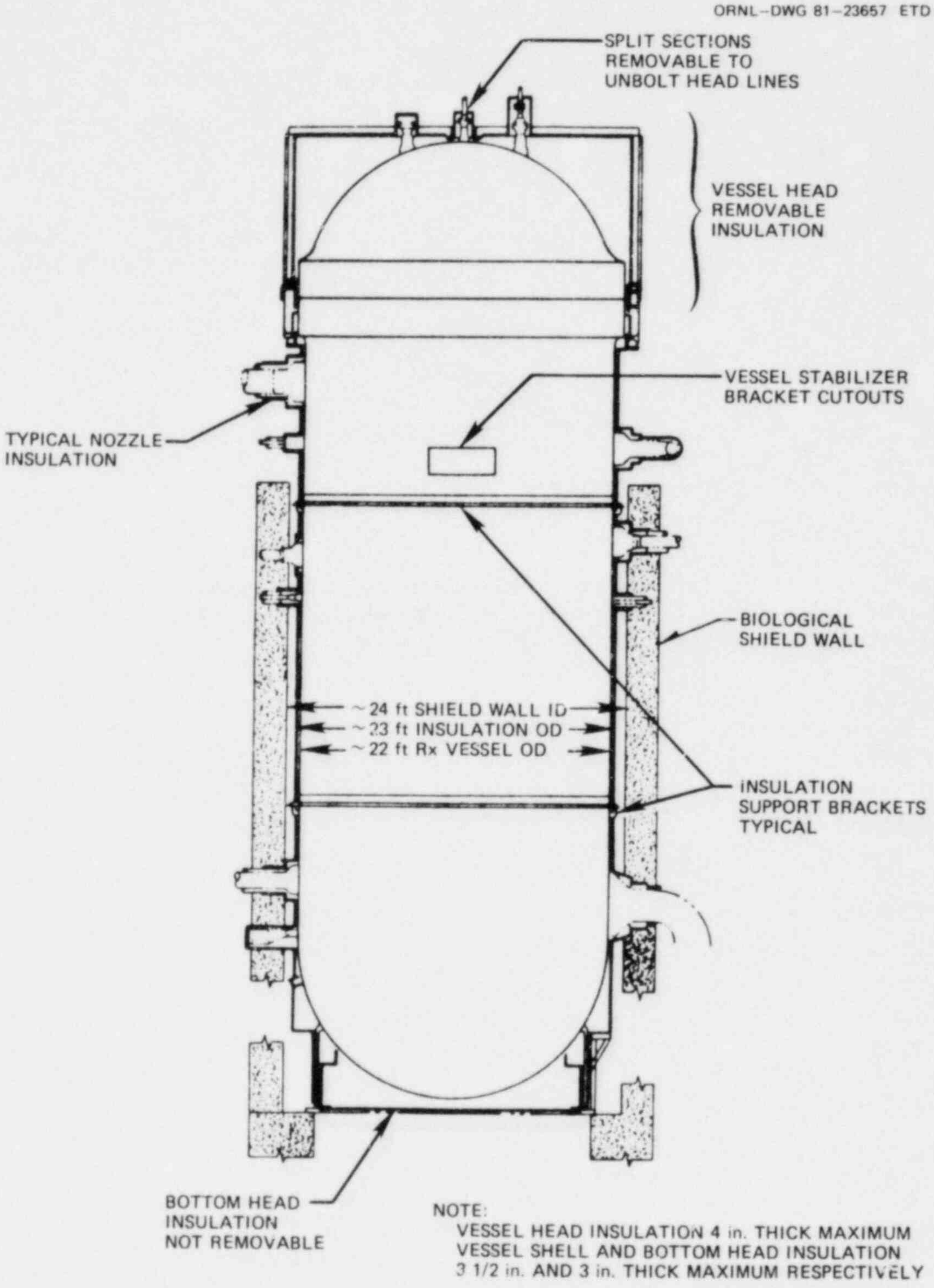

Fig. D.4. Reactor vessel insulation. Source: NRC, *Systems Manual — Boiling Water Reactors*.

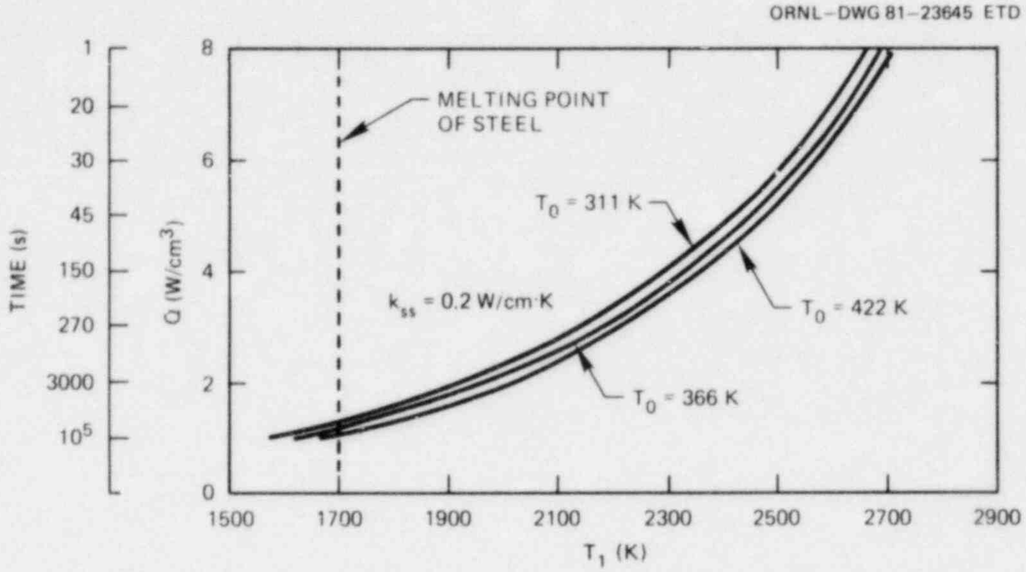

Fig. D.5. Decay heat as a function of inside vessel bottom head temperature (T_1) for various outside bottom head temperatures (T_o), and k = 0.2 W/(cm·K), Model 1.

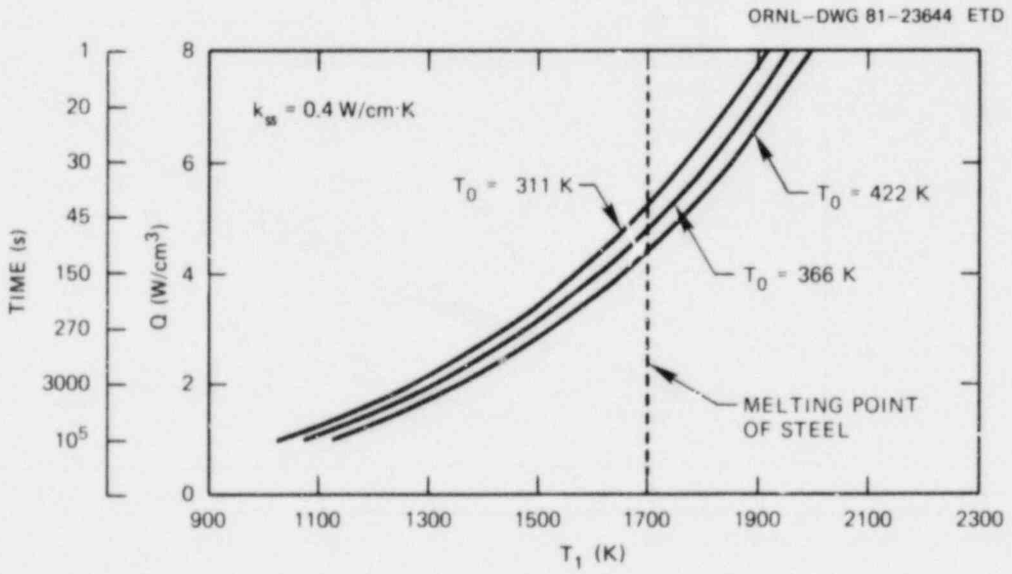

Fig. D.6. Decay heat as a function of inside vessel bottom head temperature (T_1) for various outside bottom head temperatures (T_o), and k = 0.4 W/(cm·K), Model 1.

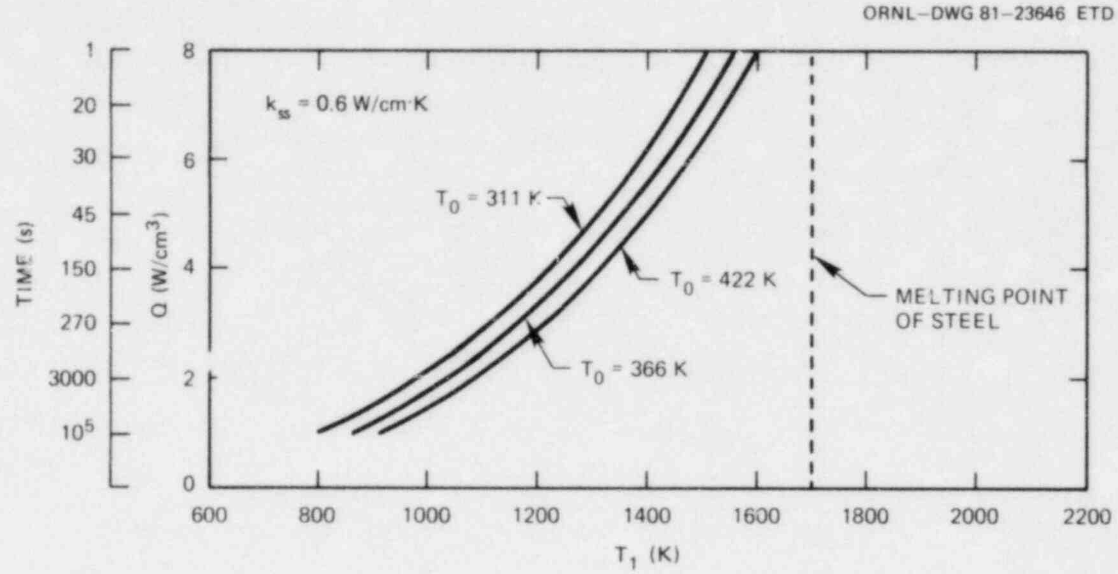

Fig. D.7. Decay heat as a function of inside vessel bottom head temperature (T_1) for various outside bottom head temperatures (T_0), and k = 0.6 W/(cm·K), Model 1.

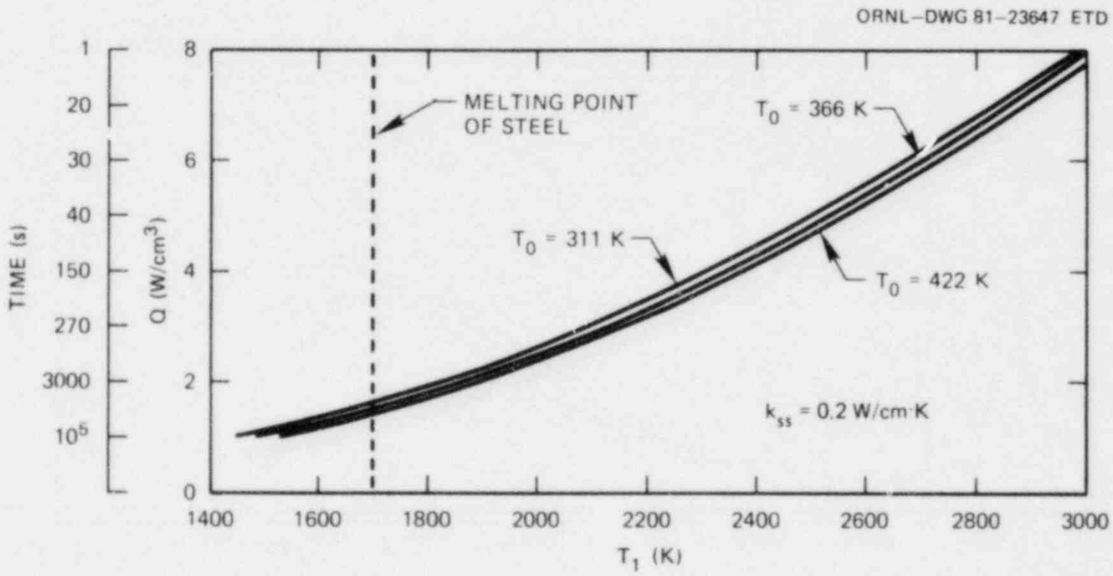

Fig. D.8. Decay heat as a function of inside vessel bottom head temperature (T_1) for various outside bottom head temperatures (T_0), and k = 0.2 W/(cm·K), Model 2.

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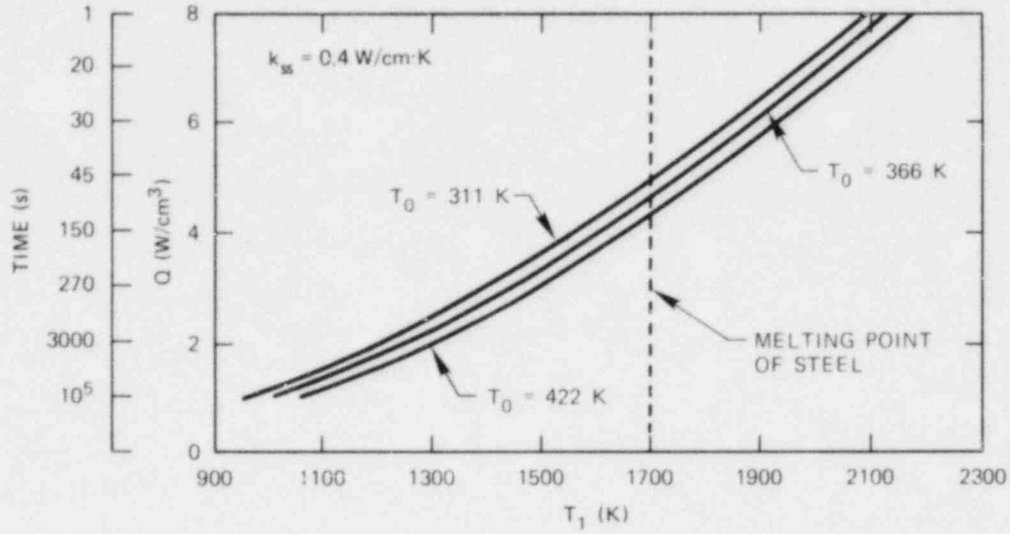

Fig. D.9. Decay heat as a function of inside vessel bottom head temperature (T_1) for various outside bottom head temperatures (T_0), and k = 0.4 W/(cm·K), Model 2.

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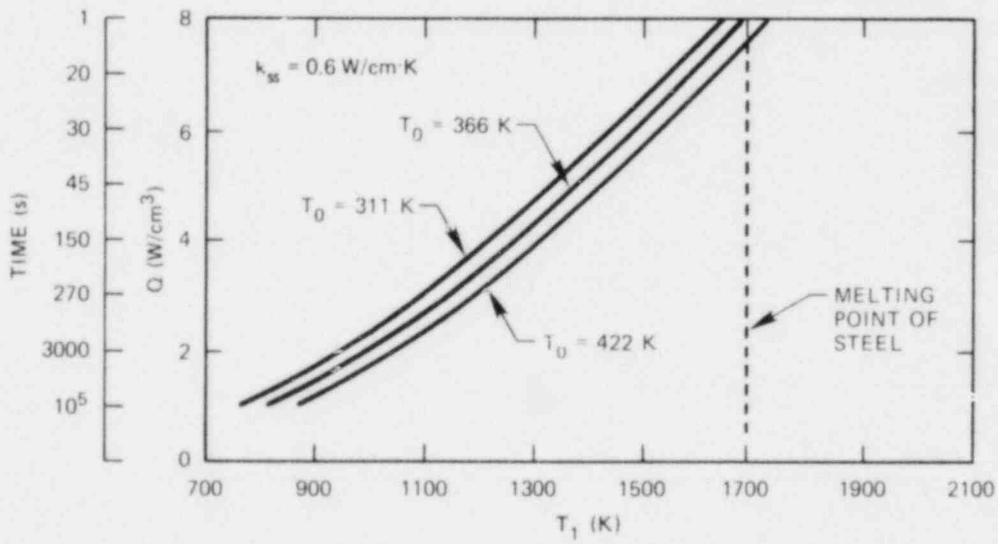

Fig. D.10. Decay heat as a function of inside vessel bottom head temperature (T_1) for various outside bottom head temperatures (T_0), and k = 0.6 W/(cm·K), Model 2.

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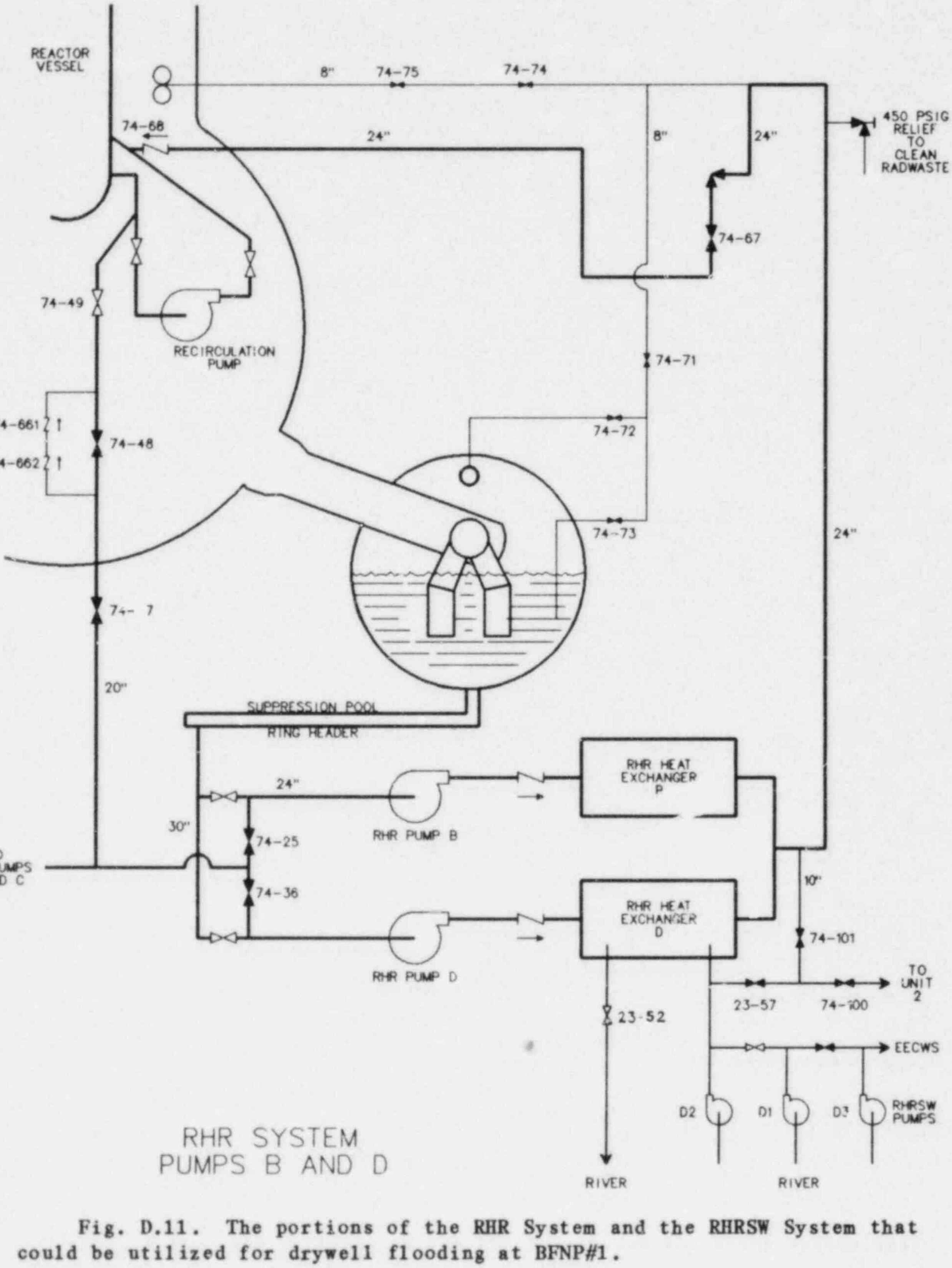

Fig. D.11. The portions of the RHR System and the RHRSW System that could be utilized for drywell flooding at BFNP#1.

Table D.1. Model 1 heat transfer equations

Layer	Equations
Convective molten steel	$\dfrac{Q(L_4 + L_5)L_6^{0.115}}{K_s(T_5 - T_6)^{1.295}} = 0.177 \left(\dfrac{g\beta_s}{\nu_s \alpha_s}\right)^{0.295}$
Solid UO_2	$Q\ L_4 + L_5 + \dfrac{L_5^2}{2} = K_u(T_4 - T_5)$
Convective molten UO_2	$\dfrac{QL_4^{1.115}}{2K_u(T_3 - T_4)^{1.295}} = 0.177 \left(\dfrac{g\beta_u}{\nu_u \alpha_u}\right)^{0.295}$
Solid UO_2 and molten UO_2, no convection	$\dfrac{QL_3^2}{2} = K_u(T_3 - T_1)$
Reactor vessel	$QL_3 = K_s \left(\dfrac{T_1 - T_0}{L_0}\right)$

Table D.2. Wetwell airspace trapped volumes at completion of drywell flooding

Initial pressure (psia)	Final pressure (psia)	Volume of trapped air (ft^3)
14.7[a]	31.0	39,790
14.7[b]	88.1	14,013
20.0	114.0	14.735
25.0	138.4	15,172
30.0	162.9	15,469

[a]The drywell is assumed to be vented during the flooding process.

[b]For this and the following cases, it is assumed that drywell venting does not occur. The initial pressure is the pressure in the containment at the time the vacuum breakers are covered, and the final pressure is the pressure in the wetwell airspace at completion of drywell flooding.

Table D.3. Drywell pressures

Initial pressure (psia)	Final pressure (psia)	Flooded volume (ft^3)
14.7[a]	14.7	208,165
14.7	71.8	233,942
20.0	97.7	233,220
25.0	122.1	232,783
30.0	146.6	232,486

[a]Case with drywell vented:

Initial free volume in drywell = 162,008 ft^3

Final free volume in drywell = 33,053 ft^3

Wetwell air volume at start of flooding = 119,000 ft^3

Final wetwell air volumes from Table D.2.

Appendix E

DESCRIPTION OF THE SCRAM DISCHARGE VOLUME AND ASSOCIATED PIPING SYSTEMS

E.1 Introduction

The function of the scram discharge volume (SDV) piping is to provide a receiver volume for the water discharged from above the control rod drive (CRD) pistons during and following a scram. As will be described, the discharged water is a mixture of hot seal leakage from the reactor vessel and relatively cold seal leakage from below the drive pistons; it is the purpose of this appendix to describe the scram discharge volume and the associated piping systems in sufficient detail to permit an understanding of the flow paths involved.

The construction and operation of the control rod drive mechanisms are briefly discussed in Sect. E.2. The control rod drive hydraulic system provides the pressurized water for mechanism operation, and this system is discussed in Sect. E.3. The scram discharge volume and the interconnected drainage, venting, and instrument piping is described in Sect. E.4. Finally, the air and electrical control logics for the valves on the boundary of the scram discharge volume are discussed in Sect. E.5.

E.2 Control Rod Drive Mechanisms

The Browns Ferry Nuclear Plant Unit 1 (BFNP#1) reactor has 185 control rods, each driven by an individual integral drive unit contained in a housing mounted on and extending below the bottom head of the reactor vessel. Figure E.1 is a schematic drawing of the drive mechanism and housing; a detailed drawing of the actual assembly is available as Figs. 2.4–2.9 of Ref. 1.

The innermost portion of the assembly shown in Fig. E.1 is the piston tube, which is immovable and fixed to the bottom flange of the drive assembly. The stationary stop piston is mounted on the upper end of the piston tube. The movable drive piston has both inner and outer seal rings and slides up or down in the annular space between the piston tube and the fixed drive cylinder.

The index tube is attached to and moves with the drive piston, transmitting its motion to the coupling spud at the upper end of the assembly, which enters a mating socket on the control rod. When the control rod is neither being inserted nor withdrawn, it is held stationary by means of collet fingers engaged with one of the notches on the index tube as shown in Fig. E.2. As explained in the following paragraphs, the collet piston is moved upward when a rod is to be withdrawn; this causes the inner sloped surface of each collet finger to slide upward and outward over the lower conical surface of the guide cap so that the collet fingers are spread apart and no longer engage the index tube notch.

When a control rod is to be inserted, drive water at a pressure considerably higher than reactor vessel pressure enters the insert line shown

on the left side of Fig. E.1. This causes the ball check valve to seat at its lower port and the drive water pressure on the lower side of the drive piston causes the drive piston and index tube to move upward, inserting the attached control rod. The water between the index tube and the piston tube in the annular space above the drive piston is displaced upward to flow through the holes at the upper end of the piston tube and then downward within the piston tube and out through the withdraw line. It should be noted that the holes at the upper end of the piston tube are progressively closed off as the drive piston approaches the stop piston; the increasing resistance to the escape of the water from above the drive piston provides a hydraulic cushion as the control rod approaches the fully inserted position near the completion of a scram stroke.

When a control rod is to be withdrawn, it is first briefly inserted by an automatic sequence timer for a period of ~1 s to lift the index tube notch off of the collet fingers. High-pressure drive water then enters the withdraw line shown on the right side of Fig. E.1, flowing simultaneously into the lower end of the piston tube and upward through the channel leading to the bottom of the collet piston. The drive pressure pushes the collet piston upward, compressing the collet spring and causing the collet fingers to move upward and outward over the lower conical surface of the guide cap, so that the index tube notch is disengaged.* Meanwhile, the drive water flowing upward through the piston tube passes outward through the holes in the upper end and then flows downward in the annular space between the piston tube and the index tube to pressurize the relatively small upper facial area of the drive piston. The drive piston moves downward, withdrawing the index tube and the attached control rod as the water below the drive piston is discharged out the insert line.

When the control rod has been withdrawn the desired distance, the withdraw drive water pressure is relieved. The collet spring then pushes the collet piston downward, causing the collet fingers to move downward and inward to press against the surface of the index tube. The downward motion of the control rod and index tube is arrested when the collet fingers have dropped into the next index tube notch, thereby locking the control rod in place.

Under all circumstances except scram, the control rods are moved one at a time, but when a scram occurs, all 185 control rods are driven into the reactor simultaneously. As discussed in Sect. E.3, the water pressure at the insert line of each of the drive mechanisms during a scram is provided by individual accumulators, initially charged to ~2.41 MPa (350 psi) above normal reactor operating pressure. Because the pressure at the withdraw line is only slightly above atmospheric during a scram, the large pressure differential across the drive piston ensures a rapid initial inward control rod movement. However, the drive pressure decreases during a scram as the accumulators discharge, and when the pressure has decreased to below reactor pressure, the ball check valve shown on the lower left side of Fig. E.1 repositions to seat at the upper port. This permits the water from the reactor vessel, which flows downward in the annular space between the control rod drive mechanism housing and the outer surface of

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*The collet fingers are shown in the disengaged position in Fig. E.1.

the drive cylinder, to pass into the volume beneath the drive piston and provide the force needed to complete the scram stroke.

The control rod drive mechanism is designed with simple seals and bushings on the drive piston and stop piston. The seals are constructed of Graphitar 14, which is inert and has a low coefficient of friction when water lubricated, as in this application, but loses strength at temperatures >120°C (250°F). Accordingly, the pressure in the insert line is maintained ~0.14 MPa (20 psi) above reactor pressure under normal operating conditions (i.e., when the control rod is not being moved and the withdraw line is blocked). This insert line pressure causes a small cooling flow to leak past the drive piston and stop piston seals into the reactor vessel.*

The brief description of the construction and operation of the control rod drive mechanisms provided in this section is intended to provide the background necessary to an understanding of the material discussed in the main body of the report. Discussions of the design and operation of these mechanisms are available in much greater detail elsewhere.[1,2]

E.3 Control Rod Drive Hydraulic System

As described in Sect. E.2, each control rod drive mechanism is a double-acting, mechanically latched hydraulic cylinder using pressurized water as the operating fluid. The pressurized drive water is provided at the proper pressure and flows by the control rod drive (CRD) hydraulic system. A schematic drawing of this system at BFNP#1 is shown in Fig. E.3, in which one of the 185 rod drive mechanisms is represented by the simple piston-in-cylinder symbol at the upper center.

As shown at the lower left side of Fig. E.3, two CRD pumps are associated with the rod drive hydraulic system of Unit 1. Pump 1A is normally in service to the Unit 1 system, whereas Pump 1B is an installed spare that is normally idle but can be valved into either the Unit 1 or the Unit 2 system. Under normal conditions, Pump 1A discharges water from the Unit 1 condensate storage tank into the pump minimum flow line back to the condensate storage tank (20 gpm), into the flushing line for the Unit 1 recirculation pump seals (8 gpm), and into the CRD hydraulic system (60 gpm).

The flow into the CRD hydraulic system is maintained at 60 gpm by a flow control station, which consists of a venturi flow element, transmitter, flow controller, and a set of parallel air-operated flow control valves (shown as one valve in Fig. E.3 for simplicity). The "charging header" taps off the main line between the venturi flow element and the flow control valves and is one of the four headers connecting the main CRD hydraulic system piping to the Unit 1 hydraulic control units (HCUs).

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*Most of the CRD mechanism cooling water flows from the insert line into the reactor vessel through the annulus formed between the housing and the outer surface of the drive cylinder. The cooling water enters this annular space through a drilled set screw (not shown in Fig. E.1) mounted in the bottom flange.

As indicated on Fig. E.3, each of the 185 control rod drive mechanisms is connected to its individual hydraulic control unit (HCU). Each HCU comprises a four-valve manifold to govern rod insertion or withdrawal, a set of scram inlet and outlet valves, and a scram accumulator, as well as several manual isolation valves which are not important to an understanding of basic system operation and not included in Fig. E.3. Figure E.4 is an isometric drawing of an actual HCU.

During normal operation, the function of the charging header is to ensure that the scram accumulators contained within each hydraulic control unit remain fully charged. As shown on Fig. E.3, the accumulators float at the CRD pump discharge pressure, which is established at ~9.76 MPa (1400 psig) by the flow control station, and there is no flow in the charging header under normal operating conditions.

When a scram occurs, the scram inlet valve and the scram outlet valve open on each HCU. The water displaced from above the drive piston of each control rod drive mechanism is displaced through the scram outlet valve to the scram discharge volume, which is described in Sect. E.4. With the scram inlet valve open, the scram accumulator discharge is passed to the volume under the drive piston, forcing the control rod up into the reactor as discussed in Sect. E.2.

When the scram accumulators are fully discharged, the flow through the 185 open scram inlet valves is maintained by a large flow established in the charging header. In attempting to maintain a flow of 60 gpm at the venturi, the flow control station valve will fully shut so that all of the CRD pump discharge except that diverted into the minimum flow and recirculation pump seal flushing lines is passed into the charging header. As shown on Fig. E.3, a restricting orifice is provided in the charging header to limit this flow to a maximum of 170 gpm. When the control rods are fully inserted, this flow continues into the control rod drive mechanisms where it leaks past the drive piston seals into the volume above the pistons; from there the flow passes through the scram outlet valves to the scram discharge volume until that volume is filled and then leaks past the stop pistons into the reactor vessel. The drive piston seal leakage is such that the scram accumulators cannot be recharged until the scram is reset, at which time the scram inlet (and outlet) valves are shut.

The second of the four headers from the main CRD hydraulic line is the drive header. The inlet to this header is located downstream from the flow control station and upstream of the drive water pressure control station, which is set to maintain a drive pressure of ~1.79 MPa (260 psi) above reactor vessel pressure. As shown on Fig. E.3, the function of the drive header is to provide pressurized water to the four-valve manifolds within each of the HCUs.

It is important to recognize that other than for a scram, the control rods are moved individually (i.e., only one is repositioned at a time). When the two valves marked "INSERT" on Fig. E.3 are opened, a flow path exists from the drive header to the volume below the drive piston of the control rod drive mechanism and a second flow path is established from the volume above the drive piston into the exhaust header, resulting in upward movement of the drive piston and the attached control rod. Alternatively, when the two valves marked WITHDRAW are opened, the drive header is connected to the above-piston volume, the below-piston volume is connected to the exhaust header, and the control rod is withdrawn. Under normal operating conditions the valves in the four-valve manifold are shut, the control rods are stationary, and there is no flow in the drive header.

It is desirable that the flow through the drive water pressure control station should remain constant during control rod manipulation, and the stabilizing valves shown on Fig. E.3 are provided for this purpose. Because of the difference in above-piston and below-piston facial areas (cf. Fig. E.1), somewhat mitigated by the seal leakage into the reactor vessel from the above-piston volume, the flow into the drive header is 4 gpm for rod insertion and 2 gpm for rod withdrawal. Thus, the INSERT stabilizing valve is adjusted to a flow of 4 gpm under normal operating conditions when there is no flow in the drive header and is automatically shut when a control rod is being inserted. Similarly, the 2-gpm WITHDRAW stabilizing valve is shut only when a control rod is being withdrawn. As shown on Fig. E.3, this arrangement results in a constant flow of 54 gpm through the drive water pressure control station.

As discussed in Sect. E.2, the control rod drive mechanism seals for the drive piston and the stop piston are composed of Graphitar-14, which is subject to degradation at elevated temperatures. Accordingly, the control rod drive mechanisms are cooled by a continuous flow from the insert line into the reactor vessel of ~0.324 gpm per mechanism under normal operating conditions. This total cooling flow of ~60 gpm is provided by the cooling header whose inlet is between the drive water pressure control station and the cooling pressure control station (Fig. E.3). The action of the flow control station to provide a flow of 60 gpm into the CRD hydraulic system also results in the proper differential pressure at the cooling water header to maintain this flow into the reactor vessel; this flow is insufficient to induce drive piston and control rod movement.

It should be noted that the 60-gpm flow into the cooling header consists of 54 gpm from the drive water pressure control station and 6 gpm from the cooling pressure control station, the latter being the same flow that has passed through the stabilizing valves. These flows differ from those intended in the original design in which valve 85-50, shown as shut on Fig. E.3, was to be open during normal reactor operation so that excess CRD hydraulic system flow not used for seal cooling would be passed into the reactor vessel via a direct connection.* With valve 85-50 shut, the water discharged from a drive mechanism during a normal control rod insertion or withdrawal flows through the exhaust header, through the cooling pressure control station, and through the cooling header into the below-drive-piston volumes of the other 184 control rod drive mechanisms that are not being operated.

As discussed in the preceding paragraphs, ~60 gpm is pumped from the condensate storage tank into the reactor vessel via the control rod drive hydraulic system as cooling flow for the control rod drive mechanisms under normal operating conditions. The control rod drive hydraulic system can also be used as a high-pressure injection system with a much higher injection flow under emergency conditions. This is accomplished by opening valve 85-50 and normally shut valve 85-551 in the pump test line and closing valve 85-519 in the pump minimum flow line. This will cause the

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*The system is now operated with valve 85-50 shut because of thermal stress problems associated with direct injection of relatively cold water into the reactor vessel.

pumped flow to bypass the system flow and pressure control stations, and at normal operating pressures, ~200 gpm will enter the reactor vessel via feedwater line B (Fig. E.3). This injection flow can be further increased by opening the CRD pump discharge cross-connect valve and starting the spare pump 1B, and by fully opening the throttling valve (not shown in Fig. E.3) in the common discharge line from pumps 1A and 1B.

E.4 Scram Discharge Volume

The function of the scram discharge volume is to receive the water discharged from above the drive pistons of the CRD mechanisms during the scram stroke and to receive and hold the subsequent seal leakage flow until the scram is reset and the scram outlet valves are shut. Because the scram outlet valves are located within the HCUs, the scram discharge volume piping is located in close proximity to these units, which are arranged in two groups on the 565-ft elevation of the reactor building, outside of and on opposite sides of the primary containment (Fig. E.5). Each group of HCUs consists of two parallel rows of units with the scram discharge volume piping located above (Fig. E.6).

A 3/4-in. scram discharge line leads from the scram outlet valve in each HCU to the scram discharge volume (SDV), which comprises the two sets of four 6-in. headers shown in an isometric view in Fig. E.7. These headers drain into a 12-in. scram discharge instrument volume (SDIV), which in turn has a drain valve into a 4-in. drain header leading to the reactor building equipment drain tank. As shown in Figs. E.6 and E.7, each set of scram discharge volume piping is vented through a 1-in. line and vent valve to the radwaste system. After the partial failure-to-scram at BFNP Unit 3 on June 28, 1980, these vent lines were also permitted ingress from the reactor building atmosphere via a vacuum breaker downstream of the vent valves.[3] The control system for the vent valves and the drain valve is discussed in Sect. E.5.

Under normal operating conditions, the vent valves and the drain valve are open so that the scram discharge volume (SDV) and the scram discharge instrument volume (SDIV) are drained and at atmospheric pressure. As indicated on Figs. E.6 and E.7, the SDIV is fitted with six float switch level sensors to detect any accumulation of water during reactor operation. These sensors protect against an inability to scram because of an accumulation of water in the scram discharge volume, providing an alarm at 3 gal, a rod block at 25 gal, and a scram should the level in the SDIV correspond to a volume of 50 gal or more.

When a scram occurs, the scram inlet and outlet valves on all of the hydraulic control units open and the two SDV vent valves and the SDIV drain valve automatically shut. During the scram strokes, ~3.3 gal of water are displaced from the volume above the drive piston in each control rod drive mechanism, and this total flow partially fills the scram discharge volume. When all control rods are fully inserted, leakage from the reactor vessel past the stop piston seals and from the charging header past the drive piston seals mixes in the volume above the drive piston in each drive mechanism and flows through the open scram outlet valves until

the scram discharge volume is filled and pressurized to full reactor pressure.

When the scram is reset by the operator, the scram inlet and outlet valves shut, and the two SDV vent valves and the SDIV drain valve automatically open, permitting the scram discharge volume and associated piping to depressurize and drain.* It is important to note that scram signal bypass provisions do not exist for several of the scram signals and that a significant time can pass before some scrams can be reset; during this period the scram discharge volume and the associated piping would remain at full reactor pressure. The signal identification, setpoints, and bypass provisions for reactor scrams from normal full power operation are listed in Table E.1. Scrams without bypass can be reset when the condition causing the scram has cleared.

E.5 Scram Discharge Volume Valve Control

The operation and control logic for the scram inlet and outlet valves, the SDV vent valves, and the SDIV drain valve are discussed in this subsection. It should be recognized that the control rod drive hydraulic system and the scram discharge volume piping were designed for maximum scram reliability, as reflected by the design provisions described in the following paragraphs.

As previously discussed, each hydraulic control unit (HCU) comprises the scram inlet valve and the scram outlet valve for its associated control rod drive mechanism. These scram valves are air-operated globe valves with Teflon seats, held closed by control air pressure during normal reactor operation and snapped open by internal springs when air pressure is removed. A schematic of the control air supply to the air-operators of these valves is included in Fig. E.8. As shown, the control air pressure is transmitted through the solenoid-operated backup scram valves and scram pilot valves.

There are two solenoid-operated scram pilot valves in each HCU, each energized from a separate reactor protection system (RPS) bus (A or) to remain in the position shown in Fig. E.8. When a scram occurs, both scram pilot valve solenoids are deenergized by the Reactor Protection System and both scram pilot valves reposition so that the air operators of the scram inlet and the scram outlet valves are vented to atmosphere, permitting the scram inlet and outlet valves to be opened by their internal springs. It should be noted that the piping arrangement provides that the scram inlet and outlet valves will remain shut if only one scram pilot valve is deenergized at a time.

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*The high SDIV level scram does not occur provided the operator has followed procedure and moved the reactor mode switch from RUN to SHUTDOWN or REFUEL and actuated the keylock switch bypass of the high SDIV level scram prior to scram reset. This bypass provides a control rod withdrawal block and sounds a control room alarm.

In contrast to the scram valves, the SDV vent valves and the SDIV drain valve are held open by control air pressure and are spring-loaded to shut. Each of the scram dump valve solenoids shown on Fig. E.8 is powered from a separate Reactor Protection System bus (A or B), and when a scram occurs, both solenoids are deenergized. Upon deenergization, the scram dump valves reposition to vent the air operators of the SDV vent and the SDIV drain valves to atmosphere, permitting these valves to be shut by their internal springs. If only one scram dump valve is deenergized, the SDV vents and the SDIV drain will remain open.

An SDV isolation test valve operable from the control room is provided to permit closure of the SDV vent valves and the SDIV drain valve during normal reactor operation so that excessive leakage through the scram outlet valves can be detected by monitoring the subsequent level increase in the scram discharge instrument volume. The SDV isolation test valve is normally deenergized and aligned as shown in Fig. E.8. When operated, the solenoid is energized from instrument and control bus A, and the valve repositions to vent the air operators of the SDV vent valves and the SDIV drain valve to atmosphere. Backup power is available to the SDV isolation test valve from the uninterruptible 120-V ac bus.

As shown in Fig. E.8, control air pressure to the air operators of both the scram pilot valves and the scram dump valves is transmitted from the control air supply through the backup scram valves. The backup scram valves are not intended to function as an alternate method for rapid scram of all control rods, but do provide assurance that air pressure will be removed from the air operators of the scram inlet and outlet valves in all HCUs and from the SDV vents and SDIV drain valve operators as protection against a common cause failure of the scram pilot valves and scram dump valves.

During normal reactor operation, the backup scram valve solenoids are deenergized and the valves are aligned as shown in Fig. E.8. Both Reactor Protection System channels A and B must trip to energize any or all of the backup scram valve solenoids and when this occurs, the backup scram valves realign to vent the control air lines leading to the scram pilot valves and the scram dump valves. Although the backup scram valves all actuate whenever the two Reactor Protection System channels trip, the operation of any one of these valves would be sufficient to vent the air from the supply line and accomplish a scram. Any scram accomplished solely through action of the backup scram valves would require from 15 to 20 s because of the large volume of air that must be vented through the small valve ports.

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

References

1. U.S. Nuclear Regulatory Commission, *Systems Manual-Boiling Water Reactors*, Inspection and Enforcement Training Center, Sect. 2.4.

2. Browns Ferry Tennessee Valley Authority Final Safety Analysis Report, Sect. 3.4.

3. *Analysis of Incomplete Control Rod Insertion at Browns Ferry 3*, Nuclear Safety Analysis Center NSAC/20, Institute of Nuclear Power Operations INPO/3 (December 1980).

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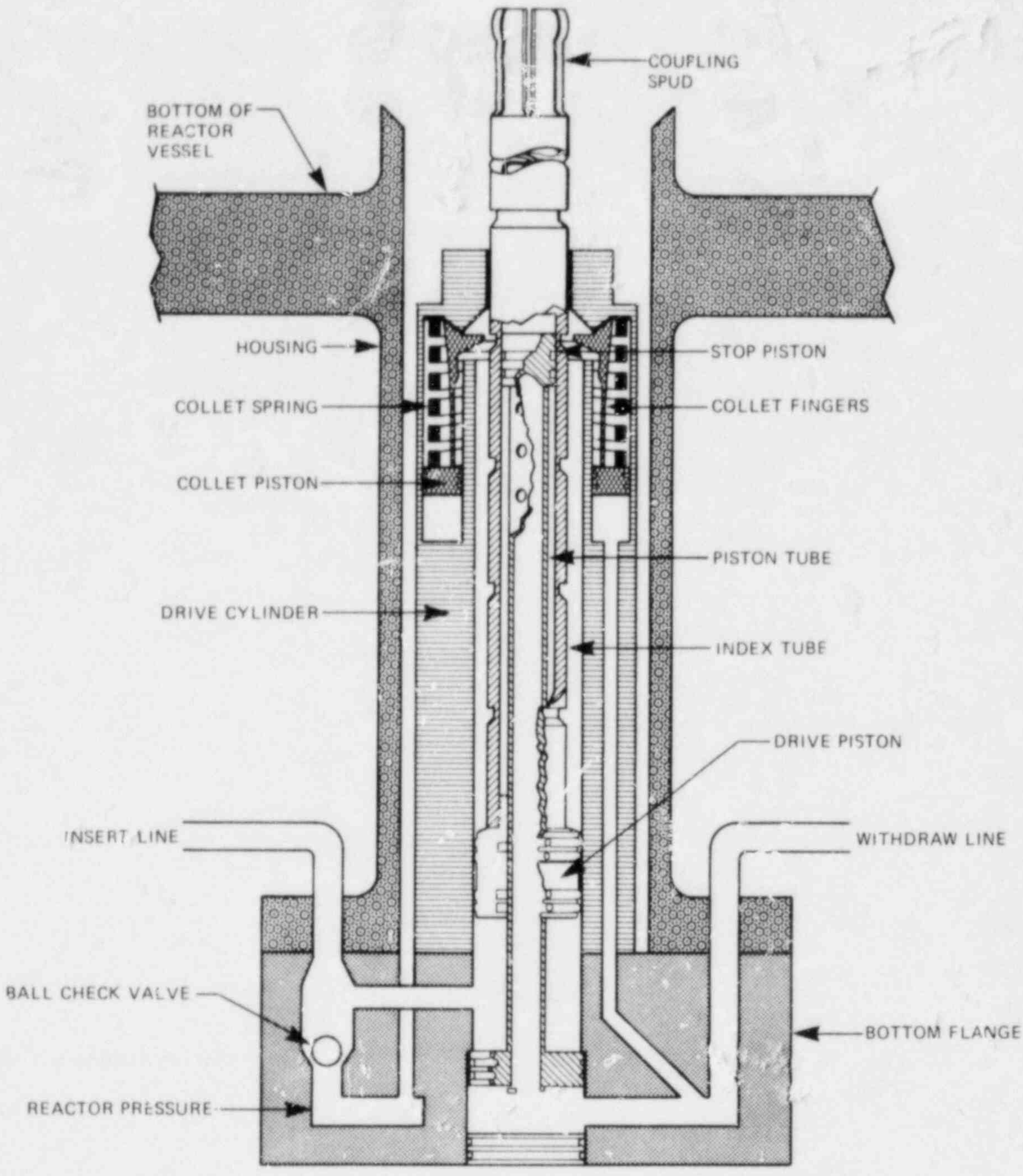

Fig. E.1. CRD schematic diagram (Adapted from NRC, *Systems Manual — Boiling Water Reactors*.

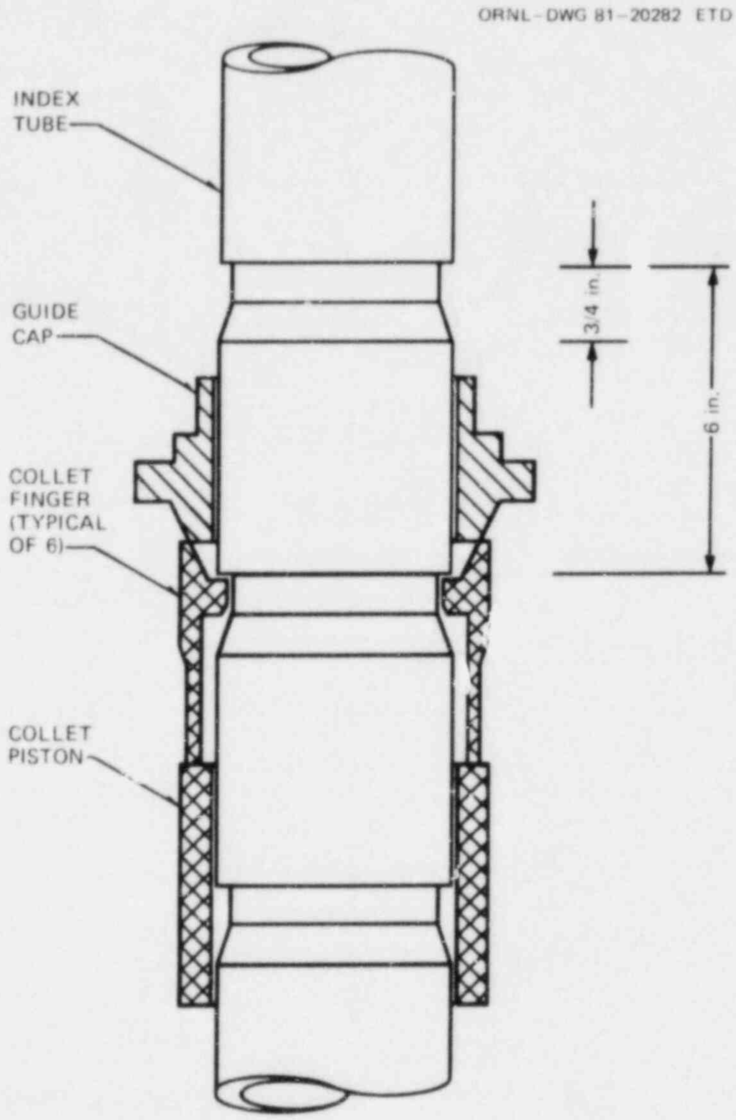

Fig. E.2. Index tube, collet piston, and guide cap arrangement (not to scale).

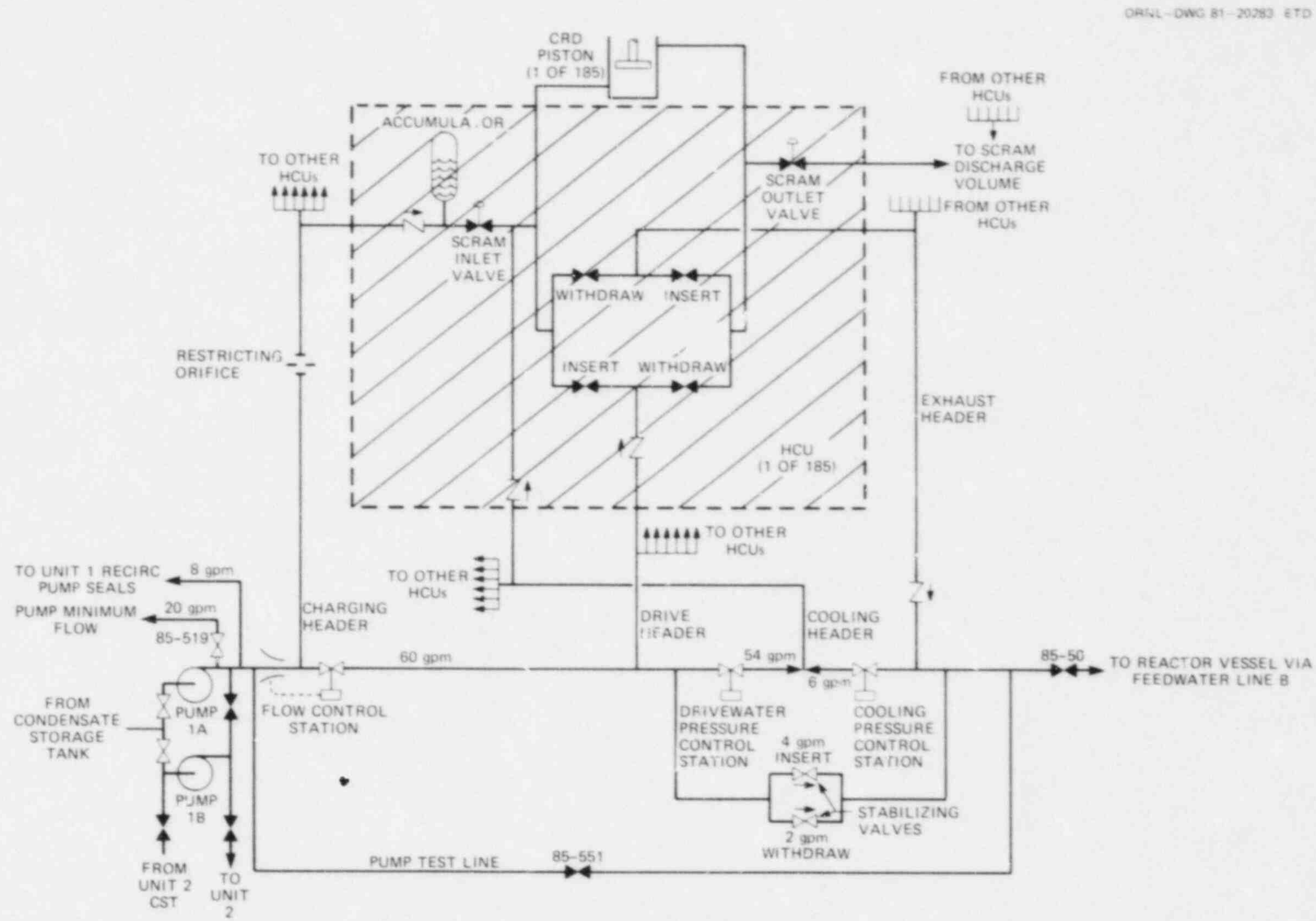

Fig. E.3. CRD hydraulic system schematic.

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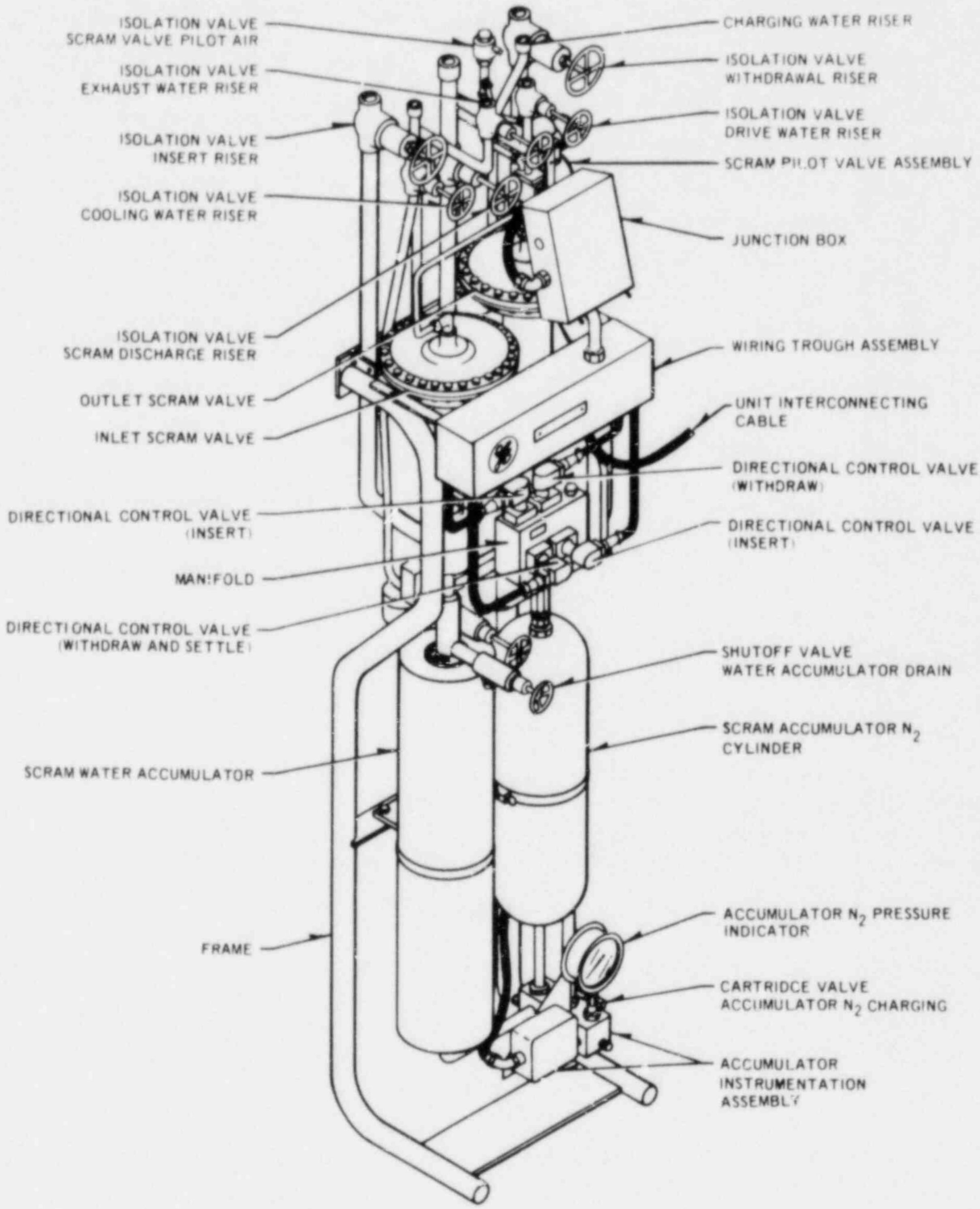

Fig. E.4. CRD hydraulic control unit. Source: *Analysis of Incomplete Control Rod Insertion at Browns Ferry 3*, NSAC/20 (December 1980).

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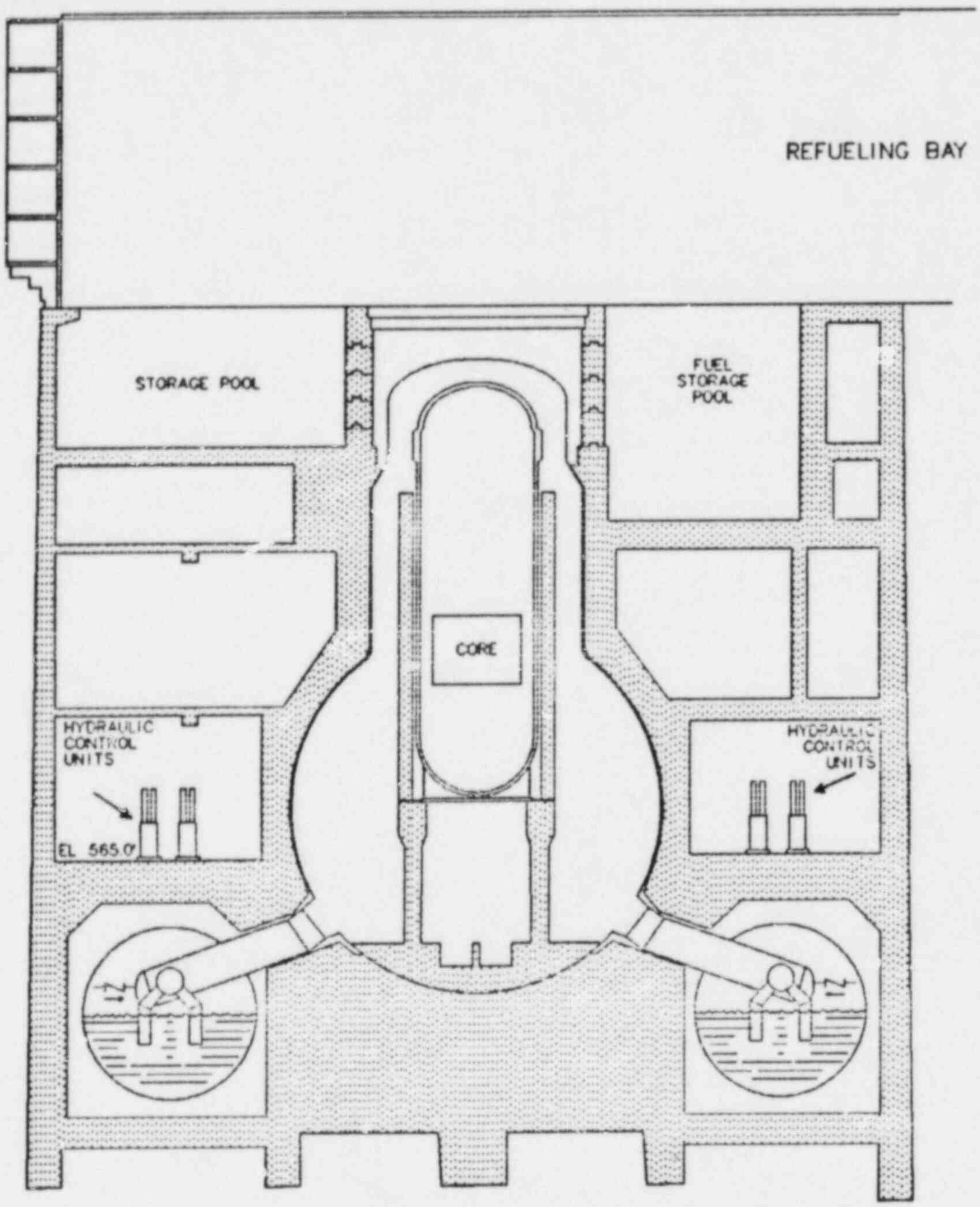

Fig. E.5. Location of hydraulic control units for the Unit 1 section of the reactor building.

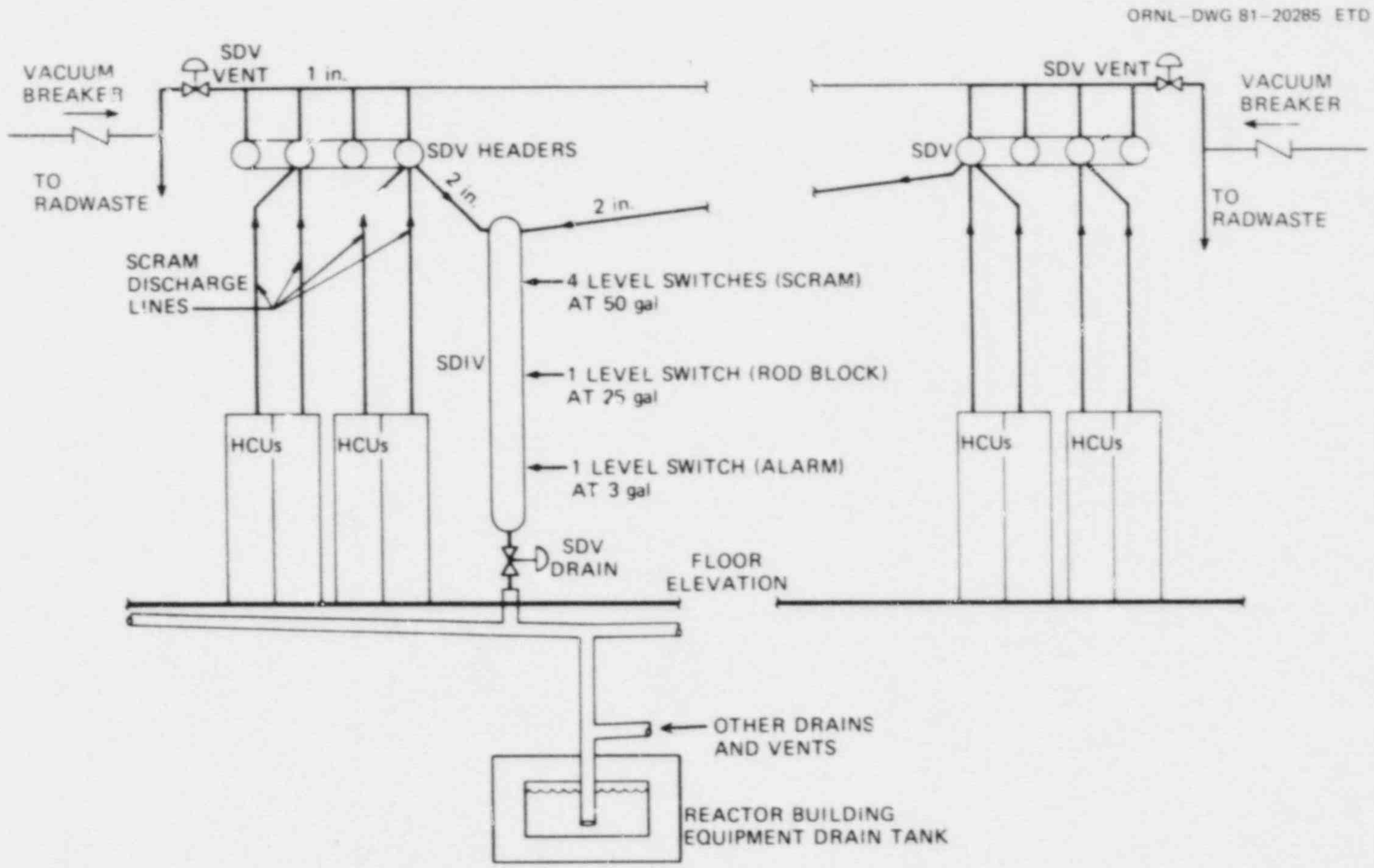

Fig. E.6. SDV piping. Source: ***Analysis of Incomplete Control Rod Insertion at Brown Ferry 3*****, NSAC/20 (December 1980).**

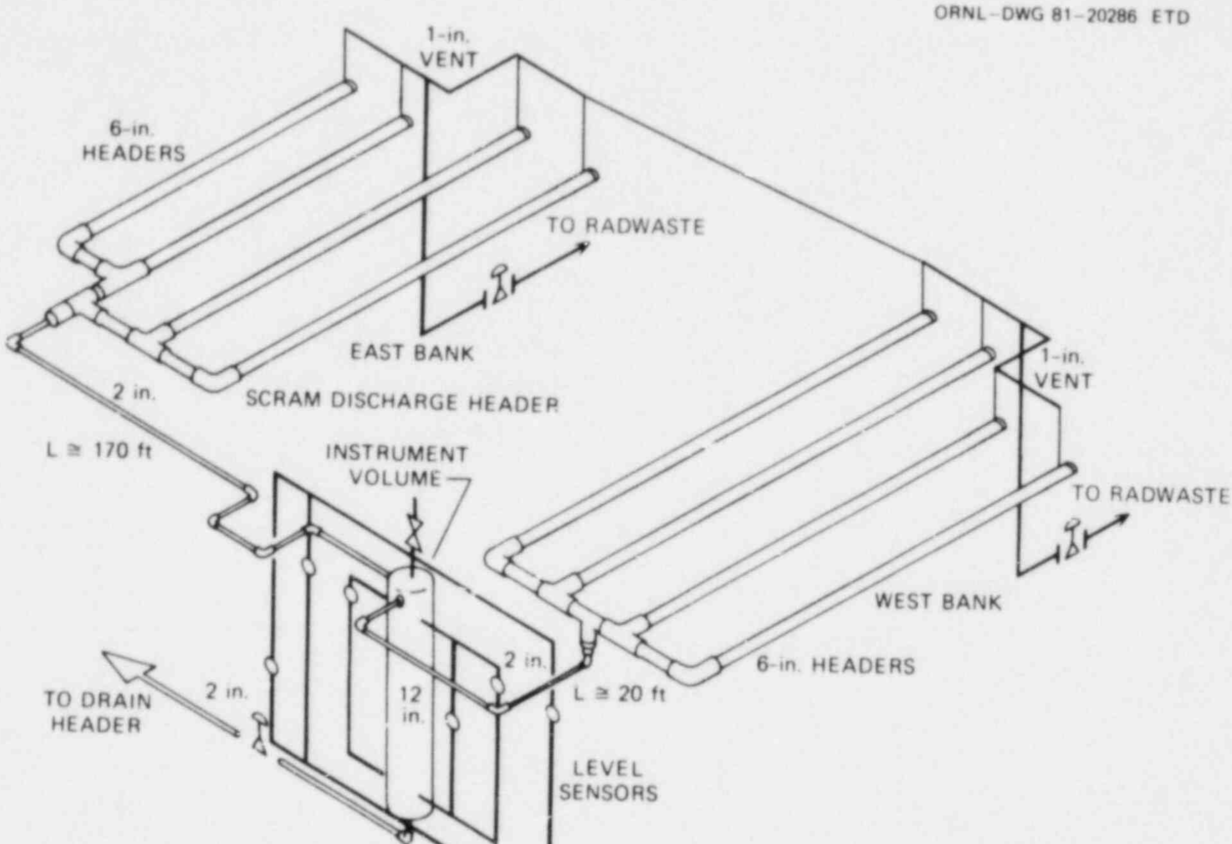

Fig. E.7. Isometric view of SDV piping. Source: ***Analysis of Incomplete Control Rod Insertion at Browns Ferry 3,*** **NSAC/20 (December 1980).**

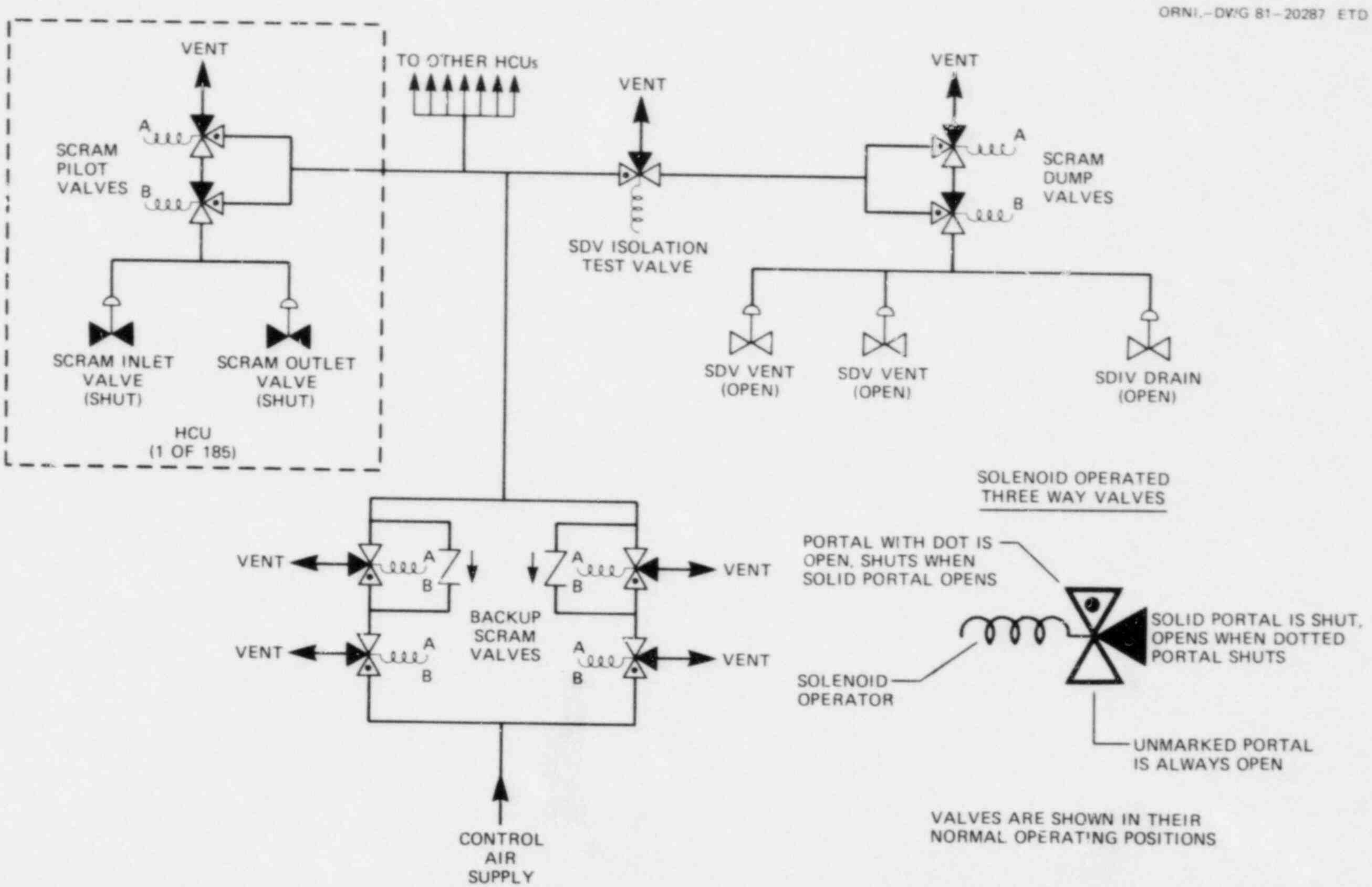

Fig. E.8. SDV valve operator air supply network.

Table E.1. BFNP scrams from full power

Signal	Setpoint	Bypass
Low reactor vessel water level	538 in.	None
Reactor vessel high pressure	1055 psig	None
Drywell high pressure	2.0 psig	None
Main steam line high radiation	3 x normal	None
High neutron flux	(0.66 W + 42)%	None
Average power range monitor	Two inoperative	None
Loss-of-RPS power		None
Loss-of-control air		None
Manual		None
Shutdown mode		None
MSIV closure	10% closed	Not in run and <1055 psig
Main condenser low vacuum	23 in. HG	Not in run and <1055 psig
Turbine stop valve closure	10% closed	<30% 1st stage pressure
Turbine control valve fast closure	850 psig	<30% 1st stage pressure
Intermediate-range monitor high or inoperative and companion average-power-range monitor downscale	APRM <3%	Not in run
Scram discharge volume level high	50 gal	Shutdown or refuel and keylock switch
Low control air pressure	60 psig	Shutdown or refuel and keylock switch

Appendix F

MARCH INPUT FOR BASE-CASE SDV BREAK ACCIDENT SEQUENCE WITH NO OPERATOR ACTION

```
SDVA NEW8 W/DESIGN ECC PARAMATERS
&CHANGE
  TRST = 1000.,
            PRST = 1000.0,
            CPSTP = 1000.,
            IS = 10,
            IFPM = 10,
            IFPV = 10,
            MEL = -1,
            FDRP = -1.0,
            TMX = -1.0,
            TFX = -1.0,
            IHOTX = 10,
            HIMX = -1.0,
            HIOX = -1.0,
            IGASX = 10,
            WALLX = -1.0,
            PFAIL = -1.0,
            ACBRK = -1.0,
            ID = 0,
            IFISH = -1,
            NCT7 = 1,
            NCRST = 1,
            LST7 = 0,
            IPLOT = 0,
&END
&FPANAL
 TIMEON=2000.0,
&END
&NLMAR
            ITRAN=1,
            IBRK=1,
            ICBRK=1,
            ISPRA=1,
  IECC=2,
            ICE=0,
            NPAIR=0,
  NINTER=18,
            IXPL=0,
            IBURN=0,
            TBURN=0.,
            H2HI=0.,
            H2LO=0.,
            IPDTL=7,
            IPDEF=0,
            IPLOT=3,
            IU=0,
  ICKV=1,
            IFPSM=2,
            IFPSV=2,
            VOLC=278000.0,
  DTINIT=0.02,
  TIME=440.0,
            TAP=1.18E6,
&END
```

```
&NLINTL
&END
 STEEL   CONCRETE
 DWLINER    CWFLOOR    UPRXPED    LORXPED    WWLINER

&NLSLAB
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          NSLAB=5,
  DEN(1)=486.924,157.481,
  HC(1)=0.1137,0.3107,
  TC(1)=25.001,0.881,
          NCD(1)=1,6,17,26,35,
          IVL(1)=1,1,1,1,2,
          IVR(1) = 1,1,1,1,2,
          NNO1(1)=5,11,9,9,5,
          NNO2(1)=0,0,0,0,0,
          MAT1(1)=1,2,2,2,1,
          MAT2(1)=1,2,2,2,1,
          SAREA(1)=18684.0,1640.3,4130.0,1815.0,17050.0,
          HIF(1)=5*0.0,
          CTOX(1)=0.0,0.0,1.0,1.0,0.0,
          X(1)=0.00,0.01,0.03,0.05,0.09375,
          X(6)=0.,0.08,0.25,0.50,0.75,1.0,1.50,2.0,3.0,4.0,4.7326,
          X(17)=0.00,0.10,0.20,0.50,1.146,1.792,2.092,2.192,2.292,
          X(26)=0.0,.25,.75,1.25,1.75,2.25,2.75,3.25,3.49,
          X(35)=0.00,0.0100,0.020,0.040,0.C625,
  TEMP(1)=34*118.0,5*120.0,
&END
&NLECC
          PUHIO=0.,
          UHIO=0.,
          PACMO=0.,
          ACMO=0.,
  TMHH=1.E8,
          PHH=1.E6,
          PHLO=100.0,
          WHHI=5000.,
          TMSIS=1.E8,
          PSIS=0.0,
          PSLO=0.,
          WSIS1=0.,
  TMLH=1.E8,
          PLH=1.E6,
          PLLO=50.0,
          WLHI=600.,
   NP = 2,
   TM(1) = 0.,  0.,
   STP(1) = 1.E6,   1.E6,
   P(1) = 295.,  289.,
   WEC(1) = -41429.,   -16346.,
   PLO(1) = 0.,   0.,
  STPHH=1.E8,
          STPSIS=1.E8,
  STPLH=1.E8,
          RWSTM=3107859.,
          ECCRC=1.,
          CSPRC=0.,
          DTSUB=-100.0,
```

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             WTCAV=1.0,
             TUHI=0.0,
             TACM=0.,
             TRWST=80.,
  STPECC=1.E7,
&END
&NLECX
&END
&NLCSX
&END
&NLCOOL
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             CQR=6.4875E6,
             CWPR=210000.,
             CTPR=150.,
             CWSR=8587.7,
             CTSR=105.,
  TCOOL=652.,
             NCOOL=0,
             CRCOOL=0.,
             PCOOL=0.0,
             PCFF=0.0,
&END
&NLMACE
  NCUB=2,
  NRPV1=3,
  NRPV2=1,
             NRPV3=0,
             ICECUB=-1,
  PO=15.7,
             FALL=0.,
  HMAX=280.0,
  DTO=0.05,
             DTS=5000.,
  DTPNT=2.,
             IDRY=-1,
             IWET=2,
             IBETA=0,
             WPOOL=7792897.,
  TPOOL=120.,
             DCF=1000.0,
             VDRY=533.7,
             VTORUS=257700.,
   WVMAX = 1282914.5,
   PRESS(1,2)=0., PRESS(2,1)=0.,
             WICE=0.,
             TICE=0.,
             TWTR=0.,
             TWTR2=0.,
             TSTM=0.,
             DCFICE=0.,
             NSMP=-2,
             NSMP2=2,
             WVMAKS=0.,
             NCAV=1,
             VCAV=133.7,
             VFLR=400.,
             FSPRA=1.,
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          IVENT=0,
          TVNT1=0.,
          TVNT2=0.,
          AVBRK=0.,
          CVBRK=0.,
  VC(1)=159000.,119000.0,
  AREA(1)=1640.3,10980.0,
          HUM(1)=0.2,1.0,
  TEMPO(1)=118.,120.,
          INERT(1)=1,1,
  N=4,
  NS(1)=                       3,     3,     2,     1,
  NC(1)=                       1,     1,     1,     1,
  NT(1)=                      -7,    -7,    -7,     7,
  C1(1)=                    400.,  500.,  177.,  652.,
  C2(1)=                    .593,  .593,  .593,  .593,
  C3(1)=                   .6374,  21.0, .6374,  .568,
  C4(1)=                      0.,    0.,    0.,    0.,
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     STPSPR=705.,
&END
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          R1 = 1,
          R2 = 10,
          NNT = 45686,
          NR = 44749,
          NDZ = 10,
          ISTR = 3,
          ISG = 0,
  MELMOD = -1,
  IMWA = 1,
  ISTM = 0,
  IHC = 0,
  IHR = 1,
  NDZDRP = 2,
          IMZ = 100,
          FR = 0.0,
          FM = 0.0,
          MWORNL = 1,
          IFP = 2,
          ISAT = 1,
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  IGRID2 = 0,
          KRPS = 0,
          TRPS = 0.0,
          ANSK = 0.0,
          TDK = 0.0,
          YT = 0.0,
          YB = 0.0,
          DTK = 1000.0,
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          TMSG1 = 1.0E06,
          TMSG2 = 1.0E06,
  TPM = 1.0,
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  AB(9)=.02222,.02234,.02247,.02  9,.02271,.02284,.02296,.023,
  TB(1)=438.,440.,443.,446.,449., 52.,455.,458.,
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        TPUMP2 = 1.0E06,
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        QPUMP2 = 0.0,
        TMUP1 = 1.0E06,
        TMUP2 = 1.0E06,
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        WMUP2 = 0.0,
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        VSHEAD=981.4,
        HSHEAD=3.72,
        ASTAND = 42.5,
        HSTAND = 6.81,
        ASEP = 173.8,
        HSEP = 7.73,
TSCT(1) = 500.0,
TSB(1) = 0.20,
        TALF1 = 1.0E10,
        TALF2 = 1.0E10,
        QZERO = 1.1242E10,
        H = 12.289,
HO = 13.64,
        DC = 15.59,
        ACOR = 108.74,
        ATOT = 261.43,
WATBH = 150103.2,
        D = 0.0424,
        DF = 0.0358,
        DH = 0.0459,
        CLAD = 0.00472,
        XOO = 0.0,
        RHOCU = 81.48,
HW = 150.0,
TGOO = 554.0,
        CSRV = 0.0,
TMELT = 4352.0,
TFUS = 5381.0,
TFAIL = 2500.,
FDROP=.75,
FCOL = .75,
FDCR = -0.50,
DPART = 0.020833,
DUO2 = 0.0358,
FZMCR = 0.05,
FZOCR = 0.08,
FZOS1 = 0.10,
F12 = 0.445,
        WFE2 = 2865.8,
TFEOO = 554.0,
        WTRSG = 0.0,
        FULSG = 0.0,
        PSG = 0.0,
PVSL = 1080.,
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  TCAV = 650.,
          YLEG = 16.0,
  ABRK=0.0,
  YBRK=-1.,
  DTPNTB = 5.0,
  DTPN = -3.0
  TPN = 1.0E06,
          VOLP = 21338.4,
  VOLS = 14482.2,
          TMAFW = 1.0E06,
          WAFW = 0.0,
          TAFW = 0.0,
          WCST = 0.0,
          F(1)=0.462,1.19,1.18,1.04,1.05,1.06,1.09,1.23,1.06,0.416,
          PF(1)=1.017,1.087,1.093,1.095,1.096,1.094,1.0875,1.128,
          PF(9)=0.9665,0.408,
          VF(1) = 10*0.1,
  TT(1) = 456.0,456.0,456.0,
          CM(1) = 2865.8,9852.3,8712. ,
          AH(1) = 286.6,5085.2,31700. ,
          DD(1) = 0.5,0.5,1.0,
          AR(1) = 286.0,332.9,72.5,
  TT(4) = 456.0,456.0,456.0,
          CM(4) = 2331.0,5259.0,23593  ,
          AH(4) = 400.0,6866.0,687.0,
          DD(4) = 0.17,0.02,0.703,
          AR(4) = 0.0,-7.083,-12.79,
  NVALVE = 13,
          RATFLO = 838900.0,
          RATPRS = 1143.0,
          RATRHO = 2.6083,
          PSET1 = 1120.0,
          TSET2 = 1.0E8,
          PSET2 = 0.0,
  WATMAS=162672.6,
&END
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          NUMVAL = 1,
          PSETDP = 0.0,
&END
&NLHEAD
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          WFEC = 26980.0,
          WUO2 = 351439.9,
          WGRID = 66750.0,
          WHEAD = 207500.0,
          TMLT = 4135.0,
          DBH = 20.915,
          THICK = 0.7031,
          CCND = 6.39,
  E1 = 0.0,
  E2 = 0.0,
  FOPEN = 0.0,
&END
&NLHOT
          IHOT = 100,
          MWR = 1,
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              DP = 0.25,
              CCN = 6.39,
              FLRMC = 3360.0,
              WTR = 0.0,
              TPOOLH = 100.0,
              NSTOP = 200,
 &END
 &NLINTR
              CAYC = 0.01524,
              CPC = 1.30,
              DENSC = 2.375,
   TIC = 308.16,
   FC1 = 0.455,
   FC2 = 0.070,
   FC3 = 0.388,
   FC4 = 0.048,
   RBR = 0.135,
              RO = 322.6,
   R = 6000.0,
              DT = 0.5,
              TF = 1.0E06,
   TPRIN = 300.0,
   DPRIN = 300.0,
              HIM = 0.20,
              HIO = 0.09,
              FIOPEN = 0.50,
              NEPS = 2,
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              EPSI(1) = 0.5,0.5,
              IWRC = 1,
              IGAS = 1,
              ZF = 1000.0,
              WALL = .001,
              TAUL = 0.5,
              TAUS = 5.0,
&END
```

Appendix G

A RELAP5 ANALYSIS OF A BREAK IN THE DISCHARGE VOLUME AT THE BROWNS FERRY UNIT ONE PLANT

EGG-NTAP-5993

A RELAP5 ANALYSIS OF A BREAK IN THE SCRAM DISCHARGE VOLUME AT THE BROWNS FERRY UNIT ONE PLANT

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ABSTRACT

As part of the Severe Accident Sequence Analysis Program, a RELAP5/MOD1 simulation of a scram discharge volume piping break at Unit 1 of the Browns Ferry Nuclear Plant has been performed. The analysis shows eventual core uncovery can occur assuming that the operator takes no action to mitigate the consequences of this small break loss of coolant accident. Included also is a comparison of results to similar calculations performed at the Oak Ridge National Laboratory.

SUMMARY

As part of the Severe Accident Sequence Analysis (SASA) Program, a scram discharge volume small break loss of coolant accident (SBLOCA) of Unit 1 of the Browns Ferry Nuclear Plant is analyzed herein. The objective of this analysis is to determine the mechanism of core uncovery assuming no operator intervention. This Idaho National Engineering Laboratory (INEL) task parallels a similar effort of the Oak Ridge National Laboratory (ORNL).

The scram discharge volume (SDV) is located in the reactor building outside of primary containment. A non-isolable SDV break is assumed to occur shortly after a reactor scram. Vessel inventory is maintained by the high pressure coolant injection (HPCI) system. As the fission decay power decreases, the HPCI depressurizes the vessel below the shut-off head of the condensate/condensate booster pumps (CBPs). The CBPs flood the vessel and deactivate the HPCI turbine.

Subsequent to vessel flooding, reactor vessel pressure remains above the shut-off head of the low pressure coolant injection and core spray systems, precluding their operation. Without means of automatic inventory replenishment, depletion can only result in core uncovery if no operator action occurs. These results agree with the ONRL results.

CONTENTS

FIGURES

TABLES

A RELAP5 ANALYSIS OF A BREAK IN THE SCRAM DISCHARGE VOLUME AT THE BROWNS FERRY UNIT ONE PLANT

1. INTRODUCTION

The basis for the investigation discussed herein is rooted in the June 13, 1980 Browns Ferry Unit 3 Boiling Water Reactor partial failure to scram incident. Subsequent studies,[1,2] conducted by the Office for Analysis and Evaluation of Operational Data (AEOD) of the Nuclear Regulatory Commission (NRC), identified a break in scram discharge volume (SDV) as a potential problem since in some instances the break cannot be isolated.

Further investigation[3] of the postulated SDV break conducted at the Oak Ridge National Laboratory (ORNL) determined that the worst SDV break scenario occurred if it is assumed that the operator did not intervene and the plant protection systems were allowed to trip automatically.

To provide an independent SDV break calculation, the worst case sequence was simulated by SASA personnel at the Idaho National Engineering Laboratory (INEL). The simplified one recirculation loop Browns Ferry model developed for the station blackout studies[4] was used to conduct the RELAP5[5] SDV break analysis assuming no-operator action.

The report provides a summary discussion of the SDV break in five sections which follow. Section 2 describes the problem, Section 3 discusses the code, the model and the initial boundary conditions. Section 4 summarizes the results. A comparison between the INEL/ORNL calculations is made in Section 5. Section 6 itemizes the conclusions and observations of the SDV break study.

2. PROBLEM DESCRIPTION

During a reactor scram the scram discharge volume (SDV) receives effluence from the control rod drives (CRDs). Normally unpressurized, the SDV becomes part of the reactor coolant system pressure boundary upon a scram and until the scram is reset. Any leakage from the SDV or its associated piping accumulates in the basement of the reactor building. Since the SDV is located outside of primary containment, as shown in Figure 1,[1] accumulation of effluence in the basement can eventually flood the vessel water inventory equipment (VWIE) located there, rendering it inoperative. In particular, the high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), residual heat removal (RHR), and core spray (CS) system pumps are located in the basement of the reactor building and will fail if flooded.

A key assumption in this accident sequence is that the operator takes no corrective or mitigating action (ORNL has demonstrated that normal operator intervention will make the transient recoverable, see Reference 3). Thus, the scram is not reset, the SDV is not isolated from the reactor, the RCIC is not reset after its injection terminates and the safety relief valves are not manually used by the operator.

Briefly, this accident proceeds as follows: The initiating event is a spurious high main steam line radiation signal which causes the main steam line isolation valves (MSIV) to close and the reactor to scram. Thirty seconds later, the SDV break occurs and effluence begins flowing into the basement. The vessel water inventory is automatically maintained by the HPCI-RCIC systems until the reactor vessel pressure falls below the shut-off head of the condensate and condensate booster pumps (CBPs). CBP injection floods the vessel and the HPCI steam turbines, thus leaving the vessel without means of inventory replenishment. The CBPs fail when their source of water i.e., the condensate hotwell is depleted.

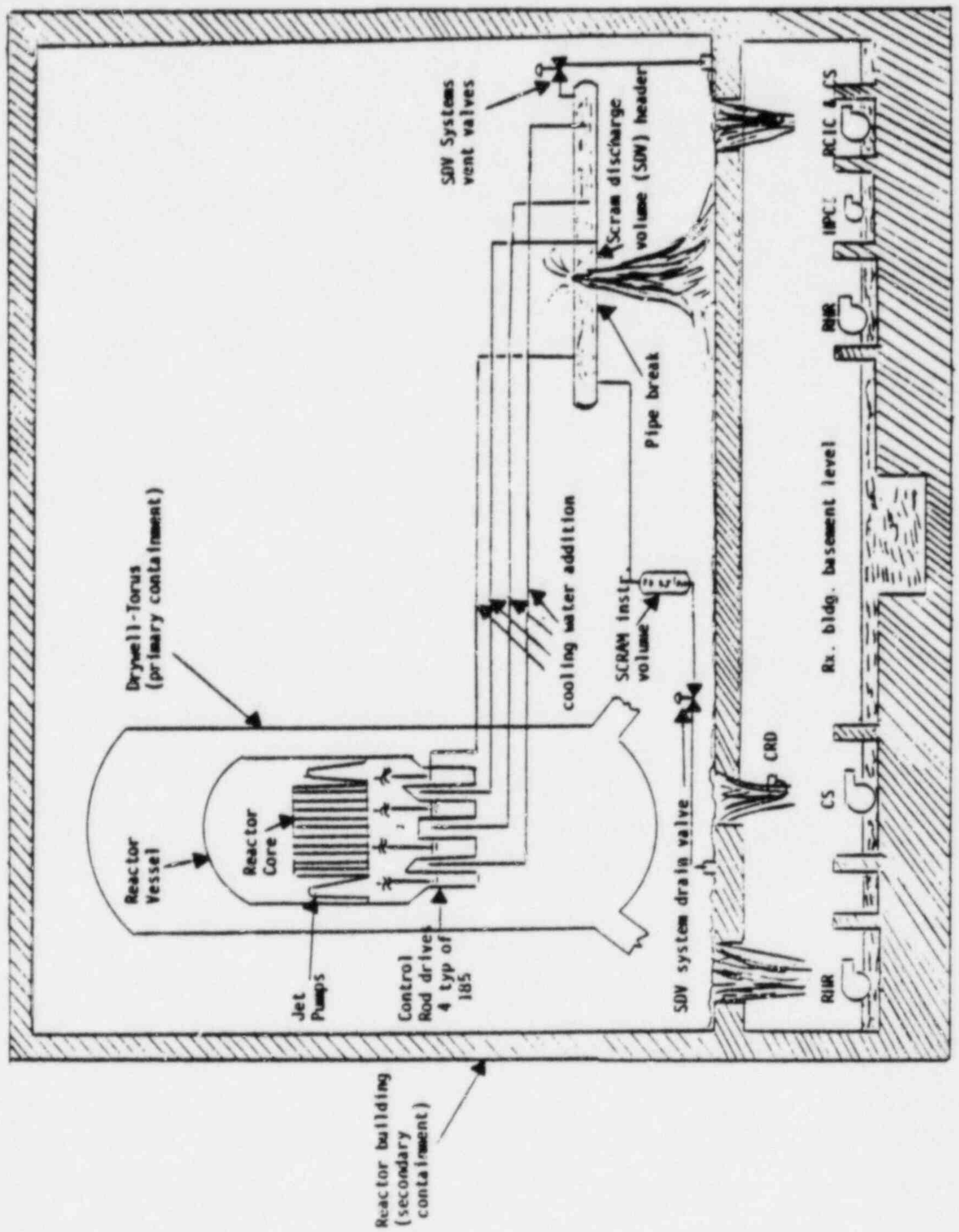

Figure 1. Reactor Building Elevation.

3. MODEL AND CODE

The overall response of the system depends upon coding, modeling, and application of boundary conditions. RELAP5[a] was used to perform the transient simulations. The model[4] used in the simulation was developed during the INEL station blackout studies. Figure 2 contains a nodalization diagram of the Browns Ferry Unit 1 model(Configuration Control No. F00930). Vessel conditions at time of reactor scram were 100 percent rated power (see Table 1).

Only minor modifications necessary to apply boundary conditions or alleviate computational difficulties were made on the Reference 4 model. The modifications implemented are listed below:

1. The break area (see junction 437 Figure 2) was sized to pass 550 gpm assuming upstream conditions of: 1050 psia, 525°F. The break occurred 30 s after the transient began (see Table 2). The break area was assumed constant for 90 min. Thereafter, the area was slowly increased over a 6.5 hr time span until 1800 gpm would leak from the vessel. Such behavior simulates the action of a line break (see Figure 1) with the break flow being limited by the CRD Graphitar seals. The slow increase assumed to begin at 90 min is physically caused by erosion of the seals. The maximum area obtained at 8 hr in the transient represents the complete erosion of the seals.

2. The level trip logic was improved from the Reference 4 model. The HPCI/RCIC systems were modeled to trip on when the downcomer water elevation was 476 in. and off at the 582 in. level. However, to prevent spurious elevation signals from either tripping the systems off or on incorrectly, the logic was modified to include a three point moving average computer;

---

a. RELAP5 MOD1 Cycle 13, Code Configuration Number F00341 with updates was used (see Reference 5)

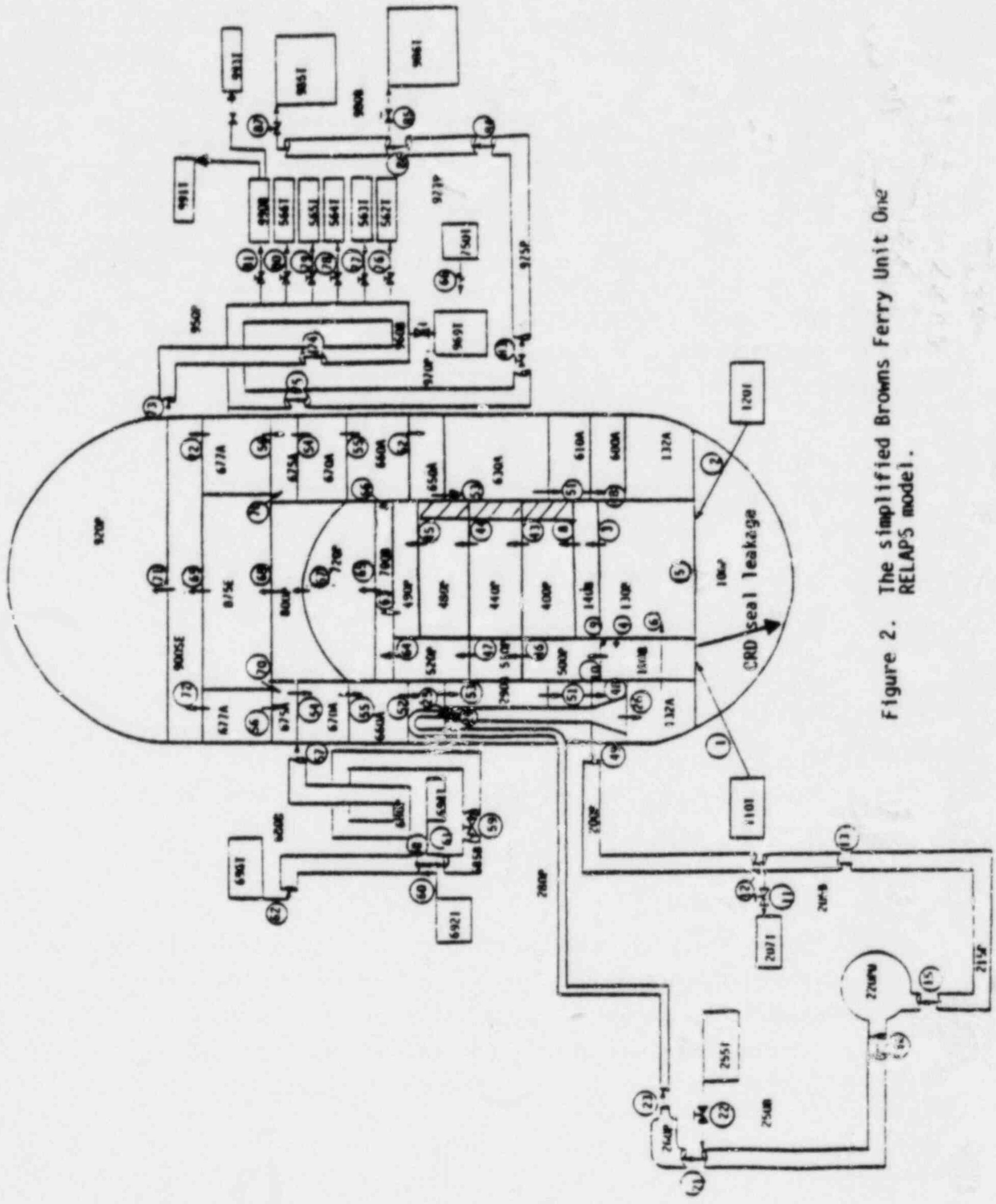

Figure 2. The simplified Browns Ferry Unit One RELAPS model.

TABLE 1. REACTOR CONDITIONS AT TIME OF SCRAM

Condition	Value
Total Reactor Power	3,357.9 MW (t)
Main Steam Flow	13.63×10^6 lbm/hr
Feedwater Flow	13.46×10^6 lbm/hr
Downcomer Level	544.5 in
Steam Dome Pressure	1016.1 psia
Lower Plenum Pressure	1053.4 psia
Lower Plenum Temperature	526.5°F
System Total Inventory	7.295×10^5 lbm
Core Mass Flow Rate	94.3×10^6 lbm/hr

TABLE 2. CHRONOLOGY OF MAJOR EVENTS

Time (sec)	Event
0	Reactor trip on high main steam line radiation.
3	Main steam line isolation valves closed.
30	SDV piping break initiates.
450	HPCI and RCIC initiate on low vessel water level.
780	HPCI and RCIC trip off on high level. The RCIC is not reset.
450-12900	Level maintained between 476 in. and 582 in. by HPCI system.
12900-18660	CBPs initiate and rapidly fill the vessel, flooding the HPCI turbine. Level maintained until CBPs fail on loss of suction.
19300	Reactor vessel pressure begins to increase after reaching a minimum of 316 psia.
19300 +	Vessel pressure continues to rise and eventually actuates the safety relief valves. A boil-off ensues, reducing vessel inventory until the core uncovered.

$$EL = \frac{1}{3}\left[EL\ (n - 2) + EL\ (n - 1) + EL(n) \right]$$

where

EL = downcomer water level

n = current time step

with the trip occurring after a 10 s delay (see Reference 4 Appendix B for a description of EL)

3. The recirculation pumps were modeled to runback to 20 percent speed upon reactor scram and trip off when the downcomer water level reaches 476 in. (low-low level).

4. The total liquid delivered to the reactor building was tracked during the transient. The liquid delivered was calculated as the sum of the break flow and a constant 170 gpm control rod drive cooling flow which bypasses the reactor vessel. The VWIE located in the basement is assumed inoperative when 470548 gal of water have accumulated there.

4. RESULTS

The initiating event of this severe accident sequence is a high main steam line radiation signal, which results in the closure of the main steam line isolation valves (MSIV) and a corresponding reactor scram. Concurrently, the feedwater pumps begin a rapid run-out as shown in Figure 3, but fail upon loss of steam. The recirculation pumps run back to 20% speed in 50 sec. The motor-driven condensate and CBPs remain operational, but are unable to inject fluid into the vessel because of their relatively low shut-off head (415 psia).

MSIV closure causes the reactor vessel pressure to rise rapidly, and the safety/relief valve (SRV) setpoints are reached in 4 s.[a] Safety/relief valve cycling initiates and continues while both the net steaming rate (from the core decay heat transfer) and level remain high. At 450 sec, the first VWIE (HPCI-RCIC) injection initiates when the vessel level falls to the low-low trip (476 inches above vessel zero). Vessel depressurization follows as shown in Figure 4. Injection terminates at 780 s when the level rises to the high water level (582 inches above vessel zero) The vessel pressure begins to increase. The RCIC is automatically shut off after the first injection cycle and is not reset.

Shortly after scram, the core and core bypass flows (Volumes 500, 510, and 520) decrease from rated value (see Figure 5). The bypass flow reverses as the liquid in the upper plenum begins to flow downward through the bypass and then into the core inlet piece and upwards through the core. Such a natural recirculation pattern is generated by core decay heat transfer.

In the natural recirculation mode of core cooling, core and core bypass flows are driven by a gravity head difference. This head is balanced by frictional losses. Introduction of subcooled emergency core cooling (ECC) injection fluid upsets this balance, and causes the bypass

a. The first SRV setpoint is 1120 psia. The remainder are staged upward. All reseat 50 psi lower than their opening setpoints.

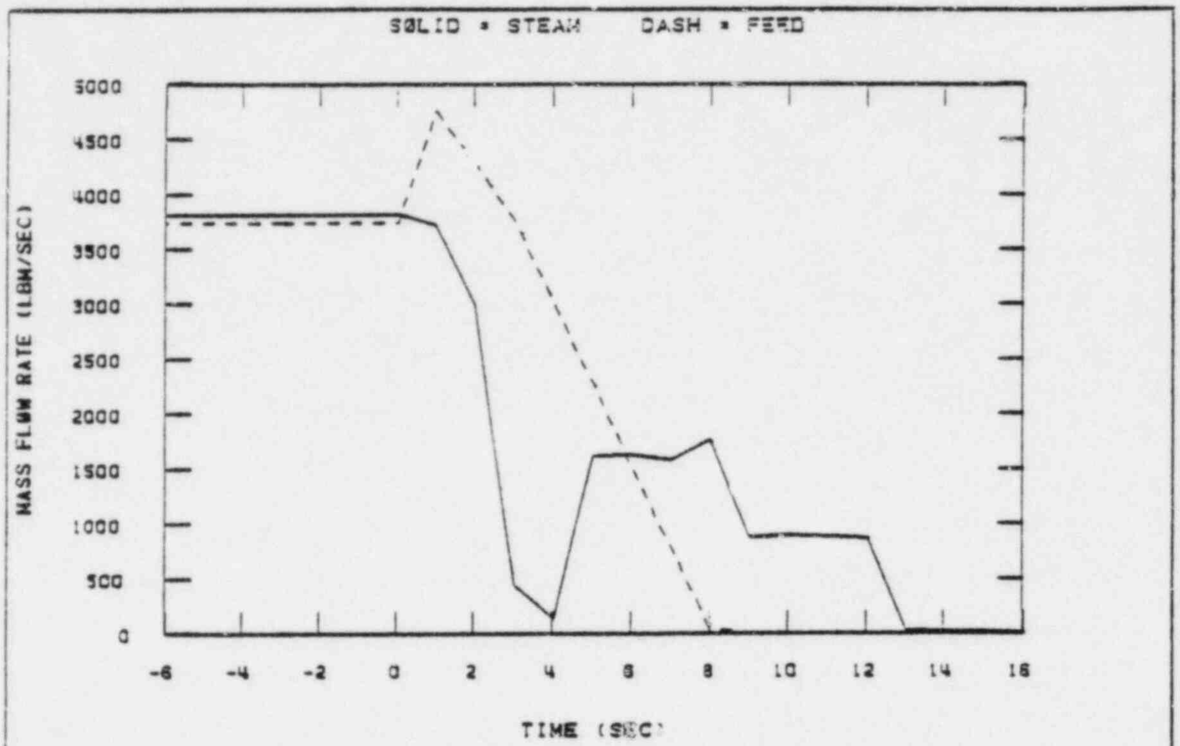

FIGURE 3: NEAR-SCRAM STEAM AND FEED

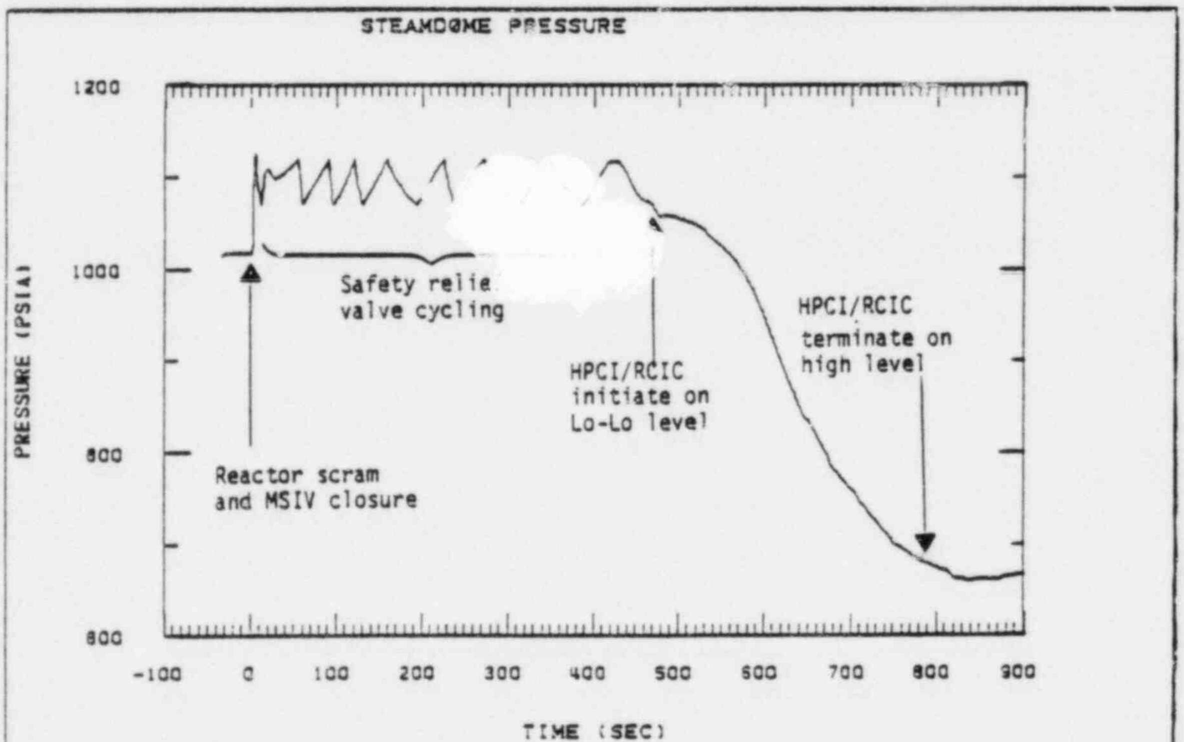

FIGURE 4: SHØRT TERM STEAMDØME PRESSURE

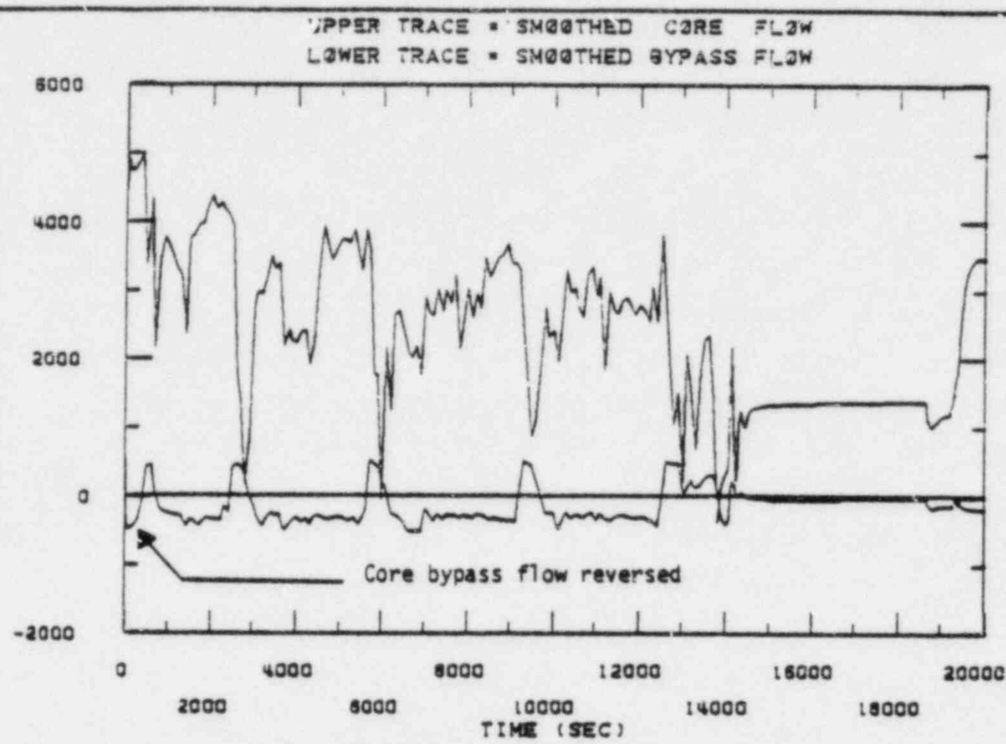

FIGURE 5: CORE AND BYPASS FLOWS

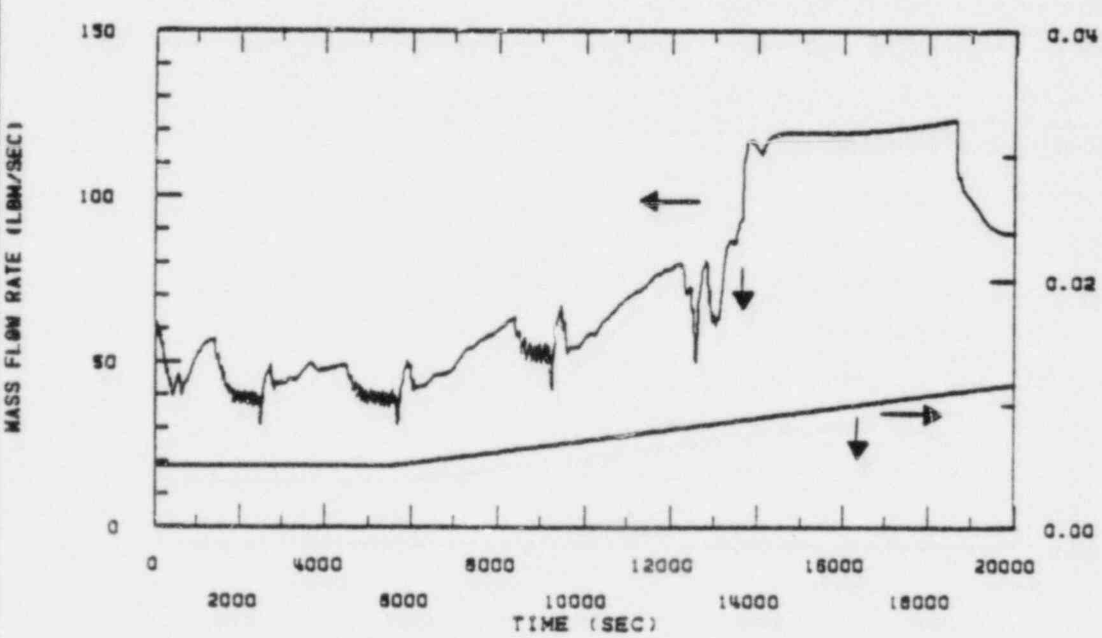

FIGURE 6: BREAK SEAL FLOW AND SIZE

flow to briefly revert to its normal direction. The magnitude of the core flow is reduced by the subcooling as the steaming rate falls, but picks up at the termination of an injection cycle.

The vessel pressure, the direction of the bypass flow, the size of the break area and the static quality of the fluid in the guide tube (Volume 180) all influence the break flow. Thus, the break flow (see Figure 6) decreases as the vessel pressure decreases, and decreases when the bypass flow is reversed since saturated fluid enters the break rather than subcooled fluid. These factors combine to give the break flow behavior shown in Figure 6. An overall increase in flow occurs after 5430 s as the break size increases.

The cycling VWIE systems (see Figure 7) take suction from the atmospheric temperature condensate storage tank. The flow of cool water into the downcomer during injection rapidly condenses steam in the dome and upper downcomer, subcooling and depressurizing the vessel (see Figure 8). In addition, core steaming is subsequently reduced and finally stopped as the fluid in the core shroud becomes subcooled. For the most part, vessel thermodynamic conditions correspond closely to saturation. The effect of localized downcomer subcooling on vessel pressure is such that core voiding tends to increase during the first few seconds of injection. Vessel pressure and temperature rise rapidly upon the termination of an injection cycle.

In total, the VWIE cycles five times (see Figure 7). The fifth and last HPCI cycle depressurizes the vessel to such an extent that the CBPs begin to flow into the vessel at t = 13000 s. A brief, high-flow period follows in which the vessel is nearly filled, as shown in Figure 7. The HPCI steam turbine is flooded and hence inoperative. The CBPs keep the vessel filled and the vessel pressure at their shut-off head until the condensate hotwell is depleted of liquid at 18570 s.

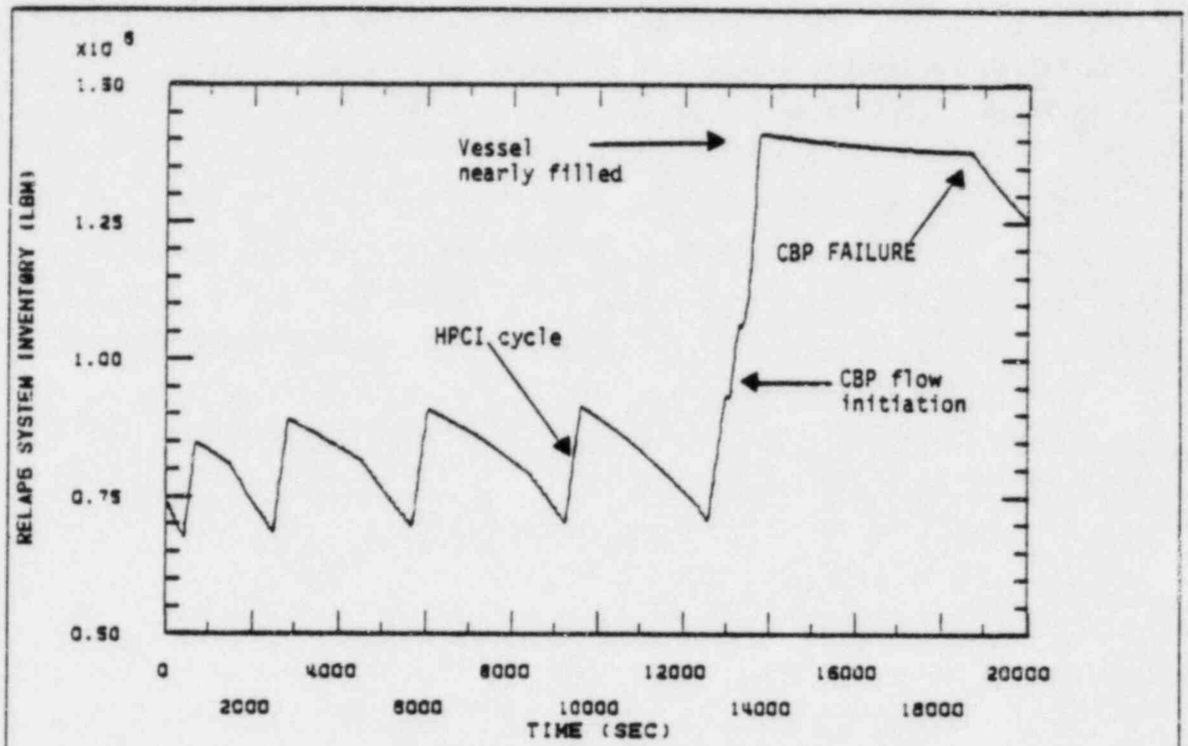

FIGURE 7: SYSTEM INVENTORY

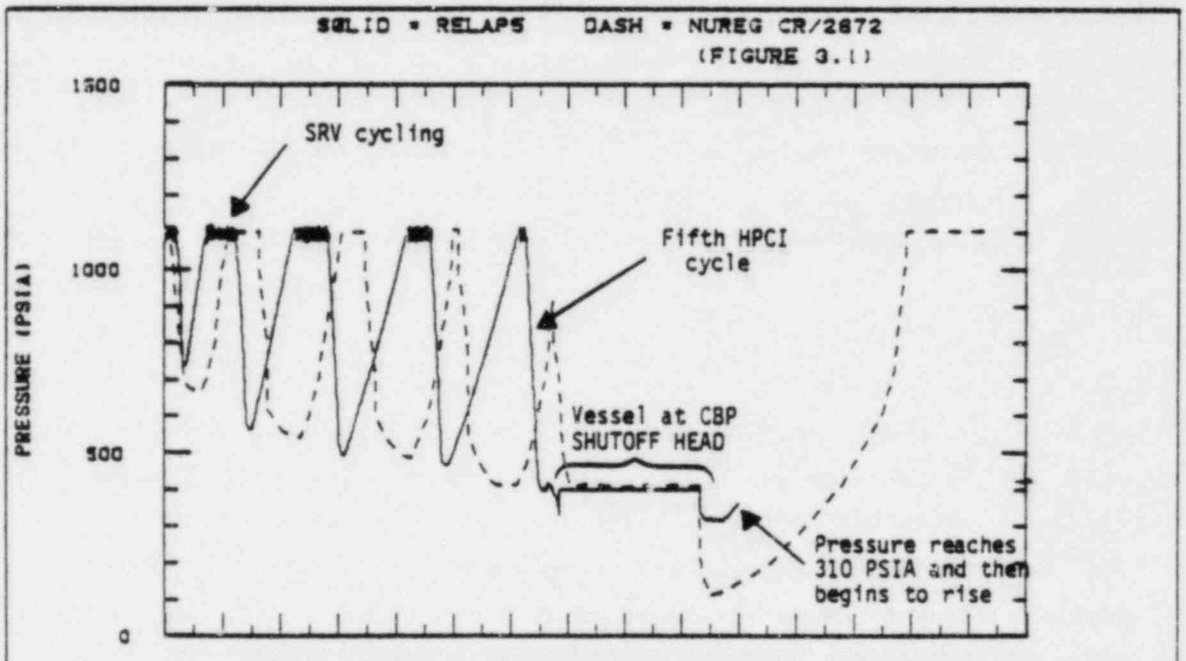

FIGURE 8: STEAM DOME PRESSURE COMPARISON

Thus, by 18570 s only the low pressure ECCs remain to provide water to the vessel. Following loss of the condensate/condensate booster pumps, the vessel pressure decreases to 316 psia, but begins to increase as the core decay heat thermally swells the vessel fluid. The low pressure systems are never able to begin injection as the vessel pressure always remains above their shut-off heads. The lack of vessel injection capability will eventually lead to core uncovery due to the continuing break inventory depletion rate. Table 2 contains a concise chronology of these results.

These RELAP5/MOD1 results are stored under Configuration Control No. F00944.

5. COMPARISON OF THE INEL AND ORNL CALCULATIONS

The RELAP5 model (INEL) used for the SDV break analysis contains a more detailed nodalization scheme than the BWR-LACP model[3]. Thus the thermo-hydraulic state of each volume (see Figure 2) is rigorously calculated in the RELAP5 model. Thermodynamic state distributions are available within a region e.g., volumes 600 through 677 represent the downcomer. As an example, the initial injection of subcooled ECC into the saturated two phase downcomer rapidly depressurizes the vessel without directly affecting the core steaming rate. As injection proceeds, core boiling is terminated as the core inventory becomes subcooled. However, core boiling begins shortly after ECC injection is terminated. Boiling occurs first at the top of the core and proceeds rapidly downward. As a result the vessel repressurizes. The BWR-LACP code treats the vessel regions on an average basis, thus the system response is dynamically slower. Such behavior is apparent if the steam dome pressure calculated by the two codes is compared (see Figure 8). The RELAP5 results show a faster time response in pressure e.g., the RELAP5 analysis predicts a faster repressurization following termination of VWIE injection. The overall vessel energy/mass balances agree quite well, however.

Figure 9 shows the seal leakage mass flow rate plotted against both BWR-LACP results and time. Although generally in good agreement, the RELAP5 results show a flow degradation due to safety/relief valve cycling induced voiding where the break takes suction, thus reducing the flow rate.

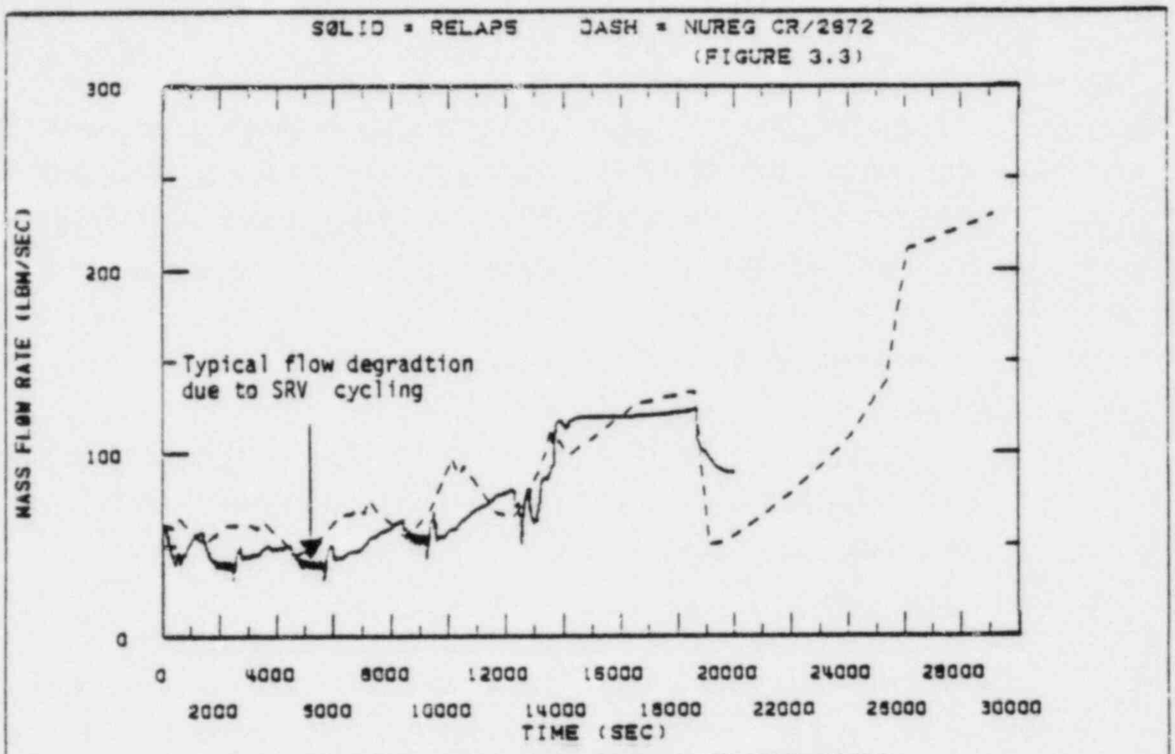

FIGURE 9: SEAL MASS FLØW RATE CØMPARISØN

6. CONCLUSIONS/OBSERVATIONS

Several conclusions and observations emerged in the SDV calculation conducted using the RELAP5 Browns Ferry model:

1. Core uncovery will occur during this transient if the operator does not act. The automatic systems are not sufficient to prevent the core from uncovery.

2. Once the CBPs have depleted the condensate hotwell, only operator action reducing the vessel pressure by opening one or more SRVs such that the low pressure ECCS could inject water into the vessel can prevent core uncovery.

3. The SDV break flow is influenced directly by the core bypass thermodynamic conditions.

4. Generally, RELAP5 and BWR-LACP code calculations show good agreement.

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5. Ransom et al., RELAP5/MOD1 Code Manual, NUREG/CR-1826, November 1980, Vol. 1 and 2.

Appendix H

ACRONYMS AND SYMBOLS

ADS	automatic depressurization system
ANS	American Nuclear Society
ANSI	American National Standards Institute
APRM	average power range monitor
ARM	area radiation monitor
BAF	bottom of active fuel
BCL	Battelle Columbus Laboratories
BFNP	Browns Ferry Nuclear Plant
BFNP#1	Browns Ferry Nuclear Plant Unit One
BWR	boiling water reactor
CBP	condensate booster pump
CILRT	Containment Integrated Leak Rate Test
CP	condensate pump
CRD	control rod drive
CS	core spray
CST	condensate storage tank
DF	decontamination factor
dp	differential pressure
DW	drywell
ECCS	emergency-core-cooling system
EPA	electrical penetration assembly
EOI	Emergency Operating Instruction
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
GE	General Electric Company
HCU	hydraulic control unit
HPCI	high-pressure coolant injection
ID	internal diameter
INEL	Idaho National Engineering Laboratory
INTER	Core-concrete interaction subroutine of the MARCH code
kPa	kilopascal

LACP	loss-of-ac power
LPCI	low-pressure coolant injection mode of the RHR system
LPECCS	low-pressure emergency-core-cooling systems
LOCA	loss-of-coolant accident
LOCA/OC	loss-of-coolant accident outside containment
LOSP	loss-of-offsite power
MARCH	meltdown accident response characteristics
MPa	megapascal
MSIV	main steam isolation valve
Mwd/te	megawatt day per tonne
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
Pa	Pascal
PCV	pressure control valve
PCIS	primary containment and reactor vessel isolation control system
PRA	probabilistic risk assessment
PSP	pressure suppression pool
PV	pressure vessel
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling system
RES	Office of Nuclear Regulatory Research
RHR	residual heat removal system
RHRSW	residual heat removal service water
RPS	reactor protection system
RPV	reactor pressure vessel
RWCU	reactor water cleanup system
SASA	severe accident sequence analysis
SGT	Standby gas treatment
SBLOCA	small-break loss-of-coolant accident
SDIV	scram discharge instrument volume
SDV	scram discharge volume
SI	International System of Units (Systeme International)
SNL	Sandia National Laboratories
SRV	safety relief valve

TAF	top of active fuel
TIP	traveling incore probe
TVA	Tennessee Valley Authority
WW	wetwell
Zr	zirconium

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