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ASSESSMENT OF BWR MITIGATION OF ATWS, VOLUME II (NUREG 0460 ALTERNATE NO. 3)

GENERAL ELECTRIC

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Sect. 1

ASSESSMENT OF BWR MITIGATION OF ATWS, VOLUME II
(NUREG 0460 ALTERNATE NO. 3)

Approved: R L Buchholz fm
G. G. Sherwood, Manager
Safety and Licensing Operation

Approved: S L Mather
S. L. Mather, Acting Manager
Projects Engineering

GENERAL ELECTRIC

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APPENDICES

1. INTRODUCTION

1.1 SUMMARY

The Nuclear Regulatory Commission (NRC) Staff issued their technical report NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," Volumes 1, 2, and 3, in 1978. Volume 3 describes recommendations for mitigation systems, for the various plant categories. In February 1979, the Staff requested General Electric to document the response of the BWR with proposed mitigation systems (February 15, 1979 letter from R. J. Mattson to G. G. Sherwood).

General Electric considers the NRC proposed anticipated transient without scram (ATWS) mitigation systems to be unwarranted in light of the high reliability of the current BWR shutdown system. Nevertheless, General Electric has performed the assessment illustrating the capability of the timed, two-pump standby liquid control system (SLCS) to mitigate the consequences of the hypothetical ATWS event. The preliminary design of this mitigation system was described in NEDE-24222, Volume 1 (May 1979), and is further defined in the conceptual functional control diagrams later in the report.

Volume 1 identified the three most likely limiting ATWS transients and evaluated them on a generic basis for a representative BWR/4-Mark I, BWR/5-Mark II, and BWR/6-Mark III. This study extends the scope to cover all ATWS transients for the three product lines given above as listed in NUREG 0460 (Volume 3). This evaluation has verified the original assumptions in Volume 1 that the selected transients were either bounding or the difference being insignificant. Accordingly this study contains sensitivity studies for the three events evaluated in Volume 1. Additionally, the analytical results are extended to examine a representative BWR/4-Mark II.

1.2 APPROACH USED IN THIS STUDY

The approach used in this study was to complete the evaluation of all the hypothesized ATWS events for the three product lines given above and to provide sensitivity studies of the three transients selected in Volume 1 for the same three product lines. For clarity, certain sections of the Volume 1 are repeated. There are some minor differences in the evaluation of Volume 1 and those in this report due to the use of more realistic parameters (see Section 3.1.1). These differences, which have negligible impact on the results and conclusions, are described at the beginning of each appropriate section of this report (Section 3.1, 3.2, 3.3).

The description of the BWR/4 Turbine Trip event has been modified from the description in the proprietary report in order to show the effects of MSIV closure, which was previously neglected.

The BWR/4-Mark II is covered on a representative basis by utilizing appropriate sensitivity studies on a representative BWR/4-Mark I and BWR/5-Mark II plants. The BWR/4-Mark II is represented by the BWR/4-Mark I for all reactor variables and bulk pool temperatures. Containment loading and local suppression pool temperatures are represented by the BWR/5-Mark II results.

Sensitivity studies are provided for the BWR/4-Mark I, BWR/5-Mark II, and BWR/6-Mark III for the key parameters affecting suppression pool temperature, fuel integrity, water level, and overpressure for the three transients evaluated in Volume 1.

Due to the extremely low probability of the occurrence of an ATWS, nominal parameters and initial conditions have been used in these analyses. This is consistent with the NRC Staff request. Additionally, since NUREG-0460 (Volume 3) suggests an implementation schedule of at least 2 years after rulemaking, this analysis whenever possible utilizes parameters expected to be present at that time, such as all-8x8 fuel. It should be noted that in spite of this approach, many conservatisms remain in the analysis.

The basis of these analyses is that the systems used to mitigate the postulated ATWS events are selected such that ATWS consequences do not result in a threat to public health and safety. Specifically, the mitigation capability meets the following criteria:

1. The reactor coolant pressure boundary shall remain below emergency pressure limits.

2. The containment pressure shall remain below design limits. The suppression pool temperature shall remain below local saturation temperature limits as defined in Section 4.2.3.

3. A coolable core geometry shall be maintained.

4. Radiological releases shall be maintained within 10CFR100 allowable limits.

5. Equipment necessary to mitigate the postulated ATWS event shall be evaluated to provide a high degree of assurance (assurance of function) that it will function in the environment (pressure, temperature, humidity, radiation) predicted to occur as a result of the ATWS event.

The assessment was performed using the following General Electric computer codes: *

*

Bars in right-hand margins indicate General Electric Company Proprietary Information deleted.

1.3 CONCLUSIONS

In response to the requirements of Alternate 3, set forth in NUREG-0460, (Volume 3) mitigation of the consequences of a postulated ATWS event have been assessed for representative BWR/4/5/6 plants. The conclusions drawn from this assessment are:

a. Recirculation Pump Trip (RPT) on high vessel pressure or low water level maintains vessel pressure within emergency limits, and quickly reduces power to well below rated.

b. Alternate Rod Insertion (ARI), utilizing diverse logic and sensors on high vessel pressure or low water level, or manual initiation, results in rod insertion after a delay of approximately 15 seconds. The combination of RPT and ARI results in low suppression pool temperatures, assures core coverage, and maintains core temperatures well within acceptable limits.

c. Two-pump SLCS, initiated on high vessel pressure or low water level, or manually, and after confirmation of an ATWS condition, results in acceptably low suppression pool temperatures, core coverage, and acceptable core temperatures for all initiating ATWS events.

d. The radiological analysis demonstrates that the limits of 10CFR100 are not exceeded for any ATWS event, and would not be exceeded, even if 100% of the fuel cladding failed.

2. DESCRIPTION OF ATWS ANALYSES

This section defines the basis for the scope and content of this submittal. Specifically, the classification of plants chosen for analysis is discussed, the reasons for choosing the particular plant transients analyzed are given, the plant conditions utilized and the assumptions employed are listed, and the equipment and systems required to be operative are specified and discussed.

2.1 CLASSIFICATION OF PLANTS ANALYZED

ATWS event analyses and sensitivity studies are provided for three classes of BWRs, consistent with General Electric product line designations. The three classes contained in this report, the BWR/4 (Mark I), BWR/5 (Mark II) and BWR/6 (Mark III), are described in Table 2.1-1. The BWR/4 (Mark II) product line is shown by sensitivity studies to be representative of the BWR/4 (Mark I) and consequently no additional evaluations have been performed.

The analyses were performed using normalized parameters such that all units within a product line can utilize these results to the maximum extent practical.

2.2 ATWS EVENTS ANALYZED

In the May ATWS submittal[1] to the NRC, results of failure to scram were presented for three initiating transients. These transients, and the rationale for their selection, are:

A. MSIV Closure Event. Previous General Electric studies[2,3] have shown that this transient, coupled with a postulated scram system

[1] Assessment of BWR Mitigation of ATWS (NUREG 0460 Alternate #3), General Electric May, 1979 (NEDE-24222, Volume 1, Class III).

[2] Analysis of Anticipated Transients Without Scram, Licensing Topical Report, March 1971 (NEDO-10349).

[3] Studies of BWR Designs for Mitigation of Anticipated Transients Without Scram General Electric Licensing Topical Report, October 1974 (NEDO-20626).

failure is the most limiting transient from the standpoint of peak vessel pressure, peak heat flux, and peak suppression pool temperature, (although design limits are not exceeded). This conclusion was reached for analyses of the BWR 4, 5 & 6 product lines assuming recirculation pump trip and automatic initiation of the SLCS.

B. Inadvertent opening of a safety/relief valve was included because, depending on the operator action taken following failure of manual scram, this transient can result in high, but acceptable, suppression pool temperatures.

C. To demonstrate acceptable performance during most other types of ATWS events, the turbine trip/load rejection transient with scram failure was presented. Lower pool temperatures result during this ATWS event due to the accessibility of the condenser heat sink and the ability of the feedwater system to make up coolant inventory in the reactor vessel. This event also results in moderately high core power for the first few minutes into the transient, and hence represents additional fuel duty considerations.

This report contains sensitivity studies which expand the previous evaluations for these three transients by analyzing the effect of variation of significant parameters impacting fuel duty, overpressure, and pool temperature about the base values in Volume 1.

All other significant anticipated operational transients have also been analyzed in this report to illustrate BWR behavior during the hypothesized ATWS events. The results of the transient analyses substantiate the conclusions in Volume 1 that the automated SLCS provides adequate mitigation.

Recent studies have shown that postulated failures of the steam bypass system with turbine trip have a frequency of less than one per plant lifetime. General Electric does not believe that the bypass failure event should be included in any required ATWS study. On the other hand, the consequences of the turbine trip without bypass ATWS event have been shown to be essentially the same as the MSIV closure event; thus, information on the turbine trip without bypass ATWS event will also be presented as requested by the NRC Staff.

2.3 PLANT CONDITIONS

Initial operating conditions for the typical plants used to represent each BWR Product Line are listed in Table 2.3-1. They are consistent with NUREG-0460 guidelines and represent nominal operating though not bounding conditions. The listing shows most parameters which are expected to influence the course of the ATWS events. Wherever possible, these parameters are normalized to the rating of the unit so that most effective generic use of this report can be made by units of all sizes within a product line (e.g., initial suppression pool volume is given in full-flow-minutes of rated feedwater).

Only Alternate 2 and Alternate 3 (NUREG 0460, Volume 3) modifications are assumed to be implemented in these analyses. The features of Alternate 3 include all Alternate 2 features plus the implementation of automated SLCS repiped to new injection locations and the simultaneous operation of both SLCS pumps. The automatic initiation of SLCS includes a two minute delay to allow for ARI and other actions to cause rod insertion, thus, avoiding unnecessary boron injection.

The analysis for acceptable ATWS performance assumes the use of quenchers on the safety/relief valve discharge piping on each plant.

2.4 OPERATIVE EQUIPMENT AND SYSTEMS

2.4.1 Systems Utilized

The systems which perform the functions during ATWS events are the same as those given and discussed in Volume I. They are repeated here for clarity. Recirculation Pump Trip and the ATWS logic for the automated standby liquid control system will be designed to meet the criteria in NUREG 0460 (Volume 3, Appendix C, Items A-H). Alternate Rod Insertion will meet IEEE-279, 1971.

a) Recirculation Pump Trip (RPT)

b) Safety/Relief Valves (S/RV)

c) Control Rod Drives

d) Alternate Rod Insertion (ARI) Valves

e) High Pressure Coolant Injection (HPCI); High Pressure Core Spray (HPCS); Reactor Core Isolation Cooling (RCIC)

f) Standby Liquid Control System (SLCS)

g) Suppression Pool and Containment

h) Residual Heat Removal System (RHR)

i) Main Condenser (not mandatory for ATWS mitigation)

j) Main Steam Isolation Valves (MSIV)

k) Feedwater System (runback function)

l) Turbine Pressure Control and Bypass System (not mandatory for ATWS mitigation)

m) Condensate Storage Tank (CST)

n) Reactor Water Cleanup System (isolation function)

o) Standby Gas Treatment System

p) Diesel Generator

q) ATWS Logic

r) Instrumentation (Primary functions)

i) Reactor Power - LPRM/APRM

ii) Control Rod Position Indication

iii) Dome Pressure

iv) Vessel Water Level

v) Suppression Pool Temperature

2.4.2 Equipment Performance Assumed in Analysis

Characteristics of the important pieces of equipment used to mitigate the consequences of failure to scram are listed in Tables 2.4-1 - 2.4-3.

The peak SRV discharge suppression pool bubble pressure during ATWS was compared to the design SRV discharge pressure with the following results:

1. The form of the forcing function for S/RV discharge phenomena is the same for both the ATWS and design cases (i.e. idealized Rayleigh bubbles).

2. The relationship between pressure amplitude and building response is linear at a given frequency.

3. Realistic S/RV discharge loads used for ATWS are bounded by current design requirements.

4. The S/RV frequency range during ATWS is bounded by the S/RV forcing function used for design.

These results are supported by Monticello data for the GE "T-quencher" and Caorso data for the "X-quencher". In the case of non-GE quenchers, a preliminary assessment indicates that a similar statement can be made for the non-GE quenchers. Thus, no unique ATWS load evaluation is necessary.

NSSS integrity is not compromised by the peak reactor vessel pressure calculated to occur during the most severe postulated ATWS event. This is demonstrated by showing that the ATWS pressures are less than that pressure which results in "Service Level C" stresses.

Operability of the systems and components necessary for ATWS mitigation will be also demonstrated by showing peak ATWS pressures are bounded by existing operability test data or are consistent with the pressures used in the definition of Plant Conditions and Load Combinations for "Service Level C" Conditions. General Electric's implementation of Regulatory Guide 1.48 requires that stresses in pumps and valves remain below yield point for both upset and emergency conditions. Operability will not be impaired by these low stresses since no permanent deformation will occur. Even if performance could be slightly degraded during the ATWS pressure peaking (a few seconds duration), no impact on ATWS mitigation would occur since such equipment is not required until the pressure is back within normal bounds. Pumps and valves will then function in a normal manner.

Based upon the foregoing, it is concluded that ATWS pressure peaks will not cause any loss of pressure boundary integrity or deformations which would prevent an ATWS mitigation system from performing its intended function. This, coupled with the fact that none of the mitigation systems which could be affected by system pressure are required until well after the pressure peaking has subsided, leads to the conclusion that ATWS pressure is not a concern with regard to mitigation system operability.

Additional information is contained in Appendix A.3.

2.4.3 ATWS Functional Control Diagrams

Conceptual functional control diagrams for potential automated standby liquid control injection and other BWR ATWS mitigation and prevention features have been evolving since the early topical reports introduced high pressure and low level recirculation pump trip (NEDO-10349). Figures 2.4.3-1 to 2.4.3-3 show one possible functional logic configuration for BWR/4,5 and 6 plants if implementation of Alternate 3 is required. Some refinements may be added as the design becomes finalized.

Table 2.3-1

TYPICAL INITIAL OPERATING CONDITIONS

Table 2.4-1

EQUIPMENT PERFORMANCE CHARACTERISTICS

(see page 2-15 for notes)

Table 2.4-1 (Continued)

EQUIPMENT PERFORMANCE CHARACTERISTICS

Table 2.4-2

BWR/5

EQUIPMENT PERFORMANCE CHARACTERISTICS

Table 2.4-2 (Continued)

BWR/5

EQUIPMENT PERFORMANCE CHARACTERISTICS

Table 2.4-3

BWR/6

EQUIPMENT PERFORMANCE CHARACTERISTICS

Table 2.4-3 (Continued)

BWR/6

EQUIPMENT PERFORMANCE CHARACTERISTICS

Figure 2.4.3-1. BWR/4 Conceptual Functional Control Diagram

Figure 2.4.3-2. BWR/5 Conceptual Functional Control Diagram

Figure 2.4.3-3. BWR/6 Conceptual Functional Control Diagram

3. RESULTS OF ATWS EVENT EVALUATIONS

3.1 RESULTS OF ATWS EVENTS - BWR/4 (MARK I)

3.1.1 Closure of Main Steam Isolation Valves (MSIV)

3.1.1.1 Overview of Response Without Scram

The behavior of the plant is separable into an early or short term transient involving a sharp pressure rise and power peak, and a longer term portion that requires evaluation of coolant and containment conditions as the reactor is ultimately brought to shutdown.

The effectiveness of RPT presented in NEDO 10349, NEDO 20626, and Volume 1 are completely reconfirmed by this analysis. It assists the relief valves in limiting the pressure disturbance acceptably and allows the estabishment of a relatively low power generation rate for the long term portion of the transient. Figure 3.1.1-1 illustrates the first period. Since Volume 1, several changes have been made to the base case calculations (shown in Tables 2-1 and 3-1). They include:

a. The increased Doppler reactivity coefficient is now more typical of all plants. (The previous value was an unrealistically low, bounding assumption.) All plants were surveyed and nearly identical coefficients are expected for them. Variations of this term are included in the sensitivity Section 3.4.

b. S/RV reclosure pressure is now 110 psi below the opening pressure setpoint which is more typical of the actual performance.

c. Feedwater flow characteristics due to automatic limiting action or loss following isolation are assumed to result in shutoff 40 seconds after isolation begins.

The ultimate resolution to the lack of scram situation must involve insertion of negative reactivity into the reactor, thereby bringing the reactor to a fully shutdown condition. ARI is provided as an effective way to mitigate common-cause failures in the logic of the scram system. The effectiveness of ARI to mitigate events is clearly in Figure 3.1.1-2 which shows its effect on the MSIV closure transient. In the case of its ineffectiveness, the automated SLCS provides further protection and shutdown capability. Coolant inventory is adequately maintained by HPCI and RCIC available in each BWR/4 to replace the coolant loss as steam flow leaves the primary system through the relief valves. Simply adding more water is not a totally satisfactory answer because it also has the effect of raising the power generation rate and the amount of inventory leaving the system as steam thus increasing suppression pool temperature. The steam reaching the suppression pool continues to heat it and pressurize the containment until the power generation/steam flow can be reduced to the RHR capacity and/or finally terminated. The RHR (pool cooling mode) ultimately cools the pool and eventually the reactor also (shutdown cooling mode) if the MSIV's cannot be reopened establishing flow to the main condenser (the preferred method of cooldown).

3.1.1.2 Sequence of Events for MSIV Closure

The MSIV closure transient provides some of the most severe conditions following a postulated failure to scram. Listed in Table 3.1.1-1 in the sequence of occurrence are significant points of the transient with representative times when each highlight occurs.

The sequence of events begins with the closure of the MSIVs in 4 seconds. With closure of the MSIVs, the pressure immediately begins to rise, resulting in a reduction in void fraction and rapid increase in power. This sequence of events is shown in Table 3.1.1-1. This power reaches a maximum of 527% of the initial value at 4.0 seconds into the event and rapidly decreases thereafter. At 4 seconds, the setpoint pressure of the relief valves is reached and they open to arrest the pressure rise. At about the same time that the relief valves are opening, it is expected some of the fuel will experience transition boiling. Shortly after 4 seconds, the vessel dome pressure reaches 1150 psig, the RPT trip; both recirculation pumps trip. A delay of 530 milliseconds exists from the time the pressure reaches 1150 psig until the time that RPT occurs. This delay time (500 milliseconds delay in the sensor and 30 milliseconds in the logic and trip) is consistent with industry experience. At the same time that the RPT occurs, the logic chain is activated which would initiate ARI.

Pressure continues to rise for a short period of time until, at approximately 9 seconds into the event, it reaches its peak and begins to decrease. The maximum pressure at the vessel bottom is 1296 psig at 8.99 seconds. The relief valves begin to close shortly after 20 seconds; pressure is then stabilized at the relief valve setpoint. This portion of the transient is shown in Figure 3.1.1-1. Peak values for key variables for all BWR/4 events are given in Table 3.1-1.

The same pressure signal (1150 psig) that initiated RPT will cause the opening of valves on the scram air header (ARI) which allows the air pressure in the header to bleed down. In the event that scram has not already occurred from any of the several available signals, this reduced pressure allows the scram discharge valves to open and the control rods to insert. Tests have shown that the pressure in the header will have been reduced sufficiently in 15 seconds to allow the control rods to insert. All rods will be fully in the core after 5 additional seconds. ARI completely mitigates the ATWS situation and 25 seconds after the event began, it is essentially over. Following ARI, normal shutdown procedures are utilized to bring the plant to cold shutdown.

If the ARI is not effective, the BWR/4 is still able to mitigate the event. With an assumed ARI failure and zero feedwater flow, at 43 seconds, the level of the bulkwater in the vessel will decrease to Level 2, the level at which HPCI and RCIC are initiated. Twenty seconds later, water from these systems will begin to enter the reactor vessel.

Following confirmation from the flux monitoring system and the rod position indicating system that scram has not taken place, the SLCS will activate. This system will be started 2 minutes after the ATWS signal; boron will reach the core after an additional 1 minute of transport time in the lines and the vessel. Therefore, nuclear shutdown begins at 3 minutes into the event using the SLCS. Additionally on confirmation that scram has not taken place, this feedwater flow is ramped to zero flow in 15 seconds. With an 86 GPM volumetric flow rate of sodium pentaborate, the reactor will be brought to hot shutdown in approximately 18 minutes from the beginning of the event. This can be seen in the lower left hand graph of Figure 3.1.1-3. The behavior of several other parameters is also depicted in Figure 3.1.1-3.

Bulkwater level within the vessel continues to decrease until approximately 4 minutes at which time HPCI and RCIC supply more water than is required to make up for steam flow out of the vessel. At this time it reaches its lowest level and begins to rise. It is important to note that adequate core cooling is maintained at all times. As the level is increasing, core flow is increased, thereby reducing the average void fraction. The various contributors to reactivity insertion and power production (boron, voids, etc.) must always be in balance with the power production. Water level is completely restored by HPCI and RCIC which cycle on at low level (L2) and off at high level (L8) to maintain adequate level in the vessel.

Following hot shutdown, the decay power will continue to generate a small amount of steam which will continue to cycle the relief valves. At 28 minutes the suppression pool temperature will reach its maximum value of 186°F. The maximum containment pressure is 10.4 PSIG. Figure 3.1.1-4 is a detailed plot of pool temperature and containment pressure.

Thus it can be seen that an MSIV closure event combined with a failure to scram is adequately mitigated for a representative BWR 4/Mark I.

3.1.2 Turbine Trip

3.1.2.1 Overview of Response Without Scram

The overview given for the MSIV closure event, section 4.1.1, is generally applicable to the Turbine Trip event. The key difference is that the main condenser is available for steam discharge during a portion of this event. During the time that steamflow is within the bypass capacity, the main condenser will be used to remove the steam from the vessel. This base case event has also been updated as given in Section 3.1.1.

3.1.2.2 Sequence of Events for Turbine Trip

The Turbine Trip event begins with the rapid closure of the turbine stop valves and the resultant opening of the turbine bypass valves. After the stop valves close in 0.1 seconds, the pressure immediately begins to rise which results in a reduction in void fraction and rapid increase in power. The sequence of events is shown in Table 3.1.2-1. This power reaches a maximum of 392% of the initial value at 0.9 seconds into the event and rapidly decreases again. At approximately 1.5 seconds, the setpoint pressure of the relief valves is reached and they begin to arrest the pressure rise. Shortly after 2 seconds, it is expected that some of the fuel will experience transition boiling, however, coolable geometry is maintained. At about the same time, the vessel dome pressure reaches 1150 psig, the maximum recirculation pump trip point; consequently, both recirculation pumps trip.

At the same time that RPT occurs the logic chain is activated to start ARI.

Pressure will continue to rise until 3 seconds when it peaks and begins to decrease. The maximum pressure occurs at the vessel bottom and is 1193 psig at 2.50 seconds. Although the feedwater pumps remain available for the turbine trip case, it is necessary to reduce the amount of power produced. Therefore, the feedwater flow will be limited to a minimum flow value, which for this design has been chosen to be zero. This minimizes power generation and resultant steam. Some of the generated steam flows through the bypass to the main condenser. The remainder of the steam is discharged through the relief valves into the suppression pool. The relief valves begin to close very early in this transient (about 9 seconds) and then open and close as needed to reduce vessel pressure. The first portion of this transient is shown in Figure 3.1.2-1. Peak values of other key variables in the system are given in Table 3.1.2-2.

The same pressure signal (1150 psig) that initiated RPT will cause the opening of valves on the scram air header (ARI) which allows the air pressure in the header to bleed down. In the event that scram has not already occurred from any of the several available signals, this reduced pressure allows the scram discharge valves to open and the control rods to insert. Tests have shown that the pressure in the header will have been reduced sufficiently in 15 seconds to allow the control rods to insert. All rods will be fully in the core after 5 additional seconds. ARI completely mitigates the ATWS situation and 25 seconds after the event began, it is over. For this event the feedwater system will continue to function and provide water to the reactor. The effectiveness of ARI to mitigate the Turbine Trip event is clearly shown in Figure 3.1.2-2.

If for some reason the ARI is also not effective, the BWR/4 is still able to mitigate the event. With an assumed ARI failure and feedwater flow now having reached zero, at 58 seconds the level of the bulkwater in the vessel will decrease to Level 2, the level at which HPCI and RCIC are initiated. At 20 seconds later water from these systems will begin to enter the reactor vessel.

Following confirmation from the flux monitoring system and the rod position indicating system that scram has not taken place, the SLCS will activate. This system will be started 2 minutes after the ATWS signal; boron will reach the core after an additional 1 minute of transient time in the lines and the vessel. Therefore, nuclear shutdown begins at 3 minutes into the event using the SLCS. With an 86 GPM volumetric flow rate of sodium pentaborate, the reactor will be brought to hot shutdown in approximately 18 minutes from the beginning of the event. This can be seen in the lower left hand graph of Figure 3.1.2-3.

Beginning at about 4 minutes, transient oscillations of neutron flux, pressure, core flow, and steam flow are calculated. This is shown in Figure 3.1.2-3. These result from interactions between the thermal, hydraulic and nuclear characteristics in the core and circulation system, and turbine pressure controls. The oscillations terminate shortly after the MSIV closure at low reactor water Level 1, which decouples the reactor from the turbine pressure control. As shown in Figure 3.1.2-4, if the MSIV closure setpoint remained at the current Level 2 instead of having been changed to Level 1, as part of the Alternate 3 plant modifications, there would be no predicted oscillations and the transient response would be similar to the MSIV closure event.

The calculated oscillations in neutron flux with the MSIV closure at Level 1 last for a short period of time, with an average power level of about 25% and peak values near 80% of rated power. Fuel thermal power swings are less than 10%. No fuel damage is expected as a result of these oscillations, and coolable geometry is maintained. The oscillations stop after a few minutes, and the reactor then proceeds to complete shutdown.

At 5 minutes, MSIV closure occurs because of low vessel water level (Level 1) and causes an increase in the reactor pressure and a momentary power peak. Bulkwater level within the vessel continues to decrease until approximately 6 minutes at which time HPCI and RCIC begin supplying more water than is required to make up for steamflow out of the vessel and the level begins to rise. As the level increases, core flow is increased, thereby reducing the average void fraction. The various contributors to reactivity insertion and power production (boron, voids, etc.) must always be in balance with the power production. Water level is completely restored by HPCI and RCIC at approximately 17 minutes. At this time HPCI and RCIC will be controlled automatically to maintain adequate level in the vessel.

Following hot shutdown, the decay power continues to generate a small amount of steam which flows into the suppression pool. Since the MSIV closure does not occur until 5 minutes into this event, some steam goes to the main condenser and the temperature rise in the suppression pool is less than the MSIV closure initiating event. The maximum suppression pool temperature calculated in this case is 162°F. The maximum containment pressure is 6.1 psig. Thus it can be seen that a Turbine Trip with bypass event combined with a failure to scram is adequately mitigated for a representative BWR 4/Mark I.

3.1.3 Inadvertent Open Relief Valve (IORV)

3.1.3.1 Overview of Response Without Scram

This event has no rapid excursions.as the previous two events, but is merely a long term depressurization. RPT does not occur until late in the event after hot shutdown is achieved.

Except for steam flow through the open relief valve and the use of the liquid boron solution for shutdown, the nuclear steam supply system is in a normal operating state. The suppression pool is the only system exposed to off-normal conditions. This base case event has also been updated as given in Section 3.1.1. The sequence of events follow.

3.1.3.2 Sequence of Events for Inadvertent Open Relief Valve

This event begins when one of the primary relief valves on the main steamlines inadvertently opens without influence from any other portion of the system. All pressure levels in the reactor coolant pressure boundary are at a nominal value prior to the event. This sequence of events is shown in Table 3.1.3-1.

At the time that the relief valve opens, there is a momentary depressurization (a few seconds) until the turbine pressure control senses it and closes slightly to control the pressure. After approximately two minutes, the suppression pool temperature, which was initially at 90°F, has risen to the alarm point of 95°F. The operator will turn on the RHR system in the pool cooling mode to maintain pool temperature. The temperature will continue to rise and at 7.5 minutes will reach 110°F at which point the operator is required to manually scram the reactor. At this point manual scram and ARI are activated.

If neither normal manual scram nor the ARI are effective, the BWR/4 is still able to mitigate the event. The ATWS logic would have determined that the control rods are not inserted and at 9 minutes into the event, the SLCS will be activated. At 9.5 minutes into the event the SLCS starts and at 10 minutes the control liquid reaches the core and shutdown begins. For this case with the recirculation pumps operating the delay time inside of the vessel is small and 0.5 minutes of boron transport time is sufficient. Within 24 minutes, the power has been reduced to the point that the amount of steam generated is less than the relief valve capacity and the pressure now begins to decrease more rapidly. The turbine control valves have closed completely due to the pressure decrease. These events are depicted in Figure 3.1.3-1. By 28 minutes, the pressure will have dropped to the low steam line pressure isolation point of 800 psig and the MSIV's will close. For plants with turbine-driven feedwater pumps, the feedwater is assumed to be lost in 20 seconds. This causes the water level in the vessel to decrease and at 29 minutes the low level point (L2) is reached where the recirculation pumps are automatically tripped and HPCI and RCIC are activated. These systems will continue to cycle on at low level (L2) and off at high level (L8) to maintain water inventory in the vessel. Depressurization of the vessel will continue with the relief valve discharging into the suppression pool; the maximum pool temperature of 183°F will occur at 95 minutes. The suppression pool temperature trace is shown in Figure 3.1.3-2. The values given here are bounding values. Each specific plant size has some features which may reduce the severity of this event.

In cases where ARI is activated (8 minutes), the maximum suppression pool temperature is 163°F.

Thus it can be seen that the inadvertent opening of a relief valve event combined with a failure to scram is adequately mitigated for a representative BWR 4/ Mark I.

3.1.4 Sensitivity Study Results - BWR/4 Base Cases

A wide variety of parameters were studied to examine the sensitivity and potential impact of plant differences and/or uncertainties on the results of the three BWR/4 base cases. While the overall objective of these sensitivity studies is to provide guidance for assessing the adequacy of plants having certain parameters different from the generic analyses, caution must be exercised when combining the results of several parameter variations, due to the non-linearities involved (see Section 3.3.4.4).

The results are documented in the following subsections:

3.1.4.1 MSIV-ATWS Sensitivity Studies

3.1.4.1.1 Variation of Boron Delay
3.1.4.1.2 Variation of Boron Capacity/Mixing
3.1.4.1.3 Variation of HPCI/RCIC Capacity
3.1.4.1.4 Variation of RHR Capacity
3.1.4.1.5 Variation of RHR Delay

3.1.4.1.6 Variation of Pool & Service Water Temperature
3.1.4.1.7 Variation of RHR Capacity and Service Water
3.1.4.1.8 Variation of Pool Size
3.1.4.1.9 Variation of S/RV Capcity
3.1.4.1.10 Variation of RPT Delay
3.1.4.1.11 Variation of RPT Inertia
3.1.4.1.12 Effect of Partial Rod Insertion
3.1.4.1.13 Variation of Void Coefficient
3.1.4.1.14 Variation of Doppler Coefficient

Table 3.1.4.1-1 summarizes the results for this event.

3.1.4.2 Turbine Trip Sensitivity Studies

3.1.4.2.1 Variation of Boron Delay
3.1.4.2.2 Variation of Boron Capacity/Mixing
3.1.4.2.3 Variation of HPCI/RCIC Capacity
3.1.4.2.4 Variation of RHR Delay
3.1.4.2.5 Variation of Void Coefficient
3.1.4.2.6 Variation of Doppler Coefficient
3.1.4.2.7 Limit Cycle Sensitivity Study

Table 3.1.4.2-1 summarizes the results for this event.

3.1.4.3 IORV Sensitivity Studies

3.1.4.3.1 Variation of S/RV Capacity
3.1.4.3.2 Variation of Boron Delay
3.1.4.3.3 Variation of Boron Capacity/Mixing
3.1.4.3.4 Variation of RHR Capacity
3.1.4.3.5 Variation of RHR Delay
3.1.4.3.6 Variation of Pool Size
3.1.4.3.7 Variation of Pool and Service Water Temperature
3.1.4.3.8 Variation of RHR Capacity and Service Water Temperature

Table 3.1.4.3-1 summarizes the results of this event.

3.1.4.1 MSIV Sensitivity Studies

3.1.4.1.1 Variation of Boron Delay

The SLCS timer delay was varied between 30 seconds (75% below the nominal timer setting of 120 seconds) and 240 seconds (100% above the nominal time) resulting in peak pool temperature 16°F less and 12°F greater respectively compared to the base case. Containment pressures decreased and increased accordingly. Figure 3.1.4.1.1 graphically shows this parameter variation.* Minimum level was increased by 2.4 feet at 30 seconds delay and was decreased 0.9 feet when the timer was extended to 240 seconds.

3.1.4.1.2 Variation of Boron Capacity/Mixing Efficiency

The effective rate of boron injection into the core is the product of the boron pumping capacity and mixing efficiency. This effective rate was varied by ±50% resulting in peak pool temperatures 16°F below and 49°F above the base case respectively. The base case represents an 86 gpm SLCS pumping rate in a 251 inch vessel with 95% assumed mixing efficiency and with vessel inventory proportional to the 218 size value given in Table 2.3-1. The -50% variation point equivalently represents 43 gpm at 95% efficiency or 86 gpm at 48% efficiency. Differences for plant size are covered by comparing the boron rate and the vessel inventory of the plant (e.g., the 86 gpm on a 251 size plant is equivalent to 66 gpm on a 218 size plant). Figure 3.1.4.1.2 graphically shows this variation.

3.1.4.1.3 Variation of HPCI/RCIC Capacity

The rated flow of the HPCI/RCIC system was varied by 20%. Figure 3.1.4.1.3-1 graphically shows variation. For the case of increased flow, pool temperature and containment pressure increased by 12°F and 2.8 psi, respectively. The increase in temperature is due to the higher power level maintained by the increased core flow. Minimum water level increased more than a foot. Decreasing

*Note that points representing calculated results of sensitivity studies are connected by lines to add clarity to general trends. This does not imply detailed knowledge of the variation between points.

HPCI/RCIC flow lowered peak temperature and pressure by 8°F and 1.6 psi. Minimum level was slightly reduced.

3.1.4.1.4 Variation of RHR Capability

To determine the effect of varying RHR heat exchanger capabilities, the base capacity of 1.85% NBR at 100°F ΔT)429 BTU/sec-°F for the 218 size plant used as the base case) was altered by ±50%. Increasing the capacity by 50% yielded a 2°F temperature reduction and lowered the peak containment pressure by 0.5 psi. For the opposite case of a 50% decrease in RHR capacity the results were a 13°F increase in temperature and a 3 psi pressure rise. Sensitivity of pool temperature is shown graphically in Figure 3.1.4.1.4.

3.1.4.1.5 Variation of RHR Delay

The effect of varying RHR start time was found to be small for the BWR/4 MSIV case. Increasing the start time from 11 (base) to 16 minutes increased peak pool temperature by only 2°F. A decrease of 2 minutes resulted in a 1"F reduction in pool temperature. This very weak sensitivity of the pool temperature to RHR startup delay is shown graphically in Figure 3.1.4.1.5.

3.1.4.1.6 Variation of Pool and Service Water Temperature

The pool and service water temperature were assumed to vary together (with the pool assumed to be 5°F above the service water). This variation was found to significantly affect peak pool temperature and containment pressure. Increasing these temperatures by 20°F (to the operating technical specification) produced a rise in pool temperature of 19°F and an increase of about 5 psig in peak pressure. Reducing the temperatures by 20°F yielded decreases of 18°F and about 4 psi respectively. Figure 3.1.4.1-7 graphically shows the pool temperature variation plotted directly vs pool and service water temperature change.

3.1.4.1.7 Variation of RHR Capacity and Pool and Service Water Temperature

Varying both parameters simultaneously was done to examine different RHR designs due to different plant site water temperatures. It showed that pool and service water temperature was the dominant variable. Simultaneous increases of +50% in RHR capacity and +20°F in pool and service water temperature (and a similar set of decreases) produced temperature changes of +17 and -4°F. Peak pressures varied accordingly by +4 and -1 psi. These variations are shown in Figures 3.1.4.1.4 and 3.1.4.1.6.

3.1.4.1.8 Variations of Pool Size

The suppression pool mass was varied by ±20% to simulate different sized plants. The larger pool mass provides a bigger heat sink, thus reducing the peak pool temperature by 13°F and peak pressure by about 3 psi. For the lower pool mass, pool temperature increases 20°F and peak pressure by 5 psi. Figure 3.1.4.1-8 graphically shows the result. BWR 4 Mark I and Mark II plants can utilize this sensitivity study to estimate the effect of different pool sizes on bulk pool temperature.

3.1.4.1.9 Variations of S/RV Capacity

The S/RV capacity was varied by ±20%. The larger and smaller valve capacity resulted in a maximum pressure (vessel bottom) of 66 psi less than and 37 psi higher than the base case, respectively. Figure 3.1.4.1.9-1 shows these results graphically. Pool temperature variation was very small (<2°F).

3.1.4.1.10 Variations of RPT Delay

The base value of RPT delay (0.53 secs) was increased by 0.5 second and 1.0 second. The maximum neutron flux and maximum average fuel heat flux remained unchanged. The minimum pressure (vessel bottom) increased by 11 and 24 psi as shown by Table 3.1.4.1-1 and Figure 3.1.4.1.9-1.

3.1.4.1.11 Variations of RPT Inertia

The recirculation pumping system inertia was increased by 50% and reduced by 20%. In both cases no noticable changes were observed in the values of maximum neutron flux and maximum average fuel heat flux. Maximum vessel pressure increased by 9 psi and decreased by 2 psi, respectively, as indicated by Table 3.1.4.1-1 and Figure 3.1.4.1.9.

3.1.4.1.12 Effect of Partial Rod Insertion

All ATWS analyses are done for the case of no rod motion. In reality, it is very likely that some rod insertion would occur. This would greatly reduce the maximum/peak values shown in the bulk of this report. Analyses for the BWR/6 product line are shown in Section 3.3.4.1.2 and similar results can be expected for the BWR/4 product line.

3.1.4.1.13 Variation of Void Coefficient

The effect of void coefficient on peak transient parameters (neutron flux, average surface heat flux, vessel pressure and suppression pool temperature) was studied for the MISV closure with bypass transient. Void coefficient was varied from -6 to -14¢/% rated voids (nominal = -11¢/%). In all cases the recirculation system was tripped on high vessel pressure. The change in total effective worth of injected boron with void fraction was accounted for. Figures 3.1.4.1.13-1 shows the flux and pressure peaks for the MSIV transient, as a function of void coefficient for several values of Doppler coefficients. Figures 3.1.4.1.13-2 shows peak suppression pool temperature as a function of void coefficient.

3.1.4.1.14 Variation of Doppler Coefficient

The effect of Doppler coefficient on transient peak neutron flux average surface heat flux and vessel pressure during an MSIV closure was studied for the range -0.20 to -0.32¢/°F (nominal = -0.283¢/°F). Figure 3.1.4.1.14-1 shows the peaks plotted as a function of Doppler coefficient. Peak pool suppression temperature is plotted against Doppler coefficient in Figure 3.1.4.1.13-2.

3.1.4.2 Turbine Trip Sensitivity Studies

The Turbine Trip event and associated sensitivity characteristics are similar to the MSIV Closure event, except that the Turbine Trip event has greater margin with respect to suppression pool temperature because more steam goes to the main condenser. Because of this, the Turbine Trip sensitivity studies on suppression pool temperature have less detail than those presented for the MSIV Closure event. Figure 3.1.4.2 shows a special case that is similar to the base turbine trip case with ARI failure but also neglects MSIV closure, thus maintaining continued access to the condenser. For this case, the amount of steam transmitted to the suppression pool is significantly less since the relief valves close at about 4 minutes without subsequent reopening. The peak suppression pool bulk temperature is only 102°F, which results in a maximum containment pressure of 0.6 psig. This special case differs from the base case in another respect; the calculated oscillations in neutron flux and several other parameters are aggravated by neglecting the MSIV closure. These predicted oscillations do not affect the ability of the BWR to successfully mitigate an ATWS. This special case has been used as a basis for much of this Turbine Trip event sensitivity study instead of the base case, because the Turbine Trip base case is so similar to the MSIV Closure base case.

3.1.4.2.1 Variation of Boron Delay

The SLCS timer delay was varied between 30 seconds and 150 seconds (nominal) delay = 120 sec). As noted previously, the effect on pool temperature or containment pressure is negligible. Water level does vary by 2.6 feet and +0.6 feet from the special case level.

3.1.4.2.2 Variation of Boron/Capacity Mixing Efficiency

This case is covered in Section 3.1.4.2.7.3 of this report for this event.

3.1.4.2.3 Variation of HPCI/RCIC Capacity

The HPCI/RCIC flow was varied from 90% to 125% of its nominal value. Minimum water level varies from about 2 feet above special case minimum for the increased HPCI/RCIC case to about 1 foot below for the reduced flow case. The minimum water level for the 125% flow case would not reach the Level 1 setpoint for MSIV closure. (ATWS analyses assume that MSIV closure on low water level has been moved from Level 2 to Level 1). If the MSIV closure occurs, the steam must then go to the suppression pool via the relief valves, and the higher temperature (by 60°F) of Table 3.1.4.2-1 will occur. For the 90% HPCI flow case, the results are shown for the MSIV closure condition only. These results are shown in Figure 3.1.4.2.3-1.

3.1.4.2.4 Variation of RHR Delay

For the base case, suppression pool temperature is bounded by the MSIV closure event. Because the flow of steam to the pool is complete for the special case before the RHR is expected to be started, there is no variation of peak containment temperature or pressure because of changes in the RHR start time.

3.1.4.2.5 Variation of Void Coefficient

The effect of void coefficient variation on the Turbine Trip with bypass transient was studied similar to the MSIV closure reported in Section 3.1.4.1.13. Figures 3.1.4.2.5-1 and 3.1.4.2.5-2 show the results. Those cases in which the peak pressure was more than 5 psi below the nominal did not reach the high pressure ATWS trip setpoint of 1150 psig. See Section 3.2.4.5 for further discussion.

3.1.4.2.6 Variation of Doppler Coefficient

Figure 3.1.4.2.6-1 shows the effect of Doppler coefficient variation on the Turbine Trip event. The effect on suppression pool peak temperature is shown in Figure 3.1.4.2.5-2.

3.1.4.2.7 Sensitivities Affecting Turbine Trip ATWS Limit Cycles

The existence of a limit cycle in previous calculations of a Turbine Trip event with ARI failure and neglecting MSIV closure prompted a study to determine how several operating parameters affected the predicted behavior. The base case Turbine Trip event with MSIV closure at reactor water Level 1 is presented in Figure 3.1.2-3. As stated in Section 3.1.4.2, MSIV closure at the existing water Level 2 eliminates the neutron flux oscillations. A special case with no MSIV closure was used as a basis for assessing the impact of limit cycles and is shown in Figure 3.1.4.2. For this special case, the oscillations in neutron flux have characteristics similar to a limit cycle for a short period of time with an average power level of about 25% and peak near 150%. Fuel thermal power swings are less than 10%. The study was performed by individually varying each parameter about its base value and noting the effect on the limit cycle magnitude and duration. Various input parameters were selected which could have impact on the event, including boron delay time, boron capacity/effectiveness, void and Doppler reactivity coefficients, and HPCI/RCIC capacity. Observed output parameters were neutron flux, average surface heat flux, and net reactivity. Vessel pressure and water level were also examined, but only very small cycling was observed. The frequency of the cycles is virtually the same for all predicted bases, giving a limit cycle period of about 8 seconds.

Table 3.1.4.2.7-1 shows the maximum/minimum values from the predicted limit cycle for key performance parameters as a function of several operating variables. They are normalized to the special case values given in Table 3.1.4.2.7-2 to give relative relationships for the sensitivity cases. Normalized neutron flux and average surface heat flux have been plotted in Figures 3.1.4.2.7-1 through 3.1.4.2.7-6. It should be noted that although neutron flux at times experiences large oscillations, the range of average surface heat flux is relatively small. A detailed discussion of each variable is presented below.

3.1.4.2.7.1 HPCI Capacity

As shown in Figure 3.1.4.2.7-1, increasing HPCI flow tends to decrease the magnitude of oscillations. The effect is significant from 18-22% (of NBR feed flow) and then tends to level out for maximum neutron flux, while minimum neutron flux continues a rapid increase. As expected, average surface heat flux can be seen to follow the same trend as neutron flux. The effect of increasing HPCI flow on average power can also be seen in the increasing value of average neutron flux due to increased water level and subcooling. Limit cycle duration was also reduced substantially at higher HPCI flow. (Table 3.1.4.2.7-1)

3.1.4.2.7.2 Boron Timer

The effect of varying the time at which the sodium pentaborate solution reaches the core can be seen in Figure 3.1.4.2.7-2. The maximum-minimum range of neutron flux increases as time to boron injection is increased. Figure 3.1.4.2.7-3 shows the time plots of plant behavior for the case with

30 second time delay near the minimum setting which still allows for ARI action before SLCS injection begins. The impact of boron injection time is due to changes in the boron concentration in the core during the limit cycle. As time to injection decreases, boron concentration during the limit cycle will increase, thereby restricting reactivity swings.

3.1.4.2.7.3 Boron Injection Rate and/or Mixing Efficiency

Figure 3.1.4.2.7-4 shows the sensitivity to changes in boron injection capacity and/or mixing efficiency. No strong effect was seen over the range studied. The special case point (100% capacity and 95% mixing efficiency) represents a 251 BWR/4 with 86 gpm SLCS capacity and the current boron solution and storage tank.

3.1.4.2.7.4 Doppler Reactivity Coefficient

Variation of the Doppler reactivity coefficient as characterized by the full power, full fuel temperature value, gave the results shown in Figure 3.1.4.2.7-5. At higher Doppler values the average power/flow conditions were shifted slightly, giving somewhat larger limit cycles. The original base case for Turbine Trip was run with a Doppler coefficient of -0.23¢/°F and no MSIV closure. It is shown in Figure 3.1.4.2.7-6. It follows closely the trend shown by this sensitivity study.

3.1.4.2.7.5 Void Reactivity Coefficient

Another parameter important to BWR behavior is the void reactivity coefficient. Figure 3.1.4.2.7-7 shows the change in limit cycle characteristics caused by variation of this term. The range and values quoted make use of the simulated full power, rated void fraction value of the coefficient to characterize this parameter, although its actual value varies throughout each event as core void fraction changes. Figure 3.1.4.2.7-8 shows the time history of the case with a more negative coefficient (-14¢ /1%).

3.1.4.2.7.6 Fuel Conditions During Limit Cycle Oscillation

The effect of the limit cycle corepower oscillations on the fuel was analyzed. This section outlines the analysis and summarizes the results.

The core power, core pressure and clad surface heat transfer conditions for the fuel were initialized at the 100 percent operating conditions. The linear heat generation rate was initialized at the technical specification limit of 13.4 kw/ft. In the first few seconds of the fuel transient the core power and core pressure were ramped down from initial values to those corresponding to average conditions expected during the predicted limit cycle oscillations in a Turbine Trip event without MSIV closure (see Table 3.1.4.2.7-1). The fuel clad heat transfer was conservatively assumed to decrease and remain in the pool film boiling conditions. For this analysis, the expected rewetting of the fuel rod was conservatively neglected. The fuel was allowed to attain steady state temperatures at the new conditions. Then power oscillations were imposed on the new steady state as shown in Figure 3.1.4.2.7-9. The core pressure was maintained constant during the power oscillation.

The precise shape of the limit cycle power oscillation is not particularly important for these relatively high frequency oscillations. Therefore, the shape was assumed to be as shown in Figure 3.1.4.2.7-10. Here "A" represents the average core power during the oscillation and "T" is the period. "FO" is the fraction of the cycle during which the power is greater than the average power. The power rise and fall during the oscillation are assumed to be linear. The two cross hatched areas shown, "abc" and "cde", are equal, thus making the total energy generated during a cycle equal to that which would be generated at the average power level in the same time interval. The fraction "F" determines how high the peak power during cycle is. This assumed shape of the cycle is considered to be representative for the fuel cycles.

The fuel condition was analyzed for various values of "F" at an average power level of 25 percent which bounds the value for the Turbine Trip special case.

The difference between the maximum and minimum clad temperatures calculated from the analysis are tabulated in Table 3.1.4.2.7-3 and shown graphically in Figure 3.1.4.2.7-11 (page 3-93b). It can be seen that for all the cases analyzed, the clad temperature only varies up to 130°F and that the temperature swings have saturated in the sense that any higher magnitudes of the limit cycles will not appreciably increase the temperature transient on the fuel. The peak cladding temperature constructed in this bounding limit cycle is near 1250°F. This clad temperature variation occurs over only a very small section of the fuel. As "F" decreases, it is seen that the difference between the maximum and minimum clad temperatures slowly increases. This can be attributed to the increase in the excess energy deposited during the first part of the cycle. (This excess energy is given by A T/2 (1-F) and is represented by the area of the triangle abc in the Figure 3.1.4.2.7-10.) Therefore, even with conservative assumptions about the limit cycle shape, the fuel peak clad temperature is within 70°F of that without limit cycles. Thus, fuel cladding temperature is dominantly determined by the average power level and fuel integrity is not significantly affected by the small temperature swings seen in the predicted oscillations.

3.1.4.3 IORV Sensitivity Studies

Table 3.1.4.3-1 summarizes the results of the sensitivities on the BWR/4 IORV - ATWS event. The effect of the listed parameters on peak suppression pool temperature and peak wetwell airspace pressure was determined.

3.1.4.3.1 Variation of S/RV Capacity

The capacity of the S/RVs was varied ±20%. Peak pool temperature changed by -8°F and +6°F, respectively. Corresponding peak containment pressures changed by +1.3 and -1.5 psi from the base case value. This change is shown in Figure 3.1.4.3.1-1.

3.1.4.3.2 Variation of Boron Delay

The boron is assumed to reach the core at 2½ minutes after the manual scram attempt. The effects of increasing this time by 5 and 10 minutes were determined. An increase of 5 minutes resulted in a suppression pool temperature increase of 7°F and an increase of 10 minutes resulted in a temperature increase of 13°F. Containment pressure varied accordingly. This change is shown in Figure 3.1.4.3.2-1.

3.1.4.3.3 Variation of Boron Capacity/Mixing

The effective rate of boron injection into the core is the product of the boron pumping capacity and mixing efficiency. This effective rate was varied by ±20% resulting in peak pool temperatures of 3°F below and 4°F above the base case value, respectively. Peak containment pressures varied accordingly. This is shown in Figure 3.1.4.3.3-1.

3.1.4.3.4 Variation in RHR Capacity

To determine the effect of varying RHR heat exchanger capacities, the base capacity of 428 BTU/sec-°F was altered by ±50%. Increasing the capacity to 642 BTU/sec yielded a 10°F temperature reduction and lowered the peak containment pressure by less than 2 psi. For the opposite case of decreasing capacity to 214 Btu/sec-°F the results were a 17°F increase in temperature and a 4 psi pressure rise. The temperature change is shown in Figure 3.1.4.3.4-1.

3.1.4.3.5 Variation of RHR Delay

The RHR system is assumed to be in operation at time zero of this event. This was varied by delaying RHR start by 5 minutes and 10 minutes from time zero. The effect on peak pool temperature was an increase of about 1°F and 1°F, respectively.

3.1.4.3.6 Variations of Pool Size

The suppression pool mass was varied by 20% to simulate different sized plants. The larger pool mass provides a bigger heat sink, thus reducing the peak pool temperature by nearly 8°F and peak pressure by 1.5 psi. For the lower pool mass, pool temperature increases 12°F and peak pressure by nearly 3 psi. BWR 4 Mark I and Mark II plants can utilize this sensitivity study to estimate the effect of different pool sizes on bulk pool temperature. The variation is shown in Figure 3.1.4.3.6-1.

3.1.4.3.7 Variation of Pool and Service Water Temperatures

The pool and service water temperatures were found to significantly affect peak pool temperature and containment pressure. Increasing these temperatures by 20°F produced a rise in pool temperature of 8°F and an increase of 1.7 psi in peak pressure. Reducing the temperatures by 20°F yielded decreases of 6°F and 1.2 psi, respectively. This is shown in Figure 3.1.4.3.7-1.

3.1.4.3.8 Variation of RHR Capacity and Service Water Temperature

In some plants, the RHR heat exchanger size is defined dependent on design service water temperature. For example, a higher service water temperature will require a larger RHR heat exchanger. In this analysis, these two items were varied together. The service water temperature was varied ±20°F and the RHR heat exchanger capacity was varied ±50% for the larger heat exchanger and higher temperature, the maximum suppression pool temperature was changed by -3°F and for the lower service water temperature and smaller heat exchanger, the maximum suppression pool temperature was changed by +12°F. Containment pressures changed by -0.6 psi and +2.7 psi, respectively. Temperature variation is shown in Figure 3.1.4.3-4 and 3.1.4.3.7-1.

3.1.5 Loss of Condenser Vacuum

3.1.5.1 Overview of Response Without Scram

This transient starts with turbine trip due to low condenser vacuum. Therefore the beginning is the same as Turbine Trip events (see Section 3.1.2). There is a rapid steam shutoff causing pressure and power increases which are limited by the action of the S/RVs and RPT. Note that direct pump trip from Turbine Trip was conservatively neglected. Since the MSIVs and turbine bypass valves also close when condenser vacuum has further dropped to their setpoint, S/RV cycling increases considerably compared to the original Turbine Trip case. Even so, the bulk pool temperature and pressure remain well within the containment design requirements. Therefore, this event is similar to the Turbine Trip event as far as the peak power and pressure characteristics are concerned and similar to the MSIV closure case with respect to suppression pool temperature and pressure.

3.1.5.2 Sequence of Events for Loss of Condenser Vacuum

The listing of significant events during this ATWS event is provided in Table 3.1.5-1. Results with and without ARI are presented.

This transient starts with the closure of all turbine stop valves (within about 0.1 second) when the unexpected decline in condenser vacuum reaches the turbine trip setpoint. If the unit has turbine driven feedwater pumps, they also trip at the same low vacuum setpoint. For the ARI failure case, the feedwater is assumed to remain as if motor-driven pumps were available until the feedwater limit action shuts them down (the most limiting case). Figure 3.1.5-1 shows the initial portions of the event for the more likely plant ATWS transient in which ARI provides a diverse logic path to quickly shut down the reactor, Figure 3.1.5-2 shows the initial portion for this case in which ARI also is assumed to fail.

In both cases, the initial power and pressure increase. Neutron flux reaches 403% NBR within 1 second; fuel average heat flux reaches 133% NBR at about 2 seconds. Some fuel may experience boiling transition, however, coolable geometry is maintained. Peak pressure occurs at the vessel bottom and is 1195 psig at 2.7 seconds. The normal reactor scram signals occur from position switches on the valves, high neutron flux, and high vessel pressure but are ignored for this analysis. The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to automatic action of RPT which is initiated when vessel dome pressure exceeds 1150 psig and the relieving action of the S/RVs which all open then start reclosing near 13 seconds. By about 30 seconds, the condenser vacuum is assumed to have fallen enough (8 in Hg) to initiate MSIV and bypass valve closure. This results in another pressure and power rise to 1134 psig and

94% NBR, respectively. Both of these peaks are lower than the earlier values. Peak heat flux rises momentarily, but remains less than 45% and fuel geometry is maintained.

The long term behavior of this transient is very much like the MSIV closure event which is discussed in detail in Section 3.1.1. Figure 3.1.5-3 shows the long term behavior predicted for this event. The peak bulk pool temperature and pressure which occur after 27 minutes are 188°F and 10.7 psig, respectively. These values remain well within the containment design requirements. Table 3.1.5-2 summarizes the significant values.

Thus it can be seen that the loss of condenser vacuum event combined with failure to scram is adequately mitigated for a representative BWR 4/Mark I.

3.1.6 Pressure Regulator Failure - Zero Steam Demand

The failure of the controlling pressure regulator to the lower limit passes control of the main turbine control valves to the backup regulator. The backup regulator is nominally set 3 to 5 psi higher than the controlling regulator. As the transfer is made, a disturbance is introduced to the system but none of the variables are disturbed sufficiently to reach any scram trip setpoint. This transient is expected to be milder than the Turbine Trip case and is not included in BWR/4 safety analysis reports. For these reasons, this event is not analyzed here as an ATWS event for BWR/4.

3.1.7 Loss of a Feedwater Heater

3.1.7.1 Overview of Response Without Scram

This is a mild transient compared to the other ATWS events. The neutron flux does not reach the scram setpoint, and the pressure rise is insignificant. Therefore, automatic ATWS logic (e.g., RPT) does not occur, nor are HPCI or RCIC initiated. This is a gradual subcooling transient. The entire transient settles out when the feedwater temperature fully stabilizes. The reactor settles out to a new equilibrium power condition at full core flow with recirculation flow assumed to be under manual control. If automatic flow control was active, the power increase would be less. Manual operator action accomplishes reactor shutdown.

3.1.7.2 Sequence of Events for Loss of a Feedwater Heater

Table 3.1.7-1 gives the sequence of events for this transient.

In this event, loss of a key group of feedwater heaters gives the reactor coolant feedwater flow (decreased 65°F) which produces an increase in core inlet subcooling leading to an increase in core power. Following the transport delay through the feedwater lines (neglected in this analysis) and the time constant for cool-down of the heater tubes, average fuel surface heat flux rises to a maximum value of 112% which is lower than the flux scram setpoint. No fuel reaches boiling transition, even if the plant was initially at thermal operating limits. The reactor is conservatively assumed to be on manual flow control, therefore, core inlet flow remains at 100%. Had the reactor been on automatic flow control, core inlet flow would have changed to decrease the severity of the transient. The peak pressure (vessel bottom) of 1046 psig occurs near 56 seconds. Figure 3.1.4.7-1 shows the response of this event. The water level remains within the normal control range throughout the transient.

When the power reaches 108% NBR near 34 seconds, a high power alarm occurs. For this analysis, it is assumed that attempts will be made to bring the power down by inserting rods. If this is not successful, manual scram will be initiated. This also initiates ARI and the SLCS timed logic. However, in this analysis, manual scram is also assumed to fail. By about 11 minutes, ARI will have been accomplished and power is terminated. If the ARI function is arbitrarily assumed to fail, as well as all other attempts to insert control rods within the two minute period, the automatic start of boron injection will begin through the jet pump instrument lines. By about 35 minutes the power has decreased below 1% NBR. Recirculation flow remains active during boron injection, providing mixing and dispersion throughout the primary system.

Thus it can be seen that a loss of feedwater heater event combined with a failure to scram can be adequately mitigated for a representative BWR 4/ Mark I. Note that this event was analyzed for a 65°F loss in feedwater heating rather than the 60°F as specified in NUREG 0460 (Volume 3).

3.1.8 Feedwater Controller Failure - Maximum Demand

3.1.8.1 Overview of Response Without Scram

The behavior of the plant is separable into an early or short term transient resulting in a gradual power increase, then a sharp pressure rise and power peak. The longer term segment requires evaluation of coolant and containment conditions as the reactor is finally brought to shutdown. The relief valves open in the early part of the transient and stay closed for the rest of the event (except for the possibility of single valve cycling very late in the event). Figures 3.1.4.8-1 and 3.1.4.8-2 illustrate the early period. The relief valves are assisted by RPT to limit the pressure disturbance acceptably. Note that the direct RPT from turbine trip was conservatively ignored. RPT also assures the establishment of a relatively low power generation rate for the long term portion of the transient. The effectiveness of RPT as presented in earlier reports is once again confirmed by this analysis.

Containment peak temperature and pressure remain well below design limits since relief valves stay open for a relatively short period. The power shutdown is achieved in either of two ways. ARI employs an alternate design of the protection logic leading to a diverse insertion of the control rods. In the unlikely event that ARI also fails, the automated SLCS provides further protection and shutdown capability.

3.1.8.2 Sequence of Events for Feedwater Controller Failure - Maximum Demand

The time sequence of events for this transient is presented in Table 3.1.8-1. Both successful ARI initiation, and ARI failure cases are considered. The initiating event is the failure of the feedwater controller to the maximum demand position (125% NBR was assumed). The feedwater flow rapidly responds, causing vessel level to rise. When the high level trip setpoint (L8) is reached near 6 seconds, the turbine and feedwater are tripped. This results in a scram signal which for purposes of this analysis fails to initiate a scram. With the occurrence of the turbine trip, this event becomes very similar to the Turbine Trip transient. Figures 3.1.8-1 and 3.1.8-2 show the early portion of the event for the cases of ARI failure, and successful ARI actuation, respectively. For each case, the peak power and flux are the same with a maximum flux of 511% NBR near 7 seconds and a peak vessel bottom pressure of 1195 psig around 9 seconds. Fuel average heat flux reaches a maximum at 8 seconds of 137% NBR. Some fuel may experience boiling transition, if the core were operating at its thermal limit. The peak cladding temperature reaches about 1530°F, and coolable geometry is maintained. Despite the assumed failure to scram based upon high neutron flux, vessel level and dome pressure generated scram signals, the transient pressure is maintained well below the 1500 psig Service Level C overpressure limit. This is accomplished through the combination of RPT (initiated on high dome pressure) and actuation of relief valves. Relief valve flow begins at 8 seconds, ceases at 37 seconds. S/RV's will open and cycle before permanently closing. This is shown along with vessel steam flow. The difference in vessel and relief steam flow is made up by the steamflow through the turbine bypass valves to the condenser.

At approximately 28 seconds, ARI will begin to insert control rods into the core thereby shutting down the reactor. This will deactivate the SLCS turning the remainder of the event into normal feedwater flow controller failure transient. No further relief valve flow will occur. The decay heat will be passed through the turbine bypass valves to the condneser.

The RHR can be activated in the pool cooling mode whenever convenient to reduce the pool temperature and any final, single valve cycles can be accommodated. Vessel level, which drops due to feedwater shutoff at high water level, is recovered and maintained in the normal water range by means of the HPCI/RCIC systems.

In the unlikely event of ARI failure, the event can still be mitigated through action of the SLCS. With confirmation from the flux monitoring system and the rod position indicating system that scram has not occurred, the SLCS will be activated. The long term behavior predicted for this event is shown in Figure 3.1.3-3. Boron first enters the core at about 3 minutes via the jet pump instrument lines and commences to shut down the system, with hot shutdown occurring near 17 minutes. Vessel level experiences slow cycles about the normal water level caused by the intermittent action of the RCIC and HPCI systems assumed to be automatically cycling between L2 and L8. Boron concentration will continue to increase until the entire inventory has been injected into the core around 50 minutes. At this point the concentration is sufficient to maintain cold nuclear shutdown conditions when the RHR system is switched to the reactor shutdown cooling mode and the plant is brought to a cold shutdown condition.

Thus it can be seen that a feedwater controller failure event (maximum demand) combined with a failure to scram is adequately mitigated for a representative BWR 4/Mark I.

3.1.9 Pressure Regulator Failure - Maximum Steam Demand

3.1.9.1 Overview of Response Without Scram

The initial portion of this transient consists of a decrease in reactor pressure and power as the turbine control valves open to the maximum position followed by a rapid rise in pressure and power due to MSIV closure on low steam line pressure. Scram is normally initiated at this time from the MSIV position switches. Should they fail, additional scram signals occur from high flux, high pressure and low water level. Once the MSIV's close, the characteristics of the remaining portion of the transient are very much the same as the MSIV event.

The power and pressure increases are limited by the action of the S/RV's and RPT. With normal scram assumed to be failed, the long term power shutdown is ashieved in either of two ways. ARI employs an alternate design of the protection logic leading to diverse insertion of the control rods. In the unlikely event that ARI fails the automated SLCS provides further protection and shutdown capability.

3.1.9.2 Sequence of Events for Pressure Regulator Failure - Maximum Demand

The significant events during this event are provided in Table 3.1.9-1. Results for both cases - with ARI and also assuming its failure - are presented.

This event begins with the failure of the pressure regulator to the maximum steam demand value. The turbine control valves open allowing an increase in vessel steam flow which results in a rapid decrease in vessel pressure. This leads to a low pressure isolation signal at about 19 seconds when the MSIV's are tripped close. Once this occurs, the transient becomes much like an MSIV closure event. The isolation is followed by a rapid rise in power and pressure. Figure 3.1.9-1 shows the event for the more likely plant ATWS transient in which ARI quickly shuts down the reactor. The initial portion of the case in which ARI also fails and the automated SLCS is called upon to shut down the reactor shown in Figure 3.1.9.2. In both cases, the peak power and pressure are the same. The neutron flux reaches 585% NBR near 25 seconds, fuel average heat flux reaches 139% NBR at about 27 seconds. Some fuel experiences boiling transition (just less than 17% if the core was initially at its operating thermal limit) however, peak cladding temperature is about 1630°F and coolable geometry is maintained. The peak pressure occurs at vessel bottom and is 1280 psig at approximately 31 seconds. The normal reactor scrams occur from position switches on MSIV's, high neutron flux, and the high vessel pressure, but are ignored for this analysis. The transient pressure is limited well within the Service Level C overpressure limit of 1500 psig. This is due to the automatic action of RPT which is initiated when vessel pressure exceeds 1150 psig near 26 seconds and the relieving action of the S/RV's which all open, then start reclosing near 47 seconds.

By about 46 seconds, the high pressure logic which began the ATWS protection will have implemented the ARI function. The remainder of the event is a normal pressure regulator failure shutdown.

If the ARI function is assumed to fail as well as all other attempts to insert control rods within the two-minute timed period, the ATWS logic will continue to sense that the APRM signals are not downscale and not enough rods are in their full-in positions, and the automatic start of boron injection will begin. The long term behavior predicted for this event is shown in Figure 3.1.9-3. Introduction of boron to the core allows the restoration of level and core flow before reaching nuclear shutdown at 18 minutes. Thereafter, only decay heat reaches the pool, giving the peak pool temperature of 189°F (11.0 psig) at about 27 minutes. These values remain well within the pressure suppression requirements. Water level inside the core shroud is a two-phase mixture which remains well above the core and up into the steam separator standpipes as RCIC and HPCI flow provide coolant inventory. The boron will continue to build the poison concentration in the vessel until it is all injected near 50 minutes making it possible for a controlled reactor cooldown. The total concentration is specified to be enough to maintain cold nuclear shutdown conditions even when the RHR system is eventually switched to the reactor shutdown cooling mode, bringing the plant by normal procedures. Listed in Table 3.1.9-2 is a summary of the maximum conditions.

Thus it can be seen that a pressure regulator failure (maximum demand) combined with a failure to scram is adequately mitigated for a representative BWR 4/Mark I.

3.1.10 Loss of Feedwater

3.1.10.1 Overview of Response Without Scram

This event has no rapid excursions as in some of the other events but is a long term power reduction and depressurization. Since the pressure begins to fall at the outset of the transient, the need for relief valves does not arise until isolation occurs very late in the event and only single valve cycling is expected to handle decay heat. Even this cycling can be avoided by taking the mode selector switch out of the "RUN" mode to prevent isolation on low steam line pressure. Thus, containment limits are not challenged. Except for the use of the liquid boron solution for shutdown, the procedure followed here is virtually identical to the normal shutdown event.

3.1.10.2 Sequence of Events for Loss of Feedwater

In this event all feedwater flow is assumed to be lost in about 5 seconds. The resulting sequence of events is shown in Table 3.1.10-1 for both cases with and without ARI. Figure 3.1.10-1 shows the initial portion of the event for the more likely plant ATWS transient in which ARI quickly shuts down the reactor. Figure 3.1.10-2 shows the early portion of the case in which ARI also fails and the automated SLCS is called upon to shut down the reactor.

In both cases, after the loss of feedwater has taken place, pressure, water level and neutron flux begin to fall. Around 19 seconds low water level (L2) is reached. This trips the recirculation drive motor breakers, initiates ARI, initiates the HPCI and RCIC and activates the SLCS timed logic. Neglected was the recirculation runback which would have occurred earlier from coincident low level logic which began the ATWS protection will have initiated the ARI function. At about 39 seconds HPCI and RCIC flows start. They replace the main feedwater system and begin to overcome the inventory loss. The vessel level continues to decrease folloing ARI and the minimum for the case is reached near 61 seconds as shown in Figure 3.1.10-1. The two-phase mixture level inside the core shroud always remains above the top of the fuel. The HPCI and RCIC will restore level to its normal range, for either automatic cycling between Level 2 and 8 setpoints or the operator takes over manual level control by using the RCIC (preferred).

If the ARI function is arbitrarily assumed to fail as well as all other scrams and attempts to insert enough control rods within the two-minute timed period, the ATWS logic will continue to sense that the APRM signals are not downscale and not enough rods are in their full-in positions, and the automatic start of boron injection will begin. SLCS injection is started at approximately 2 minutes and boron reaches the core 1 minute later. During the following 16 minute period (out to about 1100 seconds in Figure 3.1.10-3), the key result is that power is suppressed slightly, reducing the steaming rate and allowing water level to be restored. This also induces higher natural circulation core flow which follows the water level behavior. The level reaches the high level turn-off (Level 8) of the HPCI and RCIC at about 16 minutes. The turbine is also tripped at this level but since the turbine steam bypass system opens immediately, no significant pressure disturbance is experienced.

By 17 minutes the generated power is below 1% NBR and continues to decrease due to the accumulation of boron in the reactor. The net reactivity also stays negative, (nuclear hot shutdown). At this time the generated power is practically zero and the only heat in the vessel is the decay heat.

Thus it can be seen that a loss of feedwater combined with a failure to scram is adequately mitigated for a representative BWR 4/Mark I.

3.1.11 Loss of Normal AC Power

3.1.11.1 Overview of Response Without Scram

Initially in this transient, a sharp rise in reactor pressure and power occur due to MSIV closure as a result of loss of normal AC power. Scram is initiated at this time from the MSIV position switches if it had not occurred yet from loss of reactor trip system power. Should these signals fail to cause scram, additional scram signals occur from high flux, high pressure and low water level. The power and pressure increases are limited by the action of the S/RV's and RPT (which occurs at the start of this event). With normal scram assumed to have failed the long term power shutdown is achieved in two ways. ARI employs an alternate design of the protection logic leading to diverse insertion of the control rods. In the event that ARI fails, the automated SLCS provides further protection and shutdown capability.

3.1.11.2 Sequence of Events For Loss of Normal AC Power

The listing of significant events during this event is provided in Table 3.1.11-1. Results for both cases - with ARI and also assuming its failure are presented.

There are two ways of experiencing this event: 1) loss of all auxiliary power transformers; and 2) loss of all grid connections. The main difference between the two approaches is that in the latter one, load rejection occurs at the outset of the transient which results in turbine-generator trip. In both cases, MSIV closure takes place near 2 seconds. This is the earliest time isolation can occur and is based on relay-type reactor trip system (RTS) circuitry.

Since in loss of all grid connections the turbine trips first as opposed to MSIV closure in the loss of all auxiliary power transformers case, it turns out to be a less severe event in terms of peak power and pressure. Therefore the rest of the discussion is limited to the case where loss of all auxiliary power transformer occurs. The sequence of events as outlined in Table 3.1.11-1 describes the event.

This event begins with the loss of recirculation pumps and feedwater pumps. This leads to an initial reduction in power and pressure. At 2 seconds, MSIV closure is assumed to take place, which results in a rapid rise in power and pressure. Figure 3.1.11-1 shows initial portions of the event for the more likely plant ATWS transient in which ARI quickly shuts down the reactor, and Figure 3.1.11-2 shows the initial portion of the case in which ARI also fails and automated SLCS is called upon to shutdown the reactor.

In both cases, the peak power and pressure are the same. The neutron flux reaches 253% NBR near 7 seconds, however fuel average heat flux does not exceed the initial value. The peak pressure occurs at vessel bottom and is 1172 psig at 8 seconds. The normal scram signals occur due to loss of AC power and also due to position switches on MSIV's, high neutron flux and the high vessel pressure but are ignored for this analysis. The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to RPT at the start of the transient and the relieving action of the S/RV's which all open, then start reclosing near 16 seconds.

By about 28 seconds, the high pressure logic would provide ATWS protection, by activating ARI. This eliminates the automatic boron injection and allows the remainder of the event to proceed toward normal shutdown. The primary relief valve flow stops near 75 seconds, followed only by single valve cycling on the "tail" of the isolation event. The RHR can be activated in the pool cooling mode as soon as water level recovery is clearly indicated, to control pool temperature. Reactor water level is restored quickly to its normal range by RCIC and HPCI flow.

If the ARI function is assumed to fail as well as all other attempts to insert control rods within the two-minute time period, the ATWS logic will continue to sense that not enough rods are in their full-in positions, and the automatic boron injection will begin. The long term behavior predicted for this event is shown in Figure 3.1.11-3. Introduction of boron to the core around 3 minutes again restores level and core flow before decreasing power near 17 minutes when nuclear shutdown is achieved. Thereafter, only decay heat reaches the pool, giving the peak bulk pool temperature of 182°F (9.5 psig) at about 33 minutes. These values remain well within the containment design requirements. Water level inside the core shroud is a two-phase mixture which remains well above the core and up into the steam separator standpipes as RCIC and HPCI flow provide coolant inventory. The boron will continue to build the poison concentration in the vessel until it is all injected near 50 minutes making it possible for a controlled reactor cooldown. The total concentration is specified to be enough to maintain cold nuclear shutdown conditions when the RHR system is eventially switched to the reactor shutdown cooling mode, bringing the plant to cold shutdown.

Thus it can be seen that a loss of normal AC power combined with a failure to scram is adequately mitigated for a representative BWR 4/Mark I.

Table 3.1.6-1

RECIRCULATION FLOW CONTROLLER FAILURE - INCREASING FLOW

	Sequence of Events	Time
1.	M/G Speed Controller Failure	0 seconds
2.	Neutron flux reaches 120%, APRM scram assumed to fail	2 seconds
3.	Power peaks	2 seconds
4.	Maximum fuel surface heat flux occurs	5 seconds
5.	Vessel pressure peaks	6 seconds
6.	Core flow increase levels off	6 seconds
7.	New core equilibrium conditions (All parameters within normal limits, power and feedflow slowly decreasing as steady state feedwater heating is established.)	20 seconds
8.	Manual rod insertion or (if this fails) manual ARI and (if it fails) SLCS initiation	10 minutes
9.	Hot shutdown achieved	30 minutes

is operating initially at its operating limit. A small pressure rise occurs, peaking near 6 seconds with a vessel bottom pressure of 1020 psig (compared to an intial bottom pressure of 965 psig). Simultaneous with the above events, vessel level experiences a small decrease and then recovers to its initial position, and feedwater flow rises in response to the level change. As core flow levels off to approximately 61% of rated near 10 seconds, the power settles out as do all other parameters. At this point, the transient is essentially over. By 20 seconds, all parameters have reached equilibrium except power and feedwater flow which continue to slowly decrease following the warming of the feedwater heaters.

Because of the mildness of the event, no automatic pressure or level dependent actions are initiated. Containment is not affected since no relief valves are actuated. Subsequent operator action would be expected to initiate a manual shutdown, utilizing the SLCS if manual insertion or scram of rods remains unsuccessful. Initiation of ARI/SLCS from manual scram near 10 minutes would shut down the plant immediately (ARI) or by about 30 minutes (SLCS). Recirculation flow would be maintained near full flow initially and at partial flow in order to maximize boron dispersion throughout the vessel and to provide a near-normal shutdown sequence.

Thus it can be seen that a recirculation flow control failure combined with a failure to scram is adequately mitigated for a representative BWR 4/Mark I.

3.1.13 Startup of the Idle Recirculation Pump

This event is similar to the Recirculation Flow Controller Failure - Maximum Demand. Both of these events result in increased core power which results from the increased core flow. The Startup of the Idle Recirculation Pump Event has been shown in Safety Analysis Reports to be less severe than the Recirculation Flow Controller Failure and, therefore, further transient-specific analyses have not been done.

3.1.14 Inadvertent Opening of all Bypass Valves

This event will be similar to the Pressure Regulator Failure - Maximum Steam Demand. Since the turbine control valves will try to compensate for the pressure reduction, the results will be less severe. For those plants with smaller bypass capacity, the event will be even less severe.

3.1.15 Shutdown Cooling (RHR) Malfunction - Decreasing Temperature

This event can only occur at very low pressures. The shutoff head of the shutdown cooling pumps is less than 300 psig. In this condition, the reactor has almost no voids in it and therefore only little if any positive reactivity is inserted. Therefore, this event is not considered further.

3.1.16 Rod withdrawal Error From Zero and Full Power

3.1.16.1 Power Range

The control rod withdrawal error (CRWE) in the power range is not dependent on reactor scram for termination. The local power range monitors (LPRMs) will sense the local power increase due to the continuous withdrawal of a control rod. When the power exceeds the preset power point, the rod block monitor (RBM) will block further withdrawal of the control rod before the MCPR safety limit is exceeded thereby preventing fuel damage. Thus, scram is not required to insure that no fuel damage occurs due to the CRWE in the power range.

3.1.16.2 Startup Range

3.1.16.2.1 Plants Employing the Banked Position Withdrawal Sequence (Early BWR/4, All BWR/5 + 6)

In plants which employ the banked position withdrawal sequence (BPWS), only in-sequence control rods may be withdrawn in the 100% (all rods in) to 75% control rod density range. These control rods may be withdrawn from the fully inserted position to the fully withdrawn position. From the 75% control rod density point to the low power set point (i.e. 20% of rated power) the control rods must be withdrawn in the banked position mode of control rod withdrawal. (See GESSAR Section 4.3.2.6 for a description of the banked position withdrawal sequence.)

The first 25% of the control rods pulled consist of BPWS groups 1 and 2 or 3 and 4 and are withdrawn from full-in to full-out because the reactor core is generally subcritical until the control rod density is lower than 75%. In some very reactive cores, criticality may be achieved before the 75% rod density point is reached. However, it has been determined that the first control rod withdrawn in a group has the highest reactivity worth. All the remaining control rods in the group will be of lower worth. Even though the first 25% of the control rods are withdrawn from full-in to full-out, the worth of these rods need not be considered since the reactor core is subcritical when the first group (i.e. first 12.5%) of control rods is withdrawn. The subcritical condition will exist in the reactor core as the first rods of the second group of control rods are withdrawn. Criticality may be achieved after a number of control rods in the second group have been withdrawn. However, the control rods pulled at this point are very low worth (i.e., the rod worth is less than 0.003 Δk) and need not be considered in the context of a rod withdrawal error.

Between 75% rod density and the low power set point control rods are pulled in the banked position mode of rod withdrawal. The rod sequence control system (RSCS) will not allow the continuous removal of a control rod in this range of rod withdrawal.

The BPWS limits the worth of control rods by imposing contraints on the sequence in which specified control rods are withdrawn. The maximum positive reactivity which can be achieved by rod withdrawal within the constraints of the BPWS is less than 0.0035 Δk. Since the RSCS physically prohibits removal of rods that do not fall within the BPWS constraints, a control rod withdrawal error analysis without scram is not necessary.

3.1.16.2.2 Plants Employing the Group Notch Withdrawal Sequence (Early BWR/4)

In plants employing the group notch (GN) mode of rod withdrawal, only in-sequence rods may be withdrawn in the 100% to 50% control rod density range. These rods may be withdrawn from the fully inserted to the fully withdrawn position. From the 50% control rod density point to the low power set point (20% of rated power) the control rods are withdrawn in the group notch mode. Continuous control rod withdrawal is therefore not possible in the range between 50% control rod density point and the pre-set power point.

In the 100% to 50% control rod density range continuous withdrawal of in-sequence control rods is allowed. The group notch RSCS physically prohibits the withdrawal of out-of-sequence control rods. Since only in-sequence rods can be withdrawn, the control rod worth is kept low.

In-sequence control rods are normally withdrawn continuously from full-in to full-out while the core is subcritical. When the core is critical (or near critical), the control rods are generally withdrawn in the jog mode (notched out) in order to maintain a constant reactor system heatup rate. If the core is critical and a control rod is continuously withdrawn, the local power in the region around the control rod will increase more quickly and possibly cause a slight local power increase above the normal steady state power level associated with notch withdrawal. However, the intermediate range monitor (IRM) would block withdrawal of the control rod when the IRM instrument reading reached 90% of scale. Analyses indicate that even if a 1.6% Δk out-of-sequence rod were continuously withdrawn, the total amount of reactivity inserted into the core would be only 0.7% Δk before an IRM block signal stopped the rod from being withdrawn. These analyses assume the IRM system is in the worst assumed Technical Specification bypass condition (i.e., IRM instrument closest to rod being withdrawn in each IRM instrument string is bypassed).

Thus, if the core is critical and a low worth control rod is selected and continuously withdrawn, there will be no power increase of any consequence. If a relatively high worth in-sequence rod is selected and withdrawn continuously, the IRM rod block function would protect the core from any potential fuel damage. This is assuming that the in-sequence rod is of sufficient reactivity worth to cause fuel damage. As stated above, the RSCS prohibits withdrawal of out-of-sequence rods.

With the group notch RSCS and IRM rod block function, scram would not be required to terminate a control rod withdrawal error in the startup range. Hence, the CRWE without scram is of no concern.

Table 3.1-1

SUMMARY OF ATWS RESULTS - BWR/4

ARI FAILURE, 2 PUMP SLCS, 2 MINUTE LOGIC DELAY

Transient	Maximum Neutron Flux (% NBR)	Maximum Average Fuel Heat Flux (% NBR)	Maximum % of Fuel in Boiling Transition	Maximum Pressure (Vessel Bottom) (psig)	Minimum Water Level (Ft Below Separator skirt)	Maximum Suppression Pool Temp. (°F)	Maximum Containment Pressure (psig)
MSIV Closure	527	143	12	1280	11.59	186	10.3
Turbine Trip with Bypass	392	133	-	1193	11.77	162	6.1
Inadvertent Opening of a S/R Valve	100	100	-	1044	-	183	9.7
Loss of Condenser Vacuum	403	133	-	1195	10.98	188	10.7
Loss of a Feedwater Heater	113	112	-	1046	-	90	No change
Feedwater Controller Failure-Max Demand	511	137	1	1195	8.90	99	0.5
Pressure Regulator Failure-Max Steam Demand	585	139	17	1280	11.14	189	11.0
Loss of Normal Feedwater Flow	100	100	-	1044	9.35	90	No change

Table 3.1-1

SUMMARY OF ATWS RESULTS - BWR/4

ARI FAILURE, 2 PUMP SLCS, 2 MINUTE LOGIC DELAY (Continued)

Transient	Maximum Neutron Flux (% NBR)	Maximum Average Fuel Heat Flux (% NBR)	Maximum % of Fuel in Boiling Transition	Maximum Pressure (Vessel Bottom) (psig)	Minimum Water Level (Ft Below Separator Skirt)	Maximum Suppression Pool Temp. (°F)	Maximum Containment Pressure (psig)
Loss of Normal AC Power	258	100	-	1172	10.55	182	9.5
Recirculation Flow Controlled Failure-Max Demand	530	92	-	1020	-	90	No change
Turbine Trip with Bypass Failure	655	138	-	1267	10.92	191	11.4

Table 3.1.1-1

BWR/4 MSIV CLOSURE

Sequence of Events

		Time
1.	Nominal 4 second MSIV Closure - Scram Fails	0
2.	Pressure Rise Begins	0
3.	Relief Valves Lift	4 Seconds
4.	Some Fuel Experiences Transition Boiling	5 Seconds
5.	Recirculation Pumps Trip on High Pressure, ARI is initiated, and Timed SLCS Logic is Triggered	5 Seconds
6.	Vessel Pressure Peaks	9 Seconds
7.	ARI Fails	30 Seconds
8.	Feedwater Flow Coasts Down to Lower Limit	45 Seconds
9.	HPCI and RCIC Flow Starts after Level 2 Initiation	1 Minute
10.	ATWS Logic Timer Complete. SLCS Starts	2 Minutes
11.	Liquid Control Flow Reaches Core	3 Minutes
12.	Water Level Reaches Minimum and Begins to Rise	4 Minutes
13.	RHR Flow Begins (Pool Cooling)	11 Minutes
14.	Hot Shutdown Achieved*	17 Minutes
15.	Containment Temperature and Pressure Peak	28 Minutes

*Hot Shutdown is defined as generated power remaining below 1% NBR.

Table 3.1.1-2

BWR/4 MSIV CLOSURE - SUMMARY

With ARI Failure	86 GPM Boron 95% Mixing Eff 2 Min Logic Delay *MSIV
Maximum Neutron Flux (%)	527
Maximum Vessel Bottom Pressure (psig)	1296
Maximum Average Heat Flux (%)	143
Maximum Bulk Suppression Pool Temperature (°F)	186
Associated Containment Pressure (psig)	10.4

*With ARI all events occurring prior to 30 seconds remain unchanged.

Table 3.1.2-1

BWR/4 TURBINE TRIP

Sequence of Events

		Time
1.	Turbine Trips - Scram Fails	0
2.	Pressure Rise Begins	0
3.	Relief Valves Lift	2 seconds
4.	Some Fuel Experiences Transition Boiling	2 seconds
5.	Recirculation Pumps Trip on High Pressure, ARI is Initiated and Timed SLCS Logic is Triggered	2 seconds 3 seconds
6.	Vessel Pressure Peaks	3 seconds
7.	ARI Fails	30 seconds
8.	Feedwater Flow Run Back to Lower Limit Value	45 seconds
9.	HPCI and RCIC Flow Starts on Level 2 Initiation	78 seconds
10.	ATWS Logic Timer Complete, SLCS Starts	2 minutes
11.	Liquid Control Flow Reaches Core	3 minutes
12.	MSIV Closure	5 minutes
13.	Water Level Reaches Minimum and Begins to Rise	6 minutes
14.	RHR Flow Begins (Pool Cooling)	11 minutes
15.	Hot Shutdown Achieved	18 minutes

Table 3.1.2-2

BWR/4 TURBINE TRIP - SUMMARY

With ARI Failure	86 GPM Boron 95% Mixing Eff 2 Min Logic Delay Turbine Trip
Maximum Neutron Flux (%)	392
Maximum Vessel Bottom Pressure (psig)	1193
Maximum Average Heat Flux (%)	132
Maximum Bulk Suppression Pool Temperature (°F)	162
Associated Containment Pressure (psig)	6.1

Table 3.1.3-1

BWR/4 INADVERTENT OPENING OF A RELIEF VALVE,

Sequence of Events	Time
1. Relief Valve Opens Inadvertently and Fails to Close	0
2. Alarm Sounds at 95°F and Operator Initiated Pool Cooling	2 minutes
3. Suppression Pool Temperature Reaches 110°F, Operator Attempts Manual Scram. ARI and Timed SLCS Logic Initiated, Scram Fails	7.5 minutes
4. ARI Fails	8 minutes
5. SLCS Automatically Starts	9.5 minutes
6. Control Liquid Reached Core	10 minutes
7. Power is Less than Relief Valve Capacity	24 minutes
8. Isolation on Low Steam Line Pressure	28 minutes
9. Peak Suppression Pool Temperature and Pressure are Reached	95 minutes

Table 3.1.3-2

BWR/4 INADVERTENT OPENING OF A RELIEF VALVE - SUMMARY

	86 GPM Boron 95% Mixing Eff 2 Min Logic Delay
With ARI Failure	IORV
Maximum Bulk Suppression Pool Temperature Temperature (°F)	183
Associated Containment Pressure (psig)	9.7

Table 3.1.4.1-1

BWR/4 MSIV ATWS SENSITIVITY RESULTS SUMMARY

Sensitivity	Maximum Neutron Flux (% NBR)	Maximum Average Fuel Heat Flux (% NBR)	Maximum Pressure (Vessel Bottom) (psig)	Minimum Water Level (Ft. Below Separator Skirt)	Maximum Suppression Pool Bulk Temperature (°F)	Maximum Containment Pressure (psig)
MSIV Base Case	527 @ 4 sec	143 @ 5 sec	1296 @ 9 sec	11.59 @ 260 sec	186 @ 28 min	10.3 @ 28 min
MSIV Boron Delay - 90 Sec	0	0	0	-2.44	-16	-3.0
MSIV Boron Delay +120 Sec	0	0	0	+0.87	+12	+2.8
MSIV 50% RHR Capacity	0	0	0	0	+13	+3.1
MSIV 150% RHR Capacity	0	0	0	0	-2	-0.4
MSIV Delay RHR Start -2 Minutes	0	0	0	0	-1	-0.2
MSIV Delay RHR Start +5 Minutes	0	0	0	0	+2	+0.4
MSIV Service Water Temp (-20°F)	0	0	0	0	-18	-3.3
MSIV Service Water Temp (+20°F)	0	0	0	0	+19	+4.7

Table 3.1.4.1-1

BWR/4 MSIV ATWS SENSITIVITY RESULTS SUMMARY (Continued)

Sensitivity	Maximum Neutron Flux (% NBR)	Maximum Average Fuel Heat Flux (% NBR)	Maximum Pressure (Vessel Bottom) (psig)	Minimum Water Level (Ft. Below Separator Skirt	Maximum Suppression Pool Bulk Temperature (°F)	Maximum Containment Pressure (psig)
MSIV Suppression Pool Size (-20%)	0	0	0	0	+20	+5.0
MSIV Suppression Pool Size (+20%)	0	0	0	0	-13	-2.5
MSIV -50% Boron	0	0	0	+0.42	+49	+15.5
MSIV +50% Boron	0	0	0	-0.84	-16	-3.0
MSIV RPT Delay (+0.5 sec)	0	0	+11	-0.21	+5	+1.1
MSIV RPT Delay (+1 sec)	0	0	+24	-0.13	+5	+1.1
MSIV RPT Inertia (+50%)	0	0	+9	0	0	0
MSIV RPT Inertia (-20%)	0	0	-2	-0.59	+2	+0.4
MSIV S/RV Capacity (-20%)	0	0	+137	-0.86	+2	+0.4
MSIV S/RV Capacity (+20%)	0	0	-66	-0.93	-1	-0.2
MSIV Nominal HPCI Flow (-20%)	0	0	0	+0.04	-8	-1.6

Table 4.4.1-1

BWR/4 MSIV ATWS SENSITIVITY RESULTS SUMMARY (Continued)

Sensitivity	Maximum Neutron Flux (% NBR)	Maximum Average Fuel Heat Flux (% NBR)	Maximum Pressure (Vessel Bottom) (psig)	Minimum Water Level (Ft. Below Separator Skirt)	Maximum Suppression Pool Bulk Temperature (°F)	Maximum Containment Pressure (psig)
MSIV HPCI Flow (+20%)	0	0	0	-1.62	+12	+2.8
MSIV RHR Capacity (-50%) and Service Water (-20 °F)	0	0	0	0	-4	-0.8
MSIV RHR Capacity (+50%) and Service Water (+20 F)	0	0	0	0	+17	+4.1
MSIV Nominal Doppler (-.23¢/%)	+57	+3	+12	+1.40	-1	-0.2
MSIV Nominal Doppler (-.32¢/%)	-30	-2	-8	-0.42	+5	+1.1
MSIV -8 Void Coefficient (¢/%)	-96	-5	-16	-1.35	+1	+0.2
MSIV - 14 Void Coefficient (¢/%)	-4	+4	+11	+1.50	-1	-0.2

Table 3.1.4.1-2

BRW/4 MSIV ATWS NUCLEAR PARAMETRIC STUDY SUMMARY

		Change in Peak Value				
Doppler Coef ¢/°F	Void Coef ¢/%	Neutron Flux %	Average Heat Flux %	Vessel Bottom Pressure psi	Suppression Pool Temp °F	Min Level/ Time (Wide Range) ft/sec
-0.200	-8	-11	-0.9	+6		
-0.230	-8	-46	-2.5	-3		
-0.283	-8	-96	-5.1	-16	+0.6	-1.3/233
-0.320	-8	-124	-6.8	-23		
-0.200	-11	+93	+4.0	+20		
-0.230	-11	+57	+2.5	+12	-1.3	+1.4/326
-0.280+	-11+	570+	141.1+	1285+	186.2+	-11.6/261+
-0.320	-11	-30	-1.6	-8	+5.3	-0.4/243
-0.230	-14	+39	+6.5	+22		
-0.283	-14	-4	+4.2	+11	-0.6	+1.5/338

+ Values shown for nominal void and Doppler coefficients are absolute peaks. Other peaks are relative to these.

Table 3.1.4.2-1

BWR/4 TURBINE TRIP ATWS SENSITIVITY RESULTS SUMMARY*

Sensitivity	Maximum Neutron Flux (% NBR)	Average Fuel Heat Flux (% NBR)	Maximum Pressure (Vessel Bottom) (psig)	Minimum Water Level (Ft. Below Separator Skirt)	Maximum Suppression Pool Bulk Temperature (°F)	Maximum Containment Pressure (psig)
TT Base and Special Cases	402 @ .9 sec	133 @ 2.7 sec	1193 @ 2.5 sec	11.77 @ 406 sec	102 @ 4 min +60**	+0.6 @ 4 min +5.5**
TT Boron Delay (-90 sec)	0	0	0	-2.64	-2	-0.1
TT Boron (+30 sec)	0	0	0	+0.64	0	0
TT HPCI Capacity (-10%)	0	0	0	+1.03**	+56**	+5.0**
TT HPCI (+25%)	0	0	0	-3.02	+2	+0.1
TT Nominal Doppler (-.20¢/%)	+50	+3	-4	-2.50	-2	-0.1
TT Nominal Doppler (-32¢/%)	-17	-1	-1	+0.62	0	0
TT -8 Void Coeff.	-105	-4	-4	-0.53	+2	+0.1
TT -14 Void Coeff.	+136	+4	+4	-4.38	-2	-0.1

*Neglecting MSIV closure unless otherwise noted.
**With MSIV closure at Level 1

Table 3.1.4.2-2

BWR/4 TURBINE TRAP ATWS NUCLEAR PARAMETRIC STUDY SUMMARY*

Doppler Coef ¢/°F	Void Coef ¢/%	Neutron Flux %	Average Heat Flux %	Vessel Bottom Pressure psi	Suppression Pool Temp °F	Min Level/ Time (Wide Range) ft/sec
		Change in Peak Value				
-0.200	-6	-136	-5.0	-4		
-0.230	-6	-144	-6.5	-5**		
-0.283	-6	-156	-8.2	-7**		
-0.320	-6	-163	-9.4	-8**		
-0.200	-8	-77	-1.3	-1		
-0.230	-8	-88	-2.5	-2		
-0.283	-8	-105	-4.4	-4	+2.2	+0.5/260
-0.320	-8	-115	-5.6	-5**		
-0.200	-11	+50	+3.2	+4	-1.6	+2.5/385
-0.230	-11	+30	+2.0	+3		
-0.280+	-11	399+	132.0+	1193+	102.0+	-11.8/406+
-0.320	-11	-17	-1.3	-1	+0.0	-0.6/416
-0.200	-14	+216	+6.7	+8		
-0.230	-14	+183	+5.6	+7		
-0.283	-14	+136	+3.6	+4	1.7	+4.4/375
-0.320	-14	+109	+2.4	+3		

+Values shown for nominal void and Doppler coefficients are absolute peaks. Other peaks are relative to these.

*Neglecting MSIV closure at Level 1.

**These cases did not reach high pressure trip setpoint (analytical upper limit = 1150 psig dome pressure). See Section 3.2.4.2.5 for a discussion of this situation.

Table 3.1.4.2.7-1

NORMALIZED RESULTS BWR/4 TURBINE TRIP LIMIT CYCLE CHARACTERISTICS*

Parameter Varied	Change (%)	Neutron Flux Max/Min	Average Surface Heat Flux Max/Min	Net Reactivity Max/Min	Vessel Pressure Max/Min	Vessel Level Max/Min	Limit Cycle Duration (Normalized)	Average Neutron Flux
HPCI Capacity	+25	2.17/0.65	1.45/1.15	0.55/-0.96	0.98/0.97	1.36/1.67	0.25	1.29
(19.8% of FW)	-10	25.9/0.11	2.31/0159	1.15/-9.34	1.0/0.95	1.04/2.27	1.07	0.89
Doppler Coefficient	+14.3	29.7/0.1	1.96/0.59	0.93/-8.47	1.0/9.97	1.04/2.14	1.03	1.0
(-0.28¢/°F)	-28.6	3.75/0.24	1.42/0.75	0.77/-2.72	0.99/0.97	0.96/1.60	0.81	1.0
Void Coefficient	+27.3	6.7/0.13	1.68/0.63	0.95/-6.14	0.99/0.97	0.36/1.85	1.24	0.94
(-$.11/%)	-18.2	4.56/0.12	1.71/0165	0.93/-6.63	1.0/0.97	1.17/2.14	0.88	1.01
Boron Timer Delay	+25 (150 sec)	21.1/0.11	1.95/0.65	1.0/-8.07	1.0/0.97	1.03/2.15	1.04	1.0
(2 Min)	-75 (30 sec)	2.98/0.3	1.35/0.81	0.72/-2.04	0.99/0.97	1.08/1.58	0.66	1.0
Boron Mixing Efficiency and/or Capacity	+30	5.27/0.13	1.55/0.57	0.89/-5.42	0.99/0.97	0.36/0.96	0.96	0.91
86 gpm @ (95%)	-10.5	10.32/0.15	1.81/0.69	1.0/-6.24	1.0/0.97	0.97/2.10	1.15	1.01

*Neglecting MSIV closure at Level 1

Table 3.4.2.7-2

NORMALIZATION FACTORS CHOSEN FROM SPECIAL CASE

Parameter	Special Case Value	Average Power During Limit Cycle (% NBR)	Net Reactivity ($)	Duration** of Limit Cycle (sec)
HPCI + RCIC*	19.8% NBR FW	22.9	0.78	454
Doppler	-0.28¢/°F	23.1	0.90	456
Void	-11¢/%	23.1	0.90	456
Boron Delay	2 Min	23.1	0.90	456
Boron Capacity	86 GPM	23.1	0.90	456
Boron Mixing	95%	23.1	0.90	456

*Special case for this sensitivity study used -0.23¢/°F Doppler Coefficient.
**From beginning of significant cycles until elimination of S/RV cycling (≥ ±5%)

Table 3.1.4.2.7-3

EFFECT OF DIFFERENT SHAPES OF LIMIT CYCLE ON FUEL CONDITIONS

F (Fraction of Limit Cycle Where Power >P_{ave})	Maximum Power During Limit Cycle (% of NBR) [Normalized to Ave. Power]	Fuel Clad Temperature During the Cycle (Peak to Peak) (Cycle Period = 4 Seconds)	Fuel Clad Temperature Variation During the Cycle (Peak to Peak) (Cycle Period = 8 Seconds)
0.5	50%[2.0]	32°F	67°F
0.3	83.3%[3.33]	47°F	109°F
0.2	125%[5]	50°F	115°F
0.1	250%[10]	57°F	126°F
0.05	500%[20]	60°F	130°F

Table 3.1.5-1

BWR/4 LOSS OF CONDENSER VACUUM

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Main Turbine (and feedwater turbines)*Trip due to low Condenser Vacuum, Bypass opens - All Normal scrams fail.	0 Seconds	0 Seconds
2.	Pressure and power rise begins	0 Seconds	0 Seconds
3.	Peak power occurs	1 Second	1 Second
4.	Relief Valves Lift	2 Seconds	2 Seconds
5.	ATWS High Pressure Setpoint (1150 psig) is reached - Recirculations pumps tripped - ARI is initiated - SLCS Timed Logic activated	2 Seconds	2 Seconds
6.	Some fuel may experience boiling transition	2 Seconds	2 Seconds
7.	Peak Vessel pressure occurs	3 Seconds	3 Seconds
8.	ARI Control Rod Insertion	22 Seconds	Fails
9.	ATWS logic timer completed - Initiates feedwater flow limit	N/A	27 Seconds
10.	MSIV's and bypass close due to low condenser vacuum	30 Seconds	30 Seconds
11.	Reactor water level drops to level 2 - Initiates containment isolation - HPCI and RCIC start	44 Seconds	60 Seconds

*Sequence conservatively assumes motor driven feedwater pumps.

Table 3.1.5-1 (Continued)

		With ARI	With ARI Failure
12.	HPCI and RCIC flow begins	64 Seconds	80 Seconds
13.	Final ATWS logic times completed - Initiates SLCS	N/A	2 Minutes
14.	Liquid control flow reaches core	N/A	3 Minutes
15.	Reactor water level reaches minimum and begins to rise	64 Seconds	5 Minutes
16.	RHR flow begins (pool cooling)	$\geq$11 Minutes	11 Minutes
17.	Hot shutdown achieved	22 Seconds	18 Minutes
18.	Containment temperature and pressure peaks occur		27 Minutes

Table 3.1.5-2

BWR/4 LOSS OF CONDENSER VACUUM - SUMMARY

With ARI Failure	86 GPM Boron 95% Mixing Eff 2 Min Logic Delay Loss of Condenser Vacuum
Maximum Neutron Flux (%)	403
Maximum Vessel Bottom Pressure (psig)	1195
Maximum Average Heat Flux (%)	133
Maximum-Bulk Suppression Pool Temperature (°F)	188
Associated Containment Pressure (psig)	10.7

Table 3.1.7-1

BWR/4 - LOSS OF FEEDWATER HEATER

	Sequence of Events	Time
1.	Inadvertent tripping of feedwater heaters; feedwater enthalpy begins to drop	0 Seconds
2.	Reactor and turbine-generator power begins to rise	2 Seconds
3.	APRM high power alarm (108%), operator attempts to insert rods	34 Seconds
4.	Vessel pressure levels off after a small increase	56 Seconds
5.	Power levels off below the scram setpoint(s)	72 Seconds
6.	Manual scram attempted after control rod insertion attempts have failed	≤10-1/2 Minutes
7.	Manual ARI and SLCS initiated after manual scram fails	≤10-1/2 Minutes
8.	ARI control rod insertion completed, eliminating SLCS initiation, and achieving reactor shutdown	≤11 Minutes
9.	Final ATWS logic timer completed - Initiates SLCS (if ARI has failed)	≤12-1/2 Minutes
10.	Liquid control reaches core (if ARI has failed)	≤14 Minutes
11.	Hot shutdown achieved (if ARI has failed)	≤35 Minutes

Table 3.1.7-2

BWR/4 LOSS OF FEEDWATER HEATER - SUMMARY

With ARI Failure	86 GPM Boron 95% Mixing Eff 2 Min + Logic Delay Loss of Feedwater Heater
Maximum Neutron Flux (%)	113
Maximum Vessel Bottom Pressure (psig)	1046
Maximum Average Heat Flux (%)	112
Maximum Bulk Suppression Pool Temperature (°F)	90
Associated Containment Pressure (psig)	no change

Table 3.1.8-1

BWR/4 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND

	Sequence of Events	Time: With ARI	Time: With ARI Failure
1.	Feedwater controller fails to maximum demand. Reactor water level begins to rise, as well as a gradual power increase.	0	0
2.	High water level (Level 8) setpoint is reached - Turbine trips, bypass opens - All normal scram assumed to fail - Feedwater pumps trip	6 Seconds	6 Seconds
3.	Pressure and power rise begins	6 Seconds	6 Seconds
4.	Some fuel experiences boiling transition	7 Seconds	7 Seconds
5.	Relief valves lift	8 Seconds	8 Seconds
6.	ATWS high pressure setpoint is reached - Recirculation pumps are tripped* - ARI is initiated - SLCS timed logic is activated	8 Seconds	8 Seconds
7.	Maximum vessel pressure occurs	9 Seconds	9 Seconds
8.	ARI control rod insertion completed, eliminating SLCS initiation	28 Seconds	Fails
9.	ATWS logic limits feedwater	N/A	33 Seconds
10.	Lowest setpoint S/RV closes and stays closed. Steam flow to suppression pool stops and peak containment and suppression pool conditions occur.	30 Seconds	37 Seconds

Table 3.1.8-1 (Continued)

	Sequence of Events	With ARI	With ARI Failure
11.	Reactor water level drops to Level 2 - Initiates containment isolation - Initiates HPCI and RCIC	49	41 Seconds
12.	HPCI and RCIC flow begins	69	61 Seconds
13.	ATWS logic timer completed - initiates SLCS	N/A	2 Minutes
14.	Liquid control flow reaches core	N/A	3 Minutes
15.	Reactor water level reaches minimum and begins to rise	N/A	7 Minutes
16.	RHR flow begins (pool cooling)	$\geq$11 Minutes	11 Minutes
17.	Reactor water level is restored shutting off and starting automatic cycling of the HPCI and RCIC between L2 and L8 (neglecting preferred operator action to manually control flow and level).	N/A	16 Minutes
18.	Hot shutdown achieved	28 Seconds	17 Minutes

*Direct recirculation pump trip from turbine stop valve closure was conservatively neglected.

Table 3.1.8-2

BWR/4 FEEDWATER CONTROLLER FAILURE (MAXIMUM DEMAND) - SUMMARY

With ARI Failure	86 GPM Boron 95% Mixing Eff 2 Min Logic Delay Feedwater Controller Failure
Maximum Neutron Flux (%)	511
Maximum Vessel Bottom Pressure (psig)	1195
Maximum Average Heat Flux (%)	137
Maximum Bulk Suppression Pool Temperature (°F)	99
Associated Containment Pressure (psig)	0.5

Table 3.1.9-1

BWR/4 PRESSURE REGULATOR FAILURE (MAXIMUM DEMAND)

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Pressure regulator to maximum demand	0	0
2.	Pressure and power begin to fall	0	0
3.	Low steamline pressure isolation setpoint reached		
	- MSIV closure	19 Seconds	19 Seconds
	- Scram normally initiated (assumed to fail)	20 Seconds	20 Seconds
4.	Pressure and power begin to rise	22 Seconds	22 Seconds
5.	Relief valves lift	26 Seconds	26 Seconds
6.	Some fuel experiences boiling transition	26 Seconds	26 Seconds
7.	ATWS high pressure setpoint is reached (1150 psig)		
	- Recirculation pumps are tripped	27 Seconds	27 Seconds
	- ARI is initiated		
	- SLCS and feedwater limit timed logic is activated		
8.	Vessel pressure peaks	31 Seconds	31 Seconds
9.	ARI control rod insertion completed	47 Seconds	Assumed to fail
10.	ATWS logic timer completed - Initiates FW limit	N/A	52 Seconds
11.	Feedwater flow runs back to lower limit value	N/A	67 Seconds
12.	Reactor water level drops to Level 2 - Initiates containment isolation - Initiates HPCI and RCIC	66 Seconds	73 Seconds
13.	HPCI and RCIC flow begins	86 Seconds	93 Seconds
14.	Final ATWS logic timer completed - Initiates SLCS	N/A	2½ Minutes
15.	Liquid control flow reaches core	N/A	3½ Minutes
16.	Reactor water level reaches minimum and begins to rise	91 Seconds	4 Minutes
17.	RHR flow begins (pool cooling)	≥11 Minutes	11 Minutes
18.	Hot shutdown achieved	47 Seconds	18 Minutes
19.	Containment temperature and pressure peak	N/A	27 Minutes

Table 3.1.9-2

BWR/4 PRESSURE REGULATOR FAILURE (MAXIMUM DEMAND) - SUMMARY

With ARI Failure	86 GPM Boron 95% Mixing Eff 2 Min Logic Delay Pressure Regulator Failure
Maximum Neutron Flux (%)	585
Maximum Vessel Bottom Pressure (psig)	1280
Maximum Average Heat Flux (%)	139
Maximum Suppression Pool Temperature (°F)	189
Associated Containment Pressure (psig)	11.0

Table 3.1.10-1

BWR/4 LOSS OF FEEDWATER

		Time	
	Sequence of Events	With ARI	With ARI Failure
1.	Feedwater flow stops (flow assumed to reduce to zero in 5 seconds) - all normal scrams assumed to fail	0	0
2.	Pressure, water level and power starts to decline	0	0
3.	Reactor water level drops to Level 2 and trips recirculation pumps*, initiates ARI and also initiates RCIC and HPCI. SLCS timed logic is also activated.	19 Seconds	19 Seconds
4.	ARI control rod insertion completed. Eliminating SLCS	39 Seconds	Fails
5.	HPCI and RCIC flow starts	39 Seconds	39 Seconds
6.	ATWS logic timer completed - initiates SLCS	N/A	2 Minutes
7.	Liquid control flow reaches the core	N/A	3 Minutes
8.	Water level reaches minimum and begins to rise. The top of the core always remains covered.	61 Seconds	5 Minutes
9.	High water level trip of HPCI and RCIC (neglecting preferred operator action to manually control level)	~10 Minutes	16 Minutes
10.	Hot shutdown achieved	39 Seconds	17 Minutes

*Trip of recirculation drive motor breakers. Recirculation runback (from low level alarm, L4, and coincident low feedwater flow) is conservatively neglected.

Table 3.1.10-2

BWR/4 LOSS OF FEEDWATER SUMMARY

With ARI Failure	86 GPM Boron 95% Mixing Eff 2 Min Logic Delay Loss of Feedwater
Maximum Neutron Flux (%)	100
Maximum Vessel Bottom Pressure (psig)	1044 (no increase)
Maximum Average Heat Flux (%)	100
Maximum Bulk Suppression Pool Temp Temperature (°F)	90
Associated Containment Pressure (psig)	No change

Table 3.1.11-1

BWR/4 LOSS OF NORMAL AC POWER

	Sequence of Events	Time: With ARI	Time: With ARI Failure
1.	Loss of all auxiliary power transformers - Recirculation pumps trip - Condensate and feedwater pumps trip	0	0
2.	Pressure and power begin to fall	0	0
3.	Normal scram due to loss of AC Power (Assumed to fail)	2 Seconds	2 Seconds
4.	MSIV's start to close due to loss of AC power (and initiate scram - also assumed to fail)	2 Seconds	2 Seconds
5.	Pressure and power begin to rise	5 Seconds	5 Seconds
6.	SRV valves lift at relief setpoints	7 Seconds	7 Seconds
7.	ATWS high pressure setpoint is reached (1150 psig) - ARI is initiated - SLCS timed logic is activated	8 Seconds	8 Seconds
8.	Vessel pressure and power peak	8 Seconds	8 Seconds
9.	Some fuel experiences boiling transition	8 Seconds	8 Seconds
10.	ARI control rod insertion completed, eliminating SLCS initiation	28 Seconds	Fails
11.	Reactor water level drops to Level 2 - Initiates containment isolation - Initiates HPCI and RCIC	34 Seconds	36 Seconds
12.	HPCI and RCIC flow begins	54 Seconds	56 Seconds
13.	ATWS logic timer completed - initiates SLCS	N/A	2 Minutes
14.	Liquid control flow reaches core	N/A	3 Minutes
15.	Reactor water level reaches minimum and begins to rise. Level inside the core shroud remains above the top of active fuel.	85 Seconds	5 Minutes
16.	RHR flow begins (pool cooling)	≥11 Minutes	11 Minutes
17.	Hot shutdown achieved	28 Seconds	17 Minutes
18.	Containment temperature and pressure peak		33 Minutes

Table 3.1.11-2

BWR/4 LOSS OF AC POWER - SUMMARY

With ARI Failure	86 GPM Boron 95% Mixing Eff 2 Min Logic Delay Loss of Normal AC Power
Maximum Neutron Flux (%)	258
Maximum Vessel Bottom Pressure (psig)	1172
Maximum Average Heat Flux (%)	100
Maximum Bulk Suppression Pool Temperature (°F)	182
Associated Containment Pressure (psig)	9.5

Table 3.1.12-1

BWR/4 RECIRCULATION FLOW CONTROLLER FAILURE - INCREASING FLOW

Sequence of Events	Time
1. Flow controller fails	0 Seconds
2. Neutron flux reaches 120%, APRM scram assured to fail	2 Seconds
3. Power peaks	3 Seconds
4. Generator and pump reach maximum speed	4 Seconds
5. Maximum fuel surface heat flux occurs	6 Seconds
6. Vessel pressure peaks	6 Seconds
7. Core flow increase levels off	10 Seconds
8. New core equilibrium conditions (All parameters within normal limits, power and feedflow slowly decreasing as steady state feedwater heating is established)	20 Seconds
9. Manual scram or (if this fails) automatic ARI and SLCS initiation	10 Minutes
10. Hot shutdown achieved	30 Minutes

Table 3.1.12-2

BWR/4 RECIRCULATION FLOW CONTROLLER FAILURE
(INCREASING FLOW) - SUMMARY

With ARI Failure	86 GPM Boron 95% Mixing Eff 2 Min Logic Delay Recirculation Flow Controller Failure
Maximum Neutron Flux (%)	520
Maximum Vessel Bottom Pressure (psig)	1020
Maximum Average Heat Flux (%)	92
Maximum Bulk Suppression Pool Temperature (°F)	90
Associated Containment Pressure (psig)	No change

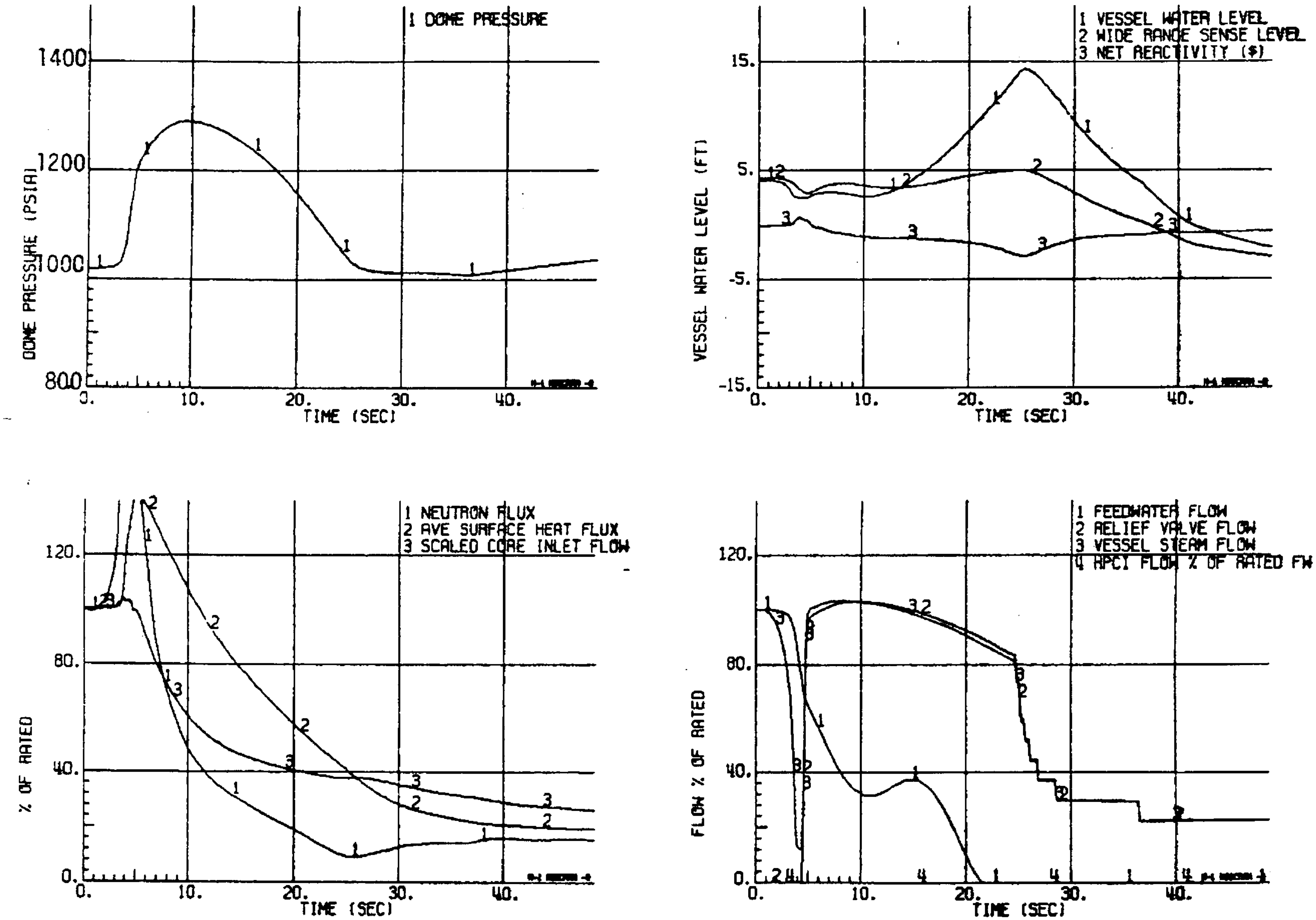


Figure 3.1.1-1. MSIV Closure with ARI Failure

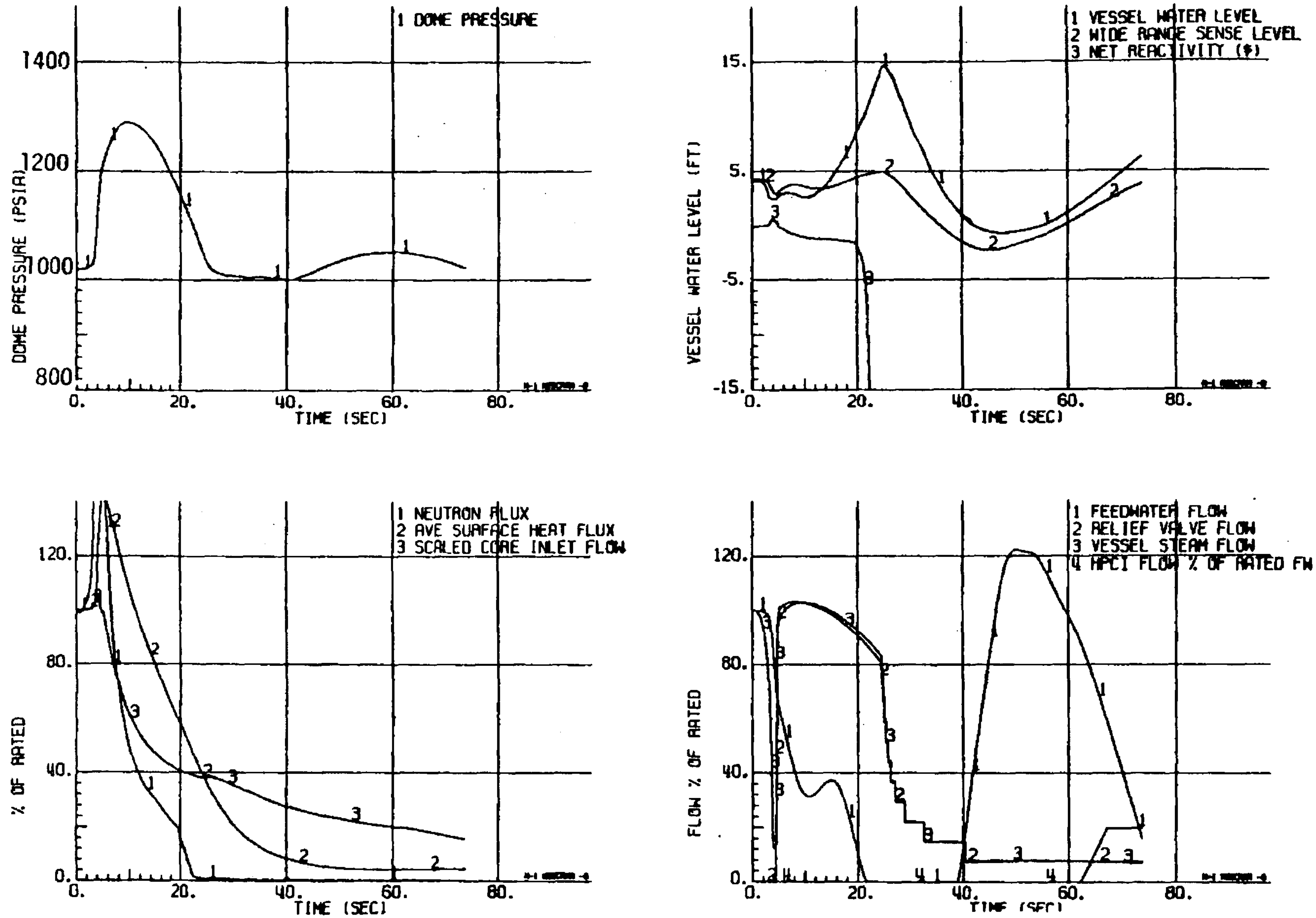


Figure 3.1.1-2. MSIV Closure with ARI

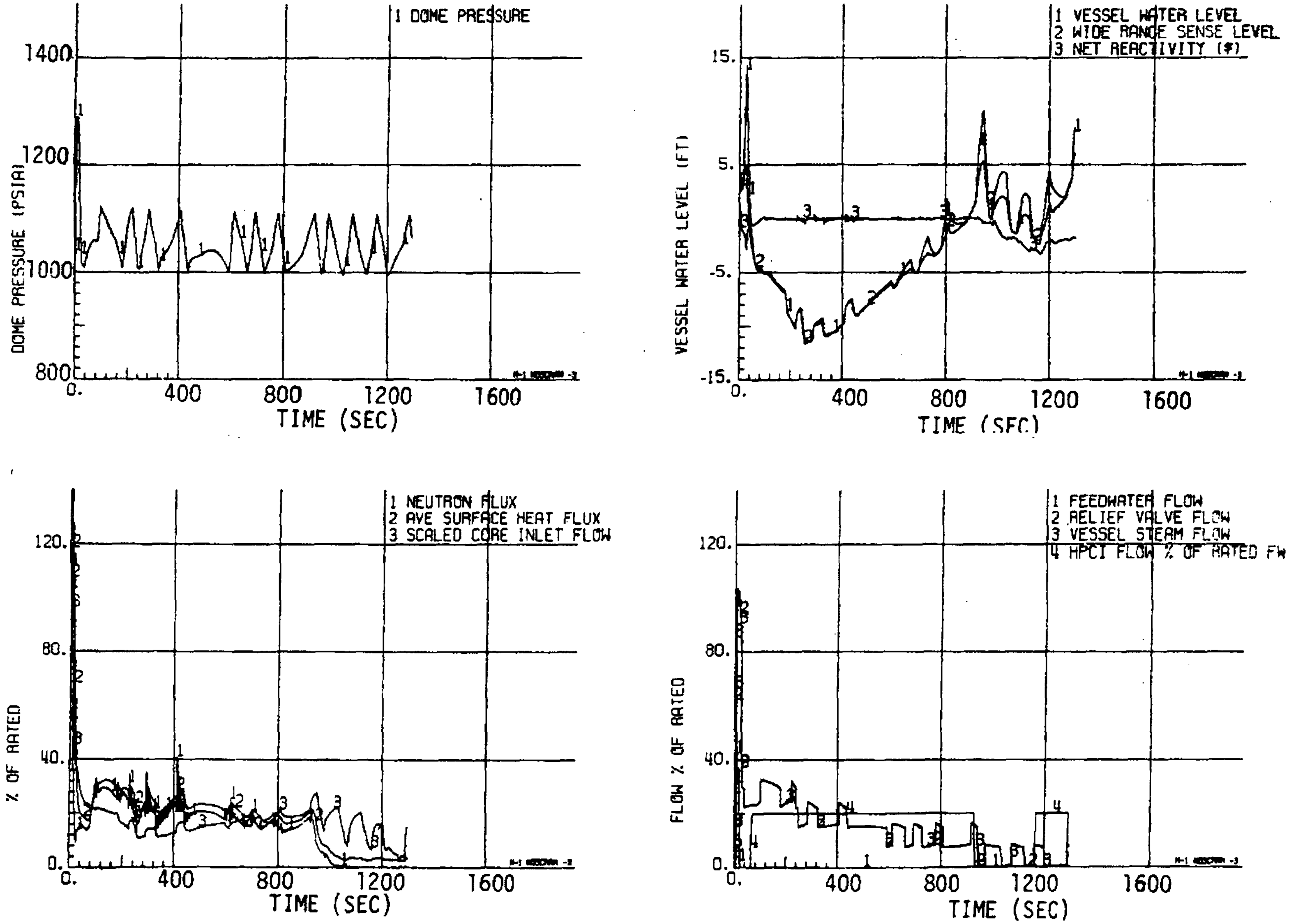


Figure 3.1.1-3. MSIV Closure with ARI Failure, 2 Minute 95% Boron Mix Eff

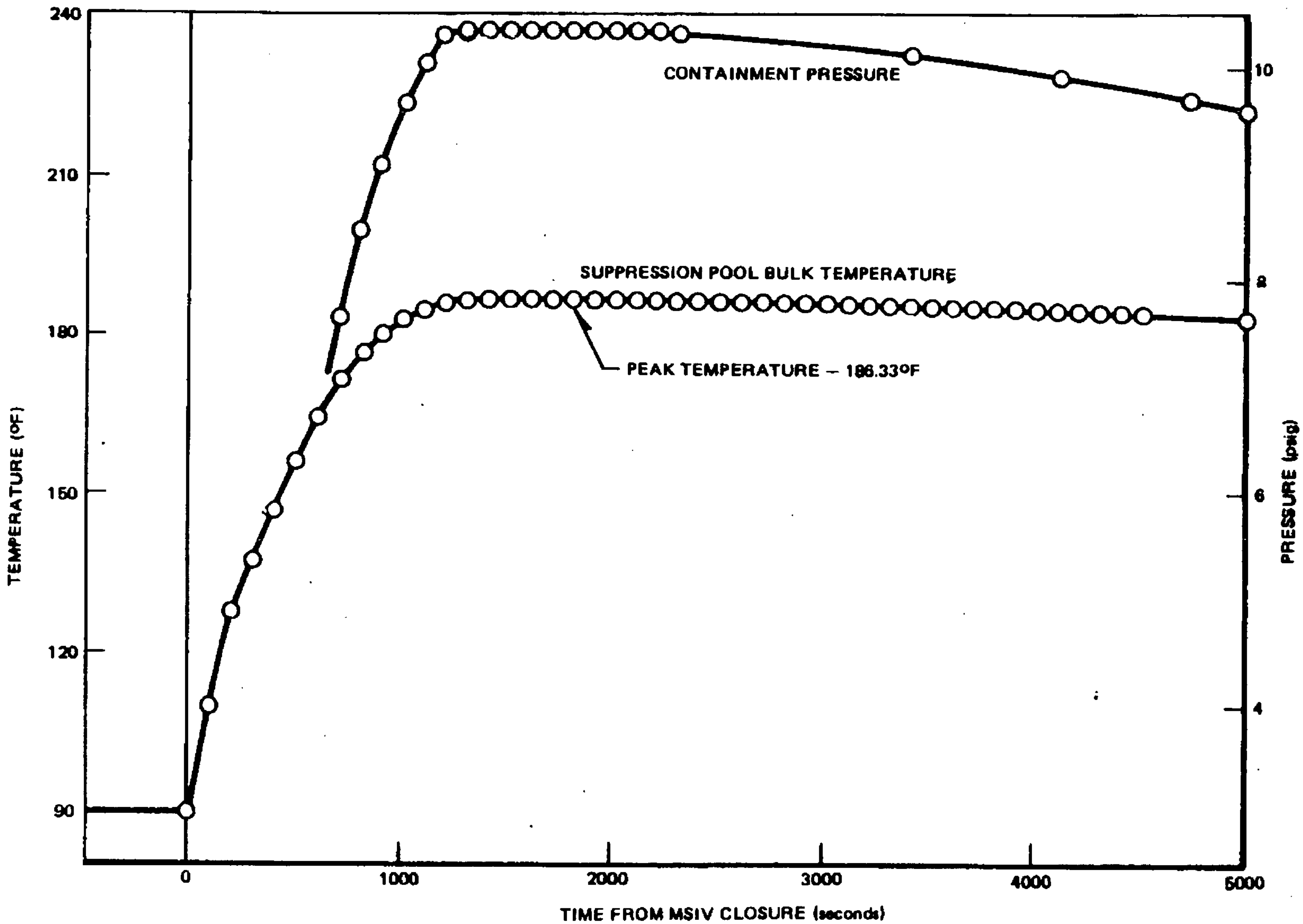


Figure 3.1.1-4. BWR-4 ATWS with AR1 Failure MSIV Base Case: Behavior of Suppression Pool Bulk Temperature and Containment Pressure Over Time

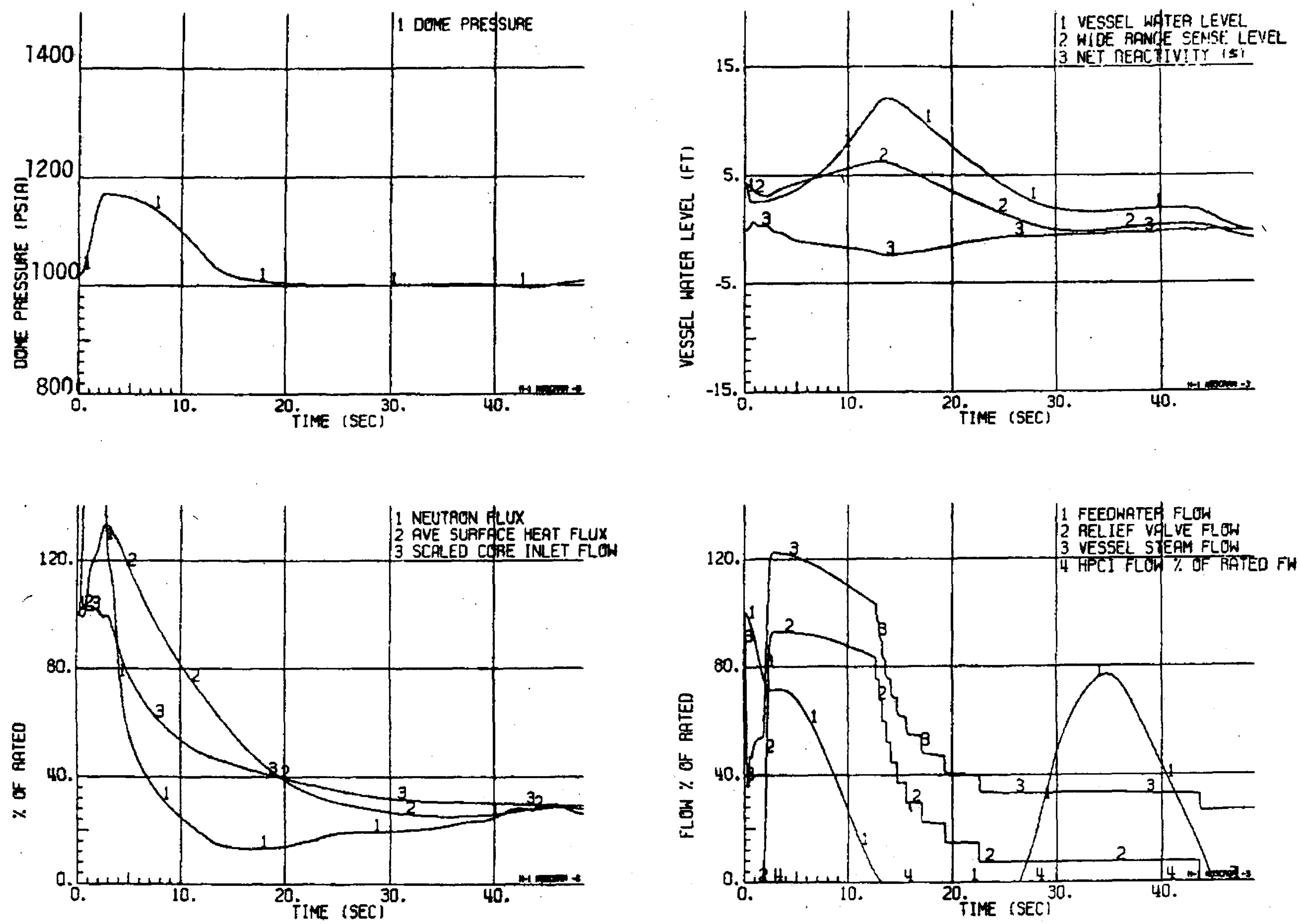


Figure 3.1.2-1. Turbine Trip - ARI Failure

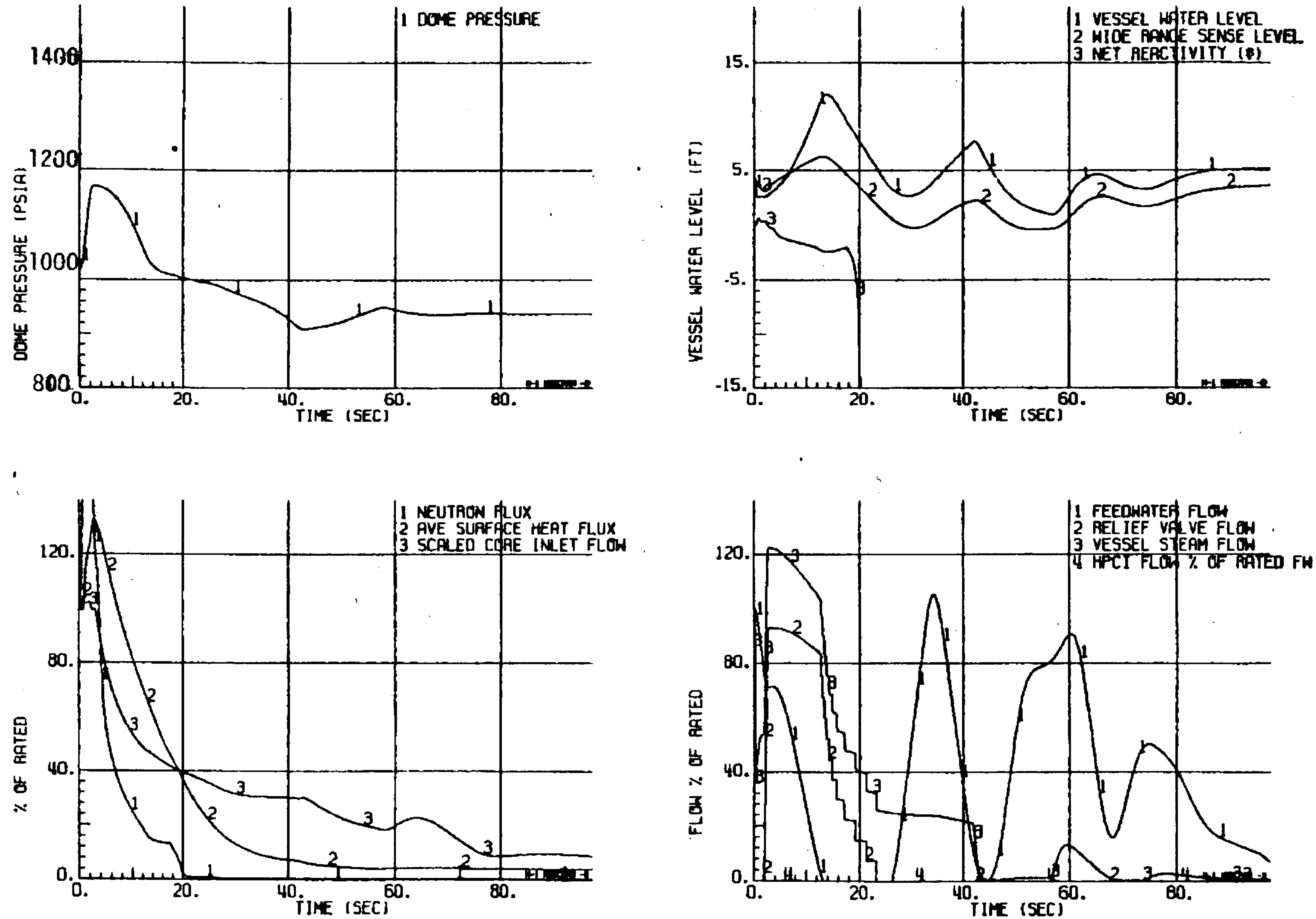


Figure 3.1.2-2. Turbine Trip with ARI

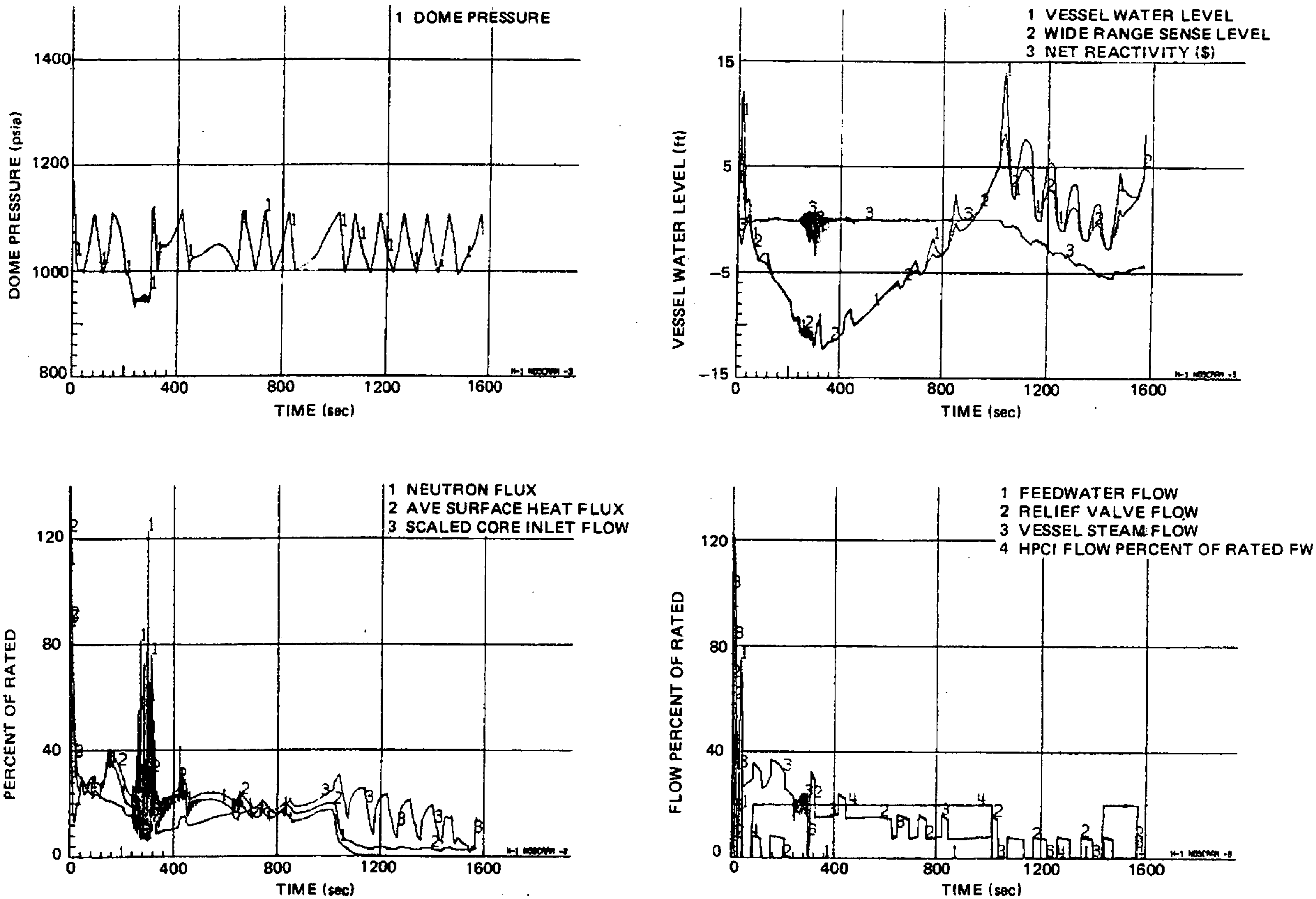


Figure 3.1.2-3. Turbine Trip, ARI Failure, 2 Minute 95% Boron Mix Eff, MSIV Level 1

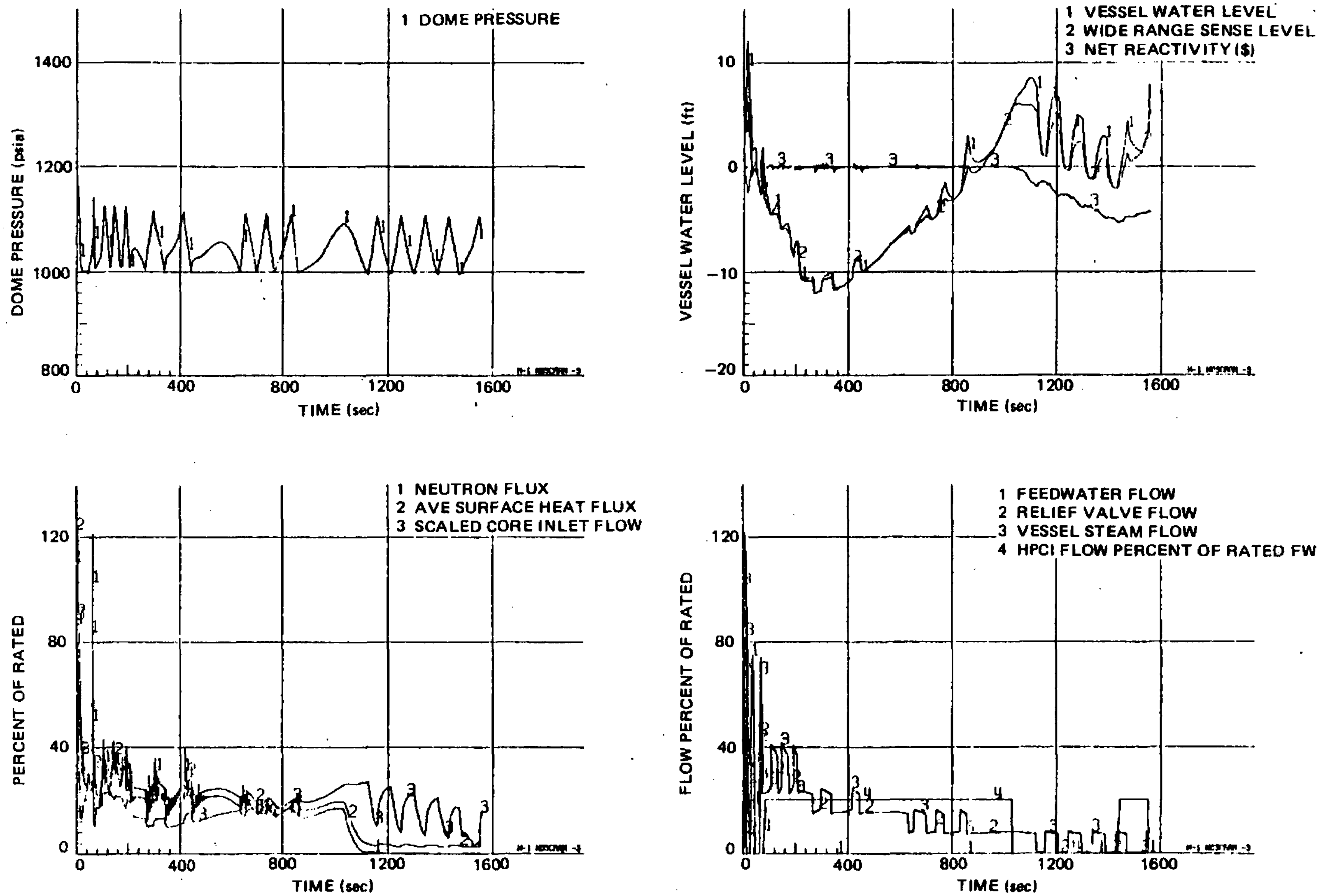


Figure 3.1.2-4. Turbine Trip, ARI Failure, 2 Minute 95% Boron Mix EFF, MSIV Level 2

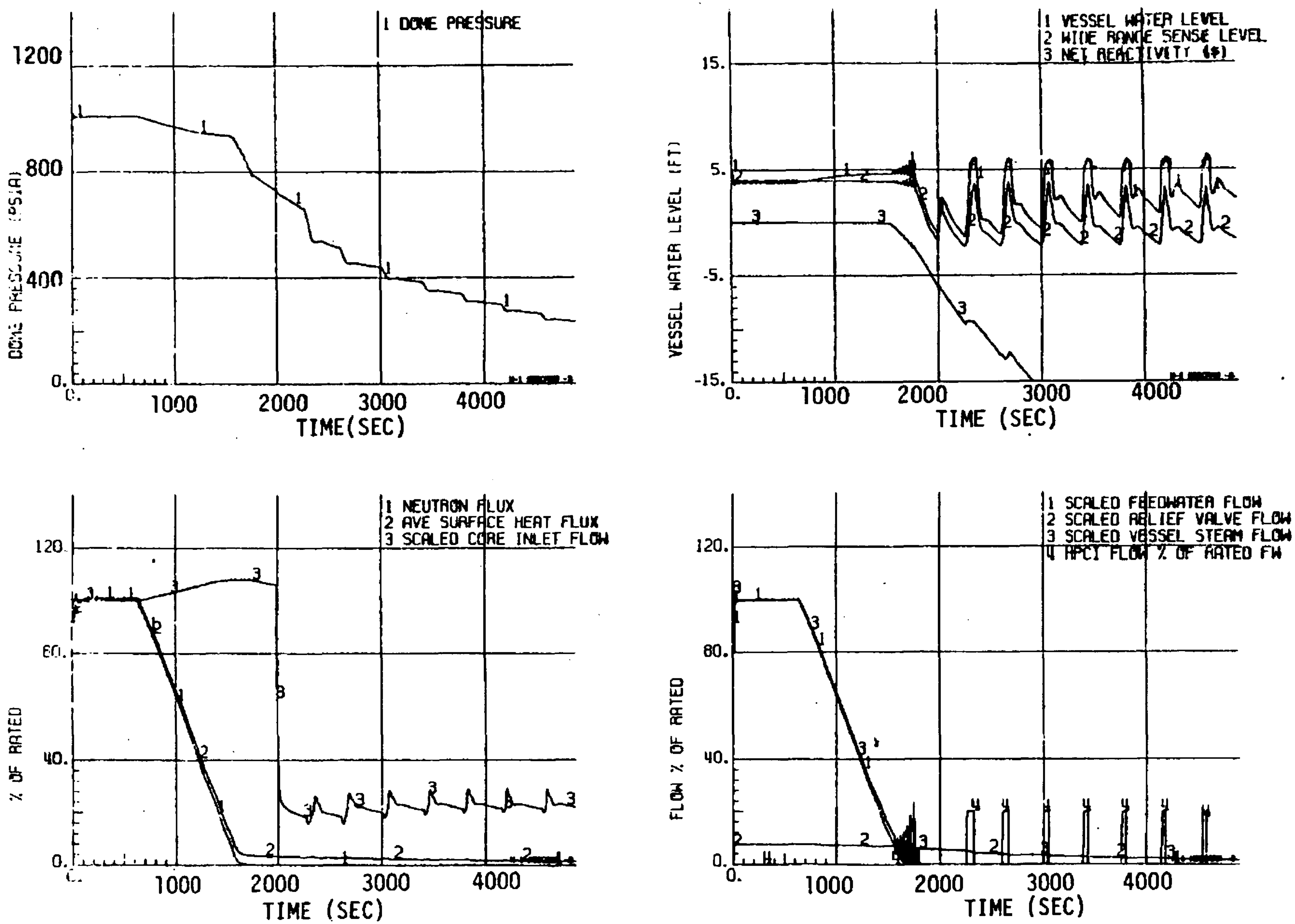


Figure 3.1.3-1. IORV with ARI Failure, 2 Minute 95% Boron Mix Eff

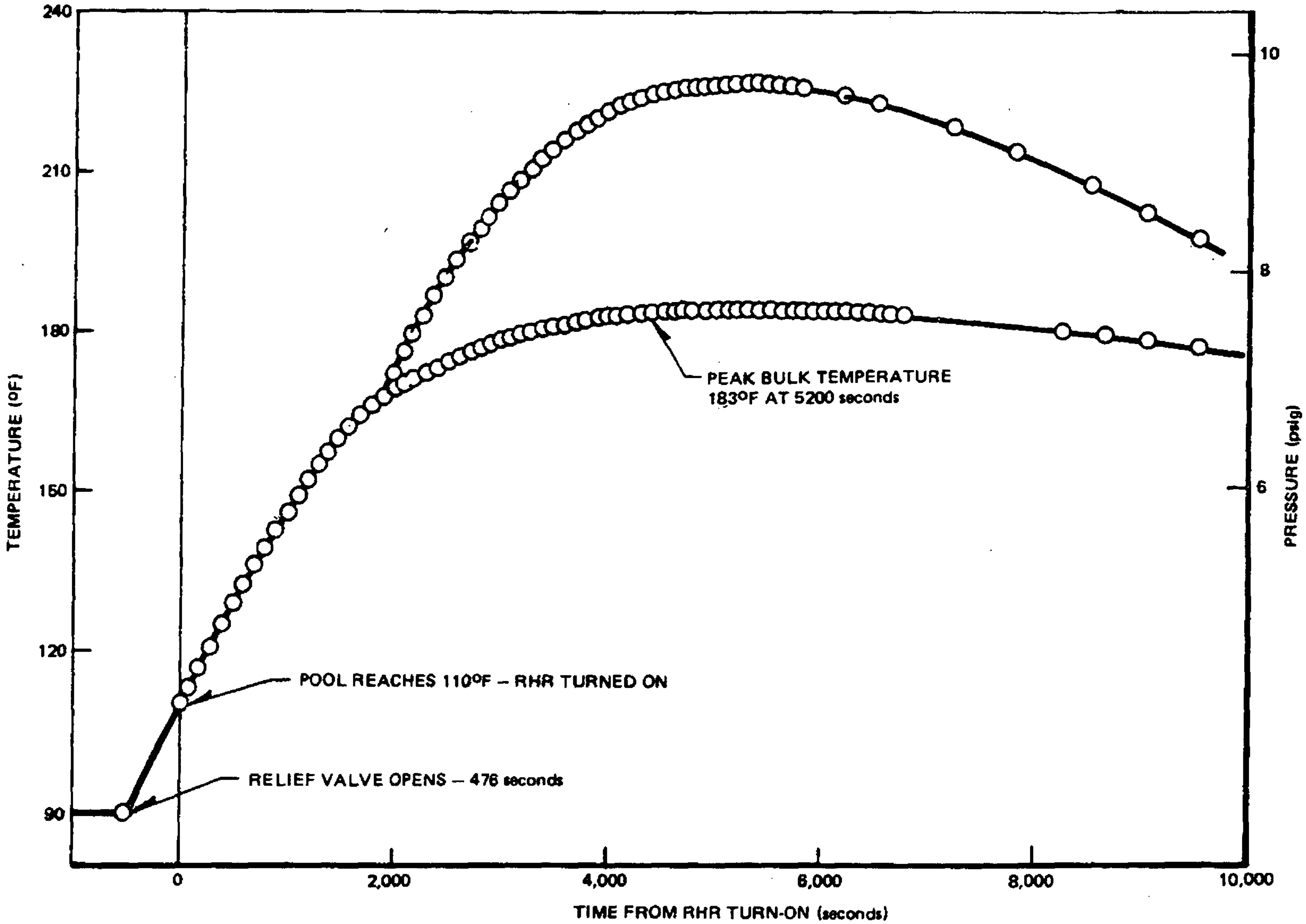


Figure 3.1.3-2. BWR-4 ATWS with ARI Failure IORV Base Case: Behavior of Suppression Pool Bulk Temperature and Containment Pressure Over Time

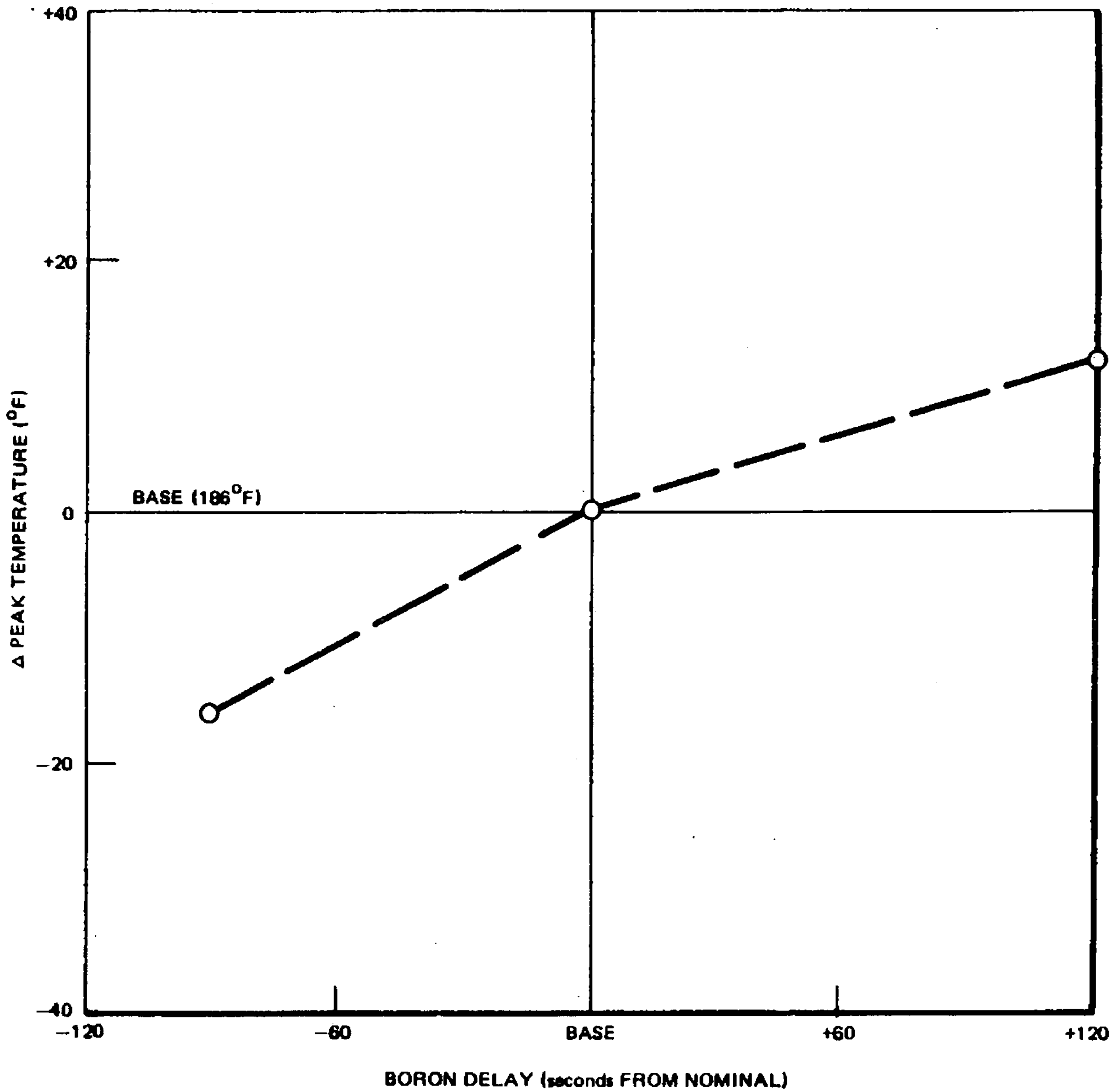


Figure 3.1.4.1-1. BWR-4 ATWS with ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to Delay in Starting Liquid Boron

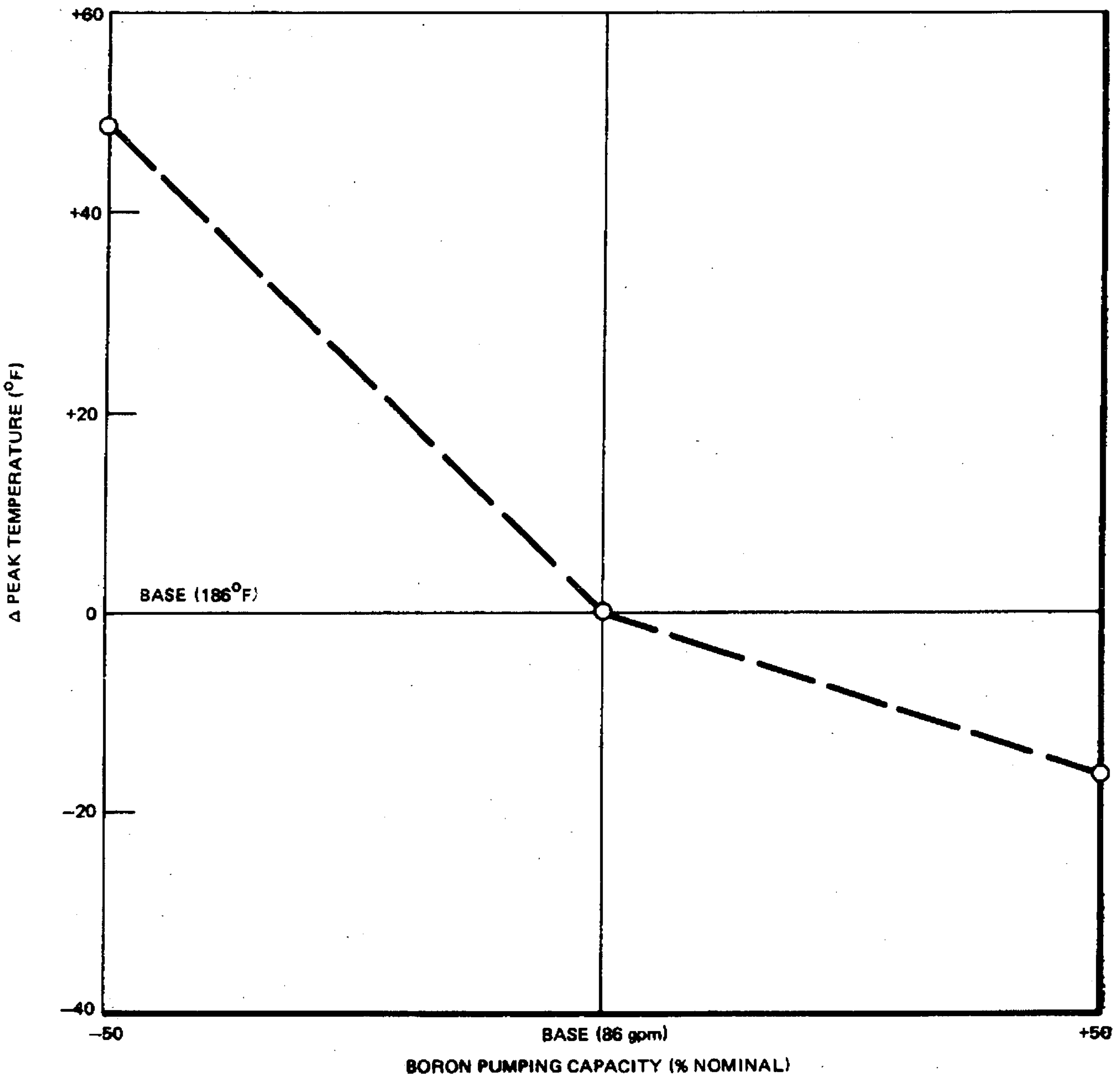


Figure 3.1.4.1.2. BWR-4 ATWS with ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to Boron Pumping Capacity

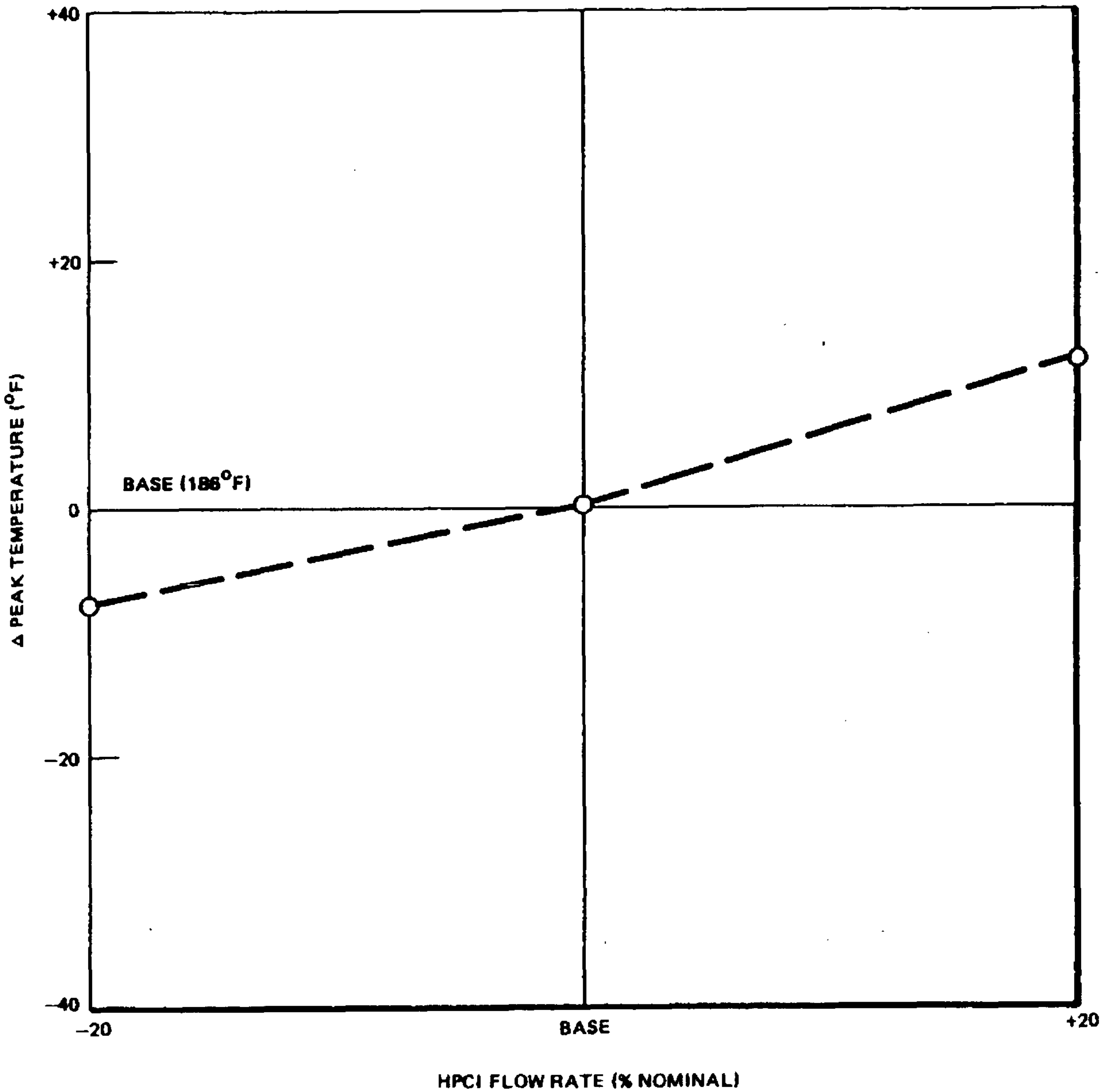


Figure 3.1.4.1.3. BWR-4 ATWS with ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to HPCI/RCIC Capacity

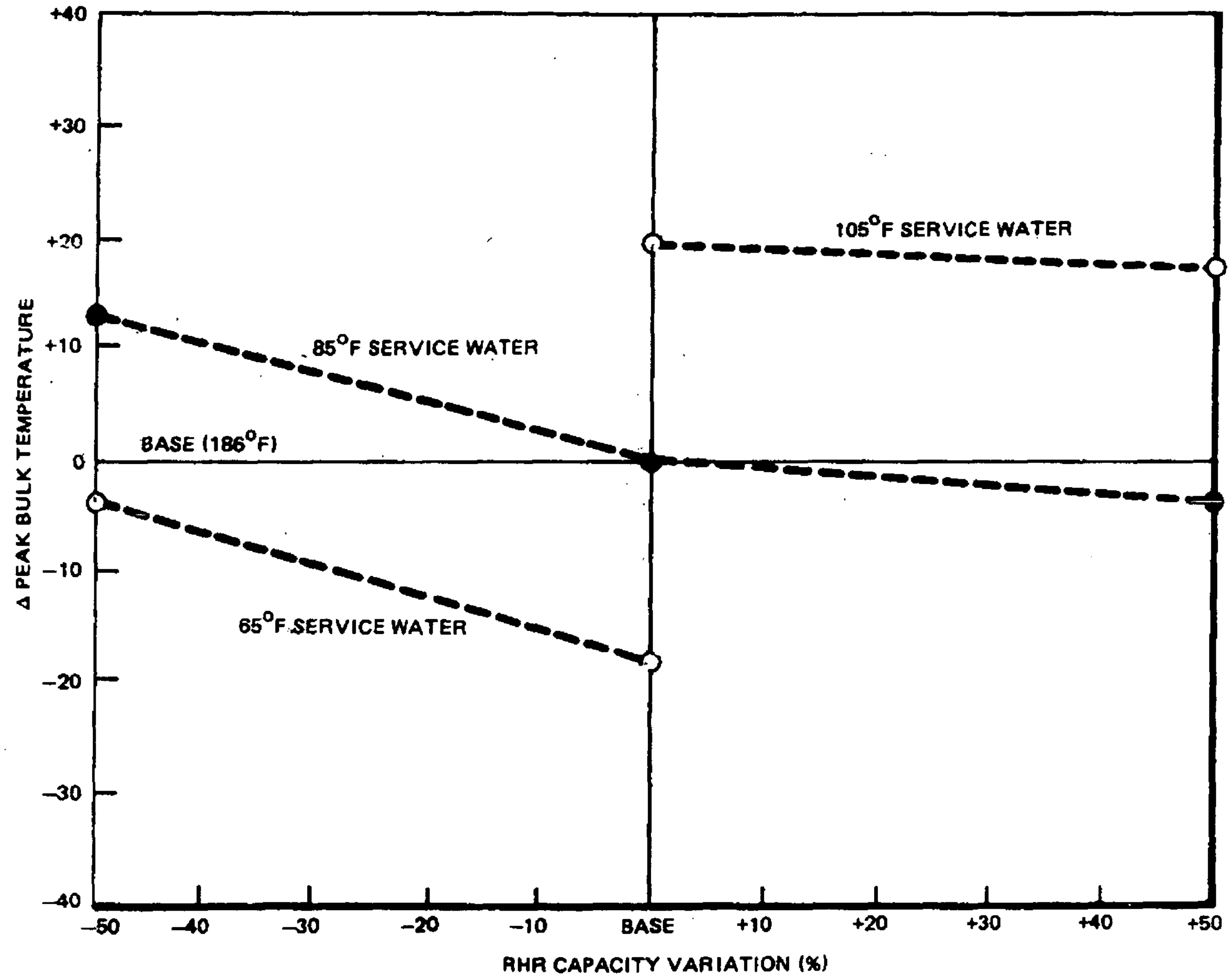


Figure 3.1.4.1.4. BWR-4 ATWS MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to a Change in the Size of the RHR Heat Exchanger

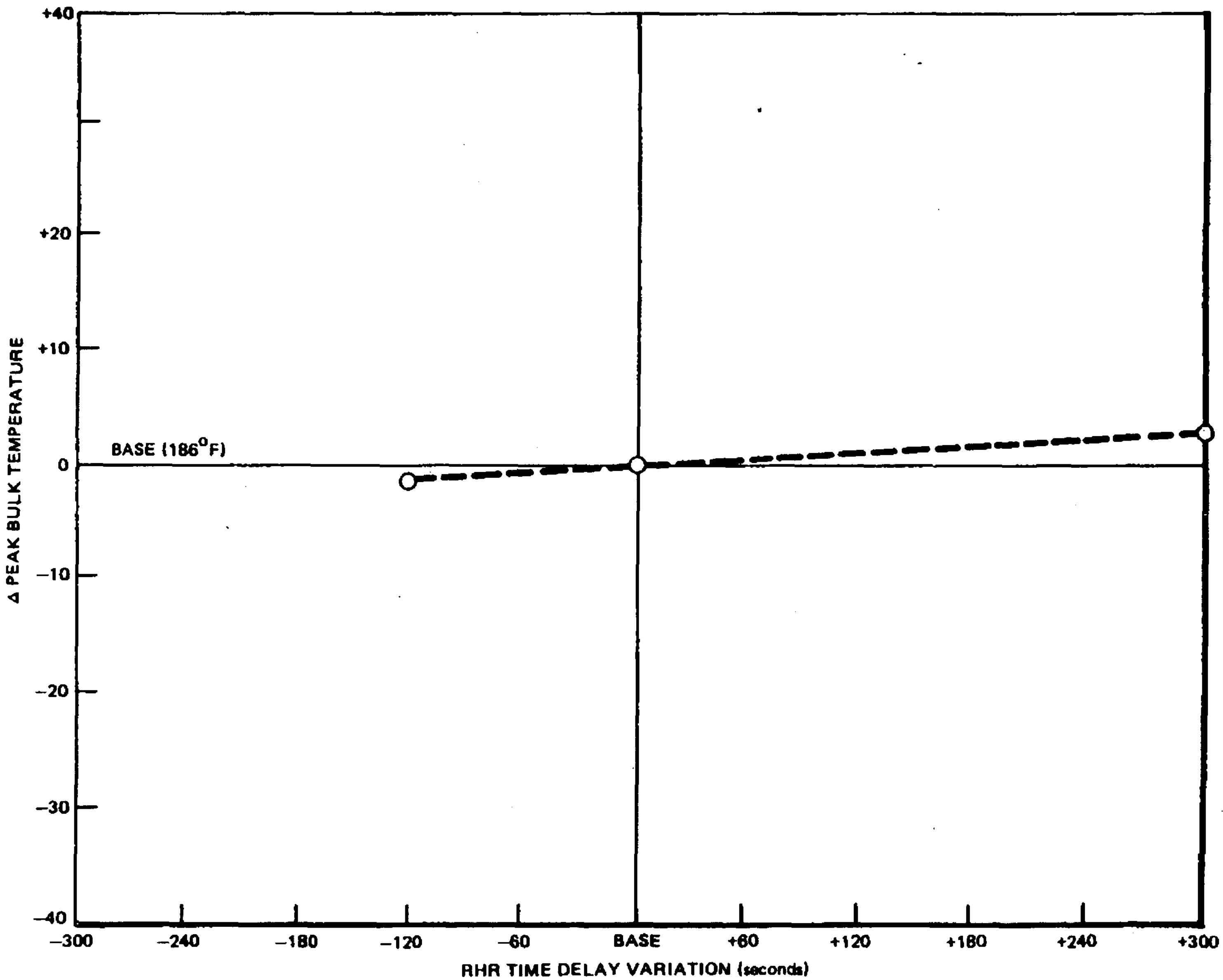


Figure 3.1.4.1.5. BWR-4 ATWS MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to a Change in the Time the RHR Heat Exchanger is Initiated

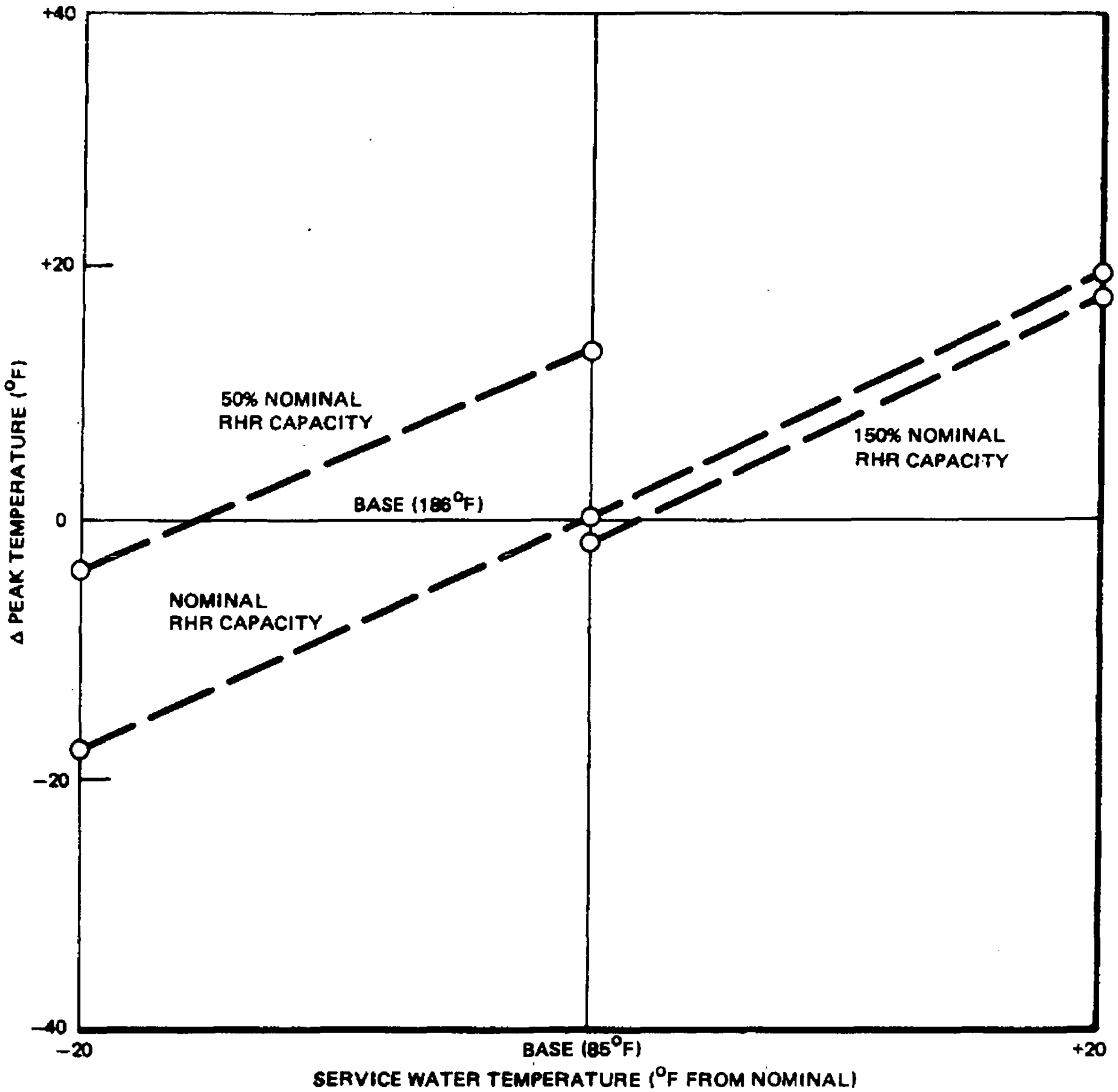


Figure 3.1.4.1.6. BWR-4 ATWS with ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to Service Water Temperature

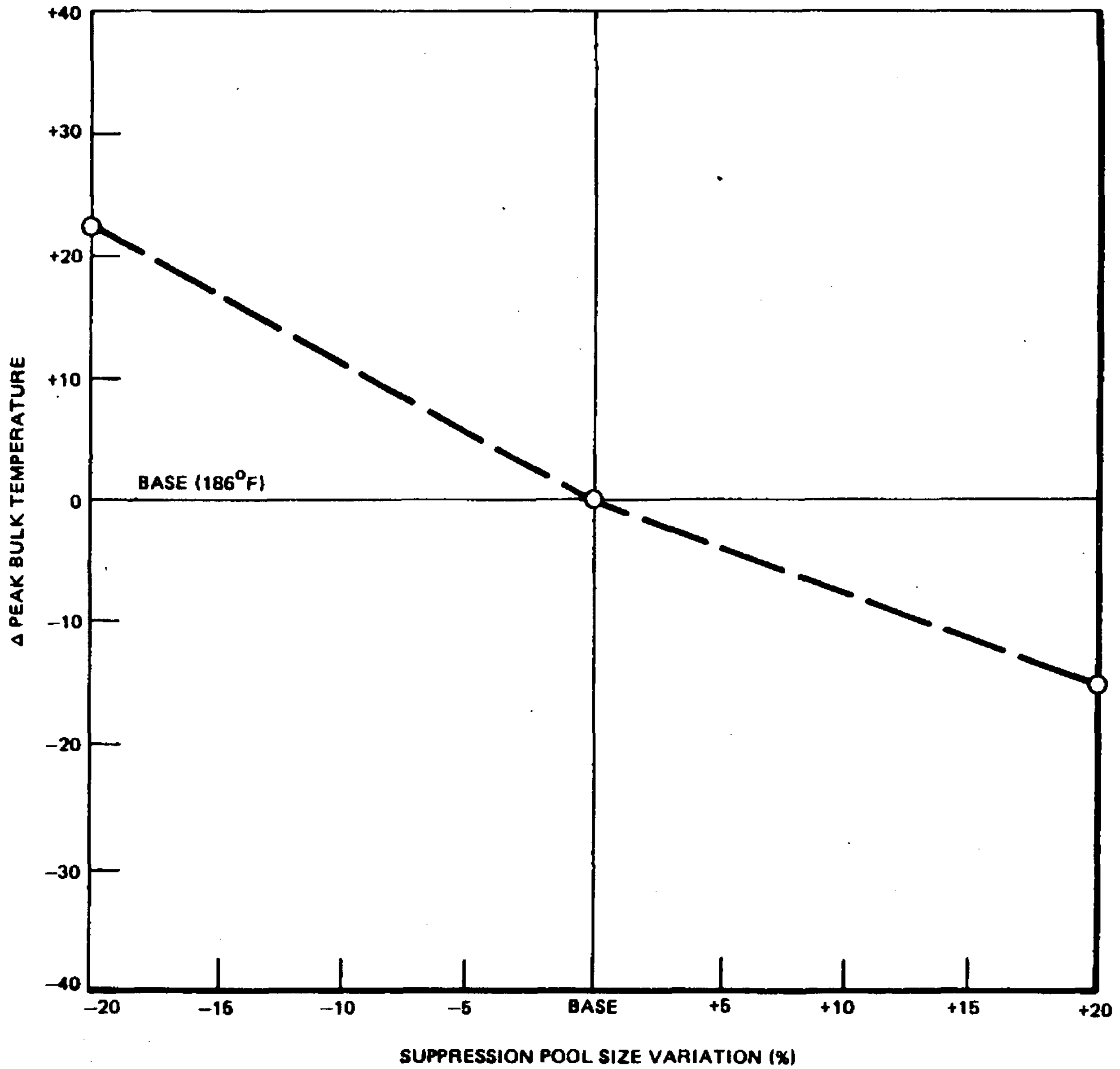


Figure 3.1.4.1.8. BWR-4 ATWS MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to a Change in the Size of the Suppression Pool

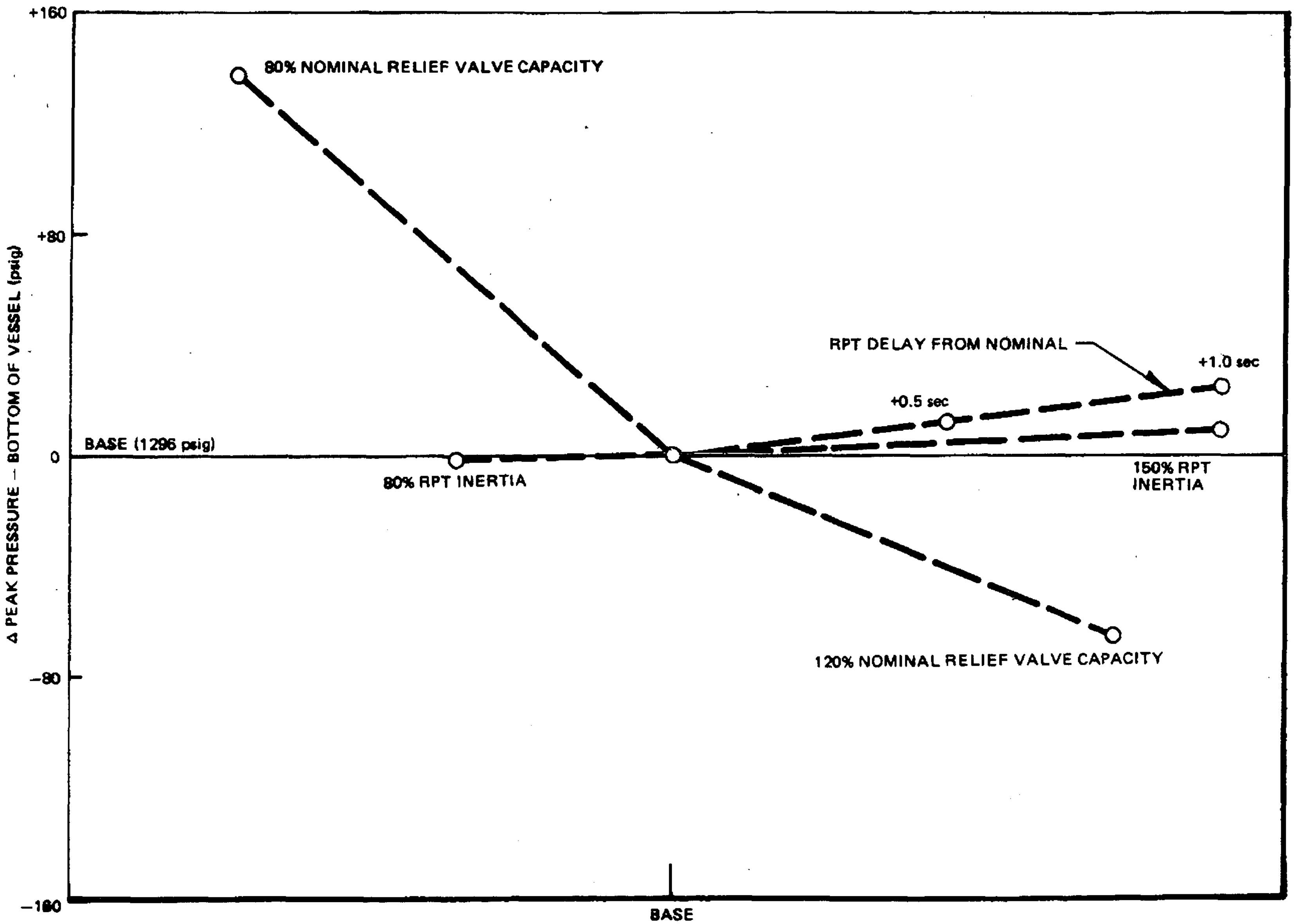


Figure 3.1.4.1.9. BWR-4 ATWS with ARI Failure MSIV Sensitivity: Vessel Pressure to Relief Valve Capacity and Recirculation Pump Delay and Inertia

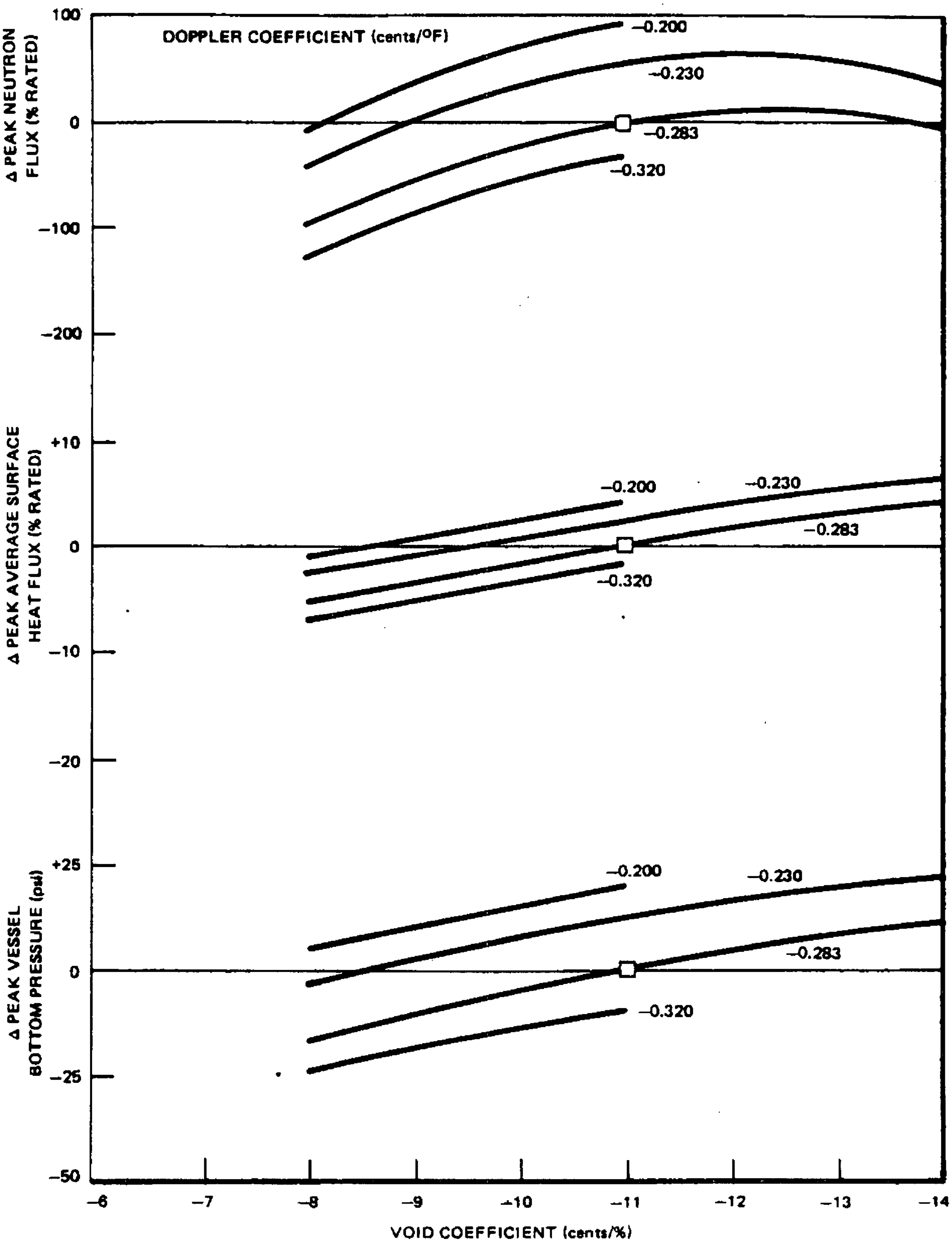
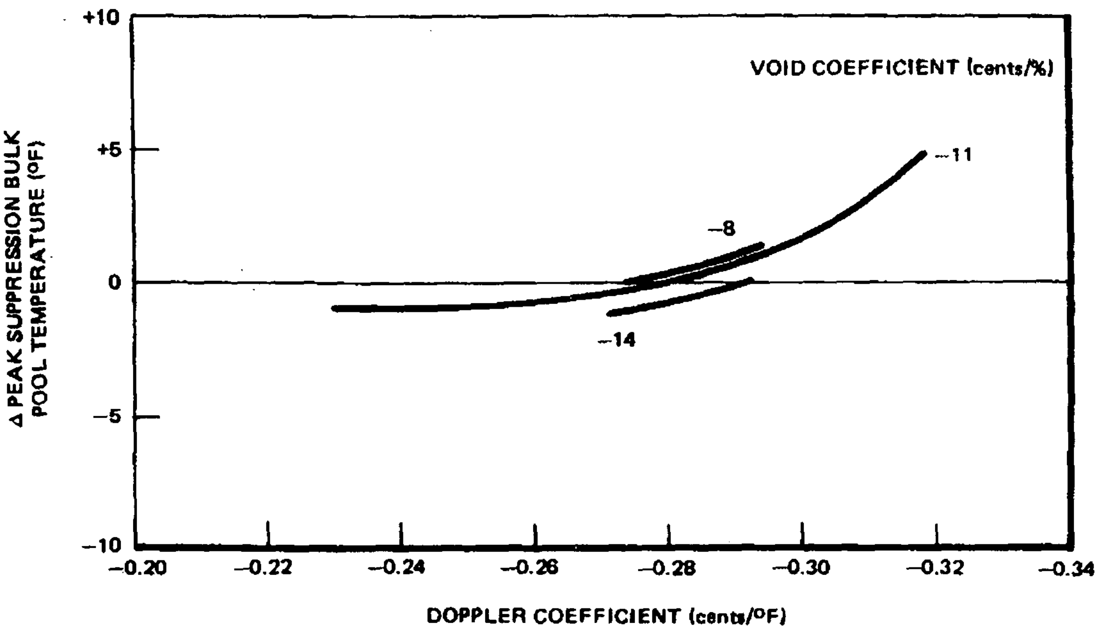


Figure 3.1.4.1.13-1. BWR/4-218 ATWS MSIV ARI Failure



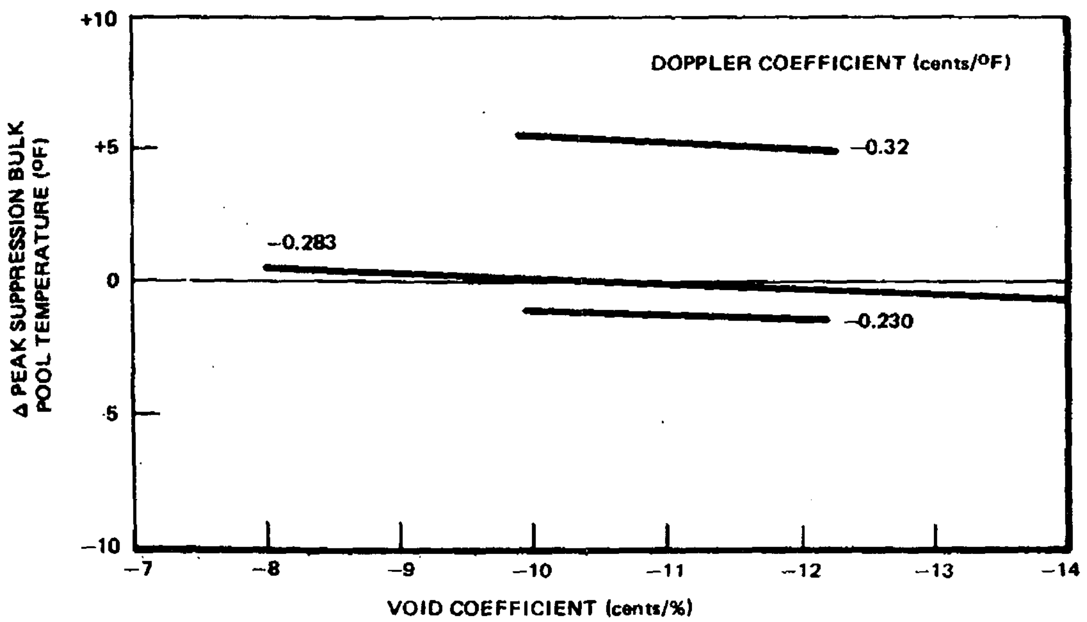


Figure 3.1.4.1.13-2. BWR/4-218 ATWS MSIV ARI Failure

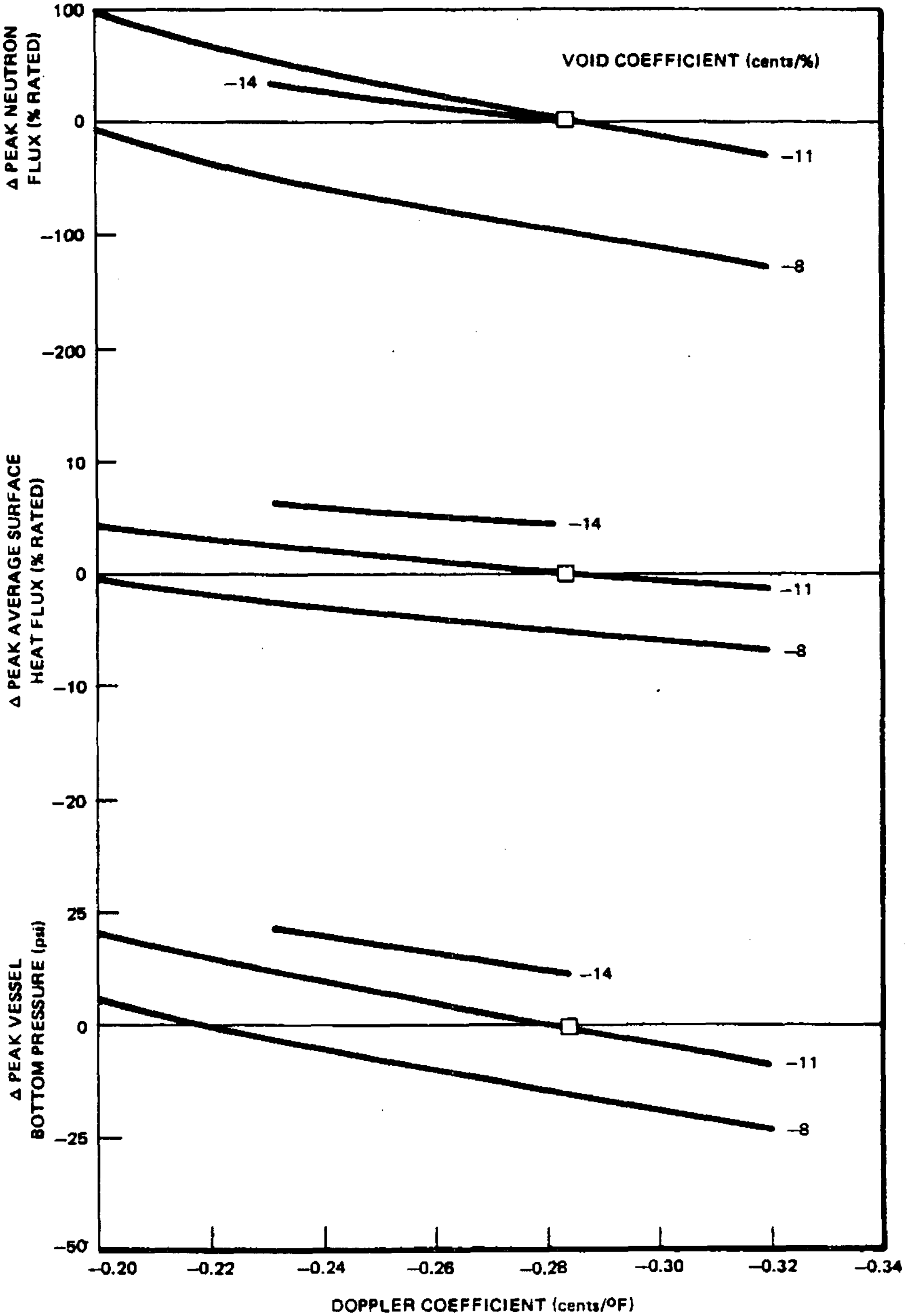


Figure 3 :.4.1.14-1. BWR-4-218 ATWS MSIV ARI Failure

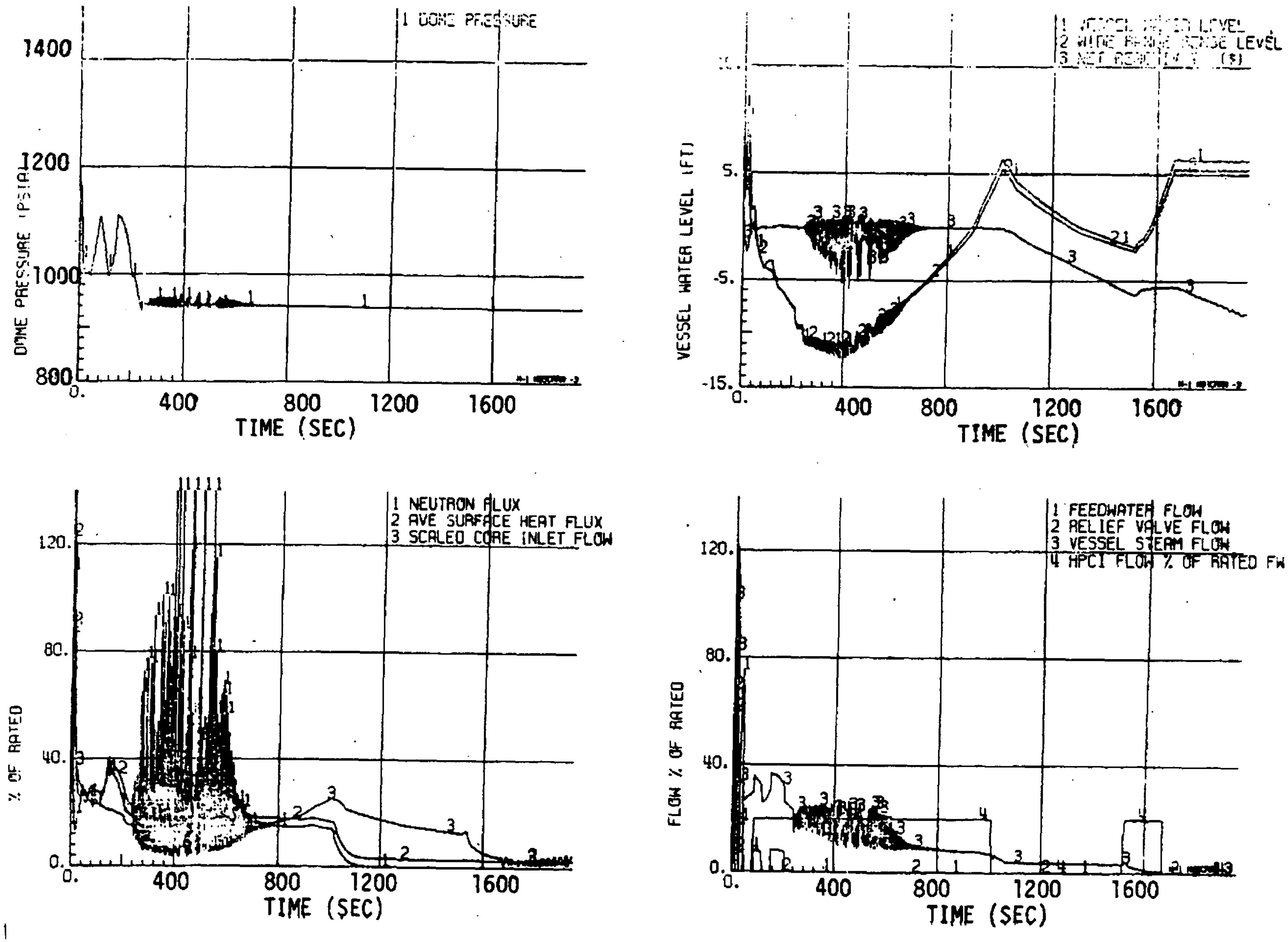


Figure 3.1.4.2. Turbine Trip, ARI Failure, 2 Minute 95% Boron Mix Eff, No MSIV Closure

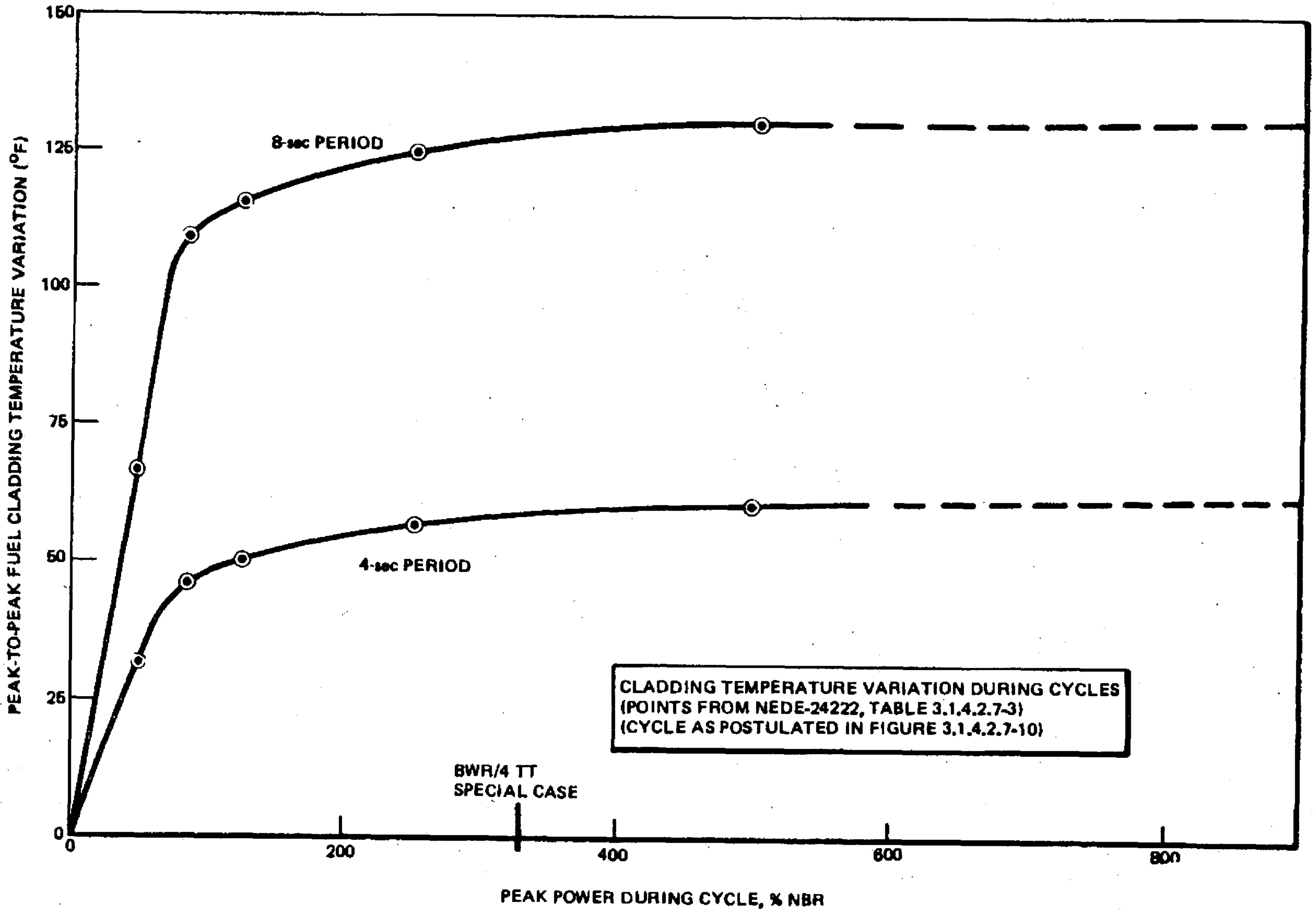


Figure 3.1.4.2.7-11. Fuel Clad Temperature Variation for Power Oscillations

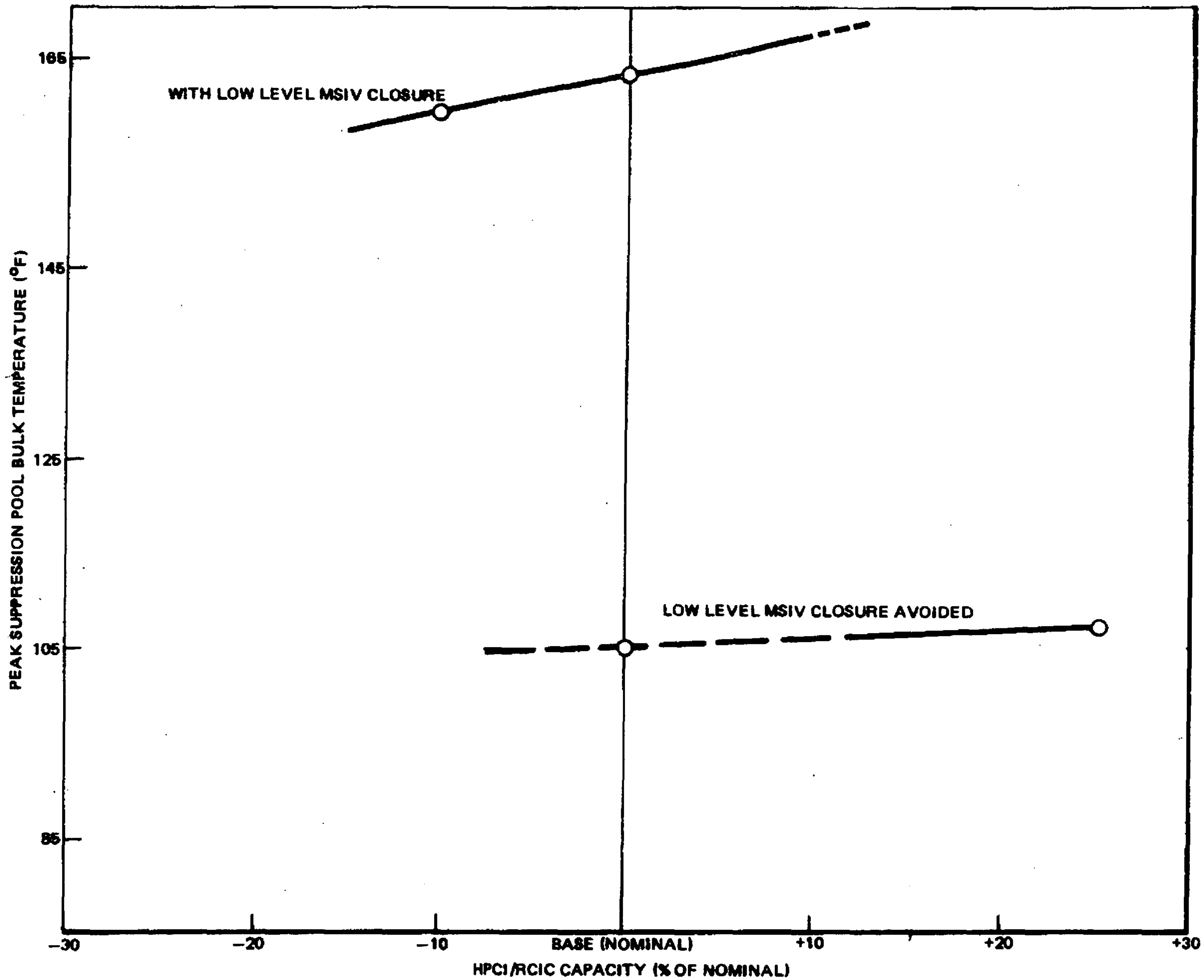


Figure 3.1.4.2.3. BWR-4 ATWS with ARI Failure Turbine Trip with Bypass Sensitivity: Peak Suppression Pool Bulk Temperature to HPCI/RCIC Capacity

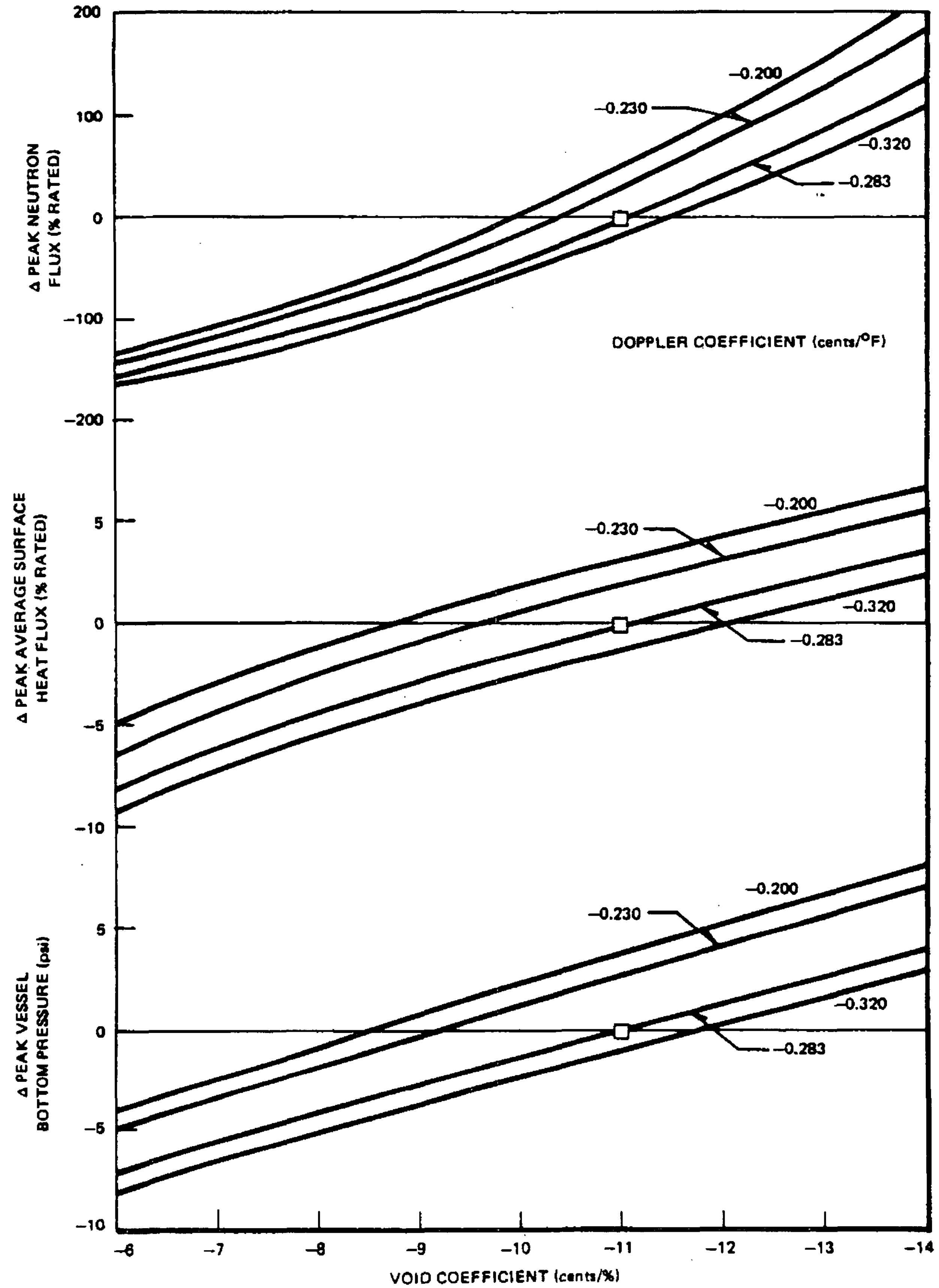


Figure 3.1.4.2.5-1. BWR/4 ATWS Turbine Trip ARI Failure

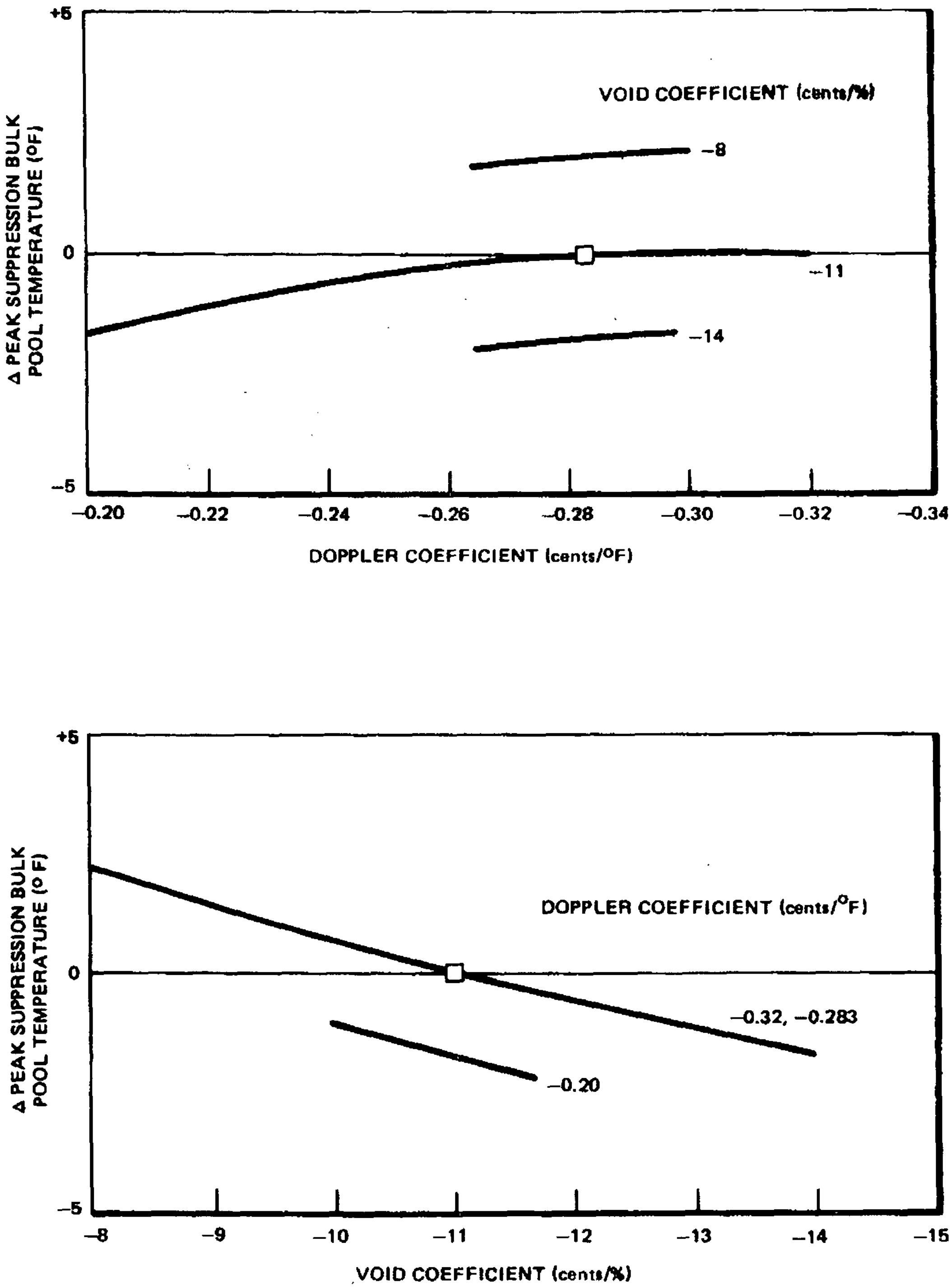


Figure 3.1.4.2.5-2. BWR/4 ATWS Turbine Trip ARI Failure, No MSIV Closure

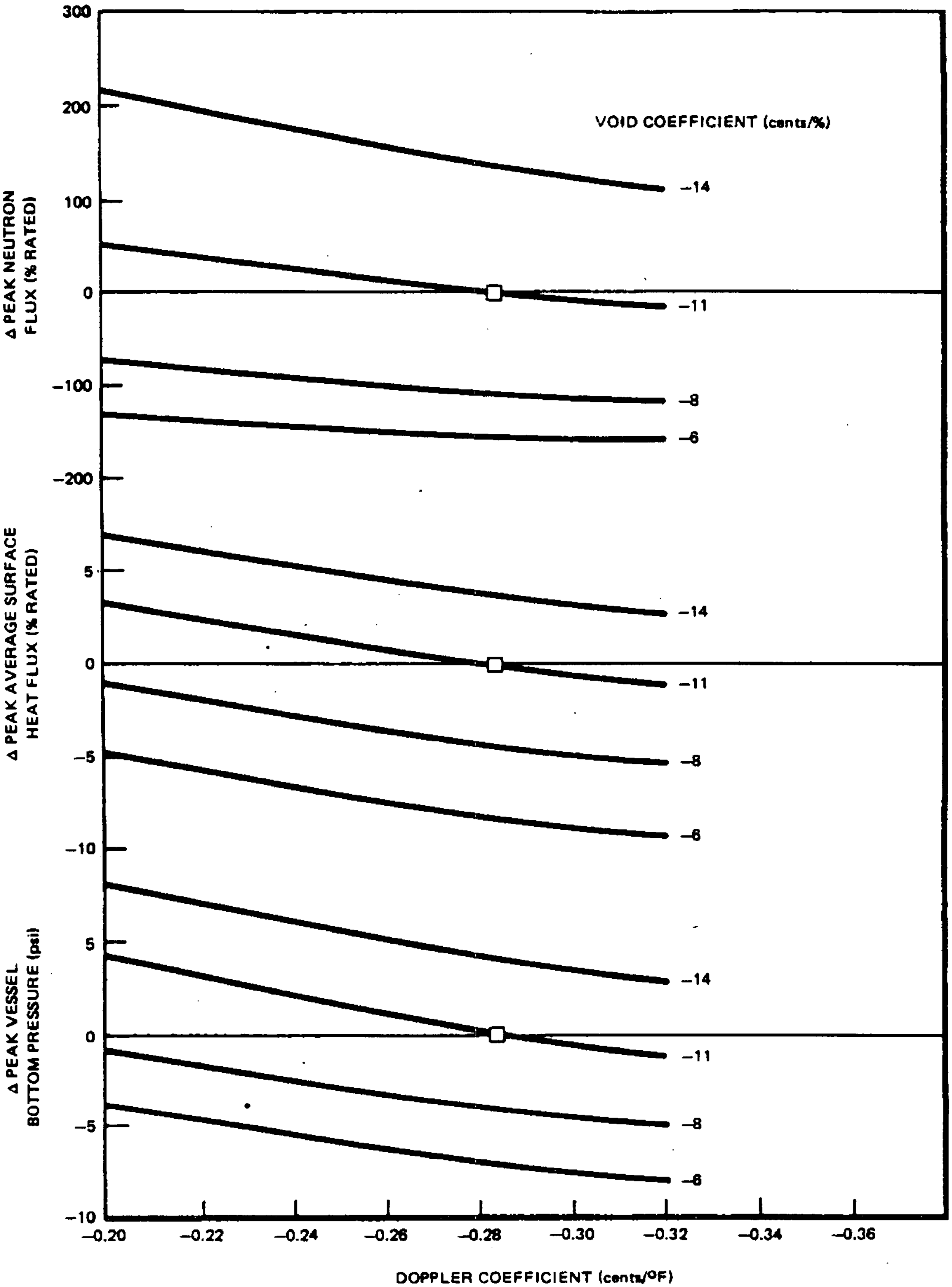


Figure 3.1.4.2.6-1. BWR/4 ATWS Turbine Trip ARI Failure

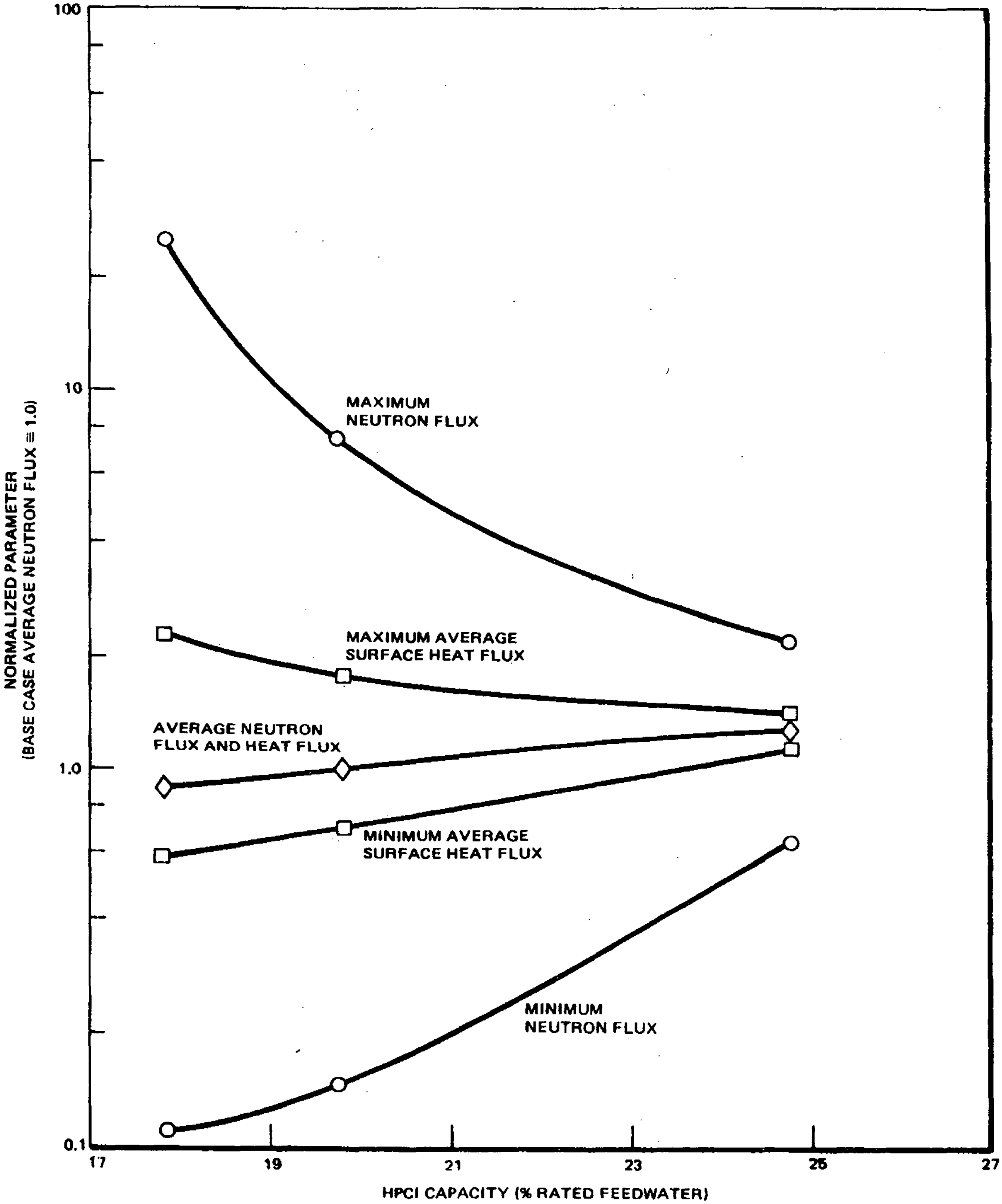


Figure 3.1.4.2.7-1. BWR/4 Turbine Trip Sensitivity Effect of HPCI Capacity on Limit Cycle, Neglecting MSIV Closure

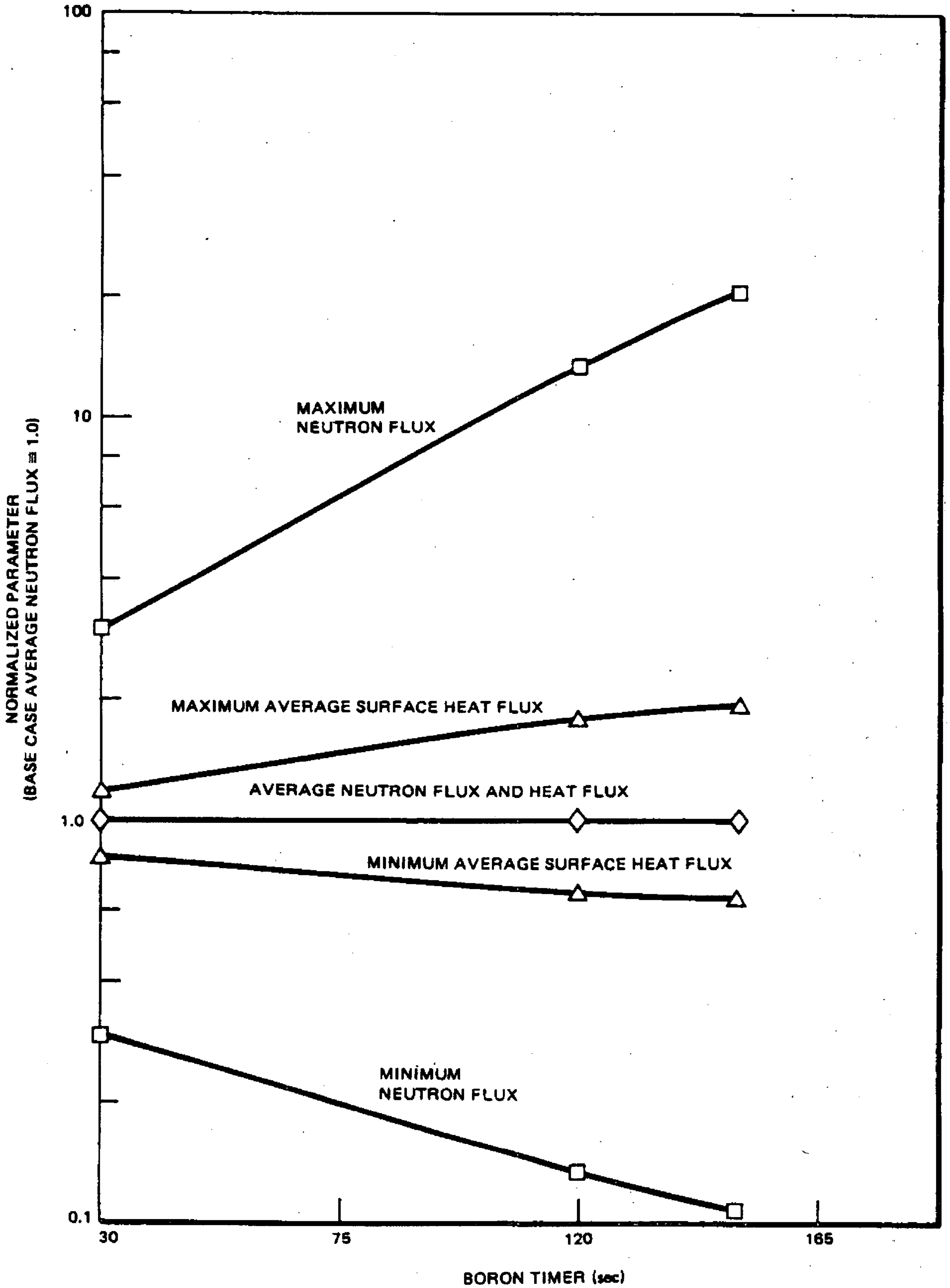


Figure 3.1.4.2.7-2. BWR/4 Turbine Trip Sensitivity Effect of Boron Delay Time on Limit Cycle

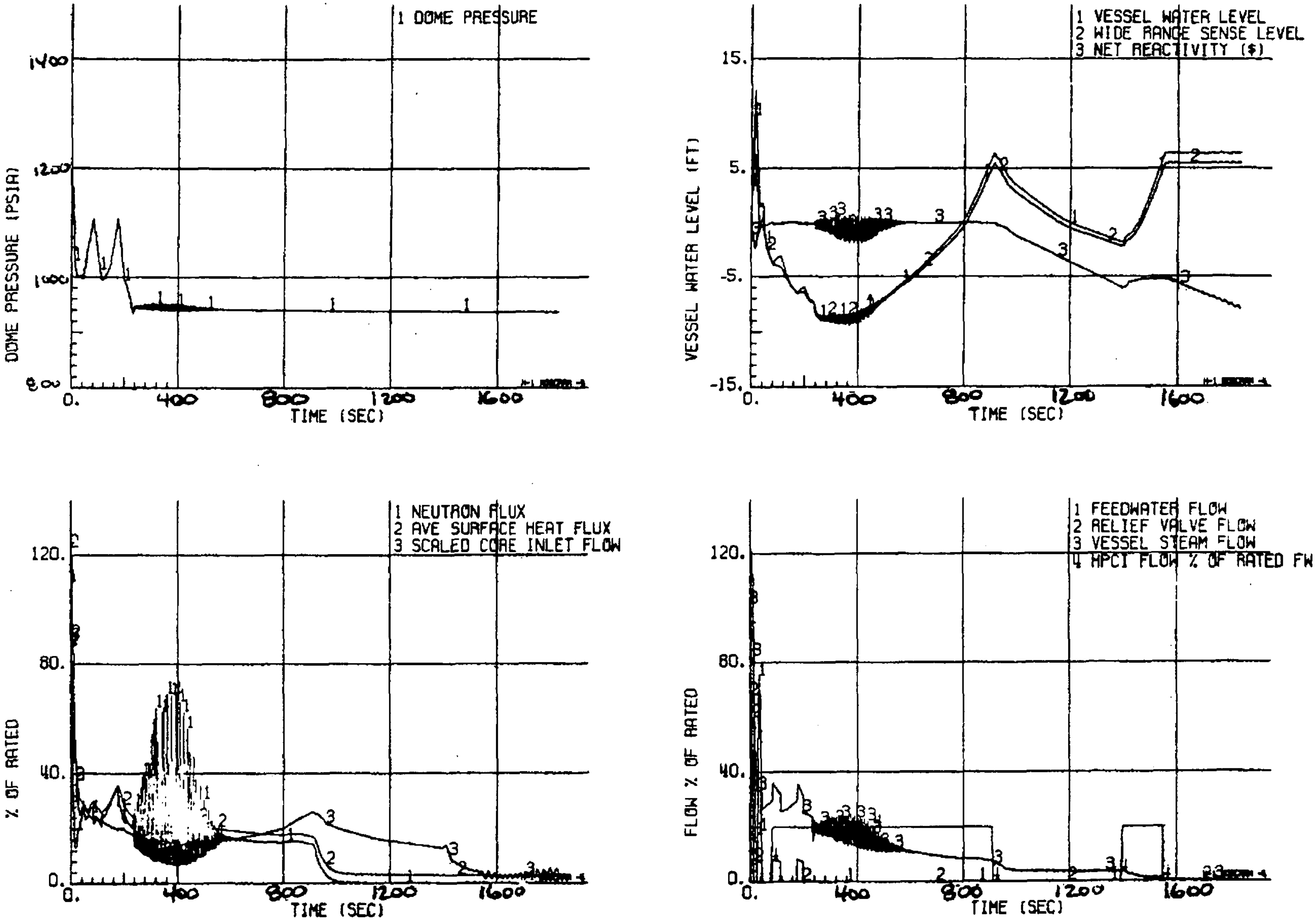


Figure 3.1.4.2.7-3. BWR/4 Turbine Trip - 30 Second Boron Delay Timer, No MSIV Closure

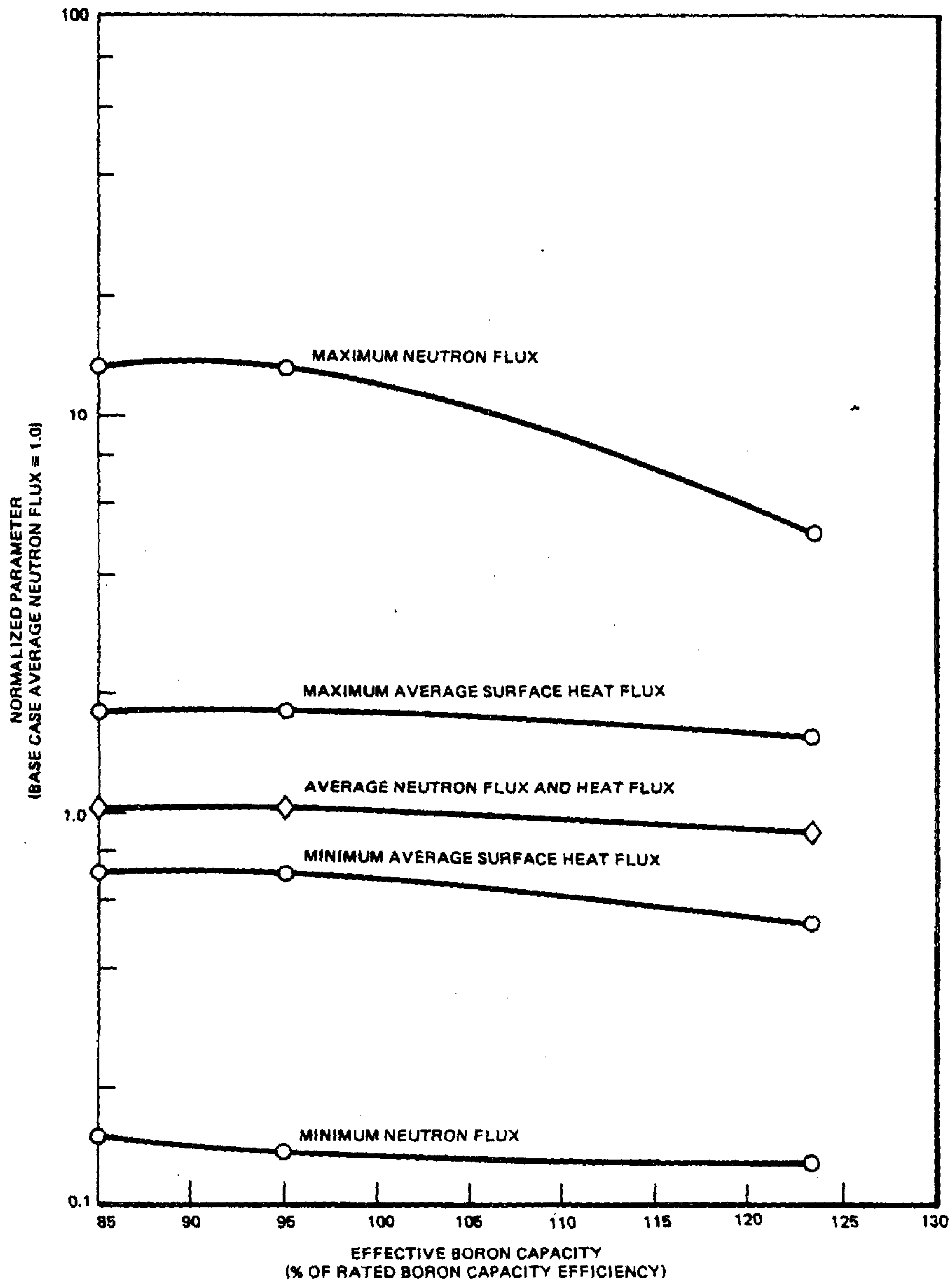


Figure 3.1.4.2.7-4. BWR/4 Turbine Trip Sensitivity Effect of Boron Capacity/Mixing Efficiency on Limit Cycle

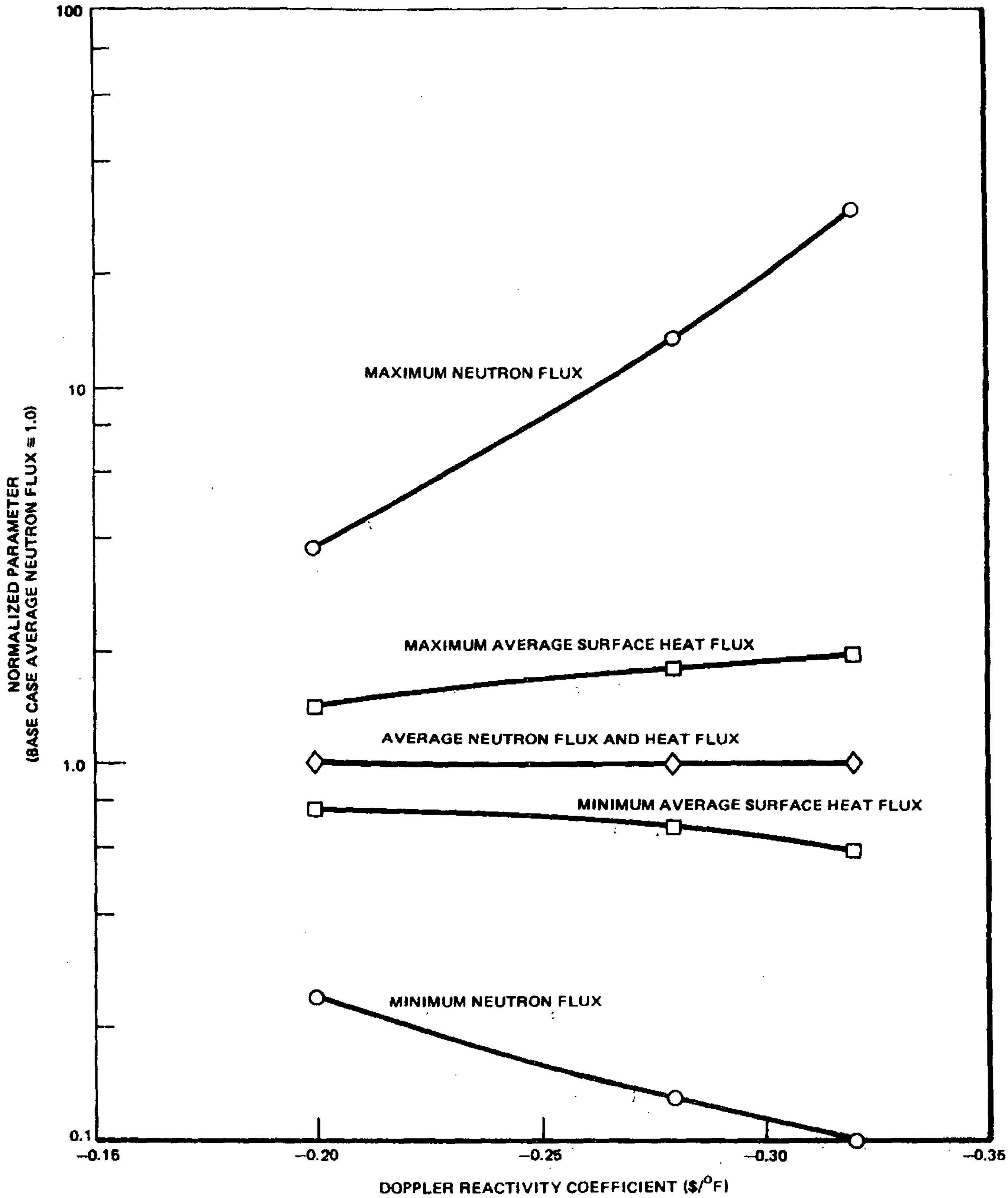


Figure 3.1.4.2.7-5. BWR/4 Turbine Trip Sensitivity Doppler Reactivity Coefficient Effect on Limit Cycle

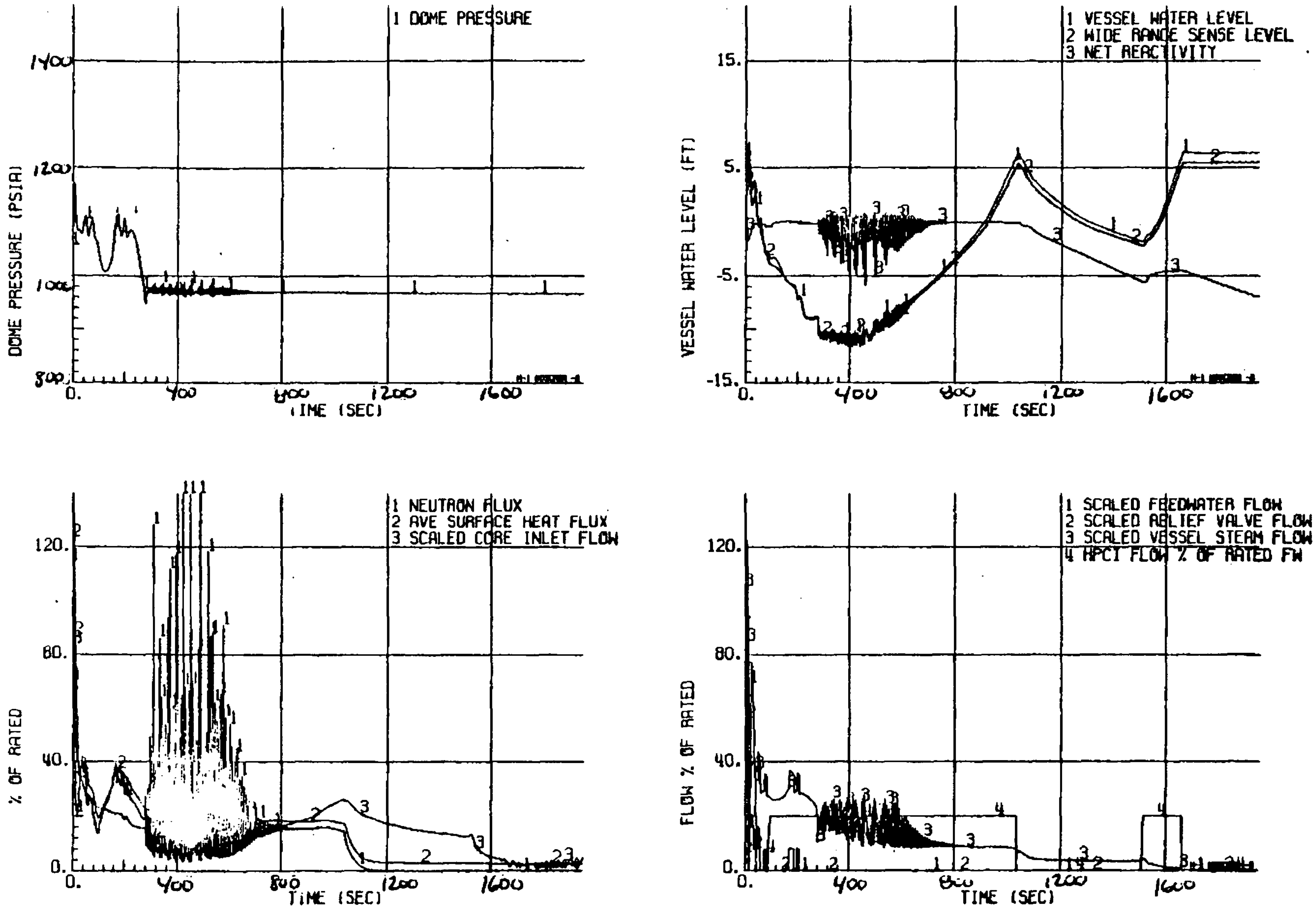


Figure 3.1.4.2.7-6. BWR/4 Turbine Trip; -0.23¢/°F Doppler Coefficient, No MSIV Closure

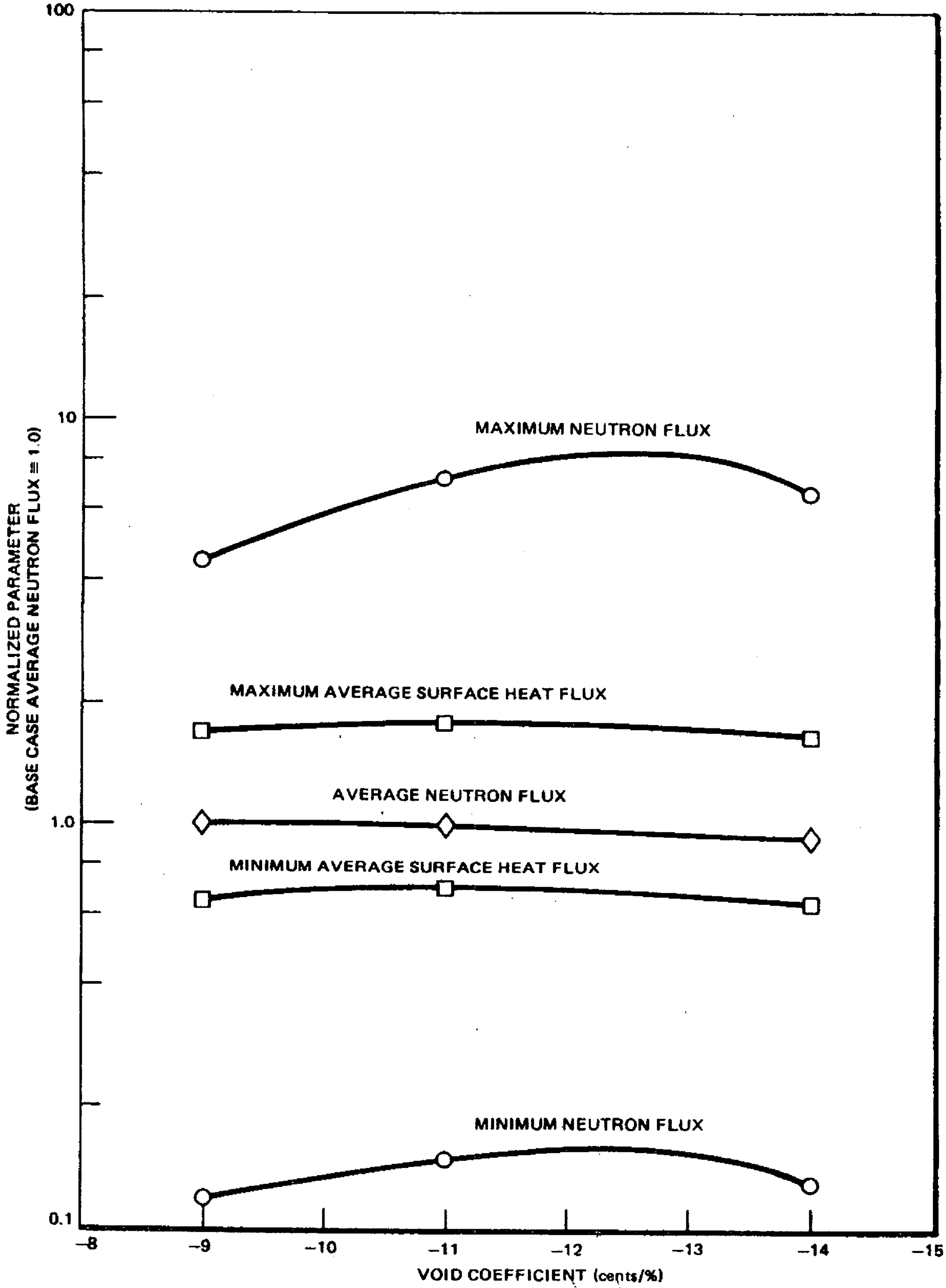


Figure 3.1.4.2.7-7. BWR/4 Turbine Trip Sensitivity Effect of Void Reactivity Coefficient on Limit Cycle

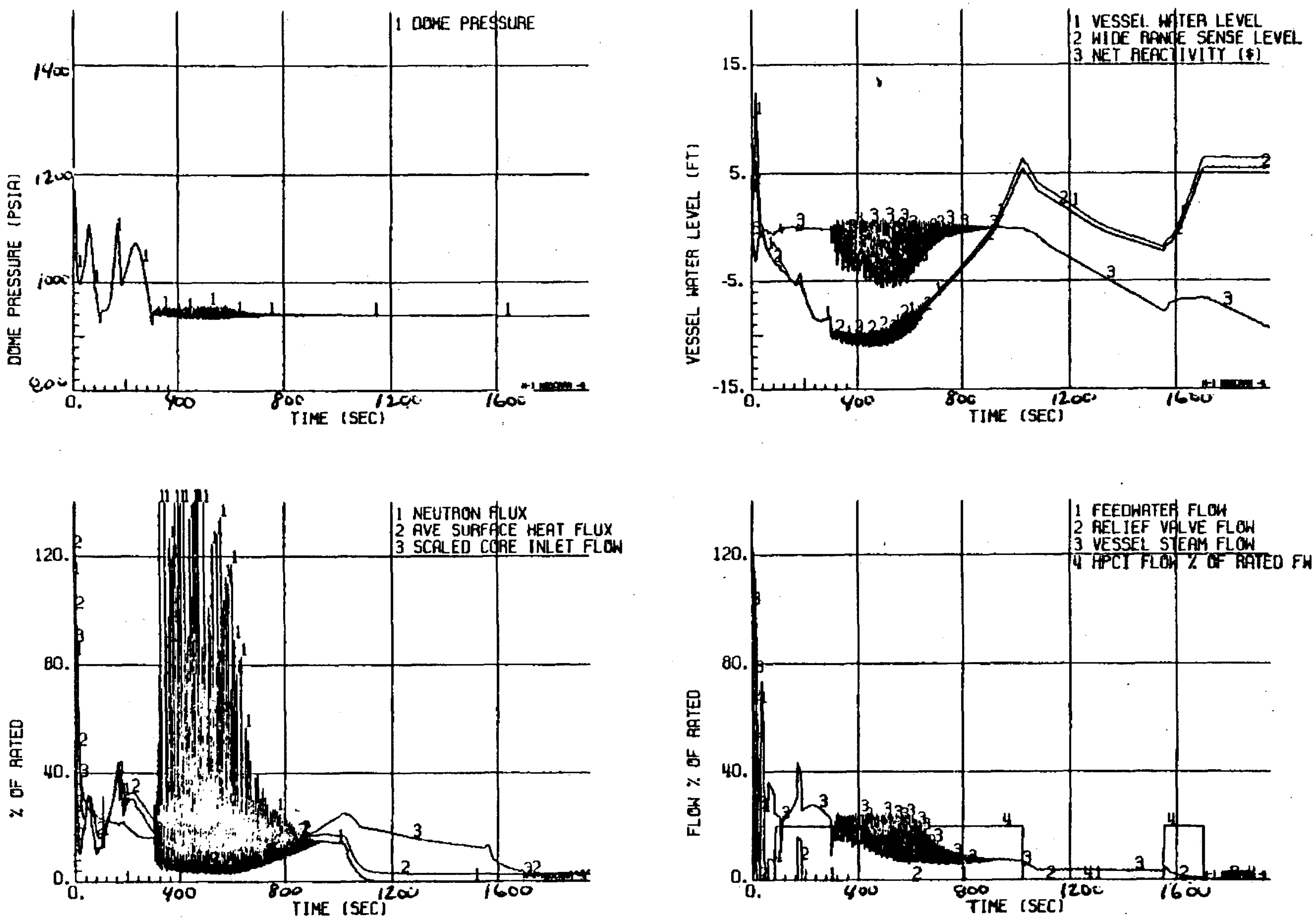


Figure 3.1.4.2.7-8. BWR/4 Turbine Trip -14¢/% Void Coefficient, No MSIV Closure

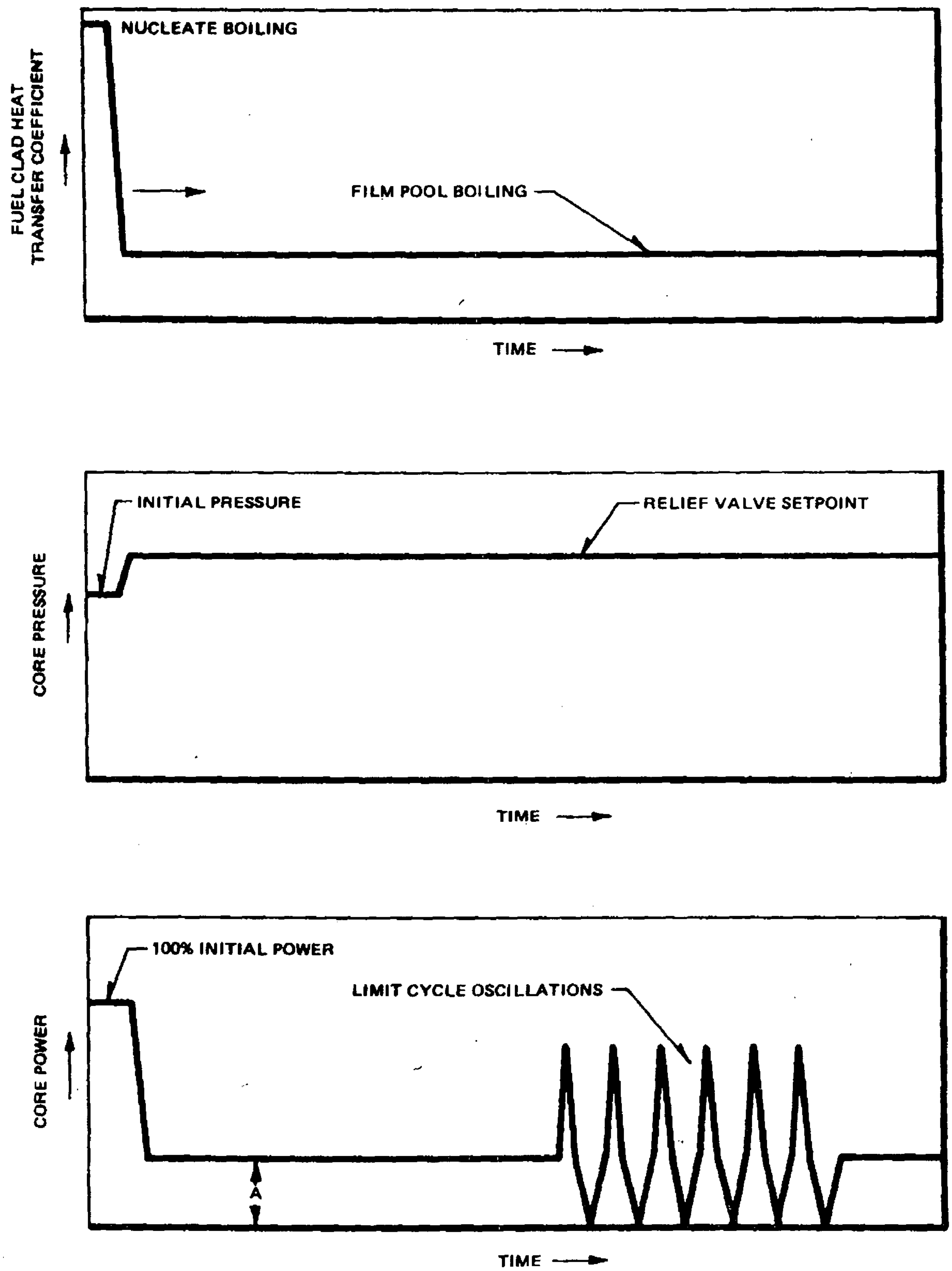
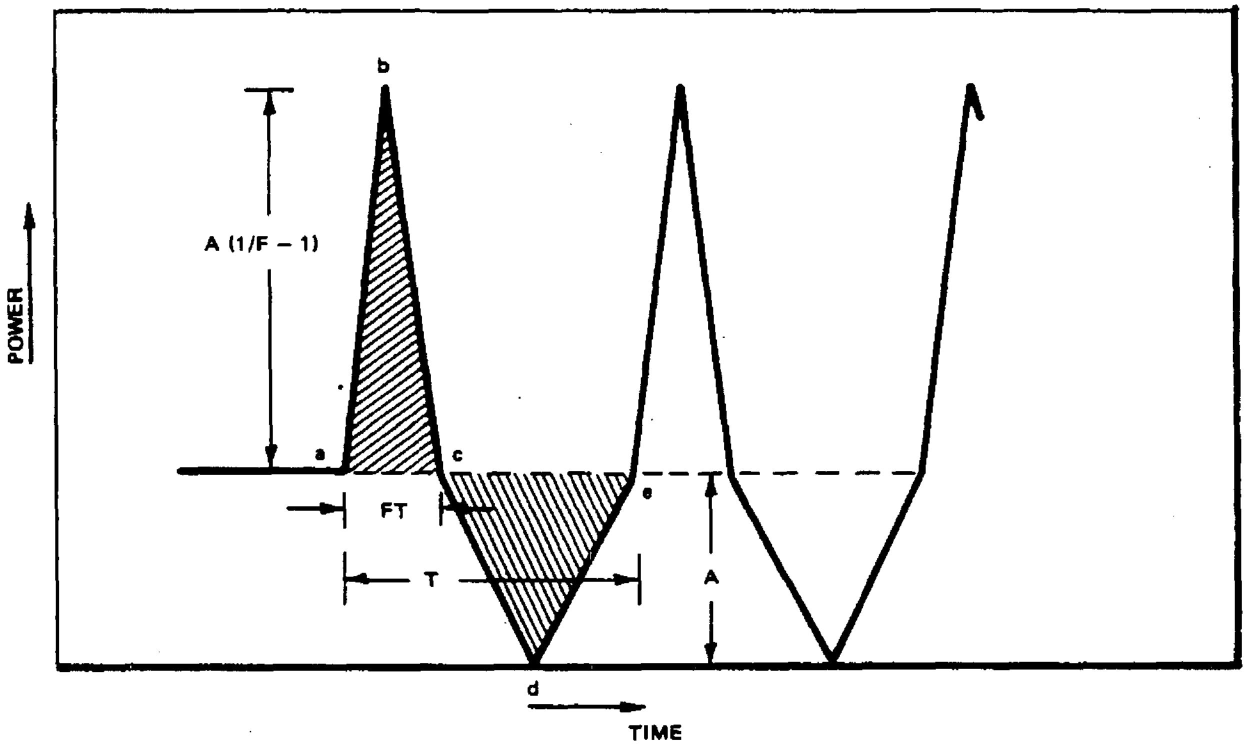


Figure 3.1.4.2.7-9. Input Conditions Used for Fuel Condition Analysis During Limit Cycle Oscillations



T = PERIOD OF CYCLE

A = AVERAGE POWER DURING LIMIT CYCLE

F = FRACTION OF LIMIT CYCLE PERIOD DURING WHICH THE POWER IS GREATER THAN AVERAGE POWER

A (1/F − 1) = MAXIMUM POWER DURING LIMIT CYCLE OSCILLATION

MINIMUM POWER DURING LIMIT CYCLE OSCILLATIONS ASSUMED TO BE ZERO

Figure 3.1.4.2.7-10. Shape of Limit Cycle Assumed for Fuel Condition Analysis

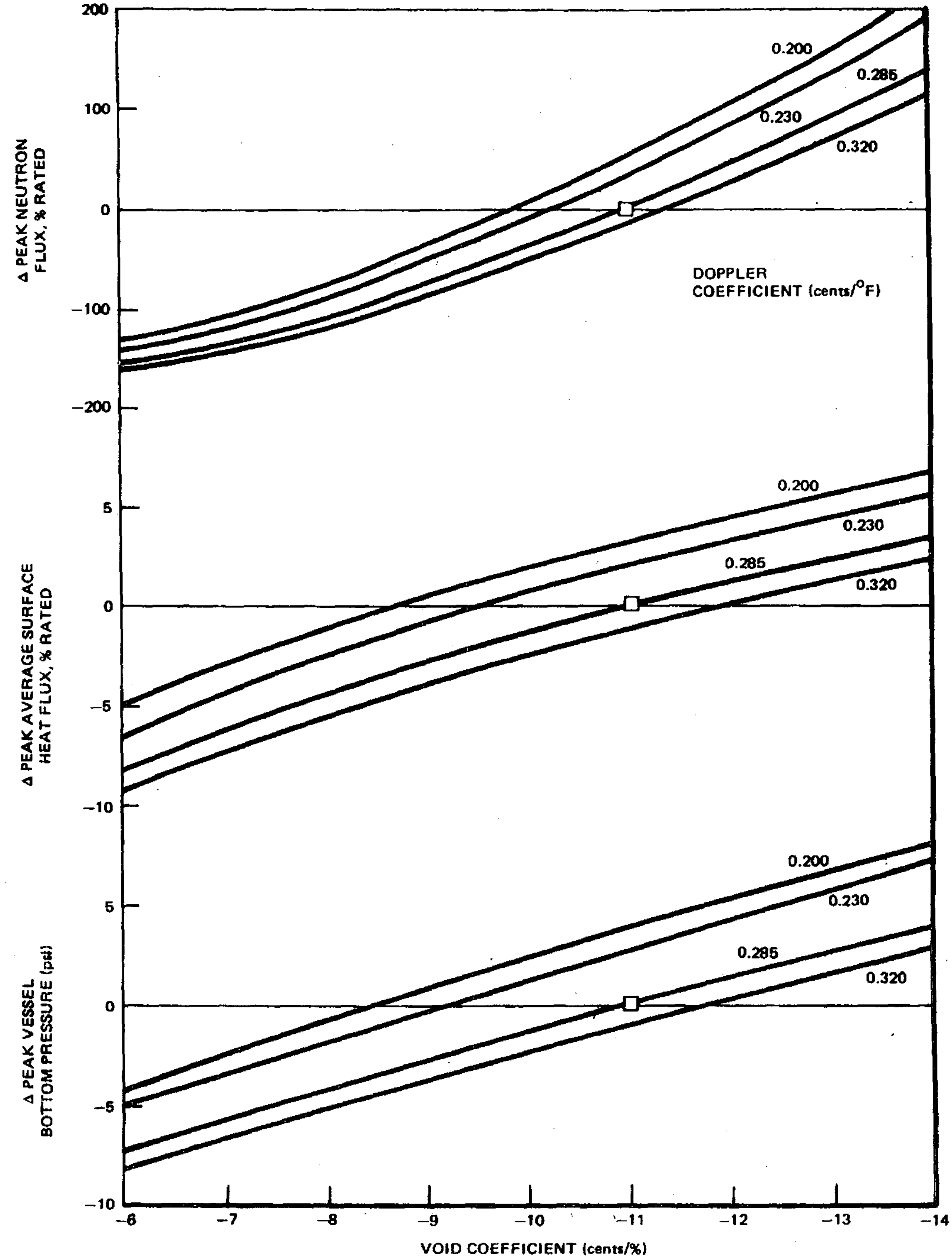


Figure 3.1.4.2.13-1. BWR/4 Turbine Trip

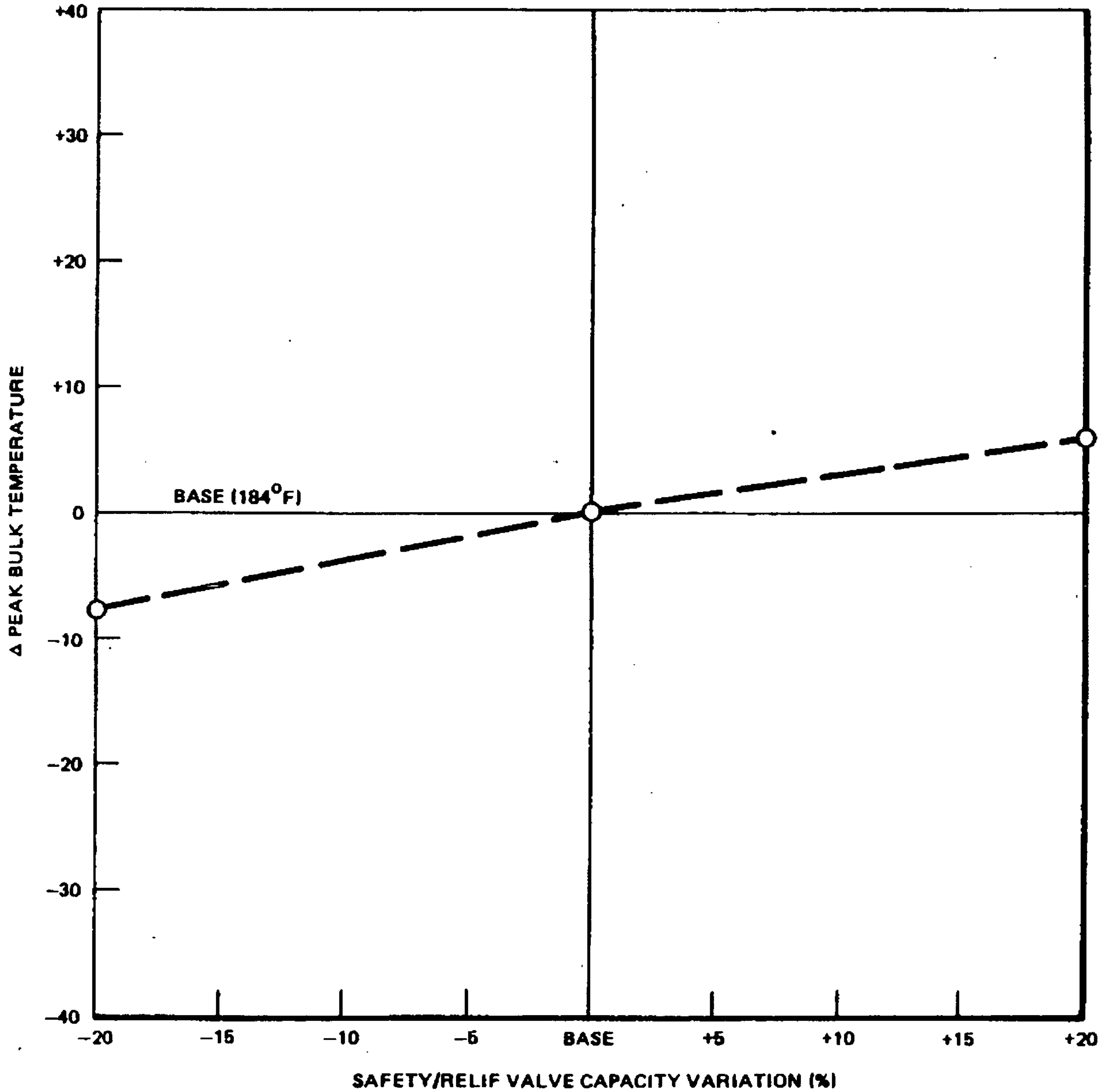


Figure 3.1.4.3.1-1. BWR-4 ATWS IORV Sensitivity: Peak Suppression Pool Bulk Temperature to a Change in Relief Valve Capacity

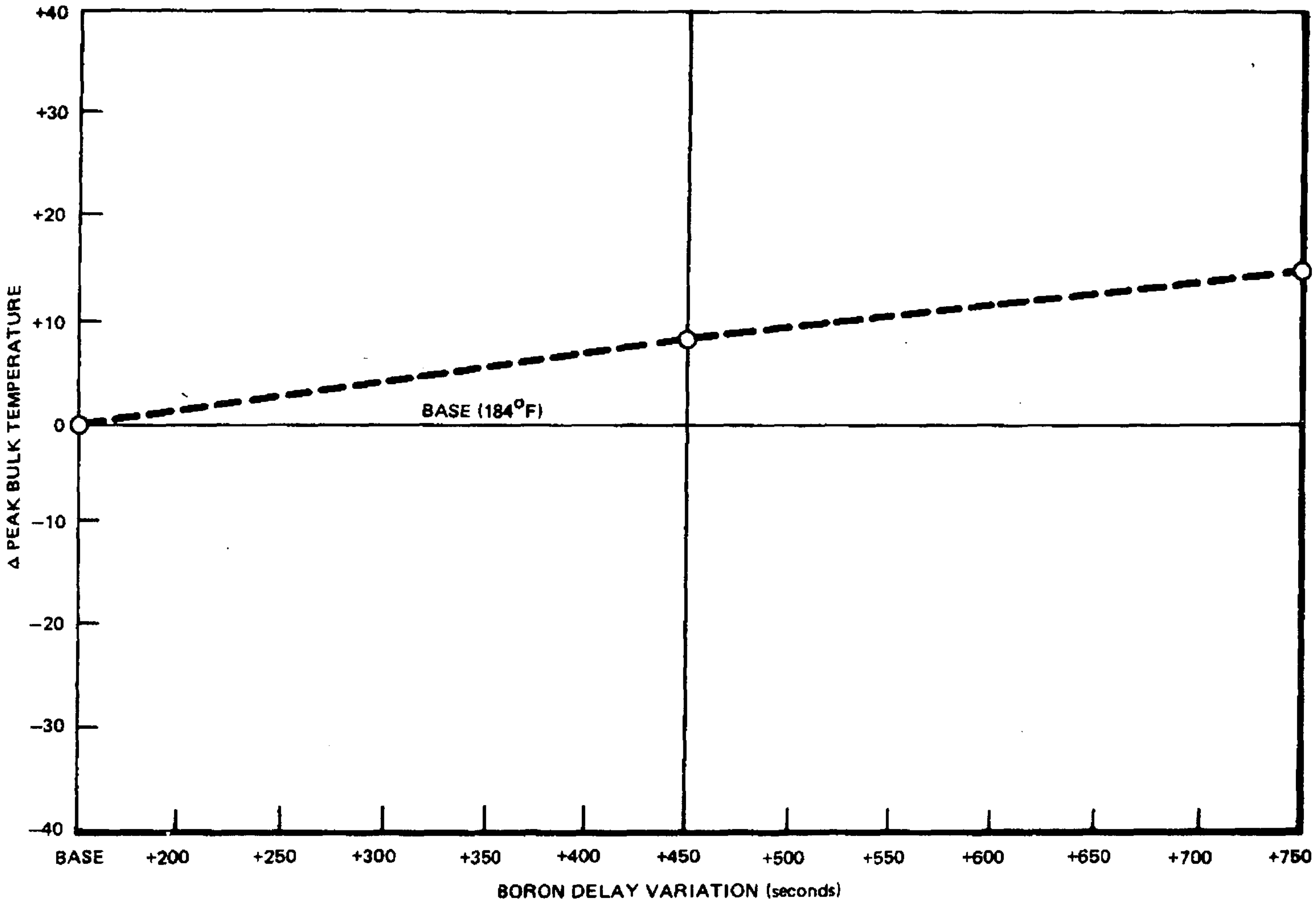


Figure 3.1.4.3.2. BWR-4 ATWS IORV Sensitivity: Peak Suppression Pool Bulk Temperature to a Change in Boron Injection Delay

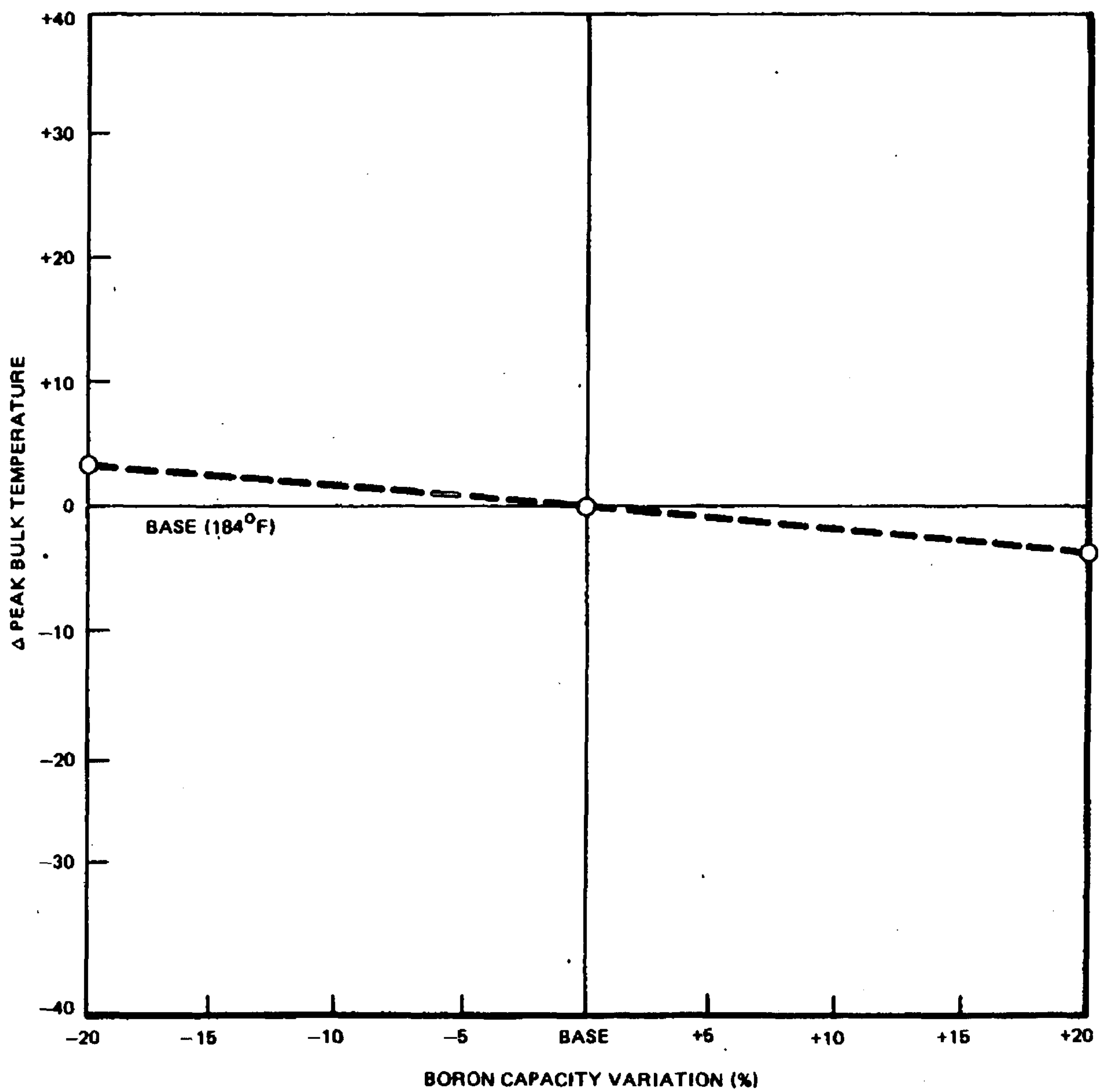


Figure 3.1.4.3.3. BWR-4 ATWS IORV Sensitivity: Peak Suppression Pool Bulk Temperature to a Change in Boron Capacity

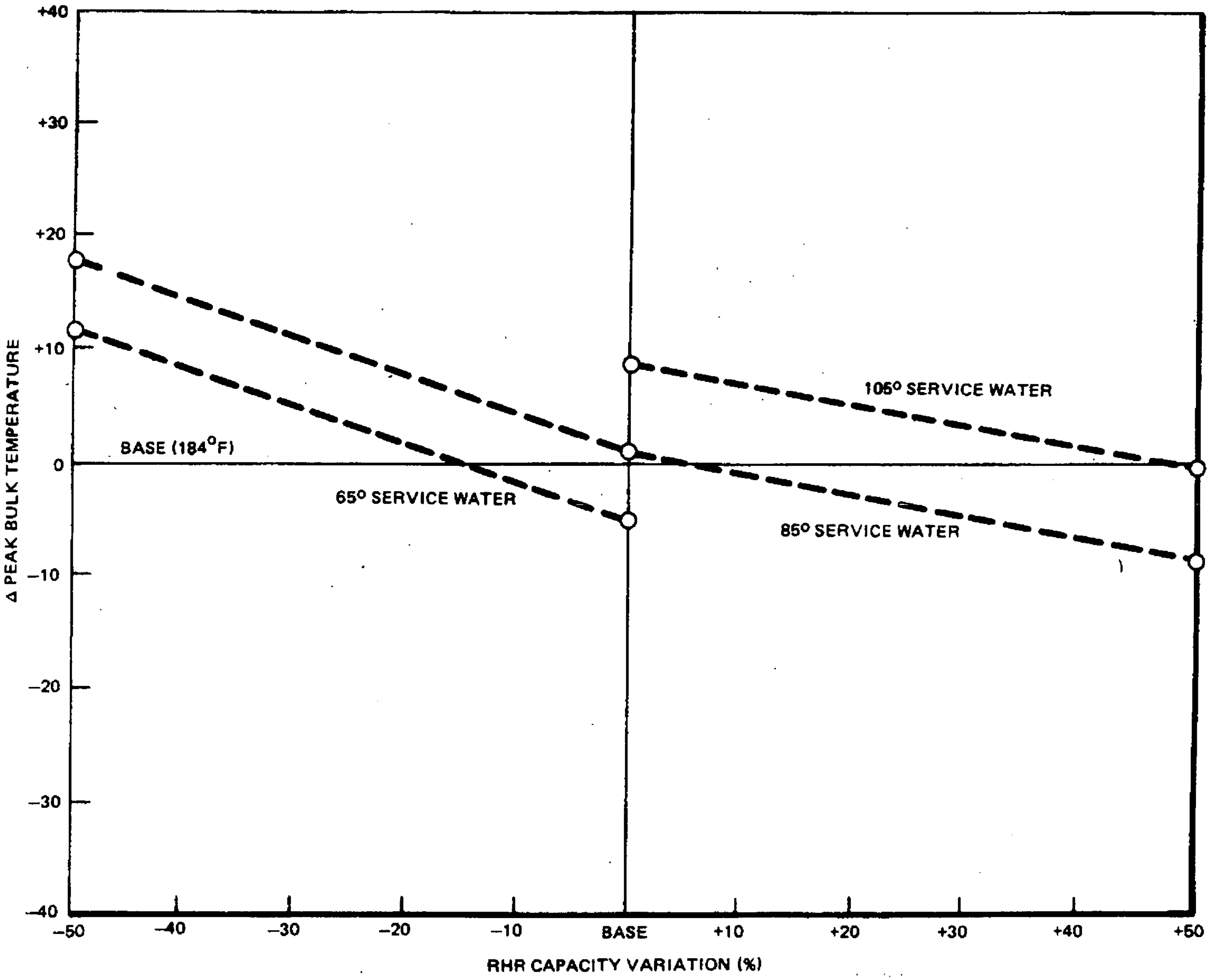


Figure 3.1.4.3.4. BWR-4 ATWS IORV Sensitivity: Peak Suppression Pool Bulk Temperature; to a Change in the Size of the Heat Exchanger, to a Change in the Service Water Temperature

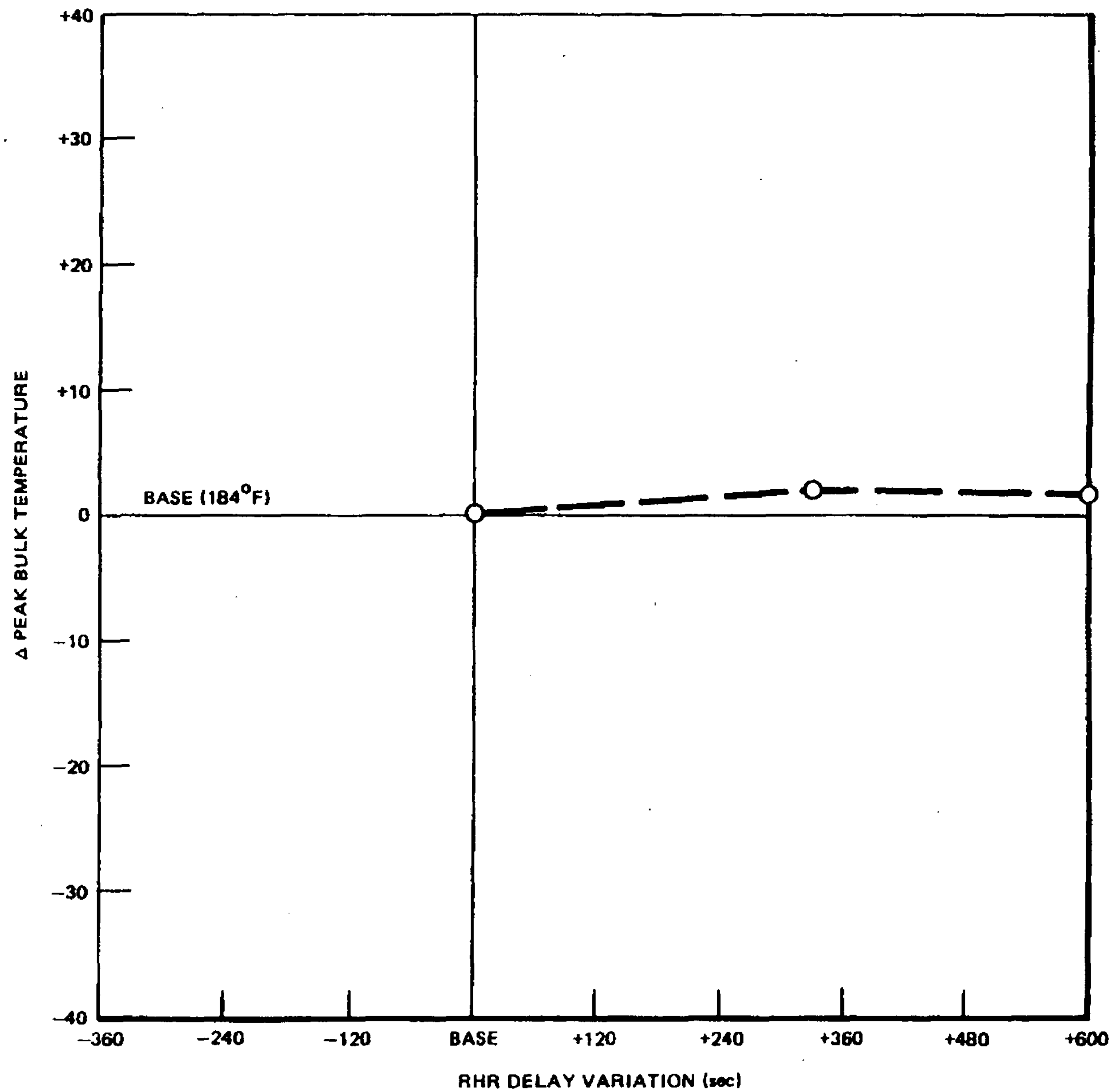


Figure 3.1.4.3.5-1. BWR/4 ATWS IORV Sensitivity: Peak Suppression Pool Bulk Temperature To a Change in Time When the RHR Heat Exchanger is Initiated

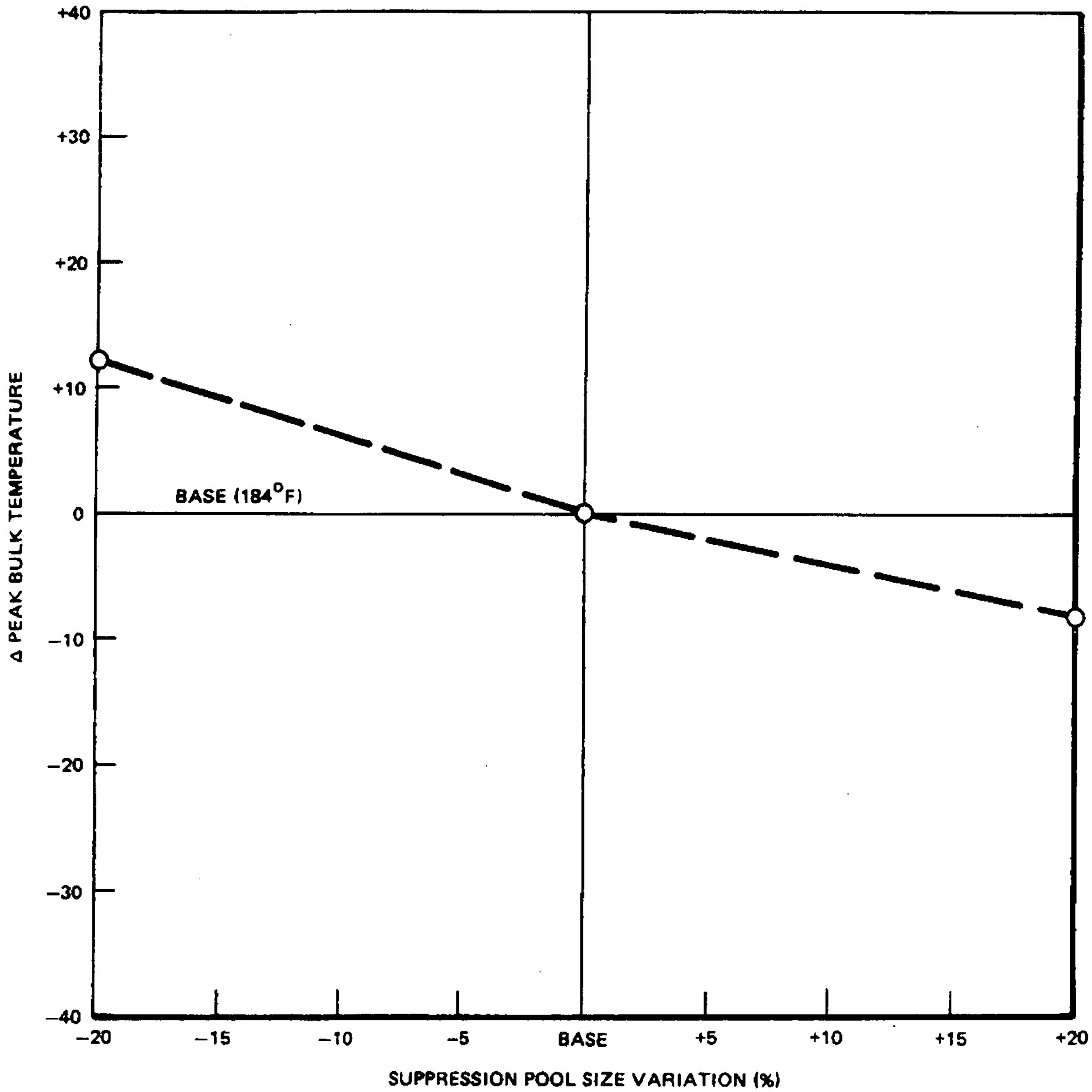


Figure 3.1.4.3.6-1. BWR/4 ATWS IORV Sensitivity: Peak Suppression Pool Bulk Temperature To a Change in Initial Size of the Suppression Pool

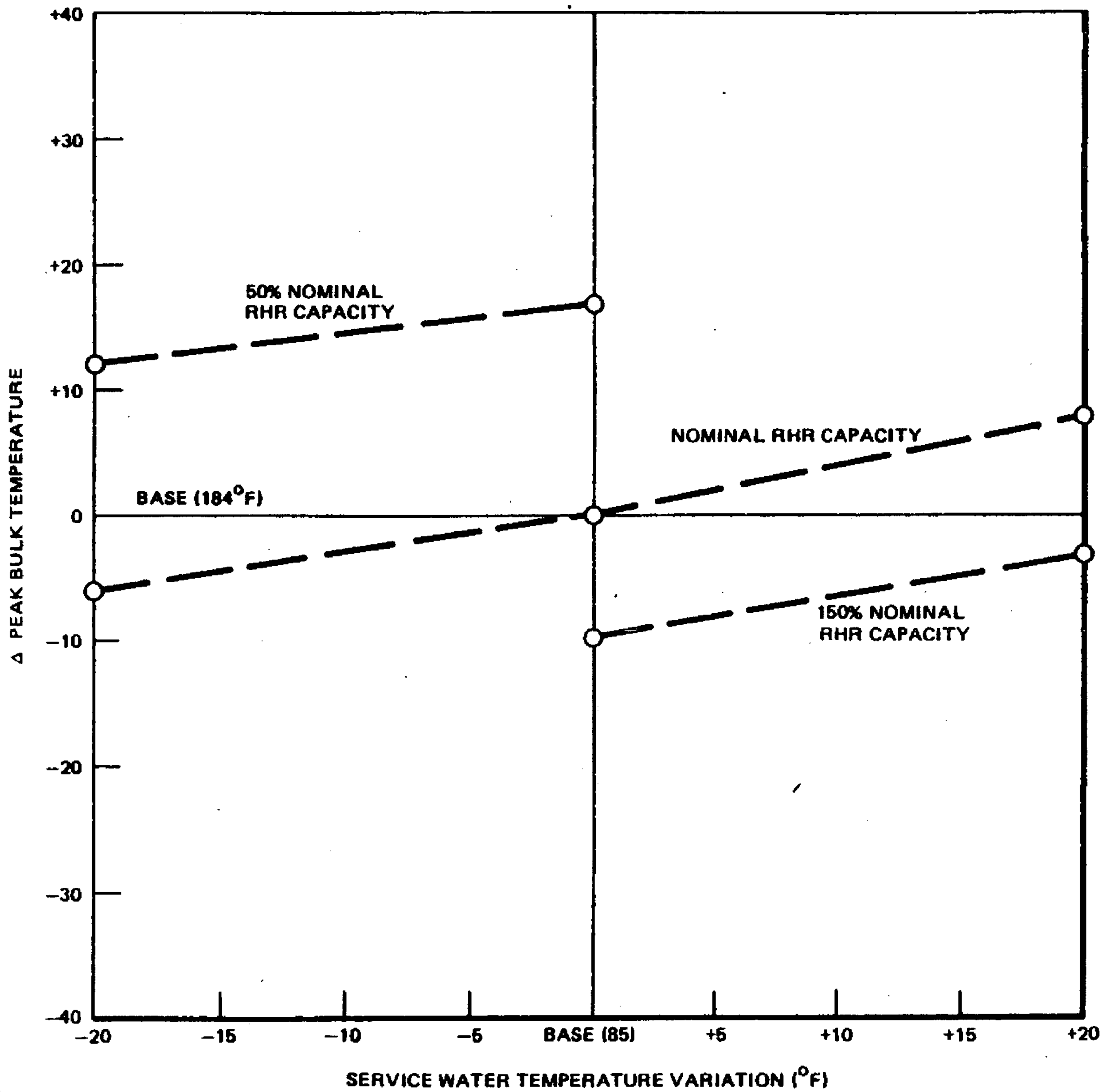


Figure 3.1.4.3.7-1. BWR/4 ATWS IORV Sensitivity: Peak Suppression Pool Bulk Temperature; To a Change in the Initial Temperature of the Service Water, To a Change in the Capacity of the RHR Heat Exchanger

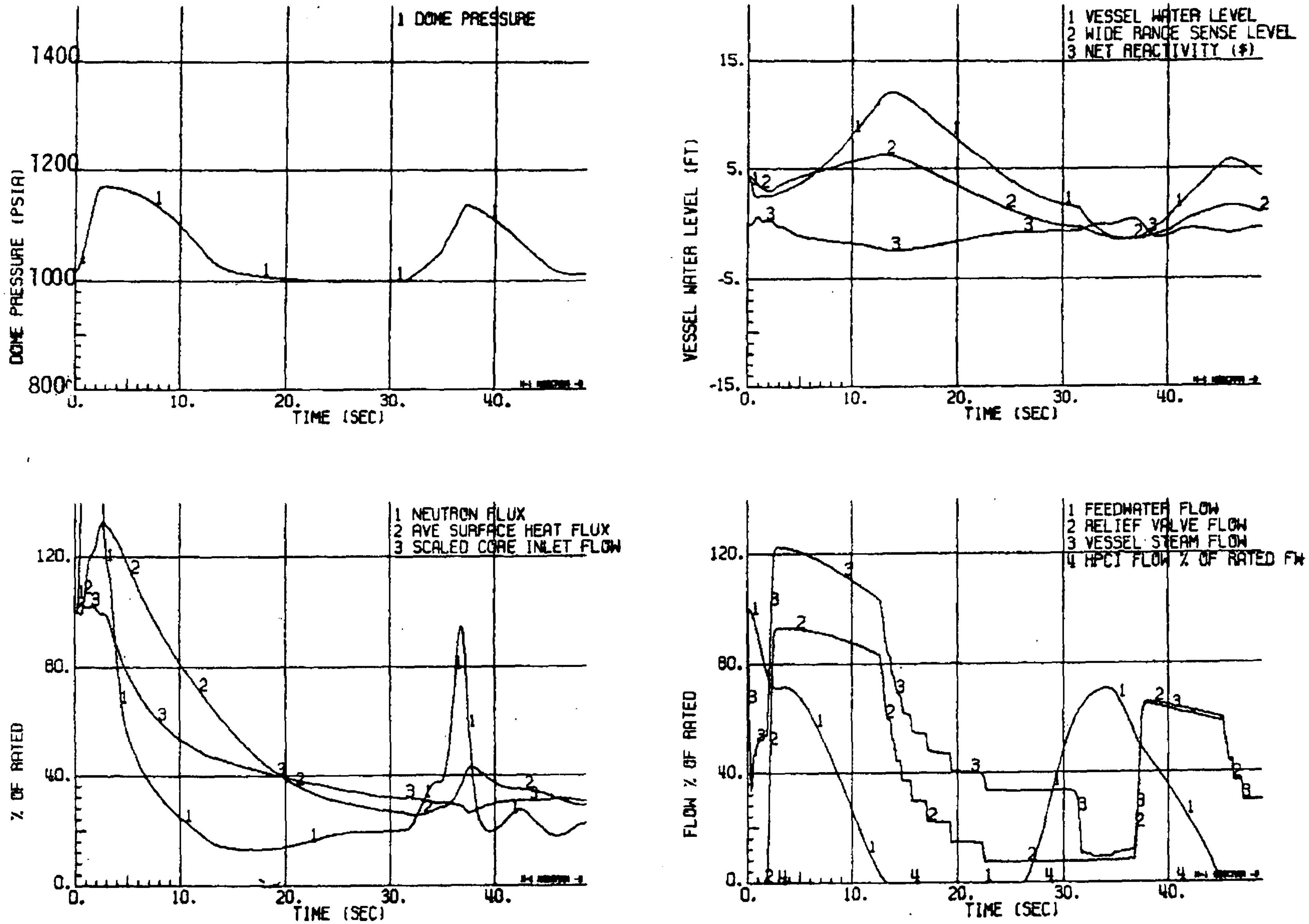


Figure 3.1.5-1. Loss of Condenser Vacuum - ARI Failure

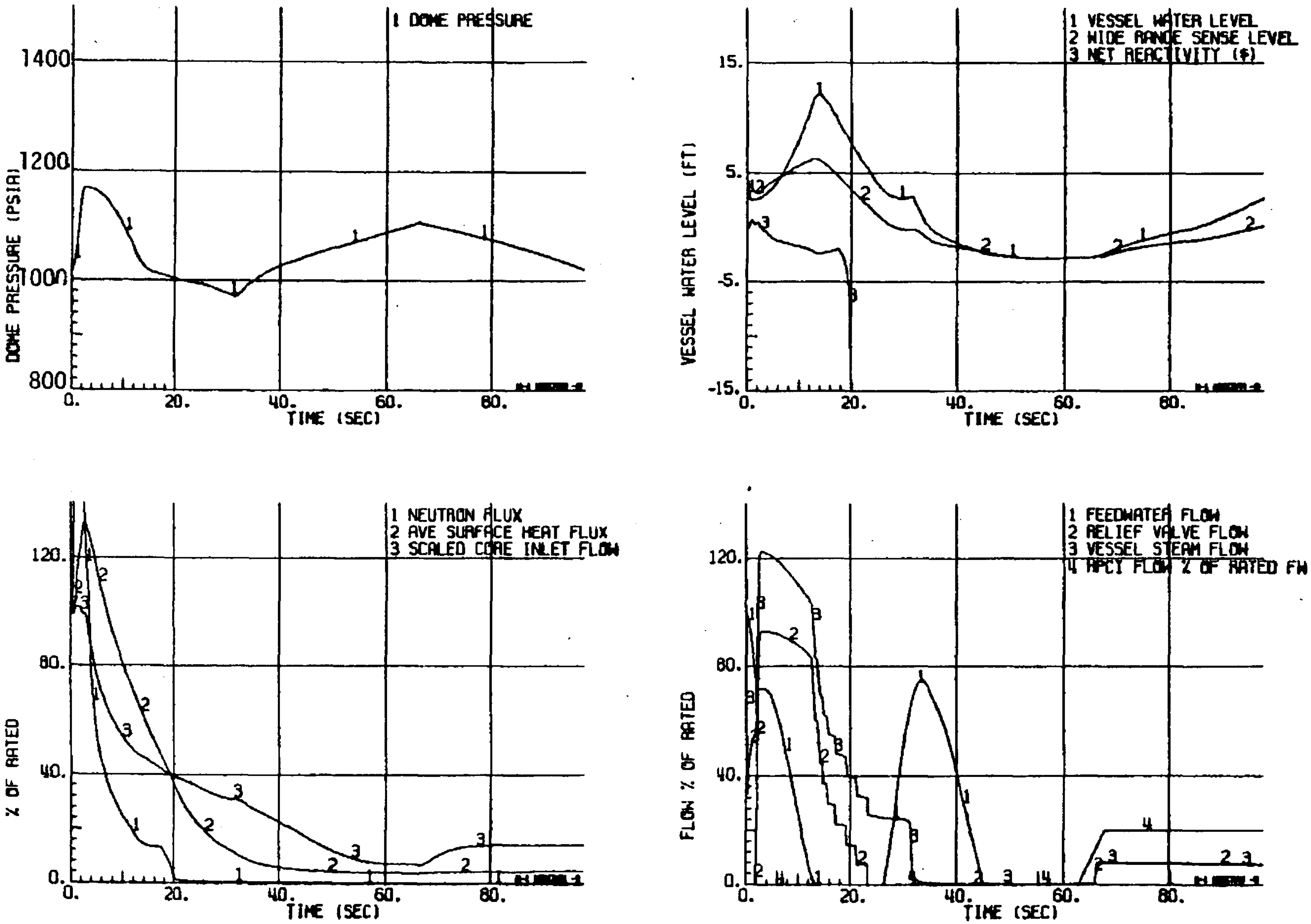


Figure 3.1.5-2. Loss of Condenser Vacuum with ARI

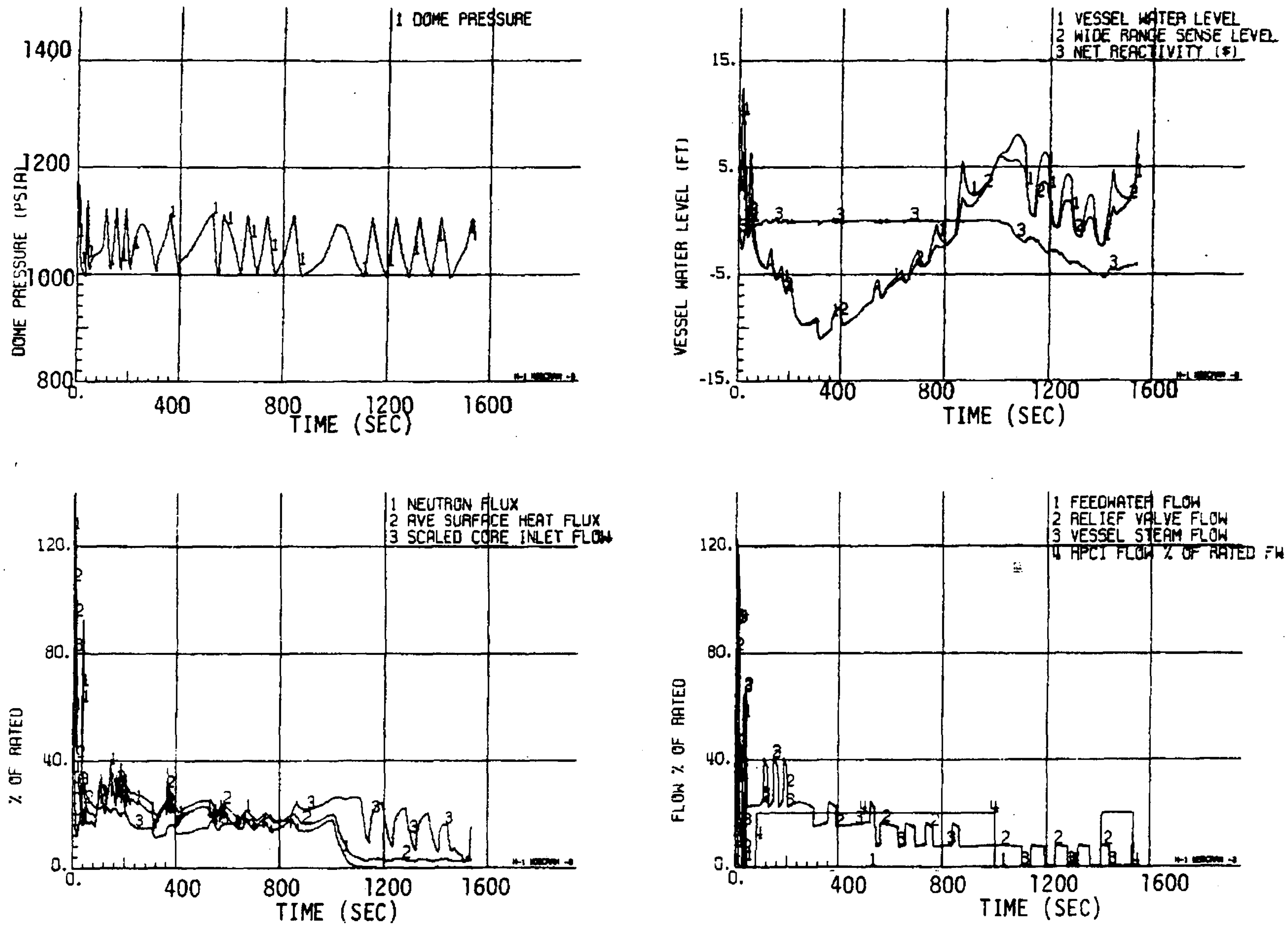


Figure 3.1.5-3. Loss of Condenser Vacuum ARI Failure

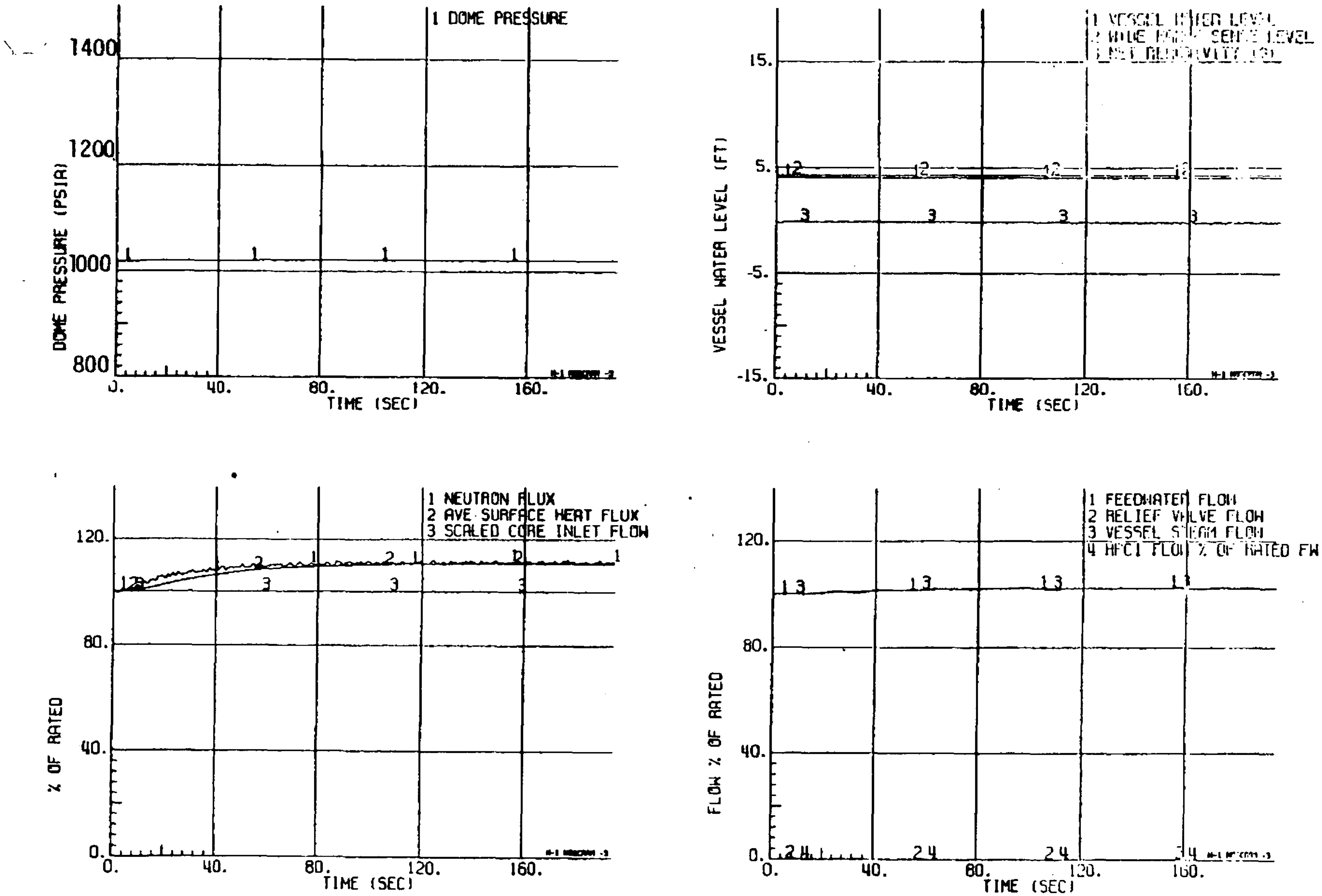


Figure 3.1.7-1. Loss of Feedwater Heater

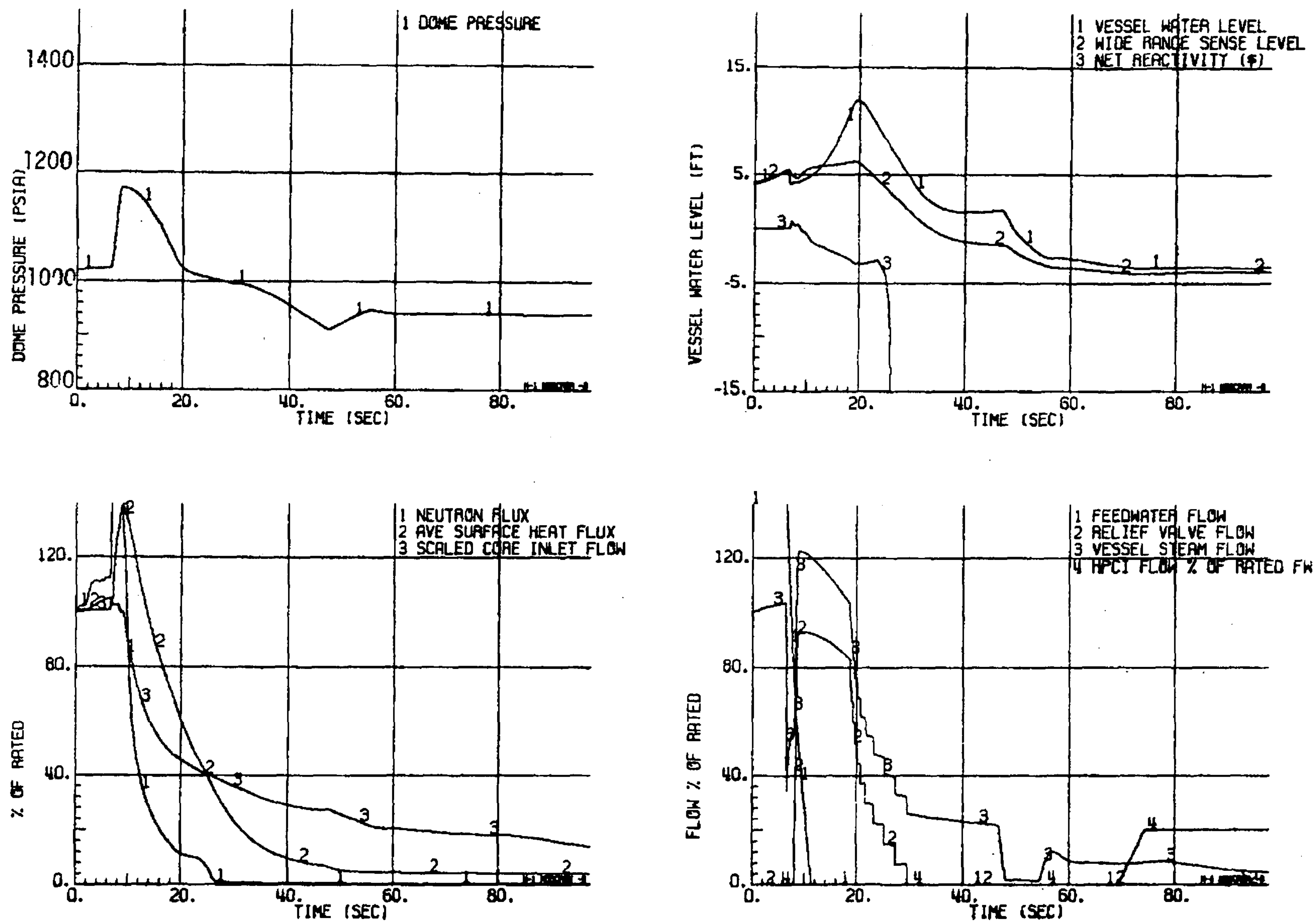


Figure 3.1.8-1. Feedwater Controller Failure - Maximum Demand with ARI

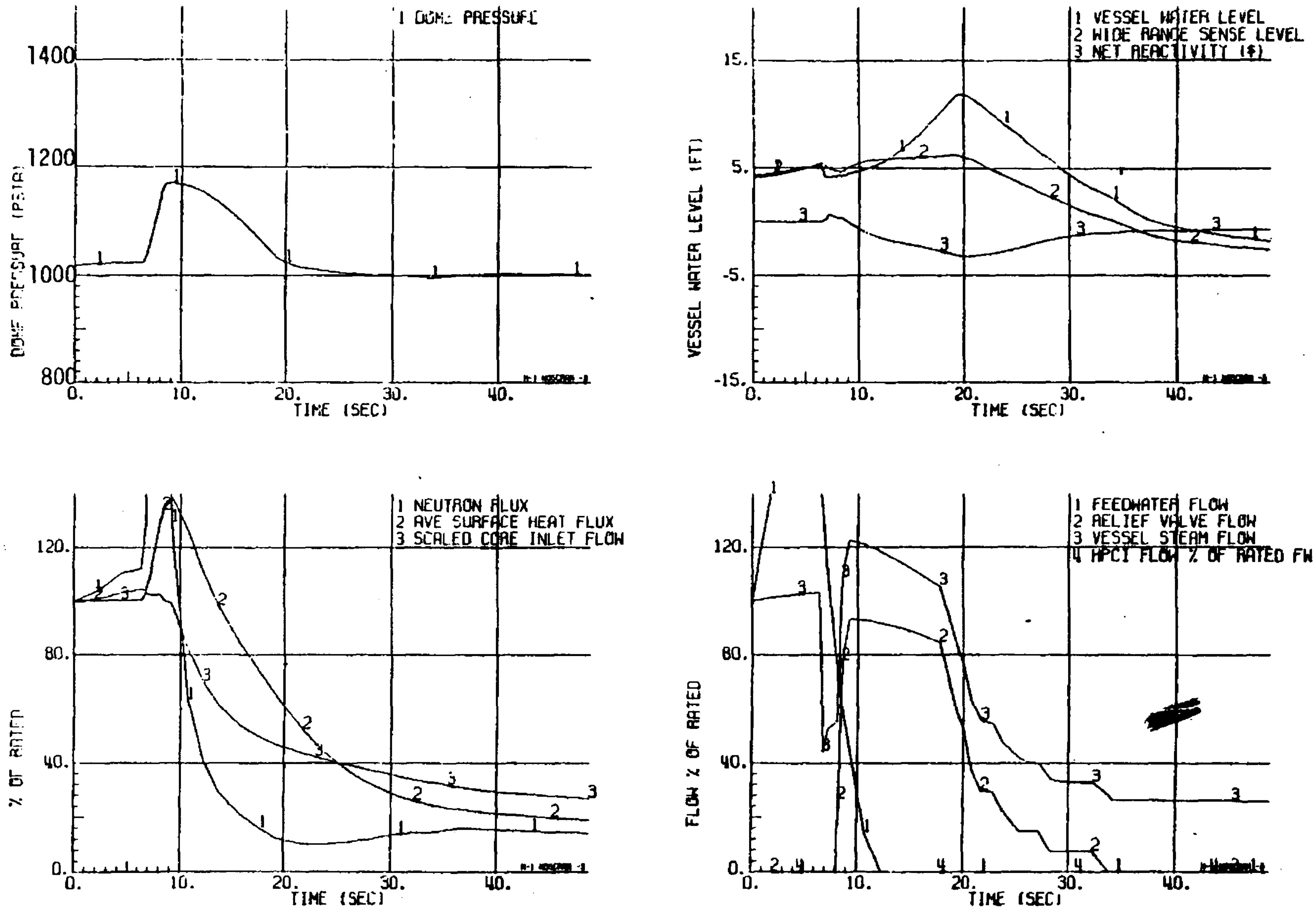


Figure 3.1.8-2. Feedwater Controller Failure - Maximum Demand ARI Failure

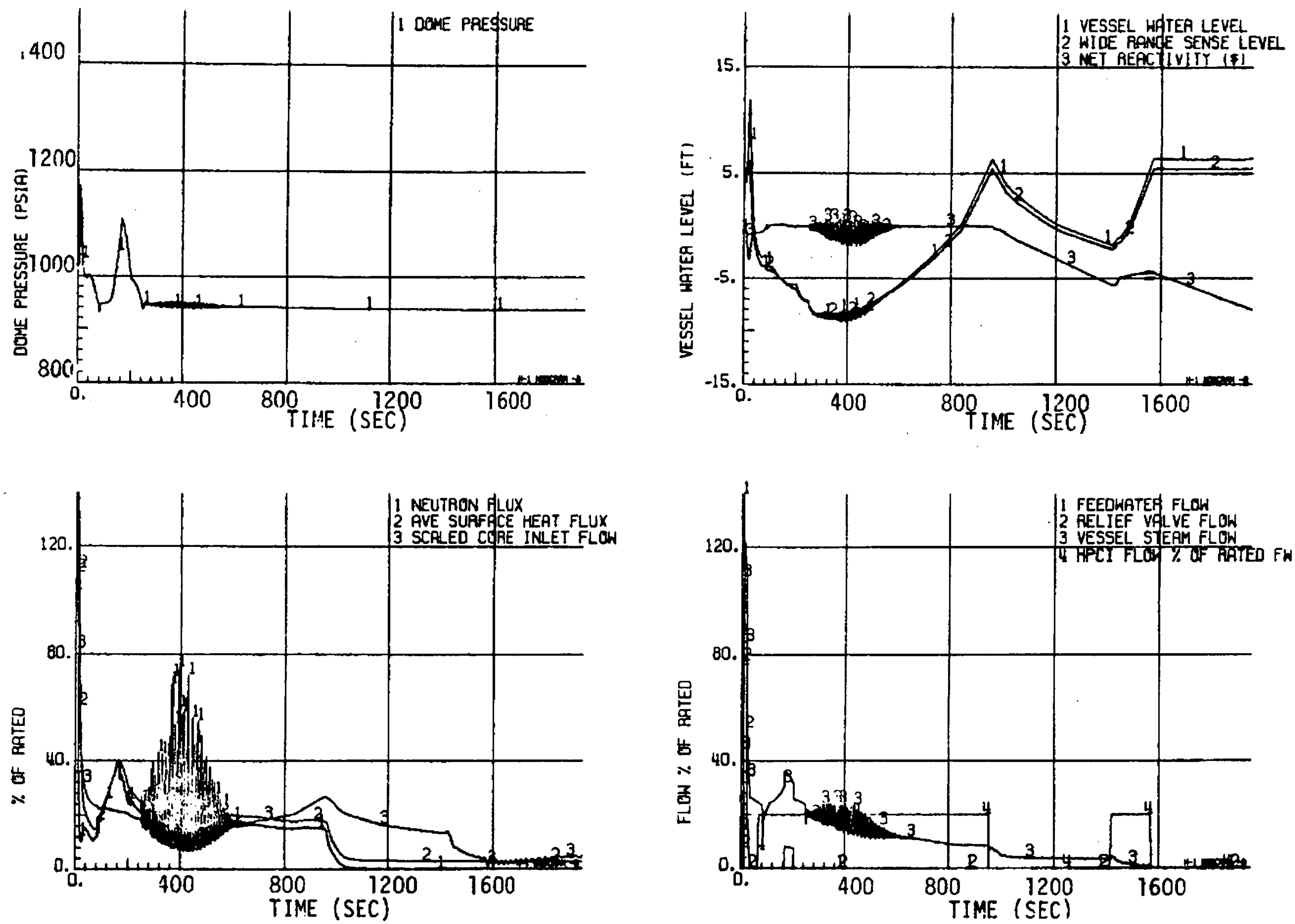


Figure 3.1.8-3. Feedwater Controller Failure Maximum Demand ARI Failure

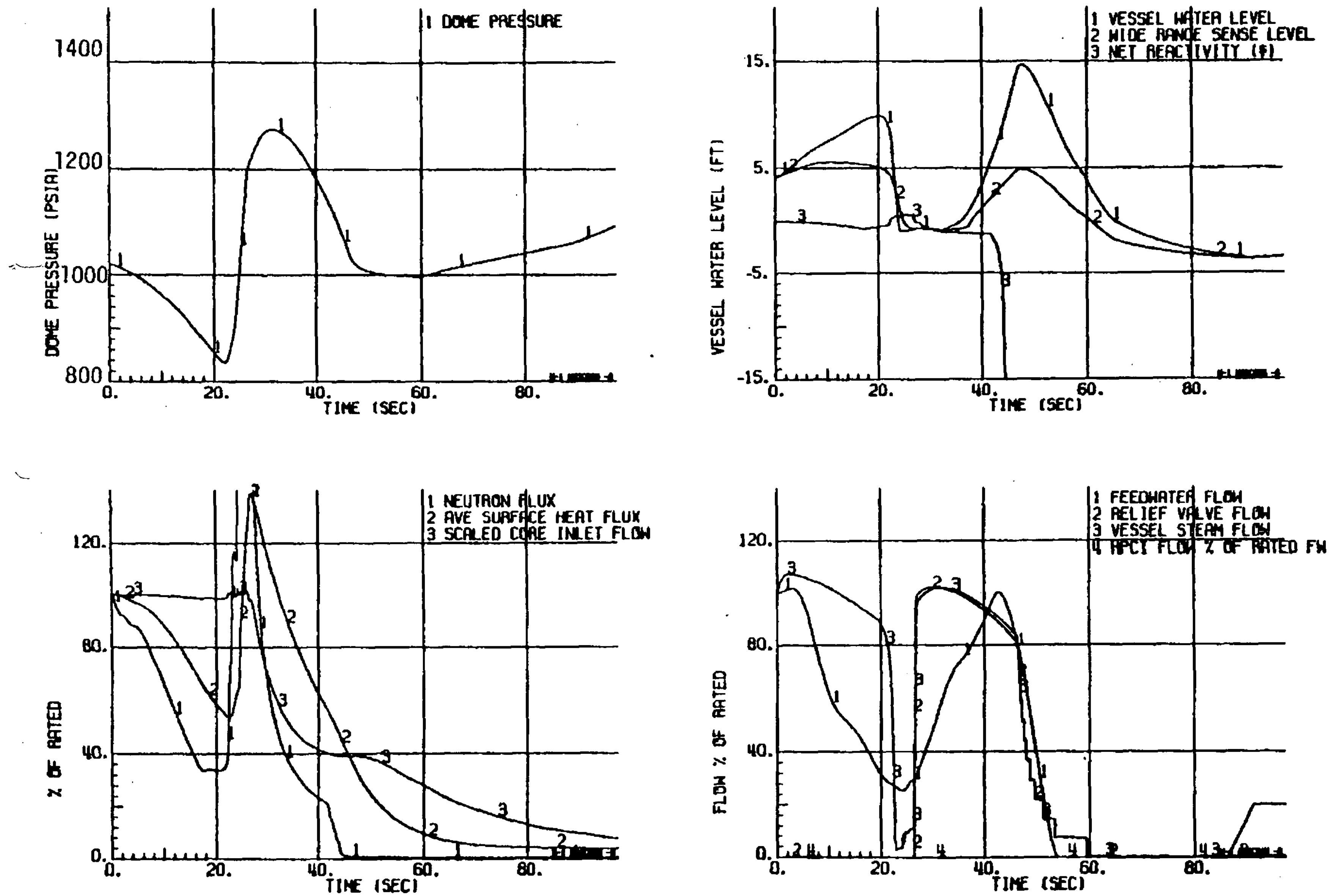


Figure 3.1.9-1. Pressure Regulator Failure - Maximum Steam Demand with ARI

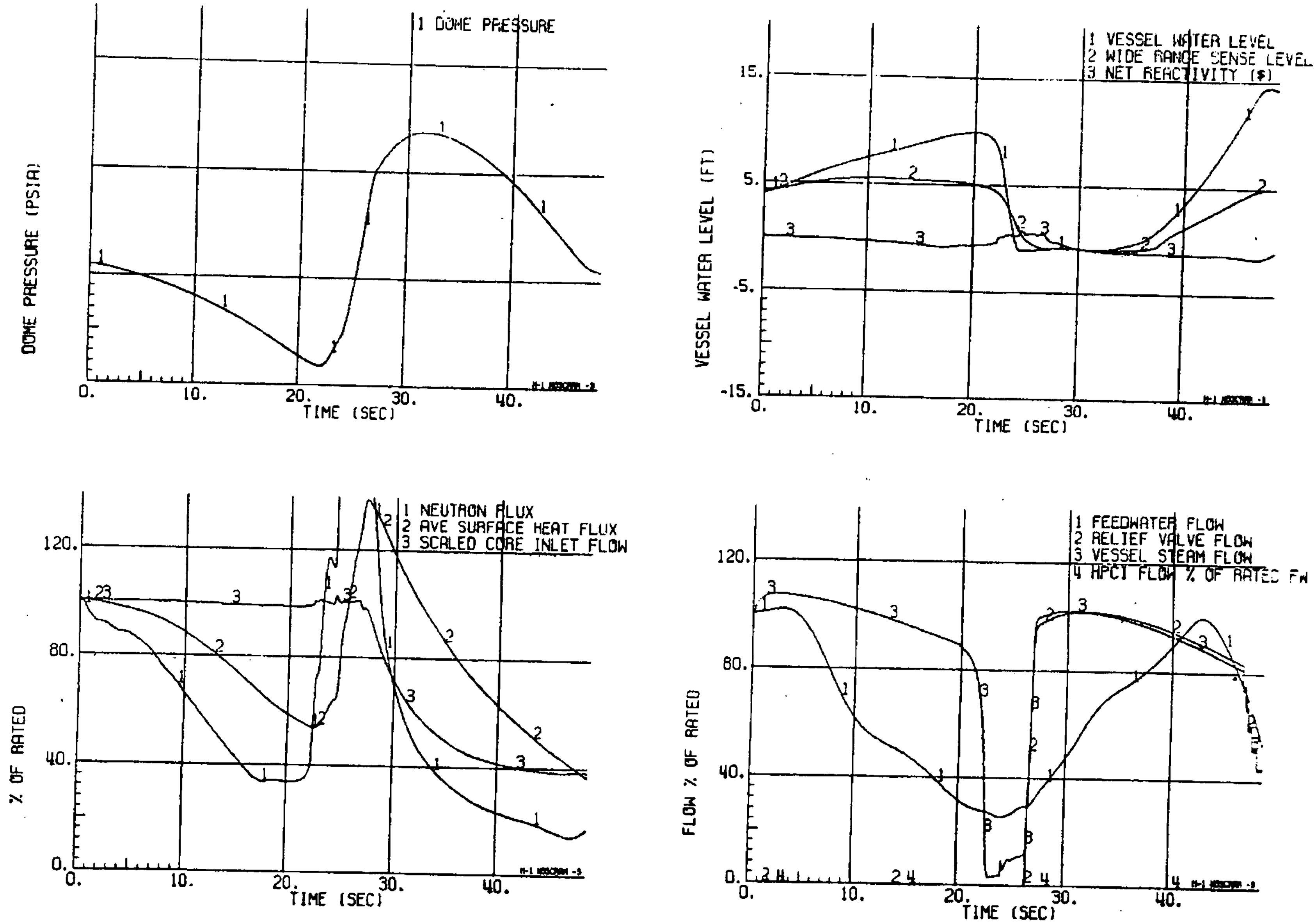


Figure 3.1.9-2. Pressure Regulator Failure - Maximum Steam Demand ARI Failure

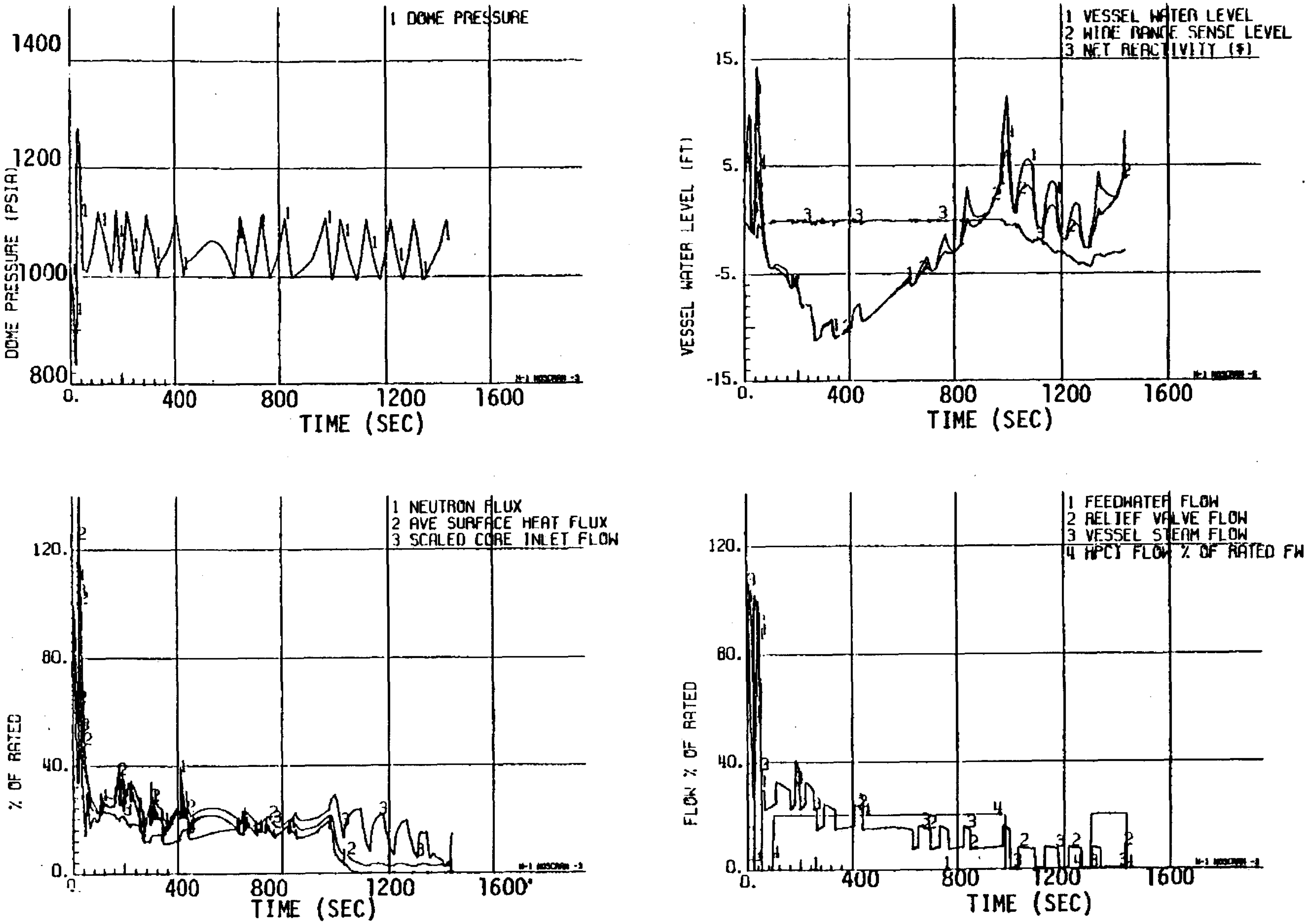


Figure 3.1.9-3. Pressure Regulator Failure - Maximum Steam Demand ARI Failure

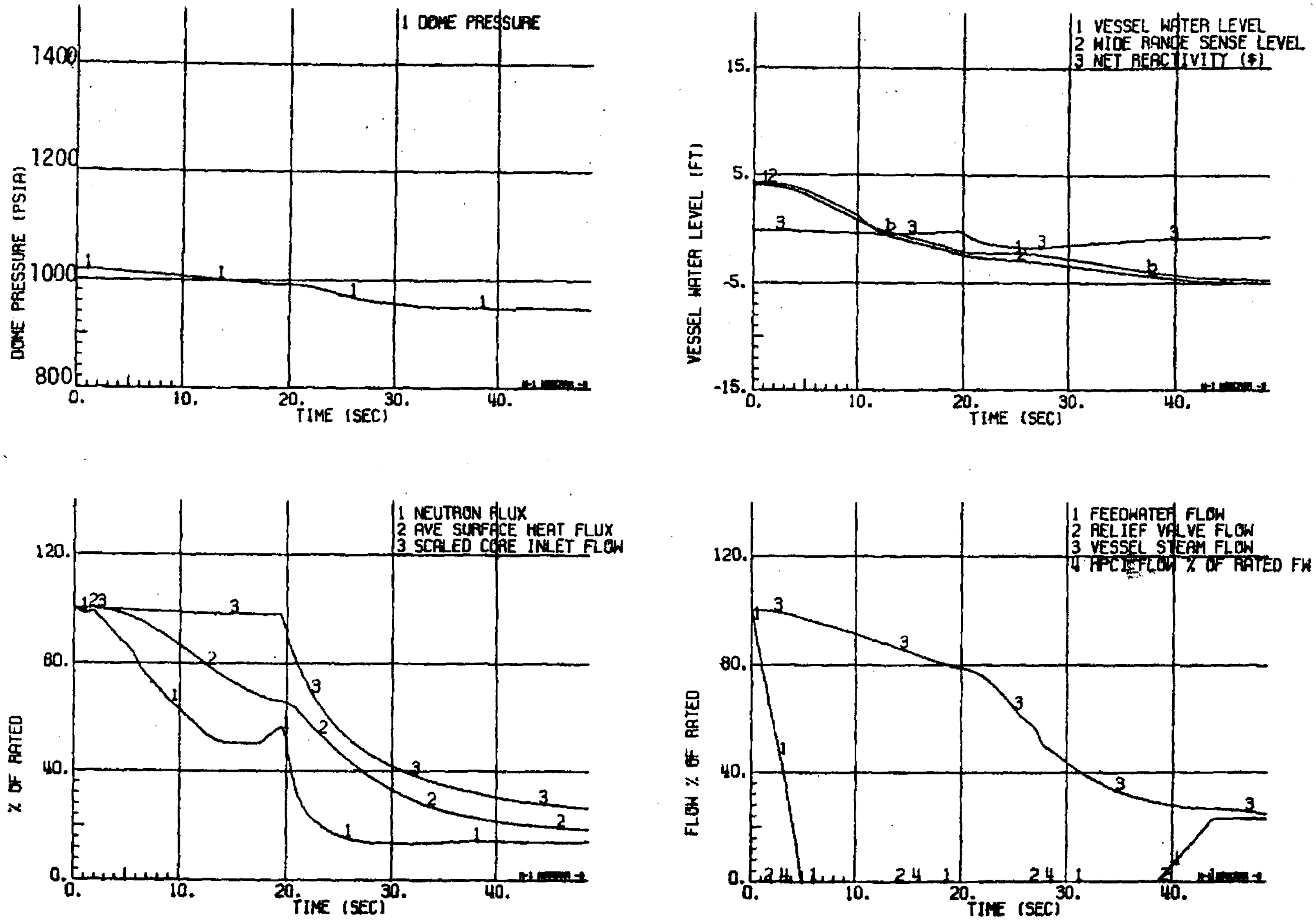


Figure 3.1.10-1. Loss of Feedwater ARI Failure

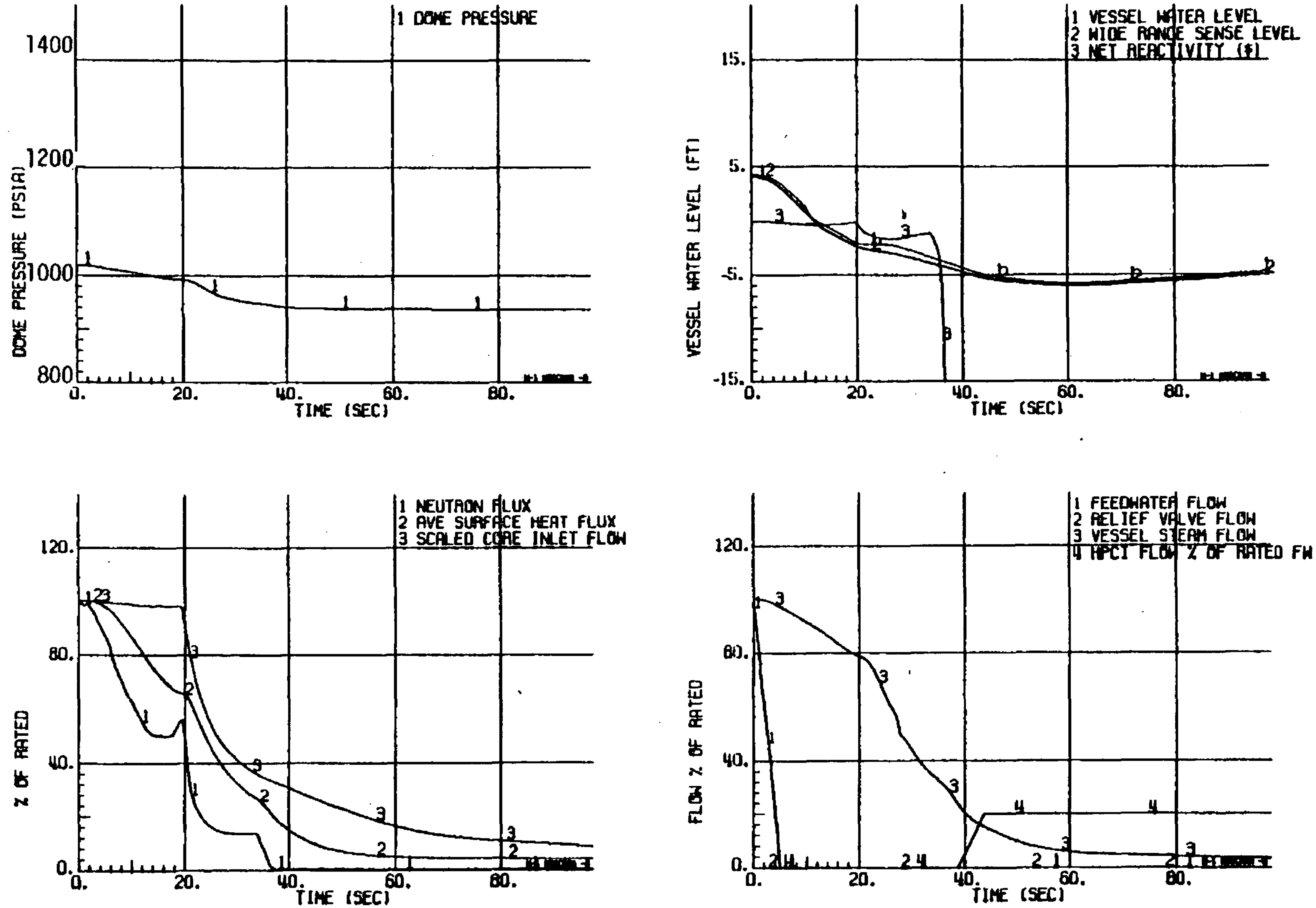


Figure 3.1.10-2. Loss of Feedwater with ARI

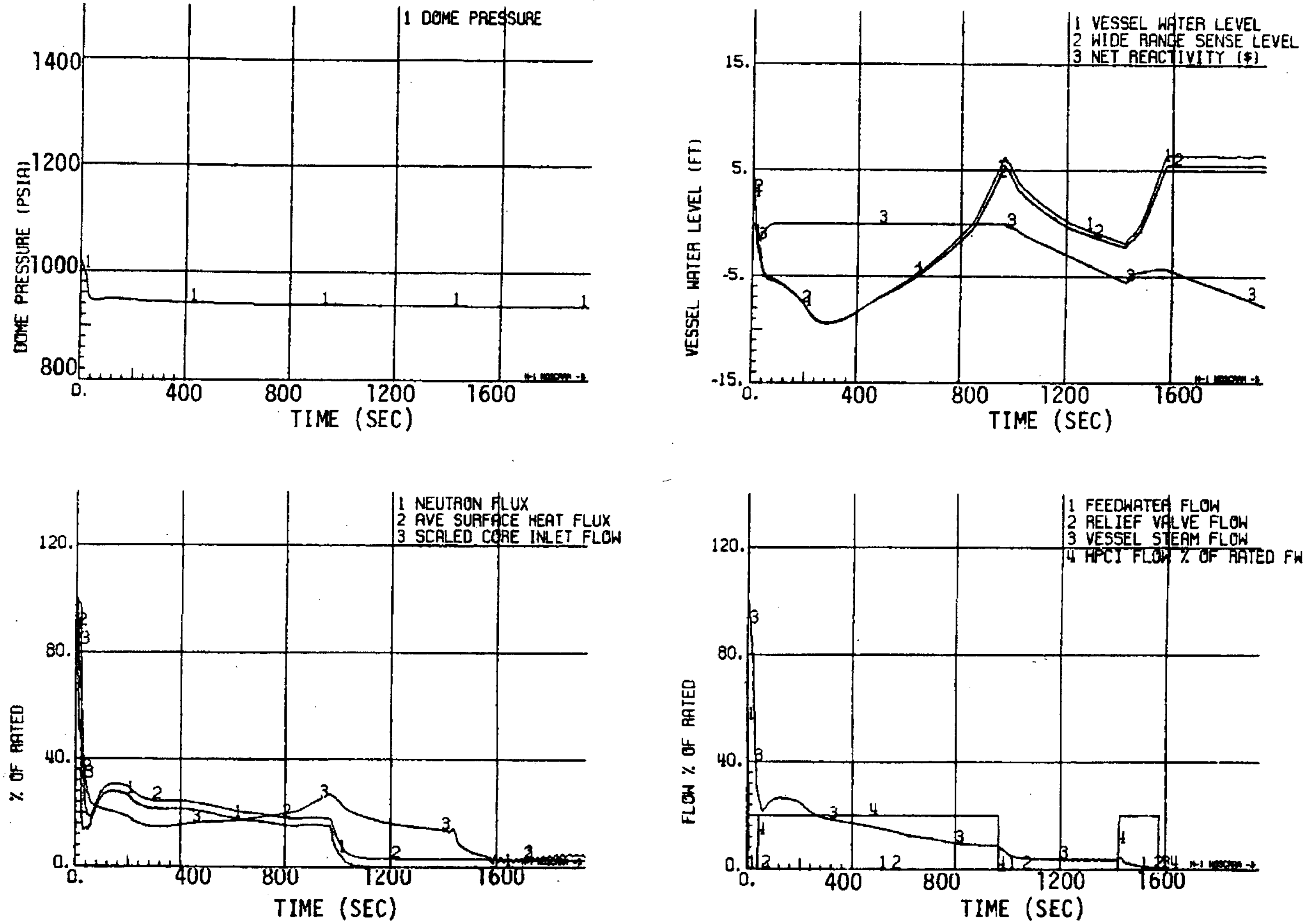


Figure 3.1.10-3. Loss of Feedwater, ARI Failure

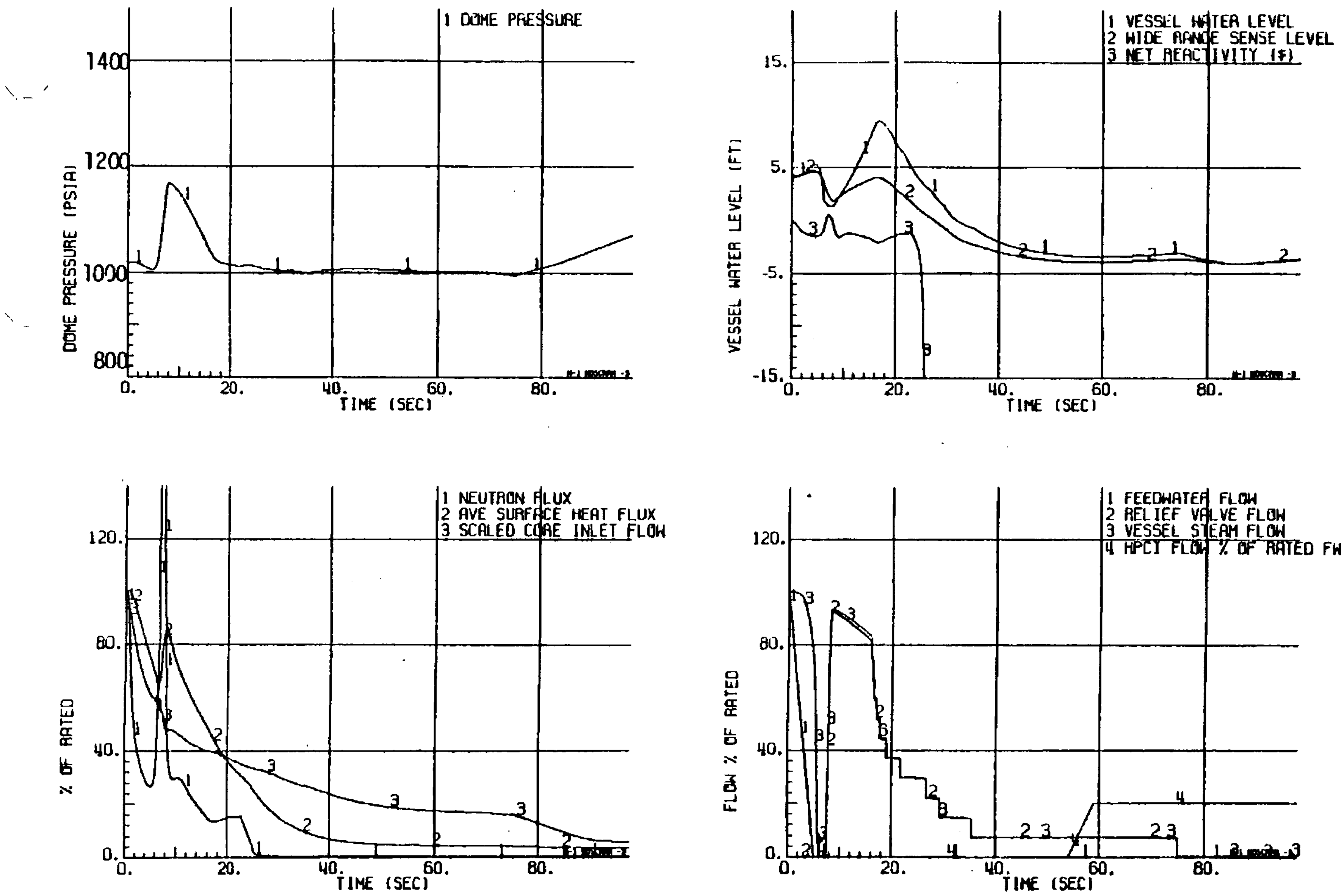


Figure 3.1.11-1. Loss of Normal AC Power, with ARI

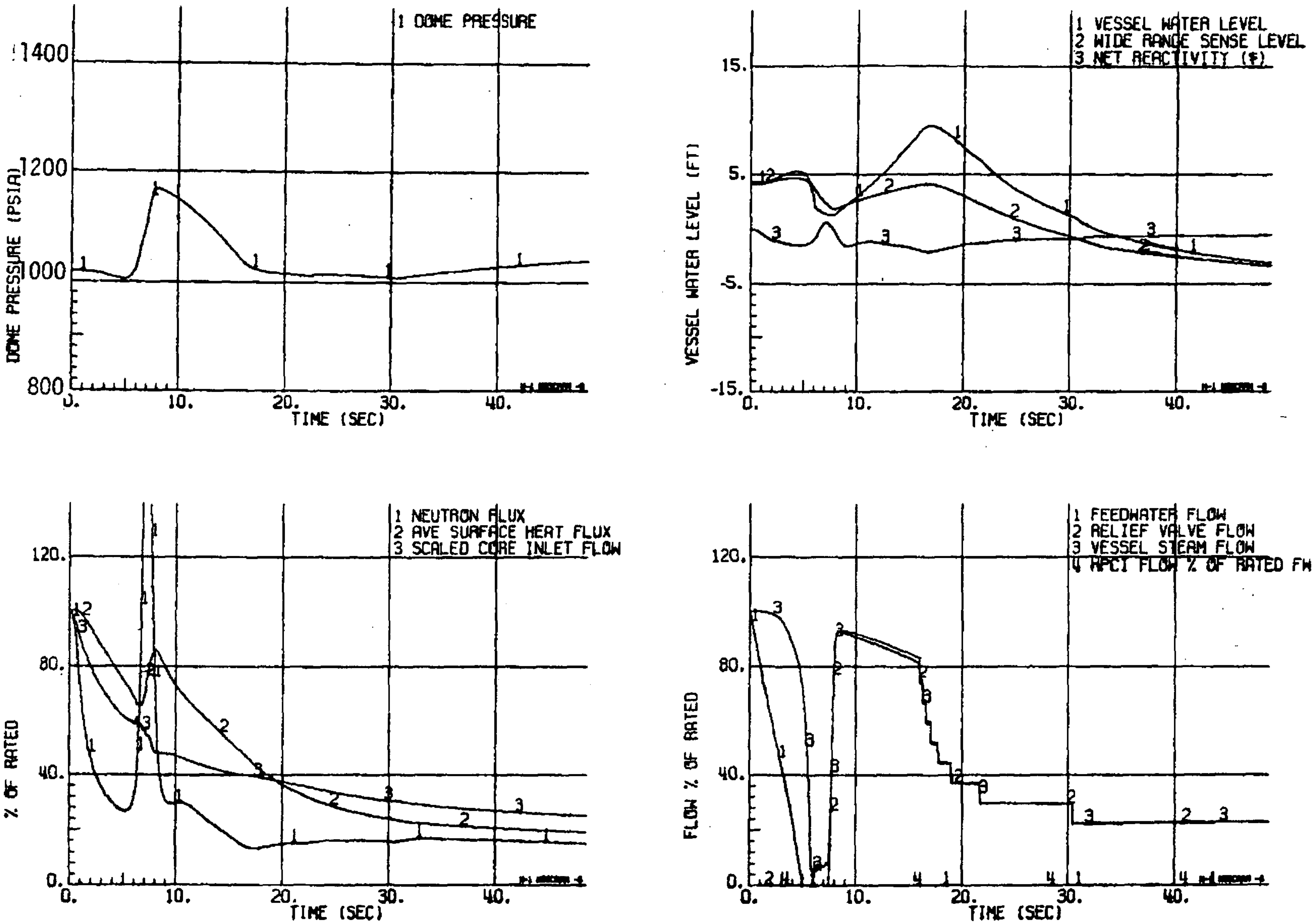


Figure 3.1.11-2. Loss of Normal AC Power, ARI Failure

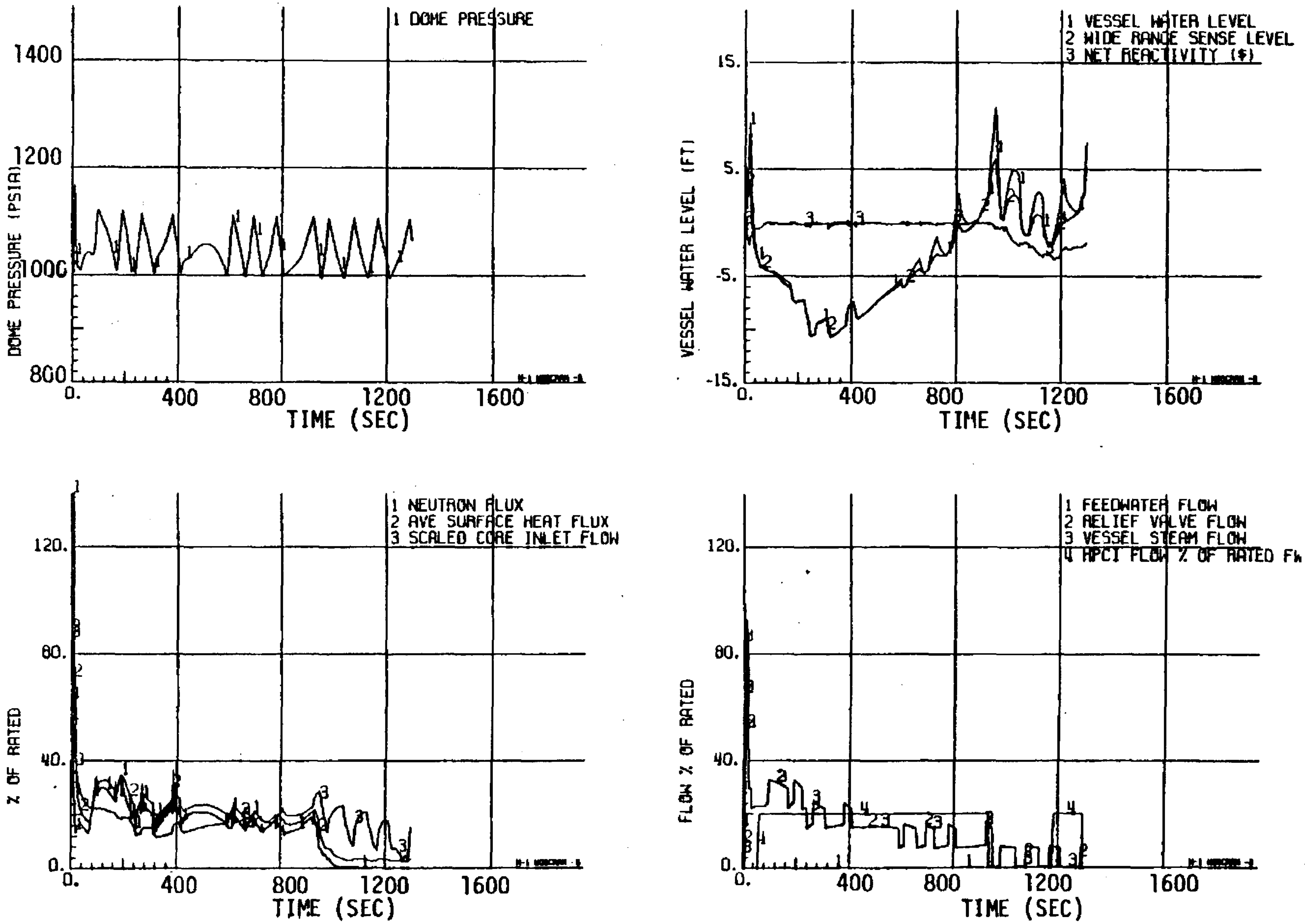


Figure 3.1.11-3. Loss of Normal AC Power, ARI Failure

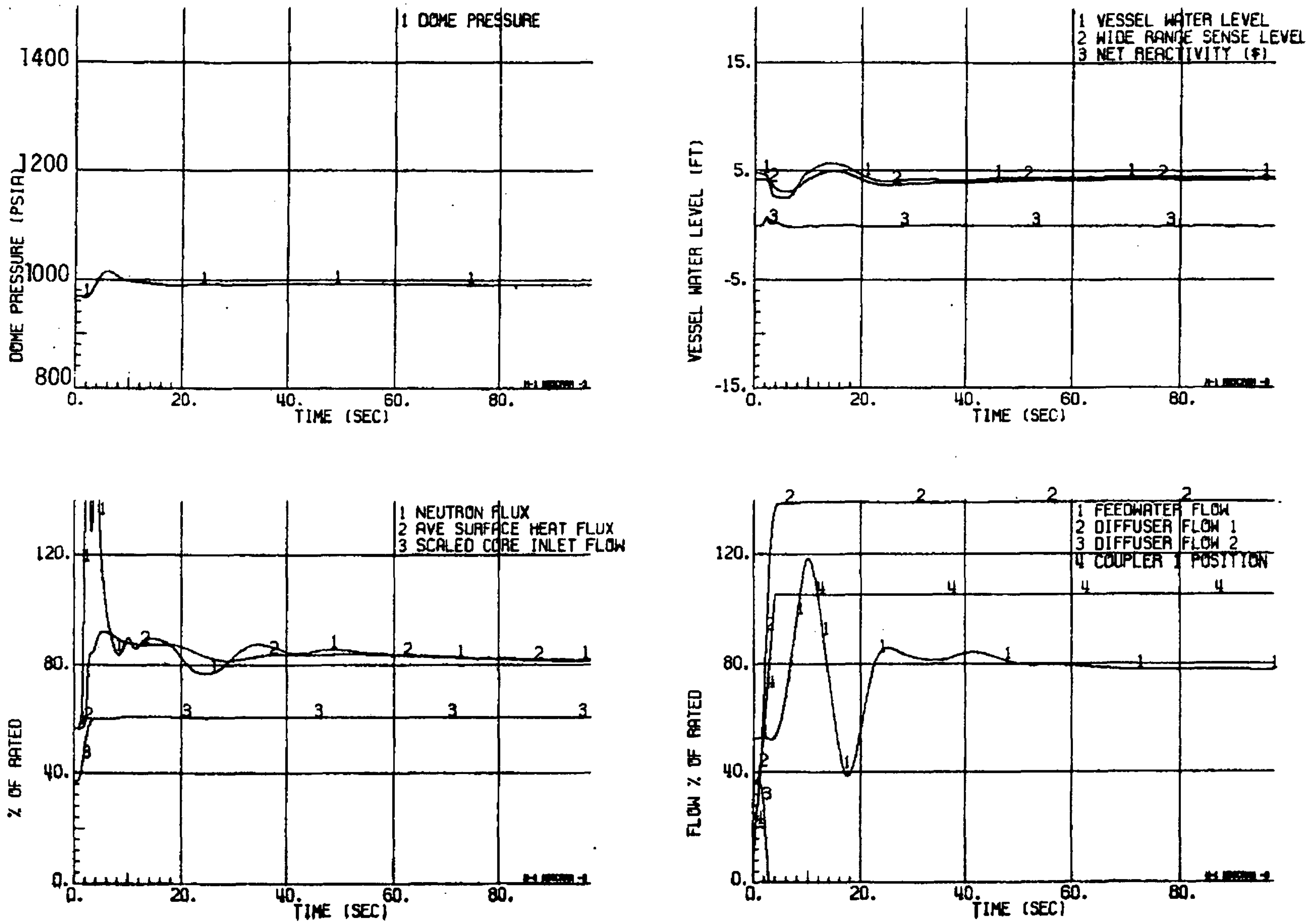


Figure 3.1.12-1. Recirculation Flow Controller Failure Maximum Demand

3.2 RESULTS OF ATWS EVENTS - BWR/5 (Mark II)

3.2.1 MSIV Closure Event

3.2.1.1 Overview of Response Without Scram

The behavior of the plant is separable into an early or short term transient involving a sharp pressure rise and power peak, and a longer term portion that requires evaluation of coolant and containment conditions as the reactor is ultimately brought to shutdown.

The effectiveness of RPT presented in NEDO-10349, NEDO-20626, and Volume 1 are reconfirmed by this analysis. It assists the relief valves in limiting the pressure disturbance acceptably and allows the establishment of a relatively low power generation rate for the long term portion of the transient. Figure 3.2.1-1 illustrates this first period. Peak values for key parameters in the system are given in Table 3.2-1 as well as for other BWR/5 transients. Since Volume 1, several changes have been made to the base case calculations. They include:

a. S/RV reclosure pressure is now 110 psi below the opening pressure setpoint which is more typical of the actual performance.

b. Feedwater flow characteristics due to automatic limiting action or loss following isolation are assumed to result in shutoff 40 seconds after isolation begins.

The ultimate resolution to the lack of scram situation must involve insertion of negative reactivity into the reactor, thereby terminating the long term aspects of the event. ARI is provided as an effective way to mitigate common-cause failures in the logic of the scram system. In the case of its ineffectiveness, the automated SLCS provides further protection and shutdown capability. Coolant inventory is adequately maintained by HPCS and RCIC available on each BWR/5 to replace the coolant loss as steam flow leaves the primary system through the relief valves. Simply adding more water is not a totally satisfactory answer because it also has the effect of raising the power generation rate and the amount of inventory leaving the system as steam, thus increasing suppression pool temperature. The steam reaching the suppression pool continues to heat it and pressurize the containment until the power generation/steam flow can be reduced to the RHR capacity and/or finally terminated. The RHR (pool cooling mode) ultimately cools the pool and eventually the reactor also (shutdown cooling mode) if the MSIV's cannot be reopened establishing flow to the main condenser (the preferred method of cooldown).

3.2.1.2 Sequence of Events for MSIV Closure

The MSIV closure transient provides some of the most severe conditions following a postulated failure to scram. Listed in Table 3.2.1-1 in sequence of occurrence are significant points of the transient with representative times when the highlight occurs.

The sequence of events begins with the closure of the MSIV's in 4 seconds. With motion of the MSIV's, the pressure immediately begins to rise resulting in a reduction in void fraction and rapid increase in power. This sequence of events is shown in Table 3.2.1-1. This power reaches a maximum of 614% of the initial value at 4 seconds into the event and rapidly decreases thereafter. At 4 seconds, the setpoint pressure of the relief valves is reached and they begin to open to arrest the pressure rise. Shortly after 4 seconds, the vessel dome pressure reaches 1150 psig, the maximum RPT trip point; both recirculation pumps trip. A delay of 530 milliseconds exists from the time the pressure reaches 1150 psig until the time that recirculation pump trip occurs. This delay time (500 milliseconds delay in the sensor and 30 milliseconds in the logic and trip) is consistent with industry experience. At the same time that the recirculation pump trip occurs, the logic chain is activated which would initiate ARI.

Pressure continues to rise for a short period of time until, at approximately 7 seconds into the event, it reaches its peak and begins to decrease. The maximum pressure at the vessel bottom is 1247 psig at 7 seconds. For those plants which have turbine-driven feed pumps, they will begin to coast down as soon as the MSIV's are closed and will have lost their ability to overcome vessel pressure head at 20 seconds. For this analysis, it was assumed that a motor-driven feedwater system was available and the feedwater shutoff occurred slightly later (near 45 seconds) as the ATWS feedwater limiter was activated. The relief valves begin to close near 22 seconds; pressure is then stabilized at the relief valve setpoint. This part of the transient is shown in Figure 3.2.1-1.

The same pressure signal (1150 psig) that initiated RPT will cause the opening of valves on the scram air header (ARI) which allows the air pressure in the header to bleed down. In the event that scram has not already occurred from any of the several available signals, this reduced pressure allows the scram discharge valves to open and the control rods to insert. Tests have shown that the pressure in the header will have been reduced sufficiently in 15 seconds to allowthe control rods to insert. All rods will be fully in the core after 4 additional seconds. ARI completely mitigates the ATWS situation and 25 seconds after the event begins, it is essentially over. Following ARI, normal shutdown procedures are utilized to bring the plant to cold shutdown. Figure 3.2.1-2 shows the expected course of the event. Water level drifts downward as decay energy generates small amounts of steam, and when Level 2 is reached, the HPCS and RCIC automatically start and replenish the vessel water inventory from the condensate storage tank, up to the high level trip. They will then reset themselves and continue to supply water to the vessel inventory as necessary.

If the ARI is not effective, the BWR/5 is still able to mitigate the event. With an assumed ARI failure and zero feedwater flow, at 39 seconds the level of the bulkwater in the vessel will decrease to Level 2, the level at which HPCS and RCIC are initiated. Near 1 minute, water from these systems will begin to enter the reactor vessel.

Following confirmation from the flux monitoring system and the rod position indicating system that scram has not taken place, the SLCS will activate. This system will be started 2 minutes after the ATWS high pressure signal; boron will reach the core after an additional 1 minute of transport time in the lines and the vessel. Therefore, nuclear shutdown begins at 3 minutes into the event using the SLCS. With an 86 GPM volumetric flow rate of sodium pentaborate (in general dependent on vessel size), the reactor will be brought to hot shutdown in approximately 22 minutes from the beginning of the event. This can be seen in the lower left hand graph of the long-term plot of this event, Figure 3.2.1-3. The behavior of several other parameters is also depicted in Figure 3.2.1-3.

Bulkwater level within the vessel continues to decrease until approximately 5 minutes at which time HPCS and RCIC supply more water than is required to make up for steam flow out of the vessel. At this time the vessel level reaches its lowest level and begins to rise. It is important to note that adeqaute core cooling is maintained at all times. As the level is increased, core flow is increased, thereby reducing the average void fraction. The various contributors to reactivity insertion and power production (boron, voids, etc) must always be in balance with the power production. Water level is completely restored by the HPCS and RCIC which cycle on at low level (L2) and off at high level (L8) to maintain adequate level in the vessel. A larger scale plot of water level is shown in Figure 3.2.1-4.

Following hot shutdown, the decay power will continue to generate a small amount of steam which will continue to cycle the relief valves. At 46 minutes the suppression pool temperature will reach its maximum value of 179°F. The maximum containmentpressure is 9.1 psig. For the case where the ARI functions as expected, the maximum suppression pool temperature is 138°F and it occurs approximately 4 hours after the event. Figures 3.2.1-5 show long-term containment conditions for these cases. Either way, the event is maintained within the prescribed limits.

Thus it can be seen that an MSIV closure event combined with a failure to scram is adequately mitigated for a representative BWR 5/Mark II.

3.2.2 Turbine Trip

3.2.2.1 Overview of Response Without Scram

The overview given for the MSIV closure event is generally applicable to the Turbine Trip event. The key difference is that the main condenser remains available for this event. From the time that steamflow is within the bypass capacity, the main condenser will be used to remove the steam from the vessel. This base case event has also been updated as given in Section 3.2.1.

3.2.2.2 Sequence of Events for Turbine Trip

The turbine trip event begins with the rapid closure of the turbine stop valves and the resultant opening of the turbine bypass valves. After the stop valves close in 0.1 seconds, the pressure immediately begins to rise which results in a reduction in void fraction and rapid increase in power. The sequence of events is shown in Table 3.2.2-1. This power reaches a maximum of 426% of the initial value at one second into the event and rapidly decreases again. At approximately 1.5 seconds, the setpoint pressure of the relief valves is reached and they begin to arrest the pressure rise. Shortly after 2 seconds, it is expected that some of the fuel will experience boiling transition, however, coolable geometry is maintained. At about the same time, the vessel dome pressure reaches 1150 psig, the maximum RPT point; both recirculation pumps trip. In this analysis, the earlier trip of recirculation pumps directly from the stop valve closure was conservatively neglected. It makes the early event results even milder. At the same time that RPT occurs the logic chain is activated to start ARI.

Pressure will continue to rise until 3 seconds when it peaks and begins to decrease. The maximum pressure occurs at the vessel bottom and is 1192 psig at 2.7 seconds. Although the feedwater pumps remain available for the turbine trip case, it is necessary to reduce the amount of power produced. Therefore, the feedwater flow will be limited to a minimum flow value which for this design has been chosen to be zero. This minimizes power generation and resultant steam discharged to the suppression pool. The relief valves began to close very early in this transient (about 9 seconds) and close for the last time in less than 2 minutes even with failure of ARI. The remainder of the generated steam flows through the bypass to the main condenser. The first portion of this transient with and without ARI is shown in Figures 3.2.2-1 and 3.2.2-2.

The same pressure signal (1150 psig) that initiated RPT will casue the opening of valves on the scram air header (ARI) which allows the air pressure in the header to bleed down. In the event that scram has not already occurred from any of the several available signals, this reduced pressure allows the scram discharge valves to open and the control rods to insert. Tests have shown that the pressure in the header will have been reduced sufficiently in 15 seconds to allow the control rods to insert. All rods will be fully in the core after 5 additional seconds. ARI completely mitigates the ATWS situation and 25 seconds after the event began, it is over. Since in this event feedwater is not los and the runback of feedwater discussed earlier does not occur until the failure of both normal scram and ARI are confirmed, the feedwater system will continue to function and provide water to the reactor.

If for some reason the ARI is also not effective, the BWR/5 is still able to mitigate the event. With an assumed ARI failure and feedwater flow now having reached zero, at 55 seconds the level of the bulkwater in the vessel will decrease to level 2, the level of which HPCS and RCIC are initiated. At 20 seconds later, water from these systems begins to enter the reactor vessel.

Following confirmation from the flux monitoring system and the rod position indicating system that scram has really not taken place, the SLCS will activate. This system will be started 2 minutes after the ATWS high pressure signal; one minute of transport time is also accounted for in the lines and the vessel. Therefore, nuclear shutdown begins at 3 minutes into the event using the SLCS. With an 86 GPM volumetric flow rate of sodium pentaborate, the reactor will be brought to hot shutdown in approximately 23 minutes from the beginning of the event. The behavior of several parameters is depicted in Figure 3.2.2-3 for the long-term event.

Bulkwater level within the vessel continues to decrease until approximately 4.5 minutes at which time HPCS and RCIC supply more water than is required to make up for steamflow out of the vessel. At this time it reaches its lowest level and begins to rise. As the level increases fore flow is increased, thereby reducing the average void fraction. The various contributors to reactivity insertion and power production (boron, voids, etc.) must always be in balance with the power production. Water level is completely restored by HPCS and RCIC at approximately 16 minutes. A detailed plot of water level is shown on Figure 3.2.2-4. Water level continues to cycle with HPCS/RCIC maintaining an adequate level in the vessel.

Following hot shutdown, the decay power continues to generate a small amount of steam which flows through the bypass to the main condenser. Since the major portion of the steam generated in this event goes to the main condenser, the temperature rise in the suppression pool is minimal. The maximum suppression pool temperature calculated in this case is 104°F which results in a maximum containment pressure of 0.7 psig.

Thus it can be seen that a Turbine Trip with Bypass event combined with a failure to scram is adequately mitigated for a representative BWR 5/Mark II.

3.2.3 Inadvertent Open Relief Valve

3.2.3.1 Overview of Response Without Scram

This event has no rapid excursion as the previous two events, but is merely a long term depressurization. RPT does not occur until late in the event after hot shutdown is achieved.

Except for steamflow through the open relief valve and the use of the liquid boron solution for shutdown, the nuclear steam supply system is in a normal operating state. The suppression pool is the only system exposed to off-normal conditions. This base case event has also been updated as given in Section 3.2.1. The sequence of events follows.

3.2.3.2 Sequence of Events for Inadvertent Open Relief Valve

This event begins when one of the primary relief valves on the main steamlines inadvertently opens without influence from any other portion of the system. All pressure levels in the reactor coolant pressure boundary are at a nominal value prior to the event. The resulting sequence of events is shown in Table 3.2.3-1.

At the time that the relief valve opens, there is a momentary depressurization (a few seconds) until the turbine pressure control senses it and closes slightly (dropping unit electrical output) to control pressure. For general application of this analysis, a relief valve capacity of 8.3% NBR rated was utilized (the nominal flow of a valve on a BWR/5-218 inch vessel plant). See Section 3.2.4 for sensitivity analyses relative to this valve size. After about two minutes, the suppression pool temperature, which was initially assumed to be at 90°F, has risen to the alarm point of 95°F. It attempts to reclose the valve are unsuccessful, the operator will turn on the RHR system in the pool cooling mode to maintain pool temperature. If attempts to close the valve continue to be unsuccessful, the temperature will continue to rise and at 7.5 minutes will reach 110°F at which point the operator is required to manually scram the reactor. For this example case, the manual scram also activates the ATWS protection paths to ARI. The logic paths shown in Section 2.4.2 are utilized.

If neither normal manual scram nor the ARI are effective, the BWR/5 is still able to mitigate the event. The ATWS logic would have determined that the control rods are not inserted and at 9.5 minutes into the event the SLCS will be activated.

For this case with the recirculation pumps operating, the boron mixing efficiency is excellent (95% is assumed), and the delay time inside of the vessel is small, so that at 10 minutes the control liquid reaches the core and shutdown begins. Within 19 minutes, the power has been reduced to the point that the amount of steam generated is less than the relief valve capability and the pressure now begins to decrease more rapidly. The turbine control valves have closed complete due to the pressure decrease.

These events are depicted in Figure 3.2.3-1. By 21 minutes, the pressure has dropped to the low steam line pressure isolation set point and the MSIV's close. Simulating plants with turbine-driven feedwater pumps, the feedwater is assumed to be lost within 20 seconds of the isolation. This causes the water level in the vessel to decrease and at 25 minutes the low level point (L2) is reached where the recirculation pumps are automatically tripped and the HPCS and RCIC are activated. These systems are shown to automatically cycle on at low level (L2) and off at high level (L8) as specified to maintain water inventory in the vessel, although manual action is expected to maintain level with the RCIC alone. Depressurization of the vessel will continue with the relief valve discharging into the suppression pool; the maximum pool temperature of 187°F will occur at about 1 hours. The peak containment pressure of 10.6 psig occurs at the same time, well below the 46 psig design pressure for the Mark II containments.

In cases where ARI is activated (8 minutes), the maximum pool temperature is 167°F.

Thus it can be seen that the inadvertent opening of a relief valve event combined with a failure to scram is adequately mitigated for a representative BWR 5/Mark II.

3.2.4 Sensitivity Study Results - BWR/5 Base Cases

A wide variety of parameters were studied to examine the sensitivity and potential impact of plant differences and/or uncertainties on the results of the three BWR/5 base cases. While the overall objective of these sensitivity studies is to provide guidance for assessing the adequacy of plants having certain parameters different from the generic analyses, caution must be exercised when combining the results of several parameter variations, due to the non-linearities involved (see Section 3.3.4.4).

3.2.4.1 MSIV - ATWS Sensitivity Studies

3.2.4.1.1 Variation of Boron Delay
3.2.4.1.2 Variation of Boron Capacity/Mixing
3.2.4.1.3 Variation of HPCS/RCIC Capacity
3.2.4.1.4 Variation of RHR Capacity
3.2.4.1.5 Variation of RHR Delay
3.2.4.1.6 Variation of Pool and Service Water Temperature
3.2.4.1.7 Variation of RHR Capacity and Service Water
3.2.4.1.8 Variation of Pool Size
3.2.4.1.9 Variation of S/RV Capacity
3.2.4.1.10 Variation of RPT Delay
3.2.4.1.11 Variation of RPT Inertia
3.2.4.1.12 Effect of Partial Rod Insertion
3.2.4.1.13 Variation of Void Coefficient
3.2.4.1.14 Variation of Doppler Coefficient

Tables 3.2.4-1, 2, and 3 summarize the results for this event. Figures 3.2.4.1-1 and 3.2.4.1-2 also graphically show the results.

3.2.4.2 Turbine Trip Sensitivity Studies

3.2.4.2.1 Variation of Boron Delay
3.2.4.2.2 Variation of Boron Capacity/Mixing
3.2.4.2.3 Variation of HPCS/RCIC Capacity
3.2.4.2.4 Variation of RHR Delay
3.2.4.2.5 Variation of Void Coefficient
3.2.4.2.6 Variation of Doppler Coefficient

Table 3.2.4.2-1 summarizes the results for this event.

3.2.4.3 IORV Sensitivity Studies

3.2.4.3.1 Variation of Boron Capacity/Mixing
3.2.4.3.2 Variation of RHR Capacity
3.2.4.3.3 Variation of RHR Delay
3.2.4.3.4 Variation of Pool and Service Water Temperature
3.2.4.3.5 Variation of Pool Size
3.2.4.3.6 Variation of S/RV Capacity
3.2.4.3.7 Variation of Boron Delay
3.2.4.3.8 Variation of RHR Capacity and Service Water Temperature

Table 3.2.4.3-1 summarizes the results of this event,

3.2.4.1 MSIV Sensitivity Studies

3.2.4.1.1 Variation of Boron Delay

The SLCS timer delay was varied between 30 seconds (75% below the nominal timer setting of 120 seconds) and 240 seconds (100% above the nominal time) resulting in peak pool temperatures, 7°F less than 6°F greater, respectively, compared to the base case. Containment pressures decreased and increased accord accordingly by 1 psi. Figures 3.2.4.1.1 graphically shows this parametric variation.* Minimum vessel level was increased by 0.9 feet at 30 seconds and decreased by 0.3 feet at 240 seconds.

3.2.4.1.2 Variation of Boron Capacity/Mixing Efficiency

The effective rate of boron injection into the core is the product of the boron pumping capacity and mixing efficiency. This effective rate was varied by ±20% resulting in peak pool temperatures 11°F below and 7°F above the base case, respectively. The base case represents an 86 gpm SLCS pumping rate in a

*While the overall objective of these sensitivity studies is to provide guidance for assessing the adequacy of plants having certain parameters different from the generic analyses, caution must be exercised when combining the results of several parameter variations, due to the non-linearities involved (see Section 3.3.4.4).

251 inch vessel with 75% assumed mixing efficiency. The -20% variation point equivalently represents 69 gpm at 75% efficiency or 86 gpm at 60% efficiency. Differences for plant size are covered by comparing the boron rate and the vessel inventory of the plant (e.g., the 86 gpm on a 251 size plant is equivalent to 66 gpm on a 218 size plant). Figure 3.2.4.1-1 graphically shows this variation.

3.2.4.1.3 Variation of HPCS/RCIC Capacity

The rated flow of the HPCS/RCIC system was varied by ±20%. Figure 3.2.4.1-1 graphically shows the variation. For the case ofincreased flow, pool temperature and containment pressure increased by 6°F and 1 psi, respectively. The increase in temperature is due to the higher power level maintained by the increased core flow. Minimum water level increased more than 2 feet. Decreasing HPCS/RCIC flow lowered peak temperature and pressure by 10°F and 2 psi, respectively. Minimum level was slightly reduced.

3.2.4.1.4 Variation of RHR Capacity

To determine the effect of varying RHR heat exchanger capabilities, the base capacity of 2.13% NBR at 100°F ΔT (670 BTU/sec-°F for the 251 size plant used as the base case) was altered by ±50%. Increasing the capacity by 50% yielded a 6°F temperature reduction and lowered the peak containment pressure by more than 1 psi. For the opposite case of 50% decrease in RHR capacity the results were a 13°F increase in temperature and a 2.5 psi pressure rise. Sensitivity of pool temperature is shown graphically in Figure 3.2.4.1.4-1.

3.2.4.1.5 Variation of RHR Delay

The effect of varying RHR start time was found to be small for the MSIV case. Increasing the start time from 11 to 16 minutes increased peak pool temperature by only 1°F. A decrease of 2 minutes resulted in less than 1°F reduction in pool temperature. Sensitivity of the pool temperature is shown graphically in Figure 3.2.4.1.4-1.

3.2.4.1.6 Variation in Pool and Service Water Temperature

The pool and service water temperatures were assumed to vary together (with the pool assumed to be 5°F above the service water). This variation was found to significantly affect peak pool temperature and containment pressure. Increasing these temperatures by 20°F (to the operating technical specification) produced a rise in pool temperature of 18°F and an increase of about 4 psi in peak pressure. Reducing the temperatures by 20°F yielded decreases of 18°F and 3 psi, respectively. Figure 3.2.4.1.4-1 graphically shows the pool tempperature variation plotted directly vs pool and service water initial temperatures.

3.2.4.1.7 Variation of RHR Capacity and Pool and Service Water Temperature

Varying both parameters simultaneously was done to examine different RHR designs due to different plant site water temperatures. It showed that pool and service water temperature was the dominant variable. Increases of ±50% in RHR capacity and +20°F in pool and service water temperature (and similar decreases) produced temperature changes of +12.5 and -4.9°F. Peak pressures varied accordingly by +2 and -1 psi.

3.2.4.1.8 Variations of Pool Size

The suppression pool mass was varied by 20% to simulate different size plants. The larger pool mass provides a bigger heat sink, thus reducing the peak pool temperature by nearly 12°F and peak pressure by 2 psi. For the lower pool mass, pool temperature increases 16°F and peak pressure 3 psi. Figure 3.2.4.1.4-1 graphically shows the result.

3.2.4.1.9 Variation of S/RV Capacity

The base case MSIV closure transient was run with S/RV variations of ±20% of nominal capacity. High pressure pump trip occurred at nominal setpoint. There was essentially no effect on the peak neutron flux or average surface heat flux. The effect on peak vessel pressure is shown in Figure 3.2.4.1.9-1.

3.2.4.1.10 Variation of RPT Delay

The delay in actual pump trip after exceeding the high pressure setpoint was varied +0.5 sec and +1.0 sec of nominal for the MSIV closure base case. Again there was not effect of peak neutron flux or average surface heat flux. That variation of pressure with delay is shown in Figure 3.2.4.1.9-1.

3.2.4.1.11 Variation of RPT Inertia

RPT inertia was varied +50% and -20% of nominal for the MSIV closure base case. There was essentially no effect on neutron flux or average surface heat flux. The effect on pressure is shown in Figure 3.2.4.1.9-1.

3.2.4.1.12 Effect of Partial Rod Insertion

No credit is taken for partial insertion of control rods in this ATWS analysis. To determine the effect of possible partial control rod insertion, cases with $1 and $2 of control rod worth were analyzed. The resulting changes in pool temperature were -10°F and -19°F, respectively. Peak containment pressures were reduced by 4.1 and 6.4 psi.

3.2.4.1.13 Variation of Void Coefficient

The effect of void coefficient on peak transient parameters (neutron flux, average surface heat flux, vessel pressure and suppression pool temperature) was studied for the MSIV closure transient. Void coefficient was varied from -6 to -14 ¢/% rated voids (nominal = -11 ¢/%). In all cases the recirculation pumps were tripped on high vessel pressure. The change in total effective worth of injected boron with void fraction was accounted for. Figures 3.2.4.1.13-1 shows the flux and pressure peaks for the MSIV transient, as a function of void coefficient for several values of Doppler coefficient. The peaks are shown relative to those with nominal nuclear coefficients. Figure 3.2.4.1.13-2 shows peak suppression pool temperature as a function of void coefficient.

3.2.4.1.14 Variation of Doppler Coefficient

The effect of Doppler coefficient on transient peak neutron flux, average surface heatflux and vessel pressure during an MSIV closure studied for the range -0.20 to -0.32 ¢/°F (nominal = -0.285 ¢/°F). Figure 3.2.4.1.14-1 shows the peaks plotted as a function of Doppler coefficient. Peak pool suppression temperature is plotted against Doppler coefficient in Figure 3.2.4.1.13-2.

3.2.4.2 Turbine Trip Sensitivity Studies

The Turbine Trip event differs from MSIV with respect to containment effects. For MSIV, steam continues to be discharged to the pool over a long period of time whereas the Turbine Trip steam dump ceases within the first 30 seconds of the event. Therefore, those sensitivities whose effect would be felt only beyond the first 80 seconds need not be considered. These include boron timer delay, boron pumping capacity/mixing efficiency and RHR capacity and start time. Since boron timer delay does affect water level, its influence on that parameter was examined. Tables 3.2.4.2-1 and 2 present the results of this study.

3.2.4.2.1 Variation of Boron Delay

The SLCS timer delay was varied between 30 seconds and 240 seconds (nominal delay = 120 sec). As noted previously, there is no effect on pool temperature or containment pressure. Water level does vary by -0.7 feet and +0.7 feet from the base case level, respectively.

3.3.4.2.2 Variation of Boron Capacity/Mixing

Variation in boron capacity/mixing is expected to have no impact on the peak pool temperature and containment pressure since the early part of the transient would remain unchanged with S/RV's closing near 71 seconds, and no isolation takes place.

3.2.4.2.3 Variation of HPCS/RCIC Capacity

The HPCS/RCIC flow was varied by ±20% about its nominal value. Since HPCS/RCIC flow is initiated approximately 10 seconds prior to closure of the last relief valve, the effect on pool temperature and containment pressure is negligible. Minimum water level varies from 1.8 feet above nominal base case minimum for the increased HPCS/RCIC case to 2.2 feet below for the reduced flow case.

3.2.4.2.4 Variation of RHR Delay

This change does not have any effect upon the pool temperature or containment pressure since peak containment temperature and pressure values occur much before the RHR is turned on. Peak values occur at about 70 seconds.

3.2.4.2.5 Variation of Void Coefficient

The effect of void coefficient variation on the turbine trip with bypass transient was studied similar to the MSIV closure reported in Section 3.2.4.1.13. Figures 3.2.4.2.5-1 and 3.2.4.2.5-2 show the results. Those cases in which the peak pressure was more than 7 psi below nominal did not reach the high pressure ATWS trip setpoint at the analytical upper limit of 1150 psig. If this happened, reactor operation at high power and continued steam dump to the pool could continue until manual action. For the pool temperature study a case was run at -9 ¢/% void coefficient and nominal Doppler, which did trip. One option to provide protection for lower void coefficients is to lower the ATWS trip setpoint. A case was run with -8 ¢/% void coefficient and the setpoint at 1091 psig, the analytical upper limit of the lowest S/RV group setpoint. This reduced setpoint would provide automatic ATWS protection for all cases, even those that open only a small amount of S/R valves.

3.2.4.2.5 Variation of Doppler Coefficient

Figure 3.2.4.2.6-1 shows the effect of Doppler coefficient variation. The effect on suppression pool peak temperature is shown in Figure 3.2.4.2.5-2.

3.2.4.3 IORV Sensitivity Studies

Table 5.4.3-1 summarizes the results of the sensitivities on the BWR/5 IORV - ATWS event. The effect of the listed parameters was determined on minimum water level, peak suppression pool temperature and peak wetwell airspace pressure. Figure 3.2.4.3-1 presents the results for peak suppression pool temperature in graphic form.

3.2.4.3.1 Variation of Boron Capacity/Mixing

The effective rate of boron injection into the core is the product of the boron pumping capacity and mixing efficiency. This effective rate was varied by ±20% resulting in peak pool temperatures of 8°F below and 26°F above the base case value, respectively. Peak containment pressures varied accordingly. Minimum water level variation was negligible for both cases.

3.2.4.3.2 Variation of RHR Capacity

To determine the effect of varying RHR heat exchanger capabilities, the base capacity of 670 BTU/sec-°F was altered by ±50%. Increasing the capacity to 1005 BTU/sec yielded a 11°F temperature reduction and lowered the peak containment pressure by 2 psi. For the opposite case of decreasing capacity the results were a 16°F increase in temperature and a 4 psi pressure rise.

3.2.4.3.3 Variation of RHR Delay

The RHR system is assumed to be in operation at time zero of this event. This was varied by delaying RHR start by 5 minutes and 10 minutes from time zero. The effect on peak pool temperature was an increase of 1°F and 2°F, respectively. Peak containment pressure varied accordingly.

3.2.4.3.4 Variation of Pool and Service Water Temperature

The pool and service water temperature were found to significantly affect peak pool temperature and containment pressure. Increasing these temperatures by 20°F produced a rise in pool temperature of 6°F and an increase of 1.5 psi in peak pressure. Reducing the temperatures by 20°F yielded decreases of 6°F and 1/2 psi, respectively.

3.2.4.3.5 Variations of Pool Size

The suppression pool mass was varied by 20% to simulate different sized plants. The larger pool mass provides a bigger heat sink, thus reducing the peak pool temperature by nearly 9°F and peak pressure by nearly 2 psi. For the lower pool mass, pool temperature increases 13°F and peak pressure by 3 psi.

3.2.4.3.6 Variation of S/RV Capacity

The capacity of the S/RV's were varied from 6.1% NBR to 10.5% NBR (base case = 8.33% NBR) Peak pool temperature changed by -10°F and +10°F, respectively. This is shown in Figure 3.2.4.3-2. Corresponding peak containment pressures changed by +2.2 and -1.9 psi from the base case value. Variation in minimum vessel water level was found to be negligible.

3.2.4.3.7 Variation of Boron Delay

The effect of increasing the time at which liquid-boron reaches the core on peak bulk average pool temperature and containment pressure is shown in Table 3.2.4.3-1 and Figure 3.2.4.3-3. These cases were run for SR/V capacities of 6.1% (typical of 251 size plants) and 8.3% (typical of 218 size plants. The base peak bulk average pool temperature for the 8.3% SR/V case is 187°F. All cases assume start of the boron injection timer at the same time pool temperaturehas reached 110°F and manual scram has been attempted. The RHR (pool cooling) is also initiated by this time.

3.2.4.3.8 Variation of RHR Capacity and Service Water Temperature

Varying both parameters simultaneously showed that RHR capacity was the dominant variable. Increases of 50% in RHR capacity and 20°F in pool and service water temperature produced temperature changes of -3 and +12°F. Peak pressures varied accordingly by -0.5 and +2.9 psi.

3.2.5 Loss of Condenser Vacuum

3.2.5.1 Overview of Response Without Scram

This transient starts a turbine trip due to low condenser vacuum, therefore, the beginning is the same as Turbine Trip events (see Section 3.2.2). There is a rapid steam shutoff causing pressure and power increases which are limited by the action of the S/RV's and RPT. Note that direct pump trip was conservatively neglected. Since the MSIV's and turbine bypass valves also close when condenser vacuum has further dropped to the setpoints, S/RV cycling increases considerably compared to the original Turbine Trip case. Even so, the bulk pool temperature and pressure are well within the containment design requirements. Therefore, this event is similar to the Turbine Trip event as far as the peak power and vessel pressure characteristics are concerned and similar to the MSIV closure case with respect so suppression pool temperature and containment pressure.

3.2.5.2 Sequence of Events For Loss of Condenser Vacuum

The listing of significant events during this ATWS event is provided in Table 3.2.5-1. Results with and without ARI are presented.

This transient starts with the closure of all turbine stop valves (within about 0.1 second) when the unexpected decline in condenser vacuum reaches the turbine trip setpoint. If the unit has turbine-driven feedwater pumps, they also trip at the same low vacuum setpoint. For the ARI failure case, the feedwater is assumed to remain as if motor-driven pumps were avilable until the feedwater limit action shuts them down (the most limiting case).

Figure 3.2.5-1 shows the initial portions of the event for the more likely plant ATWS transient in which ARI provides a diverse logic path to quickly shutdown the reactor, and Figure 3.2.5-2 shows the initial portion for the case in which ARI also is assumed to fail.

In both cases, the initial power and pressure increase. Neutron flux reaches 433% NBR near 1 second; fuel average heat flux reaches 134% NBR at about 3 seconds. Some fuel may experience boiling transition, however, coolable geometry is maintained. Peak pressure occurs at the vessel bottom and is 1193 psig near 3 seconds. The normal reactor scrams occur from position switches on the valves, high neutron flux, and high vessel pressure but are ignored for this analysis. The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to the automatic action of RPT which is initiated when vessel dome pressure exceeds 1150 psig and the relieving action of the S/RV's which all open then start reclosing near 5 seconds. By about 30 seconds, the condenser vacuum is assumed to have fallen enough to initiate MSIV and bypass valve closure. This results in another pressure and power rise to 1143 psig and 122% BNR, respectively. Both of these peaks are lower than the earlier values. Peak heat flux rises momentarily, but remains less than 65% and fuel geometry is maintained.

The long term behavior of this transient is very much like the MISV closure event which is discussed in detail in Section 3.2.1. Figure 3.2.5-3 shows the long term behavior predicted for this event. The peak bulk pool temperature and pressure which occur near 27 minutes are 176°F and 8.3 psig, respectively. These values remain well within the containment design requirement.

Thus it can be seen that the loss of condenser vacuum event combined with a failure to scram is adequately mitigated for a representative BWR 5/Mark II.

3.2.6 Pressure Regulator Failure - Zero Steam Demand

The failure of the controlling pressure regulator to the lower limit passes control of the main turbine control valves to the backup regulator. The backup regulator is nominally set 3 to 5 psi higher than the controlling regulator. As the transfer is made, a disturbance is introduced to the system but none of the variables are disturbed sufficiently to reach any scram trip setpoint. This transient is expected to be milder than the Turbine Trip case and therefore is not analyzed here as an ATWS event.

3.2.7 Loss of a Feedwater Heater

3.2.7.1 Overview of Response Without the Scram

This is a mild transient comp red to other ATWS events. The neutron flux does not reach the scram setpoint. The pressure rise is insignificantly small. Therefore, automatic ATWS logic (e.g., recirculation pump trip) does not occur, nor are HPCS or RCIC initiated. This is a gradual subcooling transient. The entire transient has settled out when the feedwater temperature stabilizes. The reactor settles out to a new equilibrium power condition at full core flow with recirculation flow assumed to be under manual control. If automatic flow control was active, the power increase would be less. Manual operator action accomplishes reactor shutdown.

3.2.7.2 Sequence of Events For Loss of a Feedwater Heater

In this event, loss of a key group of feedwater heaters gives the reactor coolant feedwater flow (decreased 65°F) which produces an increase in core inlet subcooling leading to an increase in core power. Following the transport delay through the feedwater lines (neglected in this analysis) and the time constant delay for cool-down of the heater tubes, average fuel surface heat flux rise to a maximum value of 113% which is lower than the flux scram setpoint or simulated thermal power scram setpoint. No fuel reaches boiling transition, even if the plant was initially at thermal operating limits. The reactor is conservatively assumed to be on manual flow control, therefore, core inlet flow remains at 100%. Had the reactor been on automatic flow control, core inlet flow remains at 100%. Had the reactor been on automatic flow control, core inlet flow would have changed to decrease the severity of the transient. The peak dome pressure of 1025 psig occurs near 74 seconds. Figure 3.2.7-1 shows the short term response of this event. The water level stays at normal conditions throughout the transient.

When the power reaches 108% NBR near 30 seconds, a high power alarm is sounded at which point the operator is alerted to the problem. For this analysis, it is assumed that attempts are expected to be made to bring the power down by inserting rods. If this is not successful, it is assumed that manual scram will be initiated, which will also initiate ARI and SLCS timed logic. However, in this analysis manual scram is also assumed to fail. By about 11 minutes the ARI function of inserting the rods will have been accomplished and generated power is terminated.

If the ARI function is arbitrarily assumed to fail as well as all other attempts to insert control rods within the two minute timed period after pressing the manual scram button, the automatic start of boron injection will begin through the HPCS line. An extra 30 seconds is allowed in the analysis for transport in the section of this line into the vessel without HPCS flow on. By about 35 minutes the power has been decreased below 1% NBR and nuclear hot shutdown has been achieved. Recirculation flow remains on during such a boron injection, providing total mixing and dispersion throughout the primary system.

Thus it can be seen that a loss of feedwater heater combined with a failure to scram is adequately integrated for a representative BWR 5/Mark II. Note that this event was analyzed for a 65°F loss in feedwater heating rather than 60°F as specified in NUREG 0460 (Volume 3).

3.2.8 Feedwater Controller Failure - Maximum Demand

3.2.8.1 Overview of Response Without Scram

The behavior of the plant is separable into a short and larger term behavior. The initial short term transient results in a gradual power increase, than a sharp pressure rise and power peak. The long term segment requires evaluation of coolant and containment conditions as the reactor is shut down.

Relief valve action occurs only during the early portion of the transient. RPT, acting in conjunction with the relief valves, serves to effectively limit the pressure disturbance. Note that the direct RPT from turbine trip was conservatively ignored. RPT also ensures relatively low power generation during the long term portion of the event. The effectiveness of RPT as presented in earlier reports is again confirmed by this analysis.

Containment peak temperature and pressure remain well below design limits due to the short duration of relief flow to the suppression pool. Power shutdown can be achieved in either of two ways. ARI employs an alternate design of the protection logic leading to a diverse insertion of the control rods. In the unlikely event that ARI also fails, the automated SLCS provides further protection and shutdown capability.

3.2.8.2 Sequence of Events for Feedwater Controller Failure - Maximum Demand

The time sequence of events for this transient is presented in Table 3.2.8-1. Both successful ARI initiation, and ARI failure cases are considered. The initiating event is the failure of the feedwater controller to the maximum demand position (125% NBR was assumed). The feedwater flow rapidly responds, causing vessel level to rise. When the high level trip setpoint (L8) is reached near 16 seconds, the turbine and feedwater are tripped. This results in a scram signal which, for purposes of this analysis, fails to initiate a scram. With the occurrence of the turbine trip, this event becomes very similar to the Turbine Trip transient. Figures 3.2.8-1 and 3.2.8-2 show the early portion of the event for the cases of ARI failure, and successful ARI actuation, respectively. For each case, the peak power and flux are the same with a maximum flux of 450% NBR near 17 seconds and a peak vessel bottom pressure of 1195 psig around 18 seconds. Fuel average heat flux reaches a maximum at 18 seconds of 140% NBR. Some fuel may experience boiling transition (about 6.3% if the core were operating at its thermal limit). The peak cladding temperature remains below 1520°F, and coolable geometry is maintained. Despite the assumed failure to scram based upon high neutron flux, vessel level and dome pressure generated scram signals, the transient pressure is maintained well below the 1500 psig Service Level C overpressure limit. This is accomplished through the combination of RPT (initiated on high dome pressure) and actuation of relief valves. Relief valve flow begins at 17 seconds, ceases at 70 seconds. S/RV's will open and cycle 3 times before permanently closing at 1750 seconds. This is shown along with vessel steam flow. The difference in vessel and relief steam flow is made up by the steamflow through the turbine bypass valves to the condenser.

At approximately 40 seconds, ARI will complete insertion of control rods into the core thereby shutting down the reactor. This will deactivate the SLCS turning the remainder of the event into normal feedwater flow controller failure transient. No further relief valve flow will occur. The decay heat will be passed through the turbine bypass valves to the condenser.

In the unlikely event of ARI failure, the event can still be mitigated through action of the SLCS. With confirmation from the flux monitoring system and rod position indicating that scram has not occurred, the SLCS will be activated. The long term behavior predicted for this event is shown in Figure 3.2.8-3. Boron first enters the core at about 200 seconds via the HPCS system and commences to shut down the system, with hot shutdown occurring near 23 minutes. Vessel level experiences slow cycles about the normal water level caused by the intermittent action of the RCIC and HPCS systems assumed to be automatically cycling between L2 and L8. Boron concentration will continue to increase until the entire inventory has been injected into the core around 50 minutes. At this point the concentration is sufficient to maintain cold nuclear shutdown conditions when the RHR system is switched to the reactor shutdown cooling mode and the plant is brought to a cold shut-down condition.

The peak suppression pool temperature is 108°F at 29 minutes with a corresponding peak pressure of 1.0 psig. The RHR can be activated in the pool cooling mode whenever convenient to reduce the pool temperature and any final, single valve cycles can be accommodated. Vessel level, which drops due to feedwater shutoff at high water level, is recovered and maintained in the normal water range by means of the HPCS/RCIC systems.

Thus it can be seen that a feedwater controller failure event (maximum demand) combined with a failure to scram is adequately mitigated for a representative BWR 5/Mark II.

3.2.9 Pressure Regulator Failure - Maximum Steam Demand

3.2.9.1 Overview of Response Without Scram

The initial portion of this transient consists of a decrease in reactor pressure and power as the turbine control valves open to the maximum position followed by a rapid rise in pressure and power due to MSIV closure on low steamline pressure. Scram is normally initiated at this time from the MSIV position switches. Should they fail, additional scram signals occur from high flux, high pressure, and low water level. Once the MSIV's close, the characteristics of the remaining portion of the transient are very much the same as the MSIV event.

The power and pressure increases are limited by the action of the S/RV's and RPT. With normal scram assumed to be failed, the long term power shutdown is achieved in either of two ways. ARI employs an alternate design of the protection logic leading to diverse insertion of the control rods. In the unlikely event that ARI fails the automated SLCS provides further protection and shutdown capability.

3.2.9.2 Sequence of Events For Pressure Regulator Failure - Maximum Demand

The listing of significant events during this event is provided below. Results for both cases - with ARI, and also assuming its failure, are presented.

This event begins with the inadvertent failure of the pressure regulator at the maximum demand value. This causes a quick increase in vessel steam flow which results in a rapid decrease in vessel pressure leading to a low pressure isolation set point at about 22 seconds. The MSIV's are tripped closed. Once this occurs the transient is essentially much like an MSIV closure event. The isolation is followed by a rapid rise in power and pressure. Figures 3.2.9-1 and 3.2.9-2 show the initial portions of the event for the more likely plant ATWS transient in which ARI quickly shuts down the reactor, and the case in which ARI also fails and the automated SLCS is called upon to shut down the reactor. In both cases, the peak power and pressure are the same. The neutron flux reaching 399% near NBR near 30 seconds, fuel average heat flux reaches 151% NBR at about 31 seconds. Some fuel may experience boiling transition. However, peak cladding temperature is less than 1630°F and coolable geometry is maintained. The peak pressure occurs at vessel bottom and is limited to 1238 psig near 32 seconds. The normal reactor scram signals occur from position switches on MSIV's, high neutron flux, and the high vessel pressure but are ignored for this analysis. The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to the automatic action of RPT which is initiated when vessel pressure exceeds 1150 psig near 30 seconds and the relieving action of the S/RV's which all open, then start reclosing near 43 seconds.

By about 50 seconds, the high pressure logic which began the ATWS protection will have accomplished the ARI function. This deactivates the automatic boron injection and feedwater limit, and turns the remainder of the event into a normal pressure regulator failure shutdown. The relief valve flow stops near 67 seconds. Peak suppression pool temperature will occur at the time of the last relief action and will be 95°F. The RHR can be activated in pooling cooling whenever convenient to control temperature. Reactor water level is restored to its normal range by feedwater flow and RCIC and HPCS flow.

If the ARI function is arbitrarily assumed to fail as well as all other attempts to insert control rods within the two-minute timed period, the ATWS logic will continue to sense that the APRM signals are not downscale and not enough rods are in their full-in positions, and the automatic start of boron injection will begin. The long term behavior predicted for this event is shown in Figure 3.2.9-3. Introduction of boron to the core at 3 minutes restores level and core flow before dropping the power near 22 minutes when nuclear shutdown is achieved. Thereafter, only decay heat reaches the pool, giving the peak pool temperature of 175°F (8.2 psig) at about 28 minutes. These values remain well within the containment design requirements. Water level inside the core shroud is a two-phase mixture which remains well above the core and up into the steam separator standpipes as RCIC and HPCS flow provide coolant inventory. The boron will continue to build the poison concentration in the vessel until it is all injected near 50 minutes making it possible for a controlled reactor cooldown. The total concentration is specified to be enough to maintain cold nuclear shutdown conditions even when the RHR system is eventually switched to the reactor shutdown cooling mode, bringing the plant to cold shutdown by normal procedures.

Thus it can be seen that the pressure regulator failure (maximum demand) combined with a failure to scram is adequately mitigated for a representative BWR 5/Mark II.

3.2.10 Loss of Feedwater

3.2.10.1 Overview of Response Without Scram

This event does not have rapid excursions as in some of the other events but is a long term power reduction and depressurization. Since the pressure begins to fall at the onset of the transient, the need for relief valves does not arise until isolation occurs very late in the event and only single valve cycling is expected to handle decay heat. The containment limits are not approached. Except for the use of the liquid boron solution for shutdown, the procedure followed here is virtually identical to the normal shutdown event.

3.2.10.2 Sequence of Events For Loss of Feedwater

In this event all feedwater flow is assumed to be lost in about 5 seconds. The resulting sequence of events is shown in Table 3.2.10-1 for both cases with and without ARI. Figure 3.2.10-1 shows the initial portion of the event for the more likely plant ATWS transient in which ARI quickly shuts down the reactor. Figure 3.2.10-2 shows the case in which ARI also fails and the automated SLCS is called upon to shut down the reactor.

In both cases, after the loss of feedwater (0-2 seconds) has taken place the pressure, water level and neutron flux begin to fall. Around 18 seconds low water level (L2) is reached. This trips the recirculation pumps, initiates ARI, initiates the HPCS and RCIC and activates the SLCS timed logic. Neglected was the recirculation runback which would have occurred earlier from coincident low level alarm (L4) and low feedwater flow. By 38 seconds the low water level logic which began the ATWS protection will have accomplished the ARI function. This deactivates the automatic boron injection. At 32 seconds HPCS and RCIC flows start. They replace the main feedwater system and begin to overcome the inventory loss. The vessel level decreases slightly faster immediately following ARI and the minimum for the simulated case is reached near 57 seconds as shown in Figure 3.2.10-1. The two-phase mixture level always remains above the top of the fuel. Vessel pressure continues to decrease as shown in Figure 3.2.10-1, as quenching by the RCIC and HPCS continues. The HPCS and RCIC will restore level to its normal range, for either automatic cycling between Level 2 and 8 setpoints or the operator takes over manual level control by using the RCIC (preferred). Pressure is expected to increase to the lowest S/RV setpoint when HPCS/RCIC are off (level restored), and one cycling valve is expected without significant pool temperature increase.

If the ARI function is arbitrarily assumed to fail as well as all other scrams and attempts to insert enough control rods within the two-minute timed period, the ATWS logic will continue to sense that the APRM signals are not downscale and not enough rods are in their full-in positions, and the automatic start of boron injection will begin. The power is predicted to remain in the 10-20% range, with core flow and level being restored during the first part of the boron injection as shown in Figure 3.2.10-2 and extended through the long term transient in Figure 3.2.10-3. The significant features during the early part of the event are the same as the previous case. The key difference is that the minimum water level is reached around 4 minutes and stays above the Level 1 setpoint. MSIV isolation may be avoided by taking the mode switch out of the "RUN" mode. This water level behavior is attributed to higher void fraction in the core as a result of higher power relative to the previous case in which ARI reduces power and core void fraction. SLCS boron injection is started near 2 minutes and it reaches the core 1 minute later. During the following 16 minute period (out to about 1100 seconds in Figure 3.2.10-3), the key result is that power is suppressed slightly, reducing the steaming rate and allowing water level to be restored. This also induces higher natural circulation core flow which follows the water level behavior. The level reaches the high level turnoff (Level 8) of the HPCS and RCIC at about 1200 seconds. The turbine is also tripped at this level but since the turbine steam bypass system opens immediately, no significant pressure disturbance is experienced.

By 1110 seconds the generated power is below 1% NBR and continues to decrease due to the accumulation of boron in the reactor. The net reactivity also stays negative. This accomplishes hot, nuclear shutdown. The vessel pressure is steadily decreasing and around 20 minutes MSIV isolation occurs due to low vessel pressure. By this time the generated power is practically zero and the only heat in the vessel is the decay heat.

The reactor pressure is expected to return to the setpoint of the lowest S/RV when HPCS and RCIC are off and are not quenching steam. The decay heat will cycle this lowest valve, but no significant suppression pool heatup is expected. The reactor would be cooled down at normal rates using the relief valve(s) to cold shutdown.

Thus it can be seen that loss of feedwater combined with a failure to scram is adequately mitigated for a representative BWR 5/Mark II.

3.2.11 Loss of Normal AC Power

3.2.11.1 Overview of Response Without Scram

The initial portion of the transient sees a sharp rise in reactor pressure and power due to MSIV closure as a result of loss of normal AC power. Scram is initiated at this time from the MSIV position switches if it had not occurred yet from loss of reactor trip system power. Should these signals fail, additional scram signals occur from high flux, high pressure and low water level. The power and pressure increases are limited by the action of the S/RV's and RPT (which occurs at the start of this event). With normal scram assumed to have failed the long term power shutdown is achieved in either of two ways. ARI employs an alternate design of the protection logic leading to diverse insertion of the control rods. In the unlikely event that ARI fails, the automated SLCS provides further protection and shutdown capability.

3.2.11.2 Sequence of Events For Loss of Normal AC Power

The listing of significant events during this event is provided below. Results for both cases - with ARI and also assuming its failure are presented.

There are two ways of experiencing this event. These are loss of all auxiliary power transformers and loss of all grid connections. The main difference between the two approaches is that in the latter one load rejection occurs at the outset of the transient which results in turbine-generator trip. In both cases MSIV closure takes place near 2 seconds. This is the earliest time isolation can occur due to coastdown of the RTS M/G set power supply.

Since in loss of all grid connections the turbine trips first as opposed to MSIV closure in loss of all auxiliary power transformers case, it turns out to be a less severe event in terms of peak power and pressure. Therefore the rest of the discussion is limited to the case where loss of all auxiliary power transformers occur. The sequence of events as outlined in Table 3.2.11-1 describes the event. Since loss of power takes place it is assumed that the accumulators will have enough air to last one cycle of S/RV valves at their relief setpoints after which they will switch over to spring setpoints. The low-low set S/RV design (if available) actually has greater capability for cycling in the relief mode, and would give lower pressures.

This event begins with the loss of recirculation pumps and feedwater pumps since condensate and/or booster pumps are also tripped due to loss of power. This leads to in initial fall in power and pressure. Near 2 seconds MSIV closure is assumed to take place, which results in a rapid rise in power and pressure. Figure 3.2.11-1 shows initial portions of the event for the more likely plant ATWS transient in which ARI quickly shuts down, the reactor, and Figure 3.2.11-2 shows the initial portion of the case in which ARI also fails and automated SLCS is called upon to shut down the reactor.

In both cases, the peak power and pressure are the same. The neutron flux reaches 468% NBR near 7 seconds, fuel average heat flux reaches 109% NBR at about 8 seconds. No fuel experiences boiling transition. The peak pressure occurs at vessel bottom and is 1231 psig near 9 seconds.

The normal scrams occur due to loss of AC power and also due to position switches on MSIV's, high neutron flux and the high vessel pressure but are ignored for this analysis. The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to RPT at the start of the transient and the relieving action of the S/RV's which all open, then start reclosing near 14 seconds.

By about 27 seconds, the high pressure logic would provide ATWS protection by activating ARI. This deactivates the automatic boron injection and allows the remainder of the event to proceed toward shutdown. The primary relief valve flow stops near 50 seconds, followed by only single valve cycling on the "tail" of the isolation event. The RHR can be activated in pool cooling mode as soon as water level recovery is clearly indicated to control pool temperature. Reactor water level is restored quickly to its normal range by RCIC and HPCS flow.

If the ARI function is arbitrarily assumed to fail as well as all other attempts to insert control rods within the two-minute timed period, the ATWS logic will continue to sense that not enough rods are in their full-in positions, and the automatic boron injection will begin. The long term behavior predicted for this event is shown in Figure 3.2.11-3. Introduction of boron to the core around 3 minutes again restores level and core flow before decreasing power near 20 minutes when nuclear shutdown is achieved. Thereafter, only decay heat reaches the pool, giving the peak bulk pool temperature of 170°F (7.3 psig) at about 33 minutes. These values remain well within the containment design requirements.

Water level inside the core shroud is a two-phase mixture which remains well above the core and up into the steam separator standpipes as RCIC and HPCS flow provide coolant inventory. The boron will continue to build the poison concentration in the vessel until it is all injected near 50 minutes making it possible for a controlled reactor cooldown. The total concentration is specified to be enough to maintain cold nuclear shutdown conditions even when the RHR system is eventually switched to the reactor shutdown cooling mode, bringing the plant to a cold shutdown.

Thus it can be seen that a loss of normal AC power combined with a failure to scram is adequately mitigated for a representative BWR 5/Mark II.

3.2.12 Recirculation Flow Controller Failure - Maximum Demand

3.2.12.1 Overview of Response Without Scram

This transient is not severe enough to trip any ATWS logic nor initiate HPCS or RCIC flow. It is considerably milder than MSIV closure or turbine trip ATWS cases. This is a short term transient with a sudden power rise and a relatively small pressure increase. The entire transient is over within 30 seconds by which time the reactor settles out to a new equilibrium condition less than 100% rated power. Since the peak pressure stays below the lowest S/RV setpoint, steam flow to the suppression pool does not take place.

3.2.12.2 Sequence of Events for Recirculation Flow Controller Failure - Maximum Demand

This event may occur for one loop only, or both loops simultaneously. One loop failure occurs when the individual loop controller fails. The valve stroking rate is limited by the capability of the valve hydraulics. Master controller failure would affect both loops. For this case, the maximum valve stroking rate is governed by the valve actuator velocity limiter so that this event is less severe than failure in one loop. The following sequence of events describes the failure of one loop controller.

The worst case initial conditions for this event are the conditions corresponding to the minimum recirculation valve position and pump speed on the 105% NBR steamflow rod pattern flow control line. The power and core flow at this point are 53% and 33% of rated respectively.

The event is initiated by the rapid failure open of one recirculation flow control valve, the valve reaching the full open position in approximately 3.5 seconds. Figure 3.2.12-1 shows the behavior of key parameters during this event. The resulting increase in core flow causes a neutron flux power spike which reaches a maximum value of 382% NBR at 1.9 seconds. Thermal power, as demonstrated by the heat flux at the surface of the fuel, peaks at 92% NBR near 2.8 seconds. Either of these variables (high neutron flux or high simulated thermal power) would have initiated scram (neglected in this analysis). No fuel encounters boiling transition even if the core is operating initially at its operating limit. A small pressure rise occurs, peaking near 5 seconds with a vessel bottom pressure of 1007 psig (compared to an initial bottom pressure of 961 psig). Simultaneous with the above events, vessel level experiences a small decrease and then recovers to its initial position, and feedwater flow rises in response to the power increase. As core flow levels off to approximately 66% of rated near 6 seconds, the power settles out as do all other parameters. At this point, the transient is essentially over. By 20 seconds, all parameters have reached equilibrium except power and feedwater flow which continue to slowly decrease following the warming of the feedwater heaters.

Because of the mildness of the event, no automatic pressure or level dependent actions are initiated. Containment is not affected since no relief valves are actuated. Subsequent operator action would be expected to initiate a manual shutdown, utilizing the SLCS if manual insertion or scram of rods remains unsuccessful. Initiation of ARI/SLCS from manual scram near 10 minutes would shut down the plant immediately (ARI) or by about 30 minutes (SLCS). Recirculation flow would be maintained near full flow initially and at partial flow (low frequency M/G sets on) in order to maximize boron dispersion throughout the vessel and to provide near-normal shutdown sequence.

Thus it can be seen that a recirculation flow control failure combined with a failure to scram is adequately mitigated for a representative BWR 5/Mark II.

3.2.13 Startup of the Idle Recirculation Pump

This event is similar to the recirculation flow controller failure - maximum demand, both of these vents result in increased core power which results from the increased core flow. The Startup of the Idle Recirculation Pump event has been shown in safety analysis reports to be less severe than the recirculation flow controller failure and therefore, further transient-specific analyses have not been done.

3.2.14 Inadvertent Opening of All Bypass Valves

This event will be similar to the pressure regulator failure - maximum steam demand. Since the turbine control valves will try to compensate for the pressure reduction, the results will be less severe. For those plants with smaller bypass capacity, this event will be even less severe.

3.2.15 Shutdown Cooling (RHR) Malfunction - Decreasing Temperature

This event can only occur at very low pressure. The shutoff head of the shutdown cooling pumps is less than 300 psig. In this condition, the reactor has almost no voids in it and therefore only little if any positive reactivity is inserted. Therefore, this event is not considered further.

Table 3.2-1

SUMMARY OF BWR/5 RESULTS

Event	Maximum Neutron Flux (% NBR)	Maximum Average Surface Heat Flux (% BNR)	Maximum Vessel Bottom Pressure (psig)	Maximum Steamline Pressure (psig)	Suppression Pool Maximum Bulk Temp. (°F)	Containment Peak Pressure (psig)
MSIV Closure	614	150	1247	1193	179	9.1
Turbine Trip With Bypass	426	134	1192	1126	104	0.7
IORV	---	---	---	---	187	10.6
Loss of Condenser Vacuum	433	134	1193	1126	176	8.3
Loss of Feedwater Heater	114	113	1071	987	90	no change
Feedwater Controller Failure - Max Demand	450	140	1196	1127	108	1.0
Pressure Regulator Failure - Open	399	151	1238	1180	175	8.2
Loss of Normal Feedwater Flow	100	100	1056	988	90	no change
Loss of Normal AC Power	468	109	1205	1191	170	7.3
Recirculation Flow Failure - Open	382	92	1007	969	90	no change
Turbine Trip Without Bypass	643	143	1230	1171	178	9.1

Table 3.2.1-1

BWR/5 MSIV CLOSURE

SEQUENCE OF EVENTS	Time
1. Nominal 4 Second MSIV Closure - Scram Fails	0
2. Pressure Rise Begins	0
3. Relief Valves Lift	4 seconds
4. Recirculation Pumps Trip on High Pressure, ARI is Initiated and Timed SLCS Logi is Triggered	5 seconds
5. Vessel Pressure Peaks	7 seconds
6. ARI Assumed to Fail	25 seconds
7. Water Level Reaches Level 2, Beginning HPCS and RCIC Startup	34 seconds
8. Feedwater Flow Coasts Down to Lower Limit	45 seconds
9. HPCS and RCIC Flow Reaches the Vessel	54 seconds
10. ATWS Logic Timer Complete, SLCS Starts	2 minutes
11. Liquid Control Flow Reaches Core	3 minutes
12. Water Level Reaches Minimum and Begins to Rise	5 minutes
13. RHR Flow Begins (Pool Cooling)	11 minutes
14. Hot Shutdown Achieved	22 minutes
15. Maximum Containment Bulk Temperature and Pressure Occur	46 minutes

Table 3.2.1-2

BWR/5 MSIV CLOSURE SUMMARY

With ARI Failure	86 GPM-2 Min Logic Delay MSIV
Maximum Neutron Flux (%)	614
Maximum Vessel Bottom Pressure (psig)	1246
Maximum Average Heat Flux (%)	150
Maximum Bulk Suppression Pool Temperature (°F)	179
Associated Containment Pressure (psig)	9.1

Table 3.2.2-1

BWR/5 TURBINE TRIP

SEQUENCE OF EVENTS	Time
1. Turbine Trips - Scram Fails	0
2. Pressure Rise Begins	0
3. Relief Valve Lift	2 seconds
4. Some Fuel Experiences Transition Boiling	2 seconds
5. Recirculation Pumps Trip on HIgh Pressure, ARI is Initiated and Timed SLCS Logis is Triggered	2 seconds
6. Vessel Pressure Peaks	3 seconds
7. ARI Assumed to Fail	22 seconds
8. Feedwater Flow Runs Back to Lower Limit Value	45 seconds
9. HPCS and RCIC Flow Starts on Level 2 Initiation	60 seconds
10. ATWS Logic Timer Complete, SLCS Starts	2 minutes
11. Containment Temperature and Pressure Peak	2 minutes
12. Liquid Control Flow Reaches Core	3 minutes
13. Water Level Reaches Minimum and Begins to Rise	5 minutes
14. RHR Flow Begins (Pool Cooling)	11 minutes
15. Hot Shutdown Achieved	23 minutes

Table 3.2.2-2

BWR/5 TURBINE TRIP SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Turbine Trip
Maximum Neutron Flux (%)	426
Maximum Vessel Bottom Pressure (psig)	1192
Maximum Average Heat Flux (%)	134
Maximum Bulk Suppression Pool Temperature (°F)	104
Associated Containment Pressure (psig)	0.7

Table 3.2.3-1

BWR/5 IORV

	Sequence of Events	Time
1.	Relief valve opens inadvertently and attempts to close it are unsuccessful	0
2.	Alarm sounds at 95°F and operator initiates pool cooling	2 minutes
3.	Suppression pool temperature reaches 110°F operator attempts manual scram; scram fails	7.5 minutes
4.	ARI assumed to fail	8 minutes
5.	SLCS automatically starts	9.5 minutes
6.	Control liquid reaches core	10 minutes
7.	Power is less than relief valve capacity	19 minutes
8.	Isolation on low steam line pressure	21 minutes
9.	Peak suppression pool temperature and pressure are reached	75 minutes

Table 3.2.3-2

BWR/5 IORV - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay IORV
Maximum Bulk Suppression Pool Temperature (°F)	187
Associated Containment Pressure (psig)	10.6

Table 3.2.4.1-1

BWR/5 MSIV SENSITIVITY STUDY RESULTS

Parameter Varied	Base Value	% Change	Change in Minimum Level (ft)	Change in Peak Pool Temperature (°F)	Change in Peak Containment Pressure (psi)
BASE CASE	-	-	-9.8	179.0	9.1
Boron Timer	120 sec	+100/-75	-0.25 /+0.87	+5.9 / -7.1	+1.0 /-1.0
HPCS/RCIC Capacity	13.6% NBR Fw Flow	+20/-20	+2.26 /-0.73	+5.6 /-10.4	+0.9 /-2.0
Boron Pumping Cap/ Mixing Efficiency	86 GPM*/ 75%	+20/-20	-0.20 /-0.19	-10.9 / +8.20	-2.10/+2.0
RHR Effectiveness	2.13% NBR at 100°F ΔT	+50/-50	N/A	-6.0 /+13.0	-1.3 /+2.5
RHR Start Time	660 sec	+45/-18	N/A	+1.1 / -0.6	0.0 / 0.0
Service Water Temperature	85°F	+20°F/-20°F	N/A	-18.0 /-18.0	+3.8 /-3.2
Pool Size	28.4 Full Flow	+20/-20	N/A	-11.5 /+15.8	-2.2 /+3.2
RHR Effectiveness and Service Water Temperature	(see above)	+50/-50 +20/-20	N/A	+12.5 / -4.9	+2.2 / -1.1

*Sized to a 251 vessel

Table 3.2.4.1-2

BWR/5 MSIV ATWS NUCLEAR PARAMETRIC STUDY SUMMARY

Doppler Coef (¢/°F)	Void Coef (¢/%)	Neutron Flux (%)	Average Heat Flux (%)	Vessel Bottom Pressure (psi)	Suppression Pool Temperature (°F)	Min Level/Time (with range) (ft/sec)
-0.200	-6	-159	-8.5	-16		
-0.230	-6	-194	-11.6	-23		
-0.285	-6	-242	-16.2	-35		
-0.320	-6	-264	-18.6	-40		
-0.200	-8	-18	+0.6	+5		
-0.230	-8	-70	-2.8	-5		
-0.285	-8	-140	-7.8	-17	+1.1	-1.4/232
-0.320	-8	-173	-10.5	-23		
-0.200	-11	+180	+8.0	+22		
-0.230	-11	+100	+4.9	+13	-3.4	+1.3/242
-0.280†	-11†	614†	150.8†	1252†	177.8†	-9.6/207†
-0.320	-11	-46	-2.7	-7	+1.7	-0.8/264
-0.200	-14	+278	+10.9	+30		
-0.230	-14	+192	+8.4	+22		
-0.285	-14	+77	+4.4	+10	-1.7	+1.2/233
-0.320	-14	+28	+2.8	+3		

†Values shown are nominal void and Doppler coefficients are absolute peaks. Other peaks are relative to these.

Table 3.2.4.1-3

BWR/5 MSIV-ATWS PARAMETRIC STUDY SUMMARY

Parameter	Variation	Neutron Flux	Average Heat Flux	Vessel Bottom Pressure
		Change in Peak Value		
	Nominal	%	%	psi
S/RV capacity	+20%	+0	+0.1	- 27
	-20%	+0	+0	+110
RPT Delay	+0.5 sec	+0	+0	+0
	+1.0 sec	+0	+0	+3
RPT Inertia	+50%	+2	+0	+1
	-20%	+0	+0	-1

Table 3.2.4.2-1

BWR/5 TURBINE TRIP SENSITIVITY RESULTS

Parameter Varied	Base Value	% Change	Change In Minimum Level (ft)
Base Case	--	--	-6.9
HPCS/RCIC	13.6% NBR Steamflow	+20	+1.8
		-20	-2.2
Boron Delay	120 sec	+100	-0.7
		-75	+0.7

Table 3.2.4.2-2

BWR/5 TURBINE TRIP ATWS NUCLEAR PARAMETRIC STUDY SUMMARY

Doppler Coef (¢/°F)	Void Coef (¢/%)	Change in Peak Value: Neutron Flux (%)	Average Heat Flux (%)	Vessel Bottom Pressure (psi)	Suppr Pool Temp (°F)	Min Level/Time (wide range) (ft/sec)
-0.200	- 6	-134	- 3.9	- 7*		
-0.230	- 6	-148	- 5.7	- 9*		
-0.285	- 6	-169	- 8.6	-11*		
-0.320	- 6	-179	- 10.2	-12*		
-0.200	- 8	- 51	+ 0.9	- 1		
-0.230	- 8	- 73	- 1.1	- 3		
-0.285	- 8	-104	- 4.5	-12**	+0.8	
-0.320	- 8	-120	- 5.7	- 7*		
-0.285	- 9	- 70	- 2.5	- 3	+1.5	-1.6/221
-0.200	-11	+ 92	+ 4.5	+ 5		
-0.230	-11	+ 54	+ 2.8	+ 3	-0.1	+0.7/235
-0.280***	-11***	427	134.4	1193	103.7	-6.9/269
-0.320	-11	- 27	- 1.6	- 2	+1.0	-1.0/252
-0.200	-14	-287	- 5.8	+ 8		
-0.230	-14	+220	- 4.3	+ 6		
-0.285	-14	+132	+ 2.0	+ 3	1.0	+0.5/266
-0.320	-14	- 89	+ 0.6	+ 2	-1.0	+0.5/266

*These cases did not reach high pressure trip setpoint (analytical upper limit = 1150 psig dome pressure) See discussion in Section 3.2.4.2.5.

**This case was run at lowered high pressure trip setpoint (1091 psig)

***Values shown for nominal void and Doppler coefficients are absolute peaks. Other peaks are relative to these.

Table 3.2.4.3-1

BWR/5 IORV SENSITIVITY STUDY RESULTS

Parameter Varied	Base Value	Change	Change in Vessel Minimum Water Level (ft)	Change in Peak Pool Temperature (°F)	Change in Peak Containment Pressure (psi)
Base Case			-2.4	187	10.6
Boron Capacity/ Mixing Efficiency	86 GPM/95%	+50%	0	- 8	-1.7
		-50%	0	+26	+6.8
RHR Capacity	670 BTU/°F-sec	+50%	N/A	-11	-21.
		-50%	N/A	+16	+4.1
RHR Start Time	0 Seconds	+5 Minutes	N/A	+ 1	+0.1
		-10 Minutes	N/A	+ 2	+0.3
Service Water Temperature	85°F	+20°F	N/A	+ 6	+1.5
		-20°F	N/A	- 6	-1.2
RHR Capacity and Service Water Temperature	(See above)	+50% +20°F	N/A	- 3	-0.5
		-50% -20°F	N/A	+12	+2.9
Pool Size	6.8 Million lbm	+20%	N/A	- 9	-1.9
		-20%	N/A	+13	+3.0
S/RV Capacity	8.33% NBR Steamflow	+26.1%	<0.5	+10	+2.2
		-27%	<0.5	-10	-1.9
Boron Timer and S/RV Capacity	2 minutes, 8.33%	+3 min-27%	~0	- 7	-1.6
		+8 min-27%	~0	- 3	-0.7

Table 3.2.5-1

BWR/5 LOSS OF CONDENSER VACUUM

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Main turbine (and feedwater turbines)* trip due to low condenser vacuum, bypass opens - All normal scrams fail.	0	0
2.	Pressure and power rise begins	0	0
3.	Peak power occurs	1 Second	1 Second
4.	Relief valves lift	2 Seconds	2 Seconds
5.	ATWS high pressure setpoint - Recirculations pumps tripped - ARI is initiated - SLCS timed logic activated	2 Seconds	2 Seconds
6.	Same fuel may experience boiling transition	2 Seconds	2 Seconds
7.	Peak vessel pressure occurs	3 Seconds	3 Seconds
8.	ARI control rod insertion completely eliminating SLCS and feedwater limiting actions	22 Seconds	Fails
9.	ATWS logic Initiates feedwater flow limit	N/A	27 Seconds
10.	MSIV's and bypass close due to low condenser vacuum	30 Seconds	30 Seconds
11.	Reactor water level drops to Level 2 - Initiates containment isolation - HPCS and RCIC start	36 Seconds	46 Seconds
12.	HPCS and RCIC flow begins	56 Seconds	66 Seconds
13.	ATWS logic times completed - Initiates SLCS	N/A	2 Minutes
14.	Liquid control flow reaches core	N/A	3 Minutes

*Sequence conservatively assumes motor driven feedwater pumps.

Table 3.2.5-1 (Continued)

Sequence of Events	Time With ARI	Time With ARI Failure
15. Reactor water level reaches minimum and begins to rise	50 Seconds	3-1/2 Minutes
16. RHR flow begins (pool cooling)	>11 Minutes	11 Minutes
17. Hot shutdown achieved	22 Seconds	20 Minutes
18. Containment temperature and pressure peaks occur		27 Minutes

Table 3.2.5-2

BWR/5 LOSS OF CONDENSER VACUUM - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Loss of Condenser Vacuum
Maximum Neutron Flux (%)	433
Maximum Vessel Bottom Pressure (psig)	1193
Maximum Average Heat Flux (%)	134
Maximum Bulk Suppression Pool Temperature (°F)	176
Associated Containment Pressure (psig)	8.3

Table 3.2.7-1

BWR/5 - LOSS OF A FEEDWATER HEATER

	Sequence of Events	Time
1.	Inadvertent tripping of feedwater heaters; feedwater enthalpy begins to drop	0 Second
2.	Reactor and turbine-generator power begins to rise	2 Seconds
3.	APRM high power alarm (105%), operator attempts to insert rods	29 Seconds
4.	Vessel pressure levels off after a small increase	80 Seconds
5.	Power levels off below the scram setpoint(s)	125 Seconds
6.	Feedwater enthalpy sees the assumed full change	131 Seconds
7.	Manual scram attempted after control rod insertion attempts have failed. ARI and timed SLCS logic initiated, scram fails.	$\leq$10-1/2 Minutes
8.	ARI control rod insertion complted, eliminating SLCS initiation, and achieving reactor shutdown	$\leq$11 Minutes
9.	Final ATWS logic timer completed - Initiates SLCS (if ARI has failed)	$\leq$12-1/2 Minutes
10.	Liquid control reaches core (if ARI has failed)	$\leq$14 Minutes
11.	Hot shutdown achieved (if ARI has failed)	$\leq$35 Minutes

Table 3.2.7-2

BWR/5 LOSS OF FEEDWATER HEATER - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Loss of Feedwater Heater
Maximum Neutron Flux (%)	114
Maximum Vessel Bottom Pressure (psig)	1071
Maximum Average Heat Flux (%)	113
Maximum Bulk Suppression Pool Temperature (°F)	90
Associated Containment Pressure (psig)	no change

Table 3.2.8-1

BWR/5 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND

	Sequence of Events	Time: With ARI	Time: With ARI Failure
1.	Feedwater controller fails to maximum demand. Reactor water level begins to rise, power gradually increases.	0	0
2.	High power level (Level 8) setpoint is reached - Turbine trips, bypass opens - All normal scrams fail - Feedwater pumps trip	16 Seconds	16 Seconds
3.	Pressure and power rise begins	16 Seconds	16 Seconds
4.	Relief valves lift	17 Seconds	17 Seconds
5.	ATWS high pressure setpoint is reached (1150 psig) - Recirculation pumps are tripped* - ARI is initiated - SLCS timed logic is activated	18 Seconds	18 Seconds
6.	Vessel pressure peaks	18 Seconds	18 Seconds
7.	Some fuel experiences transition boiling	18 Seconds	18 Seconds
8.	ARI control rod insertion completed, eliminating SLCS initiation	37 Seconds	Fails
9.	Reactor water level drops to Level 2 - Initiates containment isolation - Initiates HPCS and RCIC	40 Seconds	49 Seconds
10.	Lowest setpoint S/RV closes	45 Seconds	66 Seconds
11.	HPCS and RCIC flow begins	60 Seconds	69 Seconds
12.	Liquid control flow reaches core	N/A	198 Seconds
13.	ATWS Logic timer completed	N/A	278 Seconds
14.	Reactor water level reaches minimum and begins to rise	N/A	278 Seconds
15.	RHR flow begins (pool cooling)	11 Minutes	11 Minutes
16.	MSIV's close on low line pressure	N/A	20 Minutes
17.	Hot shutdown achieved	27 Seconds	23 Minutes

Table 3.2.8-2

BWR/5 FEEDWATER CONTROLLER FAILURE (MAXIMUM DEMAND) - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Feedwater Controller Failure
Maximum Neutron Flux (%)	450
Maximum Vessel Bottom Pressure (psig)	1196
Maximum Average Heat Flux (%)	140
Maximum Bulk Suppression Pool Temperature (°F)	108
Associated Containment Pressure (psig)	1.0

Table 3.2.9-1

BWR/5 PRESSURE REGULATOR FAILURE - MAXIMUM STEAM DEMAND

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Pressure regulator fails to maximum demand	0	0
2.	Pressure and power begin to fail	0	0
3.	Low Pressure isolation setpoint reached		
	- MSIV closure	22 Seconds	22 Seconds
	- Scram normally initiated (assumed to fail)	23 Seconds	23 Seconds
4.	Pressure and power begin to rise	24 Seconds	24 Seconds
5.	Relief valves lift	30 Seconds	30 Seconds
6.	ATWS high pressure setpoint is reached (1150°F psig) - Recirculation pumps are tripped - ARI is initiated - SLCS and feedwater limit logic is activated		
7.	Vessel pressure peaks	34 Seconds	34 Seconds
8.	Some fuel experiences boiling transition	34 Seconds	34 Seconds
9.	ARI control rod insertion completed (eliminating need for SLCS initiation and feedwater limit)	49 Seconds	fails
10.	Reactor water level drops to Level 2 - Initiates containment isolation - Initiates HPCS and RCIC	49 Seconds	52 Seconds
11.	ATWS logic initiates feedwater limit	N/A	55 Seconds
12.	Feedwater flow runs back to lower value	N/A	60 Seconds
13.	HPCS and RCIC flow begins	69 Seconds	72 Seconds
14.	ATWS logic timer completed - Initiates SLCS	N/A	2-1/2 Minutes
15.	Reactor water level reaches minimum and begins to rise	74	3-1/2 Minutes
16.	Liquid control flow reaches core	N/A	3-1/2 Minutes
17.	RHR flow begins (pool cooling)	$\geq$11 Minutes	11 Minutes
18.	Hot shutdown achieved	49 Seconds	19 Minutes
19.	Containment temperature and pressure peak	N/A	30 Minutes

Table 3.2.9-2

BWR/5 PRESSURE REGULATOR FAILURE (MAXIMUM STEAM DEMAND) - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Pressure Regulator Failure
Maximum Neutron Flux (%)	399
Maximum Vessel Bottom Pressure (psig)	1238
Maximum Average Heat Flux (%)	151
Maximum Bulk Suppression Pool Temperature (°F)	175
Associated Containment Pressure (psig)	8.2

Table 3.2.10-1

BWR/5 LOSS OF FEEDWATER

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Feedwater flow stops (flow assumed to reduced to zero in 5 seconds) - All normal scrams assumed to fail	0	0
2.	Pressure, water level and power start to decline	0	0
3.	Reactor water level drops to Level 2 and trips recirculation pumps*, initiates ARI and also initiates RCIC and HPCS. SLCS timed logic is also activated.	18 Seconds	18 Seconds
4.	ARI control rod insertion completed.	38 Seconds	Fails
5.	HPCS and RCIC flow starts	38 Seconds	38 Seconds
6.	ATWS logic timer completed - Initiates SLCS	N/A	138 Seconds
7.	Liquid control flow reaches the core	N/A	198 Seconds
8.	Water level reaches minimum and begins to rise. The top of the core always remains covered.	57 Seconds	4 Minutes
9.	High water level trip of HPCS and RCIC (neglecting preferred operator action to manually control level)	10 Minutes	20 Minutes
10.	Low pressure MSIV closure	N/A	20 Minutes
11.	Hot shutdown achieved	38 Seconds	23 Minutes

*Recirculation runback (from low level alarm, L4, and coincident low FW flow) is neglected.

Table 3.2.10-2

BWR/5 LOSS OF FEEDWATER - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Loss of Feedwater
Maximum Neutron Flux (%)	100
Maximum Vessel Bottom Pressure (psig)	1056
Maximum Average Heat Flux	100
Maximum Bulk Suppression Pool Temperature (°F)	90
Associated Containment Pressure (psig)	no change

Table 3.2.11-1

LOSS OF NORMAL AC POWER TRANSIENT

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Loss of all auxiliary power transformers - Recirculation pumps trip - Condensate and feedwater pumps trip	0	0
2.	Pressure and power begin to fall	0	0
3.	Normal scram due to loss of AC (Assumed to fail)	2 Seconds	2 Seconds
4.	MSIV's start to close due to loss of AC power (and initiate scram - also assumed to fail)	2 Seconds	2 Seconds
5.	Pressure and power begin to rise	6 Seconds	6 Seconds
6.	S/RV valves lift at relief setpoints	7 Seconds	7 Seconds
7.	ATWS high pressure setpoint is reached (1150 psig) - ARI is initiated - SLCS timed logic is activated	7 Seconds	7 Seconds
8.	Vessel pressure and power peak	9 Seconds	9 Seconds
9.	Some fuel may experience boiling transition	9 Seconds	9 Seconds
10.	Reactor water level drops to Level 2 - Initiates containment isolation - Initiates HPCS and RCIC	26 Seconds	26 Seconds
11.	ARI control rod insertion completed, eliminating SLCS initiation	27 Seconds	Fails
12.	HPCS and RCIC flow begins	46 Seconds	46 Seconds
13.	Lowest relief setpoint S/RV closes and the S/RV's are assumed to switch to spring setpoints		106 Seconds
14.	ATWS logic timer completed - Initiates SLCS	N/A	2 Minutes
15.	Reactor water level reaches minimum and begins to rise. Level inside the core shroud remains above the top of active fuel.	55 Seconds	3 Minutes
16.	Liquid control flow reaches core	N/A	3 Minutes
17.	RHR flow begins (pool cooling)	11 Minutes	11 Minutes
18.	Hot shutdown achieved	27 Seconds	20 Minutes
19.	Containment temperature and pressure peak		33 Minutes

Table 3.2.11-2

BWR/5 LOSS OF NORMAL AC POWER - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Loss of Normal AC Power
Maximum Neutron Flux (%)	468
Maximum Vessel Bottom Pressure (psig)	1205
Maximum Average Heat Flux (%)	109
Bulk Suppression Pool Temperature (°F)	170
Associated Containment Pressure (psig)	7.3

Table 3.2.12-1

BWR/5 RECIRCULATION FLOW CONTROLLER FAILURE - INCREASING FLOW

	Sequence of Events	Time
1.	Valve controller fails	0
2.	Neutron flux reaches 120%, APRM scram assumed to fail	1 Second
3.	Power peaks	2 Seconds
4.	Maximum fuel surface heat flux occurs	3 Seconds
5.	Flow control valve reaches full open position	4 Seconds
6.	Vessel pressure peaks	5 Seconds
7.	Core flow increase levels off	6 Seconds
8.	New core equilibrium conditions (All parameters within normal limits, power and feedflow slowly decreasing as steady state feedwater heating is established.)	20 Seconds
9.	Manual scram or (if this fails) automatic ARI and SLCS initiation	10 Minutes
10.	Hot shutdown achieved	30 Minutes

Table 3.2.12-2

BWR/5 RECIRCULATION FLOW CONTROLLER FAILURE (INCREASING FLOW) - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Recirculation Flow Controller Failure
Maximum Neutron Flux (%)	382
Maximum Vessel Bottom Pressure (psig)	1007
Maximum Average Heat Flux (%)	95
Maximum Bulk Suppression Pool Temperature (°F)	90
Associated Containment Pressure (psig)	no change

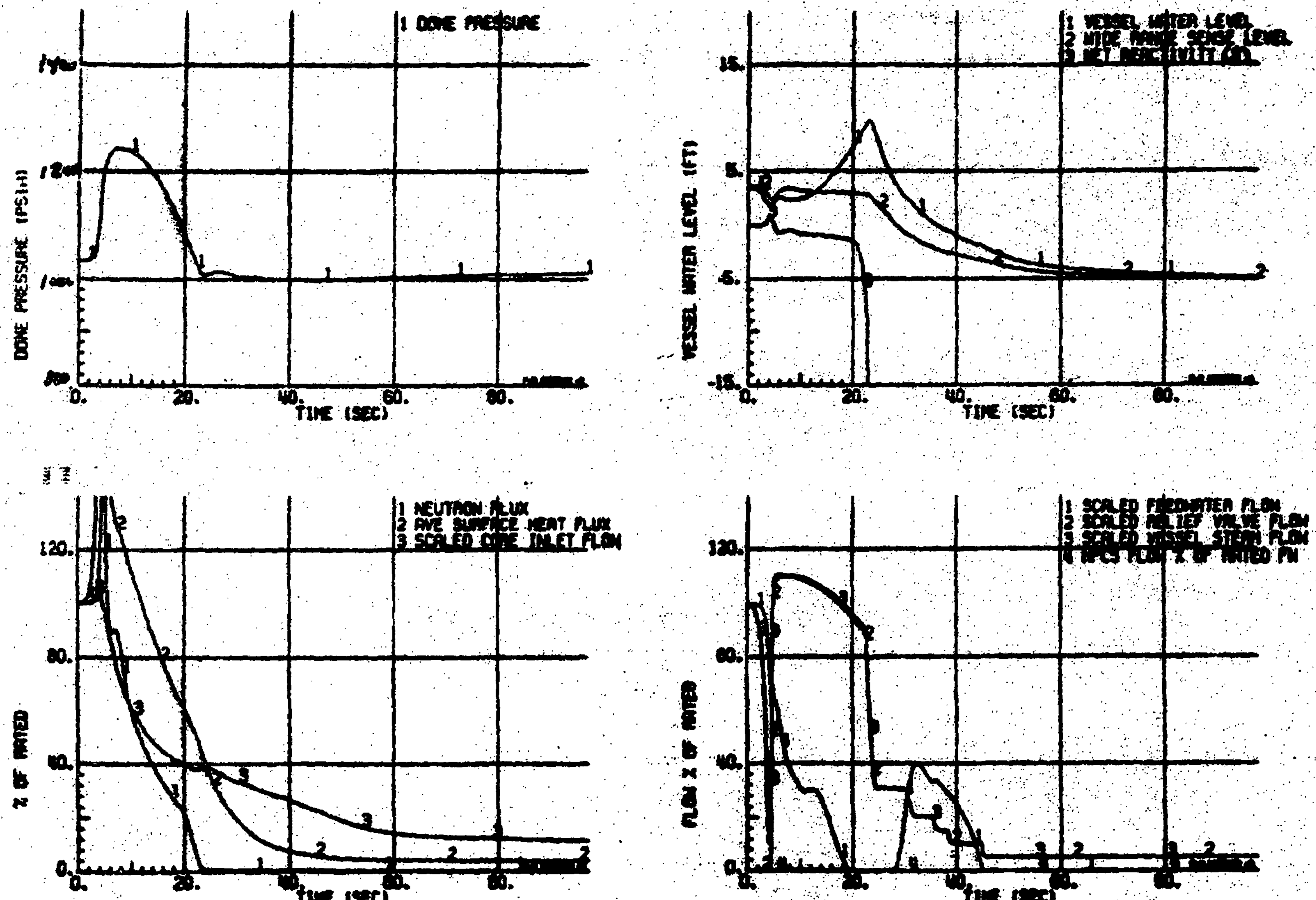


Figure 3.2.1-1. ATWS MSIV, ARI

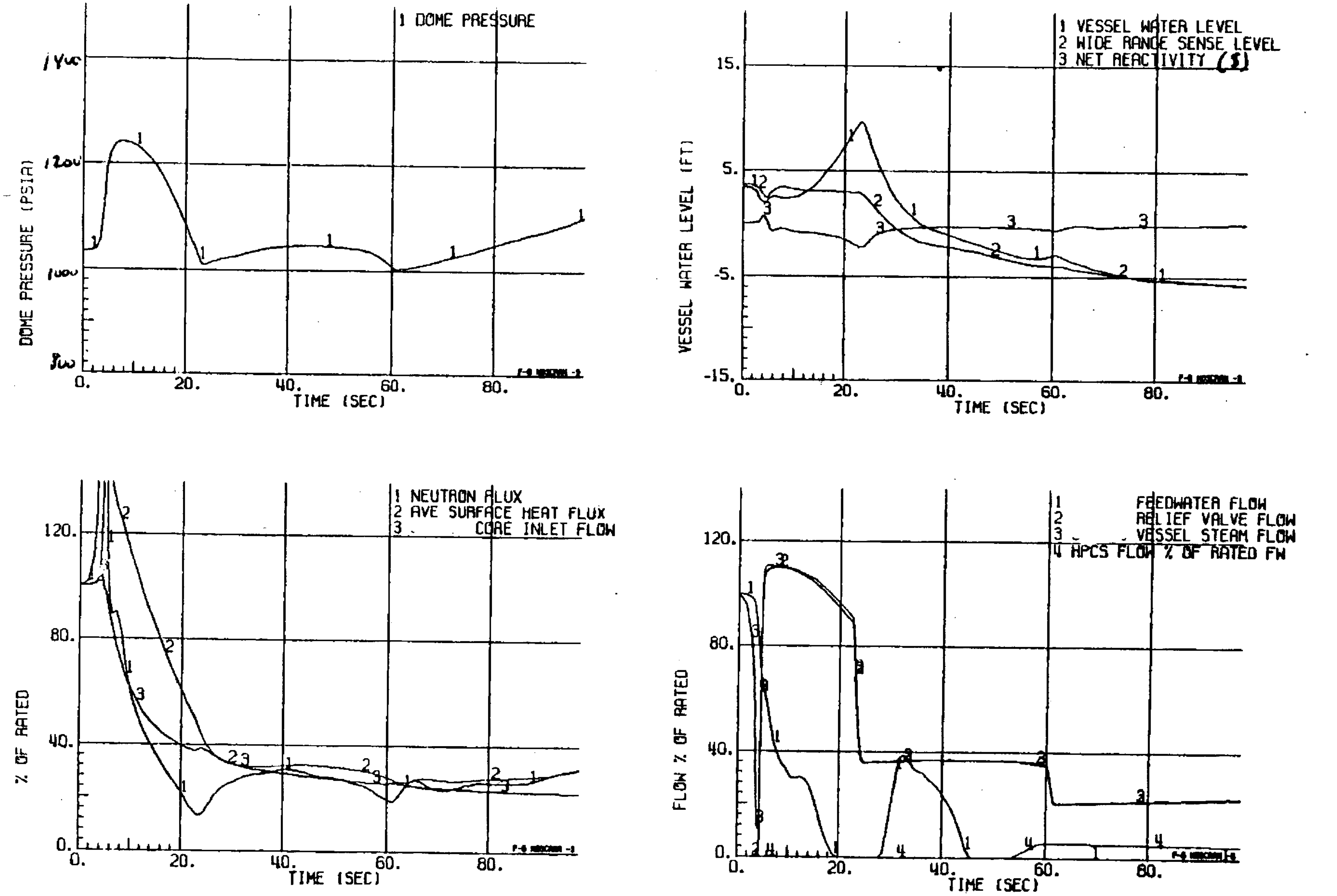


Figure 3.2.1-2. ATWS MSIV Closure, ARI Failure

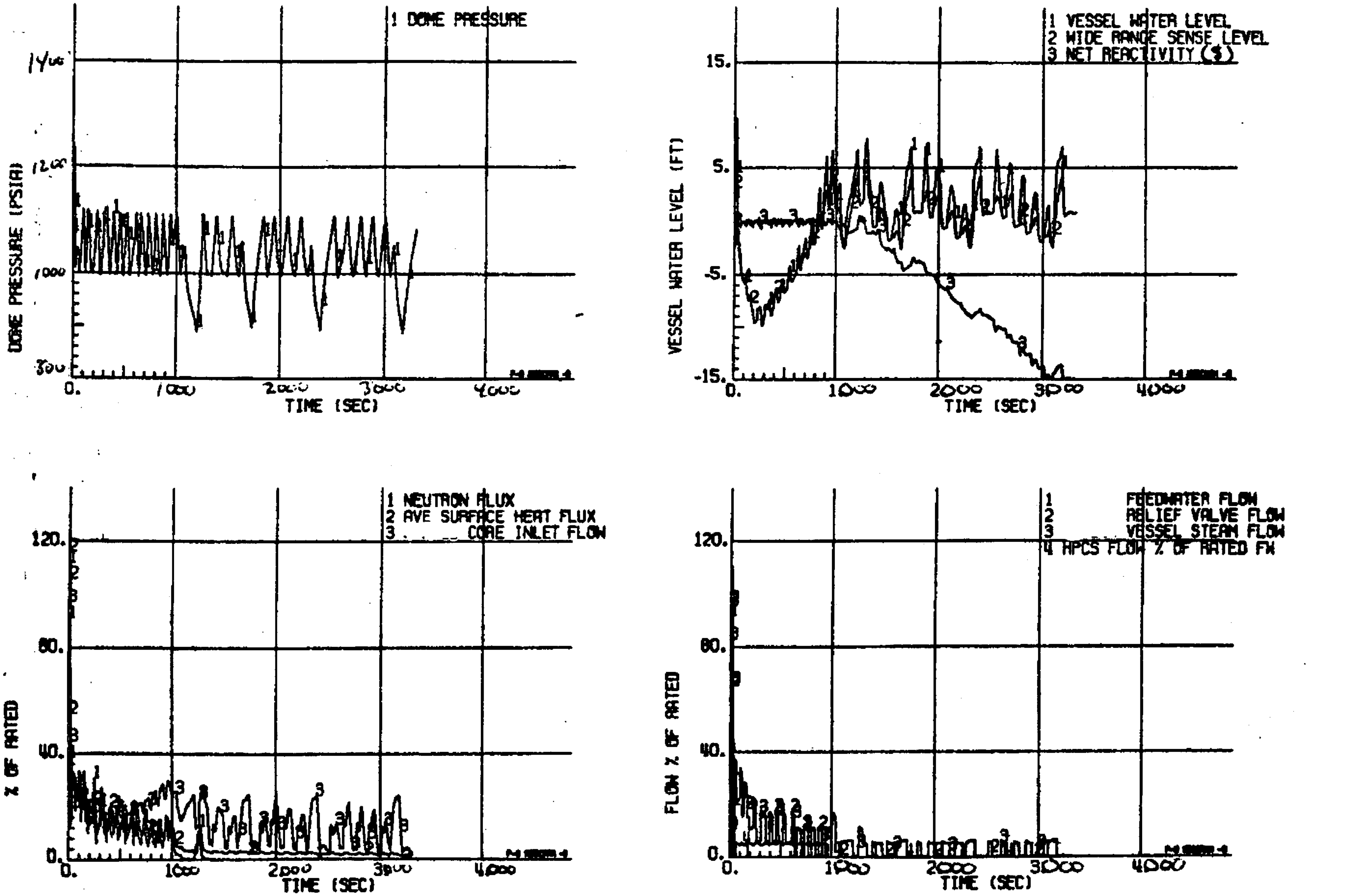


Figure 3.2.1-3. ATWS MSIV Closure, ARI Failure

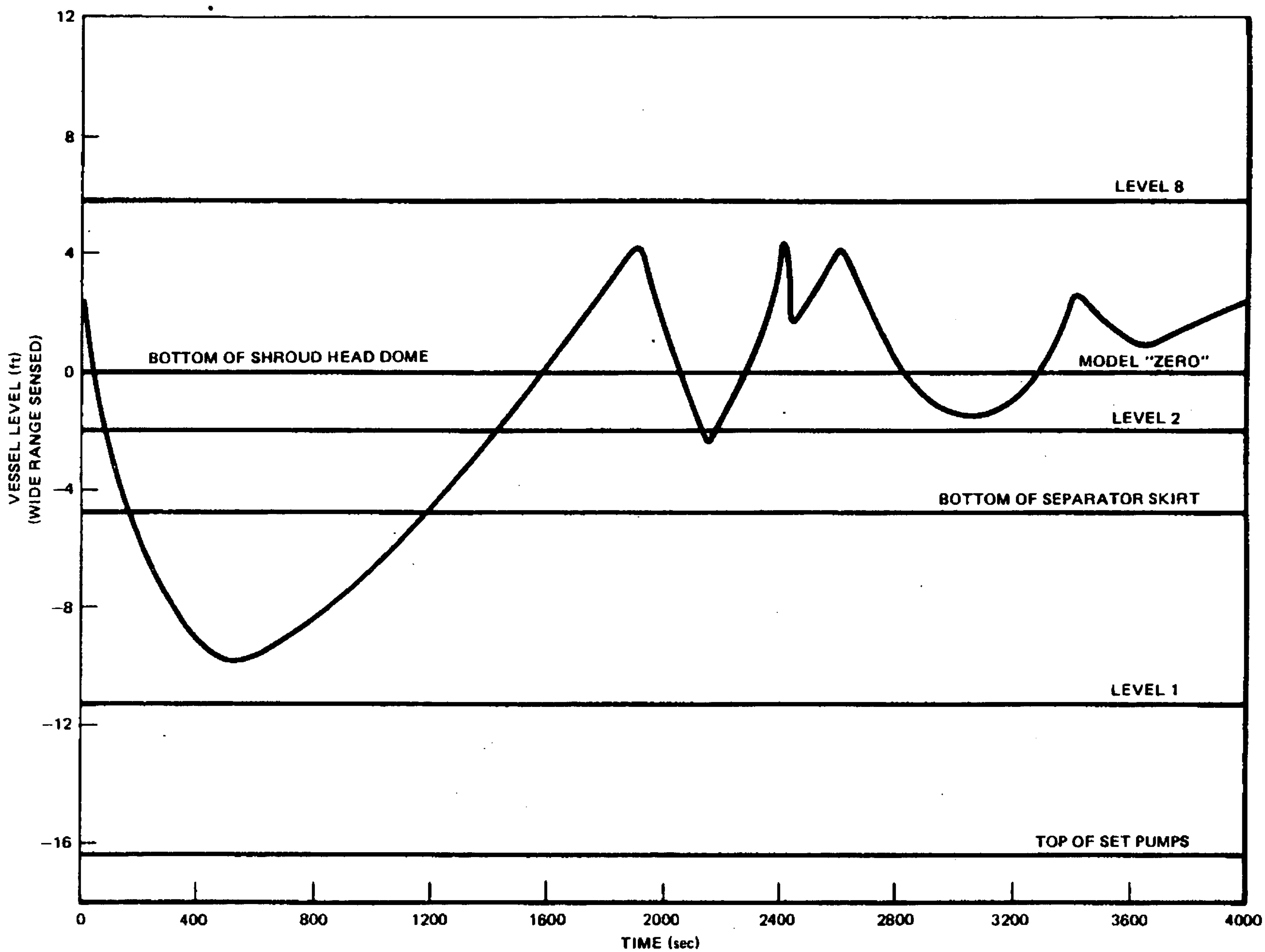


Figure 3.2.1-4. BWR/5 MSIV Vessel Water Level

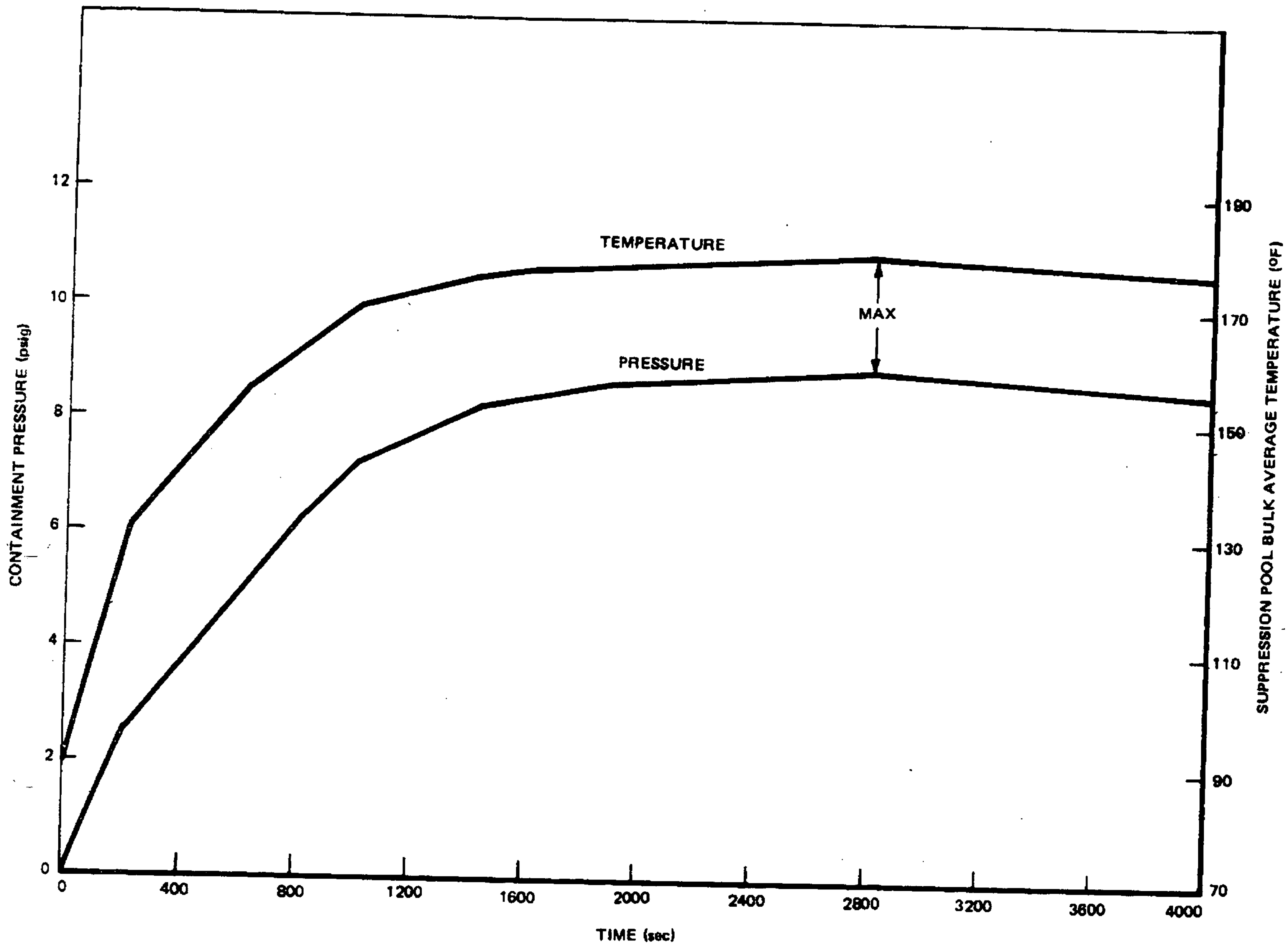


Figure 3.2.1-5. BWR/5 MSIV Bulk Pool Temperature and Containment Pressure

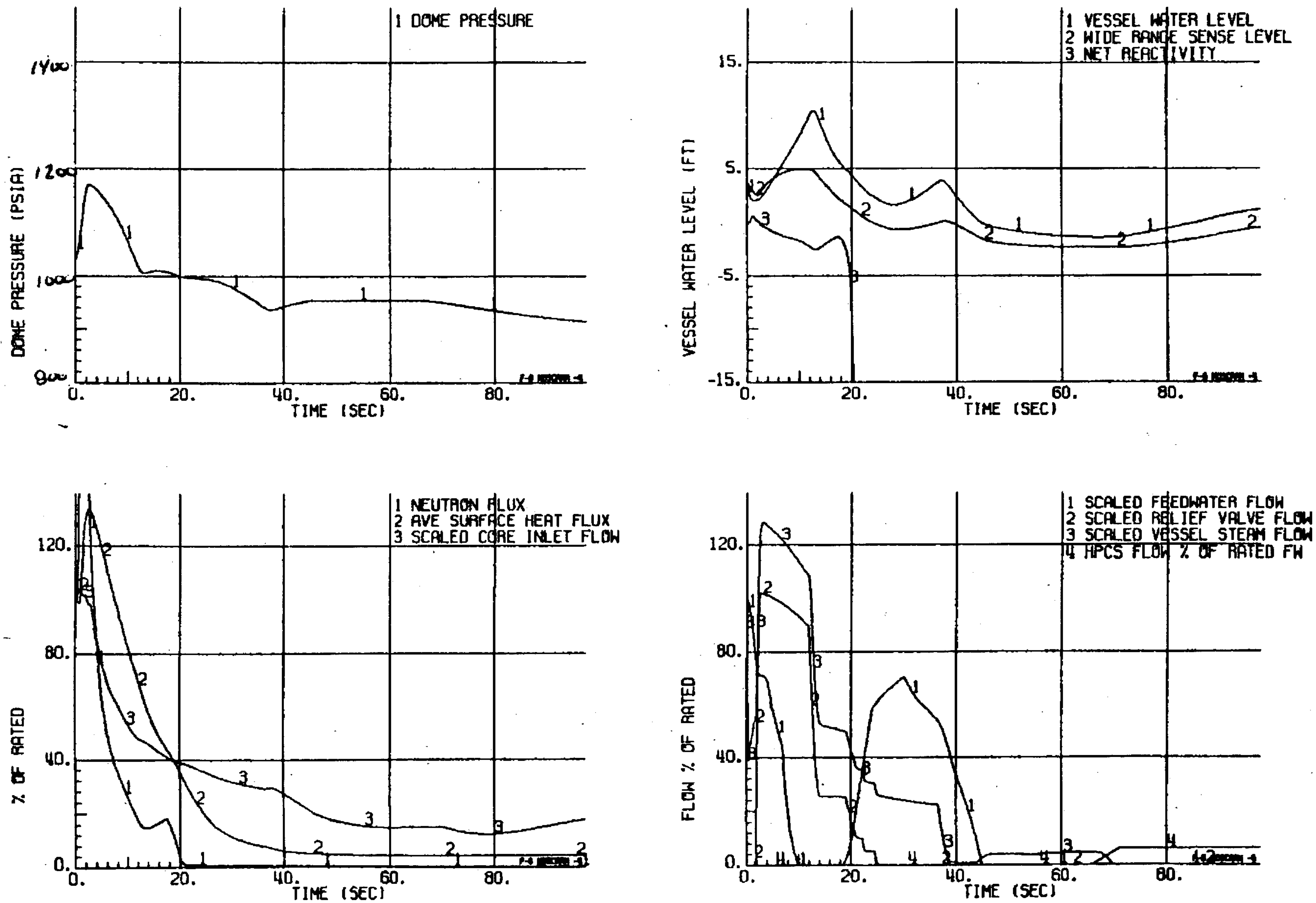


Figure 3.2.2-1. ATWS Turbine Trip, ARI Failure

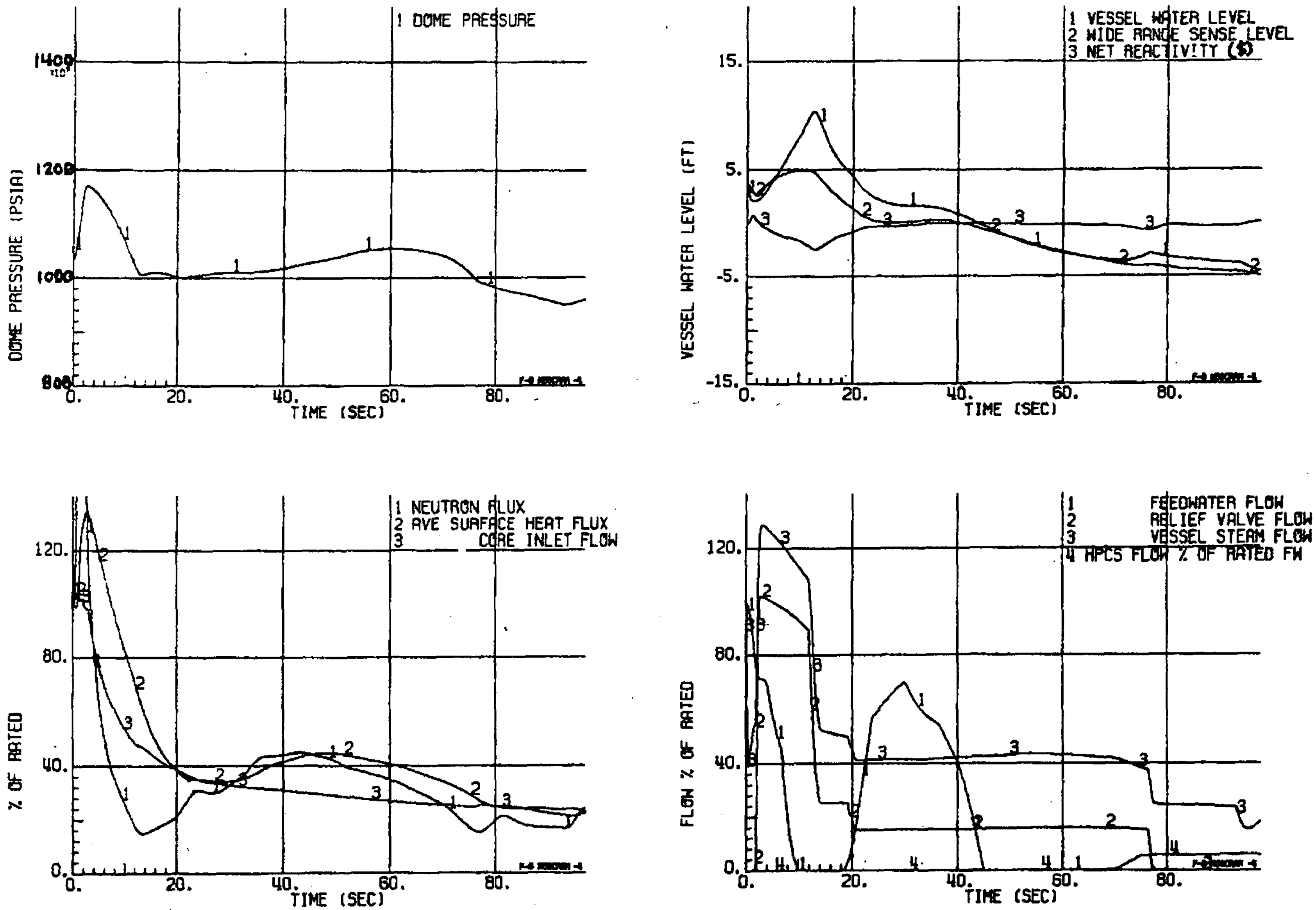


Figure 3.2.2-2. ATWS Turbine Trip, ARI Failure

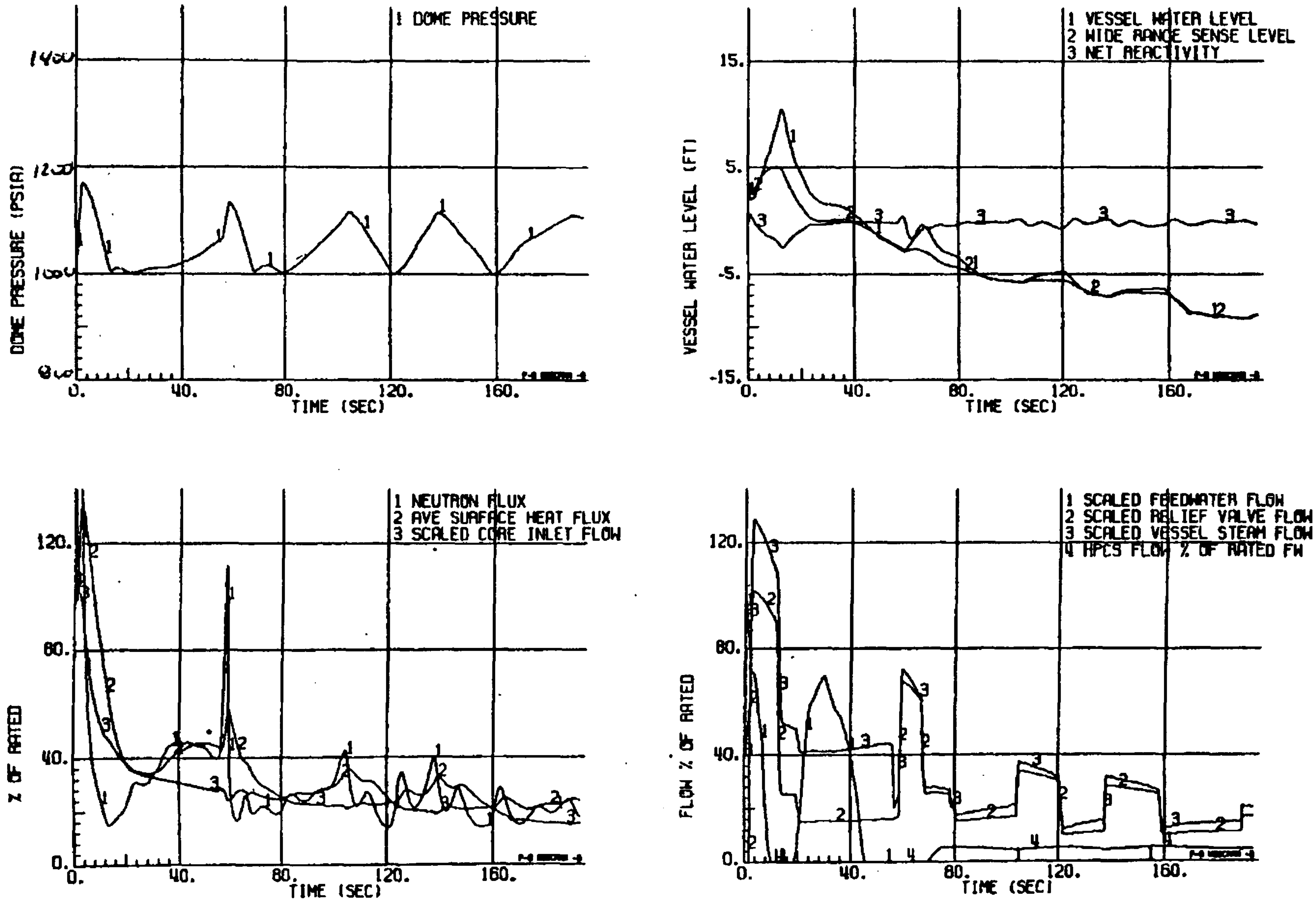


Figure 3.3.2-2. ATWS Turbine Trip, ARI Failure

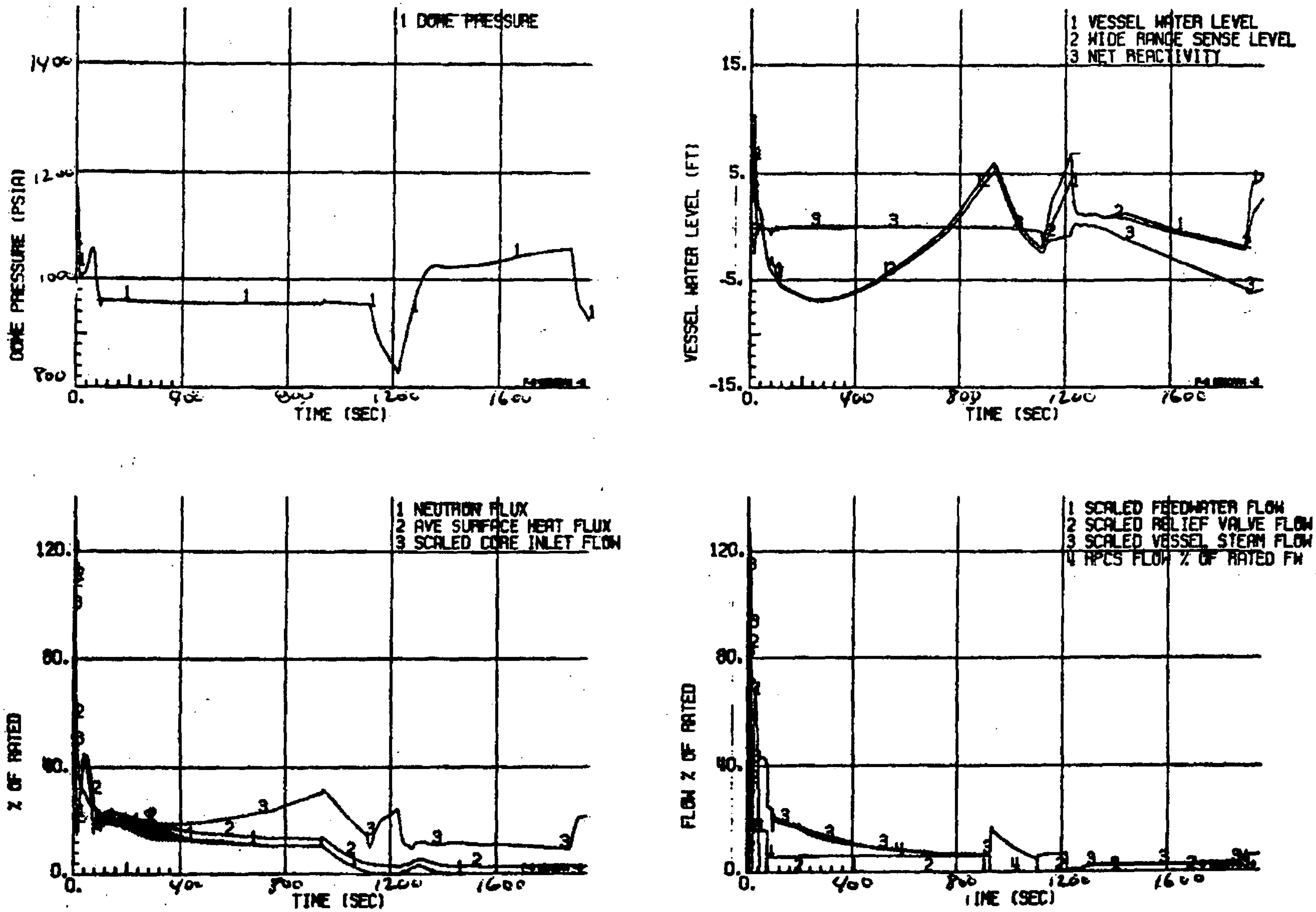


Figure 3.2.2-3. ATWS Turbine Trip, ARI Failure

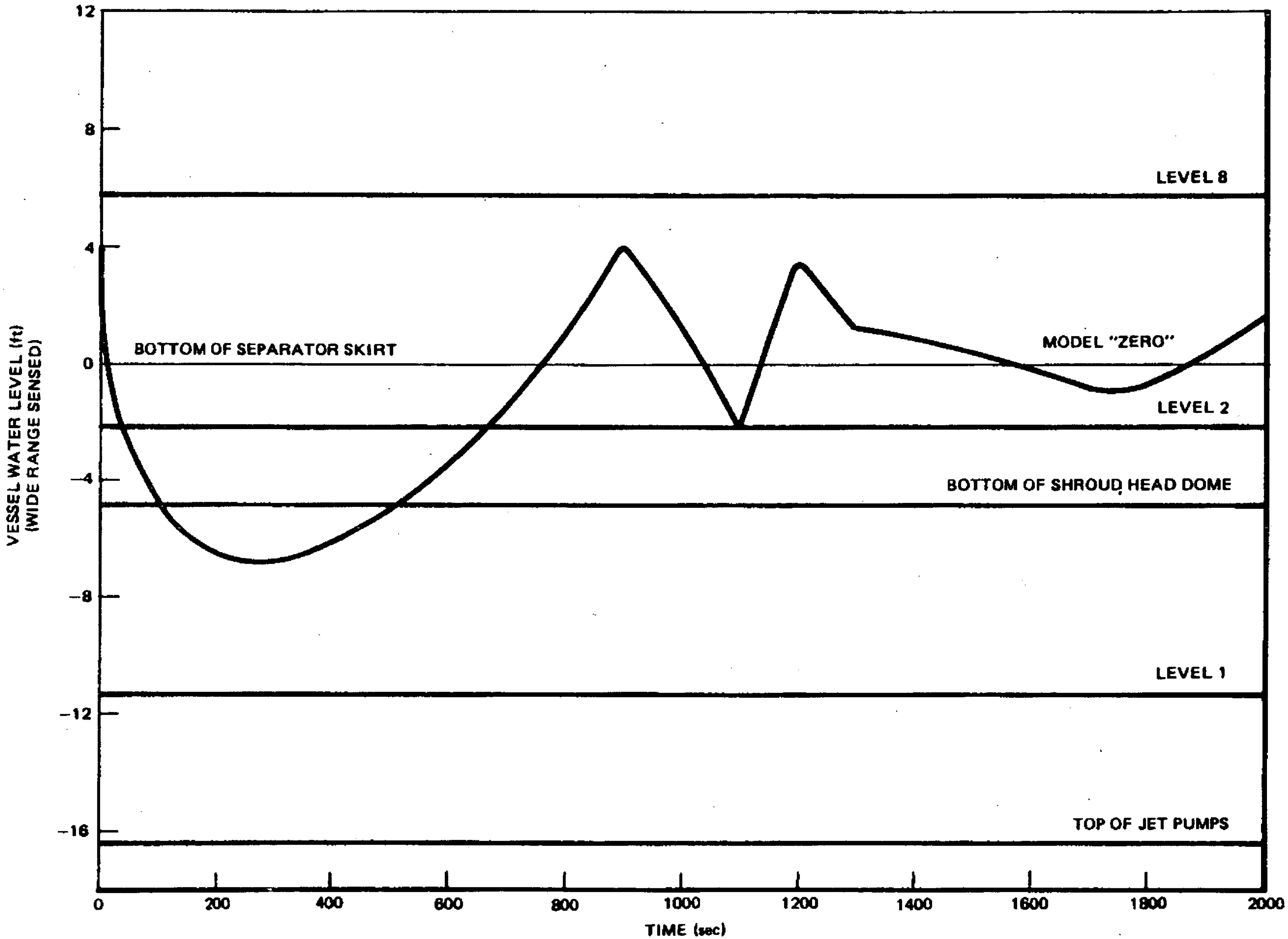


Figure 3.2.2-4. BWR/5 Turbine Trip Vessel Water Level

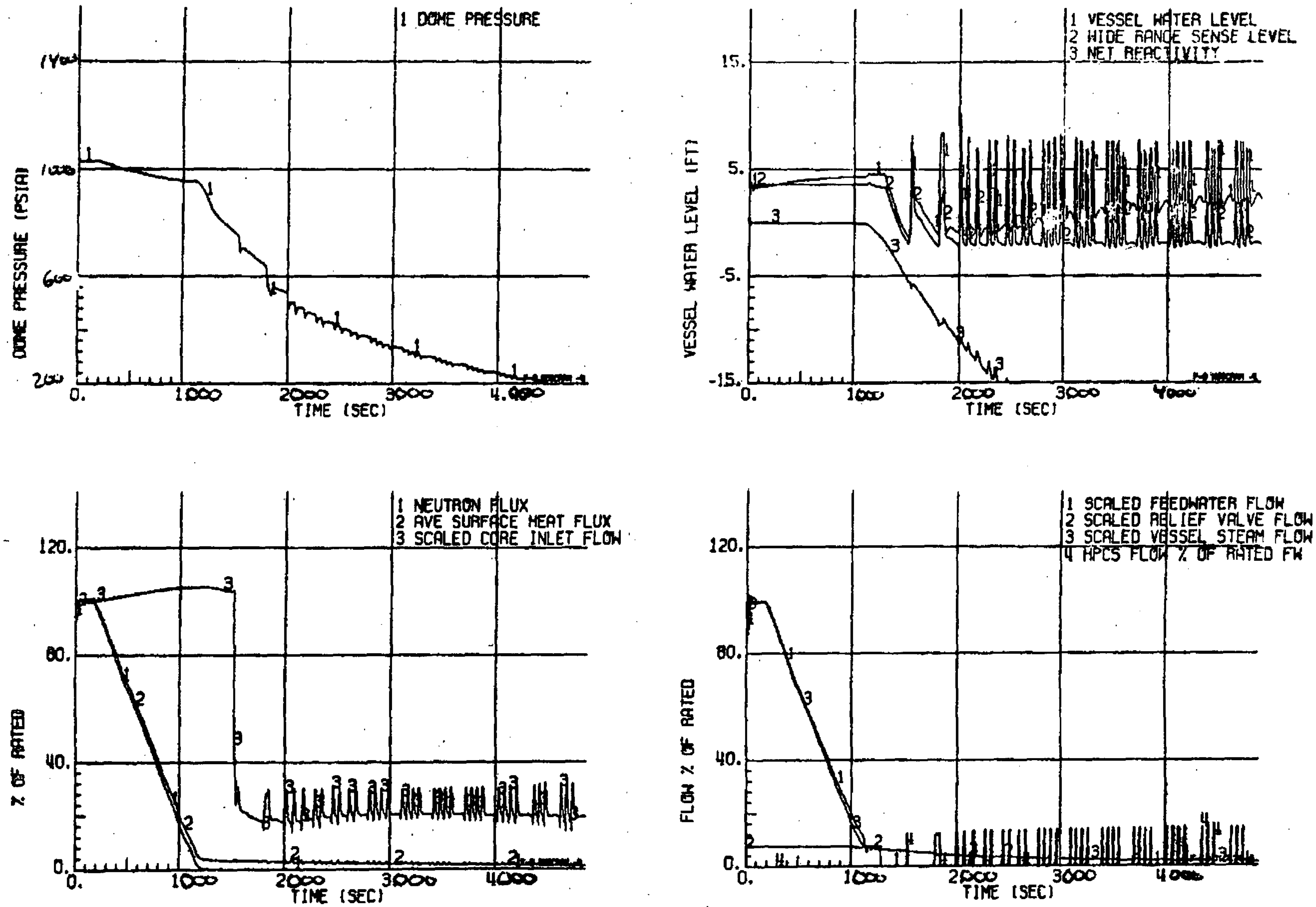


Figure 3.2..3-1. ATWS IORV, ARI Failure

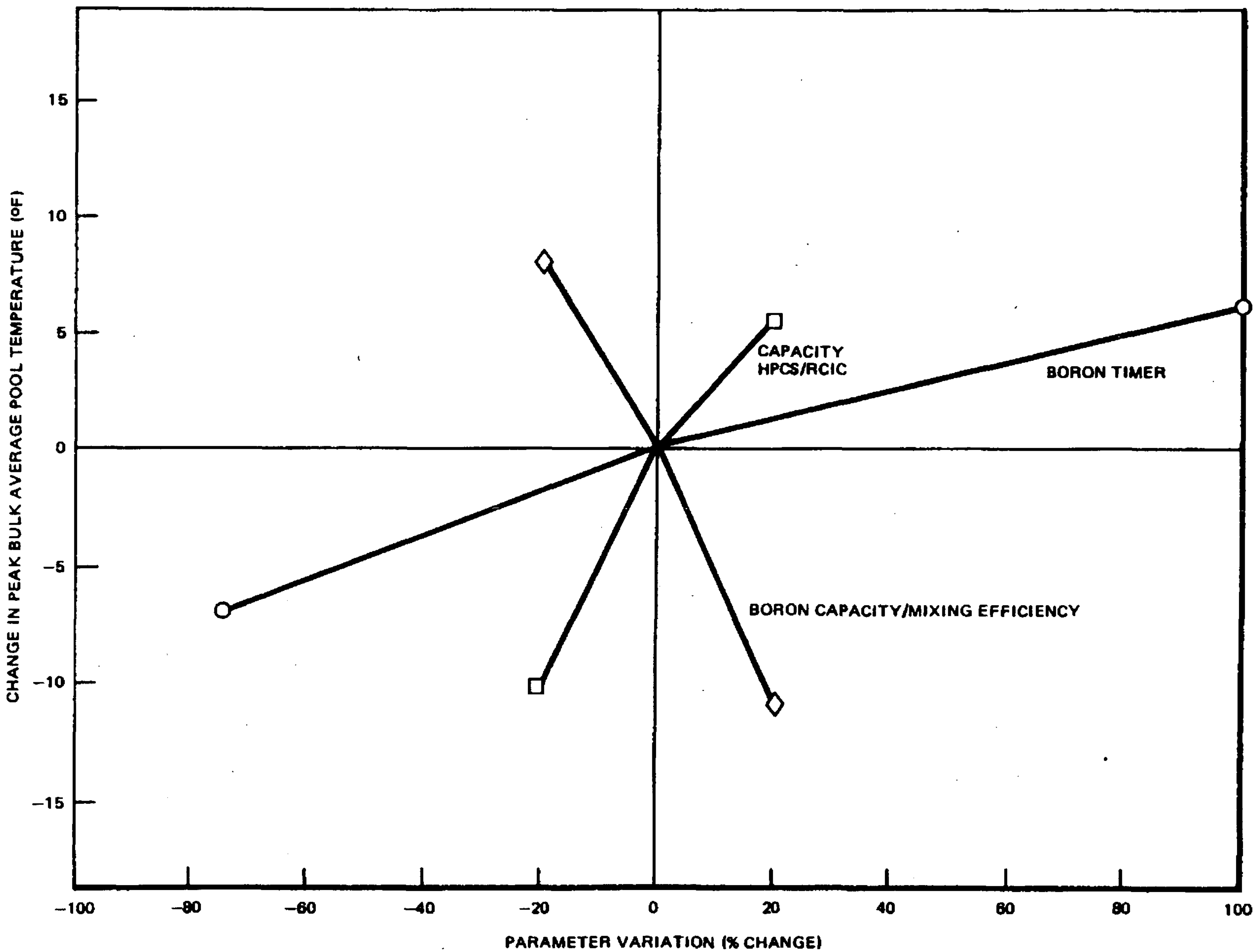


Figure 3.2.4.1-1. BWR/5 MSIV Peak Bulk Average Pool Temperature Sensitivities

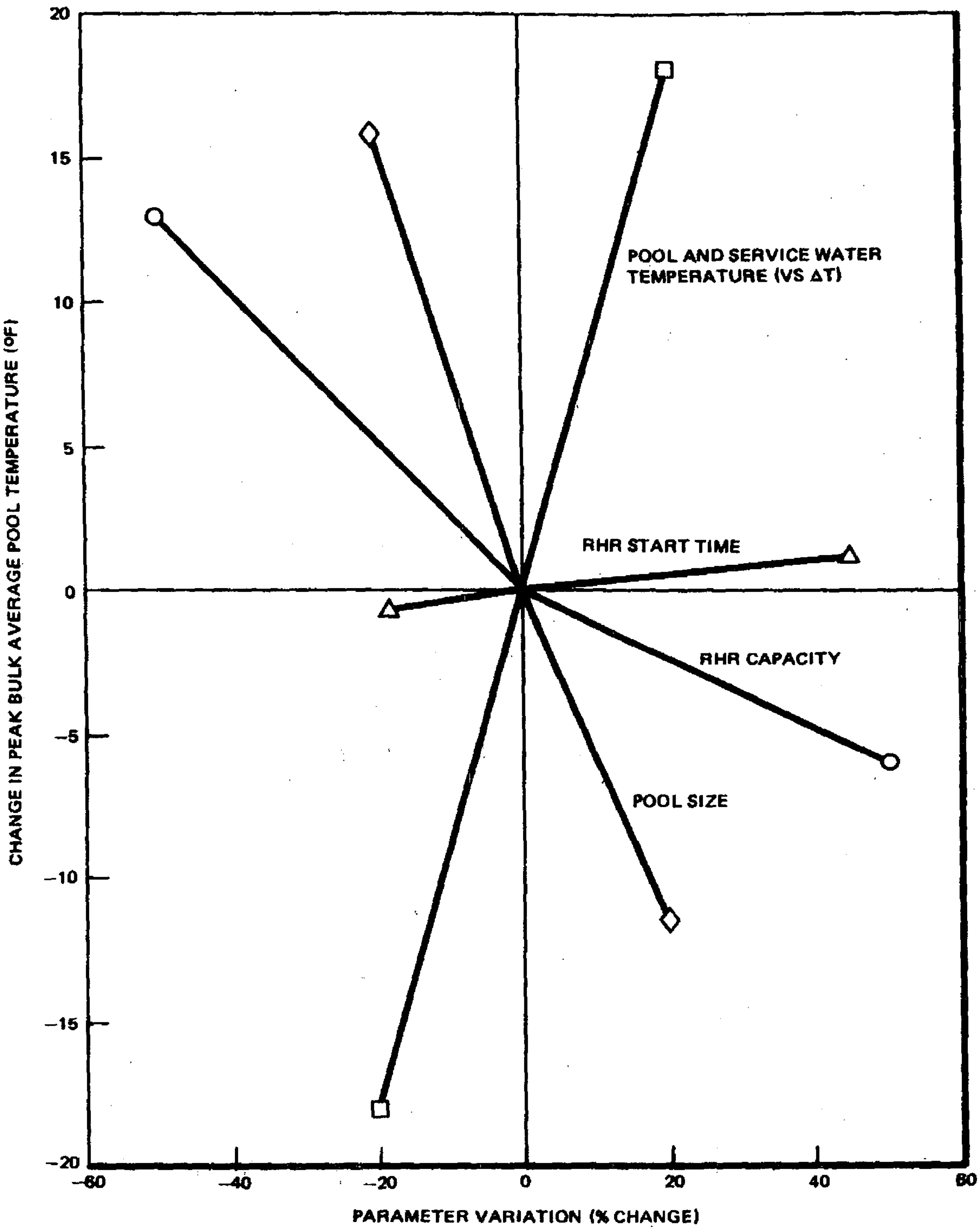


Figure 3.2.4.1.4-1. BWR/5 MSIV Peak Bulk Average Pool Temperature Sensitivities

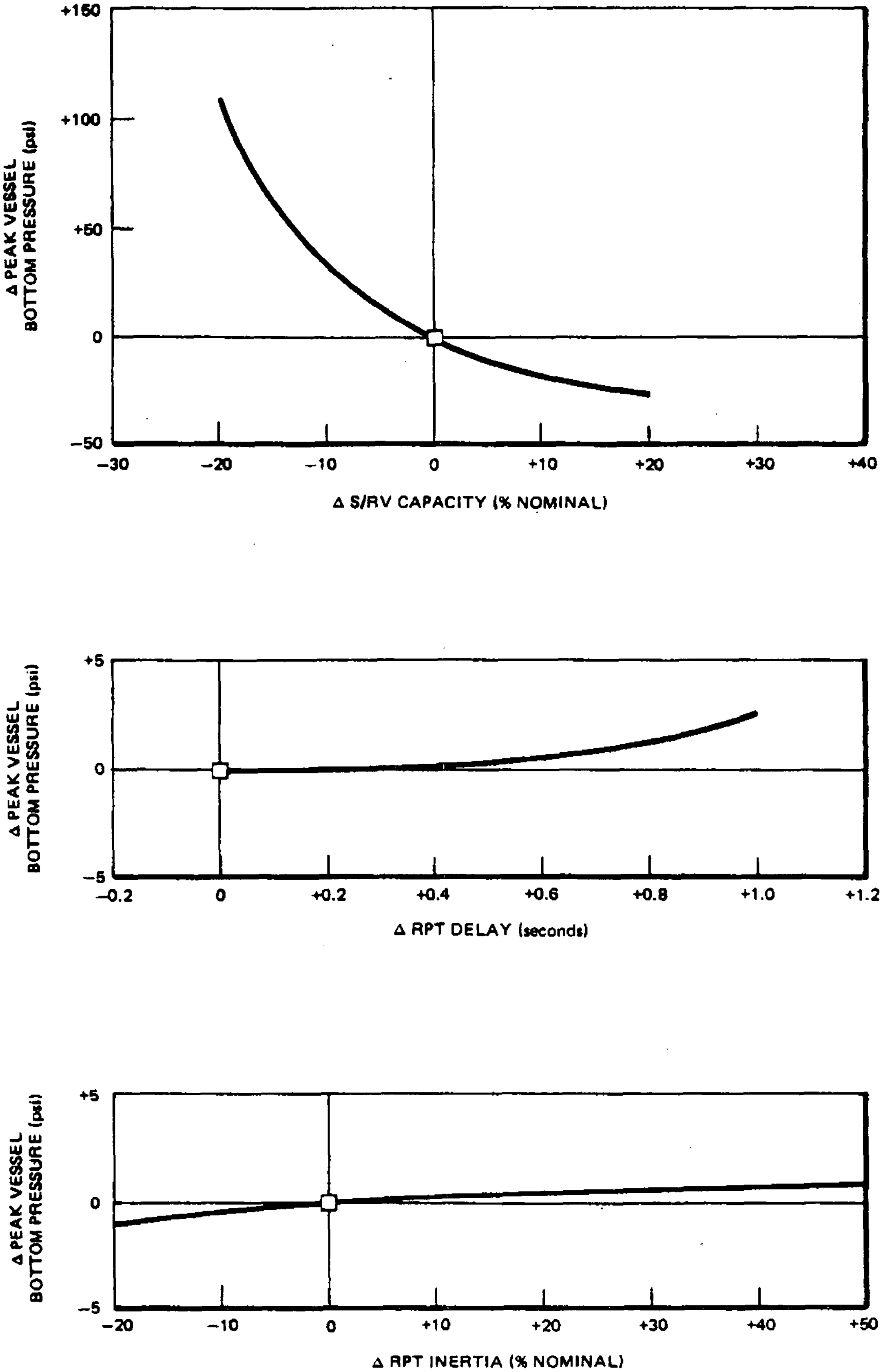


Figure 3.2.4.1.9-1. BWR/5 ATWS MSIV, ARI Failure

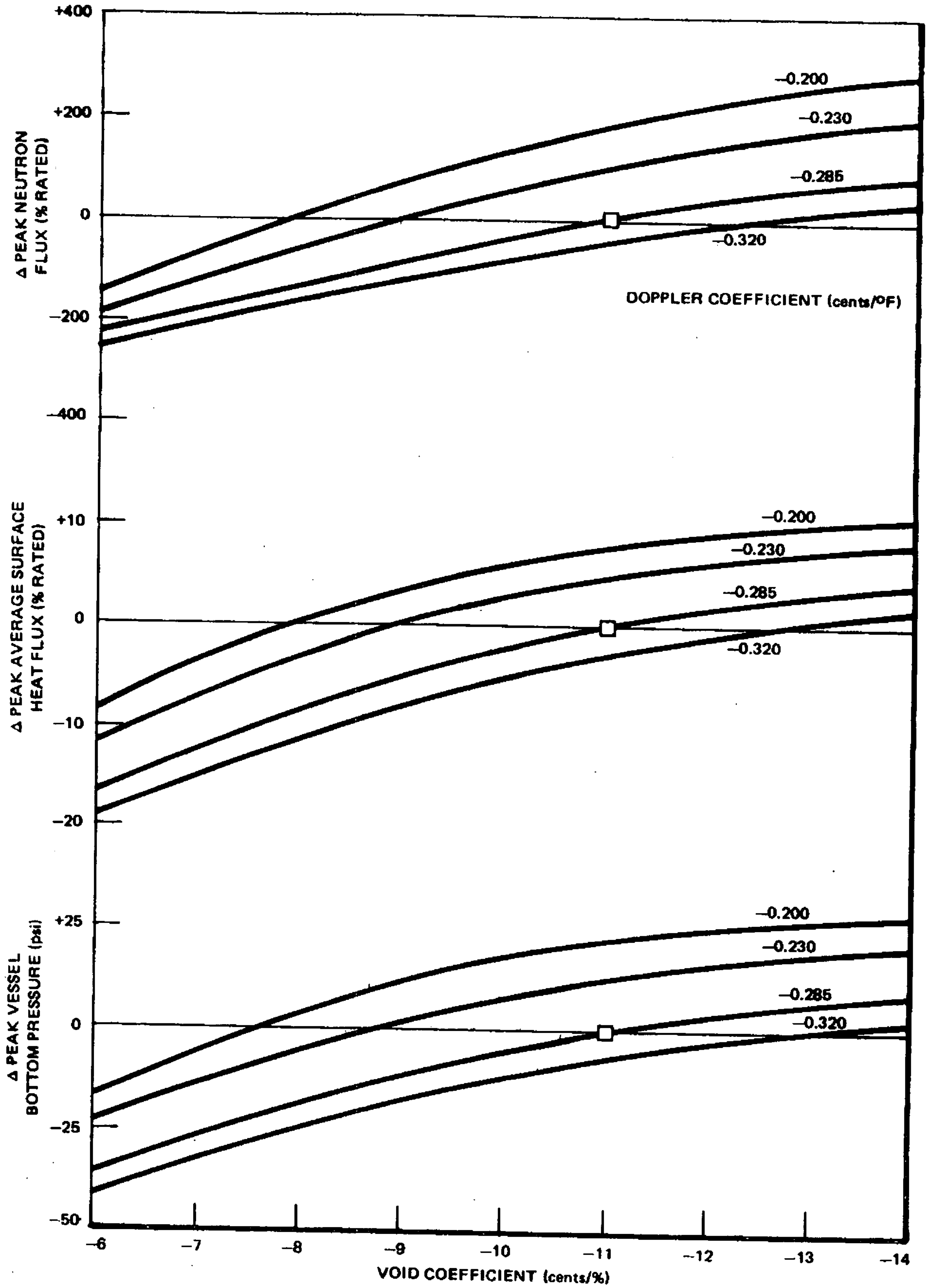
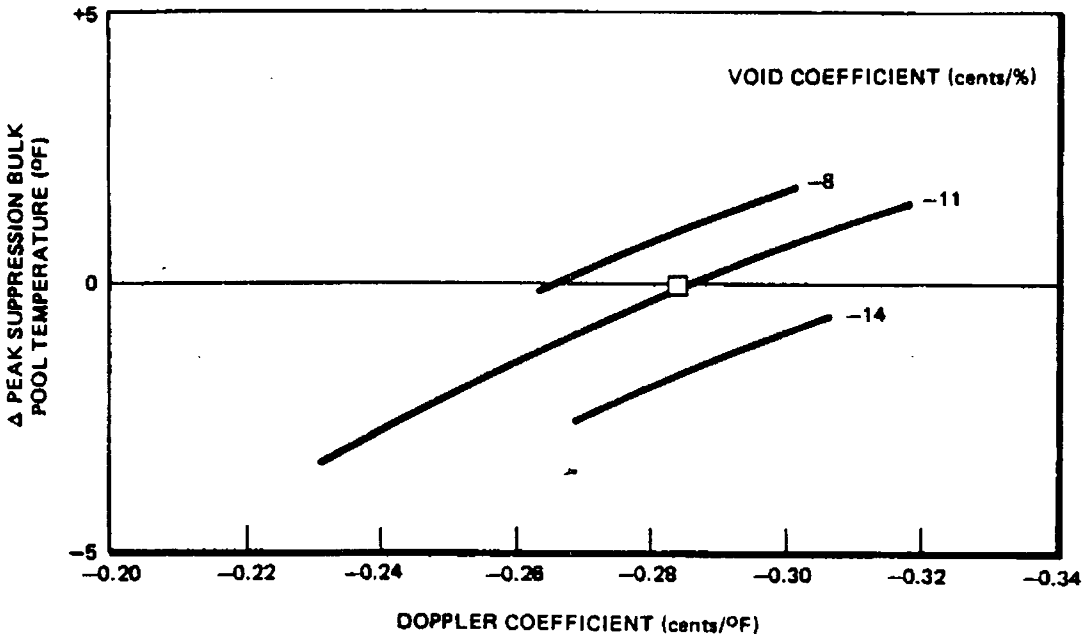


Figure 3.2.4.1.13-1. BWR/5-251 ATWS MSIV, ARI Failure



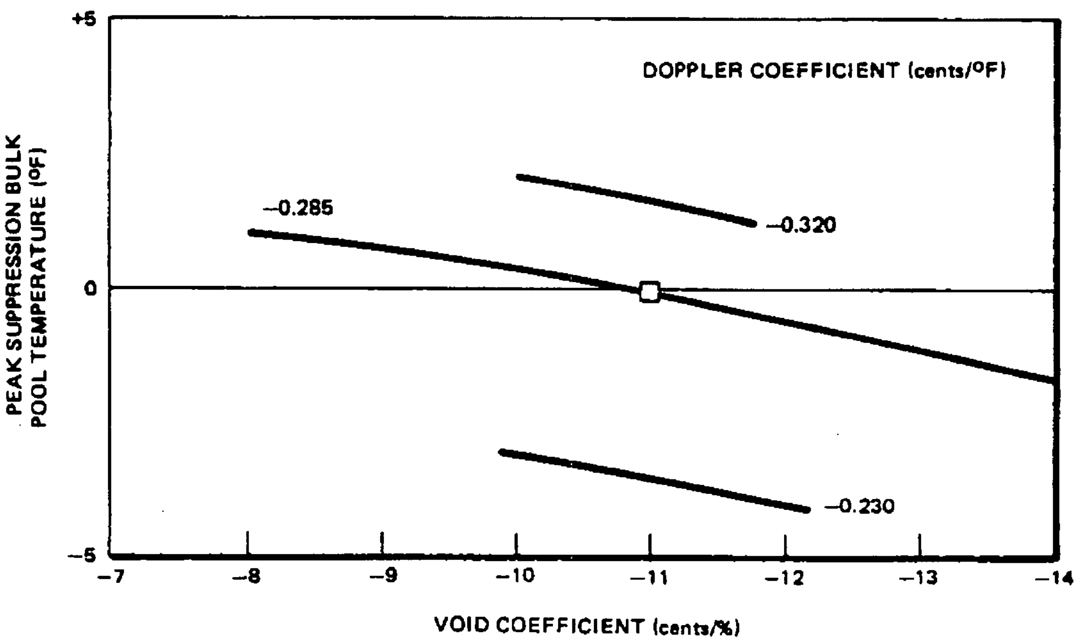


Figure 3.2.4.1.13-2. BWR/5 ATWS MSIV, ARI Failure

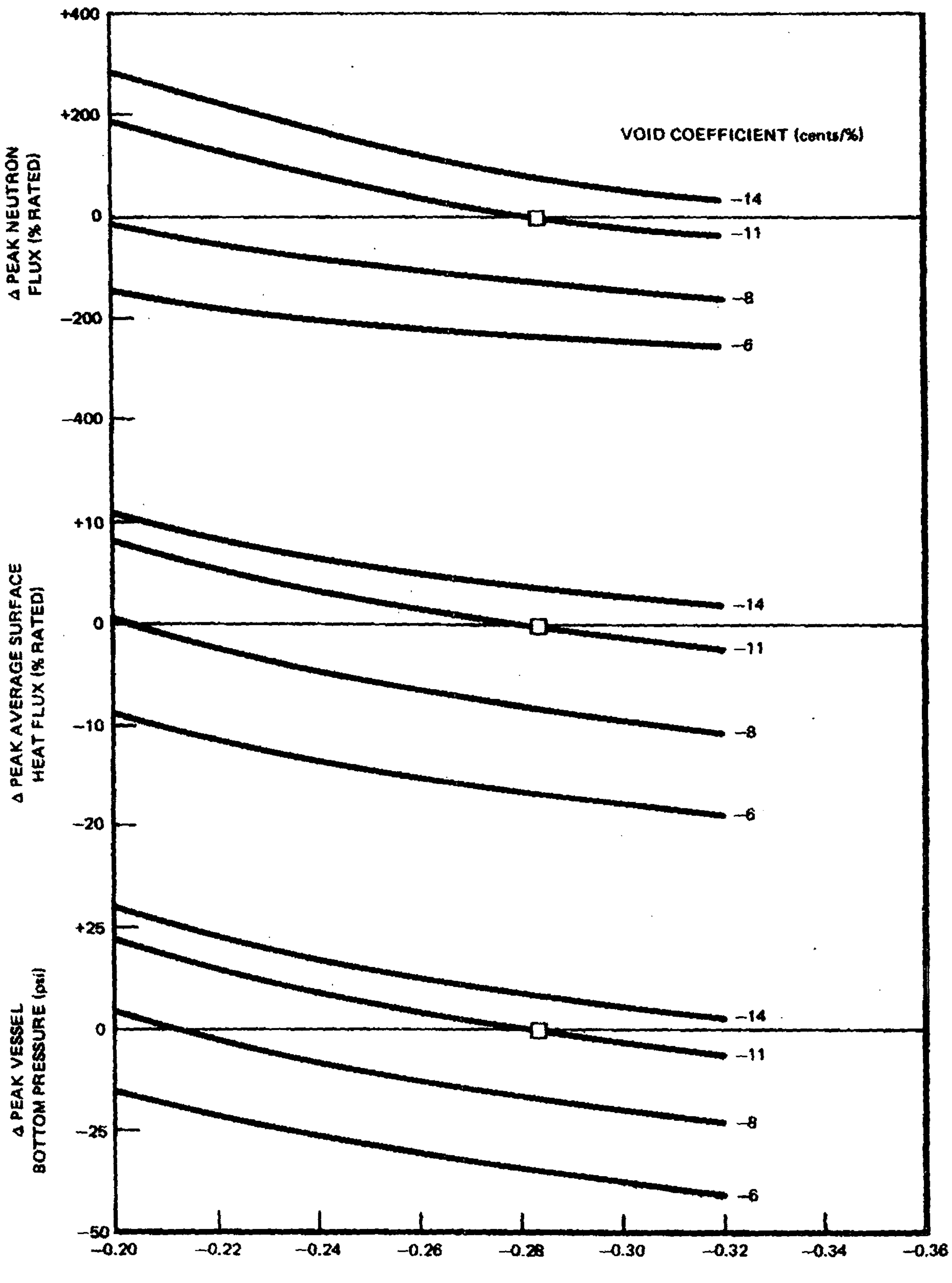


Figure 3.2.4.1.14-1. BWR/5-251 ATWS MSIV, ARI Failure

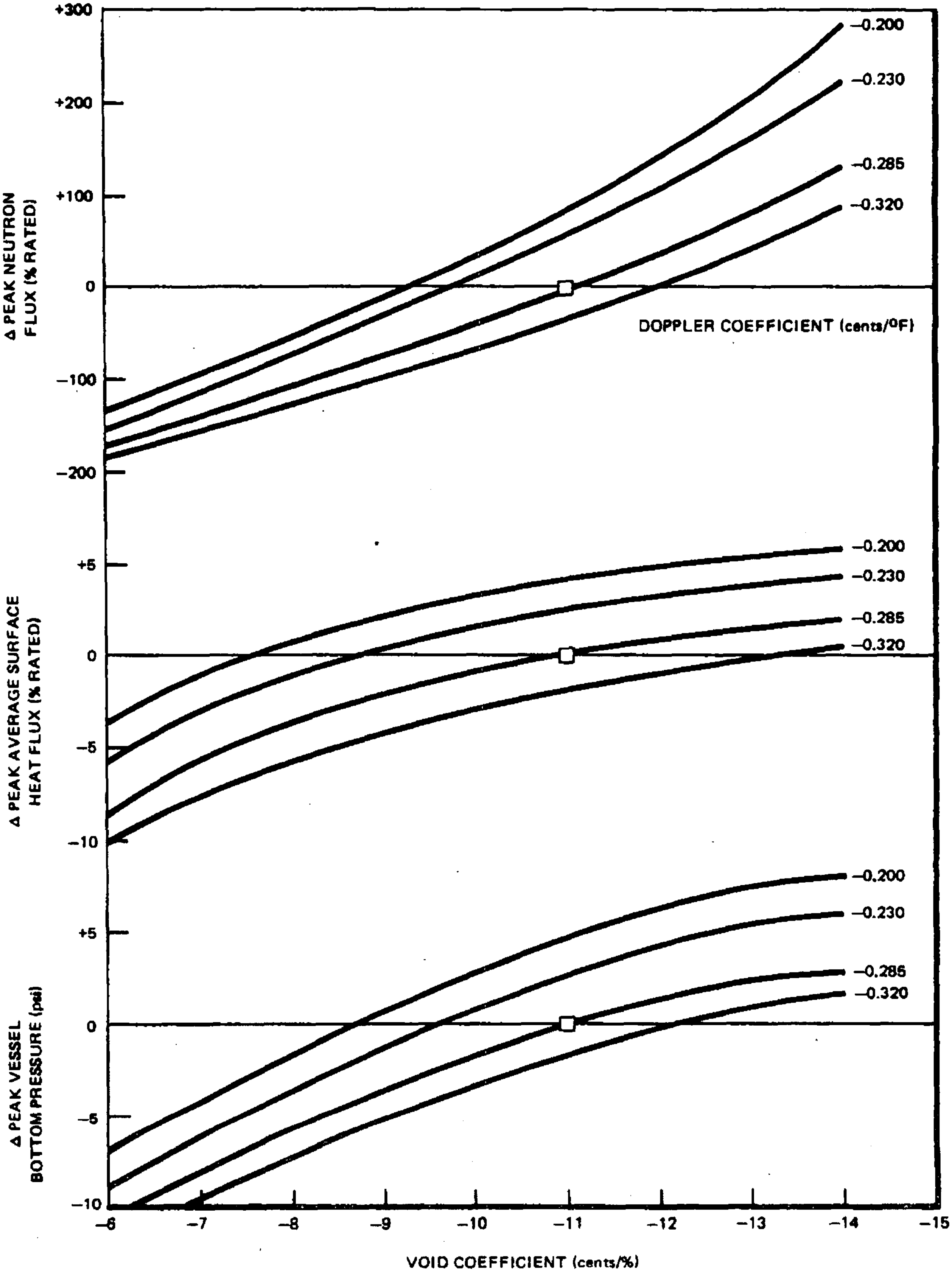
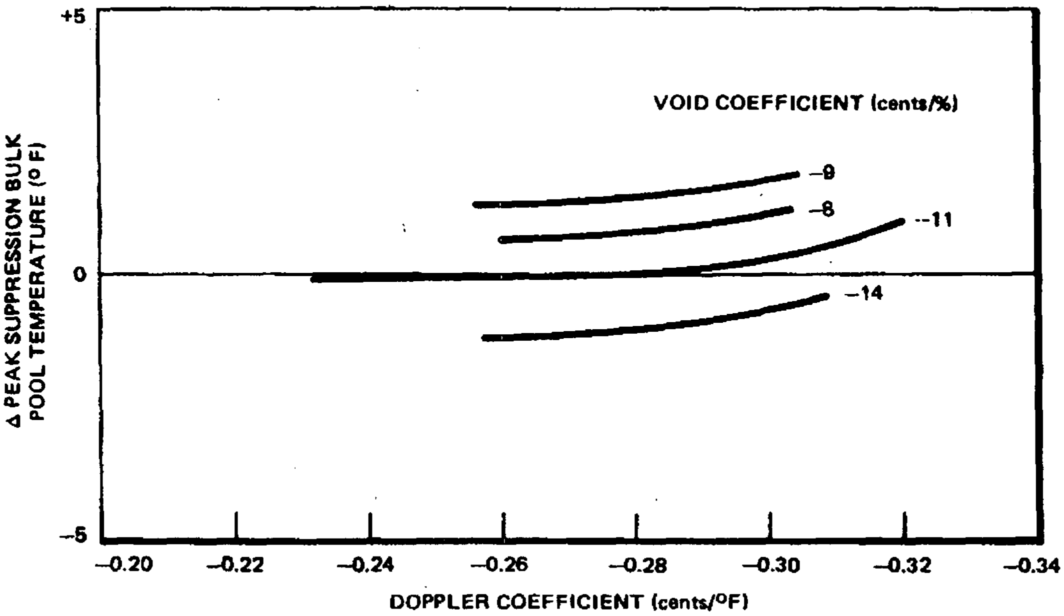


Figure 3.2.4.2.5-1. BWR/5-251 ATWS Turbine Trip, ARI Failure



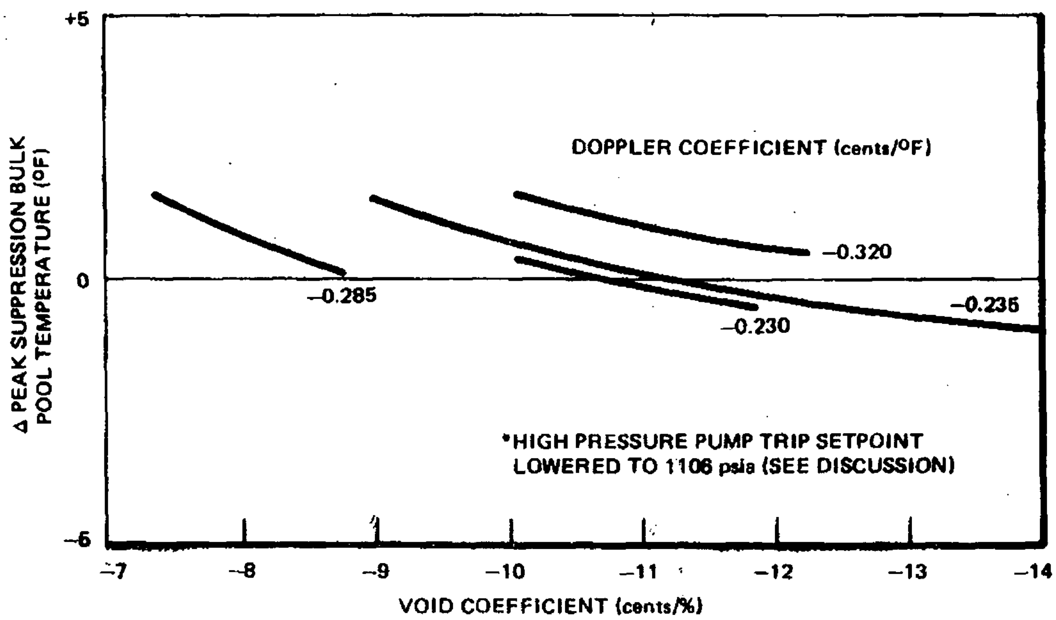


Figure 3.2.4.2.5-2. BWR/5-251 ATWS Turbine Trip

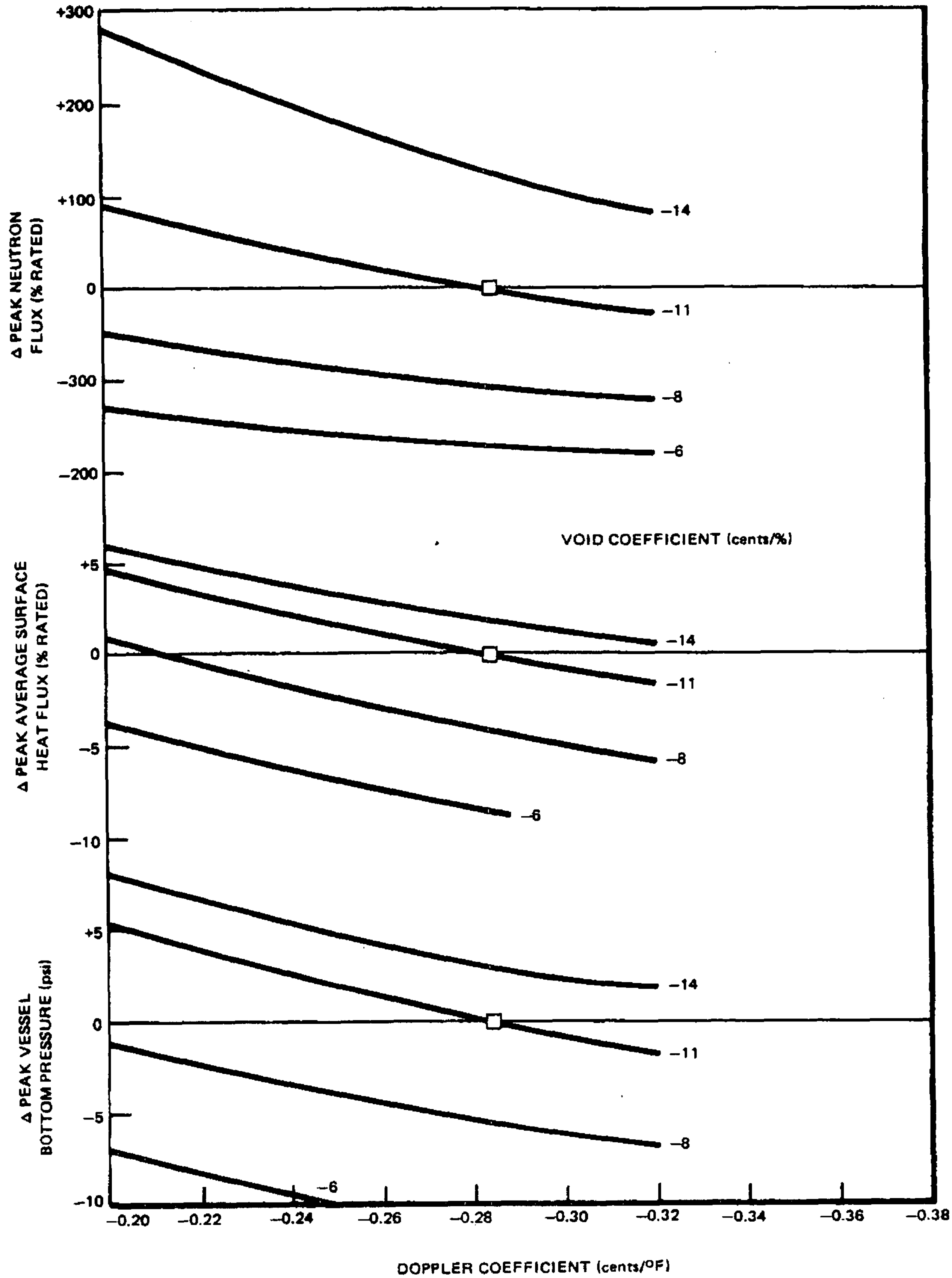


Figure 3.2.4.2.6-1. BWR/5-251 ATWS Turbine Trip

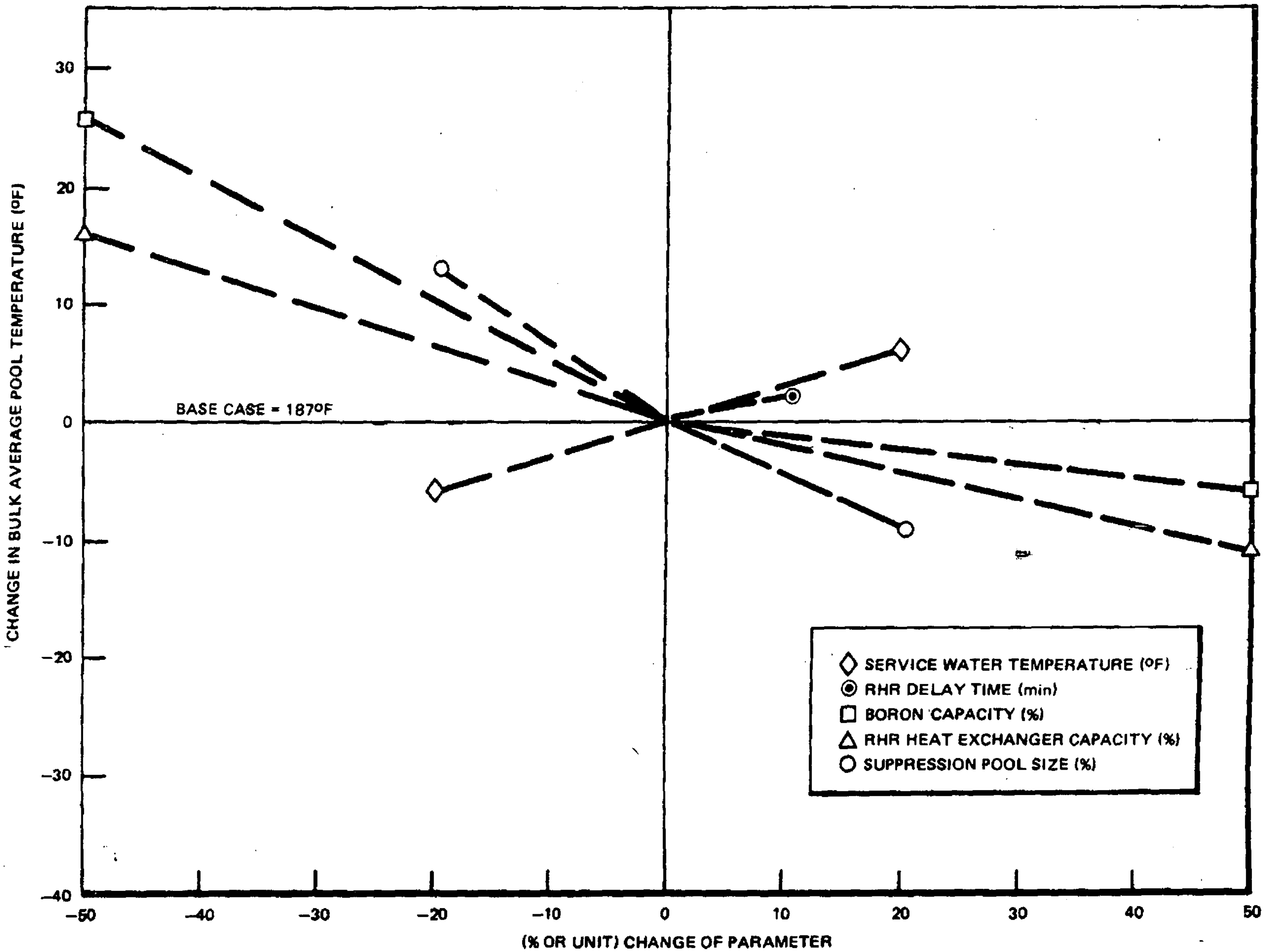


Figure 3.2.4.3-1. BWR/5 Inadvertent Opening of a Safety/Relief Valve Sensitivity

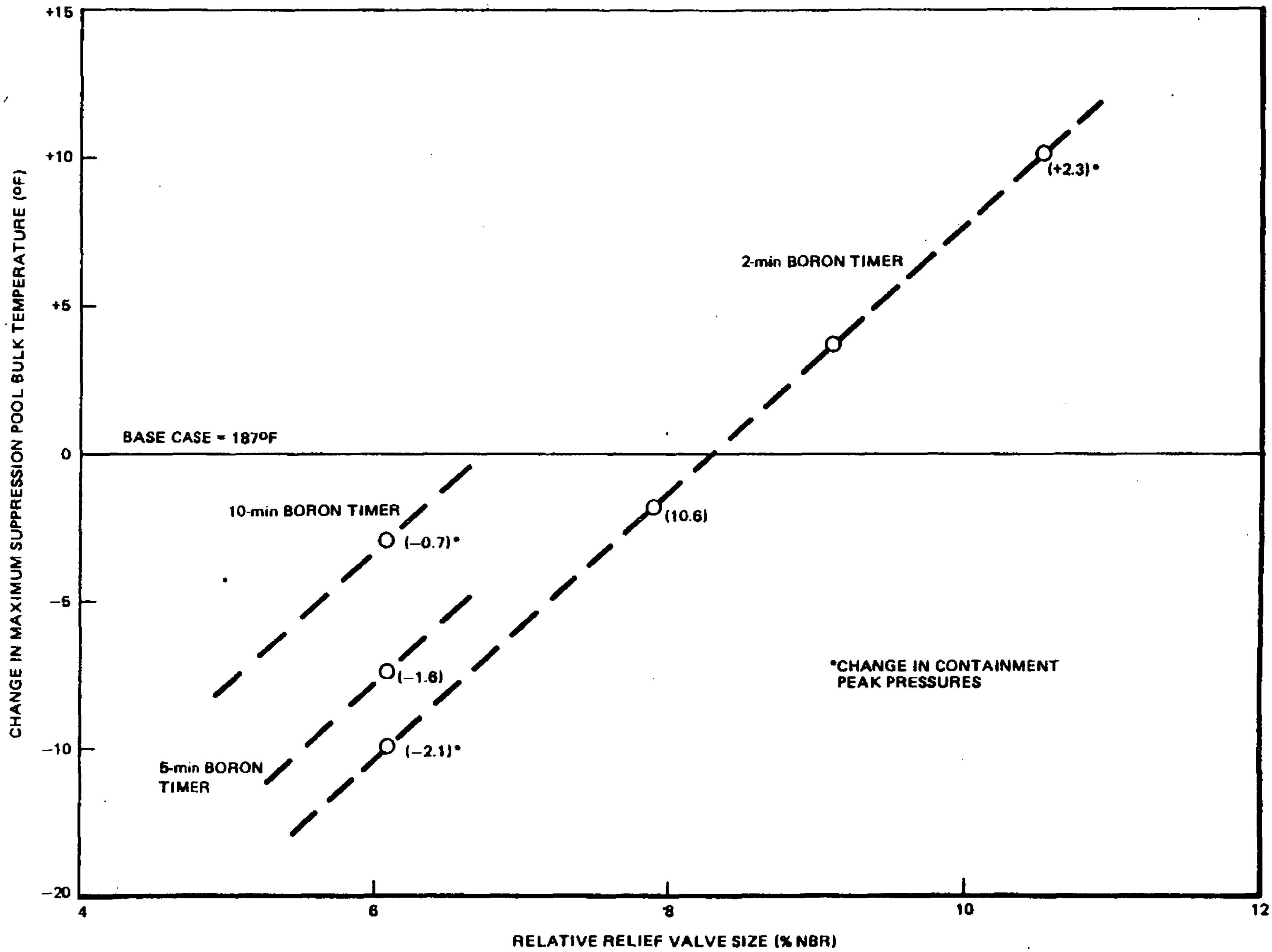


Figure 3.2.4.3-2. IORV Time For Poison to Reach Core versus Maximum Suppression Pool Bulk Temperature

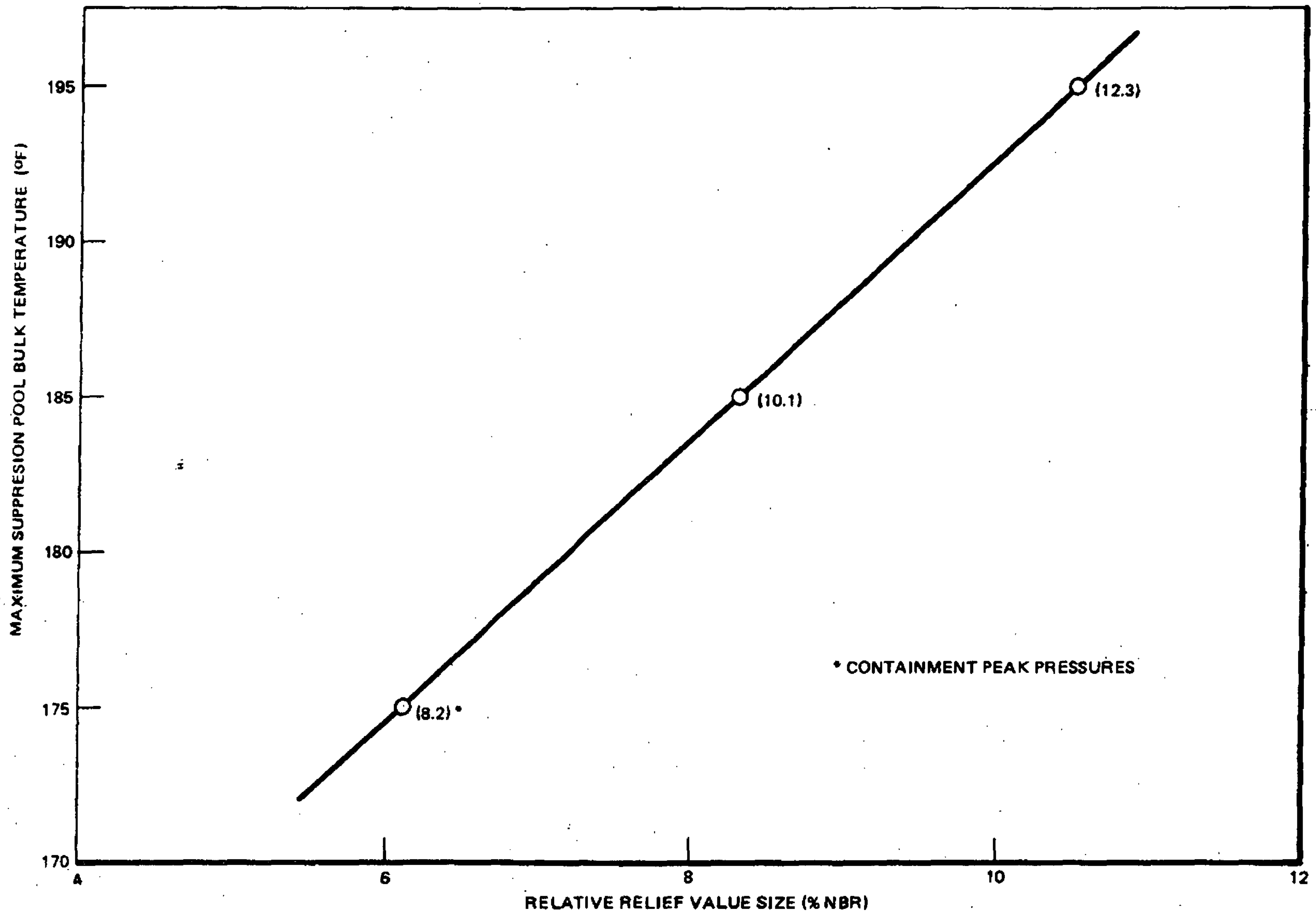


Figure 3.2.4.3-3. IORV - Relative Relief Valve Size versus Maximum Suppression Pool Bulk Temperature

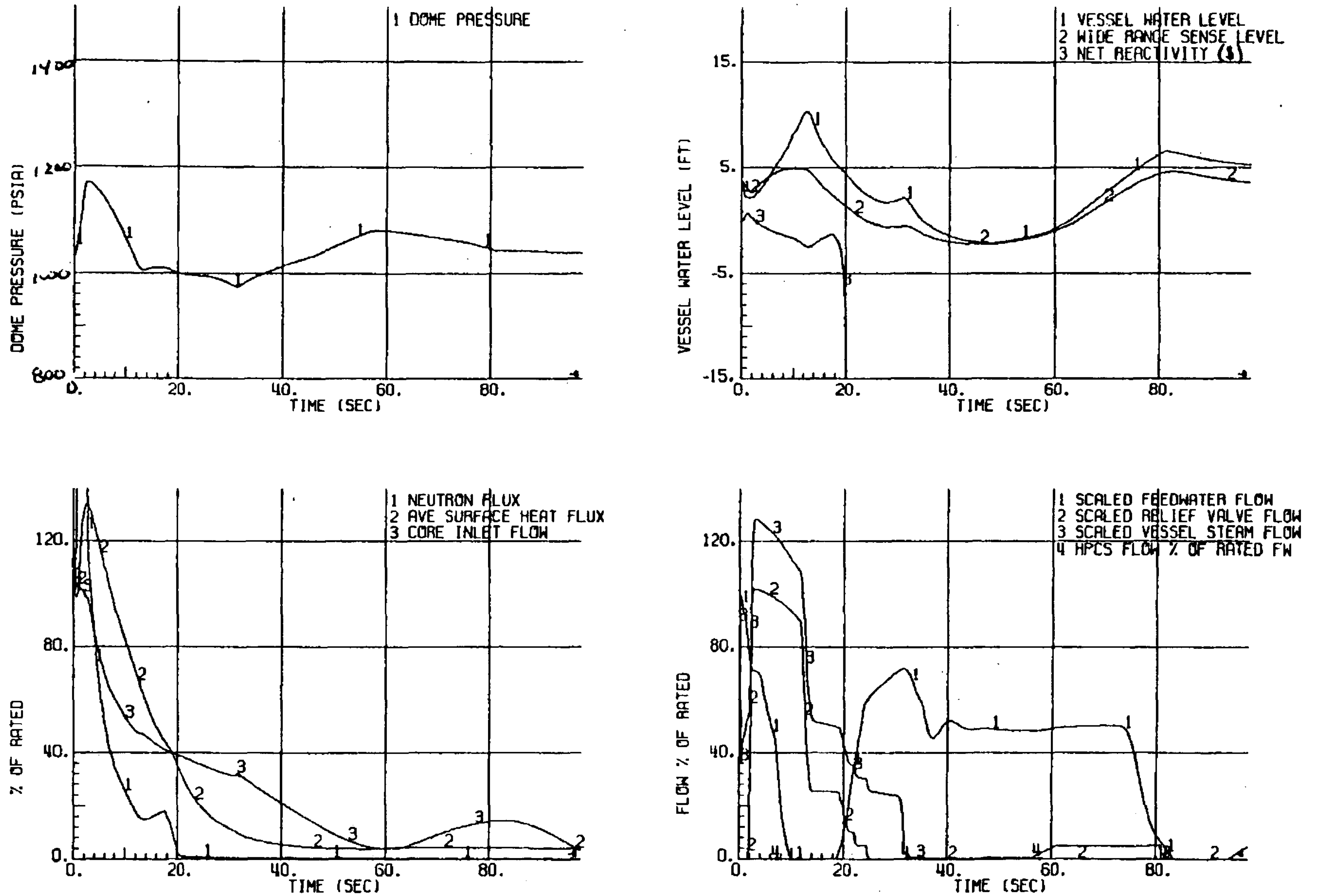


Figure 3.2.5-1. Loss of Condenser Vacuum-W/ARI

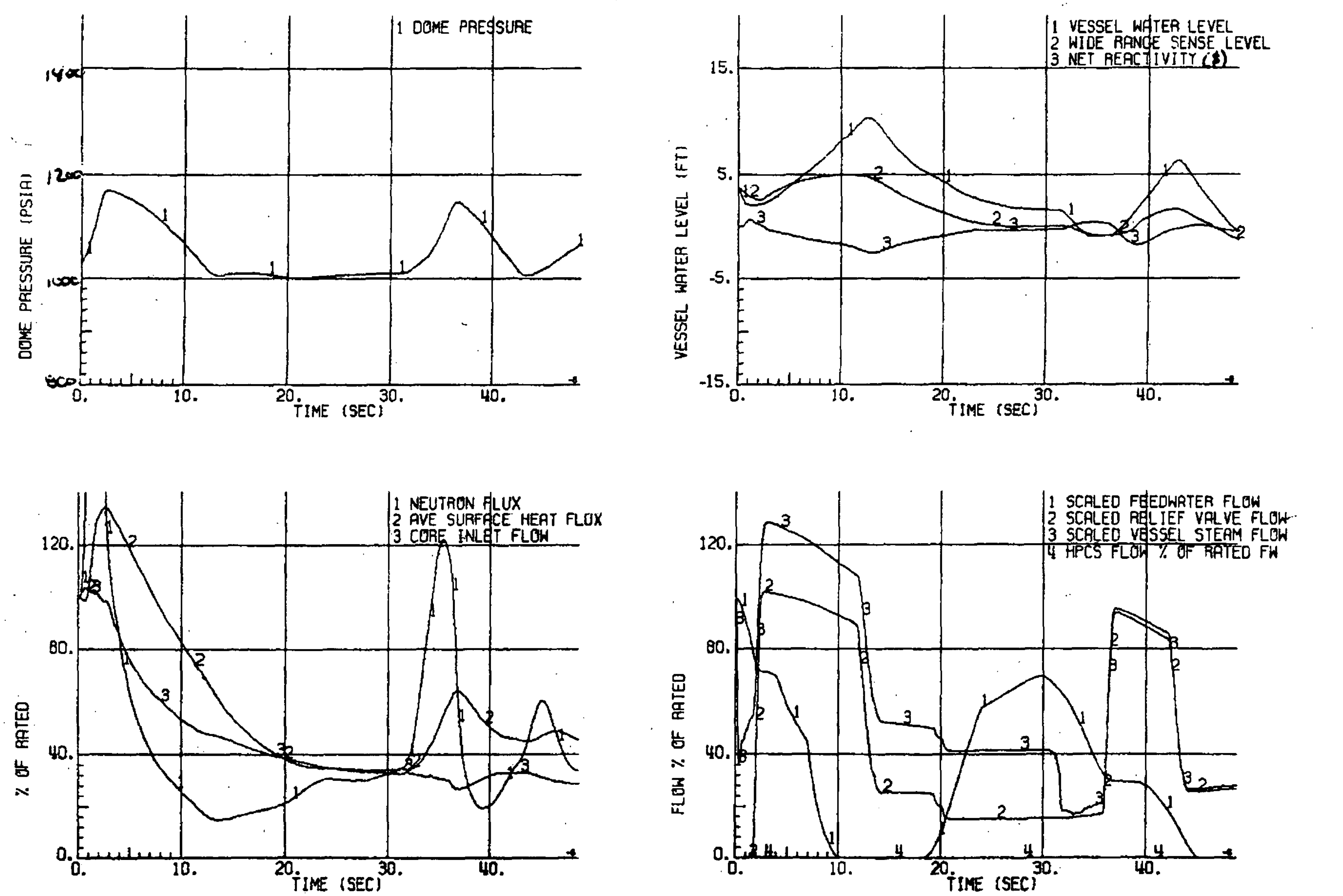


Figure 3.2.5-2. Loss of Condenser Vacuum-No ARI

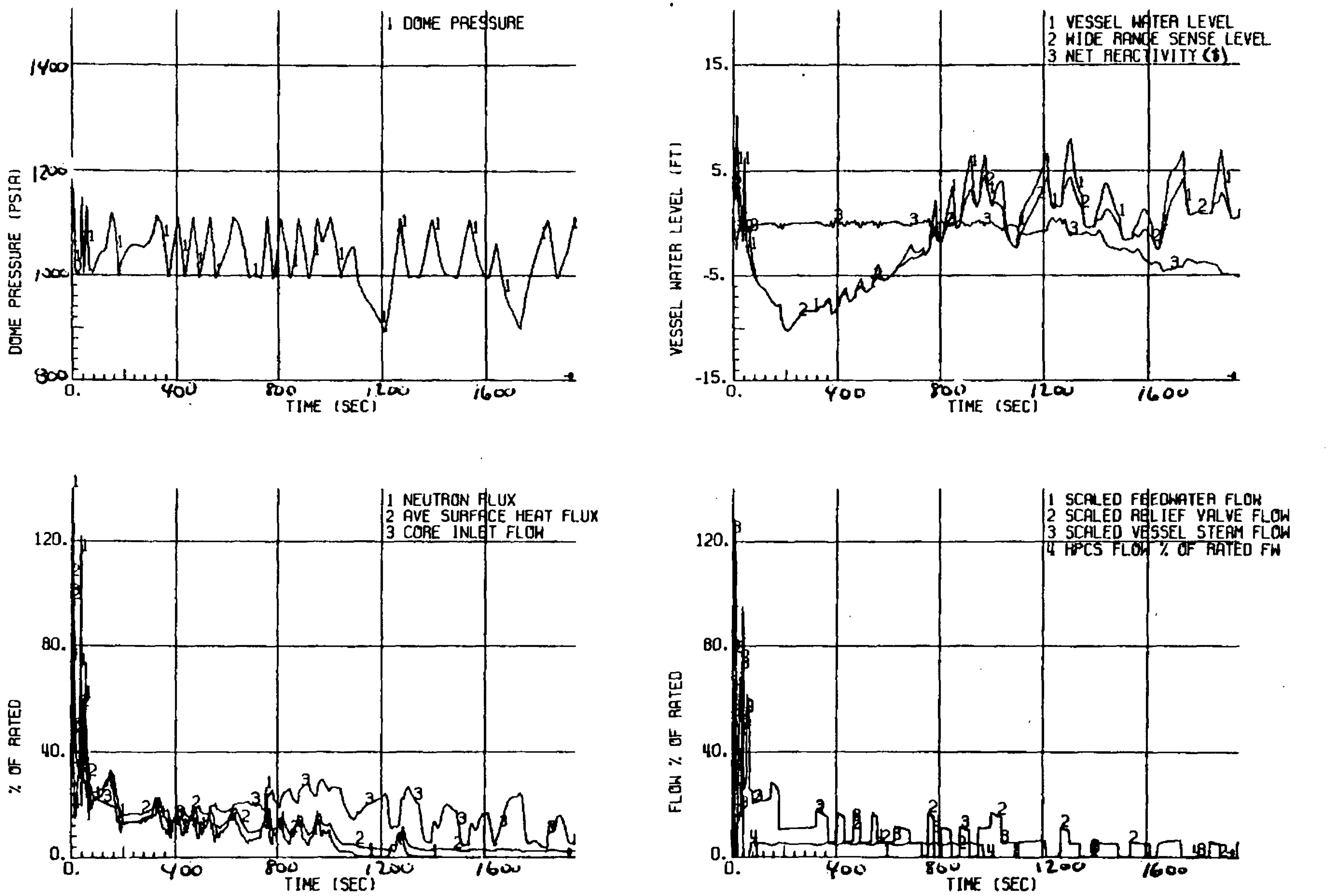


Figure 3.2.5-3. Loss of Condenser Vacuum - No ARI

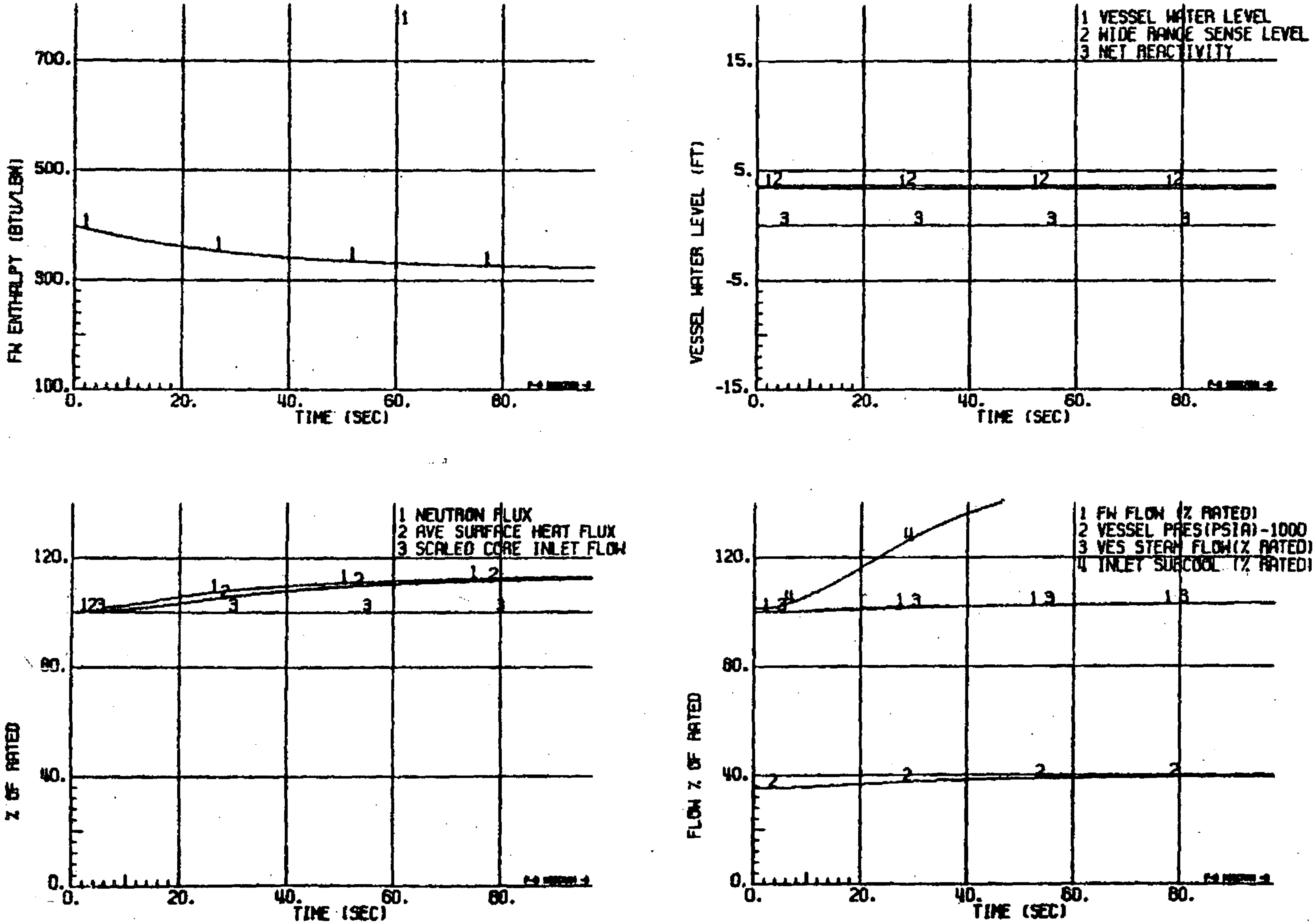


Figure 3.2.7. BWR/5 ATWS Loss of a Feedwater Heater

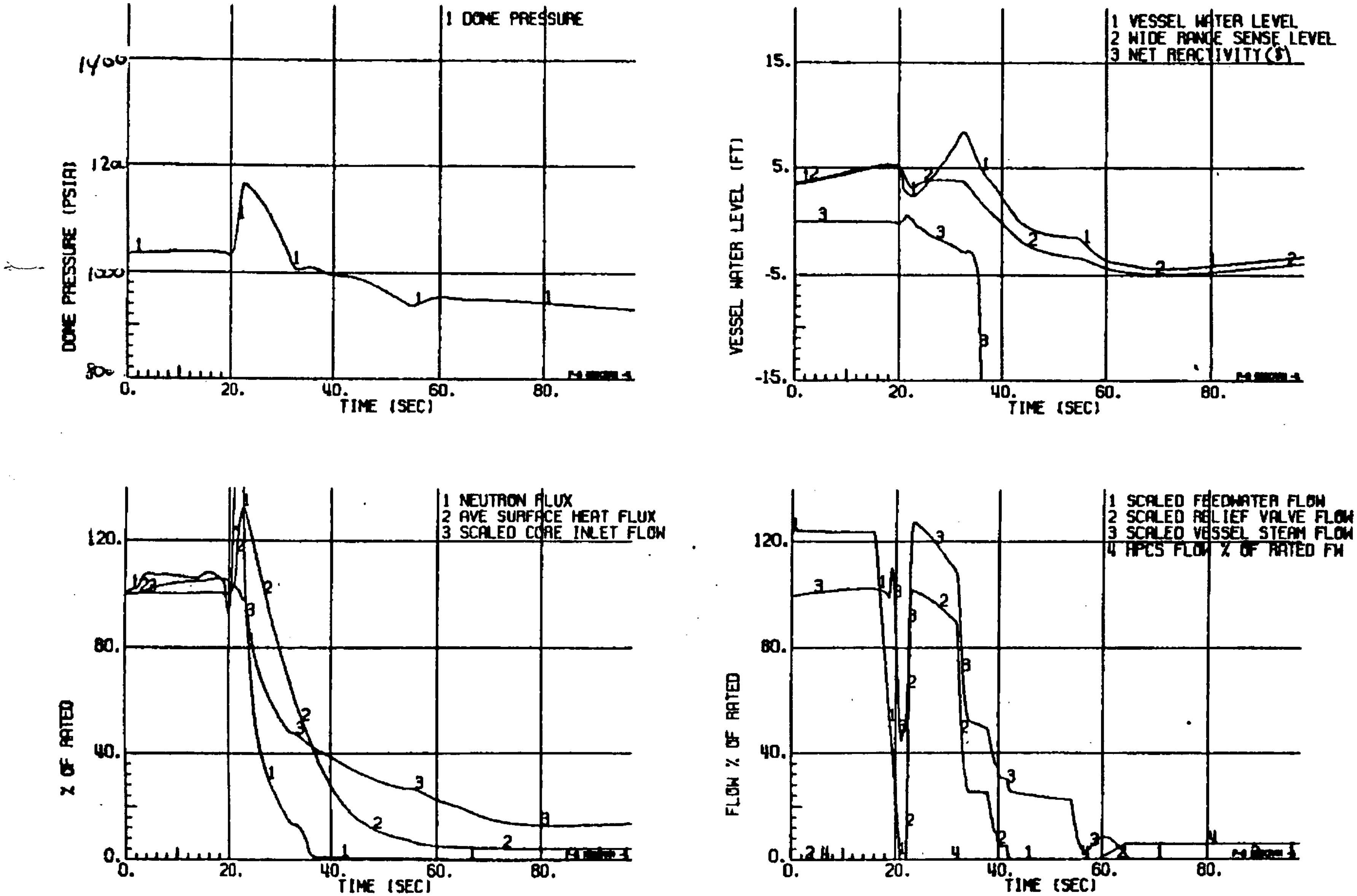


Figure 3.2.8-1. Feedwater Controller Failure, ARI Failure

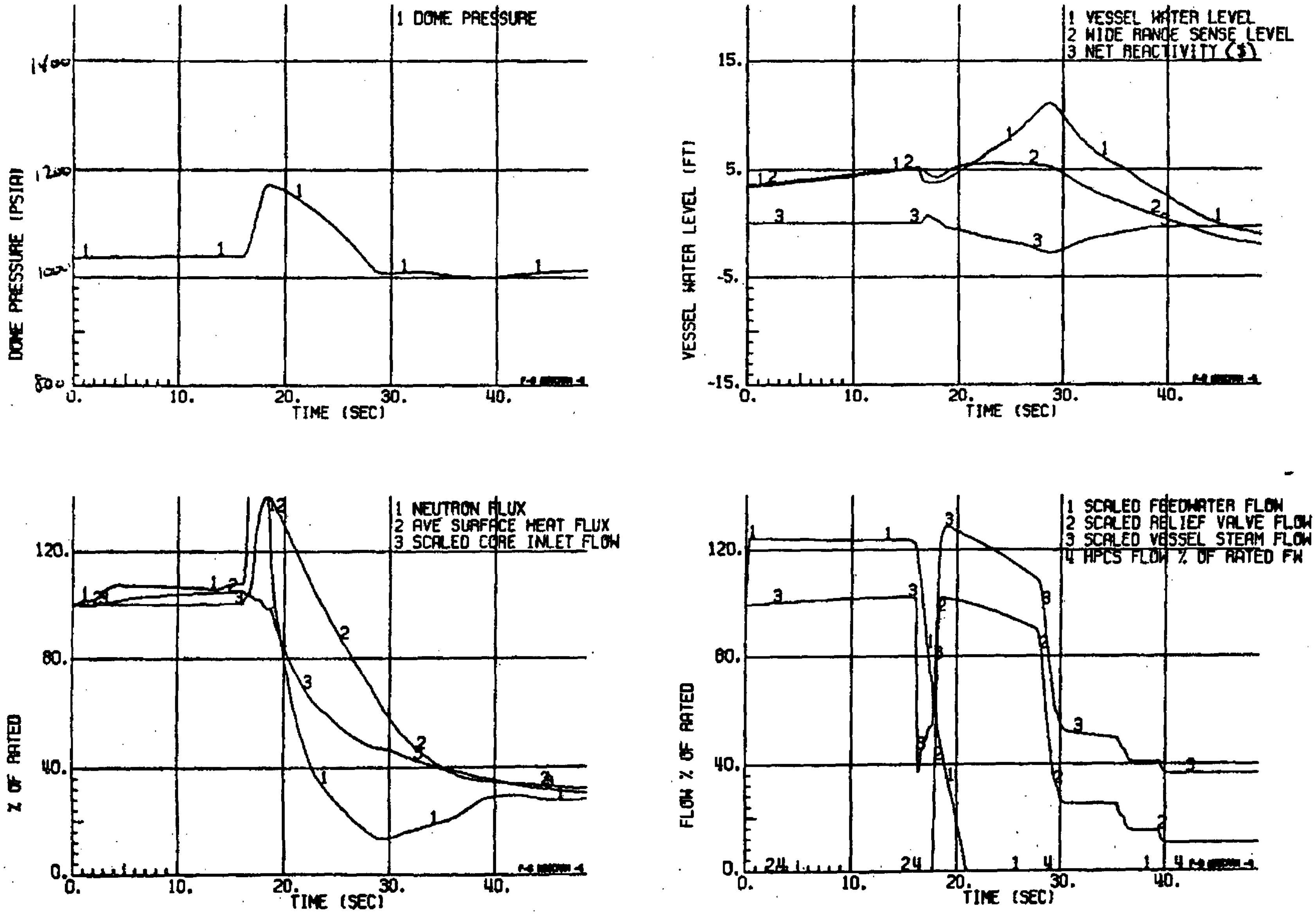


Figure 3.2.8-2. Feedwater Controller Failure, ARI Failure

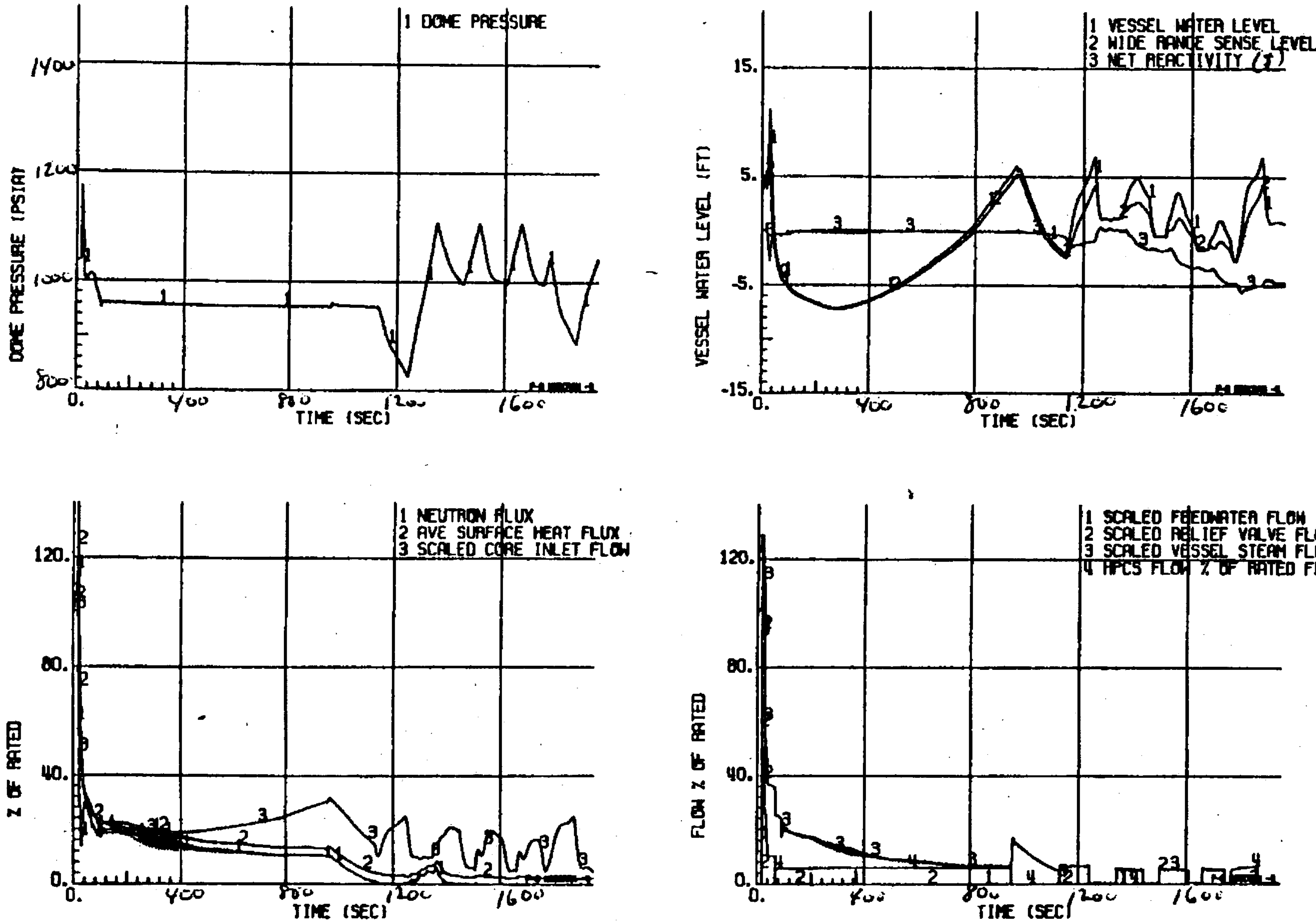


Figure 3.2.8-3. Feedwater Controller Failure, ARI Failure

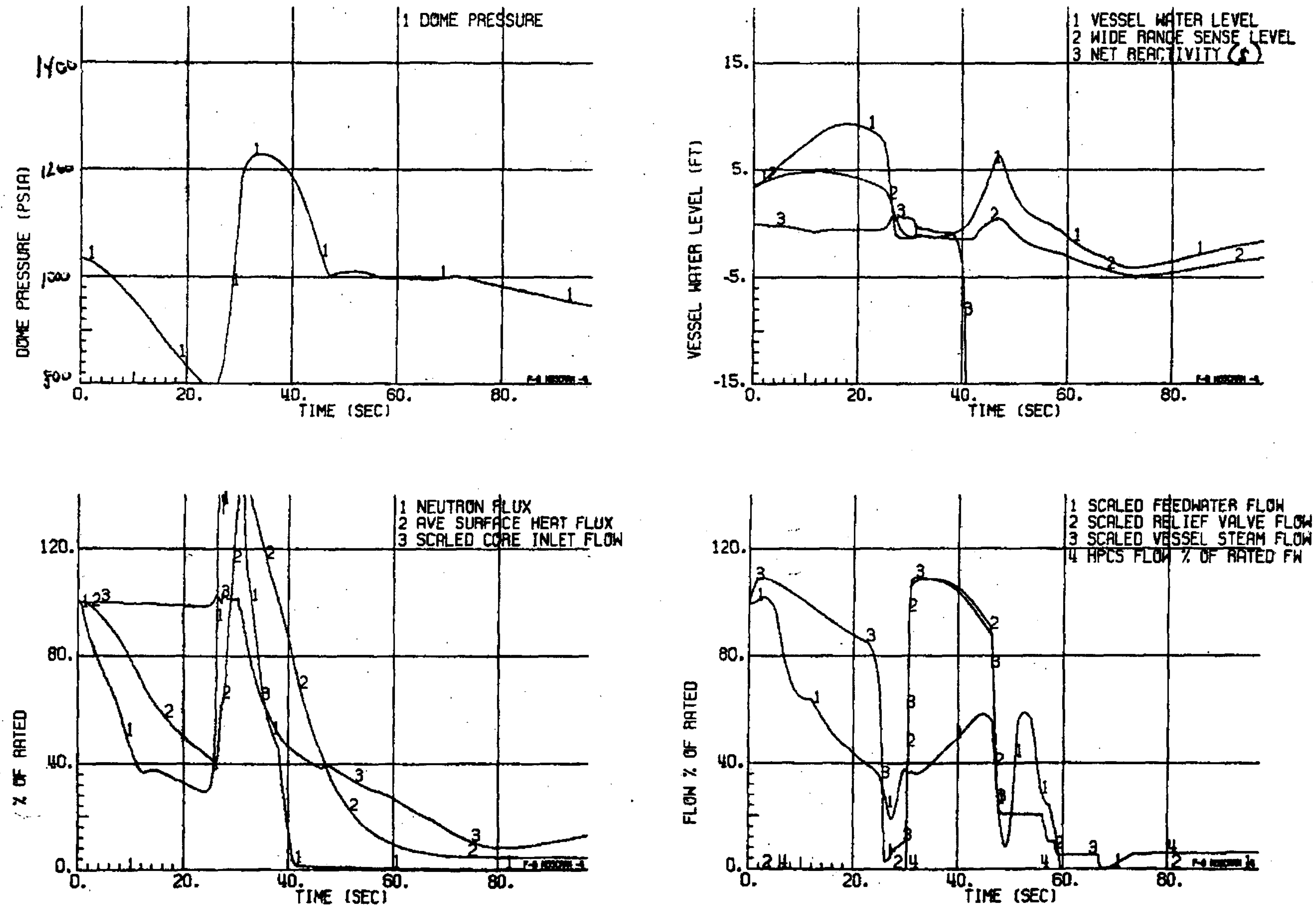


Figure 3.2.9-1. Pressure Regulator Failure, ARI

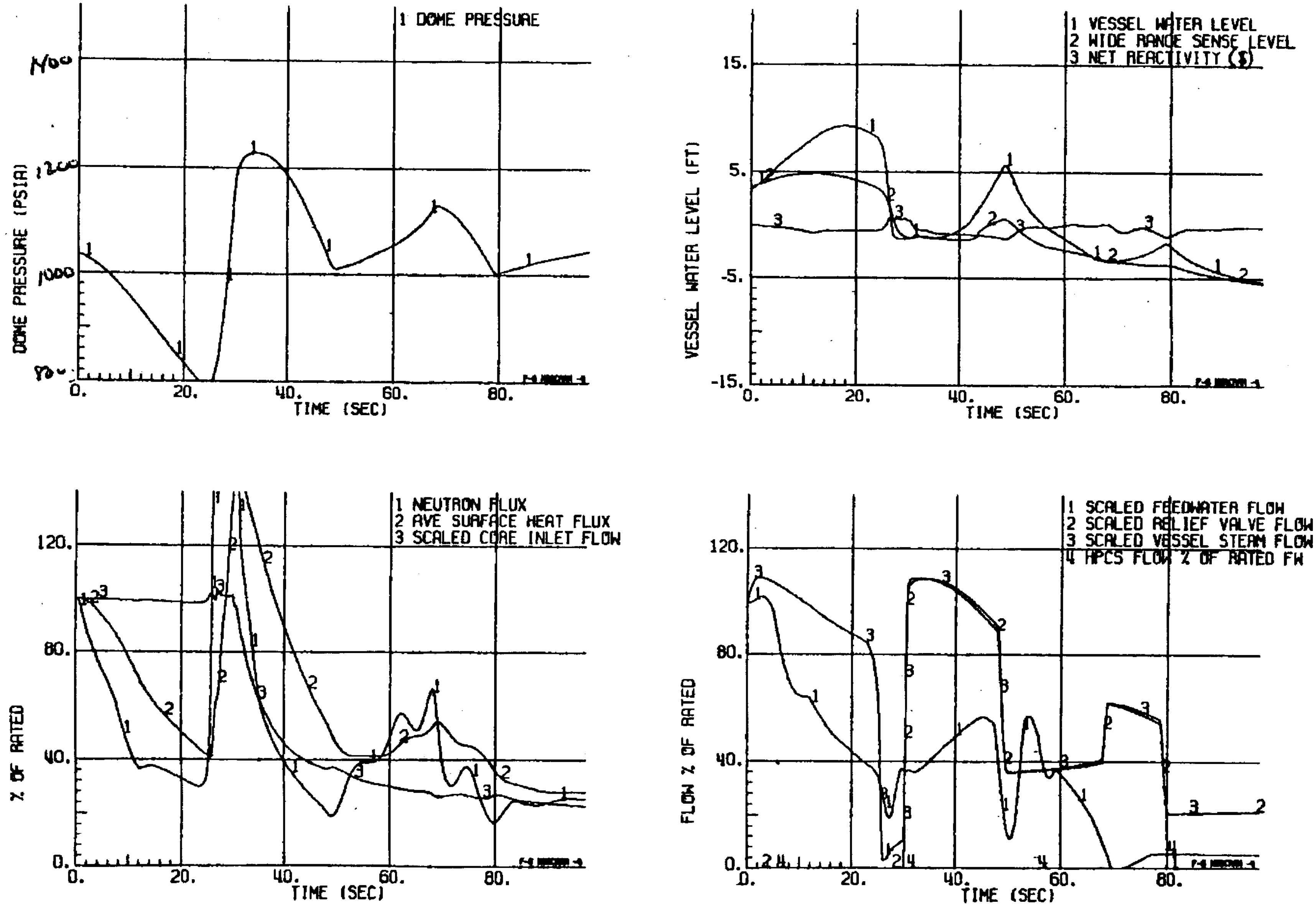


Figure 3.2.9-2. Pressure Regulator Failure, ARI Failure

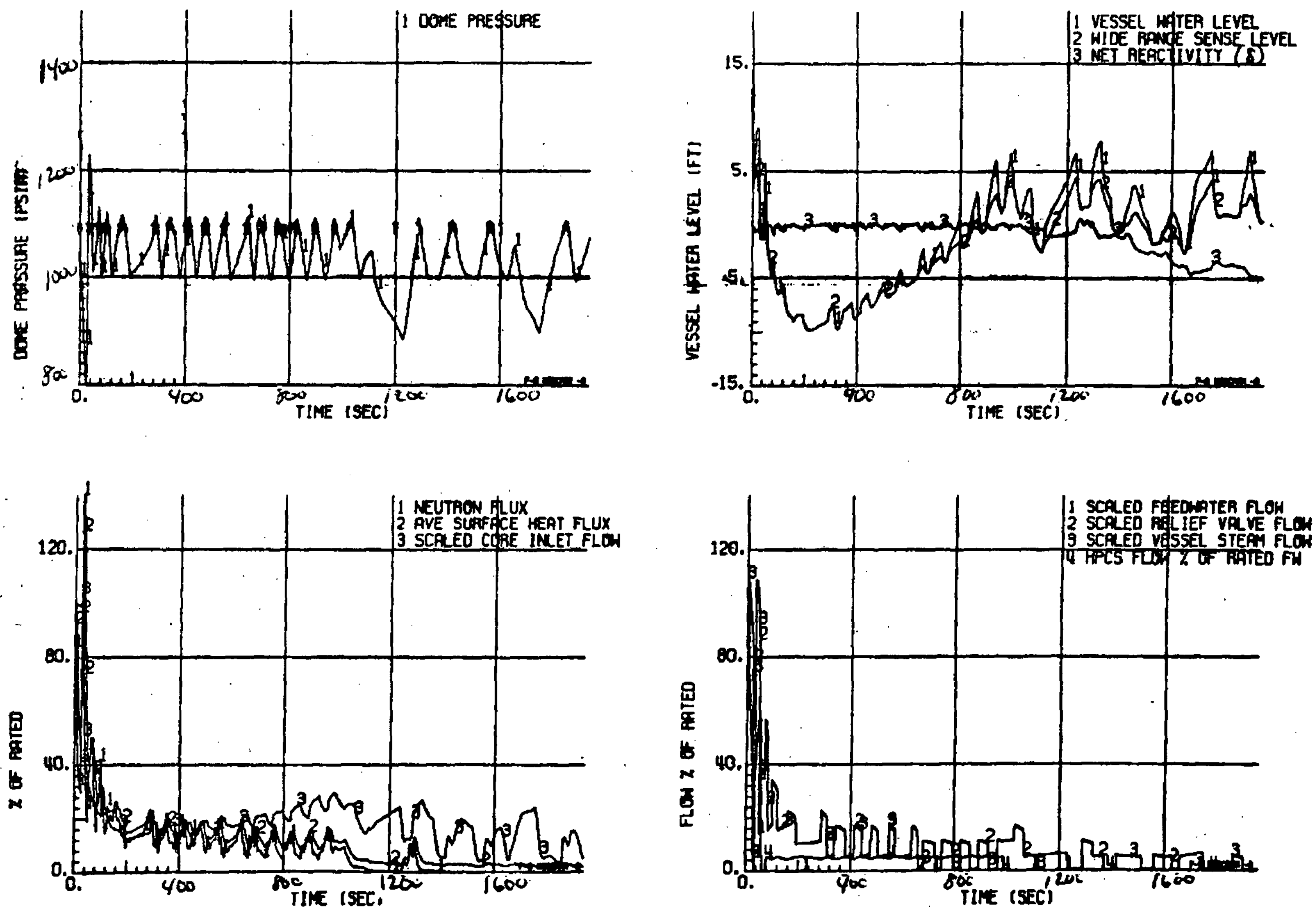


Figure 3.2.9-3. Pressure Regulator Failure, ARI Failure

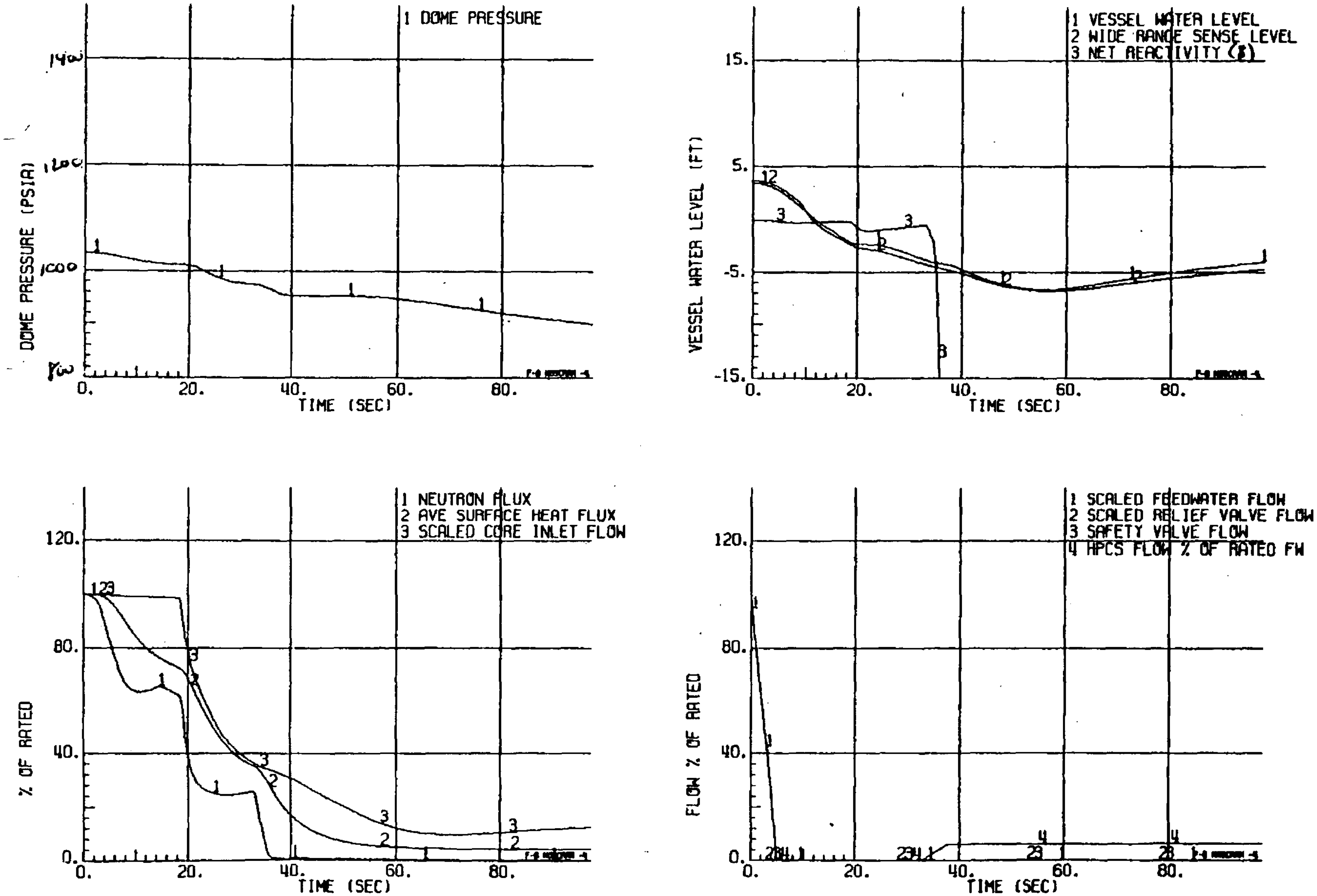


Figure 3.2.10-1. Loss of Feedwater, ARI Failure

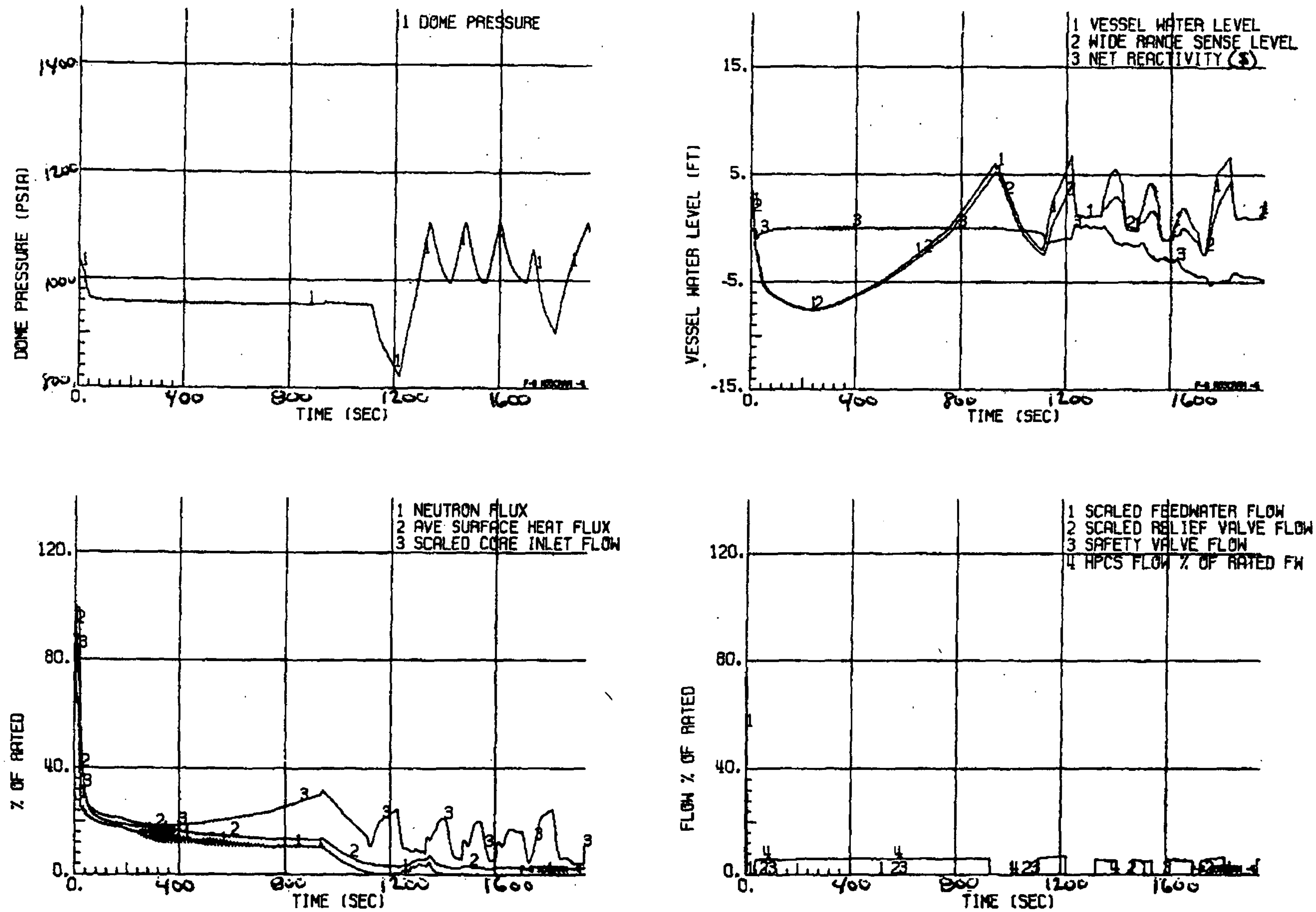


Figure 3.2.10-2. Loss of Feedwater, ARI Failure

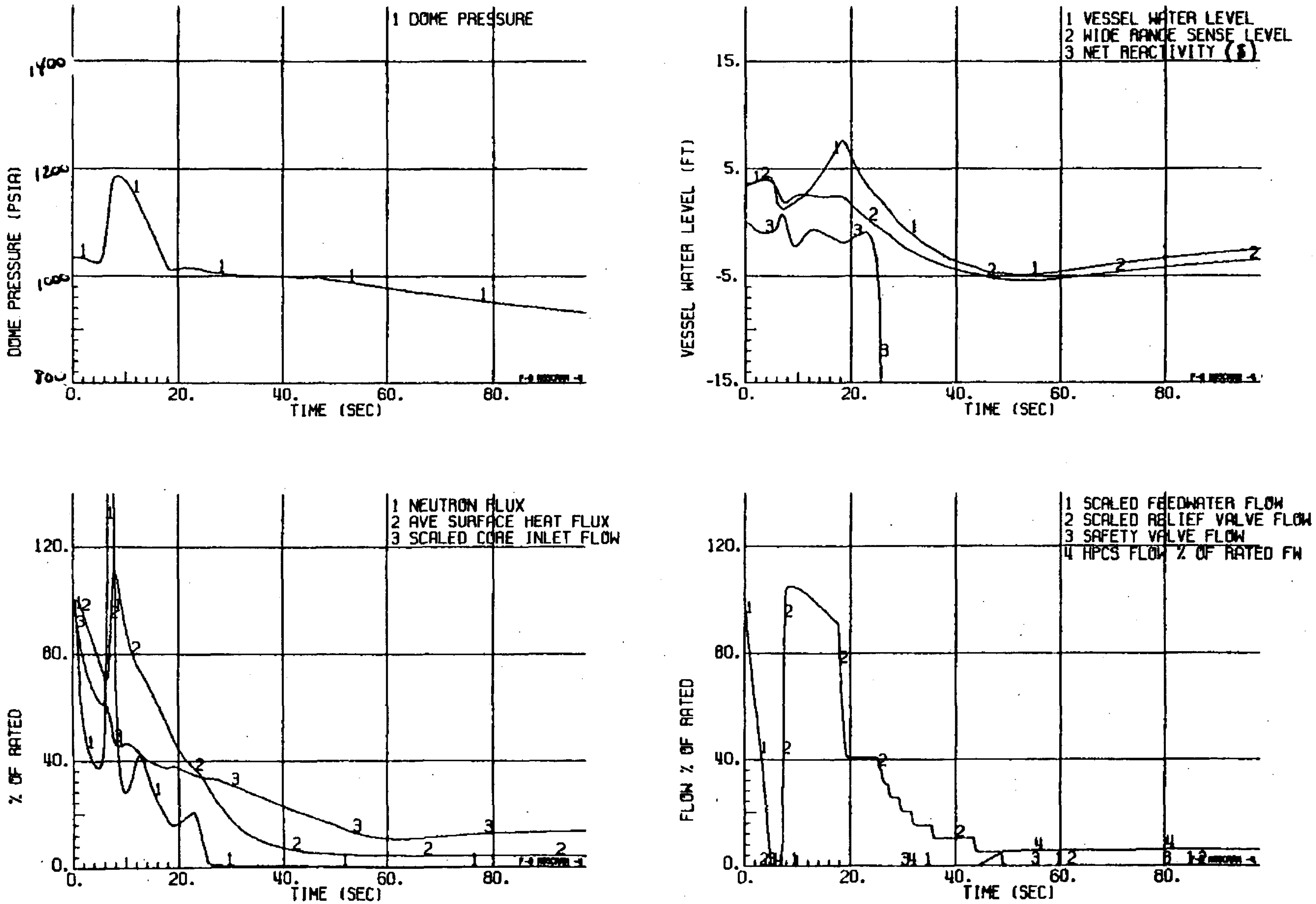


Figure 3.2.11-1. Loss of Normal AC Power, ARI

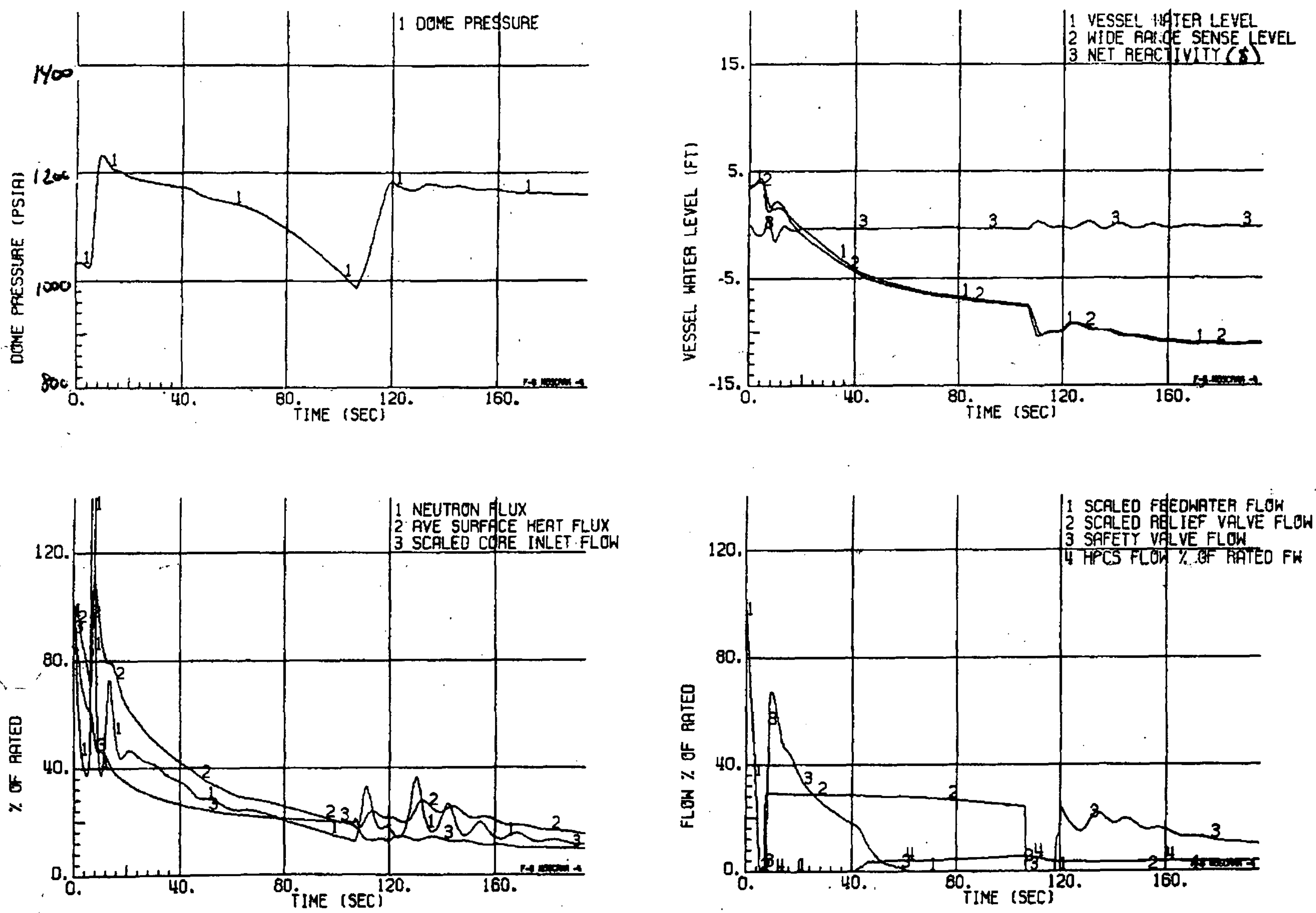


Figure 3.2.11-2. Loss of Normal AC Power, ARI Failure

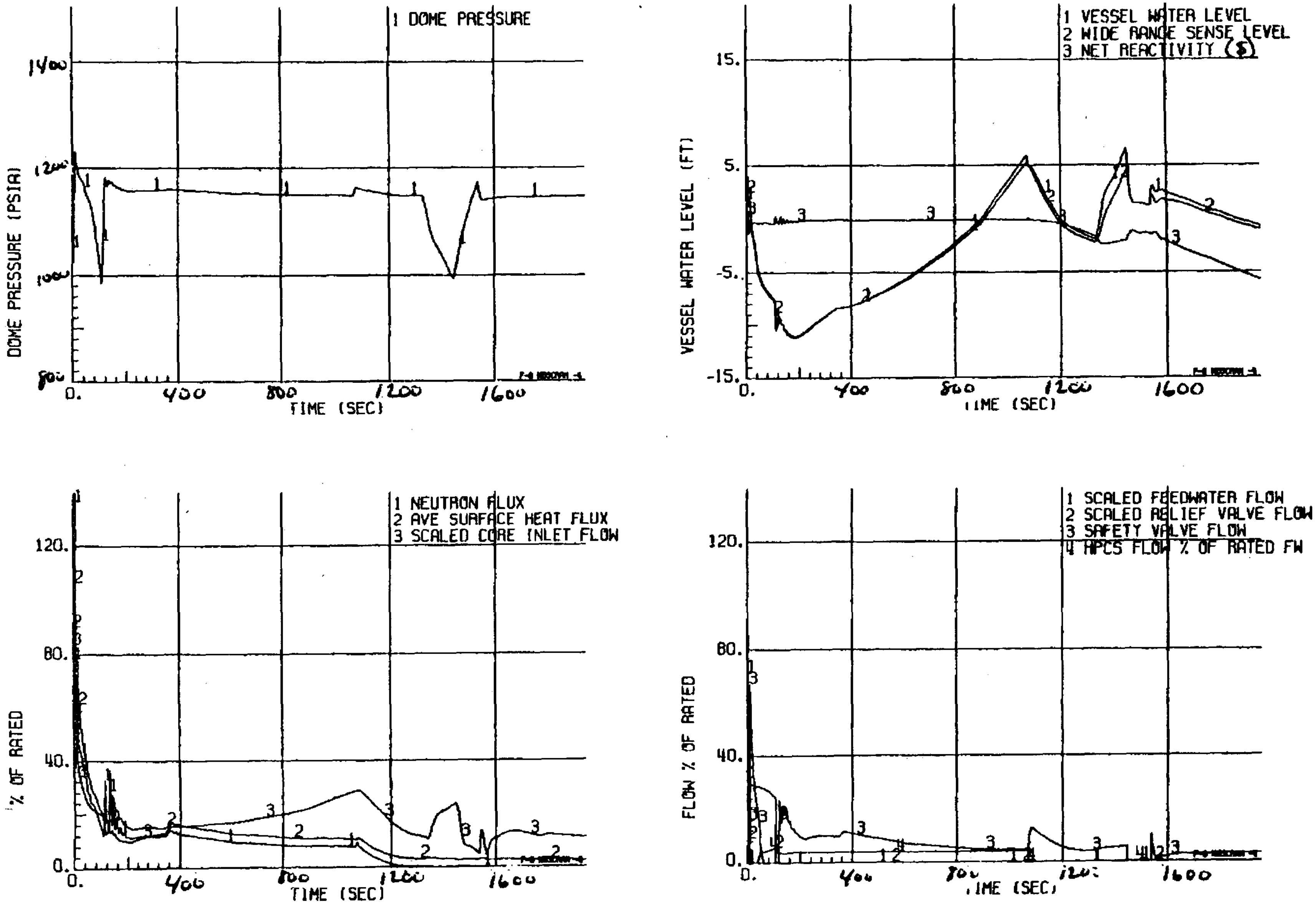


Figure 3.2.11-3. Loss of Normal AC Power, ARI Failure

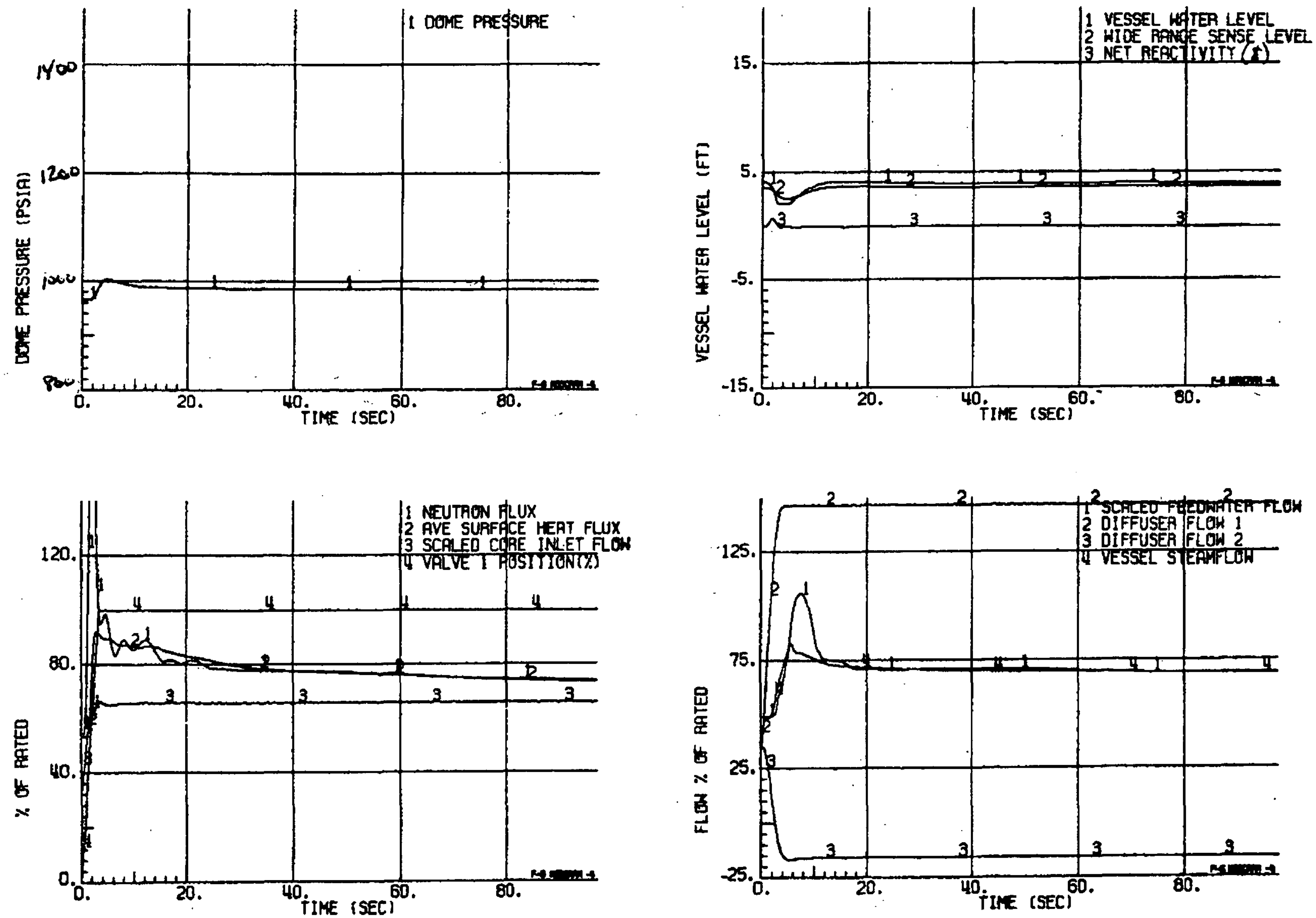


Figure 3.2-12. Recirculation Flow Control Failure, ARI Failure

3.3 RESULTS OF ATWS EVENTS - BWR/6 (MARK III)

3.3.1 MSIV Closure Event

3.3.1.1 Overview of Response Without Scram

A detailed description of all aspects of this event is given below as it has been simulated for this report. The behavior of the plant is separable into an early or short term transient involving a sharp pressure rise and power peak, and a longer term portion that requires evaluation of coolant and containment conditions as the reactor is ultimately brought to shutdown.

A summary of this event and other BWR/6 events is presented in Table 3.3-1.

The effectiveness of RPT presented in NEDO-10349, NEDO-20626 and Volume I are reconfirmed completely by this analysis. It permits the relief valves to limit the pressure disturbance acceptably, reduces the power peak which is created early in the transient, and establishes a relatively low power generation rate for the long-term portion of the transient. Since Volume I, several changes have been made to the base case calculations. They include:

a. Increased Doppler reactivity coefficient is now more typical of all plants (the previous value was an extremely low, unrealistically bounding assumption). All plants were surveyed and nearly identical coefficients are expected. Variations of this term are included in the sensitivity Section 3.3.4.

b. S/RV reclosure pressure is now 110 psi below the opening pressure setpoint which is more typical of actual performance.

c. Feedwater flow characteristics due to automatic limiting action or loss of feedwater turbines is assumed to result in shutoff about 45 seconds after isolation begins.

The ultimate resolution of the lack-of-scram situation must involve insertion of negative reactivity into the reactor thereby terminating the long-term aspects of the event. ARI is provided as an effective way to mitigate for common-cause failure in the logic of the scram system. In the very remote case of its ineffectiveness, automated SLCS provides further protection and shutdown capability. Coolant inventory is adequately maintained by HPCS and RCIC available on each BWR/6 to replace the coolant loss as steam flow leaves the primary system through the relief valves. Simply adding more water is not a totally satisfactory answer because it also has the effect of raising the power generation rate and the amount of inventory leaving the system as steam, thus increasing suppression pool temperature. The steam reaching the suppression pool continues to heat it and to pressurize the containment until the power generation/steam flow can either be reduced to the RHR capacity finally terminated. The RHR (pool cooling mode) ultimately cools the pool and eventually the reactor also (shutdown cooling mode) if the MSIV's cannot be reopened (the preferred method of cooling down).

3.3.1.2 Sequence of Events For MSIV Closure

The MSIV closure transient provides some of the most severe conditions following a postulated failure to scram. Listed below in the sequence of occurrence are significant points from the transient with representative times when each highlight occurs. Results for both cases - with ARI and also assuming its failure - are presented.

In this event, all main steam lines are assumed to isolate starting from rated power conditions with nominal valve closure speed (4 seconds). Figure 3.3.1-1 shows the initial portion of the event for the more likely plant ATWS transient in which ARI quickly shuts down the unit, and Figure 3.3.1-2 shows the case in which ARI also fails and automated SLCS is called upon to shut down the plant.

In each case, the initial power and pressure increases are the same. Neutron flux reaches 745% NBR near 4 seconds, and fuel average surface heat flux reaches 147% at 5 seconds. Some fuel may experience boiling transition, however, coolable geometry is maintained with peak fuel cladding temperature below 1500°F. Peak pressure at vessel bottom is 1299 psig near 7 seconds. The normal reactor scrams occur from position switches on the valves, high neutron flux, and high vessel pressure but are ignored for this analysis. The transient is limited within the Service Level C overpressure limit of 1500 psig through the automatic action of RPT which is initiated when vessel dome pressure exceeds 1150 psig, and by the relieving action of the S/RV's which all open, then start reclosing near 20 seconds. The action of the RPT and the recovering reactor pressure reduces neutron flux until at 90 seconds it is less than 20% NBR. Peak fuel conditions are quickly reduced with the reduction in power and no fuel damage is expected. This is true even if the analysis neglects rewetting of the fuel which should occur after the initial power transient has subsided. By 25 seconds the high pressure logic which began the ATWS protection will have accomplished the ARI function, inserting the rods and shutting off the generated power. This turns the remainder of the event into an essentially normal isolated shutdown as shown in Figure 3.3.1-1. Some relief valve cycling will occur to handle steam generated by decay heat, but peak suppression pool temperature will be only 130°F (at 2 hours, 10 minutes) assuming the RHR loops are initiated in the pool cooling mode after 11 minutes into the event. The water level in the reactor drops to the Level 3 setpoint (another scram point) at about 20 seconds, and to the Level 2 setpoint (about -2 ft on Figure 3.3.1-1) near 48 seconds, starting the RCIC and HPCS systems and initiating containment isolation. HPCS and RCIC replace the main feedwater system, which has assumed to coast down to zero flow (near 45 seconds) due to loss of steam to the turbine driven pumps or due to the action of the ATWS feedwater limiter if the plant had motor-driven pumps. The minimum level for the simulated case is reached near 71 seconds as shown in Figure 3.3.1-3, about 7 ft. above the Level 1 setpoint. HPCS and RCIC then restore level to its normal range and an essentially normal shutdown can be accomplished.

If the ARI function is arbitrarily assumed to fail as well as all other attempts to insert enough control rods within the two-minuted time period, the ATWS logic will continue to sense that the APRM signals are not downscale and not enough rods are in their full-in positions, and the automatic start of boron injection will begin. Instead of shutting off power immediately, the power is predicted to remain in the 10-30% range as shown in Figure 3.3.1-4. The significant features and peak values during the early part of the event are the same as the case with ARI, however, the key differences here is the continuing reduction of water level outside the core shroud until it reaches a minimum between the top of the jet pumps and the Level 1 setpoint at about 3 minutes. Figure 3.3.1-5 shows the level transient with more detail. The steam-water mixture inside the core shroud remains above the core and up into the steam separator standpipes as RCIC and HPCS flow provide inventory. Most of the S/RV's have been reclosed with about 3 valves handling the generated steam, and pressure is cycling near the setpoints.

Boron injection is started by the SLCS pumps at 2 minutes and it reaches the core about 1 minute later. During the following 15 minute period (to about 1100 seconds on these plots), the result is that power is suppressed slightly, reducing the steaming rate and allowing water level to be restored. This also induces higher natural circulation core flow which follows the water level behavior. The level has reached the high level turn-off (Level 8) of the HPCS and RCIC at about 1170 seconds and an off-on-off cycle of these systems is shown as level swings down to Level 2 and back up to Level 8 (between 1200 and 1300 seconds). By this time, manual operator action using the RCIC to modulate level in the normal range would be recommended and expected.

Near 1350 seconds the generated power is dropped toward zero as net reactivity (shown in the upper right plot of Figure 3.3.1-4) becomes negative and continues thereafter to be forced negative by the accumulation of boron in the reactor. At that time the water in the vessel has 345* ppm boron assuming

*Using traditional cold water density to define ppm.

a mixing efficiency of 75% of the injected poison. This accomplishes hot, nuclear shutdown, and the remainder of the steam to the pool is simply due to decay heat.

The bulk average temperature of the suppression pool and pressure in the containment are shown in Figure 3.3.1-6. They rise gradually to peaks of 167°F and 6.9 psig respectively after 20 minutes. Beyond this time the pool cooling capability of the RHR exceeds the steaming rate generated by decay heat and containment conditions are reduced. The peaks are well within the design limits: 185°F and 15 psig, containment integrity is maintained. The boron continues to build the poison concentration in the vessel until it is all injected at about 50 minutes making it possible for a controlled reactor cool-down. The total concentration is specified to be enough to maintain cold, nuclear shutdown conditions even when the RHR system is eventually switched to the reactor shutdown mode, bringing the plant successfully to cold shutdown.

Thus it can be seen that an MSIV closure event combined with a failure to scram is adequately mitigated for a representation BWR 6/Mark III.

3.3.2 Turbine Trip

3.3.2.1 Overview of Response Without Scram

This transient is described in detail in the following sections. Its initial characteristics are much like the MSIV closure described in Section 3.3.1 with a rapid steam shutoff. Pressure and power increases which are limited by the action of the S/RV's and RPT. As this event progresses, however, the availability of the main condenser makes it possible for the relief valves to be open less and close after about 60 seconds. This terminates steam flow to the pool, however, water level is close to the Level 1 isolation setpoint. If isolation occurs, the final portion of the event is similar to the MSIV closure event. This base case event has also been updated from Volume I with the new Doppler coefficient, S/RV reclosure, and feedwater flow conditions discussed in Section 3.3.1.

3.3.2.2 Sequence of Events for Turbine Trip

The listing of significant events during this ATWS event is provided below. Results for both cases - with ARI and also assuming its failure - are presented. This abnormal transient event starts with an unexpected closure of all turbine stop valves (within about 0.1 second). Figure 3.3.2-1 shows the initial portion of the event for the more likely plant ATWS transient in which ARI provides a diverse logic path to quickly shut down the reactor, and Figure 3.3.2-2 shows the case in which ARI also is assumed to fail and the automated SLCS is called upon to shut down the reactor.

In each case, the initial power and pressure increases are the same. Neutron flux reaches 358% NBR near one second and fuel average heat flux reaches 135% NBR at about 3 seconds. Some fuel may experience boiling transition, and coolable geometry is maintained since peak fuel cladding temperature is less than 1500°F. The peak pressure occurs at the vessel bottom and is 1225 psig near 3 seconds. The normal reactor scram signals occur from position switches on the valves, high neutron flux, and high vessel pressure but are ignored for this analysis. The transient pressure is limited within the Service Level C overpressure limit of 1500 psig through the automatic action of RPT which is initiated when vessel dome pressure exceeds 1150 psig and the relieving action of the S/RV's which all open, then start reclosing near 8 seconds. The plots show both the steam flow and the relief valve flow - the difference is the flow through the bypass valves to the main condenser (a typical BWR/6 bypass capacity of 35% NBR is assumed).

By about 22 seconds, the high pressure logic which began the ATWS protection will have accomplished the ARI function, inserting the rods and shutting off the generated power. This deactivates the automatic boron injection and feedwater limit, and makes the remainder of the event like a normal turbine-generator trip shutdown. No additional relief valve flow will occur as the bypass/pressure control system will handle steam generated by decay heat. Peak suppression pool temperature will occur at the time of the last relief actuation and will be only 96°F. The RHR can be activated in pool cooling whenever convenient to control the suppression pool temperature. Reactor water level remains in the normal range throughout the event by the feedwater system and no RCIC or HPCS initiation is expected.

If the ARI is also assumed to be failed, the BWR/6 is still able to mitigate the event. With an assumed ARI failure and feedwater flow now having been limited to zero, at 56 seconds the level of the bulkwater in the vessel will decrease past level 2, the level at which HPCS and RCIC are initiated. Eighteen seconds later, water from these systems is assumed to begin to enter the reactor vessel.

Following confirmation from the flux monitoring system and the rod position indicating system that scram has really not taken place, the SLCS is activated. This system will be started 2 minutes after the ATWS signal; one minute of boron transport time is also accounted for in the lines and the vessel. Therefore, nuclear shutdown begins at 3 minutes into the event using the SLCS. With an 86 GPM volumetric flow rate of sodium pentaborate, the reactor will be brought to hot shutdown in approximately 19 minutes from the beginning of the event. The behavior of several parameters is depicted in Figure 3.3.2-3 for the long-term event.

Bulkwater level within the vessel continues to decrease until approximately 2 minutes at which time HPCS and RCIC supply more water than is required to make up for steam flow out of the vessel. At this time it reaches its lowest level and begins to rise. At all times, the top of the vessel remains covered with two-phase mixture inside the core shroud. As the level is regained, core flow is increased, thereby reducing the average void fraction. The various contributors to reactivity insertion and power production (boron, voids, etc.) must always be in balance with the power production. Water level is completely restored by HPCS and RCIC at approximately 16 minutes. Water level is maintained by automatically cycling HPCS and RCIC. A larger-scale plot of water level is shown on Figure 3.3.2-4.

Following hot shutdown, the decay power continues to generate a small amount of steam which will go through the bypass to the main condenser. Since the major portion of the steam generated in this event goes to the main condenser, the temperature rise in the suppression pool is minimal. The maximum suppression pool temperature calculated in this case is 100°F, which results in a maximum pressure of 0.5 psig.

These values remain well within the containment design requirements of 185°F and 15 psig. Figures 3.3.2-5 and 3.3.2-6 show detailed plots of the water level outside the core shroud, the pool temperature, and the containment pressure through the peak portion of the event. Water level inside the core shroud is a two-phase mixture which remains above the core and up into the steam separator standpipes as RCIC and HPCS flow provide coolant inventory. The boron will continue to build the poison concentration in the vessel until it is all injected (near 50 minutes) making it possible for a controlled reactor cooldown. The total concentration is specified to be enough to maintain cold nuclear shutdown conditions even when the RHR system is eventually switched to the shutdown cooling mode, bringing the plant successfully to cold shutdown.

Thus it can be seen that a Turbine Trip with bypass event combined with a failure to scram is adequately mitigated for a representative BWR6/Mark III.

3.3.3 Inadvertent Open Relief Valve

3.3.3.1 Overview of Response Without Scram

This event has no rapid excursions as the previous two events but is merely a long term depressurization. RPT does not occur until late in the event after hot shutdown is achieved.

Except for steamflow through the open relief valve and the use of the liquid boron solution for shutdown, the nuclear steam supply system is in a normal operating state. The suppression pool is the only system exposed to off-normal conditions. This base case event has also been updated from Volume I with the new Doppler, S/RV reclosure, and feedwater flow conditions discussed in Section 3.3.1. The sequence of events follow.

3.3.3.2 Sequence of Events For Inadvertent Open Relief Valve

This event begins when one of the primary relief valves on the main steamlines inadvertently opens without influence from any other portion of the system. All pressure levels in the reactor coolant pressure boundary are at a nominal value prior to the event. The resulting sequence of events is shown in Table 3.3.3-1.

At the time that the relief valve opens, there is a momentary depressurization (a few seconds) until the turbine control valve senses it and closes slightly (dropping unit electrical output) to control the pressure. For general application of this analysis, a relief valve capacity of 7.1% NB rated was utilized (the nominal flow of a valve on a BWR/6-218 inch vessel plant). After about two minutes, the suppression pool temperature, which was initially assumed to be at 90°F, has risen to the alarm point of 95°F. If attempts to reclose the valve are unsuccessful, the operator will turn on the RHR system in the pool cooling mode to maintain pool temperature. If attempts to close the valve continue to be unsuccessful, the temperature will continue to rise and at 9 minutes will reach 110°F at which point the operator is required to manually scram the plant. Should scram fail to occur at this point, the manual scram will initiate ARI as well as start the SLCS timed logic.

If for some reason neither normal manual scram or the ARI are effective, the BWR/6 is still able to mitigate the event at this time. The ATWS logic would have determined that ARI was unsuccessful and the control rods are still not inserted, and at 11 minutes into the event will activate the SLCS. For this case, with the recirculation pumps operating, the boron mixing efficiency is excellent (95% is assumed) and the delay time inside of the vessel is small so that at 12 minutes the control liquid reaches the core and shutdown begins. Within 20 minutes, the power has been reduced to the point that the amount of steam generated is less than the relief valve capacity and the pressure now begins to decrease more rapidly. The turbine control valves have closed completely. These events are depicted in Figure 3.3.3-1. By 23 minutes, the pressure has dropped to the low steam line pressure isolation point of 800 psig and the MSIV's close. Simulating plants with turbine-driven feedwater pumps, the feedwater was assumed to be lost within 45 seconds of the isolation. This causes the water level in the vessel to decrease and at 24 minutes the low level point (L2) was reached where the recirculation pumps were automatically tripped and HPCS and RCIC were activated. These systems are shown to automatically cycle on at low level (L2) and off at high level (L8) as specified to maintain water inventory in the vessel, although manual action is expected to maintain level with the RCIC alone. The depressurization of the vessel will continue with the relief valve discharging into the suppression pool; the maximum pool temperature of 170°F will occur at 50 minutes. The peak containment pressure of 7.3 psig occurs at the same time (see Figure 3.3.3-3). Both values are well below the criteria values of 185°F and 15 psig.

In cases where ARI is activated (11 minutes), the maximum pool temperature is 155°F.

Thus is can be seen that the inadvertent opening of a relief valve event combined with a failure to scram is adequately mitigated for a representative BWR6/Mark III.

3.3.4 Sensitivity Study Results - BWR/6 Base Cases

A wide variety of parameters were studied to examine the sensitivity and potential impact of plant differences and/or uncertainties on the results of the three BWR/6 base cases. The results are documented in the following subsections:

While the overall objective of these sensitivity studies is to provide guidance for assessing the adequacy of plants having certain parameters different from the generic analyses, caution must be exercised when combining the results of several parameter variations, due to the non-linearities involved (see Section 3.3.4.4).

3.3.4.1 MSIV Sensitivity Studies

3.3.4.1.1 Variation of Boron Delay

3.3.4.1.2 Variation of Boron Capacity/Mixing

3.3.4.1.3 Variation of HPCS/RCIC Capacity

3.3.4.1.4 Variation of RHR Capacity

3.3.4.1.5 Variation of RHR Delay

3.3.4.1.6 Variation of Pool and Service Water Temperature

3.3.4.1.7 Variation of RHR Capacity and Service Water

3.3.4.1.8 Variation of Pool Size

3.3.4.1.9 Variation of S/RV Capacity

3.3.4.1.10 Variation of RPT Delay

3.3.4.1.11 Variation of RPT Inertia

3.3.4.1.12 Effect of Partial Rod Insertion

3.3.4.1.13 Variation of Void Coefficient

3.3.4.1.14 Variation of Doppler Coefficient

Table 3.3.4.1-1 and 2 summarizes the results of this event.

3.3.4.2 Turbine Trip Sensitivity Studies

3.3.4.2.1 Variation of Boron Delay

3.3.4.2.2 Variation of Boron Capacity/Mixing

3.3.4.2.3 Variation of HPCS/RCIC Capacity

3.3.4.2.4 Variation of RHR Delay

3.3.4.2.5 Variation of Void Coefficient

3.3.4.2.6 Variation of Doppler Coefficient

Table 3.3.4.2-1 and -2 summaries the results for this event.

3.3.4.3 IORV Sensitivity Studies

3.3.4.3.1 Variation of Boron Delay

3.3.4.3.2 Variation of Boron Capacity/Mixing

3.3.4.3.3 Variation of RHR Capacity

3.3.4.3.4 Variation of RHR Delay

3.3.4.3.5 Variation of Pool and Service Water Temperature

3.3.4.3.6 Variation of RHR Capacity and Service Water Temperature

3.3.4.3.7 Variation of Pool Size

3.3.4.3.8 Variation of S/RV Capacity

Table 3.3.4.3-1 summaries the results of this event

3.3.4.4 BWR/6 Multiple Sensitivity Results

3.3.4.1 MSIV Sensitivity Studies

3.3.4.1.1 Variation of Boron Delay

The SLCS timer delay was varied between 30 seconds (75% below the nominal timer setting of 120 seconds) and 240 seconds (100% above the nominal time) resulting in peak pool temperature 3°F less and 6°F greater, respectively compared to the base case. Containment pressures decreased and increased accordingly by 1 psi. Figures 3.3.4.1.1-1 and 3.3.4.1.1-2 graphically show this parameter variation.* Minimum level was increased by 0.7 feet at 30 seconds delay and was unchanged when the timer was extended to 240 seconds.

3.3.4.1.2 Variation of Boron Capacity/Mixing

The effective rate of boron injection into the core is the product of the boron pumping capacity and mixing efficiency. This effective rate was varied by ±50% resulting in peak pool temperatures 13°F below and 40°F above the base case, respectively. The base case represents an 86 gpm SLCS pumping rate in a 251 inch vessel with 75% assumed mixing efficiency. The -50% variation point equivalently represents 43 gpm at 75% efficiency or 86 gpm at 38% efficiency. Differences for plant size are covered by comparing the boron rate to the rated steam flow of the plant (e.g., the 86 gpm on a 251 size plant is equivalent to 66 gpm on a 218 size plant). Figure 3.3.4.1.2 graphically shows this variation.

3.3.4.1.3 Variation in HPCS/RCIC Capacity

The rated flow of the HPCS/RCIC system was varied by 20%. Figures 3.3.4.1.3-1 and 3.4.1.3-2 graphically show the change for the case of increased flow, pool temperature and containment pressure increased by 7°F and 1 psi, respectively. The increase in temperature is due to the higher power level maintained by the

*Note that points representing calculated results of sensitivity studies are connected by lines to add clarity to general trends. This does not imply detailed knowledge of the variation between points.

increased core flow. Minimum water level increased more than a foot. Decreasing HPCS/RCIC flow lowered peak temperature and pressure by 9°F and 2 psi. Minimum level was reduced by approximately 1.8 feet.

3.3.4.1.4 Variation in RHR Capacity

To determine the effect of varying RHR heat exchanger capacity, the base capacity of 3.2% NBR at 100°F ΔT (1272 BTU/sec-°F for the 251 size plant used as the base case) was altered by 50%. Increasing the capacity by 50% yielded a 3°F temperature reduction and lowered the peak containment pressure by less than 1 psi. For the opposite case of a 50% decrease in RHR capacity the results were a 7°F increase in temperature and a 1.1 psi pressure rise. Sensitivity of pool temperature is shown graphically in Figure 3.3.4.1.4.

3.3.4.1.5 Variation in RHR Delay

The effect of varying RHR start time was found to be small for the BWR/6 MSIV case. Increasing the start time from 11 (base) to 16 minutes increased peak pool temperature by less than 3°F. A decrease of 2 minutes resulted in less than 1°F reduction in pool temperature. This very weak sensitivity of the pool temperature to RHR startup delay is shown graphically in Figure 3.3.4.1.5.

3.3.4.1.6 Variation in Pool and Service Water Temperature

The pool and service water temperature were assumed to vary together (with the pool assumed to be 5°F above the service water). This variation was found to significantly affect peak pool temperature and containment pressure. Increasing these temperatures by 20°F (to the operating technical specification) produced a rise in pool temperature of 18°F and an increase of about 3 psi in peak pressure. Reducing the temperatures by 20°F yielded decreases of 19°F and about 3 psi respectively. Figure 3.3.4.1-6 graphically shows the pool temperature variation plotted directly vs ΔT change.

3.3.4.1.7 Variation of RHR Capacity and Pool and Service Water Temperature

Varying both parameters simultaneously was done to examine different RHR design due to different plant site water temperatures. It showed that pool and service water temperature was the dominant variable (see Figure 3.3.4.1-6). Simultaneous increases of +50% in RHR capacity and +20°F in pool and service water temperature (and a similar set of decreases) produced temperature changes of +16 and -12°F. Peak pressures varied accordingly by +2 and -2 psi.

3.3.4.1.8 Variations in Pool Size

The suppression pool mass was varied by ±20% to simulate different sized plants. The larger pool mass provides a bigger heat sink, thus reducing the peak pool temperature by nearly 11°F and peak pressure by about 2 psi. For the lower pool mass, pool temperature increases 16°F and peak pressure by 3 psi. Figure 3.3.4.1-8 graphically shows the result.

3.3.4.1.9 Variations in S/RV Capacity

The total S/R valve capacity was varied by ±20%. The larger and smaller valve capacity resulted in a maximum pressure (level bottom) of 33 psi less than and 148 psi higher than the base case respectively. Figure 3.3.4.1.9-1 shows these results graphically. Pool temperature variation was very small (<2°F).

3.3.4.1.10 Variations in RPT Delay

The base value of RPT delay (0.43 sec) was varied by 0.5 second and 1.0 second. Due to the large S/RV capacity for BWR/6 the maximum neutron and maximum average fuel heat flux remained unchanged. The maximum pressure (vessel bottom) increased by less than 7 psi as shown by Table 3.3.1.4.1-1 and Figure 3.3.

3.3.4.1.11 Variation of RPT Inertia

The RPT Inertia was increased by 50% and reduced by 20%. In both cases noticeable changes were observed in the values of maximum neutron flux, maximum average fuel heat flux and maximum vessel pressure as indicated by Table 3.3.4.1-1.

3.3.4.1.12 Effect of Partial Rod Insertion

All ATWS analysis are performed assuming that no rod motion takes place. It is very likely that some rod insertion would occur. To explore the effects of partial rod insertion, an analysis was performed assuming that rods equivalent to $2 worth of reactivity actually insert. As expected, this significantly reduced peak values for vessel pressure, neutron flux, etc. The peak pool temperature reduced by about 16°F with a corresponding reduction of 2 psi in containment pressure.

3.3.4.1.13 Variation of Void Coefficient

The effect of void coefficient on peak transient parameters (neutron flux, average surface heat flux, vessel pressure and suppression pool temperature) was studied for the MSIV closure transient. Void coefficient was varied from -6 to -14¢/% rated voids (nominal = -11¢/%). In all cases the recirculation pumps were tripped on high vessel pressure. The change in total effective worth of injected boron with void fraction was accounted for. Figure 3.3.4.1.13-1 shows the flux and pressure peaks for the MSIV transients, as a function of void coefficient for several values of Doppler coefficient. The peaks are shown relative to those at nominal nuclear coefficients. Figure 3.3.4.1.13-2 shows peak suppression pool temperature as a function of void coefficient.

3.3.4.1.14 Variation of Doppler Coefficient

The effect of Doppler coefficient on transient peak neutron flux average surface heat flux and vessel pressure during an MSIV closure was studied for the range 0.20 to 0.32¢/°F (nominal of minus 0.28¢/°F). Figure 3.3.4.1.14-1 shows the peaks plotted as a function of Doppler coefficient. Peak pool temperature is plotted against Doppler coefficient in Figure 3.3.4.1.13-2.

3.3.4.2 Turbine Trip Sensitivity Studies

3.3.4.2.1 Variation of Boron Delay

The SLCS timer delay logic was varied between 30 sec (from nominal value of 120 seconds) to 240 seconds. Suppression pool temperature and containment pressure remain unchanged, since the early part of the transient is the same as the base case. S/RV's close at about 70 seconds and no isolation takes place. Tables 3.3.4.2-1 and -2 summarize all of the sensitivity studies for the BWR/6 Turbine Trip cases.

3.3.4.2.2 Variation of Boron Capacity/Mixing

Variation in boron capacity/mixing is expected to have no impact on the peak pool temperature and containment pressure since the early part of the transient would remain unchanged with S/RV's closing near 71 seconds, and no isolation takes place.

3.3.4.2.3 Variation of HPCS/RCIC Flow Capacity

The HPCS flow capacity was varied ±20%. The -20% case results in isolation since water level falls below the Level 1 setpoint, leading to increased S/RV cycling. This gives a suppression pool (bulk) temperature 45°F higher than the base case. Containment pressure correspondingly increased by 3 psi. The +20% HPCS flow case however remains unchanged. Since isolation is avoided and S/RVs close at the some time as the base case. These results are graphically shown in Figure 3.3.4.2.3-1.

3.3.4.2.4 Variation in RHR Delay

The start time of the RHR was varied from 9 minutes to 16 minutes (the base value was 11 minutes). This change does not have any effect on the pool temperature and containment pressure since peak values occur (∿70 seconds) much before RHR is turned on.

3.3.4.2.5 Variation of Void Coefficient

The effect of void coefficient variation on the Turbine Trip transient was studied similar to the MSIV closure reported in Section 3.3.4.1.13. Figures 3.3.4.2.5-1 and 3.3.4.2.5-2 show the results.

3.3.4.2.6 Variation of Doppler Coefficient

Figure 3.3.4.2.6-1 shows the effect of Doppler coefficient variation on the Turbine Trip transient. The effect on suppression pool peak temperature is shown in Figure 3.3.4.2.5-2.

3.3.4.3 IORV Sensitivity Studies

3.3.4.3.1 Variation of Boron Delay

Increasing and decreasing the boron time delay by 5 minutes results in a peak bulk pool temperature 5°F higher and lower, respectively, compared to the base case. Containment pressures increased and decreased accordingly by ∿1 psi. Minimum water level essentially remained unchanged due to the long term characteristic of the transient. Figure 3.3.4.3.1-1 shows the results graphically. Table 3.3.4.2-1 lists values found for this variation as well as other sensitivities studied for the IORV event.

3.3.4.3.2 Variation of Boron Pumping Capacity/Mixing

The effective rate of boron injection into the core is the product of the boron pumping capacity and mixing efficiency. This effective rate was varied by

±50% resulting in peak pool temperature 4°F below and 14°F above the base case respectively. Containment pressures decreased and increased accordingly by 1 psi and 3 psi, respectively. Figure 3.3.4.3.2-1 graphically shows this variation.

3.3.4.3.3 Variation of RHR Capability

To determine the effect of varying RHR heat exchange capabilities, the base capacity of 3.2% NBR at 100°F ΔT (2172 BTV/sec-°F for the 251 size plant used as the base case) was altered by ±50%. Increasing the capacity by 50% yielded a 10°F temperature reduction and lowered the peak containment pressure by less than 2 psi. For the opposite case of a 50% decrease in RHR capacity the results were a 16°F increase in temperature and 3 psi pressure risk. Sensitivity of pool temperature is shown graphically in Figure 3.3.4.3.3-1.

3.3.4.3.4 Variation of RHR Delay

As in MSIV case the effect of varying RHR start time was found to be small for the BWR/6 IORV case. Increasing the start time by 5 minutes and 10 minutes resulted in a pool temperature increase of less than 2°F. This very break sensitivity of the pool temperature to RHR startup delay is shown graphically in Figure 3.3.4.3.4-1.

3.3.4.3.5 Variation of Service Water Temperature

This variation in service water temperature was found to have less effect on peak pool temperature and containment pressure than the MSIV closure case. Increasing these temperatures by 20°F (to the operating limit) produced a rise in pool temperature of 7°F and an increase of about 1 psi in peak pressure. Reducing the temperatures by 20°F yielded decreases of 7°F and about 1 psi, respectively. Figure 3.3.4.3.5-1 graphically shows the pool temperature variation plotted directly versus service water temperature.

3.3.4.3.6 Variation of RHR Capacity and Service Water

These two parameters were changed simultaneously to s tions of RHR design and plant site water temperature pool temperature is more sensitive to RHR capacity th temperature. Simultaneous increases of +50% in RHR c service water temperature (and a similar set of decr temperature changes of -2°F and +11°F, respectively. accordingly by -0.1 and +2 psi. Figure 3.3.4.3.5-1 graphically.

3.3.4.3.7 Variation of Pool Size

The suppression pool mass was varied by ±20% to simu plants. Larger pool mass results in a temperature r corresponding decrease of 1 psi in peak pressure. Sr a higher peak pool temperature and pressure by 9°F a Figure 3.3.4.3.7-1 graphically shows the result.

3.3.4.3.8 Variation of S/RV Capacity

The S/RV capacity was varied by ±20%. From the 7.1% in the base case, which is typical of a 218 size pl pool temperature was experienced when S/RV capacity 1°F lower peak pool temperature when the capacity i tainment pressure correspondingly changes by +1.6 p Figure 3.3.9.3.8-1 shows the results graphically.

3.3.4.4 BWR/6 Multiple Sensitivity Results

The majority of the sensitivity studies presented impact of varying a single parameter from the typi istics given in Section 2.3. They were presented sharp sensitivities are present for BWR ATWS perfo

Although summation of more than one variation may be in the right range if the parameter variations are small, this multiple-parameter use of the single-variation results has high risk of yielding misleading conclusions if changes are too large, or non-linear interactions are missed.

A few cases are included with variation of two parameters in the general sensitivity sections:

a. Simultaneous variation of the void and Doppler reactivity coefficients,

b. Simultaneous variation of RHR heat exchanger capacity and pool/service water temperatures.

Within the ranges provided, these terms appear to combine linearly, so that reasonable interpolations can be made for variations of both parameters. For example, see Figures 3.3.4.1.6-1 (Peak Bulk Pool Temperature) or 3.3.4.1.13-1 (Peak Pressure).

In order to check more complex combinations, two special multiple-variation cases were constructed from the BWR/6 MSIV closure base case. The parameters that were varied, and their expected individual impacts on peak bulk pool temperature are tabulated in Table 3.3.4.4-1 and Table 3.3.4.4-2, and compared to the calculated combined variation result. In the first case, the combined variation was expected to produce a net result near the base case (compensating deviations), and the actual calculation shows close agreement.

The second case was constructed in such a way that all the parameters contributed to increased severity of the event. In this situation, the simple summation of the individual terms was less accurate (the combined calculation showed a more severe result).

In either situation, but especially the latter case, extreme caution must be exercised in attempting to combine these sensitivity studies. These two cases constitute a very small smaple of all possible combinations. While "ballpark" results may be possible, significant deviations or discontinuities may be present and detailed conclusions should have further support.

3.3.5 Loss of Condenser Vacuum

3.3.5.1 Overview of Response Without Scram

This transient starts with a turbine trip due to low condenser vacuum, therefore the beginning is the same as the Turbine Trip event (see Section 3.3.2). There is a rapid steam shutoff causing pressure and power increases which are limited by the action of the S/RV's and RPT. Note that direct recirculation pump trip from turbine trip was conservatively neglected. Since the MSIV's and turbine bypass valves also close when condenser vacuum has further dropped to each respective setpoint, S/RV cycling increases considerably compared to the original Turbine Trip case. Even so, the bulk pool temperature and pressure are well within the containment design requirements. Therefore, this event is similar to the Turbine Trip event as far as the peak power and pressure characteristic are concerned and similar to the MSIV closure case with respect to suppression pool temperature and pressure.

3.3.5.2 Sequence of Events For Loss of Condenser Vacuum

The listing of significant events during this ATWS event is provided in Table 3.3.5-1. Results with ARI and also assuming its failure are presented.

This transient starts with the closure of all turbine stop valves (within about 0.1 second) when the unexpected decline in condenser vacuum reaches the turbine trip setpoint. If the unit has turbine-driven feedwater pumps, they would also trip at the same low vacuum setpoint. For the ARI failure case, the feedwater is assumed to remain as if motor-driven pumps were available until the feedwater limit action shuts them down (the most limiting case). Figure 3.3.5-1 shows the initial portions of the event for the more likely plant ATWS transient in which ARI provides a diverse logic path to quickly shut down the reactor, and Figure 3.3.5-2 shows the initial portion for the case in which ARI also is assumed to fail.

In both cases, the initial power and pressure increase. Neutron flux reaches 367% NBR near 1 second, fuel average heat flux reaches 135% NBR at about 3 seconds. Some fuel may experience boiling transition. However, coolable geometry is maintained. Peak pressure occurs at the vessel bottom and is 1235 psig near 3 seconds. The normal reactor scrams would occur from position switches on the valves, high neutron flux, and high vessel pressure but are not considered in this analysis. The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to the automatic action of RPT which is initiated when vessel dome pressure exceeds 1150 psig, and the relieving action of the S/RV's which all open then start reclosing near 8 seconds. By about 30 seconds, the condenser vacuum is assumed to have fallen enough to initiate MSIV and bypass valve closure. This results in another pressure and power rise to 1202 psig and 230% NBR, respectively. Both of these peaks are lower than the earlier values. Peak heat flux rises momentarily, but remains less than 80% and fuel geometry is maintained.

The long term behavior of this transient is very much like the MSIV closure event which is discussed in detail in Section 3.3.1.1. Figure 3.3.5-3 shows the long term behavior predicted for this event. The peak bulk pool temperature and pressure which occur near 27 minutes are 160°F and 6.3 psig, respectively. These values remain well within the containment design requirements of 185°F and 15 psig.

Thus it can be seen that the loss of condenser vacuum event combined with a failure to scram is adequately mitigated for a representative BWR6/Mark III.

3.3.6 Pressure Regulator Failure - Zero Steam Demand

3.3.6.1 Overview of Response Without Scram

This transient is described in detail in the following sections. Its initial characteristics are much like the MSIV closure event. A rapid steam shutoff and pressure and power increases are limited by the action of the S/RV's and RPT. The relatively slower closure of the control valves in this case makes it less severe than the MSIV events in terms of peak neutron flux, vessel pressure and fuel heat flux. Even though this event experiences S/RV cycling due to isolation of the system, the bulk pool temperature and pressure are well within the containment design requirements. The final portion of the event is similar to the MSIV closure event.

3.3.6.2 Sequence of Events For Pressure Regulator Failure - Zero Steam Demand

The listing of significant events during this ATWS event is provided. Results with and without ARI are presented.

This event is assumed to occur when the normal pressure regulator fails in a closed state with the backup regulator also assumed to have failed to take over control. Having both regulators in a failed state simultaneously with the reactor in a failure-to-scram condition is extremely improbable. However, a failure (closed) without backup action causes control valve closure without the bypass being available. The normal closure speed characteristics of the control valves coupled with the extra steamline volume available to take up part of the pressure disturbance, makes this event less severe than MSIV closure case. Figure 3.3.6-1 shows the initial portions of the event for the more likely plant transient in which ARI provides a diverse logic path to quickly shut down the unit and Figure 3.3.6-2 shows the case in which ARI also fails and the automated SLCS is called upon to shut down the plant.

In each case, the initial power and pressure are the same. Neutron flux reaches 404% near 2.5 seconds, fuel average heat flux reaches 143% NBR at about 3 seconds. Some fuel may experience boiling transition. However, all fuel maintains coolable geometry. The peak pressure occurs at the vessel bottom and is 1283 psig near 5 seconds. The normal reactor scram signals would occur from position switches on the valves, high neutron flux and high vessel pressure but are not considered in this analysis. The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to the automatic action is RPT which is initiated when vessel closure pressure exceeds 1150 psig and the relieving action of the S/RV's which all open, then start reclosing near 19 seconds.

The long term behavior of this transient is very much like the MSIV closure case given in detail in Section 3.3.1.2. Figure 3.3.6-3 shows the long term behavior predicted for this transient. The peak bulk pool temperature which occurs near 33 minutes is 167°F (6.9 psig). These values remain well within the containment design requirements of 185°F and 15 psig.

Thus is can be seen that the pressure regulator failure (zero demand) event combined with a failure to scram is adequately mitigated for a representative BWR 6/Mark III.

3.3.7 Loss of a Feedwater Heater

3.3.7.1 Overview of Response Without the Scram

This is a mild transient compared to the other ATWS events. The neutron flux does not reach the scram setpoint. The pressure rise is insignificantly small. Therefore, automatic ATWS logic (e.g., RPT) does not occur, nor are HPCS or RCIC initiated. This is a gradual subcooling transient. The entire transient settles out when the feedwater temperature fully stabilizes. The reactor settles out to a new equilibrium power condition at full core flow with recirculation flow assumed to be under manual control. If automatic flow control was active, the power increase would be less. Manual operator action accomplishes reactor shutdown.

3.3.7.2 Sequence of Events For Loss of a Feedwater Heater

In this event, loss of a key group of feedwater heaters gives the reactor feedwater flow (decreased 70°F) which produces an increase in core inlet subcooling leading to an increase in core power. Following the transport delay through the feedwater lines (neglected in this analysis) and the time constant delay for cool-down of the heater tubes, neutron flux and average fuel surface heat flux rise to a maximum value of 114% which is lower than the flux scram setpoint. No fuel reaches boiling transition, even if the plant was initially at thermal operating limits. The reactor is conservatively assumed to be on manual flow control, therefore, core inlet flow remains at 100%. Had the reactor been on automatic flow control, core inlet flow would have changed to decrease the severity of the transient. The peak pressure (vessel bottom) of 1071 psig occurs near 74 seconds. Figure 3.3.8-1 shows the short term response of this event. The water level remains within the normal control range throughout the transient.

When the power reaches 108% NBR near 30 seconds, a high power alarm occurs. For this analysis, it is assumed that attempts will be made to bring the power down by inserting rods. If this is not successful manual scram will be initiated at 10 minutes. This action also initiates ARI and SLCS timed logic. However, in this analysis, manual scram is also assumed to fail. By about 11 minutes, ARI will have been accomplished and power is terminated.

If the ARI function is arbitrarily assumed to fail, as well as all other attempts to insert control rods within the two minute period, the automatic start of boron injection will begin through the HPCS line. An extra 30 seconds is allowed in the analysis for transport in the section of this line into the vessel without HPCS flow on. By about 35 minutes the power has decreased below 1% NBR. Recirculation flow remains active during boron injection, providing mixing and dispersion throughout the primary system.

Thus it can be seen that a loss of feedwater heater event combined with a failure to scram is adequately mitigated for a representative BWR 6/Mark III. Note that this event was analyzed for a 70°F loss in feedwater heating rather than 60°F as specified in NUREG 0460 (Volume 3).

3.3.8 Feedwater Controller Failure - Maximum Demand

3.3.8.1 Overview of Response Without Scram

The initial short term portion of this transient results in a gradual power increase, then a sharp pressure rise and power peak. The longer term segment requires evaluation of coolant and containment conditions as the reactor is shut down.

Relief valve action occurs only during the early portion of the transient. RPT, acting in conjunction with the relief valves, serves to effectively limit the pressure disturbance. Note that the direct RPT from turbine trip was conservatively ignored. RPT also ensures relatively low power generation during the long term portion of the event. The effectiveness of RPT as presented in earlier reports is again confirmed by this analysis.

Containment peak temperature and pressure remain well below design limits due to the short duration of relief flow to the suppression pool. Power shutdown can be achieved in two ways. ARI employs an alternate design of the protection logic leading to a diverse insertion of the control rods. In the event that ARI also fails, the automated SLCS provides further protection and shutdown capability.

3.3.8.2 Sequence of Events for Feedwater Controller Failure - Maximum Demand

The time sequence of events for this transient is presented in Table 3.3.8-1. Both successful ARI initiation, and ARI failure cases are considered. The initiating event is the failure of the feedwater controller to the maximum demand position (125% NBR was assumed). The feedwater flow rapidly responds, causing vessel level to rise. When the high level trip setpoint (L8) is reached near 16 seconds, the turbine and feedwater are tripped. This results in a scram signal which, for purposes of this analysis, fails to initiate a scram. With the occurrence of the turbine trip, this event becomes very similar to the Turbine Trip transient. Figures 3.3.8-1 and 3.3.8-2 show the early portion of the event for the cases of ARI failure, and successful ARI actuation, respectively. For each case, the peak power and flux are the same with a maximum flux of 396 NBR near 7 seconds and a peak vessel bottom pressure of 1214 psig around 9 seconds. Fuel average heat flux reaches a maximum at 9 seconds of 126% NBR. Some fuel may experience boiling transition, however, coolable geometry is maintained. Despite the assumed failure to scram based upon high neutron flux, vessel level and dome pressure generated scram signals, the transient pressure is maintained well below the 1500 psig Service Level C overpressure limit. This is accomplished through the combination of RPT (initiated on high dome pressure) and actuation of relief valves. Relief valve flow begins at 8 seconds. S/RV's will open and cycle before permanently closing. This is shown along with vessel steam flow. The difference in vessel and relief steam flow is made up by the steamflow through the turbine bypass valves to the condenser.

At approximately 28 seconds, ARI will begin to insert control rods into the core thereby shutting down the reactor. This will deactivate the SLCS turning the remainder of the event into normal feedwater flow controller failure transient. No further relief valve flow will occur. The decay heat will be passed through the turbine bypass valves to the condenser.

Peak suppression pool temperature is 95°F at 200 seconds with a corresponding peak pressure of less than 1 psig. The RHR can be activated in the pool cooling mode whenever convenient to reduce the pool temperature and any final, single valve cycles can be accommodated. Vessel level, which drops due to feedwater shutoff at high water level, is recovered and maintained in the normal water range by means of the HPCS/RCIC systems.

In the unlikely event of ARI failure, the event can still be mitigated through action of the SLCS. With confirmation from the flux monitoring system and the rod position indicating system that scram has not occurred, the SLCS will be activated. The long term behavior predicted for this event is shown in Figure 3.3.8-3. Boron first enters the core at about 200 seconds via the HPCS system and commences to shut down the system, with hot shutdown occurring near 17 minutes. Vessel level experiences slow cycles about the normal water level caused by the intermittent action of the RCIC and HPCS systems assumed to be automatically cycling between L2 and L8. Boron concentration will continue to increase until the entire inventory has been injected into the core around 50 minutes. At this point the concentration is sufficient to maintain cold nuclear shutdown conditions when the RHR system is switched to the reactor shutdown cooling mode and the plant is brought to a cold shutdown condition.

Thus it can be seen that a feedwater controller failure event (maximum demand) combined with a failure to scram is adequately mitigated for a representative BWR 6/Mark III.

3.3.9 Pressure Regulator Failure - Maximum Steam Demand

3.3.9.1 Overview of Response Without Scram

The initial portion of the transient consists of a decrease in reactor pressure and power as the turbine control valves open to the maximum position followed by a rapid rise in pressure and power due to MSIV closure on low steam line pressure. Scram is normally initiated at this time, from the MSIV position switches. Should these signals fail, additional scram signals occur from high flux, high pressure and low water level. Once the MSIV's close, the characteristics of the remaining portion of the transient are very much the same as the MSIV event.

The power and pressure increases are limited by the action of the S/RV's and RPT. With normal scram assumed to be failed, the long term power shutdown is achieved in two ways. ARI employs an alternate design of the protection logic leading to diverse insertion of the control rods. In the unlikely event that ARI fails the automated SLCS provides further protection and shutdown capability.

3.3.9.2 Sequence of Events For Pressure Regulator Failure - Maximum Demand

The listing of significant events during this event is provided below. Results for both cases - with ARI and also assuming its failure - are presented.

The event begins with the inadvertent failure of the pressure regulator to the maximum demand value. This causes a quick increase in vessel steam flow which results in a rapid decrease in vessel pressure leading to a low pressure isolation setpoint at about 16 seconds. The MSIV's are tripped closed. Once this occurs, the transient is essentially much like an MSIV closure event. The isolation is followed by a rapid rise in power and pressure. Figures 3.3.9-1 and 3.3.9-2 show the initial portions of the event for the more likely plant ATWS transient in which ARI quickly shuts down the reactor and the case in which ARI also fails and automated SLCS is called upon to shut the reactor down. In both cases, the peak power and pressure are the same. The neutron flux reaches 509% NBR near 23 seconds, fuel average heat flux reaches 153% NBR at about 24 seconds. Some fuel may experience boiling transition however, coolable geometry is maintained. The peak pressure occurs at vessel bottom and is 1296 psig near 27 seconds. The normal reactor scrams occur from position switches on MSIV's, high neutron flux, and the high vessel pressure but are not considered for this analysis. The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to the automatic action of RPT (which is initiated when vessel pressure exceeds 1150 psig near 23 seconds) and the relieving action of the S/RV's which all open, then start reclosing near 39 seconds.

By about 43 seconds, the high pressure logic which began the ATWS protection will have accomplished the ARI function. This deactivates the automatic boron injection and feedwater limit and turns the remainder of the event into normal pressure regulator failure shutdown. The relief valve flow stops near 63 seconds.

Peak suppression pool temperature will occur at the time of the last relief action and will be 95°F. The RHR can be activated in pool cooling whenever convenient to control temperature. Reactor water level is restored to its normal range by feedwater flow and RCIC and HPCS flow.

If the ARI function is arbitrarily assumed to fail as well as all other attempts to insert control rods within the two-minute timed period, the ATWS logic will continue to sense that the APRM signals are not cownscale and not enough rods are in their full-in positions, and the automatic start of boron injection will begin. The long term behavior predicted for this event is shown in Figure 3.3.9-3. Introduction of boron to the core at 3 minutes and 23 seconds again restores level and core flow before dropping the power near 20 minutes when nuclear shutdown is achieved. Thereafter, only decay heat reaches the pool, giving a peak pool temperature of 167°F (6.9 psig) at about 30 minutes. These values remain within the containment design requirements of 185°F and 15 psig. Water level inside the core shroud is a two-phase mixture which remains well above the core and up into the steam separator standpipes as RCIC and HPCS flow provide coolant inventpry. The boron will continue to build the poison concentration in the vessel until it is all injected near 50 minutes making it possible for a controlled reactor cooldown. The total concentration is specified to be enough to maintain cold nuclear reactor shutdown conditions even when the RHR system is eventually switched to the reactor shutdown cooling mode, bringing the plant to cold shutdown by normal procedures.

Thus it can be seen that the pressure regulator failure (maximum demand) combined with failure to scram is adequately mitigated for a representative BWR 6/Mark III.

3.3.10 Loss of Feedwater

3.3.10.1 Overview of Response Without Scram

This event has no rapid excursions as in some of the other events but is a long term power reduction and depressurization. Since the pressure begins to fall at the outset of the transient, the need for relief valves does not arise until isolation occurs very late in the event, and only single valve cycling is expected to handle decay heat. The containment limits are not approached. Except for the use of the liquid boron solution for shutdown, the procedure followed here is virtually identical to the normal shutdown event.

3.3.10.2 Sequence of Events For Loss of Feedwater

In this event all feedwater flow is assumed to be lost in about 5 seconds. The resulting sequence of events is shown in Table 3.3.10-1 for both cases with and without ARI. Figure 3.3.10-1 shows the initial portion of the event for the more likely plant ATWS transient in which ARI quickly shuts down the reactor. Figure 3.3.5-2 shows the case in which ARI also fails and the automated SLCS is called upon to shut down the reactor.

In both cases, after the loss of feedwater has taken place the pressure, water level and neutron flux begin to fall. Around 13 seconds low water level (L2) is reached. This trips the recirculation pumps, initiates ARI, initiates the

HPCS and RCIC and activates the SLCS timed logic. Not included in the analysis was the recirculation runback which would have occurred earlier from coincident low level alarm (L4) and low feedwater flow. By 33 seconds the low water level logic which began the ATWS protection will have accomplished the ARI function. This deactivates the automatic boron injection. At about 32 seconds HPCS and RCIC flows start. They replace the main feedwater system flow and begin to overcome the inventory loss. The vessel level decreases slightly faster immediately following ARI and the minimum level is reached near 47 seconds as shown in Figure 3.3.10.1. The two-phase mixture level inside the core shroud always remains above the top of the fuel, however, the primarily single phase level outside the shroud does drop below the Level 1 setpoint, causing MSIV's to start closing at about 45 seconds. Since drywell pressure is not elevated, ADS logic is not actuated. Since by this time the neutron flux has already been decreased well below 1% NBR by the ARI action, there is no significant steam generation. Vessel pressure continues to fall as shown in Figure 3.3.10-1, as quenching by the RCIC and HPCS continues. The HPCS and RCIC will restore level to its normal range, for either automatic cycling between Level 2 and 8 setpoints or the operator takes over manual level control by using the RCIC (the preferred method). Pressure is expected to increase to the lowest S/RV setpoint when HPCS/RCIC are off (level restored), and one cycling valve is expected without significant pool temperature increase.

If the ARI function is arbitrarily assumed to fail as well as all other scrams and attempts to insert enough control rods within the two-minute timed period, the ATWS logic will continue to sense that the APRM signals are not downscale and not enough rods are in their full-in positions, and automatic boron injection will begin. The power is predicted to remain in the 10-20% range, with core flow and level being restored during the first part of the boron injection as shown in Figure 3.3.10-2 and extended through the long term transient transient in Figure 3.3.10-3. The significant features during the early part of the event are the same as the previous case (ARI). The key difference is that the minimum water level is reached around 71 seconds and stays slightly above the Level 1 setpoint. MSIV isolation may be avoided by taking the mode switch out of the "RUN" mode. This water level behavior is attributed to higher void fraction in the core as a result of higher power relative to the previous case in which ARI reduces power and core void fraction. SLCS boron injection is started near 2 minutes and it reaches the core 1 minute later. During the following 16 minute period (out to about 1100 seconds in Figure 3.3.10-3), the key result is that power is suppressed slightly, reducing the steaming rate and allowing water level to be restored. This also induces higher natural circulation core flow which follows the water level behavior. The level reaches the high level turn-off (Level 8) of the HPCS and RCIC at about 100 seconds. The turbine is also tripped at this level but since the turbine steam bypass system opens immediately, no significant pressure disturbance is experienced.

By 1125 seconds the generated power is below 1% NBR and continues to decrease due to the accumulation of boron in the reactor. The net reactivity also stays negative. This accomplishes nuclear hot shutdown. The vessel pressure is steadily decreasing and at around 19 minutes, MSIV isolation occurs due to low vessel pressure. By this time the generated power is practically zero and the only heat in the vessel is the decay heat.

The reactor pressure is expected to return to the setpoint of the lowest S/RV when HPCS and RCIC are off and are not quenching steam. The decay heat will cycle this lowest valve, but no significant suppression pool heatup is expected. The reactor would be cooled down at normal rates using the relief valve(s) to cold shutdown.

Thus it can be seen that loss of feedwater combined with a failure to scram is adequately mitigated for a representative BWR 6/Mark III.

3.3.11 Loss of Normal AC Power

3.3.11.1 Overview of Response Without Scram

The initial portion of the transient sees a sharp rise in reactor pressure and power due to MSIV closure as a result of loss of normal AC power. Scram is initiated at this time from the MSIV position switches if it had not occurred yet from loss of reactor trip system power. Should these signals fail, additional scram signals occur from high flux, high pressure and low water level. The power and pressure increases are limited by the action of the S/RV's and RPT (which occurs at the start of this event). With normal scram assumed to have failed the long term power shutdown is achieved in either of two ways. ARI employs an alternate design of the protection logic leading to diverse insertion of the control rods. In the event that ARI also fails, the automated SLCS provides further protection and shutdown capability.

3.3.11.2 Sequence of Events For Loss of Normal AC Power

The listing of significant events during this event is provided in Table 3.3.11-1. Results for both cases - with ARI and also assuming its failure are presented.

There are two ways of initiating this event. These are loss of all auxiliary power transformers and loss of all grid connections. The main difference between the two approaches is that in the latter, load rejection occurs at the outset of the transient which results in turbine-generator trip. In either case MSIV closure takes place near 2 seconds. This is the earliest time isolation can occur and is based on relay-type RTS circuitry. In the case of solid state RTS circuitry MSIV closure takes place later, and the event is less severe.

Since in loss of all grid connections the turbine trips first as opposed to MSIV closure in the loss of all auxiliary power transformers case, it turns out to be a less severe event in terms of peak power and pressure. Therefore the rest of the discussion is focused on the case where loss of all auxiliary power transformers occur. The sequence of events as outlined in Table 3.3.11-1 describes the event. Since loss of power takes place it is assumed that accumulator will have enough air to last one cycle of the S/RV valves at their relief setpoints after which they will switch over to spring setpoints. The low-low set S/RV design actually has greater capability for cycling in the relief mode, and would give lower pressures.

This event begins with the loss of recirculation pumps and feedwater pumps since the condensate and/or booster pumps are also tripped due to loss of power. This leads to an initial fall in power and pressure. Near 2 seconds MSIV closure is assumed to take place, which results in a rapid rise in power and pressure. Figure 3.3.11-1 shows initial portions of the event for the more likely ATWS transient in which ARI quickly shuts down the reactor, and Figure 3.3.11-2 shows the initial portion of the case in which ARI also fails and automated SLCS is called upon to shut the reactor down.

In both cases, the peak power and pressure are the same. The neutron flux reaches 546% NBR near 7 seconds, fuel average heat flux reaches 101% NBR at about 8 seconds. Some fuel may experience boiling transition. The peak pressure occurs at vessel bottom and is 1218 psig near 8 seconds. The normal scram signals occur due to loss of AC power and also due to position switches on MSIV's, high neutron flux and the high vessel pressure but are not considered for this analysis. The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to RPT at the start of the transient and the relieving action of the S/RV's which all open, then start reclosing near 14 seconds.

By about 27 seconds, the high pressure logic would provide ATWS protection by activating ARI. This deactivates the automatic boron injection and allows the remainder of the event to proceed toward shutdown. The primary relief valve flow stops near 42 seconds, followed only by single valve cycling on the "tail" of the isolation event. The RHR can be activated in pool cooling mode as soon as water level recovery is clearly indicated, to control pool temperature. Reactor water level is restored quickly to its normal range by RCIC and HPCS flow.

If the ARI function is arbitrarily assumed to fail as well as all other attempts to insert control rods within the two-minute timed period, the ATWS logic will continue to sense that not enough rods are in their full-in positions, and the automatic boron injection will begin. The long term behavior predicted for this event is shown in Figure 3.3.11-3. Introduction of boron to the core around 3 minutes again restores level and core flow before decreasing power near 22 minutes when nuclear shutdown is achieved. Thereafter, only decay heat reaches the pool, giving the peak bulk pool temperature of 150°F (4.5 psig) at about 33 minutes. These values remain well within the containment design requirements of 185°F and 15 psig. Water level inside the core shroud is a two-phase mixture which remains well above the core and up into the steam separator standpipes as RCIC and HPCS flow provide coolant inventory. The boron will continue to build the poison concentration in the vessel until it is all injected near 50 minutes making it possible for a controlled reactor cool-down. The total concentration is specified to be enough to maintain cold nuclear shutdown conditions even when the RHR system is eventually switched to the reactor shutdown cooling mode, bringing the plant to cold shutdown.

Thus it can be seen that a loss of normal AC power combined with a failure to scram is adequately mitigated for a representative BWR 6/Mark III.

3.3.12 Recirculation Flow Controller Failure - Maximum Demand

3.3.12.1 Overview of Response Without Scram

This transient is not severe enough to trip any ATWS logic nor initial HPCS or RCIC flow. It is considerably milder than the MSIV closure or turbine trip ATWS cases. This is a short term transient with a sudden power rise and relatively small pressure increase. The entire transient is over within 30 seconds by which time the reactor settles out to a new equilibrium condition of less than 100% rated power. Since the peak pressure stays below the lowest S/RV setpoint, steam flow to the suppression pool does not take place.

3.3.12.2 Sequence of Events For Recirculation Flow Controller Failure - Maximum Demand

This event can take place in either of two ways: 1) failure occurs within either loop's flow controller; or 2) malfunction of master controller results in increased core coolant flow in both loops. The most severe case of increasing coolant flow results when failure of flow controller of one recirculation loop results in a maximum valve stroking rate while at the lowest initial power flow conditions on the automatic flow control line. Figure 3.3.12-1 shows responses to such a failure. The initial reactor condition corresponds to 54% of rated power and 34% of rated flow which is based on minimum valve position and minimum recirculation pump speed on the 105% NBR steamflow rod pattern flow control line.

The rapid increase in core inlet flow causes a large neutron flux peak which would initiate a reactor scram signal (assumed not to scram). As is depicted in Figure 3.3.12-1 neutron flux then peaks at 2 seconds to a value of 247% and settles to a steady state value of 71% which is higher than the initial value yet within the normal power-flow range of the reactor. Surface average heat flux increases to 88% and also settles to a value of 71%. No fuel reaches boiling transition even if the core is operating initially at its operating limit. The entire transient is over within 30 seconds. The peak pressure (vessel bottom) of 1013 psig occurs near 5 seconds. Core inlet flow increases to 65% and is held at that value, thereby insuring adequate core coverage.

As stated above, the transient is not severe enough to trip the ATWS logic nor initiate HPCS or RCIC flow since normal feedwater and level control is maintained. If manual rod insertion or scram is not possible, manual ARI/SLCS will be utilized to shut down the plant. Assuming manual ARI/SLCS is initiated at 10 minutes, ARI would shut the plant down like a normal scram, but if it also failed, the boron injection would reduce power to below 1% and achieve nuclear shutdown by about 30 minutes after the initial event. Recirculation flow would be maintained near full flow initially and at partial flow (low frequency M/G sets on) in order to maximize boron dispersion throughout the vessel, and to provide a near-normal shutdown sequence.

Thus it can be seen that a recirculation flow controller failure combined with a failure to scram is adequately mitigated for a representative BWR 6/Mark III.

3.3.13 Startup of the Idle Recirculation Pump

This event is similar to the recirculation flow controller failure - maximum demand. Both of these events result in increased core power which results from the increased core flow. Startup of the idle recirculation pump event has been shown in safety analysis reports to be less severe than the recirculation flow controller failure and therefore further transient-specific analyses have not been done.

3.3.14 Inadvertent Opening of All Bypass Valves

This event will be similar to the pressure regulator failure - maximum steam demand. Since the turbine control valves will try to compensate for the pressure reduction, the results will be less severe. For these plants with smaller bypass capacity, the event will be even less severe.

3.3.15 Shutdown Cooling (RHR) Malfunction - Decreasing Temperature

This event can only occur at very low pressures. The shutoff head of the shutdown cooling pumps is less than 300 psig. In this condition, the reactor has almost no voids in it and therefore only little if any positive reactivity is inserted. Therefore, this event is not considered further.

Table 3.3-1

SUMMARY OF ATWS RESULTS - BWR/6

ARI FAILURE, 2 PUMP SLCS, 2 MINUTE LOGIC DELAY

Transient	Maximum Neutron Flux (% NBR)	Maximum Average Fuel Heat Flux (% NBR)	Maximum % of Fuel in Boiling Transition	Maximum Pressure (Vessel Bottom) (psig)	Maximum Suppression Pool Temperature (°F)	Maximum Containment Pressure (psig)
MSIV Closure	745	147	-	1299	167	6.9
Turbine Trip with Bypass	358	135	-	1225	100	0.5
Inadvertent Opening of a S/R Valve	-	-	-	-	170	7.3
Loss of Condenser Vacuum	367	135	-	1235	163	6.3
Pressure Regulator Failure - Zero Steam Demand	404	143		1283	167	6.9
Loss of a Feedwater Heater	115	114	-	1071	90	No change
Feedwater Controller Failure - Max Demand	396	126	16	1214	95	0.3

Table 3.3-1

SUMMARY OF ATWS RESULTS - BWR/6

ARI FAILURE, 2 PUMP SLCS, 2 MINUTE LOGIC DELAY (Continued)

Transient	Maximum Neutron Flux (% NBR)	Maximum Average Fuel Heat Flux (% NBR)	Maximum % of Fuel in Boiling Transition	Maximum Pressure (Vessel Bottom) (psig)	Maximum Suppression Pool Temperature (°F)	Maximum Containment Pressure (psig)
Pressure Regulator Failure - Max Steam Demand	509	153	-	1296	167	6.9
Loss of Feedwater	100	100	-	1061	90	No change
Loss of Normal AC Power	546	101	5	1218	150	4.5
Recirculation Flow Controlled Failure - Maximum Demand	247	88	-	1013	90	No change
Turbine Trip with Bypass Failure	773	144	-	1285	168	7.0

Table 3.3.1-1

BWR/6 MSIV CLOSURE

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Nominal (4-sec) MSIV closure begins - All normal scrams fail	0	0
2.	Pressure and power rise begins	0	0
3.	Relief valves lift	4 Seconds	4 Seconds
4.	ATWS high pressure setpoint is reached (1150 psig): Recirculation pumps are tripped, ARI is initiated, and SLCS timed Logic is activated	5 Seconds	5 Seconds
5.	Some fuel may experience boiling transition	5 Seconds	5 Seconds
6.	Vessel pressure peaks	8 Seconds	8 Seconds
7.	ARI control rod injection completed eliminating SLCS and feedwater limit actions	25 Seconds	Fails
8.	ATWS logic initiates feedwater limit	N/A	30 Seconds
9.	Feedwater flow stops (or limited)	45 Seconds	45 Seconds
10.	Reactor water level drops to Level 2, initiates RCIC and HPCS and containment isolation.	48 Seconds	52 Seconds
11.	HPCS and RCIC flow starts	68 Seconds	72 Seconds
12.	ATWS logic timer initiates SLCS	N/A	2 Minutes
13.	Water level reaches minimum and begins to rise. At all times, fuel remains covered.	71 Seconds	3 Minutes
14.	Liquid control flow reaches core	N/A	3 Minutes
15.	RHR flow begins (pool cooling)	11 Minutes	11 Minutes
16.	Hot shutdown achieved	25 Seconds	20 Minutes
17.	Peak containment pressure and pool temperature	130 Minutes	30 Minutes

Table 3.3.1-2

BWR/6 MSIV CLOSURE - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay MSIV
Maximum Neutron Flux (%)	745
Maximum Vessel Bottom Pressure (psig)	1299
Maximum Average Heat Flux (%)	147
Maximum Bulk Suppression Pool Temperature (°F)	167
Associated Containment Pressure (psig)	6.9

Table 3.3.2-1

BWR/6 TURBINE TRIP EVENT

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Turbine Trips, Bypass Opens - All Normal Scrams Fail	0	0
2.	Pressure and Power Rise Begins	0	0
3.	Relief Valves Lift	2 Seconds	2 seconds
4.	ATWS High Pressure Setpoint is Reached (1150 psig): Recirculation Pumps are Tripped*, ARI is Initiated, and SLCS Timed Logic is Activated	2 Seconds	2 seconds
5.	Peak Vessel Pressure Occurs	3 Seconds	3 Seconds
6.	Some Fuel Experiences Boiling Transition	3 Seconds	3 seconds
7.	ARI Control Rod Insertion Completed, Eliminating SLCS and FW-Limiting Actions	22 seconds	Fails
8.	ATWS Logic Timer Initiates Feedwater Flow Limit	N/A	27 seconds
9.	Reactor Water Level Drops to Level 2, Initiates HPCS and RCIC, and Containment Isolation	N/A	56 seconds
10.	Peak Containment and Peak Bulk Temperature Occur when S/RV's all reclose	22 seconds	65 seconds
11.	HPCS and RCIC Flow Begins	N/A	76 seconds
12.	Reactor Water Level Reaches Minimum and Begins to Rise. Fuel always remains covered.		2 minutes
13.	ATWS Logic Timer Initiates SLCS	N/A	2 minutes
14.	Liquid Control Flow Reaches Core	N/A	3 minutes
15.	RHR Flow Begins (Pool Cooling)	$\geq$11 minutes	11 minutes
16.	Hot Shutdown Achieved	22 seconds	19 minutes

*Direct recirculation pump trip from turbine stop valve closure was conservatively neglected.

Table 3.3.2-2

BWR/6 TURBINE TRIP - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Turbine Trip
Maximum Neutron Flux (%)	358
Maximum Vessel Bottom Pressure (psig)	1225
Maximum Average Heat Flux (%)	135
Maximum Bulk Suppression Pool Temperature (°F)	100
Associated Containment Pressure (psig)	0.5

Table 3.3.3-1

BWR/6 INADVERTENT OPENING OF A RELIEF VALVE

	Sequence of Events	Time
1.	Relief valve opens inadvertently and fails to close	0
2.	Alarm sounds at 95°F and operator initiates pool cooling	2 minutes
3.	Suppression pool temperature reaches 110°F operator attempts manual scram, scram fails, initiating ARI and SLCS logic	9 minutes
4.	ARI fails	9.5 minutes
5.	SLCS starts	11 minutes
6.	Control liquid reached core	12 minutes
7.	Power is less than relief valve capacity	20 minutes
8.	Isolation on low steamline pressure (800 psig)	23 minutes
9.	Peak suppression pool temperature and pressure are reached	50 minutes

Table 3.3.3-2

BWR/6 INADVERTENT OPENING OF A RELIEF VALVE - SUMMARY

With ARI Failure	86 GPM - 2Min Logic Delay IORV
Maximum Bulk Suppression Pool Temperature (°F)	170
Associated Containment Pressure (psig)	7.3

Table [illegible].3.4.1-1

BWR-6 MSIV ATWS SENSITIVITY RESULTS SUMMARY

Sensitivity	Change in Maximum Neutron Flux From Base Value (% NBR)	Change in Maximum Heat Flux From Base Value (% NBR)	Change in Maximum Pressure (Vessel Bottom) From Base Value (psi)	Change in Minimum Water Level From Base Value (ft)	Change in Maximum Bulk Suppression Pool Temperature From Base Value (°F)	Change in Maximum Containment Pressure From Base Value (psi)
MSIV Base Case	745(1) @ 4 Sec	147(1) @ 4.9 Sec	1299(1) @ 7.3 Sec	13.28(1) @ 175 Sec	167°(1) @ 1200 Sec	6.85(1)
MSIV-Sensitivity to Boron Delay - 30 Sec	0	0	0	-0.68 @ 123 Sec	-3 @ 1700 Sec	-0.45
MSIV-Sensitivity to Boron Delay - 240 Sec	0	0	0	0	+6 @ 1400 Sec	+0.98
MSIV-43 GPM* (-50%) Boron Pump Capacity	0	0	0	0(2)	+40 @ 2200 Sec	+5.89
MSIV-129* GPM (+50%) Boron Pump Capacity	0	0	0	0	-13 @ 1700 Sec	-1.82
MSIV-55% Boron (-27%) Mixing Efficiency	0	0	0	0	+17 @ 1700 Sec	+3.05
MSIV-95% Boron (+27%) Mixing Efficiency	0	0	0	0	-8 @ 1700 Sec	-1.16
MSIV-80% Nom HPCS Flow	0	0	0	+1.81 @ 144 Sec	-9 @ 2000 Sec	-1.3
MSIV-120% Nom HPCS-Flow	0	0	0	-1.13 @ 210 Sec	+7 @ 1100 Sec	+1.15
MSIV-50% RHR Capacity	0	0	0	0 @ 175 Sec	+7 @ 4200 Sec	+1.15
MSIV-150% RHR Capacity	0	0	0	0 @ 175 Sec	-3 @ 1200 Sec	-0.45
MSIV-Delay RHR Turn-on 9 Minutes	0	0	0	0 @ 175 Sec	-1 @ 1800 Sec	-0.15
MSIV-Delay RHR Turn-on 16 Minutes	0	0	0	0 @ 175 Sec	+3 @ 1200 Sec	+0.32

Table 3.3.4.1-1 (Continued

BWR-6 MSIV ATWS SENSITIVITY RESULTS SUMMARY

Sensitivity	Change in Maximum Neutron Flux From Base Value (% NBR)	Change in Maximum Heat Flux From Base Value (% NBR)	Change in Maximum Pressure (Vessel Bottom) From Base Value (psi)	Change in Minimum Water Level From Base Value (ft)	Change in Maximum Bulk Suppression Pool Temperature From Base Value (°F)	Change in Maximum Containment Pressure From Base Value (psi)
MSIV-Service Water Initially - 65°F	0	0	0	0 @ 175 Sec	-19 @ 1200 Sec	-4.54
MSIV-Service Water Initially - 105°F	0	0	0	0 @ 175 Sec	18 @ 1200 Sec	+3.25
MSIV-50% RHR Capacity 65°F Service Water	0	0	0	0	-12 @ 5200 Sec	-1.69
MSIV-150% RHR Capacity 105°F Service Water	0	0	1299 @ 7.3 Sec	0	+16 @ 1201 Sec	+2.84
MSIV-80% Suppression Pool Size	0	0	0	0 @ 175 Sec	+16 @ 1200 Sec	+2.84
MSIV-120% Suppression Pool Size	0	0	0	0 @ 175 Sec	-11 @ 1800 Sec	-1.56
MSIV-80% SRV Capacity	0	0	148 @ 12.2 Sec	-0.22(2) @ 170 Sec	+2 @ 1600 Sec	+0.32
MSIV-120% SRV Capacity	0	0	-33 @ 5 Sec	+0.84 @ 135 Sec	+2 @ 1600 Sec	+0.32
MSIV-0.93 Sec RPT Delay	0	0	+3 @ 7.75 Sec			
MSIV-1.43 Sec RPT Delay	0	0	-7 @ 8.20 Sec			
MSIV-150% RPT Inertia	0	0	+2 @ 7.6 Sec			
MSIV-80% RPT Inertia	0	0	0			

(1) Absolute values for base case
(2) FT. below bottom of Separator Skirt
* Based on 251-size plant

Table 3.3.4.1-2

BWR/6 MSIV ATWS NUCLEAR PARAMETRIC STUDY SUMMARY

				Change in Peak Value		
Doppler Coef (¢/°F)	Void Coef (¢/%)	Neutron Flux (%)	Average Heat Flux (%)	Vessel Bottom Pressure (psi)	Suppression Pool Temperature (°F)	Min Level/Time (wide range) (ft/sec)
−0.200	-6	-181	-7.5	-17		
−0.230	-6	-223	-10.0	-27		
−0.280	-6	-277	-13.3	-39		
−0.320	-6	-311	-15.7	-46		
−0.200	-8	-13	-1.0	+5		
−0.230	-8	-74	-3.4	-6		
−0.280	-8	-149	-6.9	-21	+2.1	-1.0/199
−0.320	-8	-197	-9.5	-30		
−0.200	-11	+158	+5.2	+26		
−0.230	-11	+88	+3.2	+15	-1.3	+0.2/157
−0.280*	-11*	745*	147.3*	1299*	167.1*	-13.1/180*
−0.320	-11	-57	-2.4	-9	+3.1	-1.0/143
−0.200	-14	+113	+9.9	+42		
−0.230	-14	+63	+8.1	+31		
−0.280	-14	0	+5.1	+16	-1.5	+0.1/143
−0.320	-14	-43	+2.9	+6		

*Values shown for nominal void and Doppler coefficients are absolute peaks. Other peaks are relative to these.

Table 3.3.4.2-1

BWR/6 TURBINE TRIP - ATWS SENSITIVITY RESULTS SUMMARY

Sensitivity	Change in Maximum Neutron Flux From Base Value (% NBR)	Change in Maximum Heat Flux From Base Value (% NBR)	Change in Maximum Pressure (Vessel Bottom) From Base Value (psi)	Change in Minimum Water Level From Base Value (ft)	Change in Maximum Bulk Suppression Pool Temperature From Base Value (°F)	Change in Maximum Containment Pressure From Base Value (psi)
TT Base Case	358(1)	135(1)	1225(1)	10.42(1)(2) @ 117 Sec	100(1)	0.5(1)
TT - 30 Sec Boron Delay	0	0	0	+60.33 @ 112 Sec	0	0
TT - 240 Sec Boron Delay	0	0	0	+0.28 @ 112 Sec	0	0
TT - 80% Nominal HPCS	0	0	0	+4.82 @ 153 Sec	+43 @ 2000 Sec	+3.27
TT - 120% Nominal HPCS Flow	0	0	0	-0.65 @ 95 Sec	100° @ 71 Sec	0
TT - RHR Turn-on 9 minutes	0	0	0	0	100° @ 66 Sec	0
TT - RHR Turn-on -16 minutes	0	0	0	0	100° @ 66 Sec	

(1) Absolute values for Base Case
(2) FT. below Separator Skirt

Table 3.3.4.2-2

BWR/6 TURBINE TRIP ATWS NUCLEAR PARAMETRIC STUDY SUMMARY

		Change in Peak Value				
Doppler Coef (¢/°F)	Void Coef (¢/%)	Neutron Flux (%)	Average Heat Flux (%)	Vessel Bottom Pressure (psi)	Suppression Pool Temperature (°F)	Min Level/Time (wide range) (ft/sec)
−0.200	-6	-97	-6.3	-8		
−0.230	-6	-106	-7.7	-9		
−0.280	-6	-119	-9.8	-12		
−0.320	-6	-128	-11.4	-13		
−0.200	-8	-53	-1.4	-2		
−0.230	-8	-65	-3.0	-3		
−0.280	-8	-81	-5.5	-6	+1.1	-1.8/123
−0.320	-8	-92	-7.3	-8		
−0.200	-11	+44	+3.9	+4		
−0.230	-11	+25	+2.4	+2	-0.8	+1.0/119
−0.280*	-11*	358*	134.3*	1225*	99.8*	-10.3/126*
−0.320	-11	-17	-1.6	-2	-0.2	-1.3/110
−0.200	-14	+181	+7.8	+8		
−0.230	-14	+50	+6.3	+7		
−0.280	-14	+108	+4.0	+4	-1.0	+1.5/127
−0.320	-14	+81	+2.3	+3		

*Values shown for nominal void and Doppler coefficients are absolute peaks. Other peaks are relative to these.

Table 3.3.4.3-1

BWR-6 IORV - ATWS SENSITIVITY RESULTS SUMMARY

Sensitivity	Change in Maximum Neutron Flux From Base Value (% NBR)	Change in Maximum Heat Flux From Base Value (% NBR)	Change in Maximum Pressure (Vessel Bottom) From Base Value (psi)	Change in Minimum Water Level From Base Value (ft)	Change in Maximum Bulk Suppression Pool Temperature From Base Value (°F)	Change in Maximum Containment Pressure From Base Value (psi)
IORV - Base Case	101%	101%	1064	2.98 @ 63 Min	170° @ 60 Min	7.33
IORV - Boron Delay Nominal - 5 Minutes	0	0	0	0	-5 @ 58 Min	-0.78
IORV - Boron Delay Nominal +5 Minutes	0	0	0	0 @ 62 Min	+5	+0.85
IORV - 129 GPM Boron Capacity (150% Nom)	0	0	0	0	-4 @ 60 Min	-0.63
IORV - 43 GPM Boron Capacity (50% Nom)	0	0	0	5.35 @ 1870 Sec	+14 @ 60 Min	+2.57
IORV - 50% Nominal RHR Capacity (1212 Btu/Sec/°F Nominal)	0	0	0	0	+16 @ 88 Min	+3
IORV - 150% Nominal RHR Capacity	0	0	0	0	-10 @ 47 Min	-1.51
IORV - 5 Minutes RHR Delay	0	0	0	0	+1 @ 60 Min	+0.2
IORV - 10 Minutes RHR Delay	0	0	0	0	+2 @ 60 Min	+0.4
IORV - 65°F Service Water (85°F Nominal)	0	0	0	0	-7 @ 55 Min	-1
IORV - 105°F Service Water	0	0	0	0	+7 @ 67 Min	+1.2
65°F Service Water 50% Nominal RHR Capacity	101%	101%	1064	0	11 @ 77 Min	+2
105°F Service Water 50% Nominal RHR Capacity	0	0	0	0	-4 @ 55 Min	-0.2
80% Nominal Pool Size	0	0	0	0	+9 @ 55 Min	+1.6
120% Nominal Pool Size	0	0	0	0	-7 @ 65 Min	-1.1
80% SR Valve Capacity	0	0	0	-0.351 @ 40 Min	-10 @ 67 Min	-1.5
120% SR Valves Capacity	0	0	0	0	+9 @ 63 Min	+1.6

Table 3.3.4.4-1

EFFECT OF MULTIPLE (COMPENSATING) VARIATIONS ON PEAK POOL BULK TEMPERATURE

	Parameter (Base Value)	Variation (Final Value)	Pool Temperature Change	Reference
1.	Boron Timer Delay (2 Minutes)	+2 Min (4 Min)	+6°F	Figure 3.3.4.1.1-1
2.	Boron Efficiency (75%)	+27% (95%)	-8°F	Figure 3.3.4.1.2-1
3.	HPCS/RCIC Capacity (10.6% NBR FW)	-20% (8.48% NBR)	-9°F	Figure 3.3.4.1.3-1
4.	RHR Capacity (3.2% NBR*)	-50% (1.6% NBR*)	+7°F	Figure 3.3.4.1.4-1
5.	Doppler Coefficient (-0.28¢/°F)	+14% (-0.32¢/°F)	+3°F	Figure 3.3.4.1.13-2
		Net Results by Simple Summation of Individual Variations =	-1°F	
		Result Calculated for Simultaneous Variations =	-1°F	

*At 100°F ΔT

Table 3.3.4.4-2

EFFECT OF MULTIPLE (NON-COMPENSATING) VARIATIONS ON PEAK POOL BULK TEMPERATURE

	Parameter (Base Value)	Variation (Final Value)	Pool Temperature Change	Reference
1.	Boron Timer Delay (2 Minutes)	+2 Min (4 Min)	+6°F	Figure 3.3.4.1.1.-1
2.	Boron Efficiency (75%)	-27% (55%)	+17°F	Figure 3.3.4.1.2-1
3.	HPCS/RCIC Capacity Capacity (10.6% NBR FW)	+20% (12.7% NBR)	+7°F	Figure 3.3.4.1.3-1
4.	RHR Capacity (3.2% NBR*)	-50% (1.6% NBR*)	+7°F	Figure 3.3.4.1.4-1
5.	Doppler Coefficient (-0.28¢/°F)	+14% (-0.32¢/°F)	+3°F	Figure 3.3.4.1.3-2
		Net Result by Simple Summation of Individual Variations =	+40°F	
		Result Calculated for Simultaneous Variations =	+48°F	

*At 100°F ΔT

Table 3.3.5-1

BWR/6 LOSS OF CONDENSER VACUUM

	Sequence of Events	Time: With ARI	Time: With ARI Failure
1.	Main Turbine (and Feedwater Turbines) Trip Due to Low Condenser Vacuum, Bypass Opens - All Normal Scrams Fail.	0	0
2.	Pressure and Power Rise Begins	0	0
3.	Peak Power Occurs	1 Second	1 Second
4.	Relief Valves Lift	2 Seconds	2 Seconds
5.	ATWS High Pressure Setpoint (1150 psig) is Reached -Recirculation Pumps Tripped -ARI is Initiated -SLCS Timed Logic Activated	2 Seconds	2 Seconds
6.	Same Fuel May Experience Boiling Transition	2 Seconds	2 Seconds
7.	Peak Vessel Pressure Occurs	3 Seconds	3 Seconds
8.	ARI Control Rod Insertion, Eliminating SLCS and Feedwater Limiting Actions	22 Seconds	Fails
9.	ATWS Logic Initiates Feedwater Flow Limit	N/A	27 Seconds
10.	MSIV's and Bypass Close Due to Low Condenser Vacuum	30 Seconds	30 Seconds
11.	Reactor Water Level Drops to Level 2 - Initiates Containment Isolation - HPCS and RCIC Start		53 Seconds

Table 3.3-1 (Continued)

	Sequence of Events	With ARI	With ARI Failure
12.	HPCS and RCIC Flow Begins		73 Seconds
13.	Reactor Water Level Drops to Level 1	N/A	85 Seconds
14.	Reactor Water Level Reaches Minimum and Begins to Rise		107 Seconds
15.	ATWS Logic Initiates SLCS	N/A	2 Minutes
16.	Liquid Control Flow Reaches Core	N/A	3 Minutes
17.	RHR Flow Begins (Pool Cooling)	$\geq$11 Minutes	11 Minutes
18.	Hot Shutdown Achieved	22 Seconds	20 Minutes
19.	Containment Temperature and Pressure Peaks Occur		27 Minutes

Table 3.3.5-2

BWR/6 LOSS OF CONDENSER VACUUM - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Loss of Condenser Vacuum
Maximum Neutron Flux (%)	367
Maximum Vessel Bottom Pressure (psig)	1235
Maximum Average Heat Flux (%)	135
Maximum Bulk Suppression Pool Temperature (°F)	163
Associated Containment Pressure (psig)	6.3

Table 3.3.6-1

BWR/6 PRESSURE REGULATOR FAILURE - ZERO STEAM DEMAND

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Normal pressure regulator fails to closed position, backup regulator also assumed to fail eliminating opening turbine bypass valves.	0	0
2.	Turbine control valves start to close - All normal scrams fail	0	0
3.	Pressure and power rise begins	0	0
4.	Relief valves lift	2 Seconds	2 Seconds
5.	ATWS High pressure setpoint (1150 psig) is reached -Recirculation pumps tripped -ARI initiated -SLCS times logic activated	2 Seconds	2 Seconds
6.	Some fuel experiences transition boiling	2 Seconds	2 Seconds
7.	Peak Vessel pressure occurs	5 Seconds	5 Seconds
8.	ARI control rod insertion completed, eliminating SLCS and feedwater limiting actions	22 Seconds	Fails
9.	ATWS logic Initiates feedwater flow limit*	N/A	27 Seconds
10.	Reactor water level drops to Level 2 -Initiates containment isolation -HPCS and RCIC initiated		53 Seconds

*Assumes motor-driven feedwater pumps.

Table 3.3.6-1 (Continued)

	Sequence of Events	With ARI	With ARI Failure
11.	HPCS and RCIC flow begins		73 Seconds
12.	Reactor water level drops to Level 1		101 Seconds
13.	ATWS logic -Initiates SLCS	N/A	2 Minutes
14.	Reactor water level reaches minimum and begins to rise		155 Seconds
15.	Liquid control flow reaches core	N/A	3 Minutes
16.	RHR flow begins (pool cooling)	≥11 Minutes	11 Minutes
17.	Hot shutdown achieved	22 Seconds	20 Minutes
18.	Containment temperature and pressure peak		33 Minutes

Table 3.3.6-2

BWR/6 PRESSURE REGULATOR FAILURE (ZERO STEAM DEMAND) - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Pressure Regulator Failure (Closed)
Maximum Neutron Flux (%)	404
Maximum Vessel Bottom Pressure (psig)	1283
Maximum Average Heat Flux (%)	143
Maximum Bulk Suppression Pool Temperature (°F)	167
Associated Containment Pressure (psig)	6.9

Table 3.3.7-1

BWR/6 LOSS OF A FEEDWATER HEATER

	Sequence of Events	Time
1.	Inadvertent tripping of feedwater heater; feedwater enthalpy begins to drop	0 Seconds
2.	Reactor and turbine-generator power begins to rise	2 Seconds
3.	APRM high power alarm (108%), operator attempts to insert rods	30 Seconds
4.	Vessel pressure levels off after a small increase	74 Seconds
5.	Power levels off below the scram setpoint(s)	124 Seconds
6.	Feedwater enthalpy change complete	138 Seconds
7.	Manual scram attempted after control rod insertion attempts have failed, manual scram fails, ARI and SLCS logic initiated	10-1/2 Minutes
8.	ARI control rod insertion completed, eliminating SLCS initiation, and achieving reactor shutdown	11 Minutes
9.	ATWS logic timer Initiates SLCS (if ARI has failed)	12-1/2 Minutes
10.	Liquid control reaches core (if ARI has failed)	14 Minutes
11.	Hot shutdown achieved (if ARI has failed)	35 Minutes

Table 3.3.7-2

BWR/6 LOSS OF FEEDWATER HEATER - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Loss of Feedwater Heater
Maximum Neutron Flux (%)	115
Maximum Vessel Bottom Pressure (psig)	1071
Maximum Average Heat Flux (%)	114
Maximum Bulk Suppression Pool Temperature (°F)	90
Associated Containment Pressure (psig)	no change

Table 3.3.8-2

BWR/6 FEEDWATER CONTROLLER FAILURE
(MAXIMUM DEMAND - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Feedwater Controller Failure
Maximum Neutron Flux (%)	396
Maximum Vessel Bottom Pressure (psig)	1214
Maximum Average Heat Flux (%)	126
Maximum Bulk Suppression Pool Temperature (°F)	95
Associated Containment Pressure (psig)	1

Table 3.3.9-1

BWR/6 PRESSURE REGULATOR FAILURE - MAXIMUM STEAM DEMAND

	Sequence of Events	Time: With ARI	Time: With ARI Failure
1.	Pressure regulator to maximum demand	0	0
2.	Pressure and power begin to decrease	0	0
3.	Low steamline pressure isolation setpoint reached -MSIV closure -Scram normally initiated (assumed to fail	16 Seconds 17 Seconds	16 Seconds 17 Seconds
4.	Pressure and power begin to rise	18 Seconds	18 Seconds
5.	Relief valves lift	23 Seconds	23 Seconds
6.	ATWS high pressure setpoint is reached (1150 psig) -Recirculation pumps are tripped -ARI is initiated -SLCS and feedwater limit timed logic is activated	23 Seconds	23 Seconds
7.	Vessel pressure peaks	26 Seconds	26 Seconds
8.	Some fuel experiences boiling transition	27 Seconds	27 Seconds
9.	ARI control rod insertion completed, eliminating SLCS initiation and feedwater limit	43 Seconds	Fails
10.	ATWS logic Initiates feedwater limit	N/A	48 Seconds
11.	Feedwater flow runs back to lower limit value	N/A	63 Seconds
12.	Reactor water level drops to Level 2 -Initiates containment isolation -Initiates HPCS and RCIC	59 Seconds	68 Seconds
13.	HPCS and RCIC flow begins	79 Seconds	88 Seconds
14.	Reactor water level reaches minimum and begins to rise	83 Seconds	137 Seconds
15.	ATWS logic timer Initiates SLCS	N/A	2-1/2 Minutes
16.	Liquid control flow reaches core	N/A	3-1/2 Minutes
17.	RHR flow begins (pool cooling)	$\geq$11 Minutes	11 Minutes
18.	Hot shutdown achieved	43 Seconds	20 Minutes
19.	Containment temperature and pressure peak	N/A	30 Minutes

Table 3.3.9-2

BWR/6 PRESSURE REGULATOR FAILURE
(MAXIMUM STEAM DEMAND) - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Pressure Regulator Failure (Open)
Maximum Neutron Flux (%)	590
Maximum Vessel Bottom Pressure (psig)	1296
Maximum Average Heat Flux (%)	153
Maximum Bulk Suppression Pool Temperature (°F)	167
Associated Containment Pressure (psig)	6.9

Table 3.3.10-1

BWR/6 LOSS OF FEEDWATER

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Feedwater flow stops (flow assumed to reduce to zero in 5 seconds) - all normal scram fail	0	0
2.	Pressure, water level and power starts to decline	0	0
3.	Reactor water level drops to Level 2 and trips recirculation pumps*, initiates ARI and also initiates RCIC and HPCS. SLCS timed logic is also activated.	13 Seconds	13 Seconds
4.	ARI control rod insertion completed. Eliminating SLCS initiation	33 Seconds	Fails
5.	HPCS and RCIC flow starts	33 Seconds	33 Seconds
6.	Water level reaches Level 1. MSIV closure initiated	45 Seconds	N/A
7.	Water level reaches minimum and begins to rise. The top of the core always remains covered.	47 Seconds	71 Seconds
8.	ATWS logic timer initiates SLCS	N/A	2 Minutes
9.	Liquid control flow reaches the core	N/A	3 Minutes
10.	High water level trip of HPCS and RCIC (neglecting preferred operator action to manually control level)	10 Minutes	16 Minutes
11.	Hot shutdown achieved	32 Seconds	19 Minutes
12.	MSIV closure on low pressure	N/A	19 Minutes

*Recirculation runback (from low level alarm, L4, and coincident low FW flow) is conservatively neglected.

Table 3.3.10-2

BWR/6 LOSS OF FEEDWATER - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Loss of Feedwater
Maximum Neutron Flux (%)	100
Maximum Vessel Bottom Pressure (psig)	1061
Maximum Average Heat Flux (%)	100
Maximum Bulk Suppression Pool Temperature (°F)	90
Associated Containment Pressure (psig)	No change

Table 3.3.11-1

BWR/6 LOSS OF NORMAL AC POWER

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Loss of all auxiliary transformers -Recirculation pumps trip -Condensate and feedwater pumps trip	0	0
2.	Pressure and power begin to fall	0	0
3.	Normal scram due to loss of AC (Assumed to fail)	2 Seconds	2 Seconds
4.	MSIV's start to close due to loss of AC power (and initiated scram - also assumed to fail)	2 Seconds	2 Seconds
5.	Pressure and power begin to rise	6 Seconds	6 Seconds
6.	S/RV valves lift at relief setpoints	7 Seconds	7 Seconds
7.	ATWS high pressure setpoint is reached (1150 psig) -ARI is initiated -SLCS timed logic is activated	7 Seconds	7 Seconds
8.	Vessel pressure and power peak	7 Seconds	7 Seconds
9.	Some fuel experiences boiling transition	8 Seconds	8 Seconds
10.	Reactor water level drops to Level 2 -Initiates containment isolation -Initiates HPCS and RCIC	20 Seconds	20 Seconds
11.	ARI control rod insertion completed, eliminating SLCS initiation	27 Seconds	Fails
12.	HPCS and RCIC flow begins	40 Seconds	40 Seconds
13.	Lowest relief setpoint SRV closes and the S/RV's are assumed to switch to spring setpoints		66 Seconds

Table 3.3.11-1 (Conti...

Sequence of Events

14. Reactor water level drops to Level 1

15. Reactor water level reaches minimum an begins to rise. Level inside the core shroud remains above the top of active fuel.

16. ATWS logic timer Initiates SLCS

17. Liquid control flow reaches core

18. RHR flow begins (pool cooling)

19. Hot shutdown achieved

20. Containment temperature and pressure peak

Table 3.3.11-2

BWR/6 LOSS OF NORMAL AC POWER - SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Loss of Normal AC Power
Maximum Neutron Flux (%)	546
Maximum Vessel Bottom Pressure (psig)	1218
Maximum Average Heat Flux (%)	101
Maximum Bulk Suppression Pool Temperature (°F)	150
Associated Containment Pressure (psig)	4.5

Table 3.3.12-1

BWR/6 RECIRCULATION FLOW CONTROLLER FAILURE
(MAXIMUM DEMAND)

	Sequence of Events	Time
1.	One recirculation flow controller fails with maximum valve starting rate	0
2.	Power and flow begin to rise	0
3.	Neutron flux reaches 120%, APRM scram assumed to fail	2 Seconds
4.	Power peaks	2 Seconds
5.	Flow control valve reaches full open position	3 Seconds
6.	Maximum average fuel surface heat flux occurs	4 Seconds
7.	Vessel Pressure peaks	5 Seconds
8.	Core flow increase stabilizes	10 Seconds
9.	Manual scram or (if failed) ARI initiation and (if failed) SLCS logic initiation	10 Minutes
10.	Hot shutdown achieved	30 Minutes

Table 3.3.12-2

BWR/6 RECIRCULATION FLOW CONTROLLER FAILURE (MAXIMUM DEMAND) – SUMMARY

With ARI Failure	86 GPM - 2 Min Logic Delay Recirculation Flow Controller Failure
Maximum Neutron Flux (%)	247
Maximum Vessel Bottom Pressure (psig)	1013
Maximum Average Heat Flux (%)	88
Maximum Bulk Suppression Pool Temperature (°T)	90
Associated Containment Pressure (psig)	No change

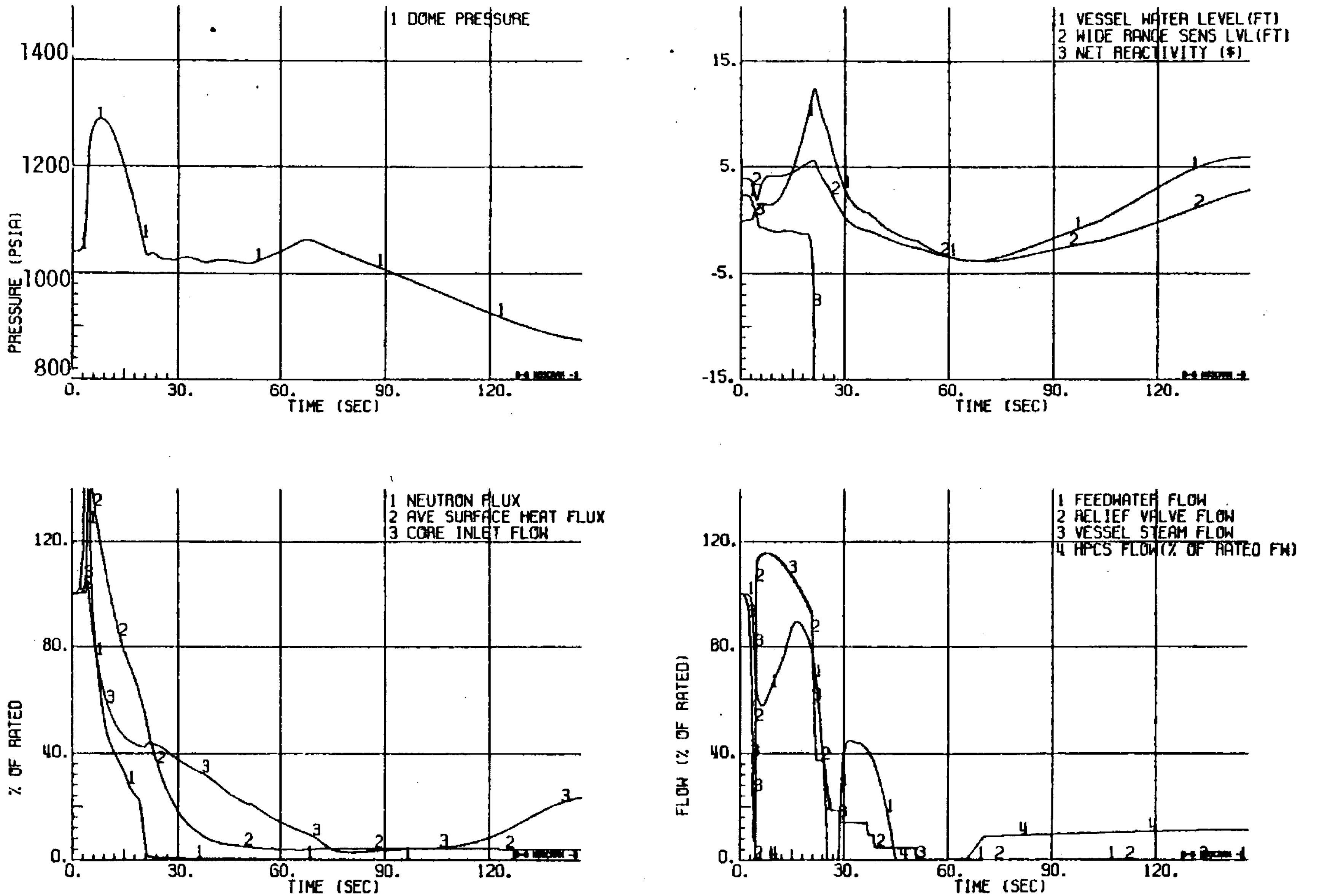


Figure 3.3.1-1. BWR/6 MSIV Closure, ARI

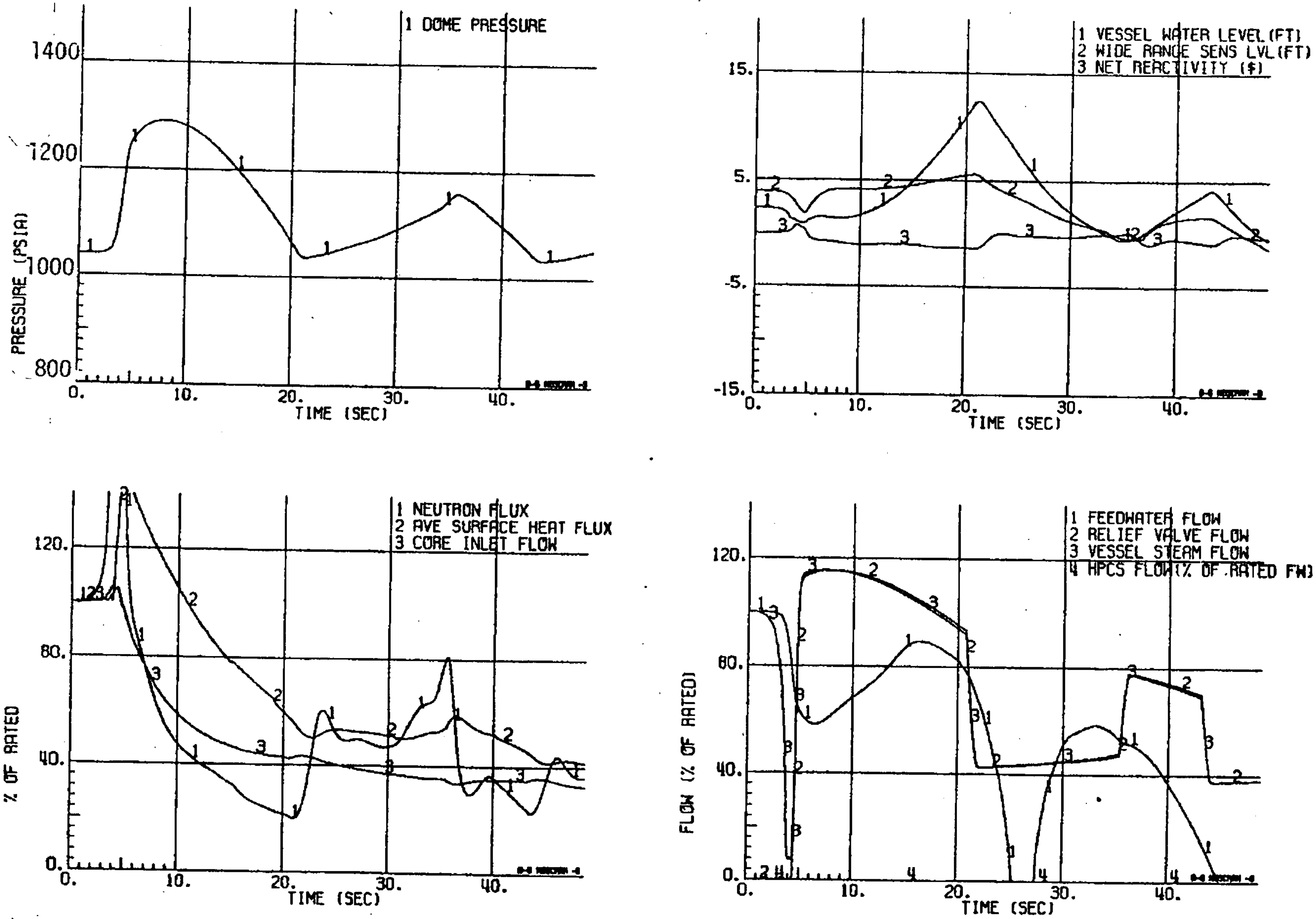


Figure 3.3.1-2. BWR/6 MSIV Closure, ARI

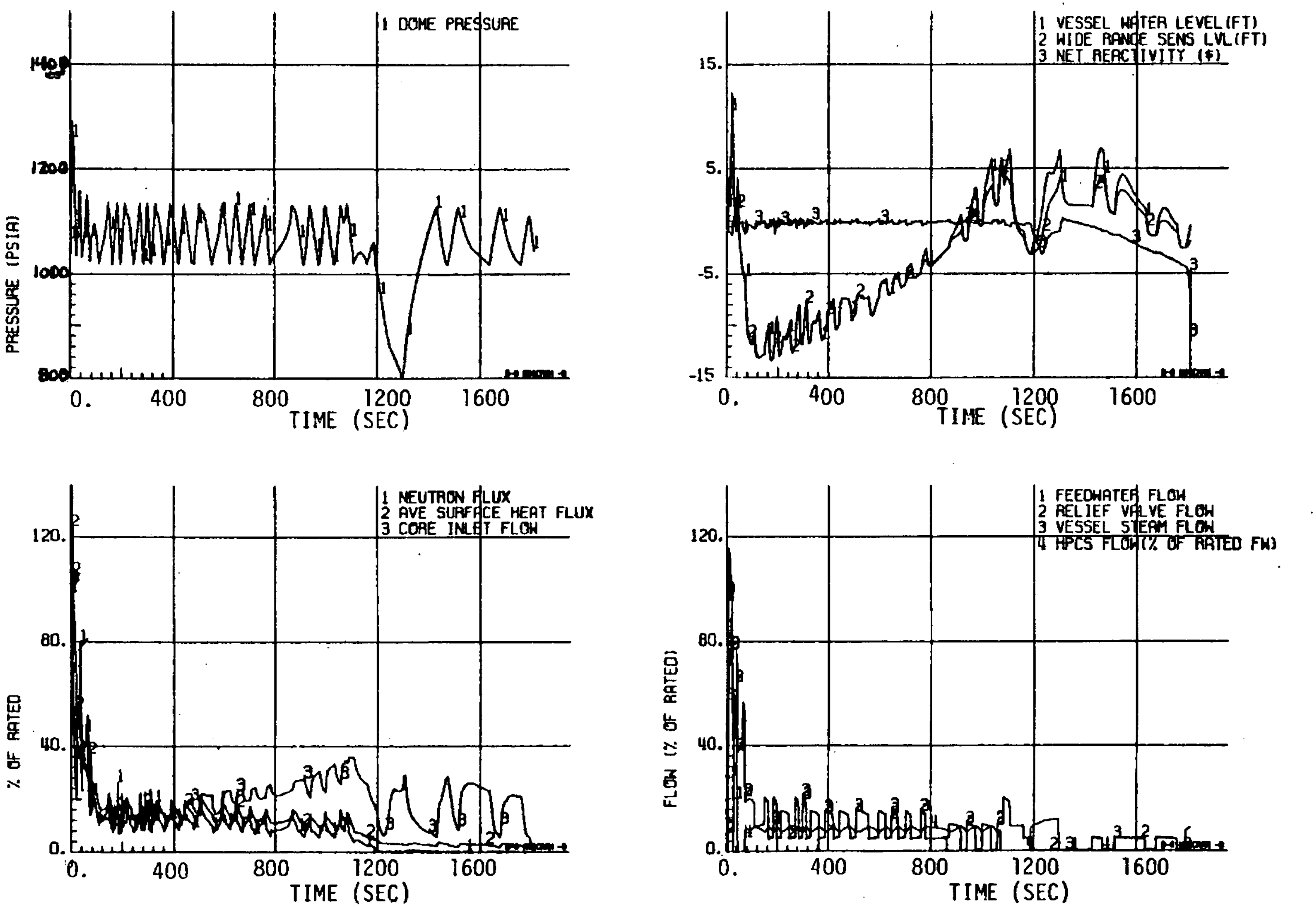


Figure 3.3.1-3. BWR/6 MSIV Closure, ARI Failure

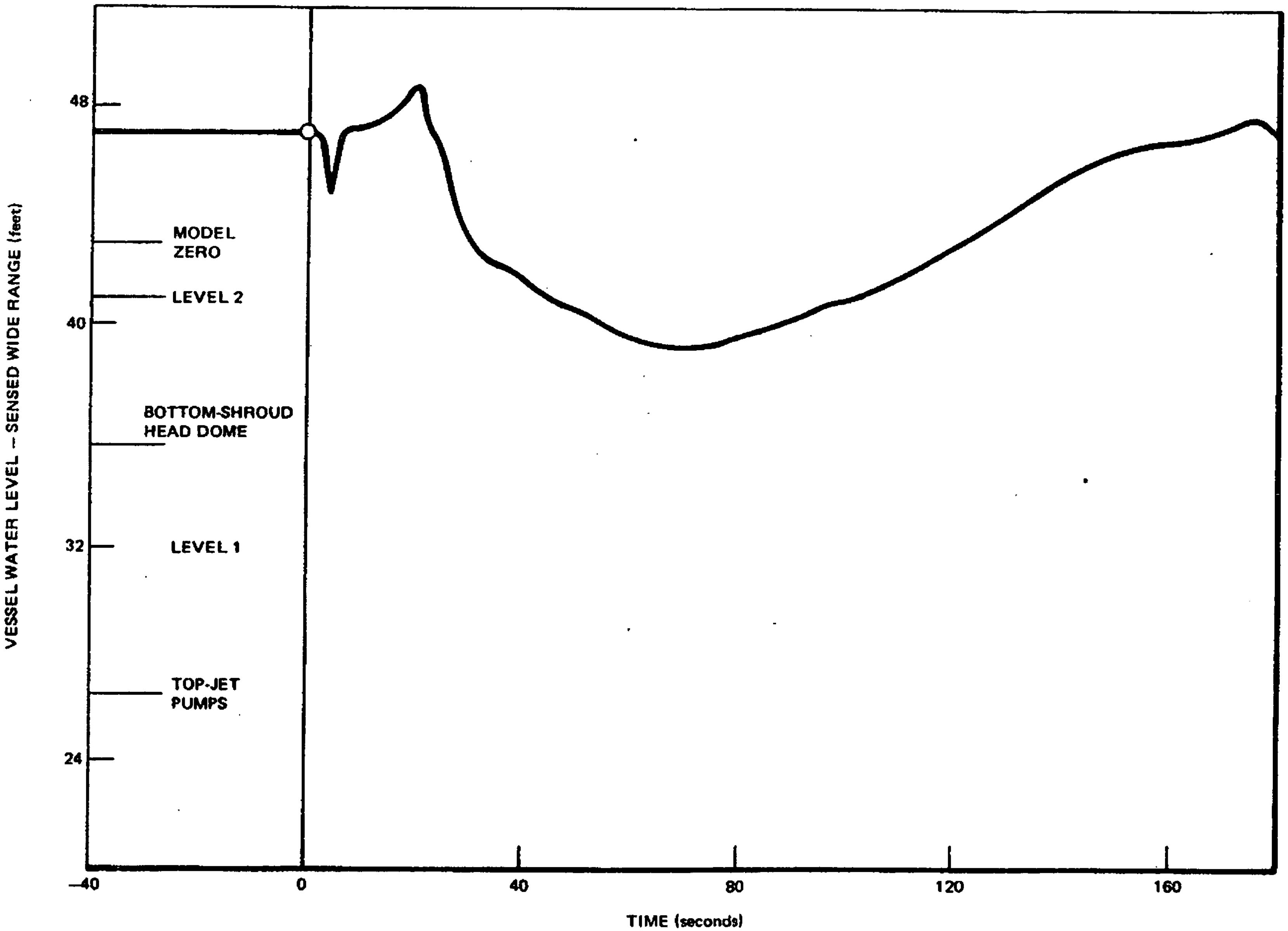


Figure 3.3.1-4. BWR-6 ATWS With ARI MSIV Base Case: Behavior of Vessel Water Level Over Time

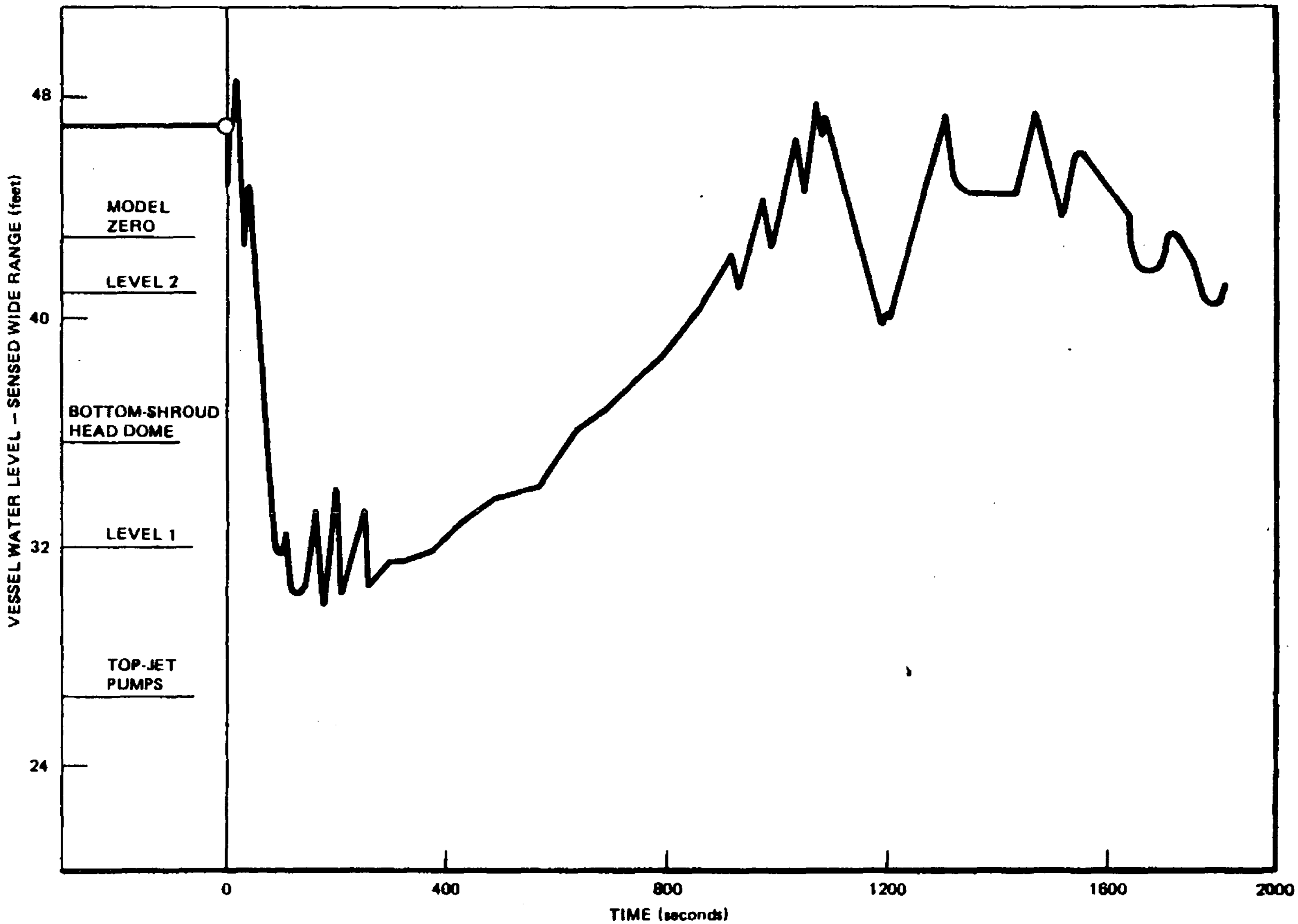


Figure 3.3.1-5. BWR-6 ATWS With ARI Failure MSIV Closure: Behavior of Vessel Water Level Over Time

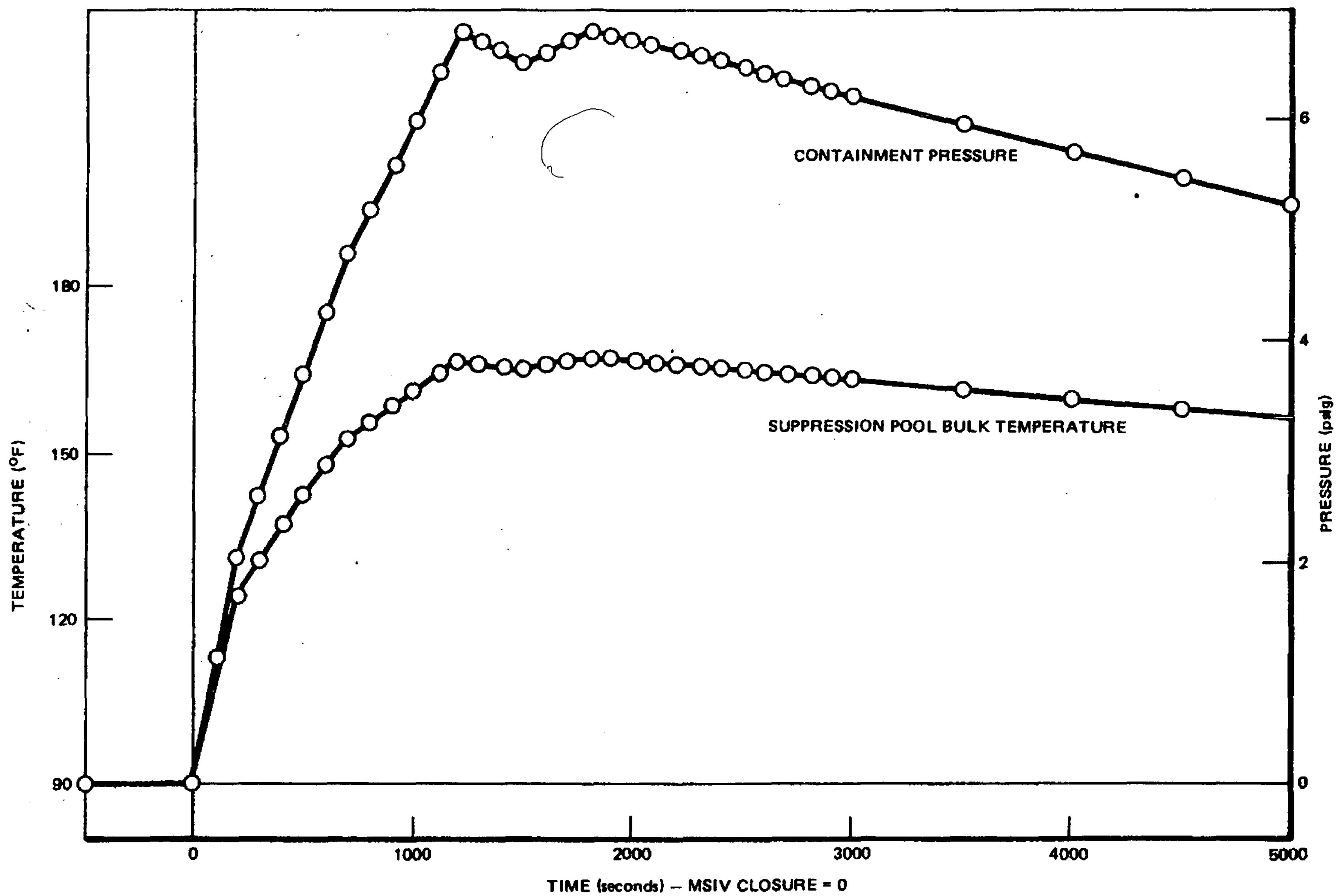


Figure 3.3.1-6. BWR-6 ATWS With ARI Failure MSIV Base Case: Behavior of Suppression Pool Temperature and Containment Pressure Over Time

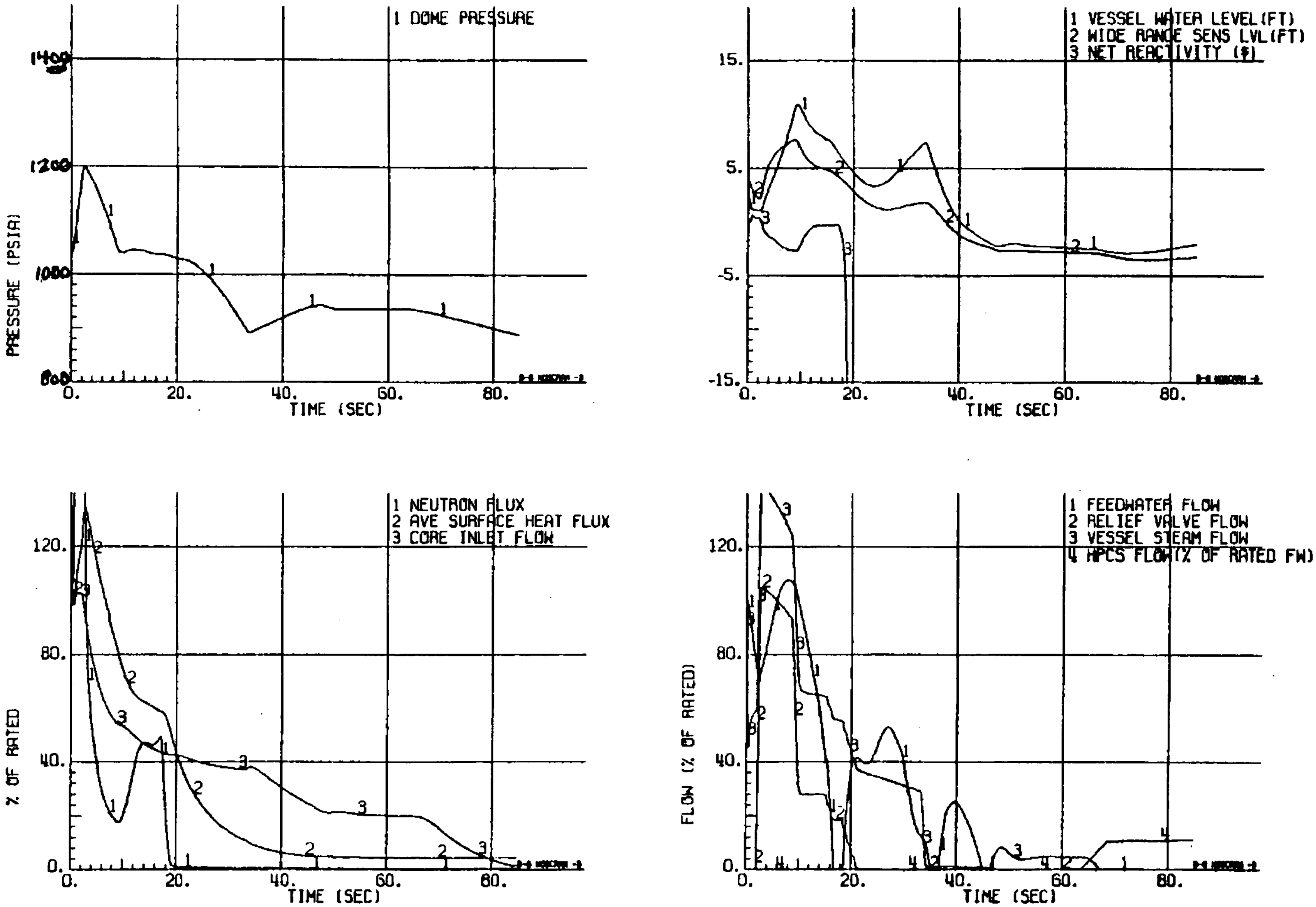


Figure 3.3.2-1. BWR/6 Turbine Trip, ARI Pump Trip

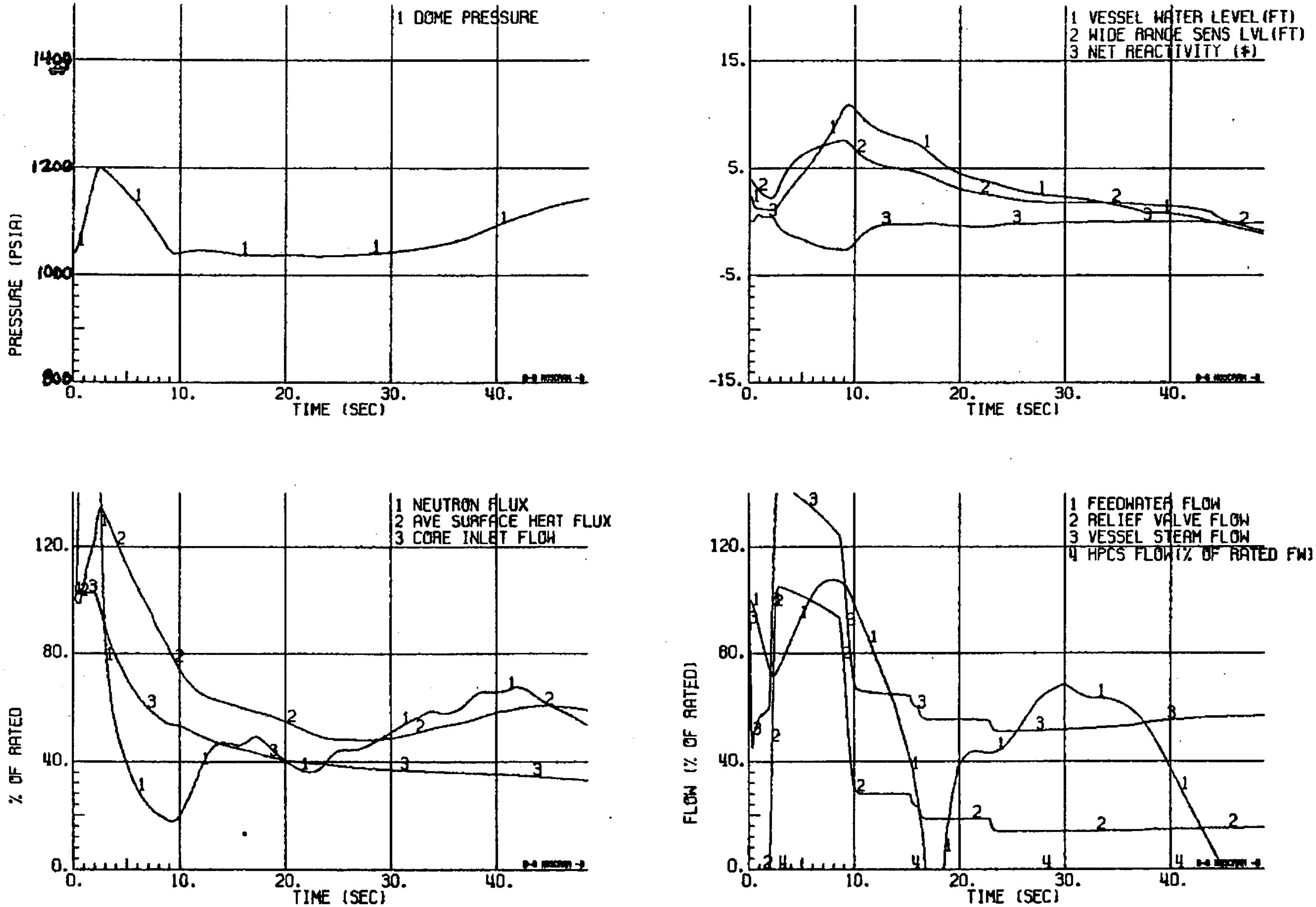


Figure 3.3.2-2. BWR/6 Turbine Trip, ARI Failure

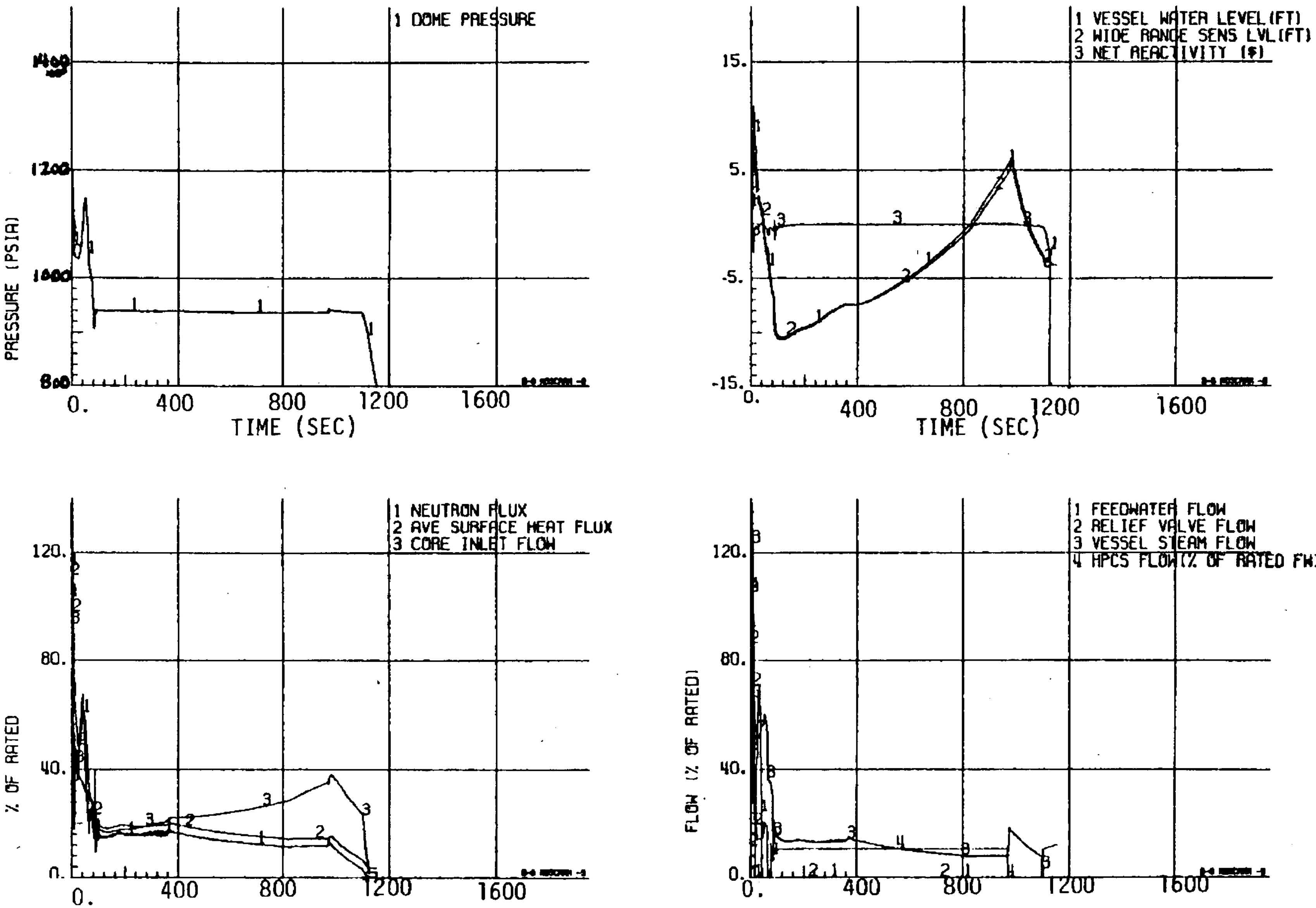


Figure 3.3.2-3. BWR/6 Turbine Trip, ARI Failure

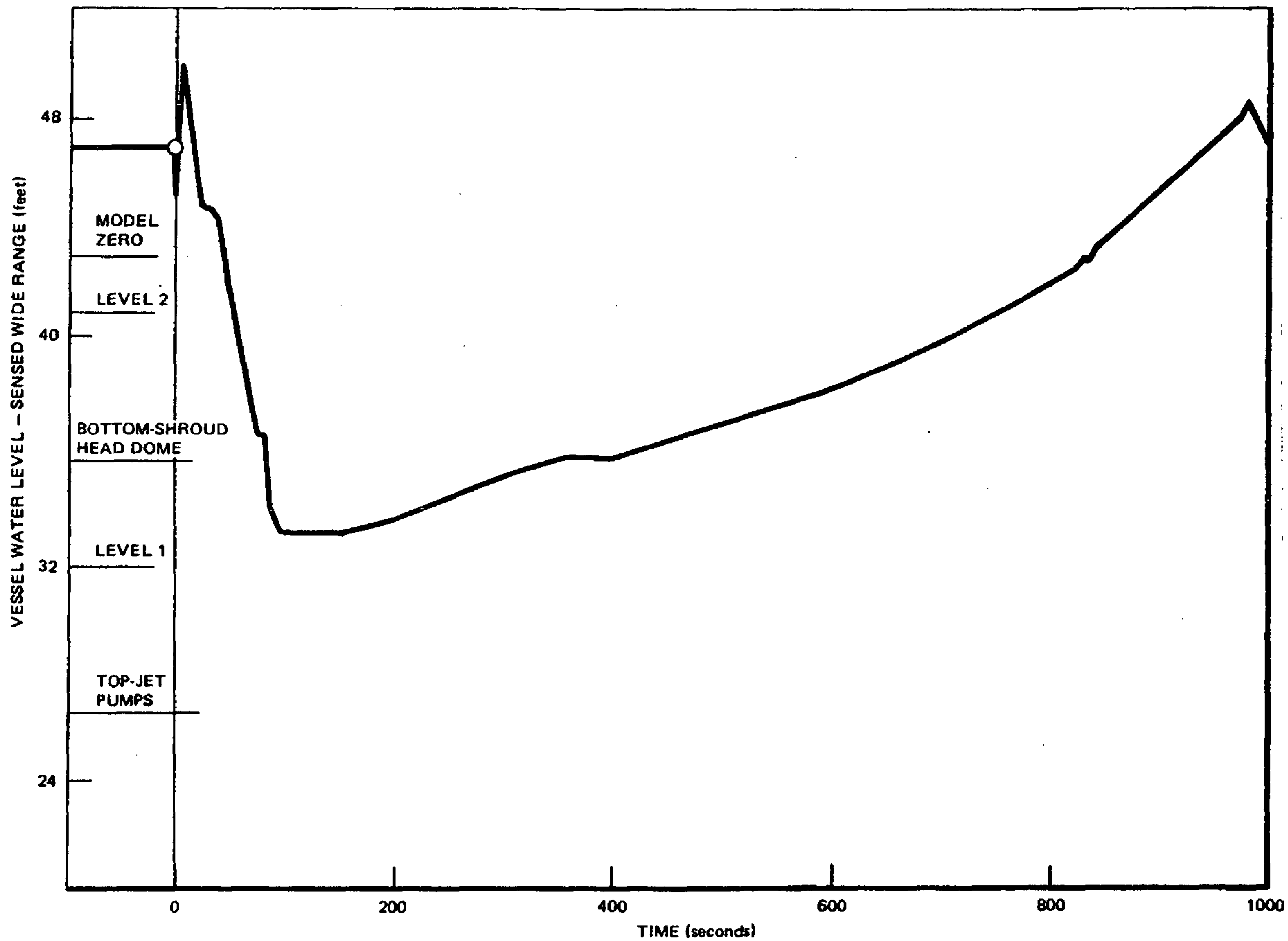


Figure 3.3.2-4. BWR-6 ATWS With ARI Failure Turbine Trip With Bypass Base Case: Behavior of Vessel Water Level Over Time

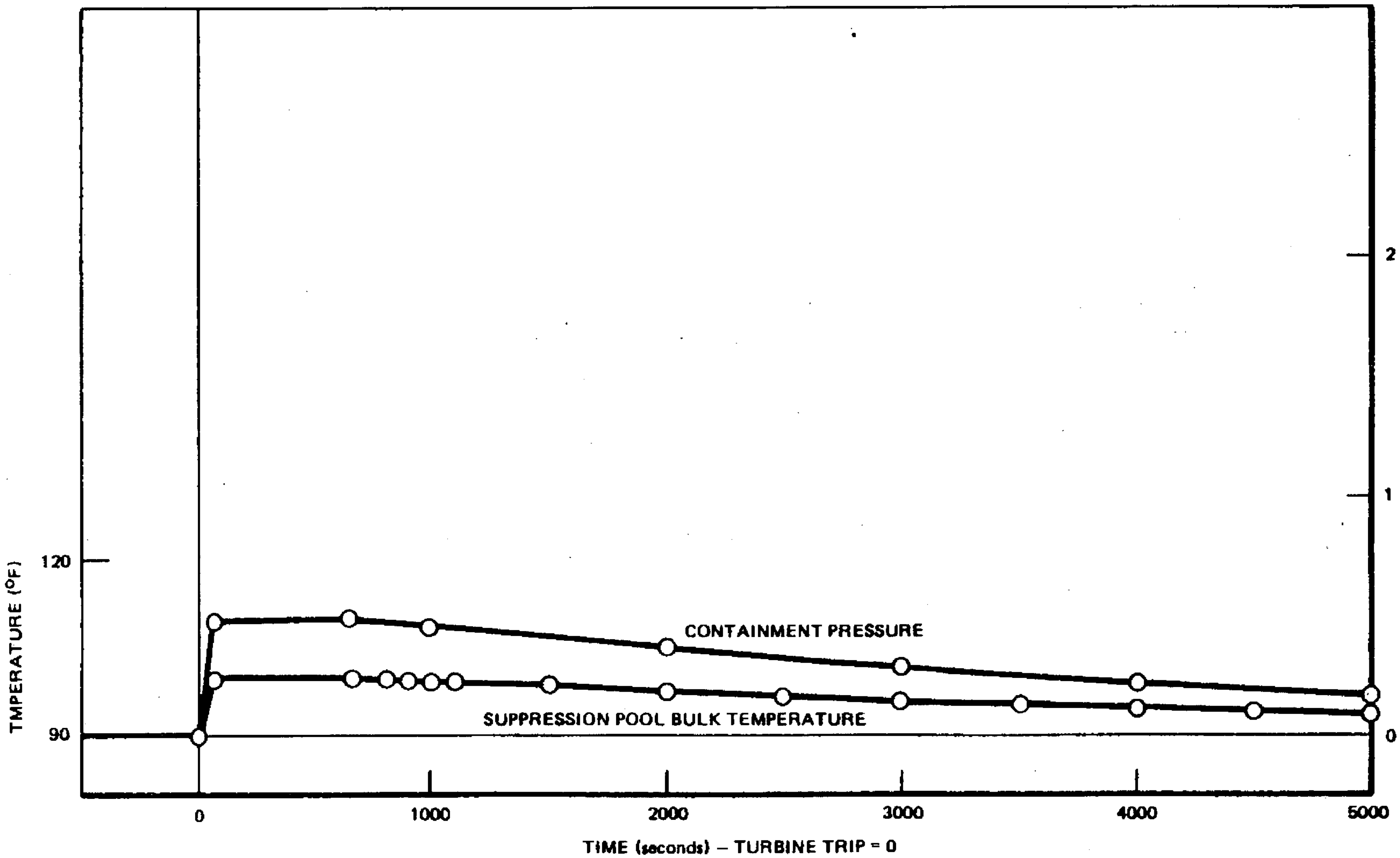


Figure 3.3.2-5. BWR-6 ATWS With ARI Failure Turbine Trip Base Case: Behavior of Suppression Pool Temperature and Containment Pressure Over Time

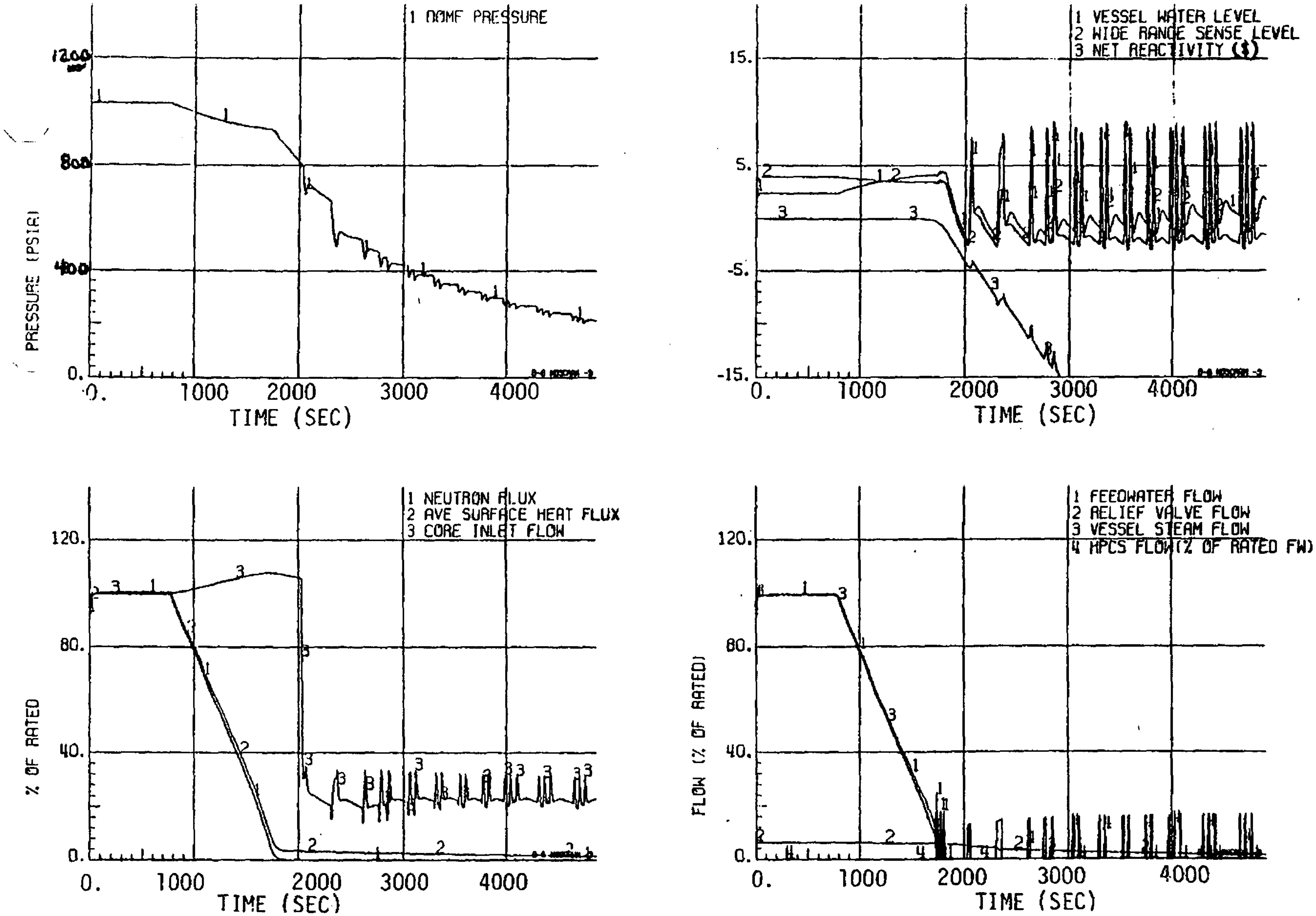


Figure 3.3.3-1. BWR/6 Inadvertent Open Relief Valve, ARI Failure

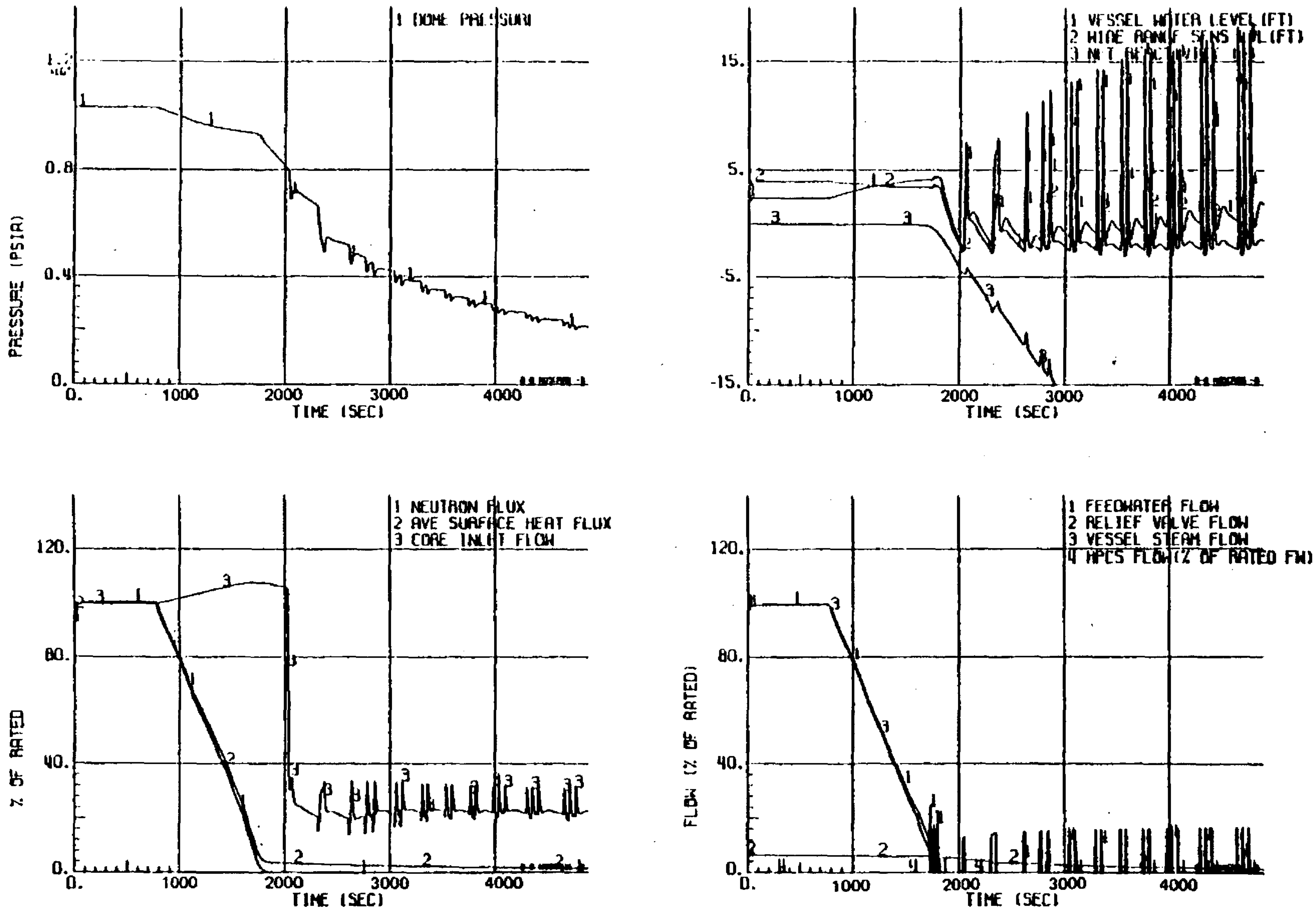


Figure 3.3.3-2. Inadvertent Open Relief Valve with ARI

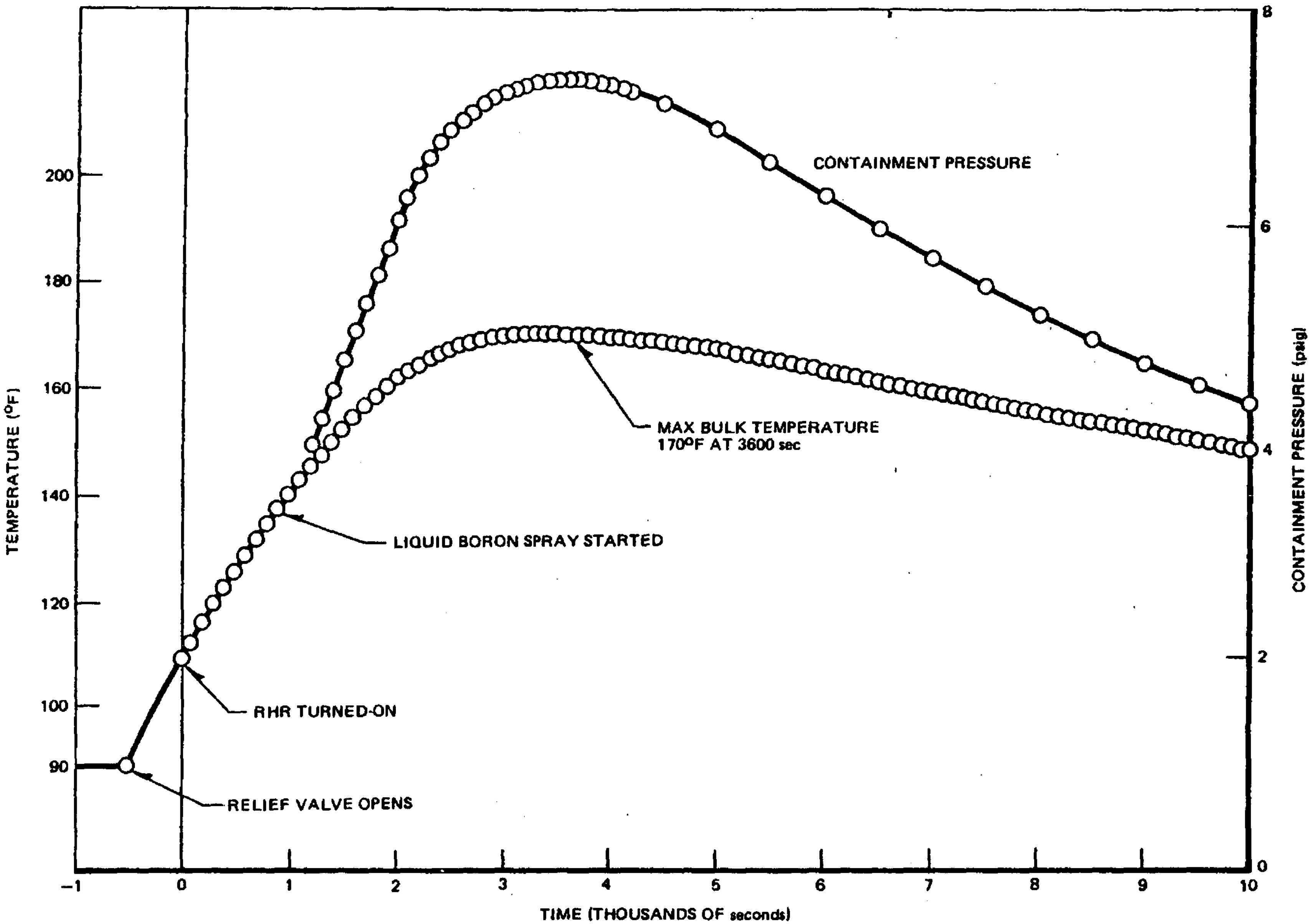


Figure 3.3.3-3. BWR-6 ATWS With ARI Failure IORV Base Case: Behavior of Suppression Pool Temperature and Containment Pressure Over Time

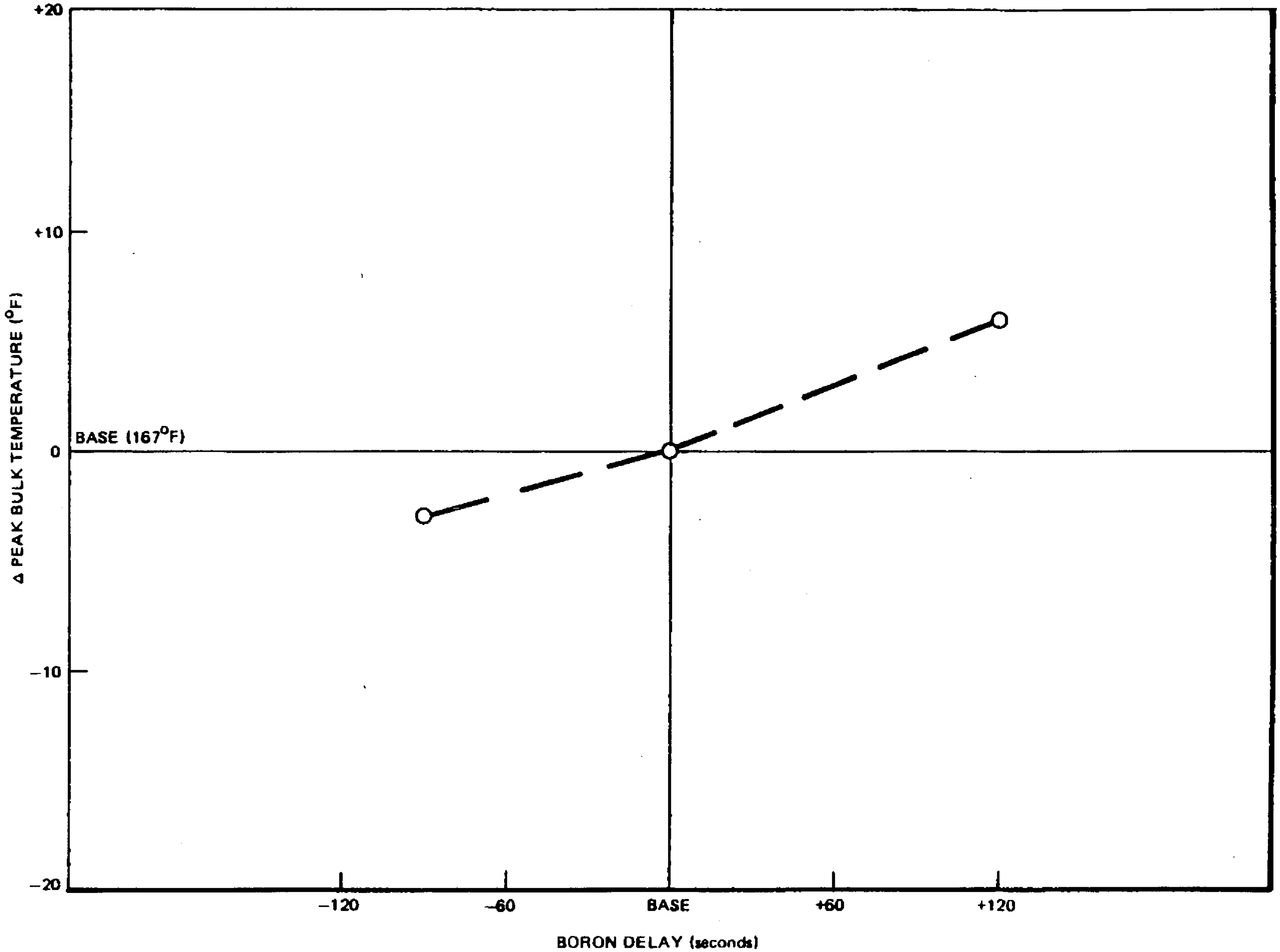


Figure 3.3.4.1.1-1. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to Delay of Liquid Boron Injection

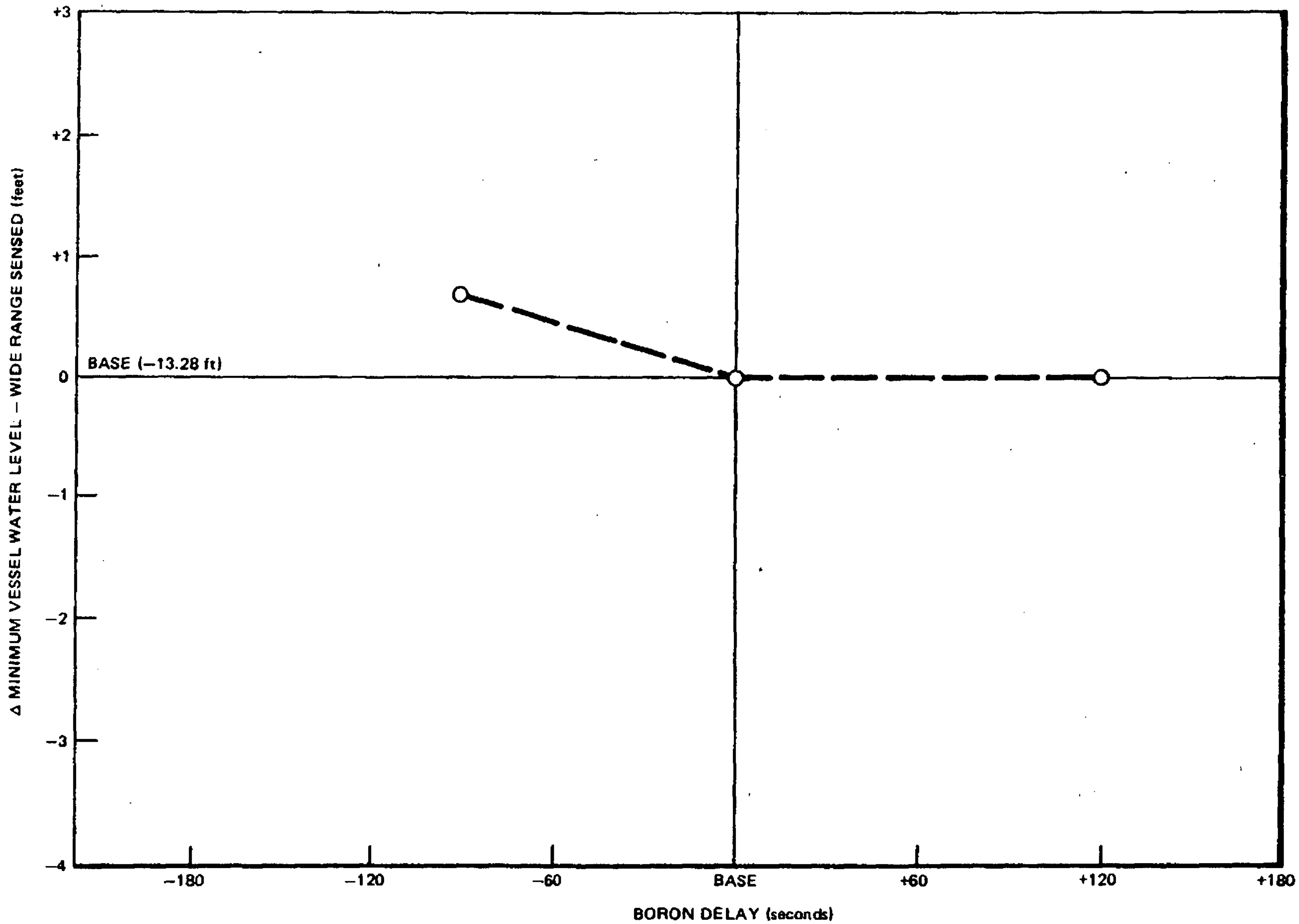


Figure 3.3.4.1.1-2. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Minimum Vessel Water Level to Delay of Liquid Boron

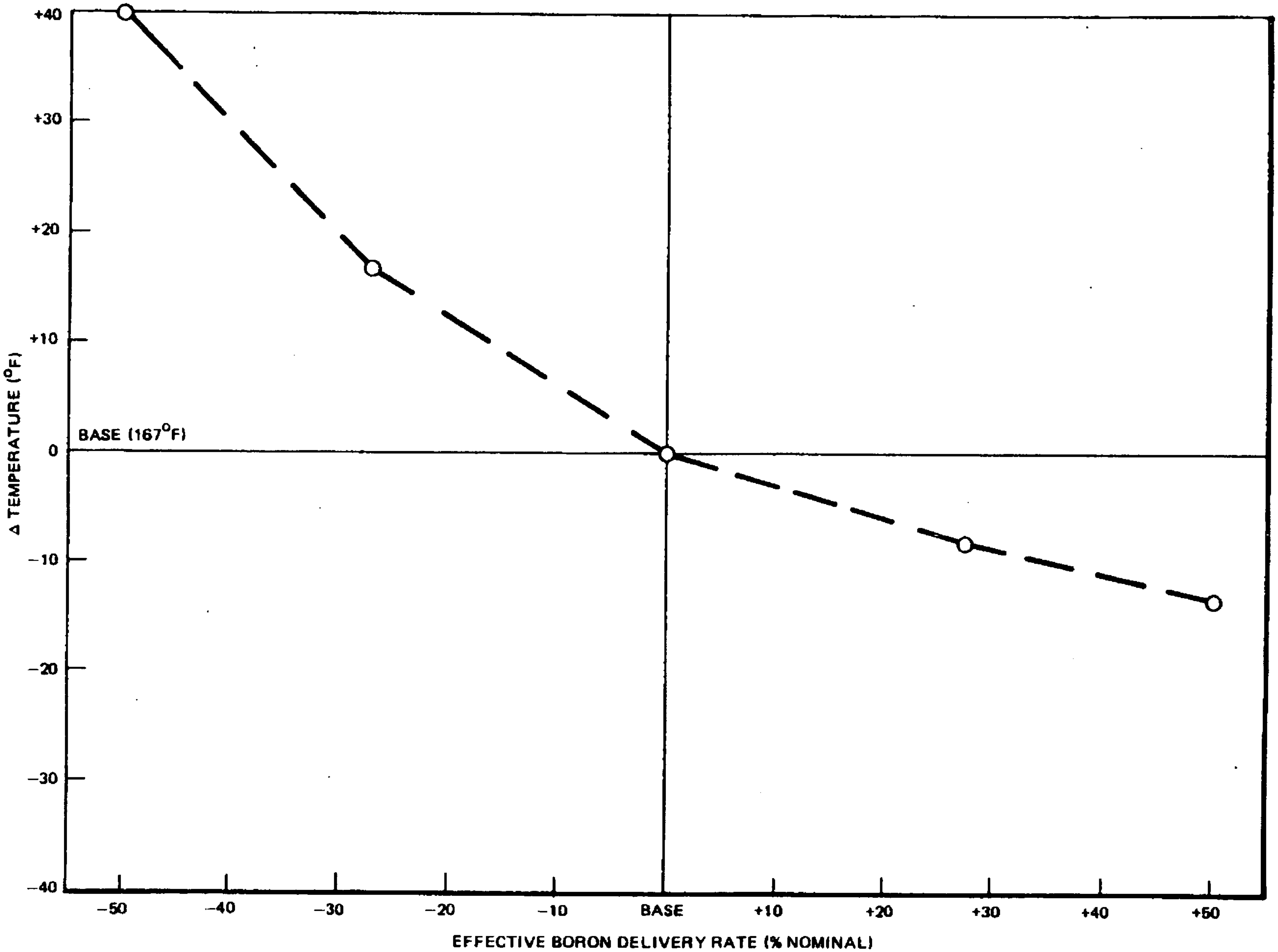


Figure 3.3.4.1.2. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to Boron Pumping Capacity and Mixing Efficiency

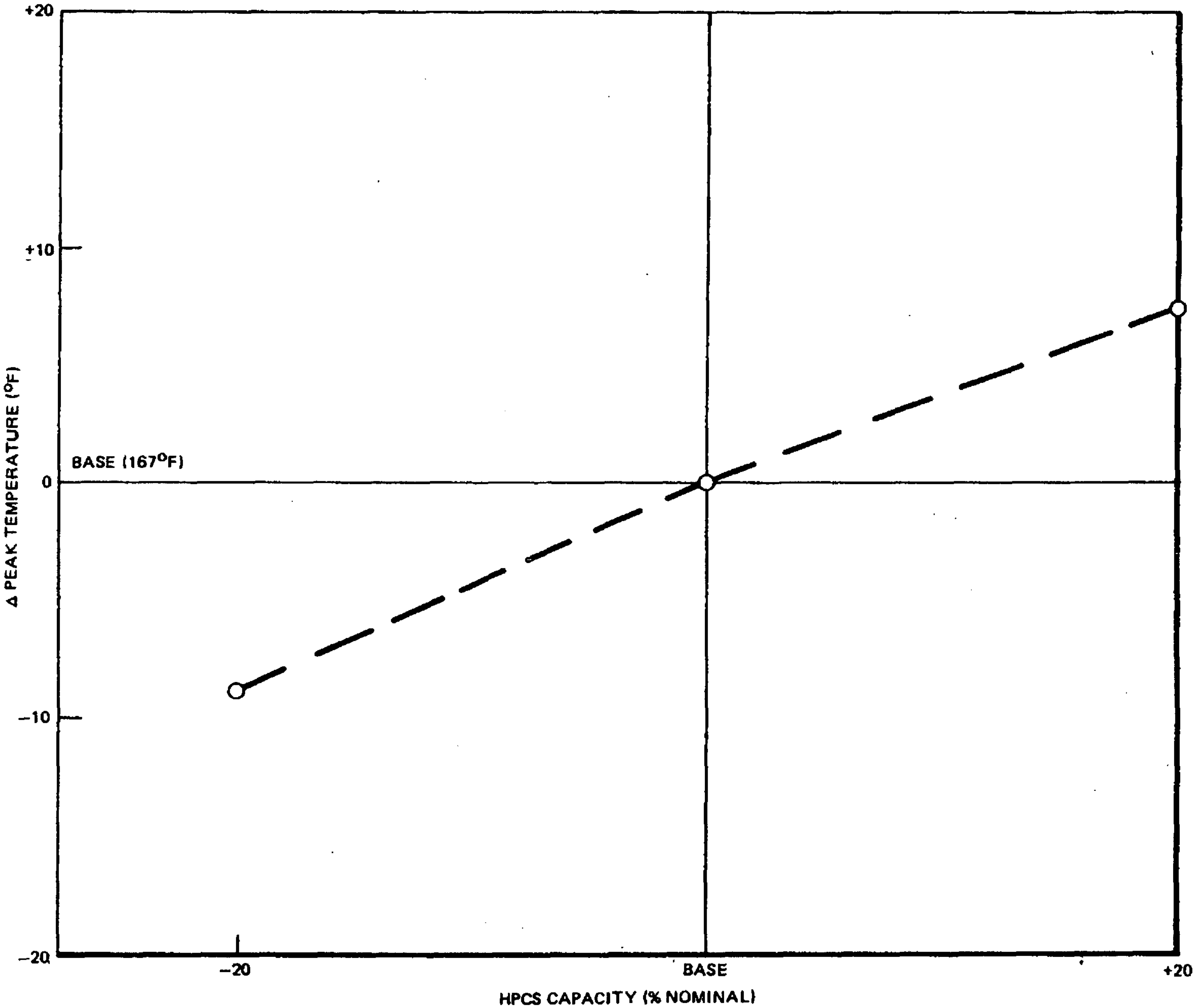


Figure 3.3.4.1.3-1. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to HPCS Capacity

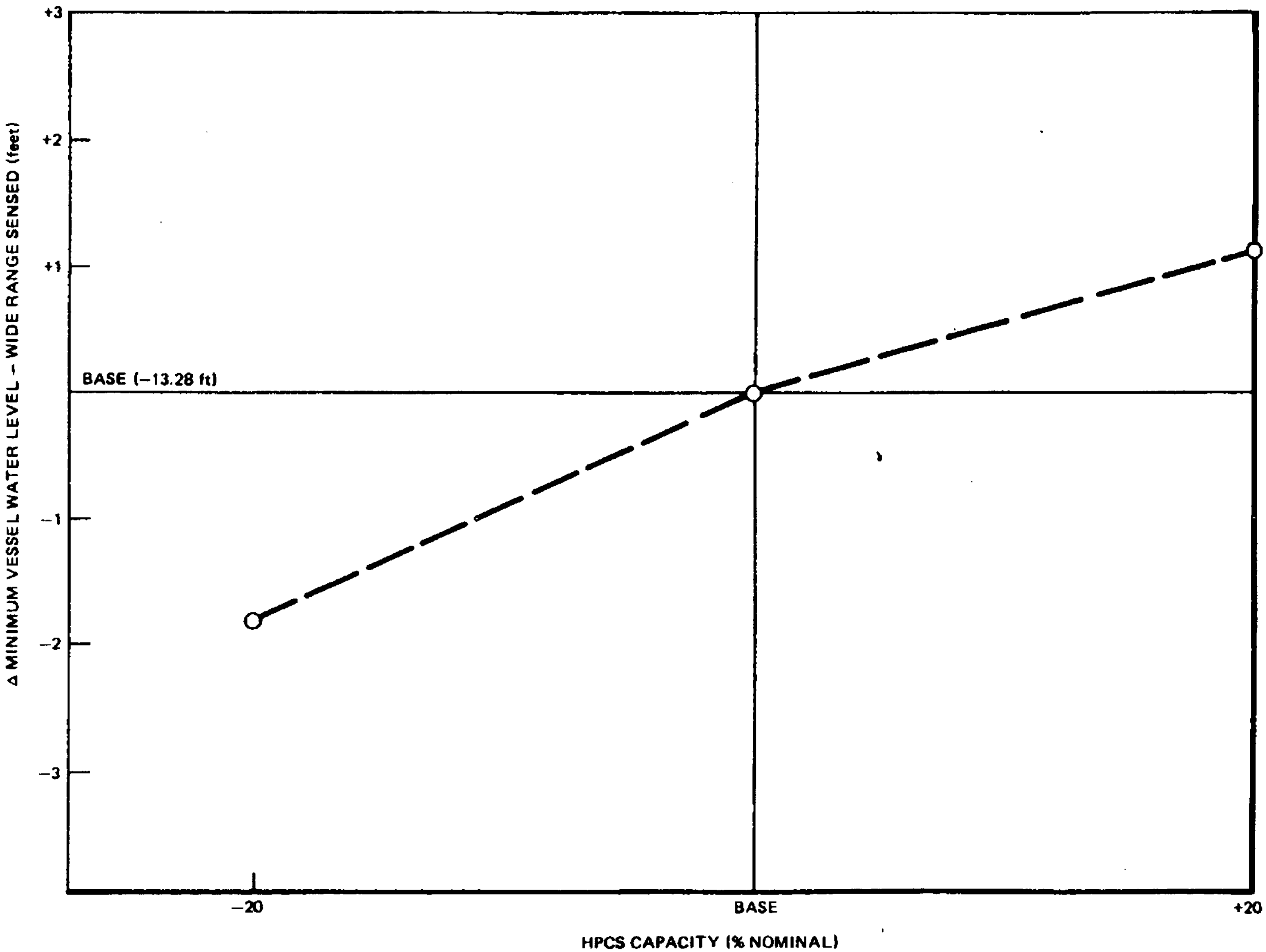


Figure 3.3.4.1.3-2. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Minimum Vessel Water Level to HPCS Capacity

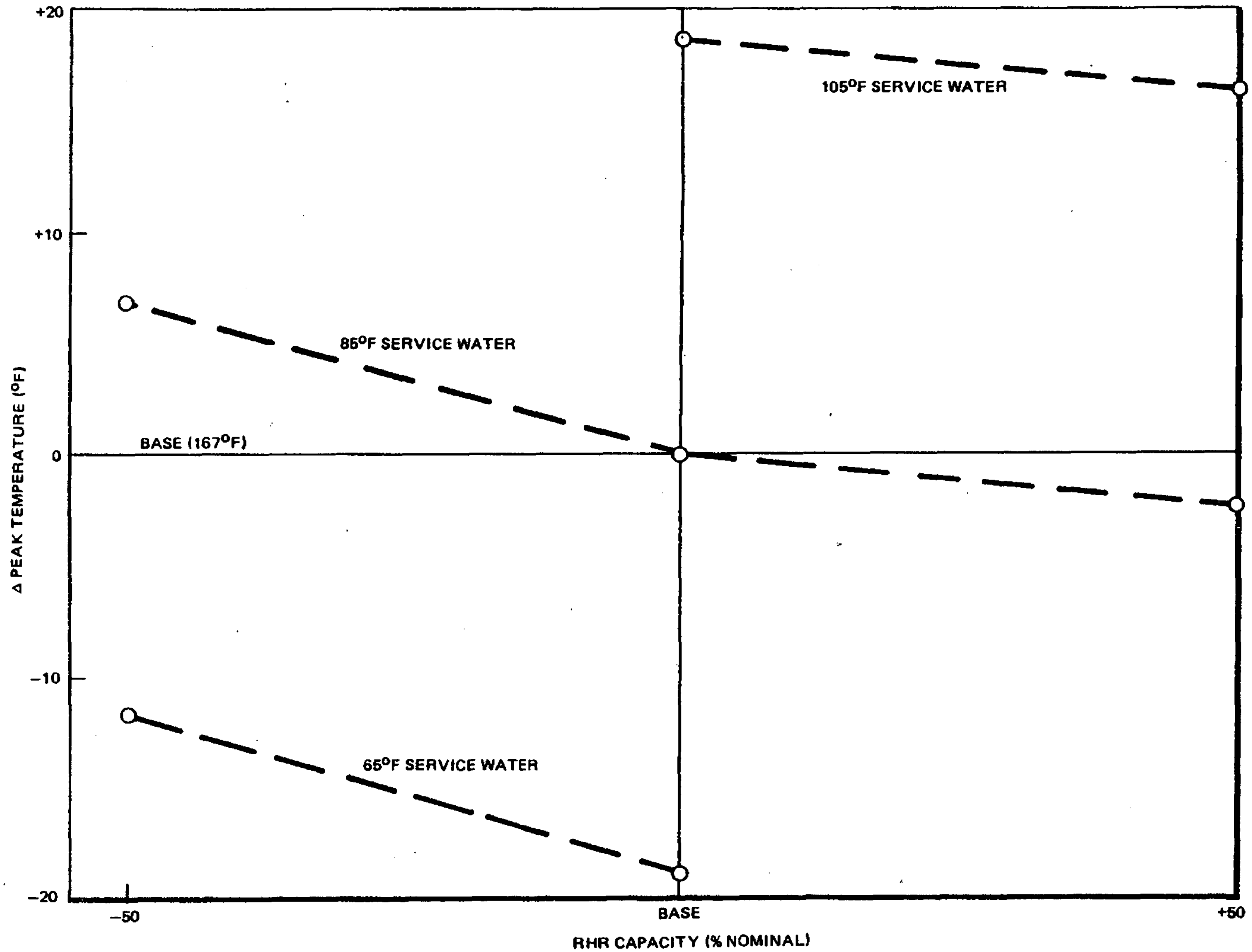


Figure 3.3.4.1-4. BWR-6 ARWS With ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to RHR Heat Capacity

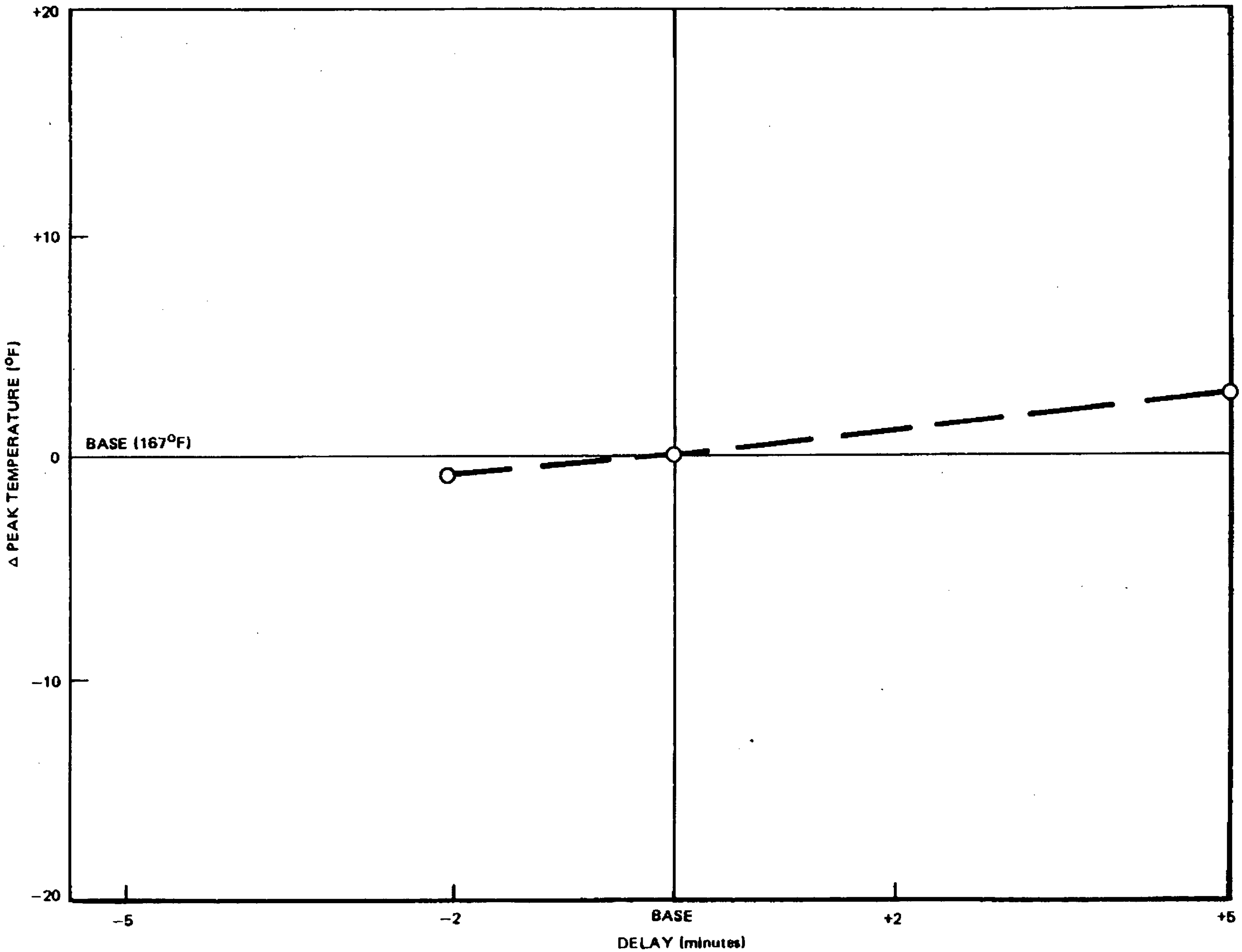


Figure 3.3.4.1-5. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to Delay in Starting RHR

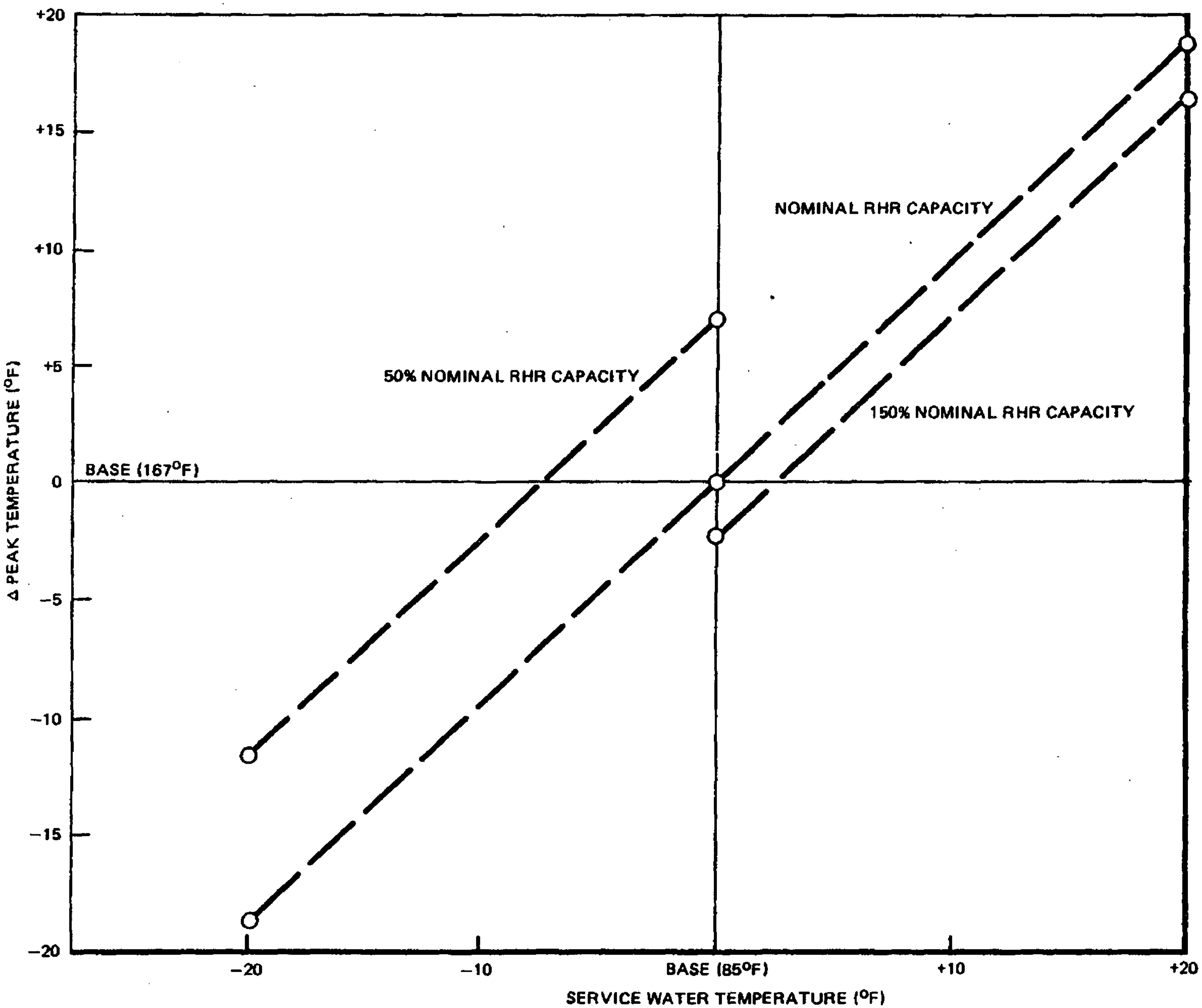


Figure 3.3.4.1-6. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to Service Water Temperature

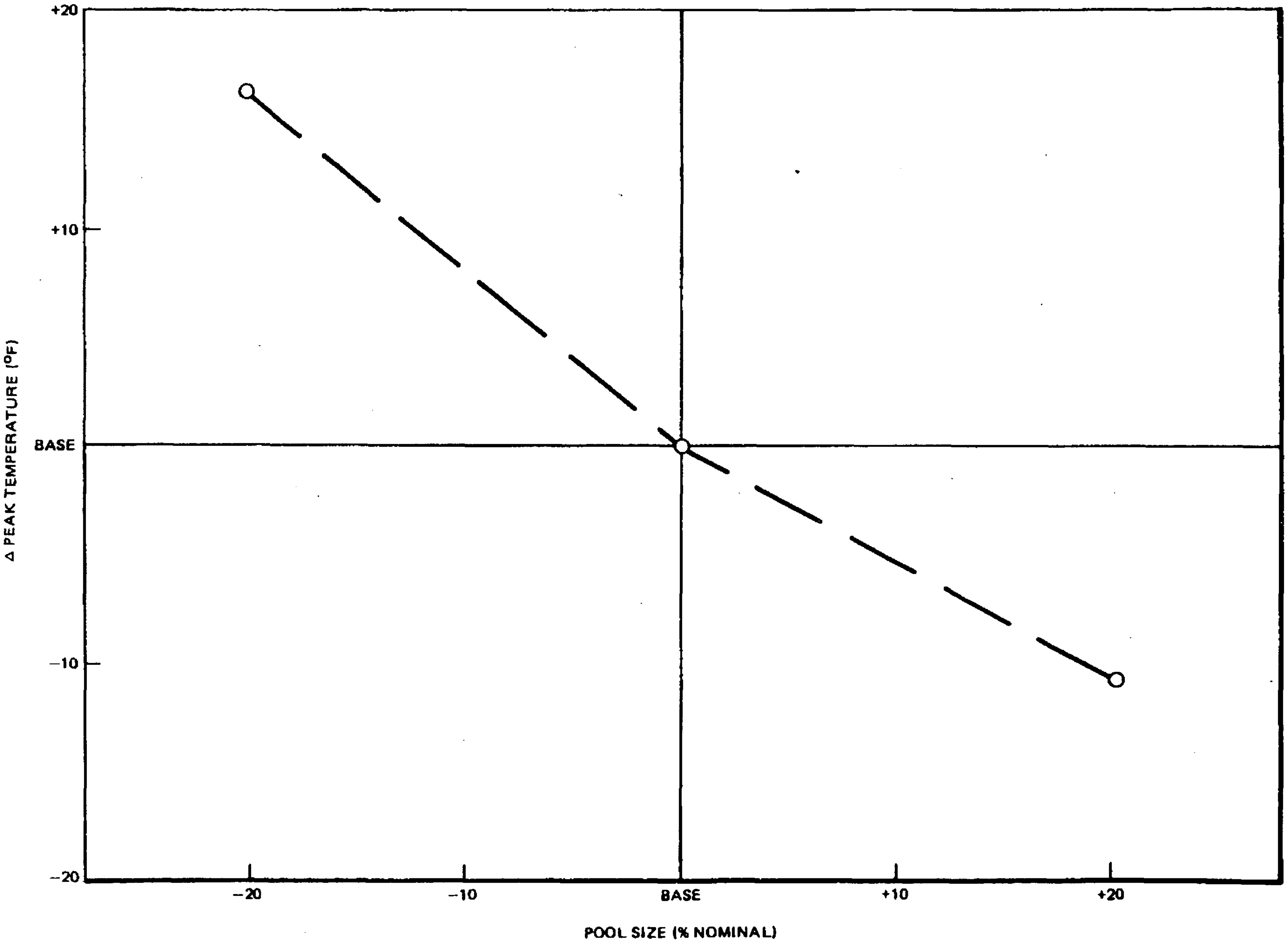


Figure 3.3.4.1-8. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to Initial Mass of the Suppression Pool

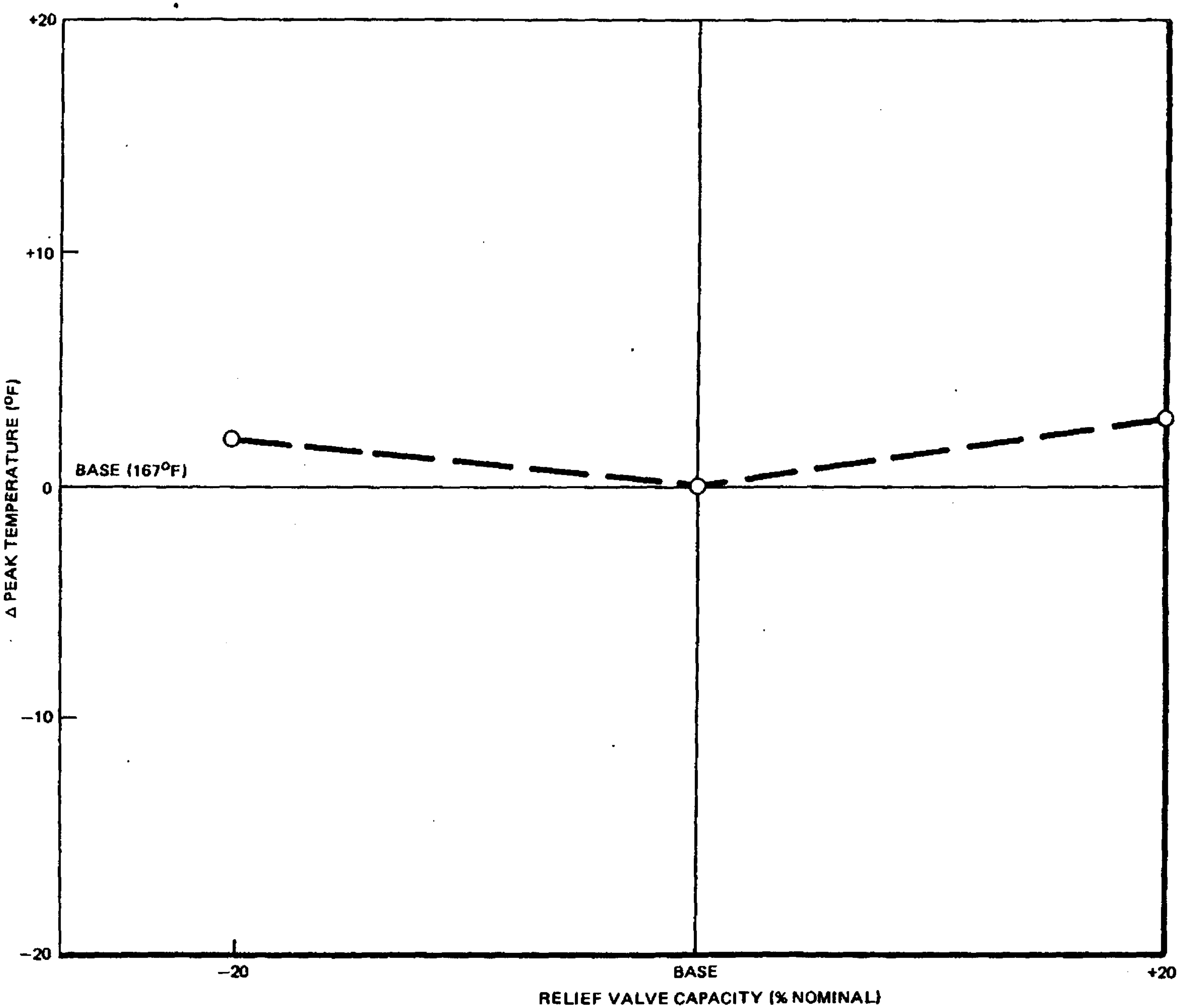


Figure 3.3.4.1-9. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to Relief Valve Capacity

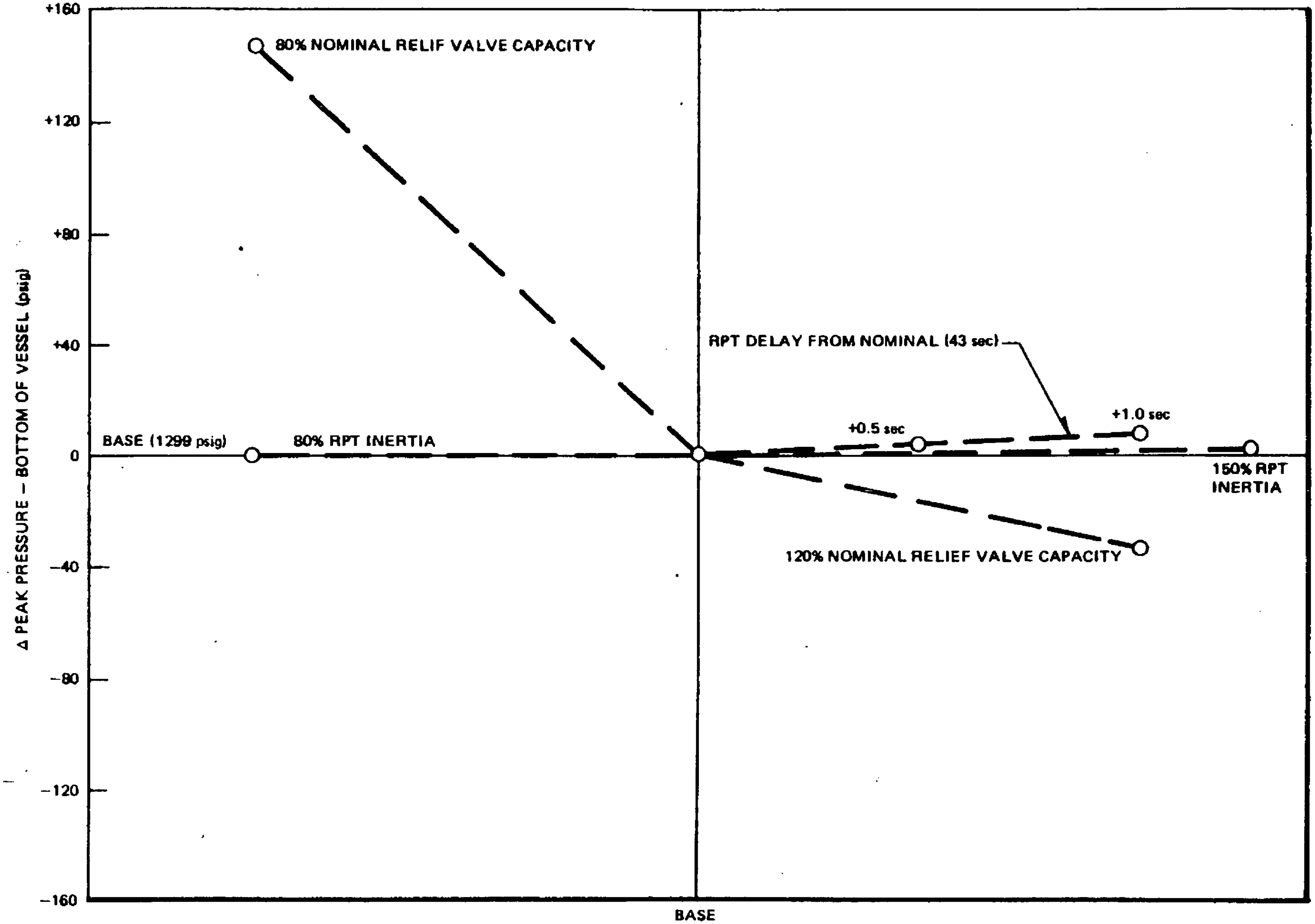


Figure 3.3.4.1.10. BWR-6 ATWS ARI Failure MSIV Sensitivity: Vessel Pressure to Relief Valve Capacity and Recirculation Pump Delay and Inertia

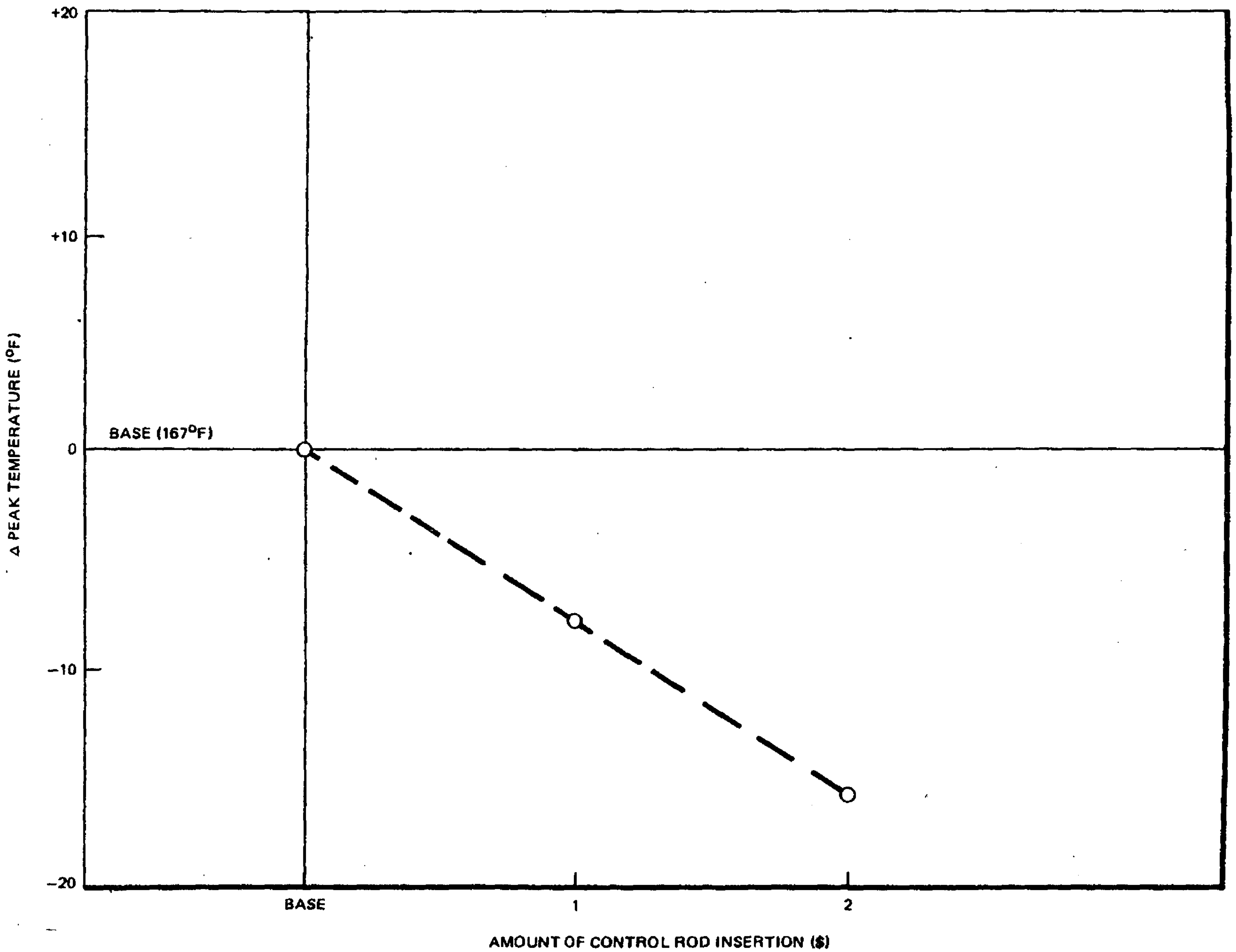


Figure 3.3.4.1.12-1. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Peak Suppression Pool Bulk Temperature to Partial Control Rod Insertion

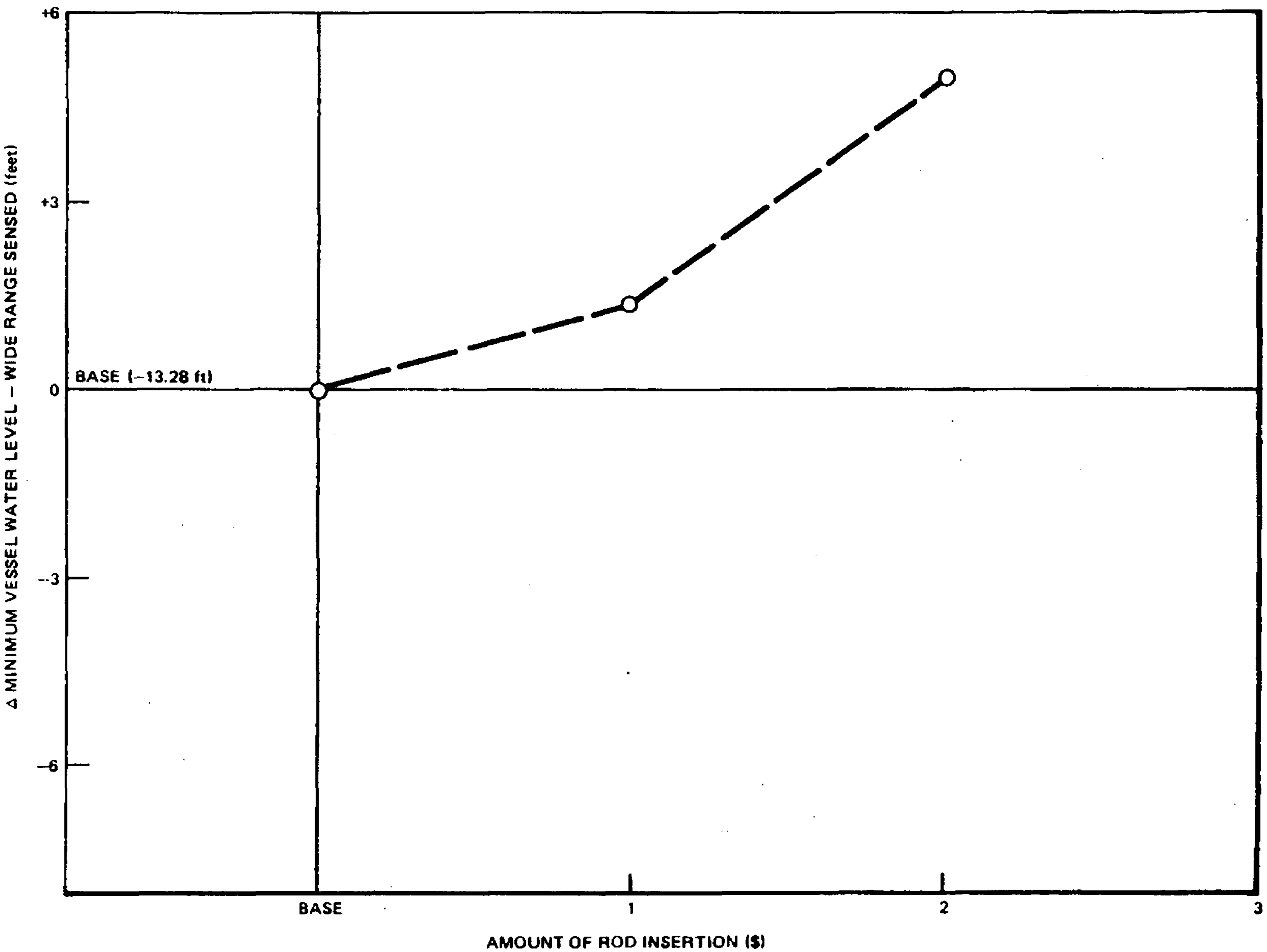


Figure 3.3.4.1.12-2. BWR-6 ATWS With ARI Failure MSIV Sensitivity: Minimum Vessel Water Level to Partial Control Rod Insertion

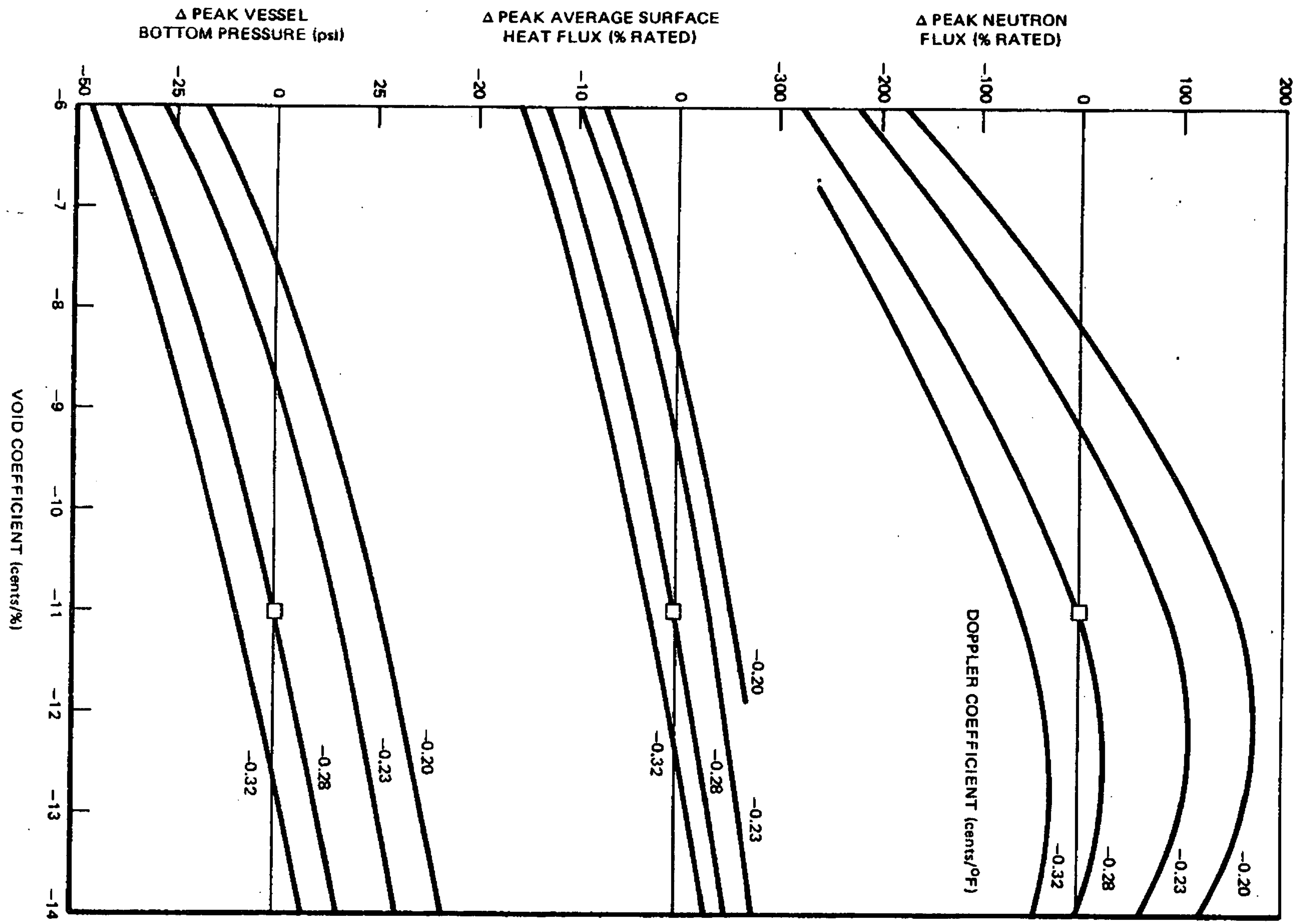


Figure 3.3.4.1.13-1. BWR/6-251 ATWS MSIV

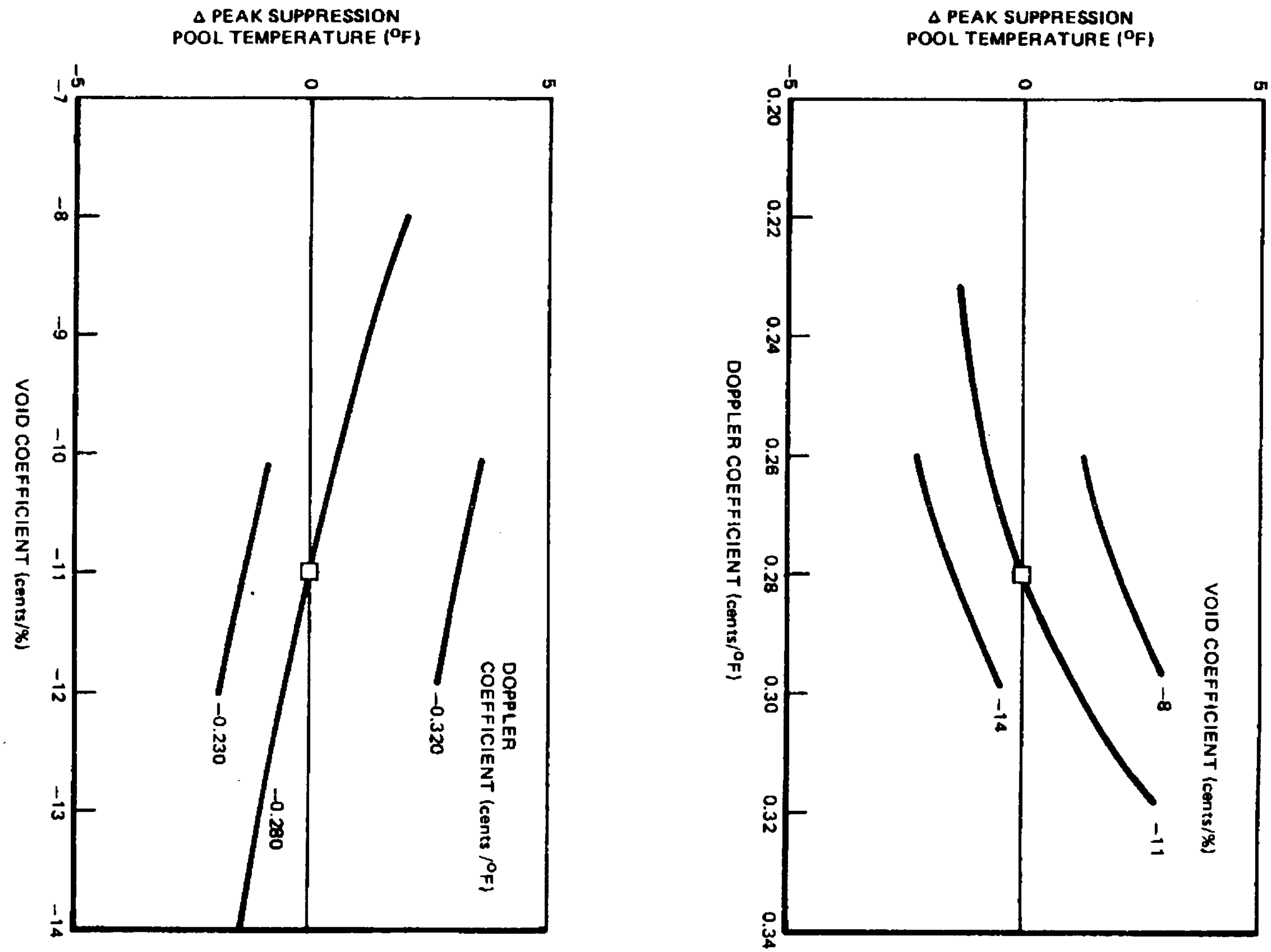


Figure 3.3.4.1.13-2. BWR/6-251 ATWS MSIV

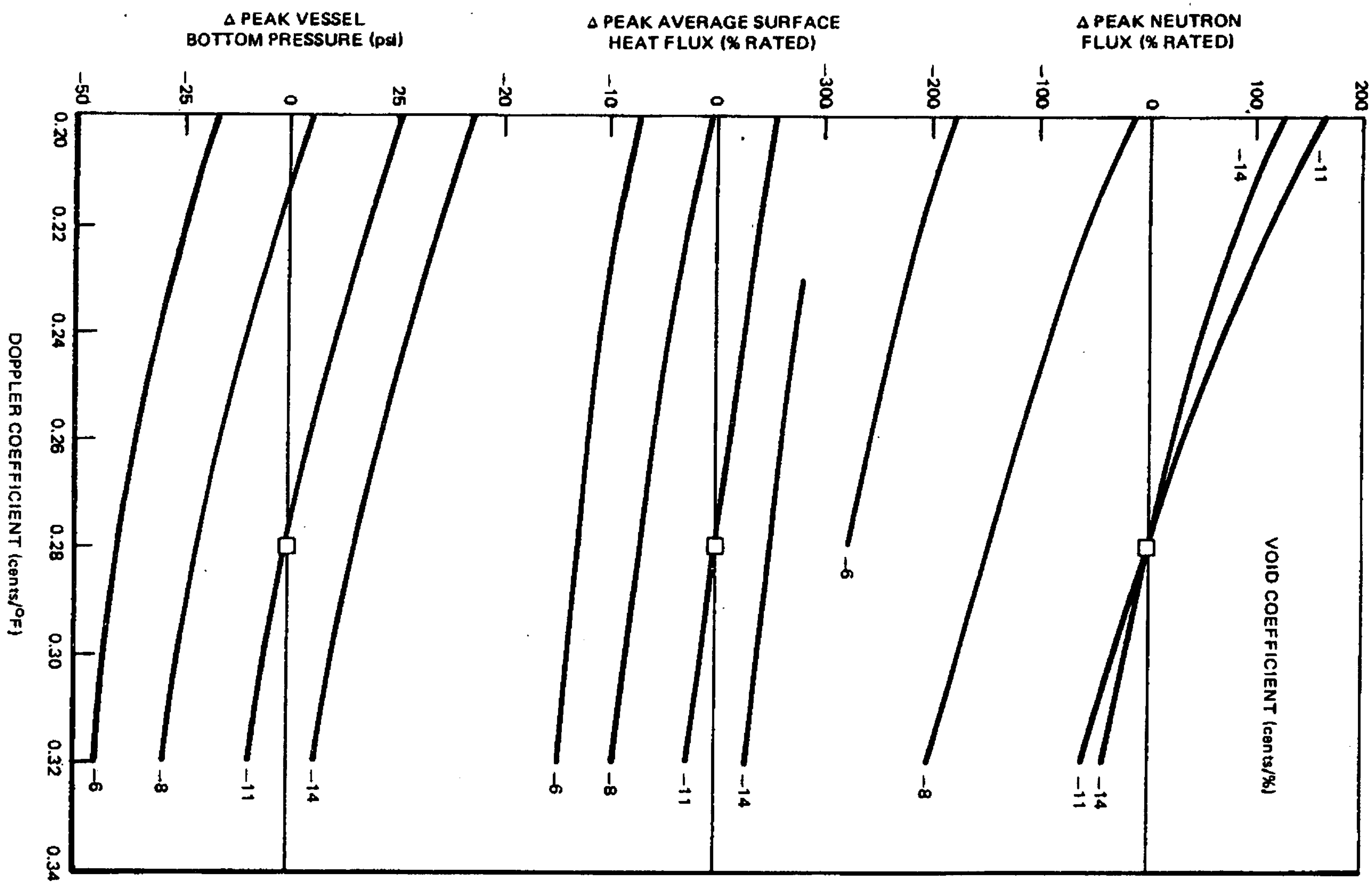


Figure 3.3.4.1.14-1. BWR/6-251 ATWS MSIV

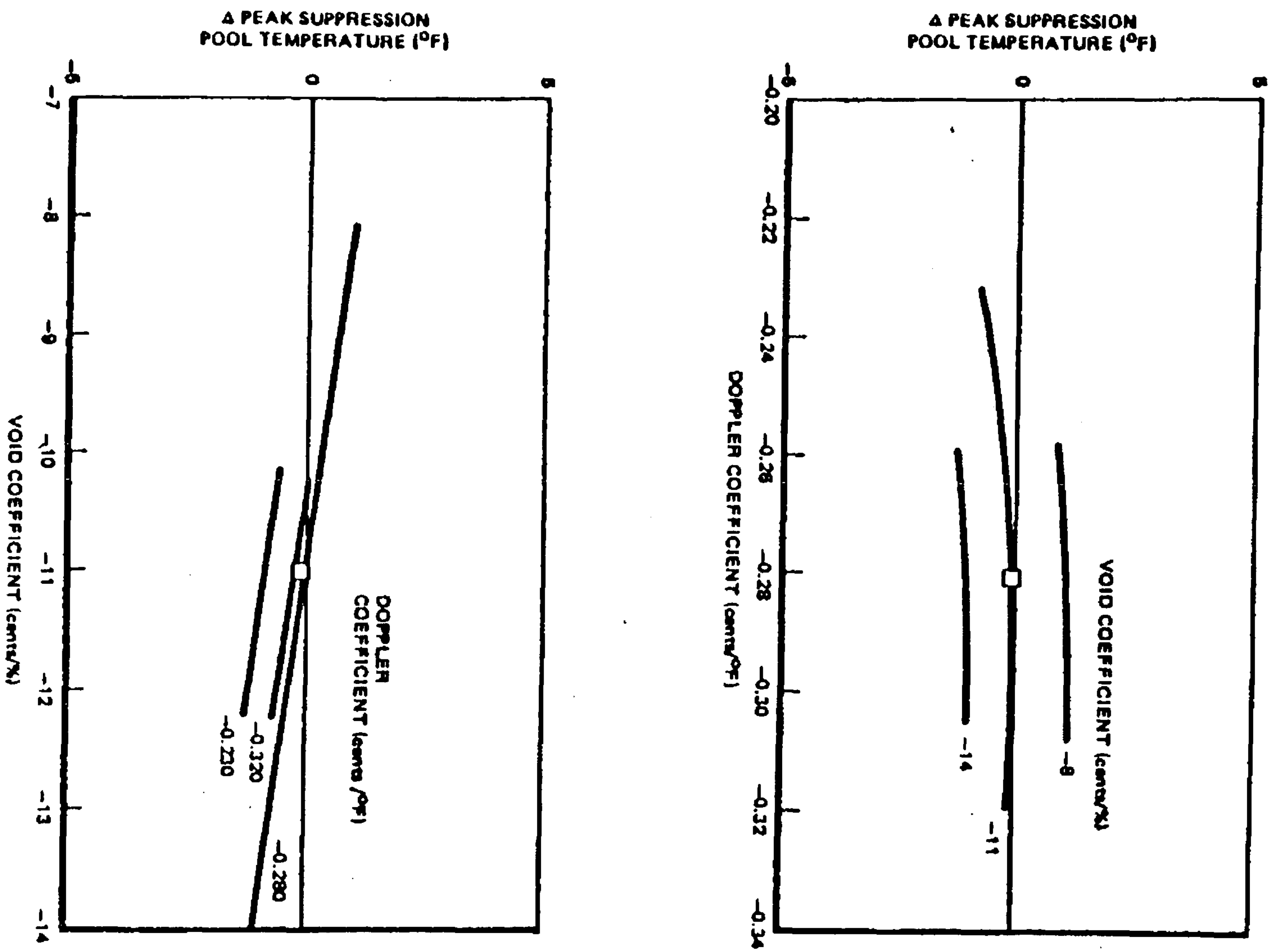


Figure 3.3.4.2.5-2. BWR/6-251 ATWS Turbine Trip

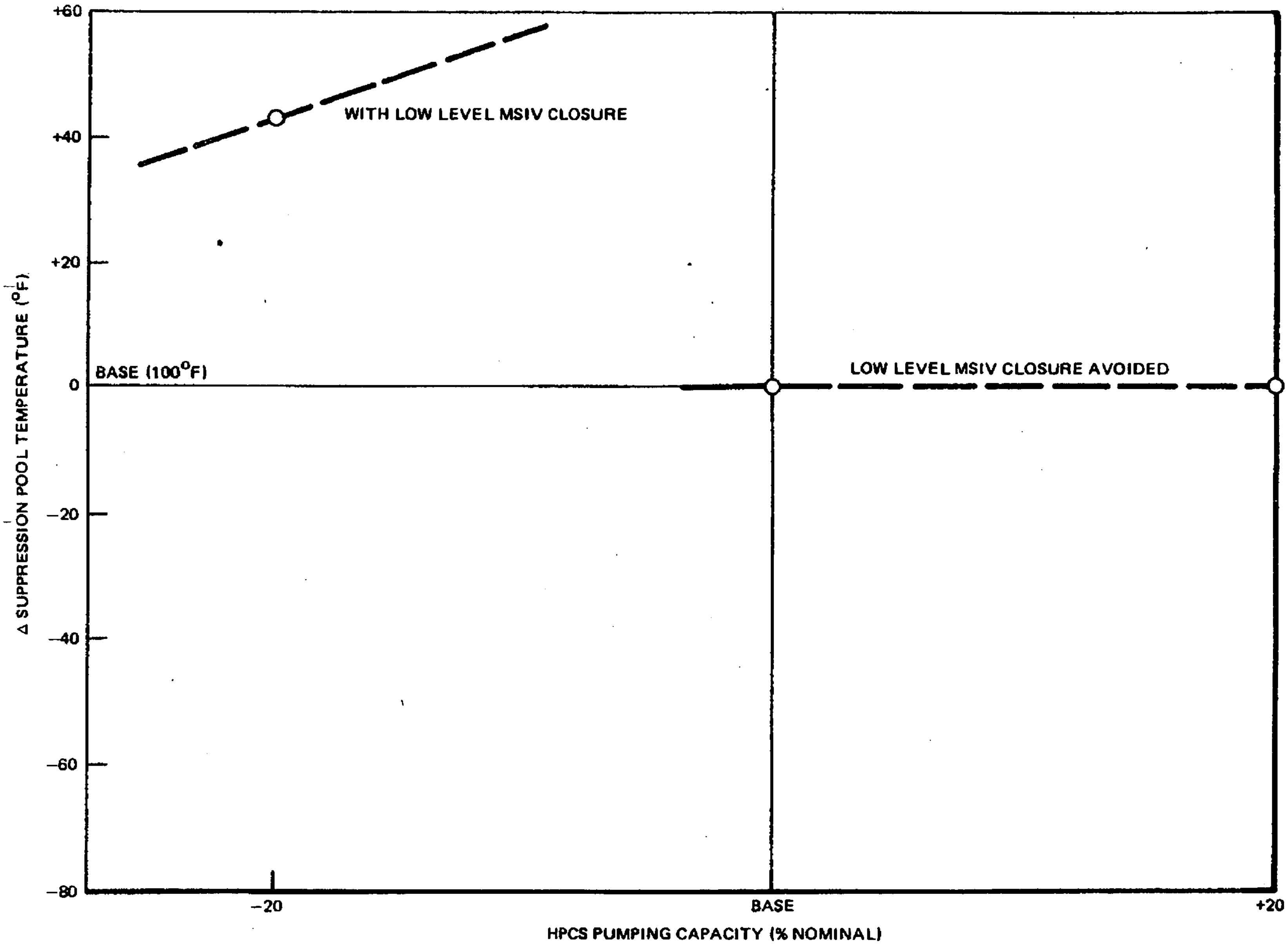


Figure 3.3.4.2.3-1. BWR-6 ATWS With ARI Failure Turbine Trip With Bypass Sensitivity: Peak Suppression Pool Bulk Temperature to HPCS Capacity

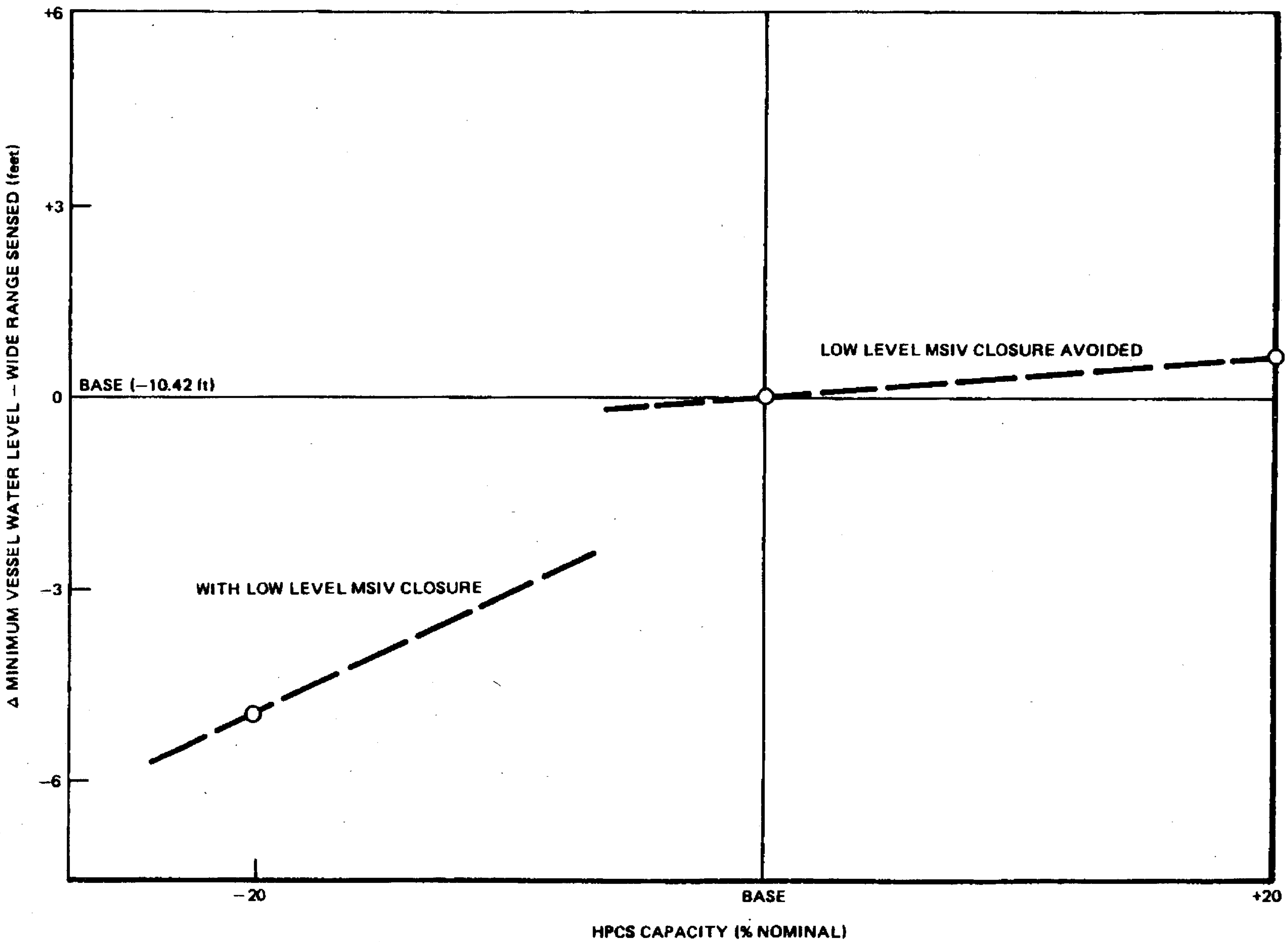


Figure 3.3.4.2.3-6. BWR-6 ATWS With ARI Failure Turbine Trip With Bypass Sensitivity: Minimum Vessel Water Level to HPCS Capacity

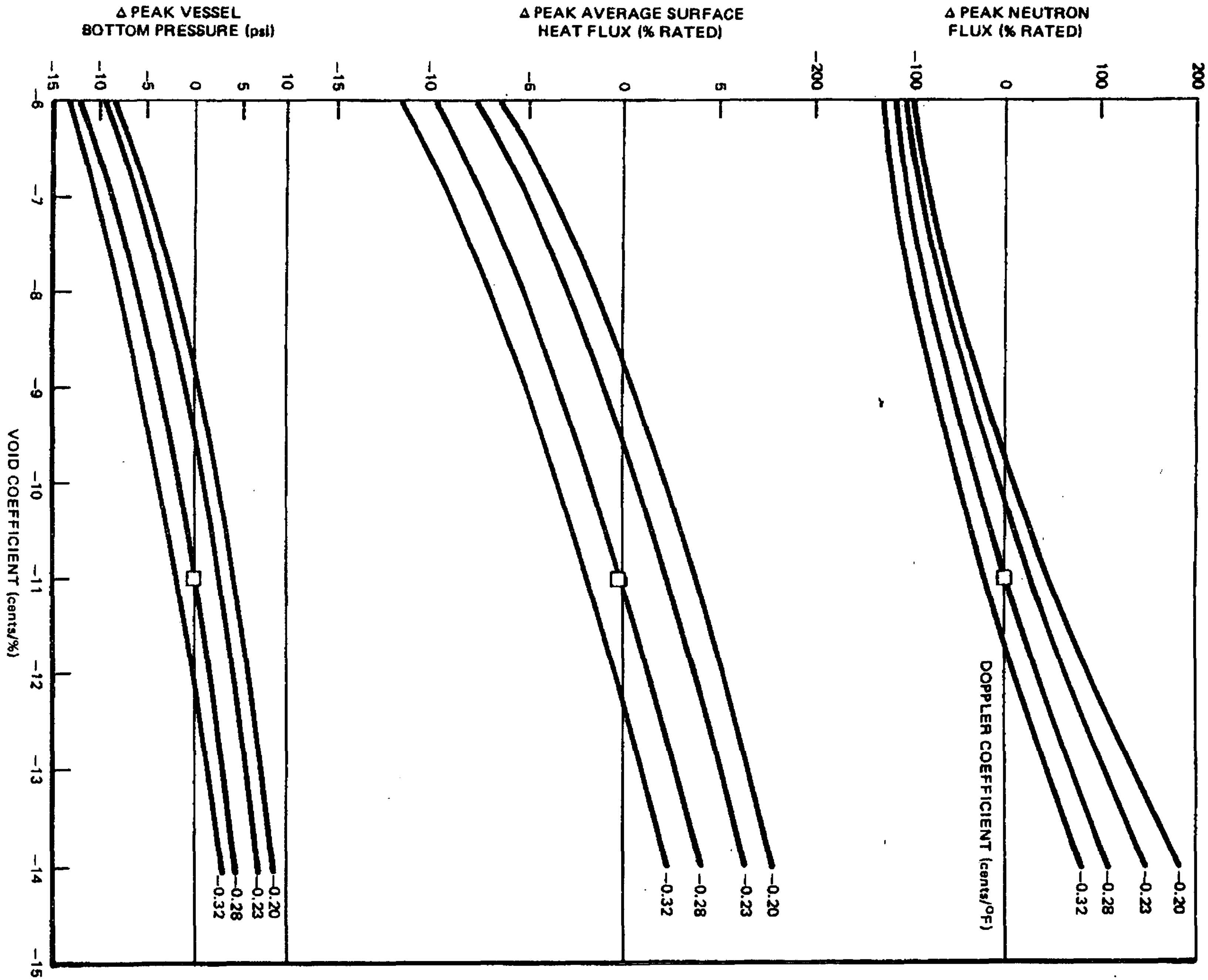


Figure 3.3.4.2.5-1. BWR/6-251 ATWS Turbine Trip

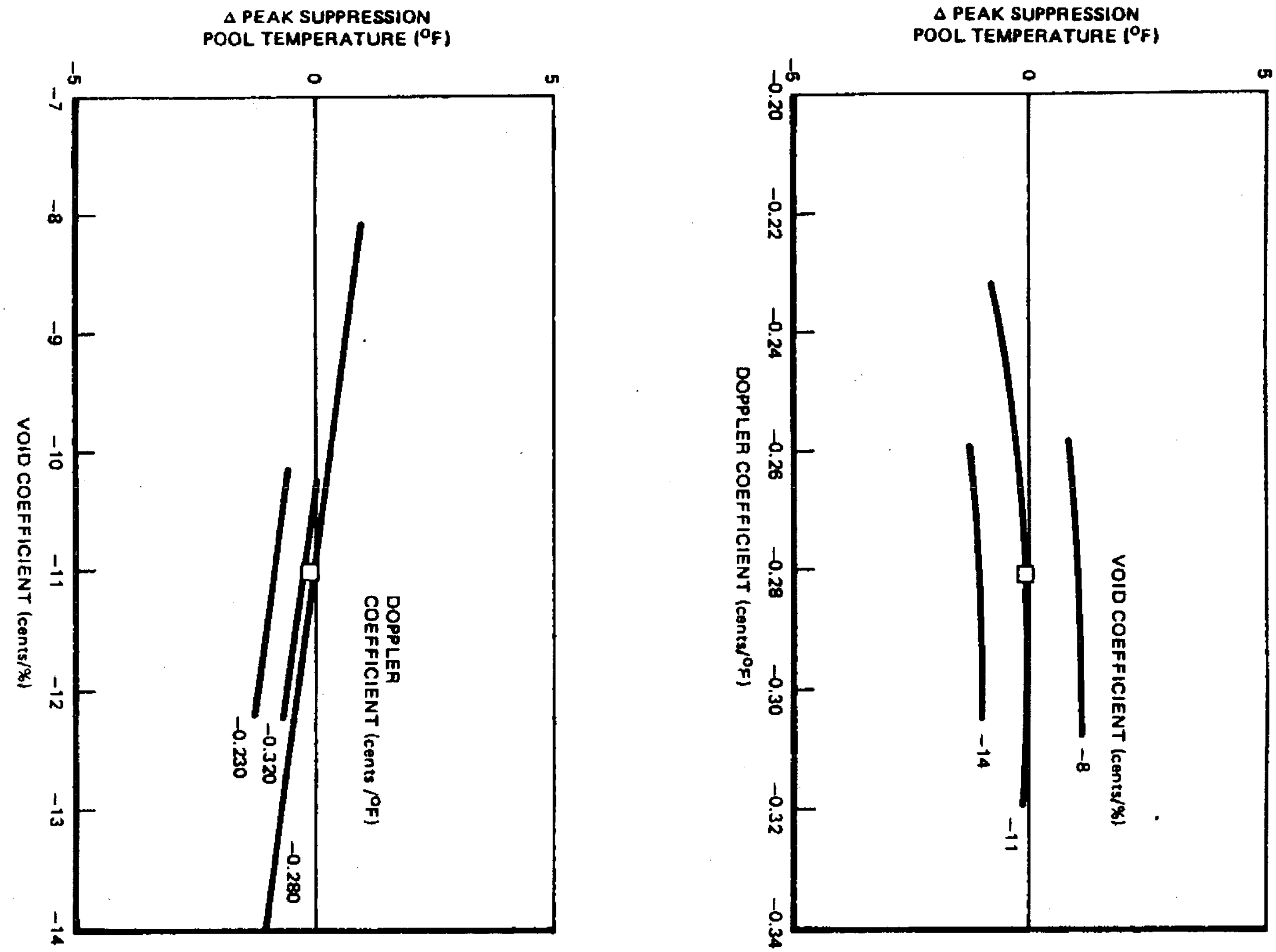


Figure 3.3.4.2.5-2. BWR/6-251 ATWS Turbine Trip

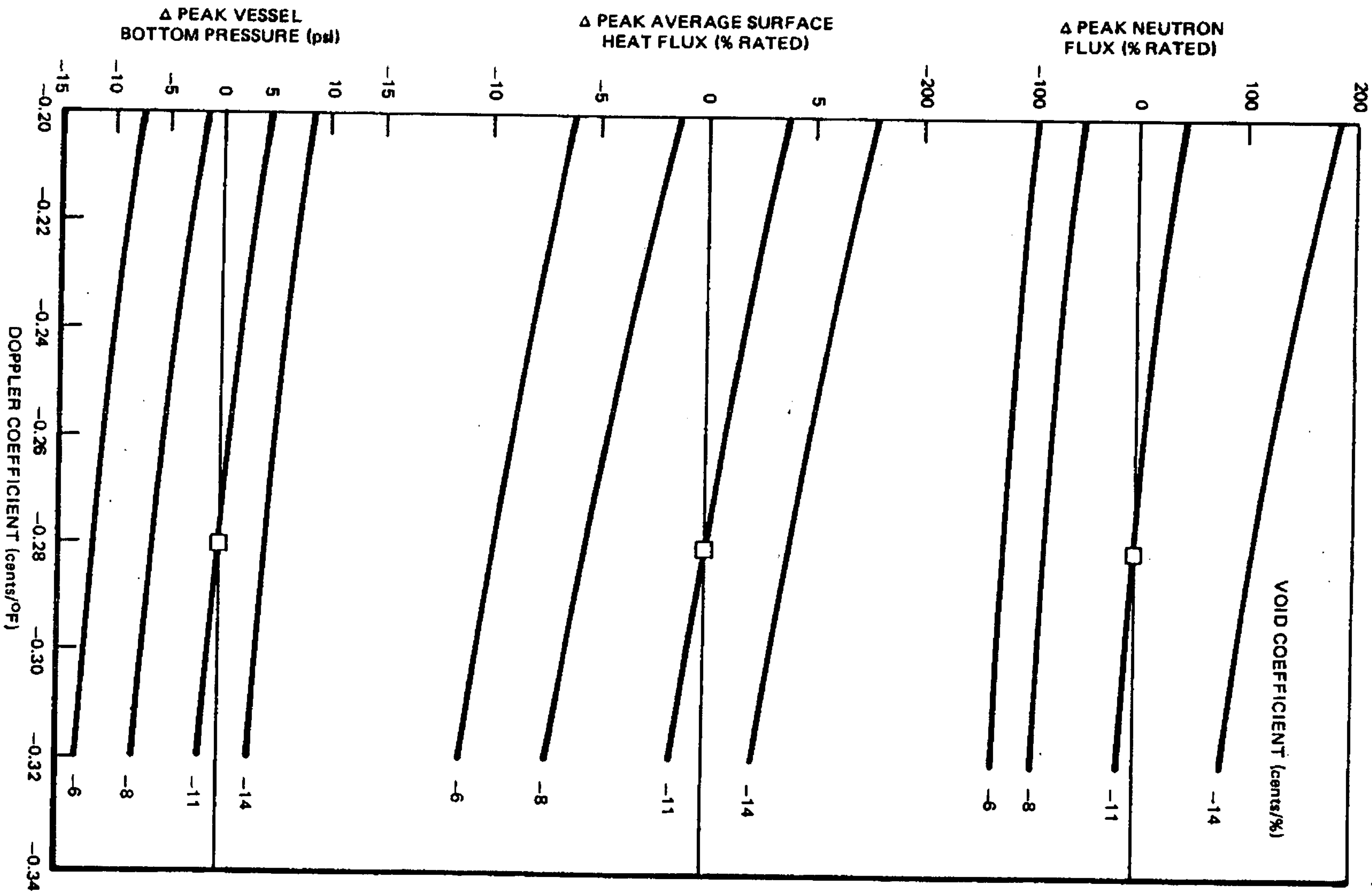
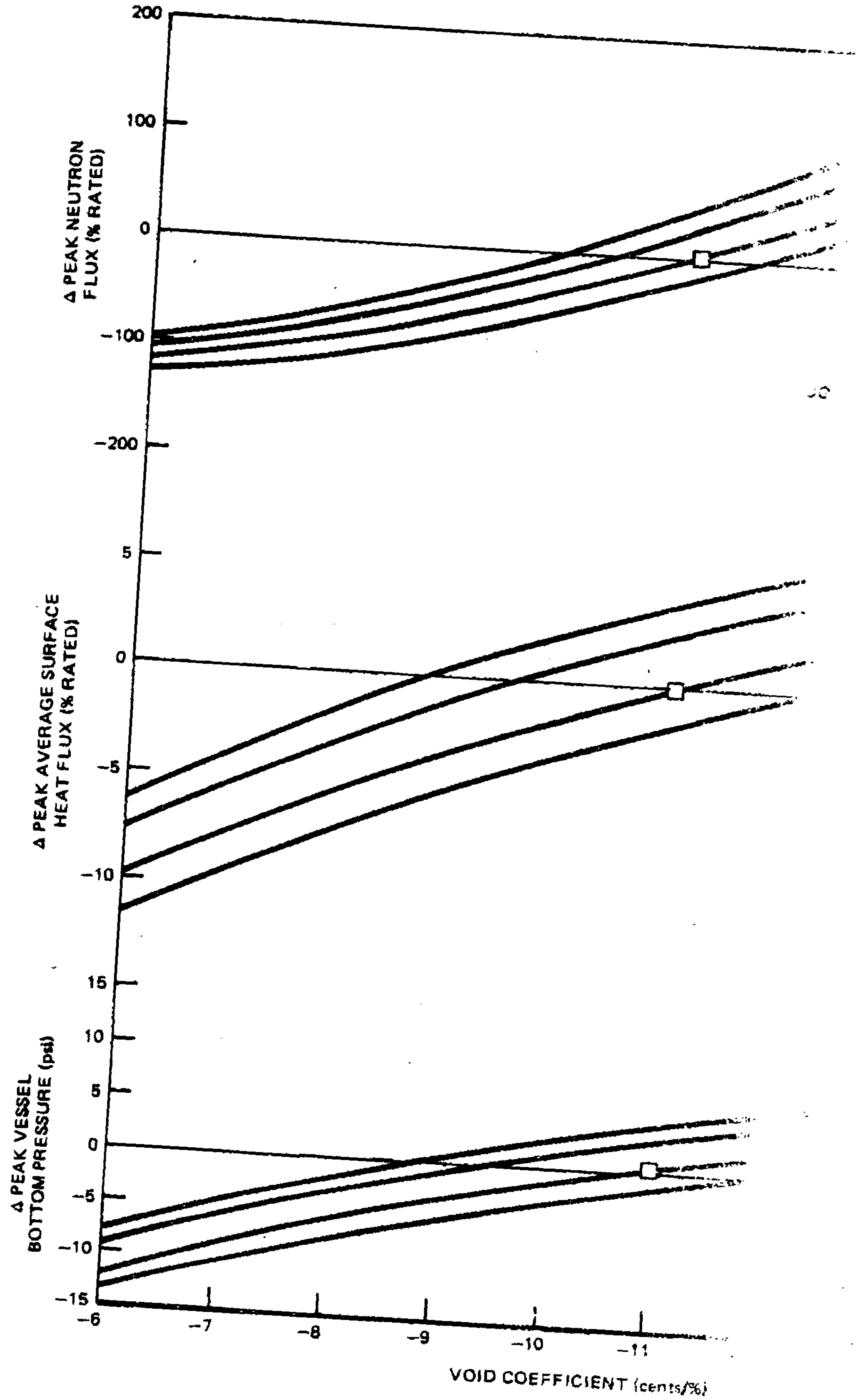


Figure 3.3.4.2.6-1. BWR/6-251 ATWS Turbine Trip


Δ PEAK NEUTRON FLUX (% RATED)
200
100
0
−100
−200
Δ PEAK AVERAGE SURFACE HEAT FLUX (% RATED)
5
0
−5
−10
Δ PEAK VESSEL BOTTOM PRESSURE (psi)
15
10
5
0
−5
−10
−15
−6
−7
−8
−9
−10
−11
VOID COEFFICIENT (cents/%)

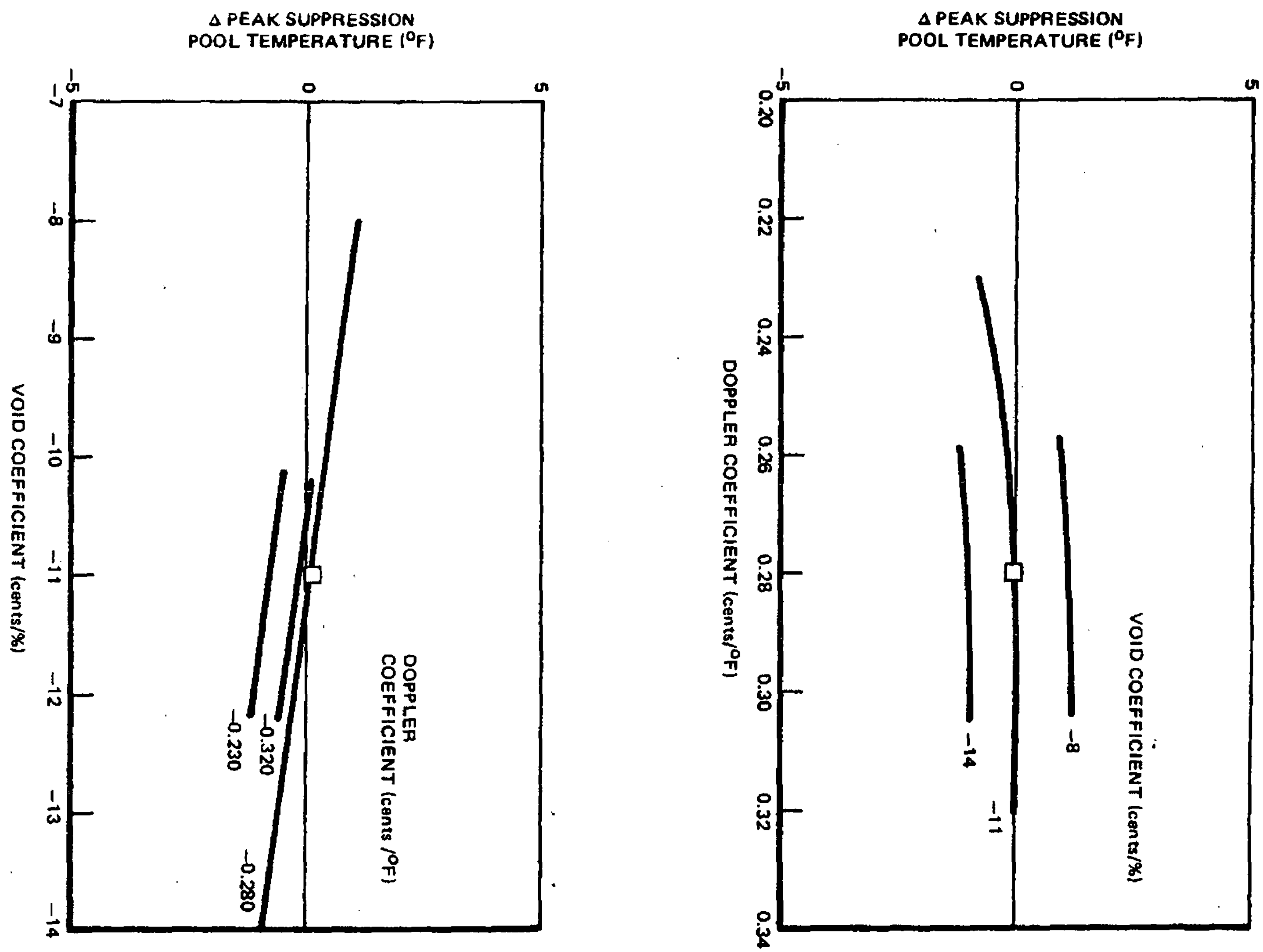


Figure 3.3.4.2.6-3. BWR/6-251 ATWS Turbine Trip

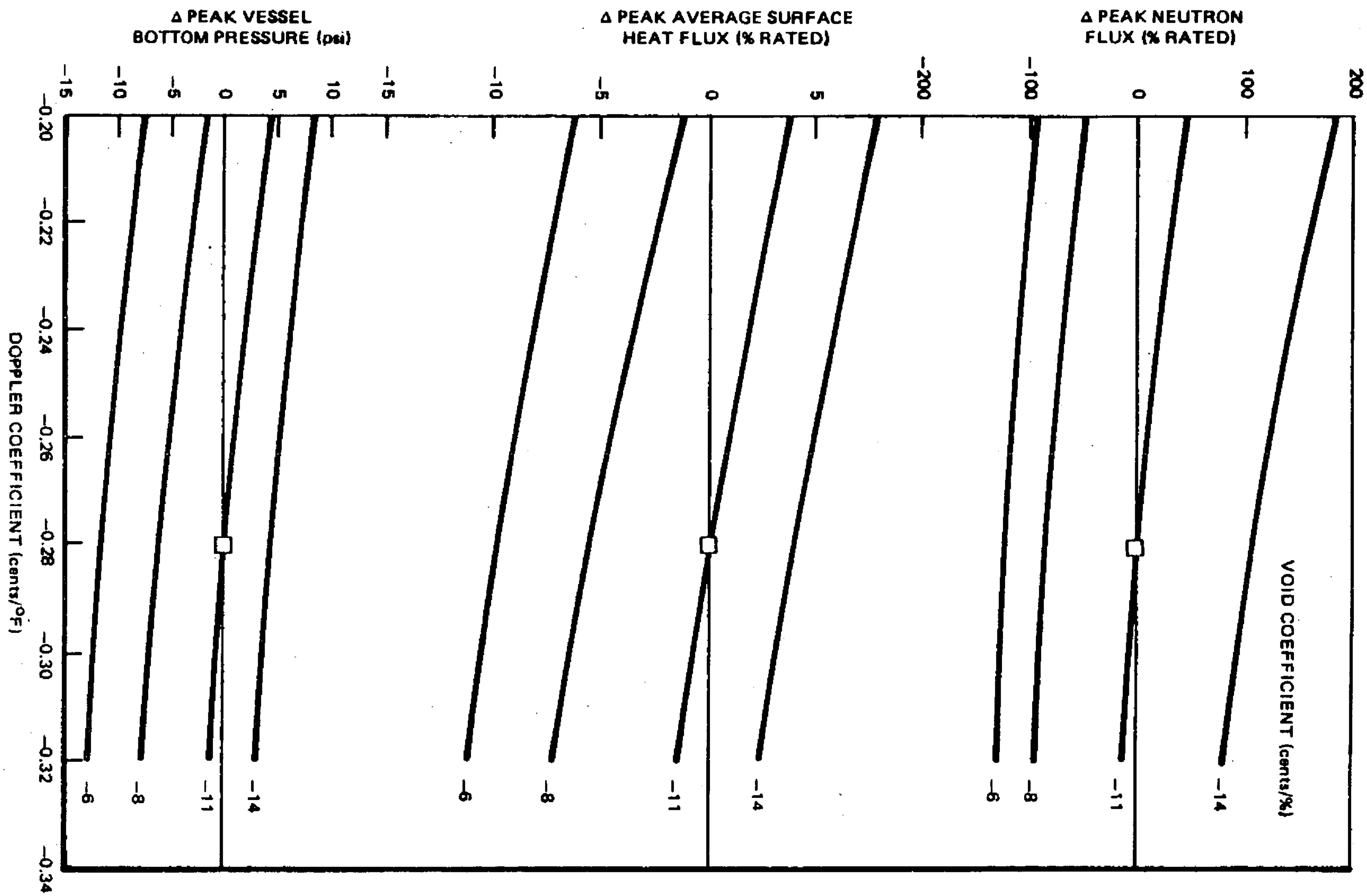


Figure 3.3.4.2.7-1. BWR/6-251 ATWS Turbine Trip

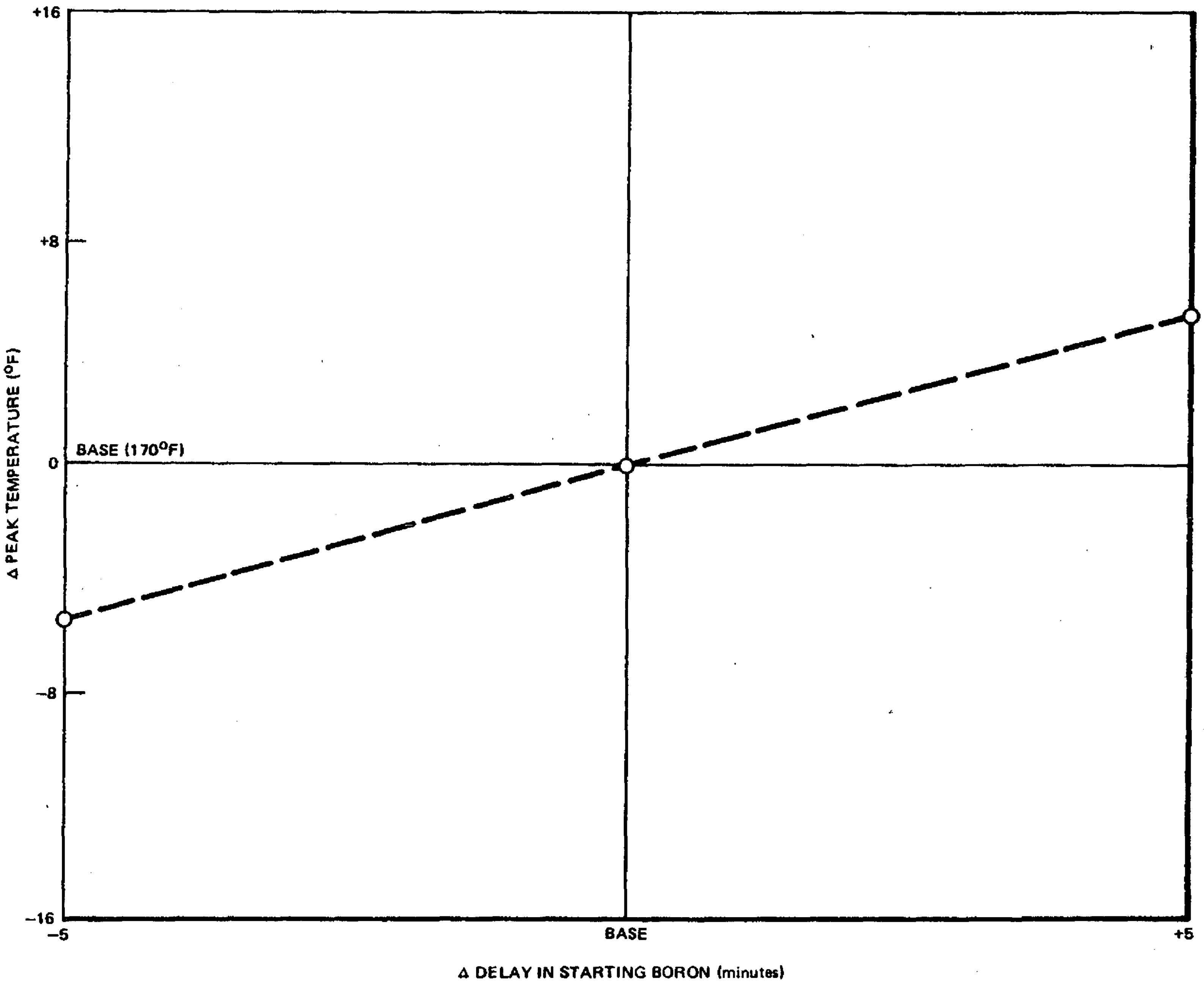


Figure 3.3.4.3-1. BWR-6 ATWS With ARI Failure IORV Sensitivity: Peak Suppression Pool Bulk Temperature to Delay in Starting Liquid Boron Injection

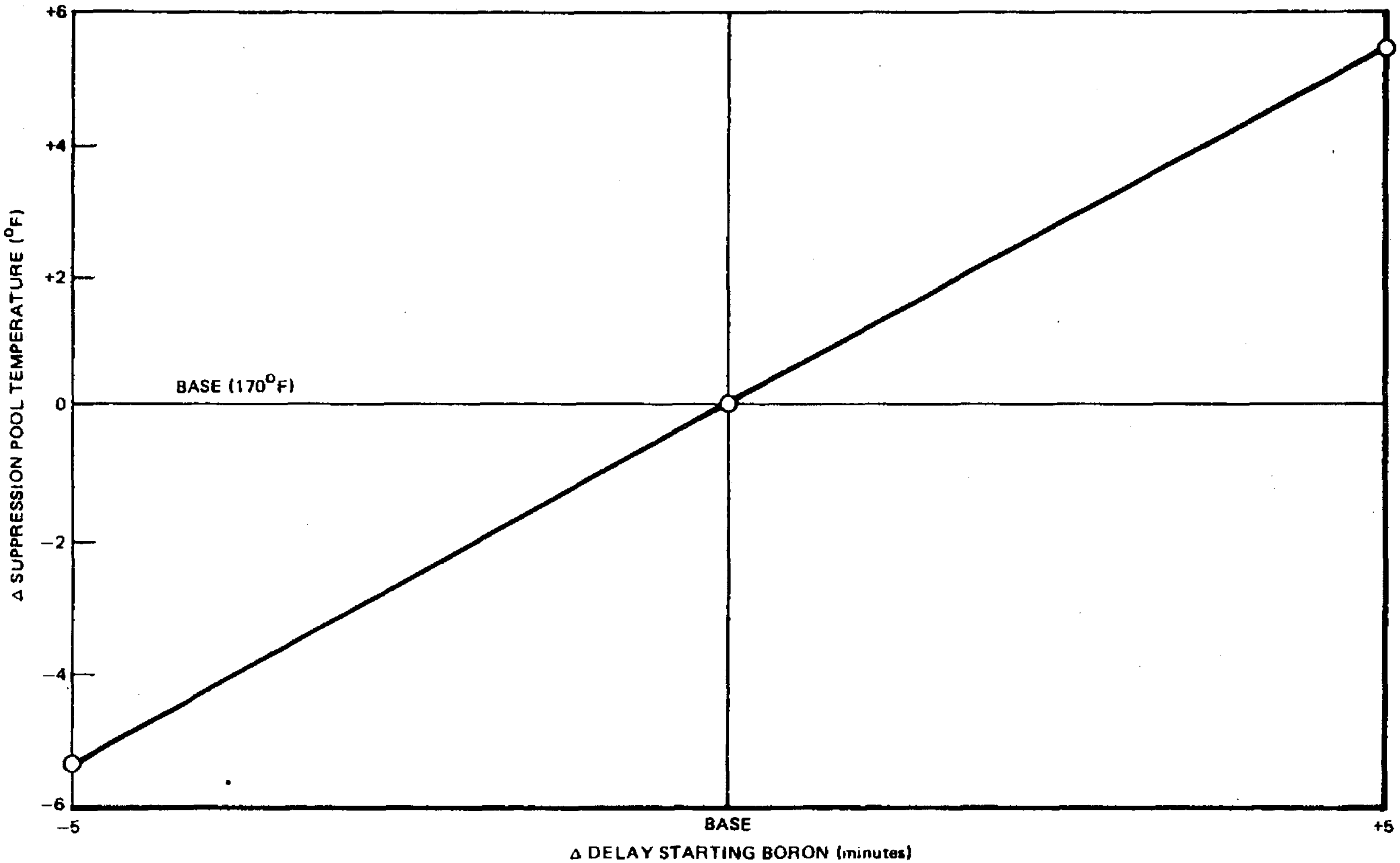


Figure 3.3.4.3.2. BWR-6 ATWS IORV Sensitivity: Peak Suppression Pool Temperature to Delay in Starting Liquid Boron

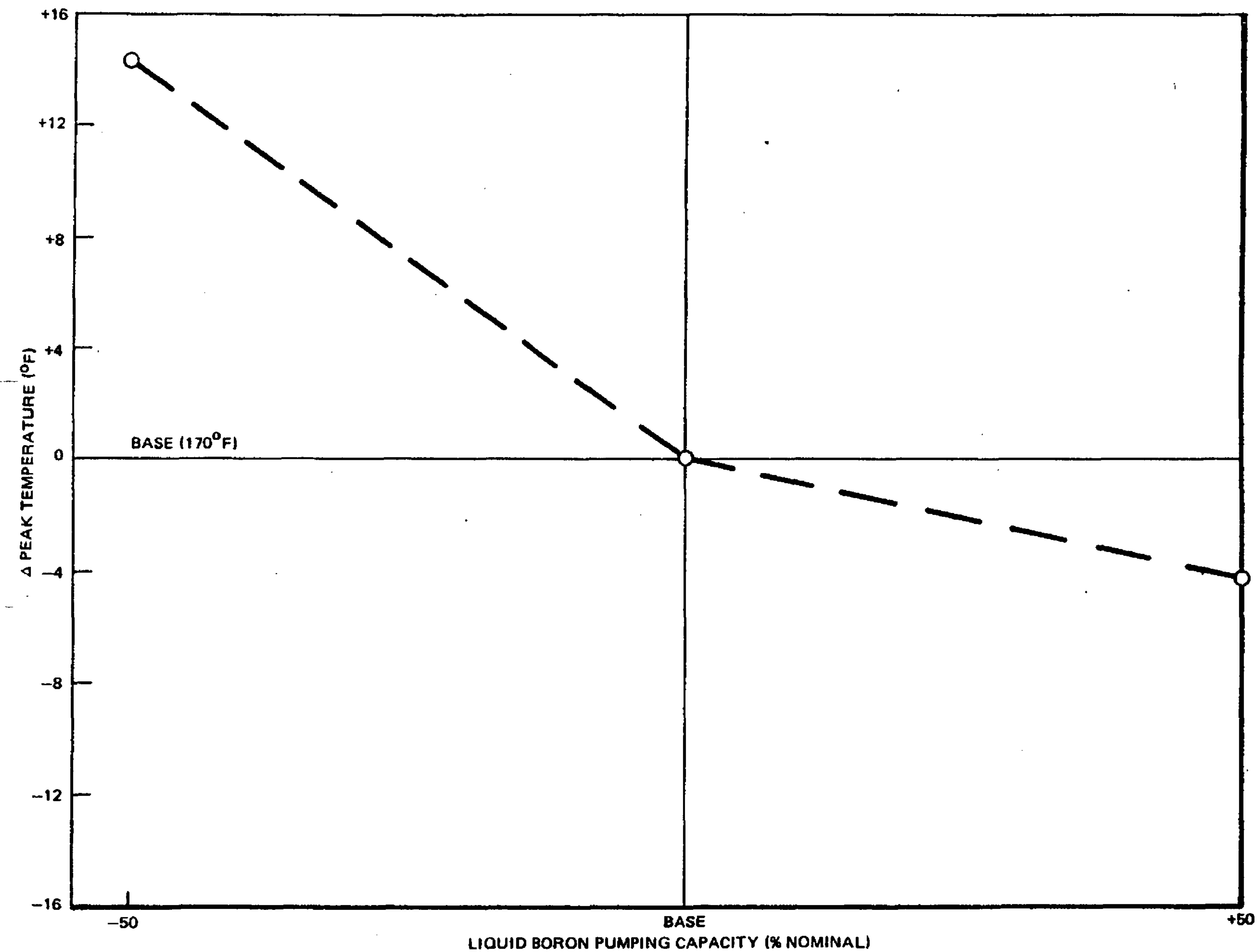


Figure 3.3.4.3-3. BWR-6 ATWS With ARI Failure IORV Sensitivity: Peak Suppression Pool Bulk Temperature to Boron Pump Capacity

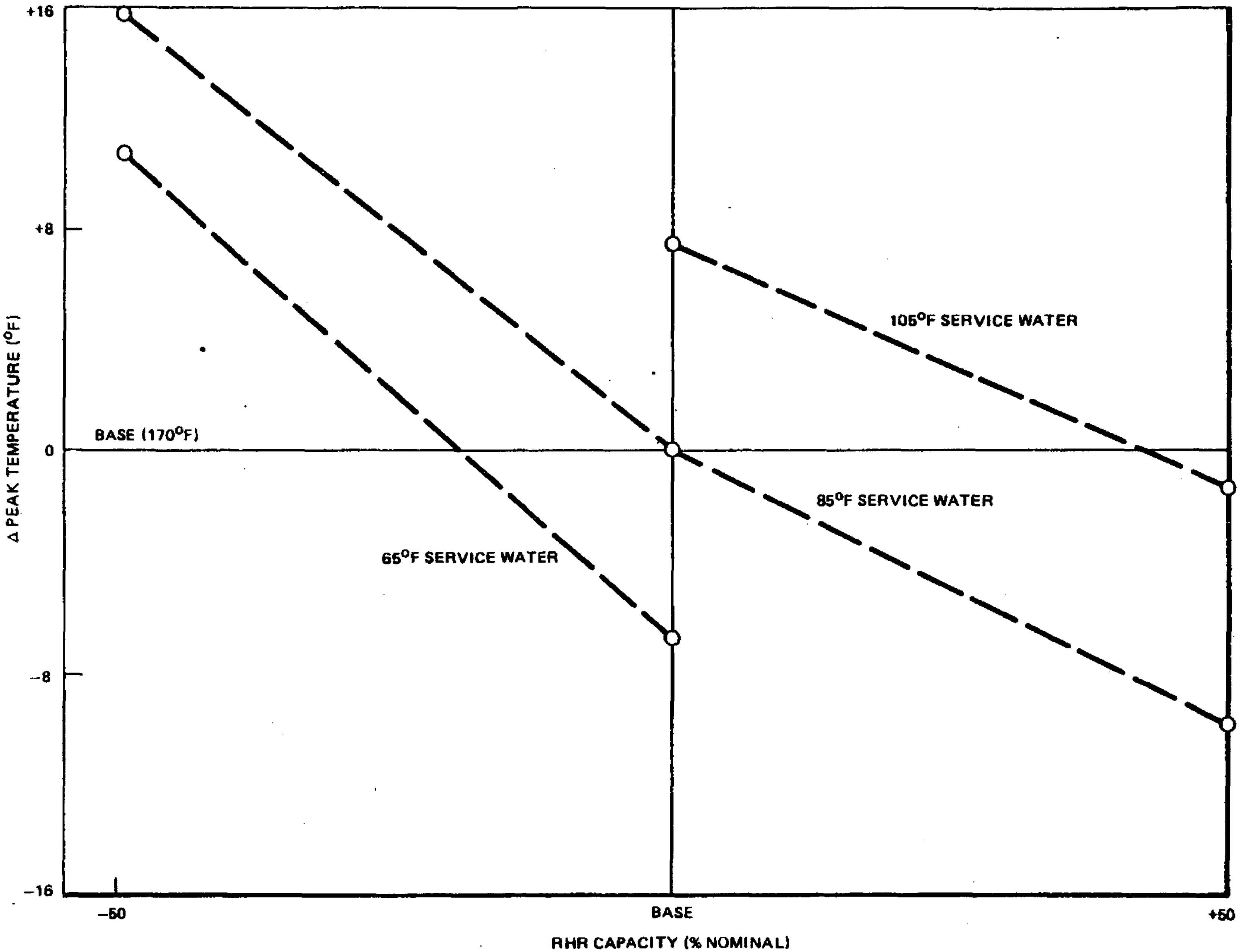


Figure 3.3.4.3-4. BWR-6 ATWS With ARI Failure IORV Sensitivity: Peak Suppression Pool Bulk Temperature to RHR Capacity

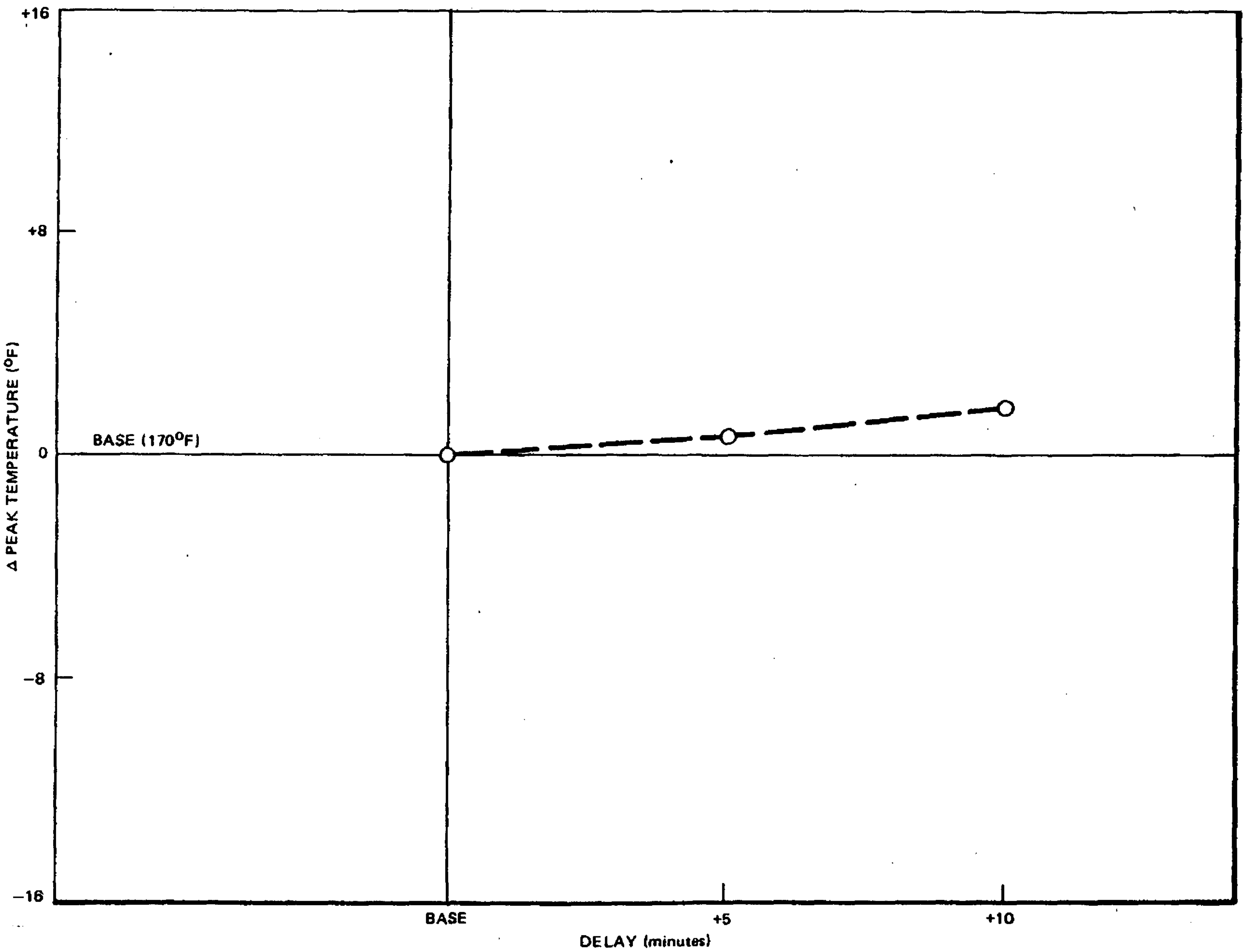


Figure 3.3.4.3-5. BWR-6 ATWS With ARI Failure IORV Sensitivity: Peak Suppression Pool Bulk Temperature to Delaying Start of RHR

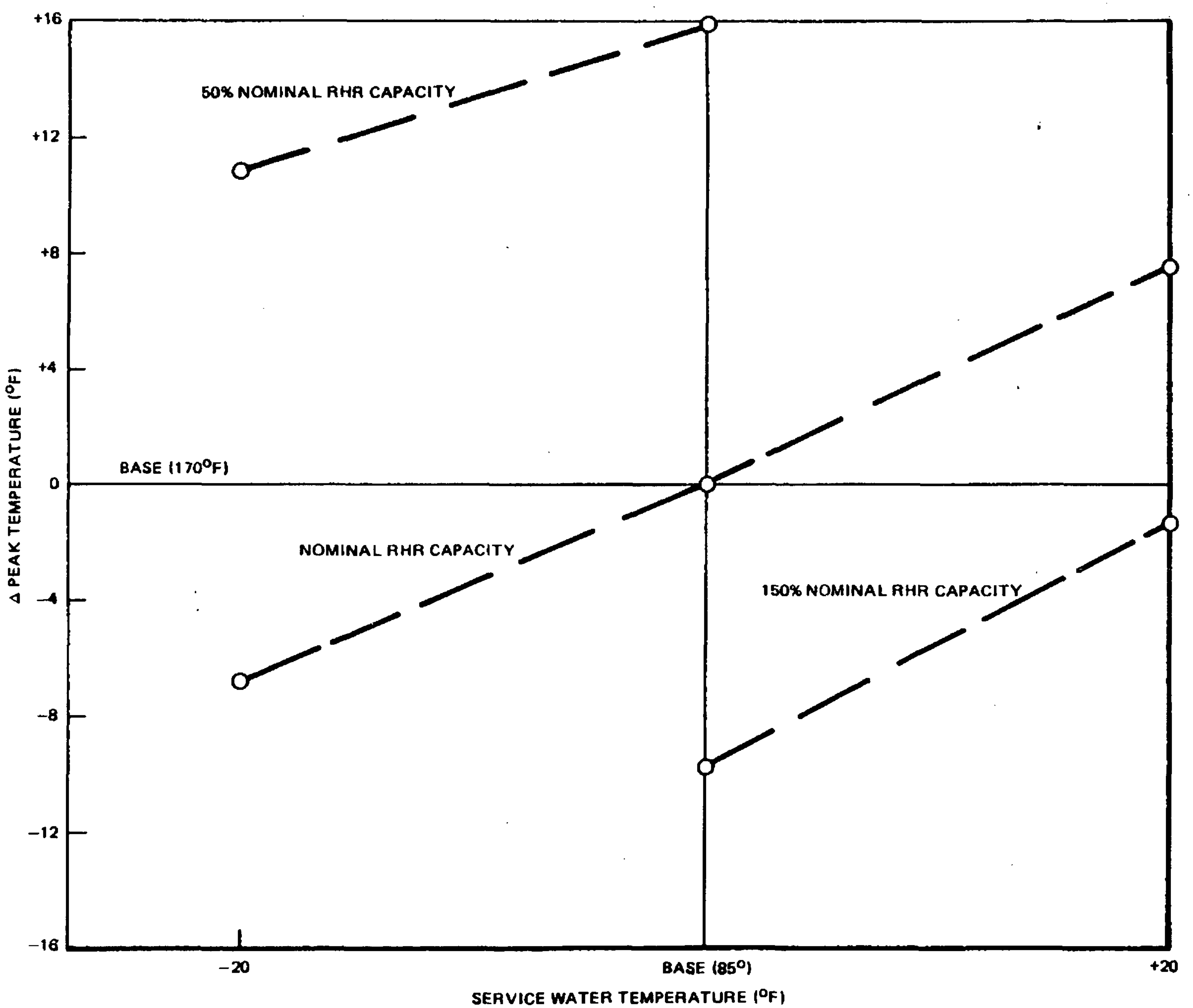


Figure 3.3.4.3.6. BWR-6 ATWS With ARI Failure IORV Sensitivity: Peak Suppression Pool Bulk Temperature to Service Water Temperature

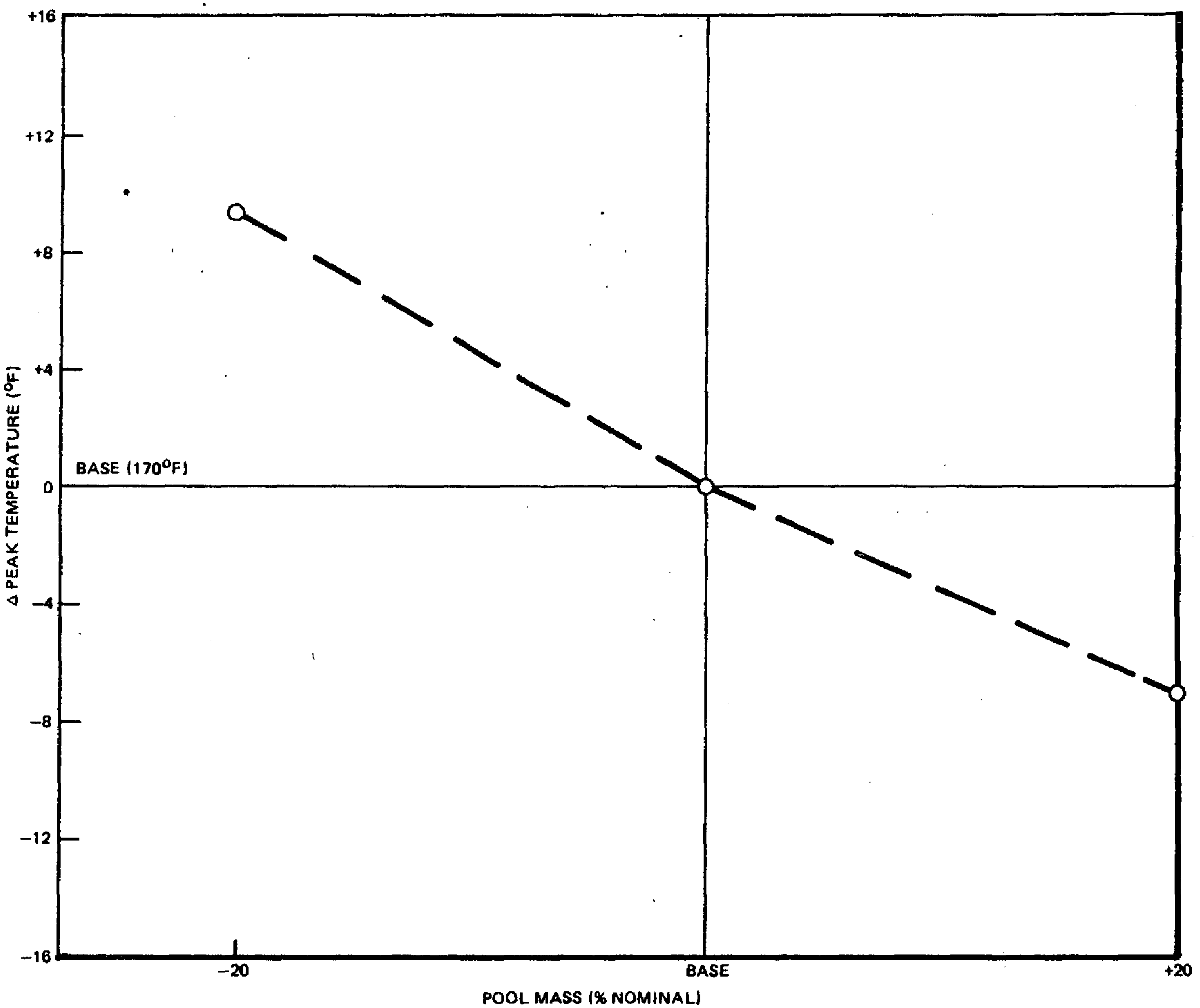


Figure 3.3.4.3-7. BWR-6 ATWS With ARI Failure IORV Sensitivity: Peak Suppression Pool Bulk Temperature to Initial Mass of Suppression Pool

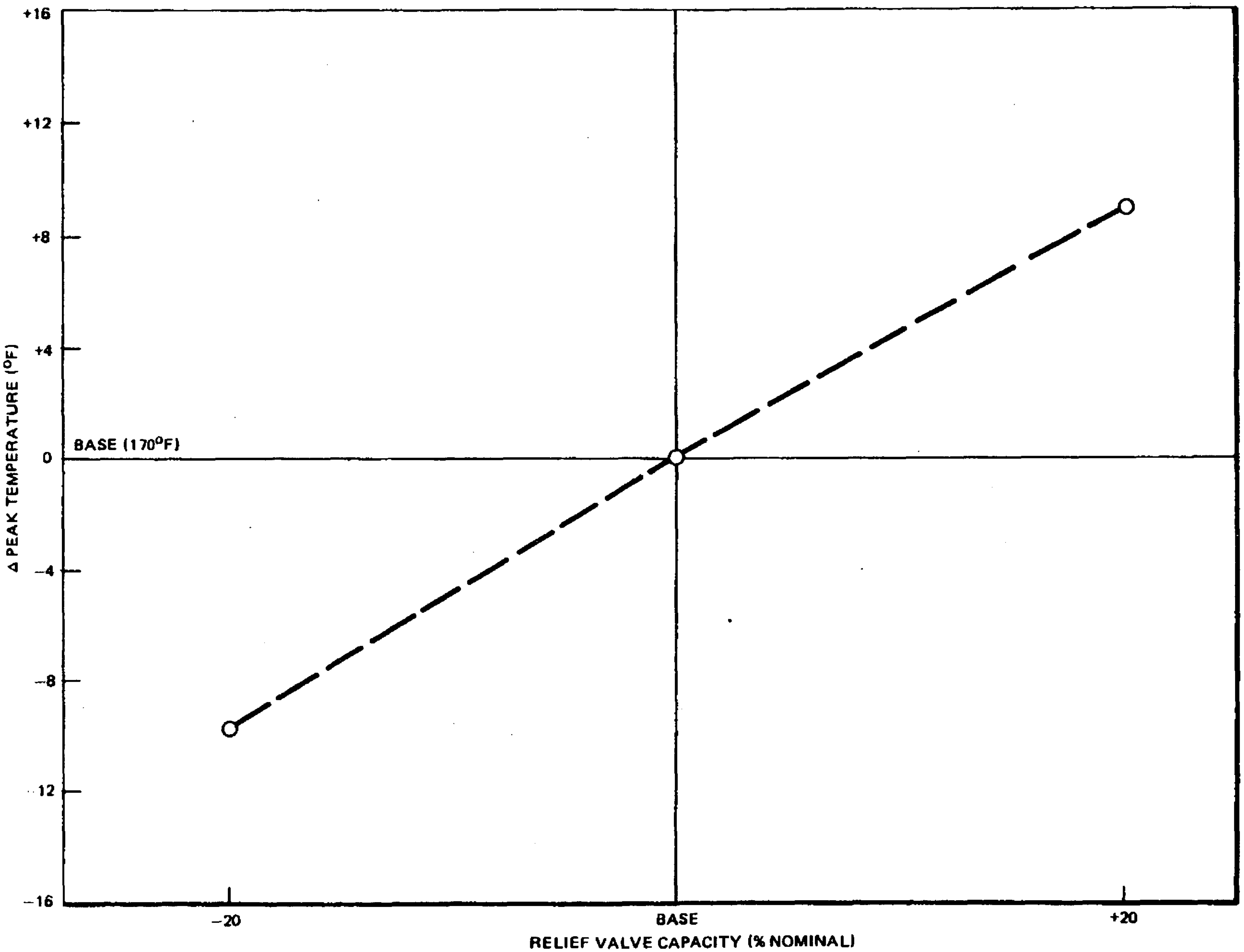


Figure 3.3.4.3-8. BWR-6 ATWS With ARI Failure IORV Sensitivity: Peak Suppression Pool Bulk Temperature to Relief Valve Capacity

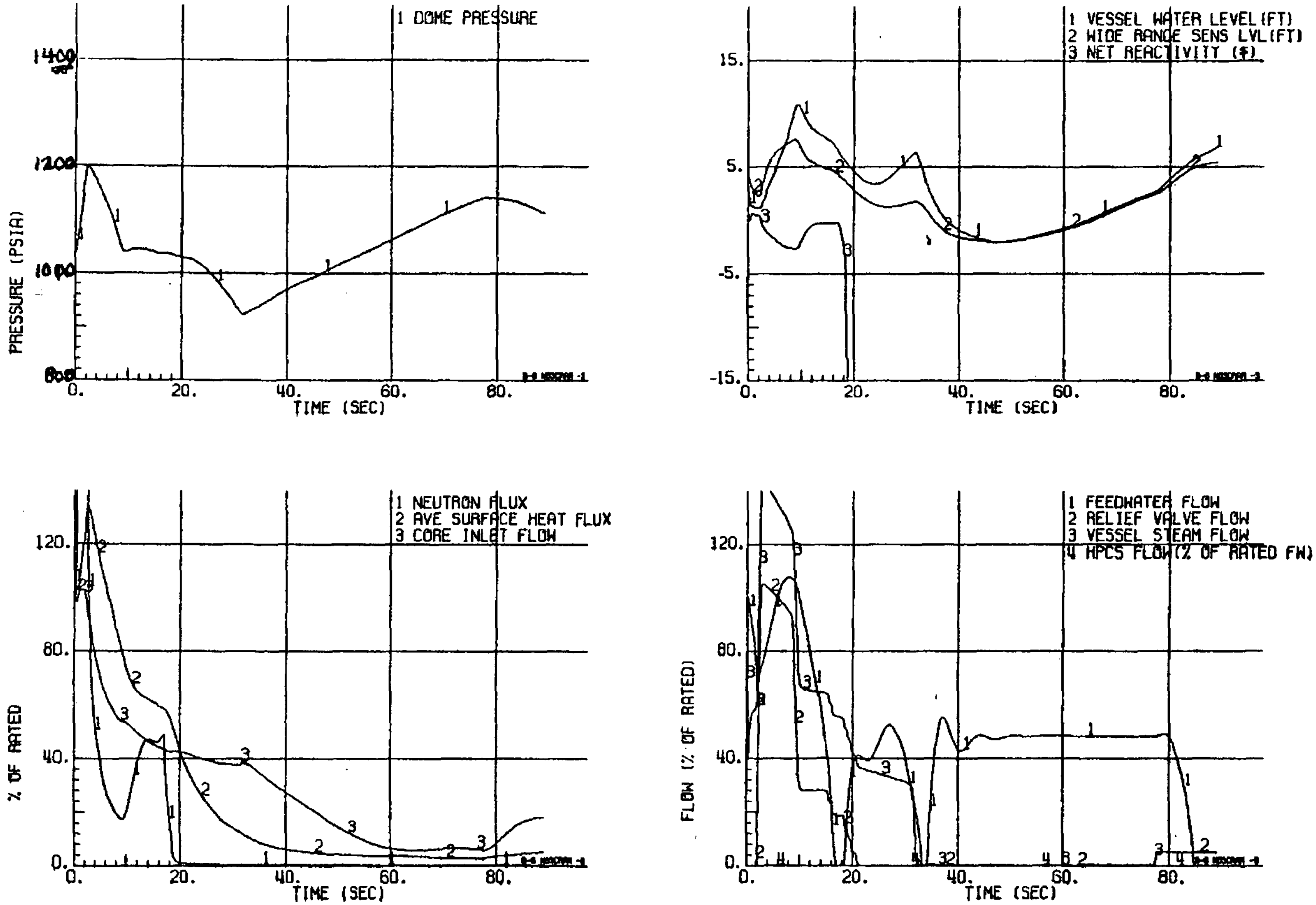


Figure 3.3.5-1. BWR/6 Loss of Condenser Vacuum, ARI

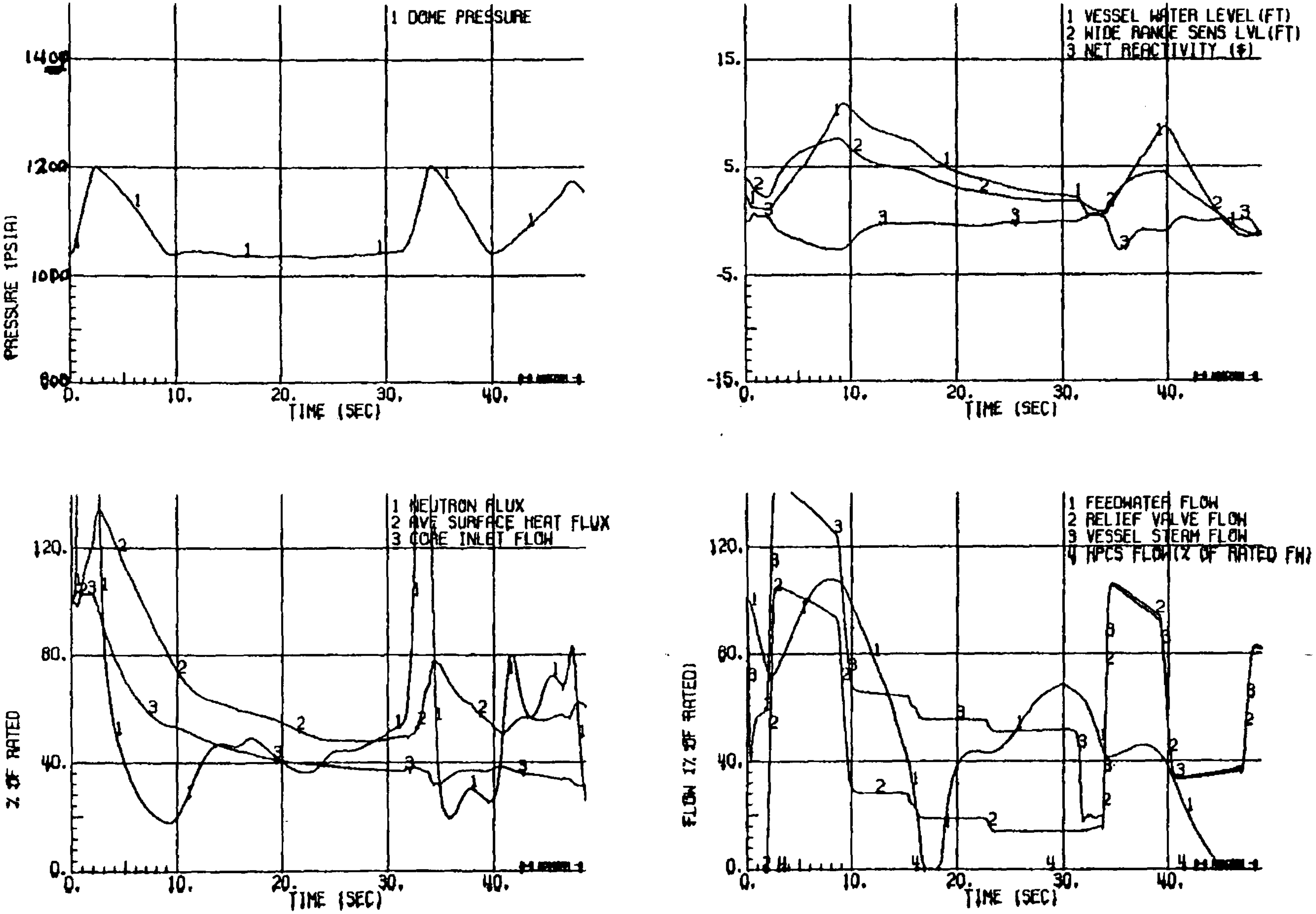


Figure 3.3.5-2. BWR/6 Loss of Condenser Vacuum, ARI Failure

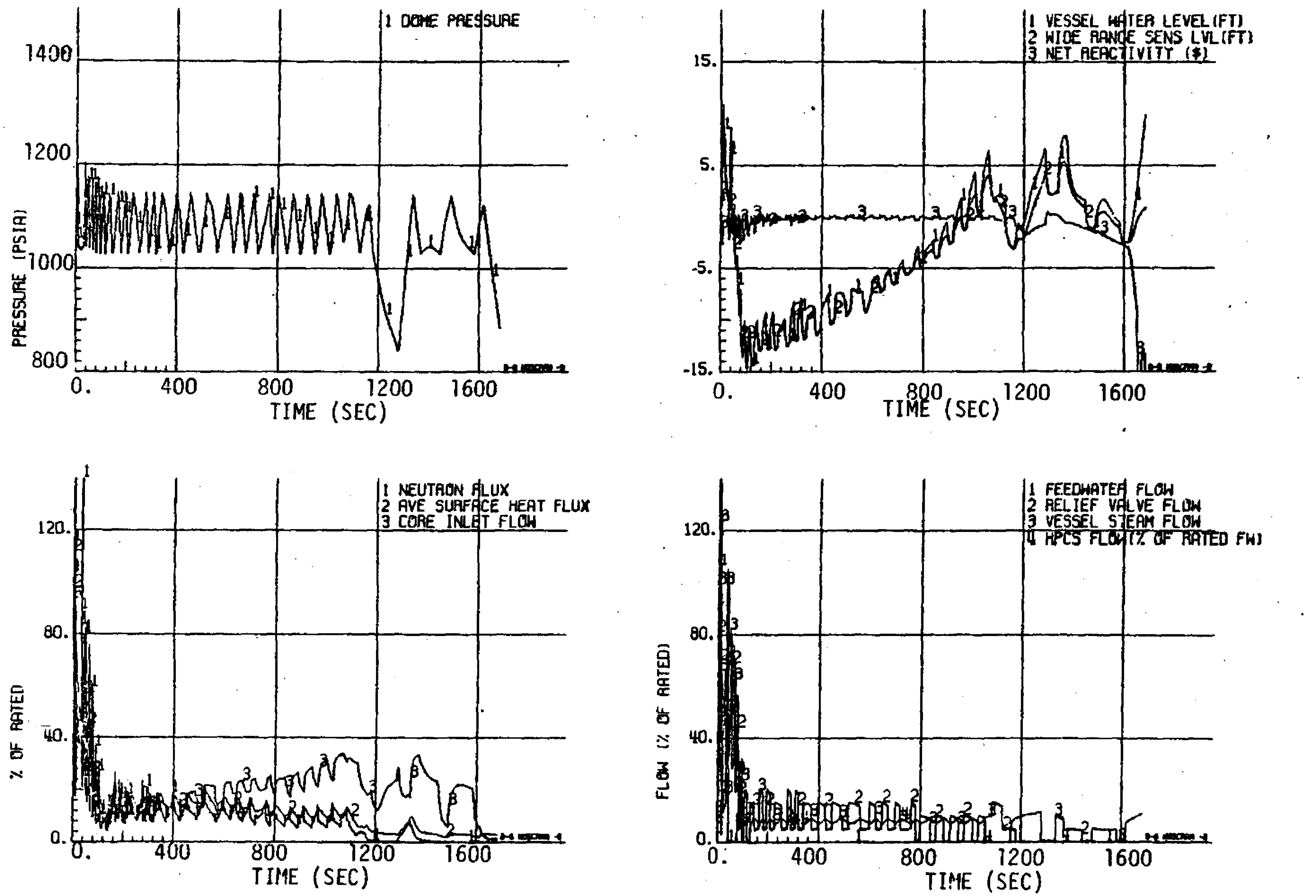


Figure 3.3.5-3. BWR/6 Loss of Condenser Vacuum, ARI Failure

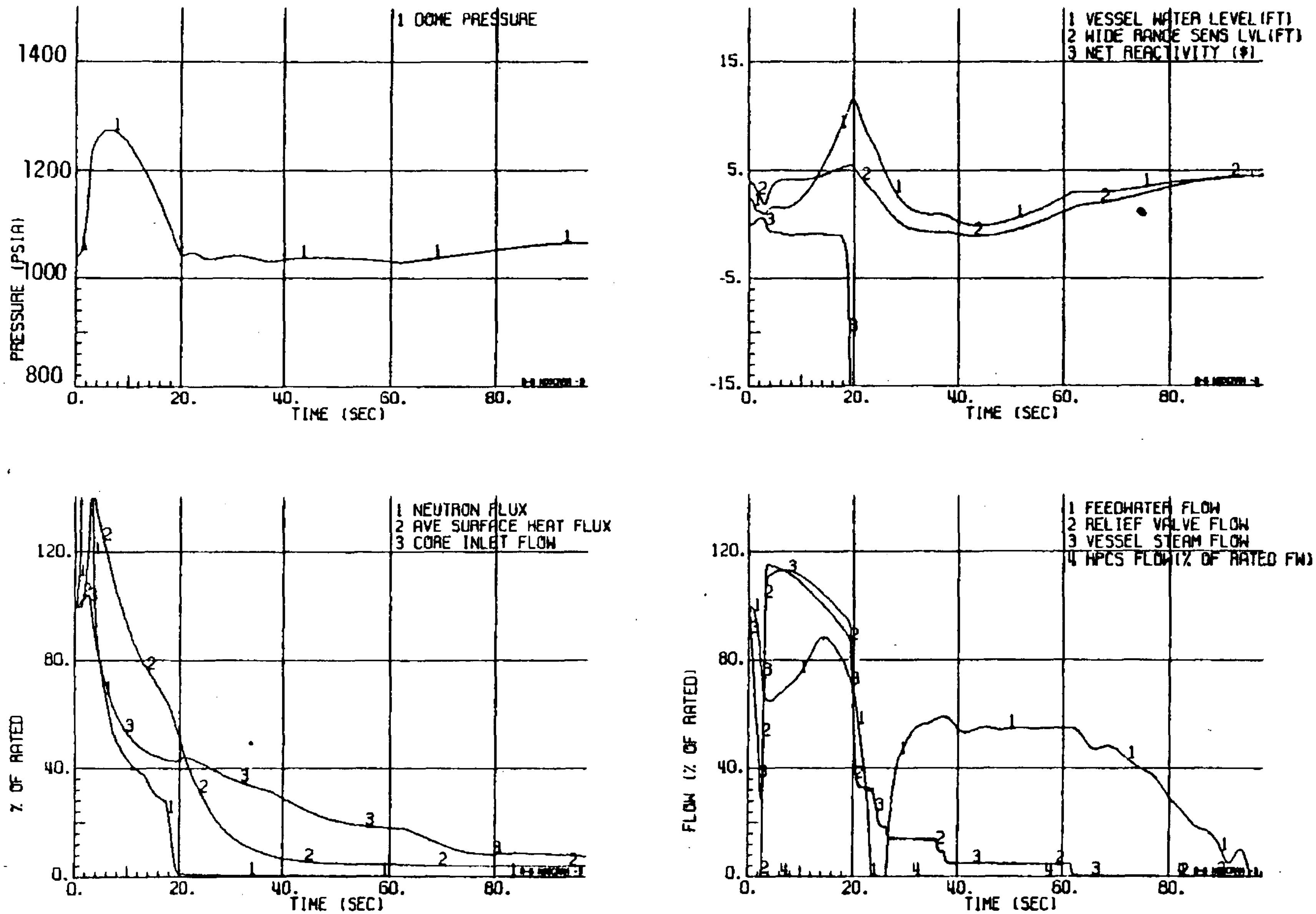


Figure 3.3.6-1. BWR/6 Pressure Regulator Failure Zero Steam Demand, ARI

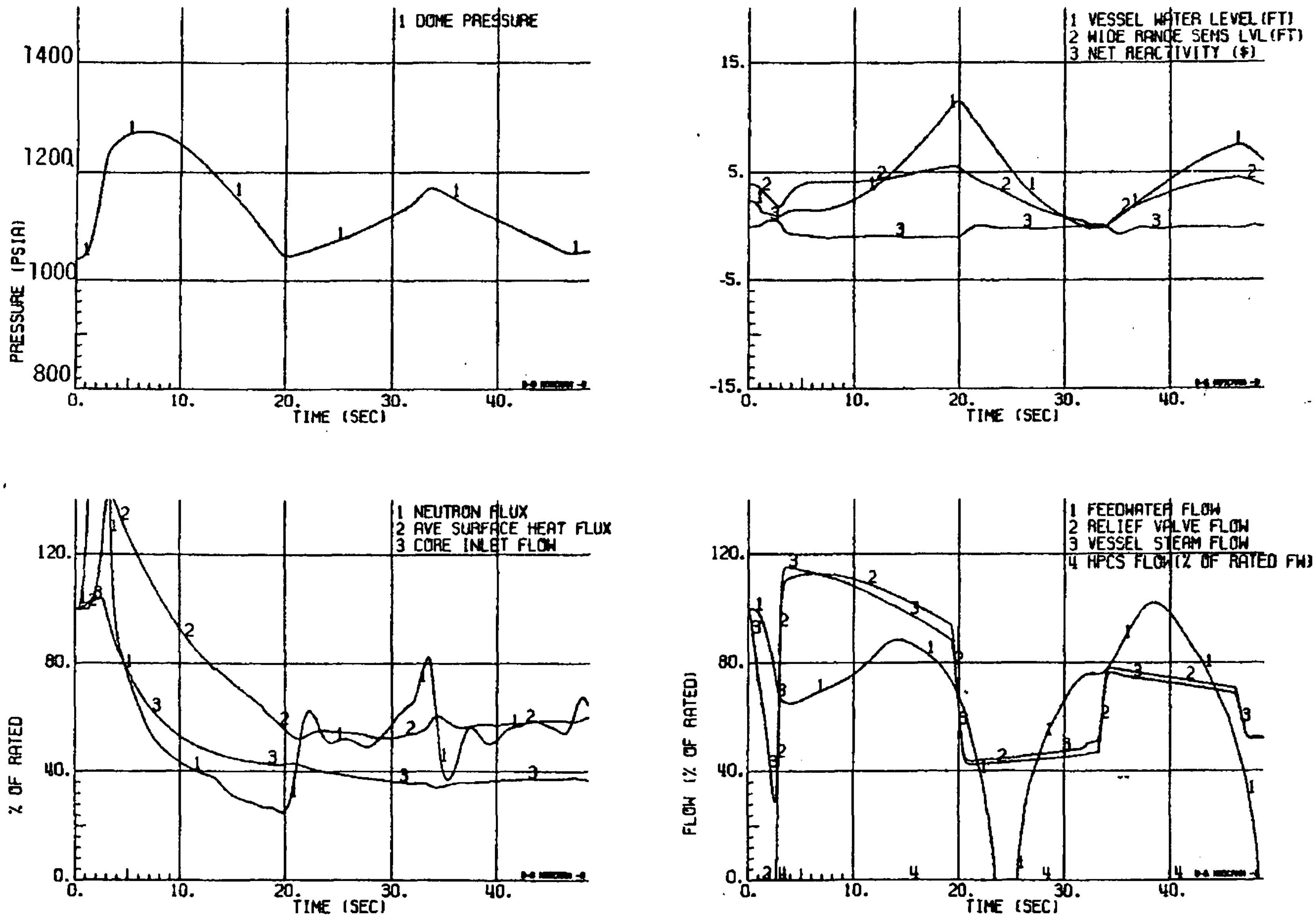


Figure 3.3.6-2. BWR/6 Pressure Regulator Failure - Zero Steam Demand, ARI Failure

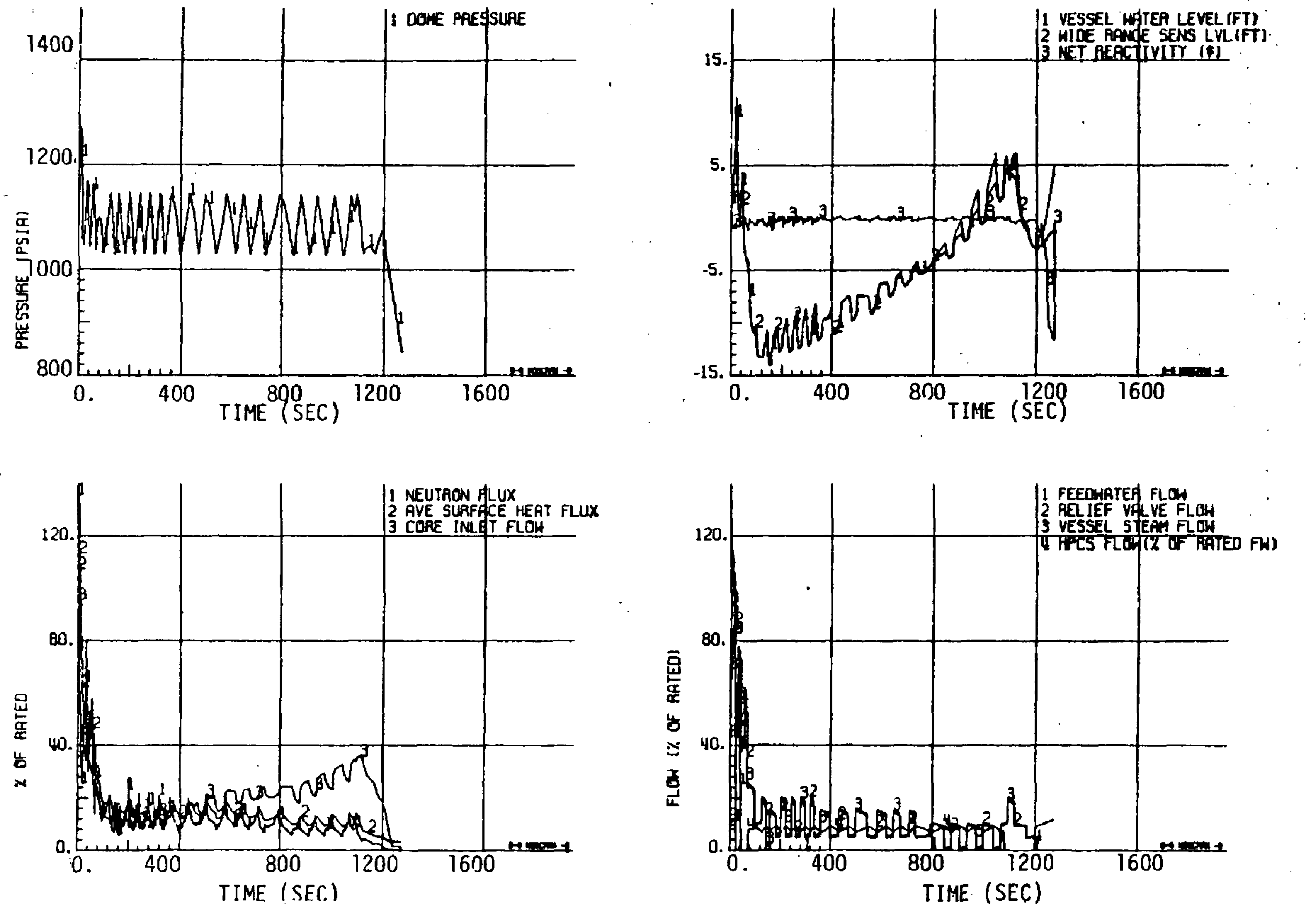


Figure 3.3.6-3. BWR/6 Pressure Regulator Failure Zero Steam Demand, ARI Failure

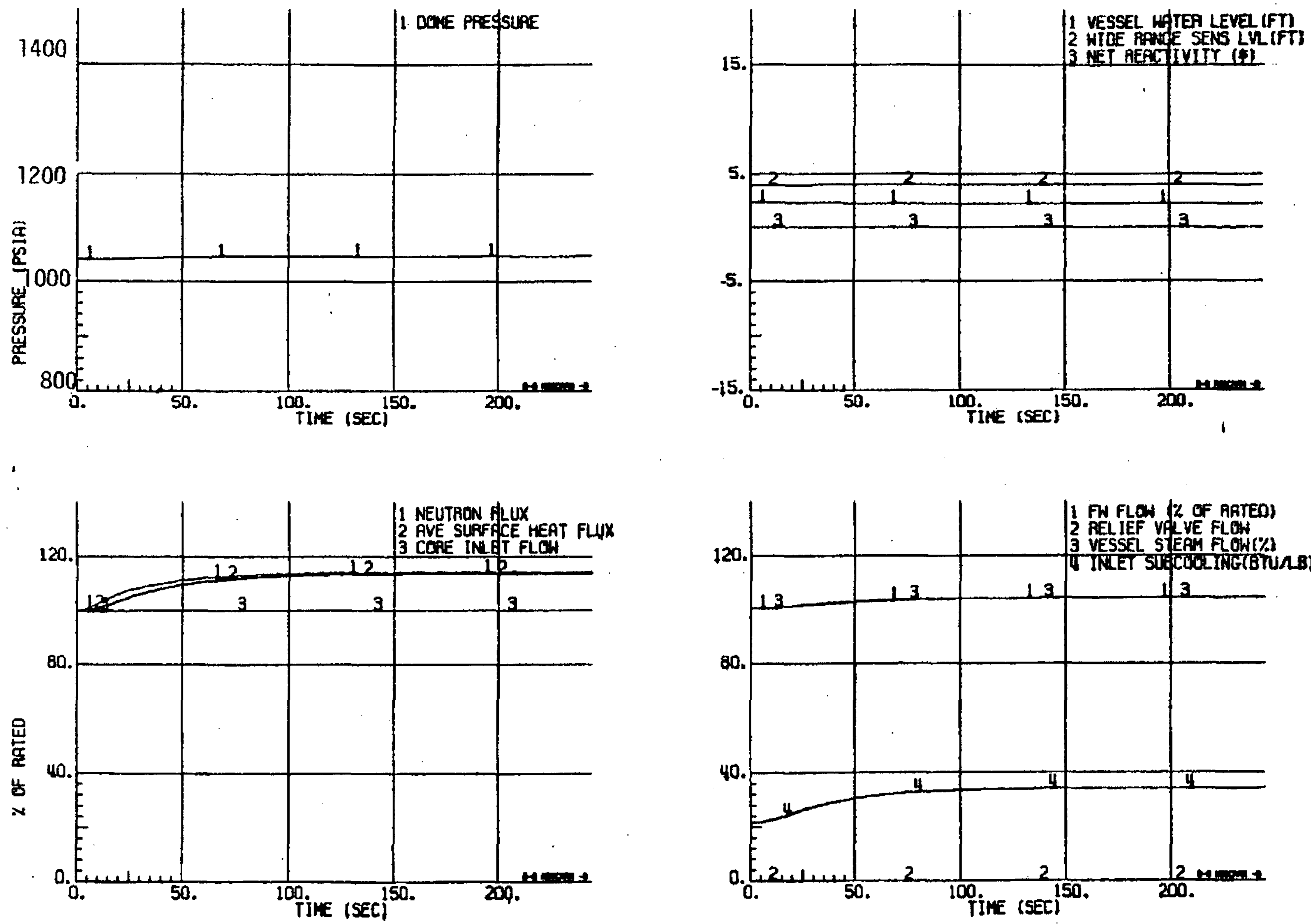


Figure 3.3.7-1. BWR/6 Loss of Feedwater, ARI Failure

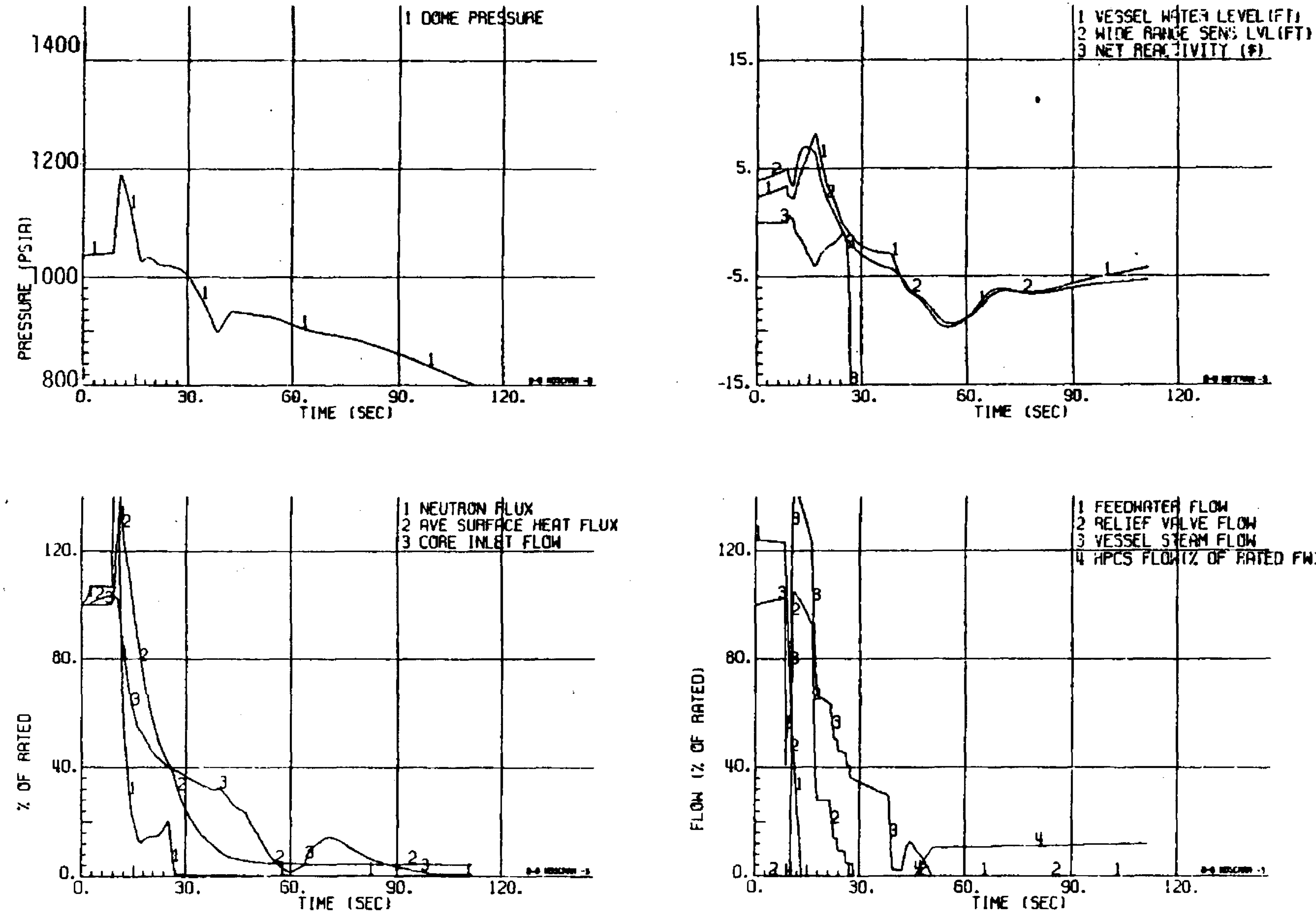


Figure 3.3.8-1. BWR/6 Feedwater Controller Failure Maximum Demand, ARI

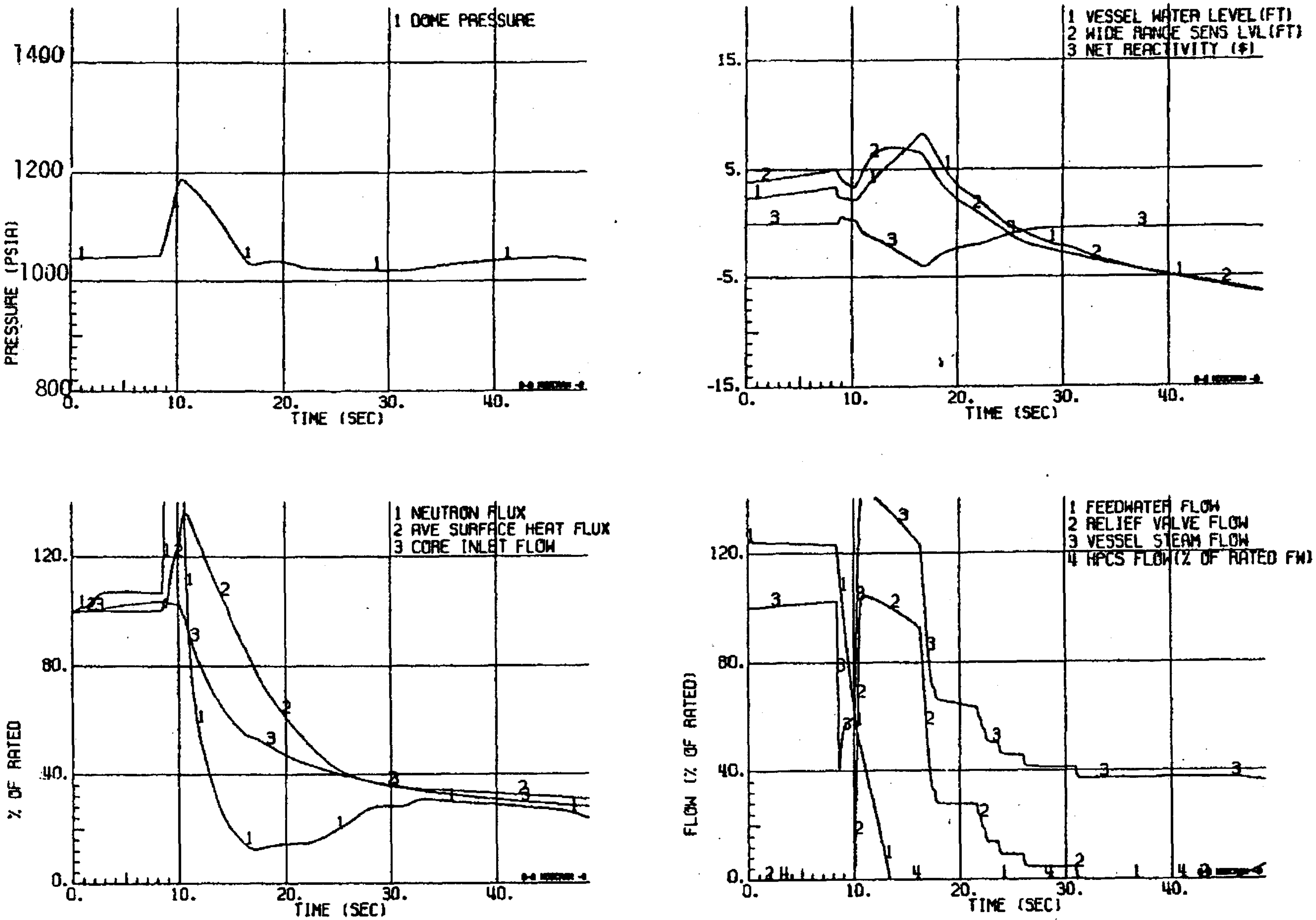


Figure 3.3.8-2. BWR/6 Feedwater Controller Failure Maximum Demand, ARI Failure

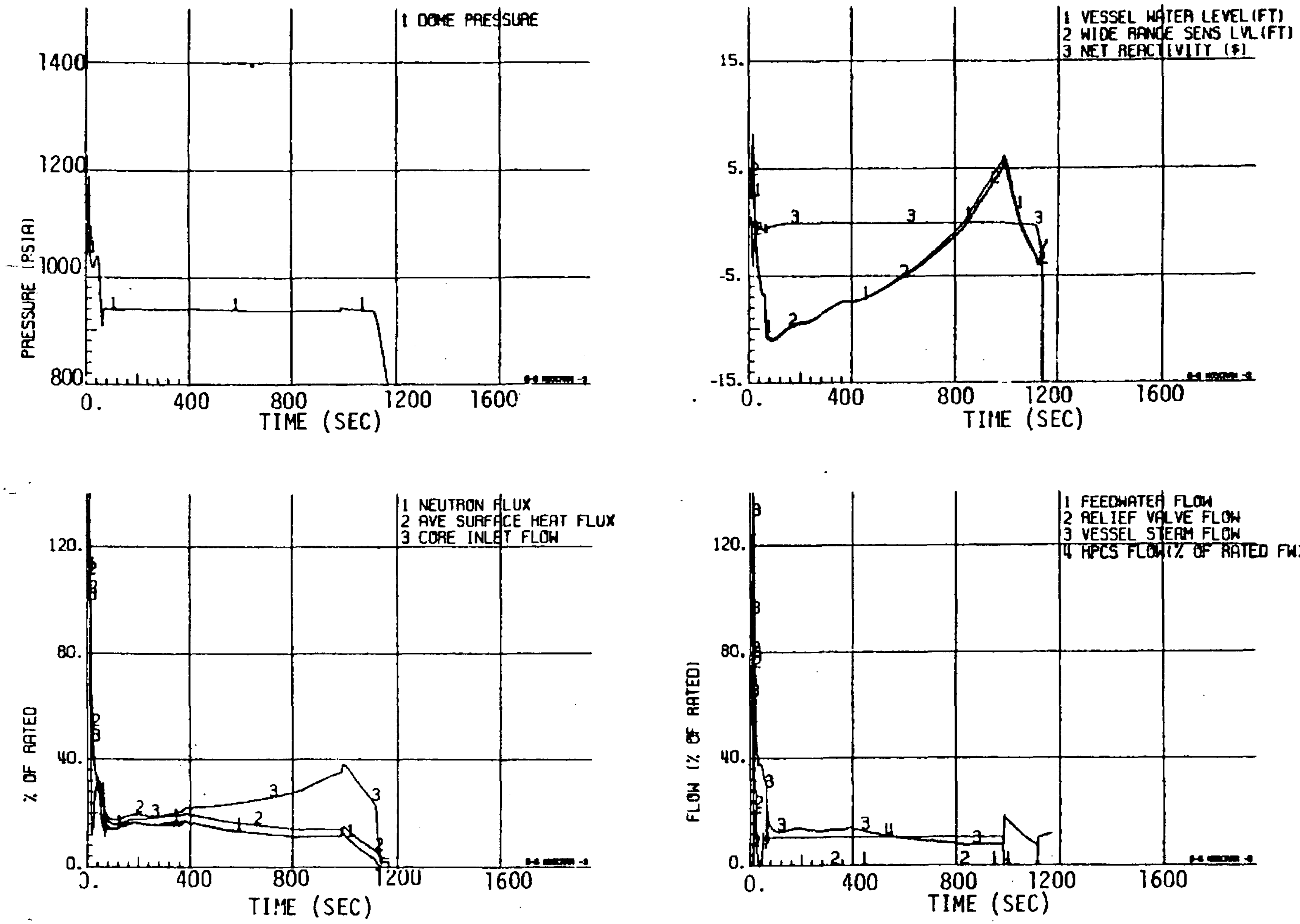


Figure 3.3.8-3. BWR/6 Feedwater Controller Failure Maximum Demand, ARI Failure

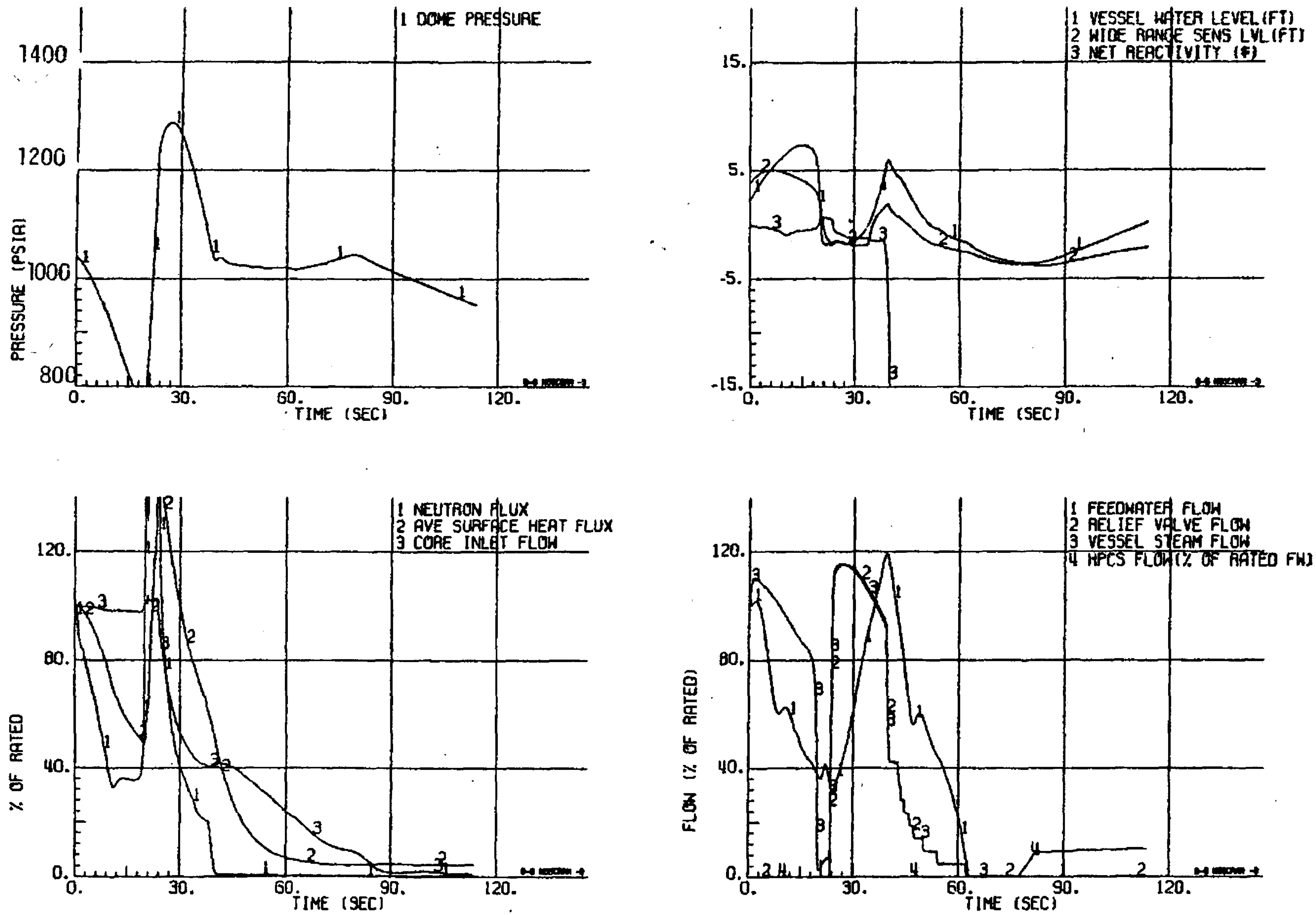


Figure 3.3.9-1. BWR/6 Pressure Regulator Failure Maximum Steam Demand, ARI

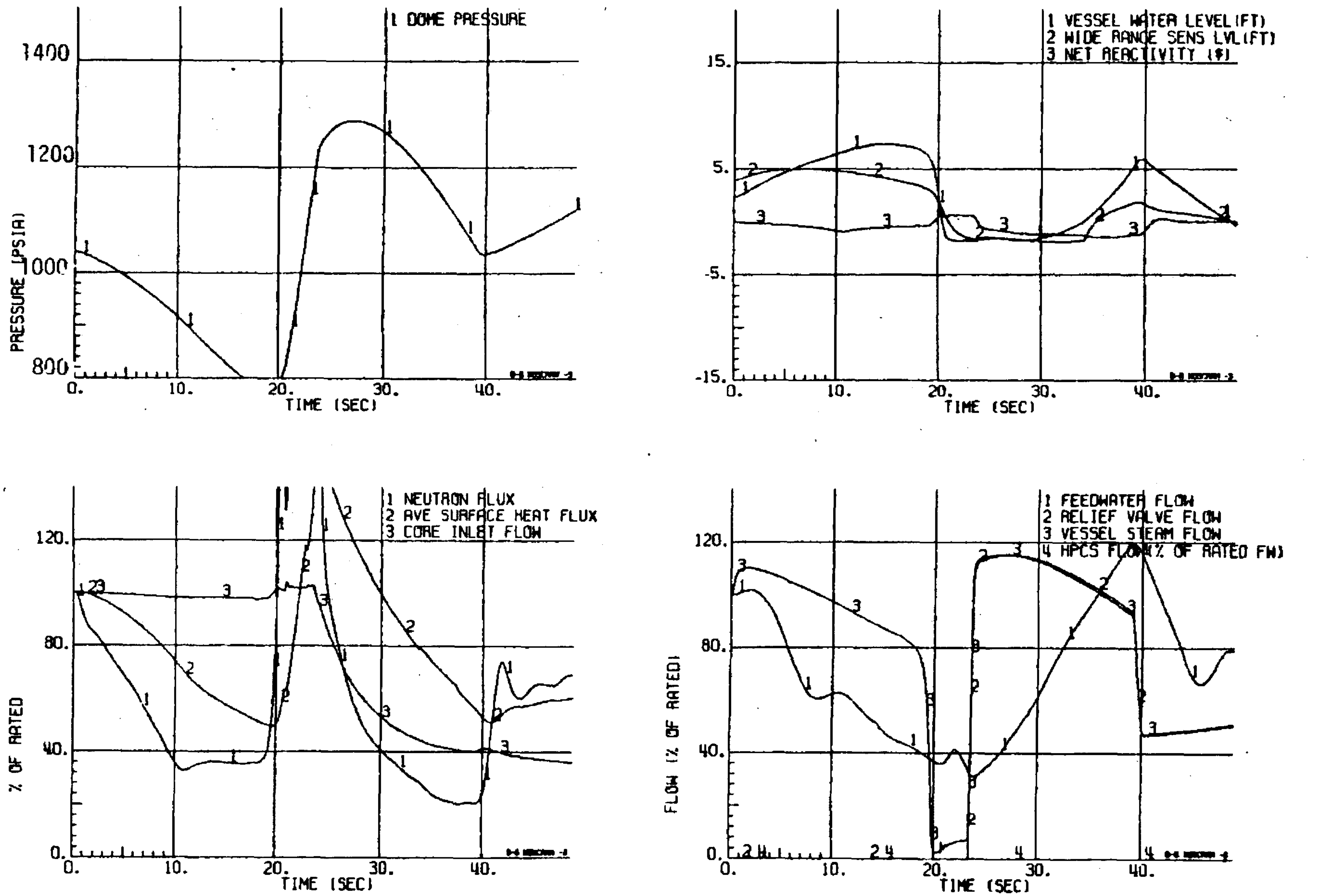


Figure 3.3.9-2. BWR/6 Pressure Regulator Failure Maximum Steam Demand, ARI Failure

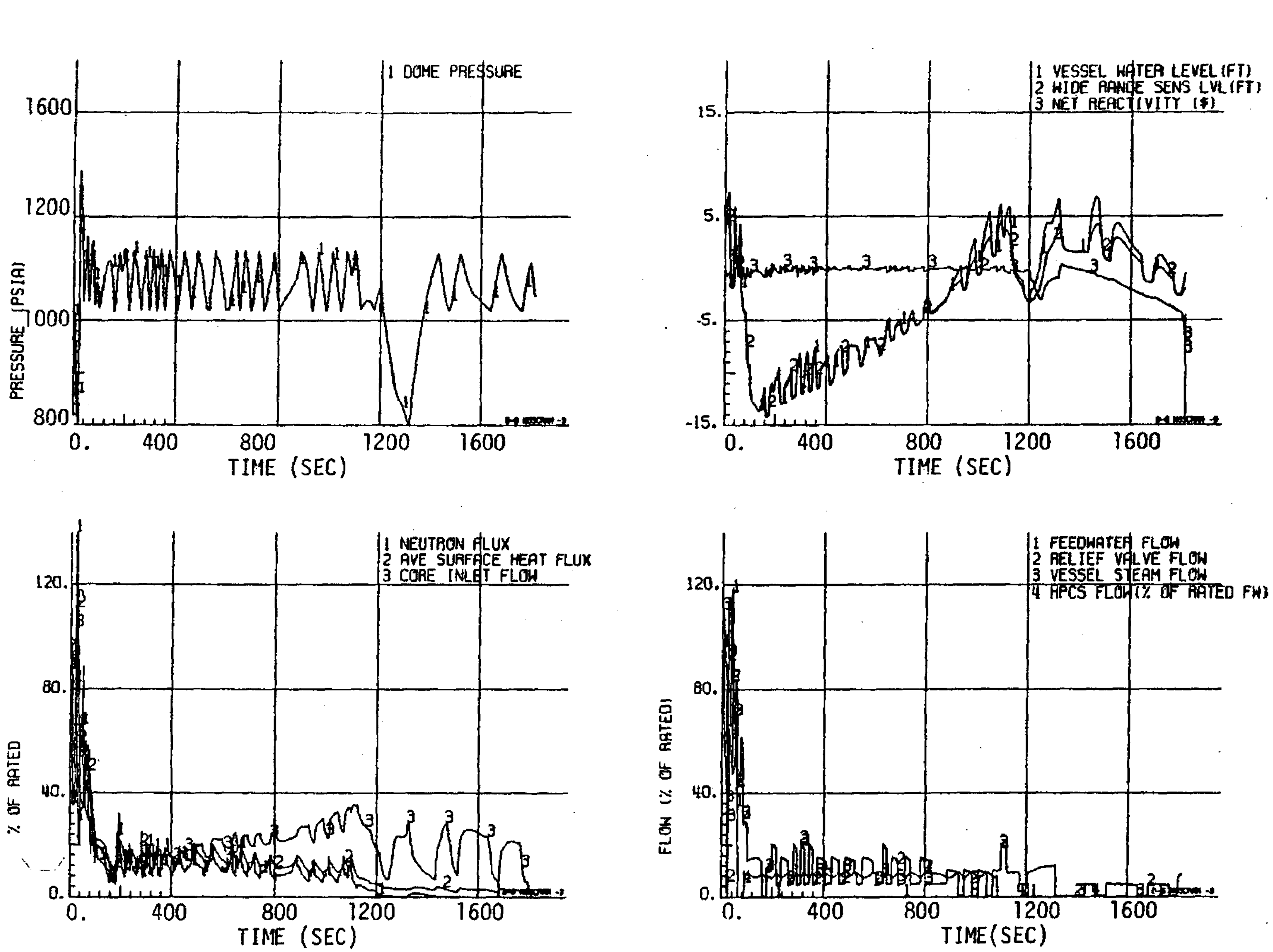


Figure 3.3.9-3. BWR/6 Pressure Regulator Failure Maximum Steam Demand, ARI Failure

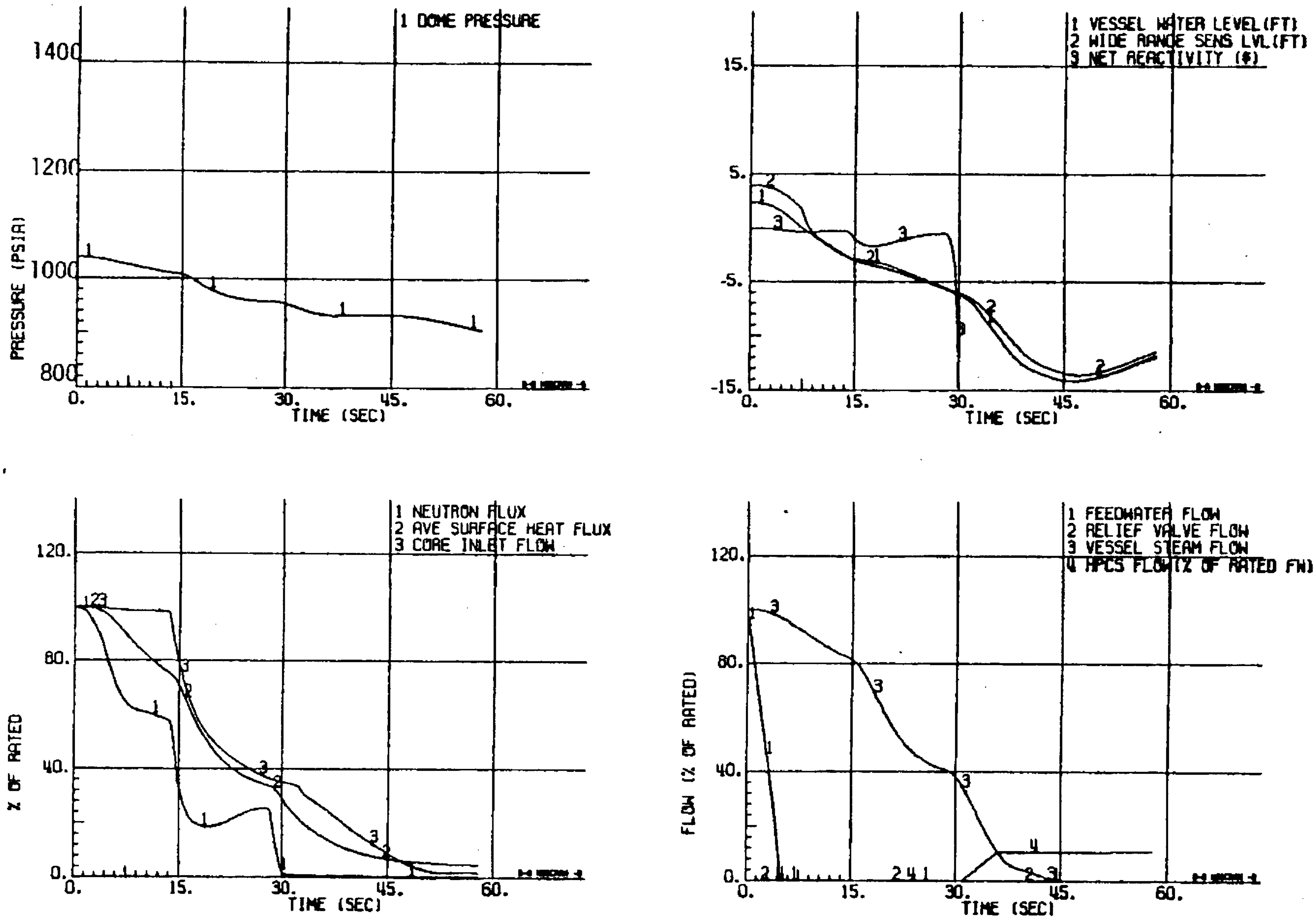


Figure 3.3.10-1. BWR/6 Loss of Normal Feedwater Flow, ARI

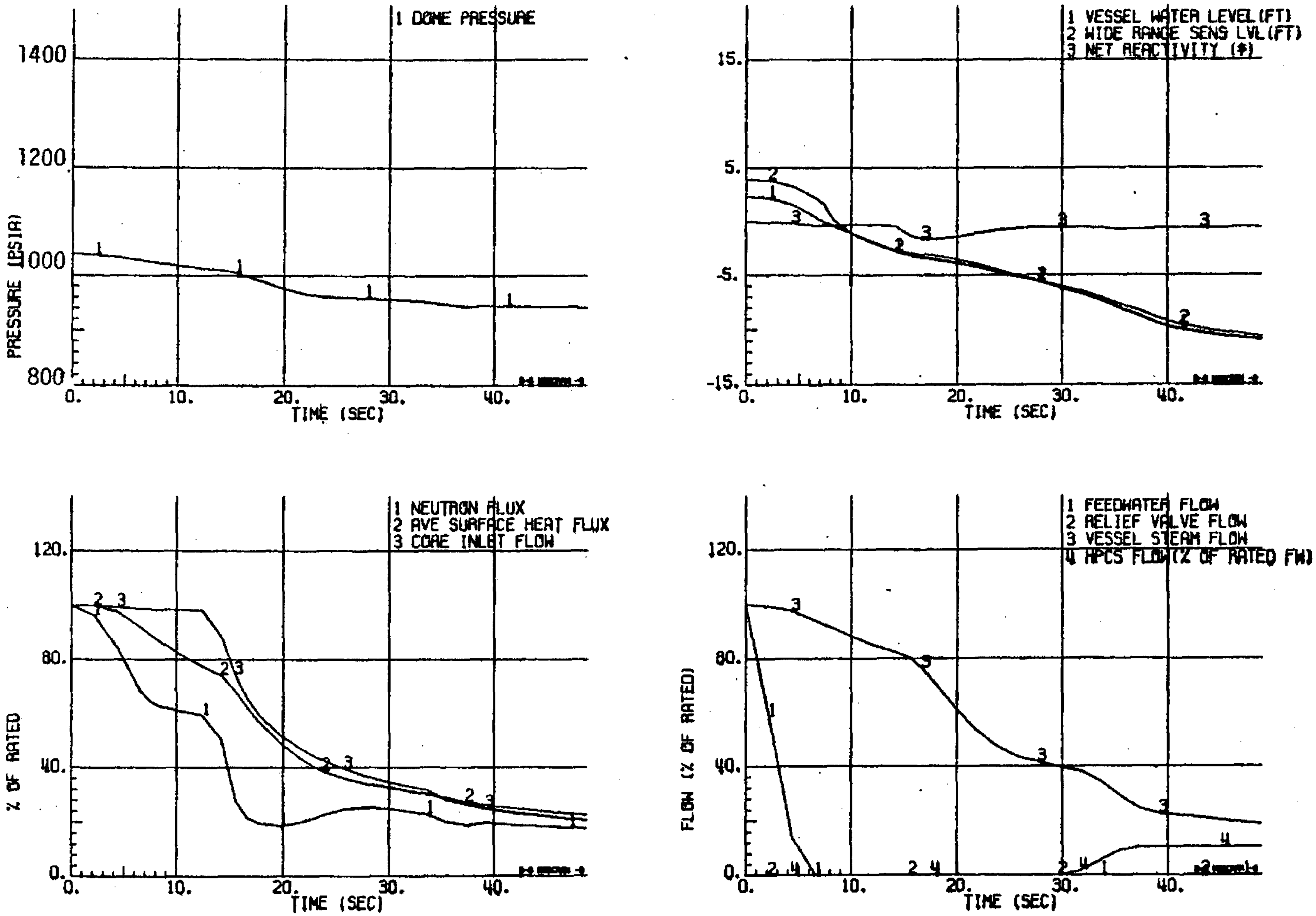


Figure 3.3.10-2. BWR/6 Loss of Normal Feedwater Flow, ARI Failure

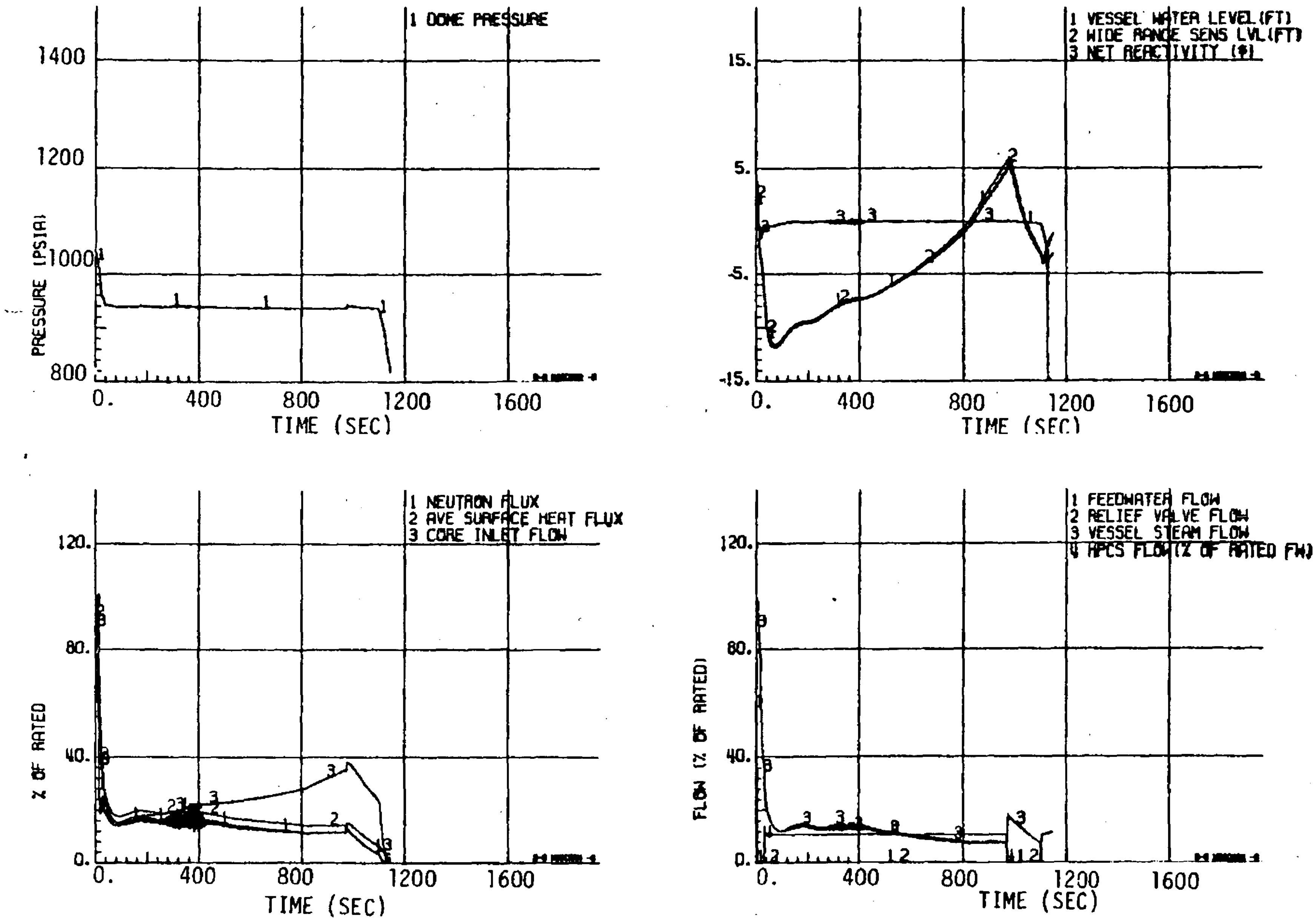


Figure 3.3.10-3. BWR/6 Loss of Normal Feedwater Flow, ARI Failure

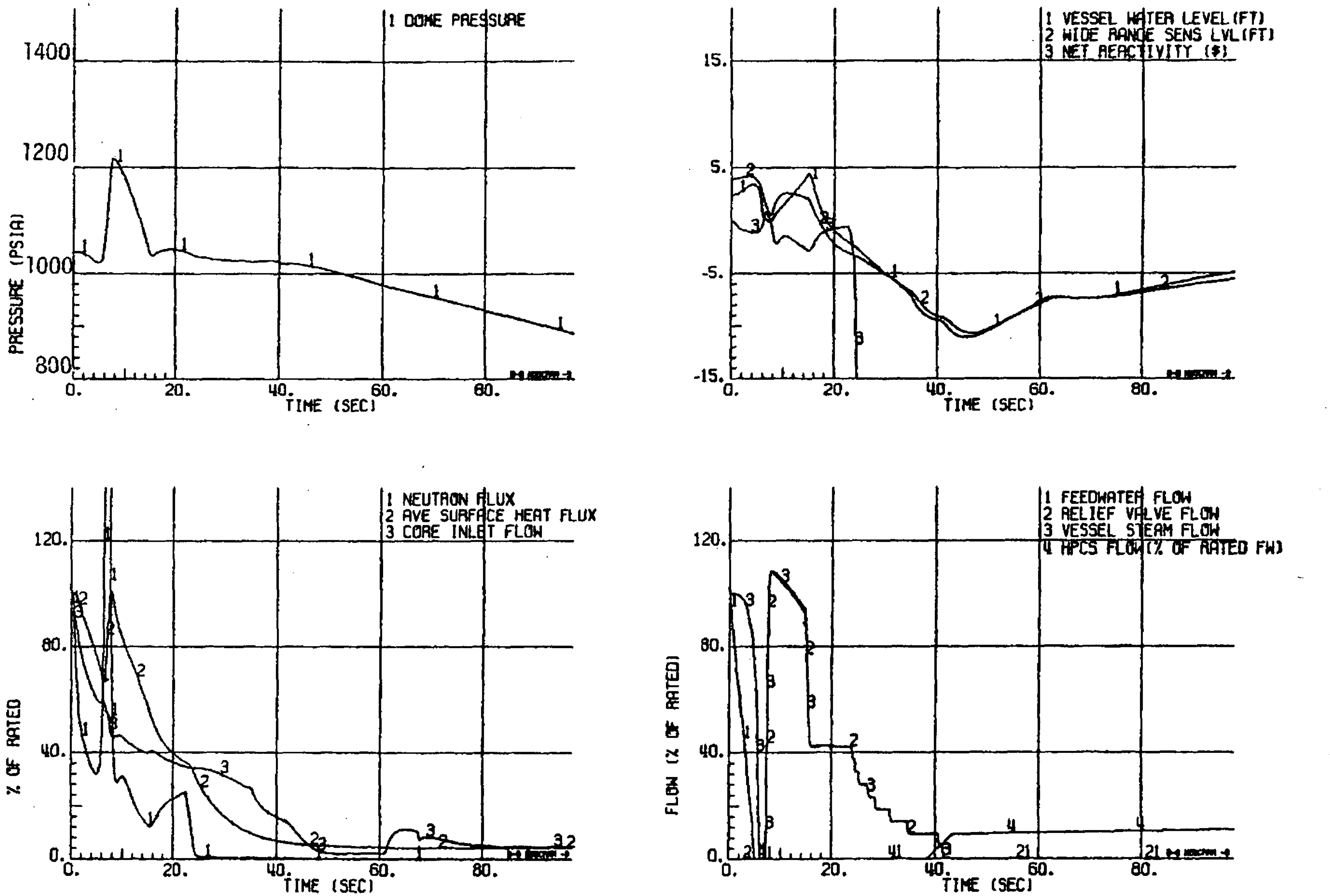


Figure 3.3.11-1. BWR/6 Loss of Normal AC Power, ARI

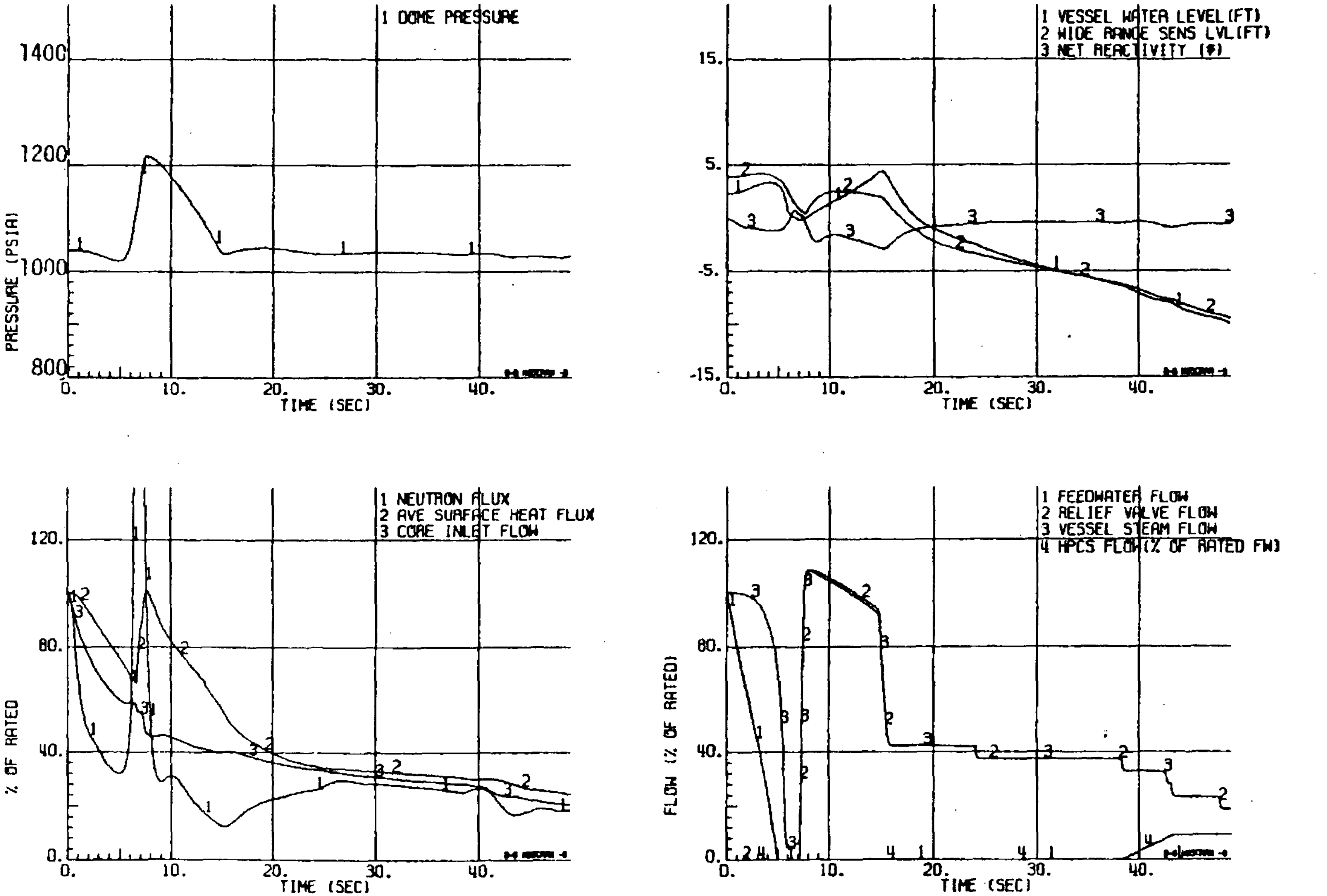


Figure 3.3.11-2. BWR/6 Loss of Normal AC Power, ARI Failure

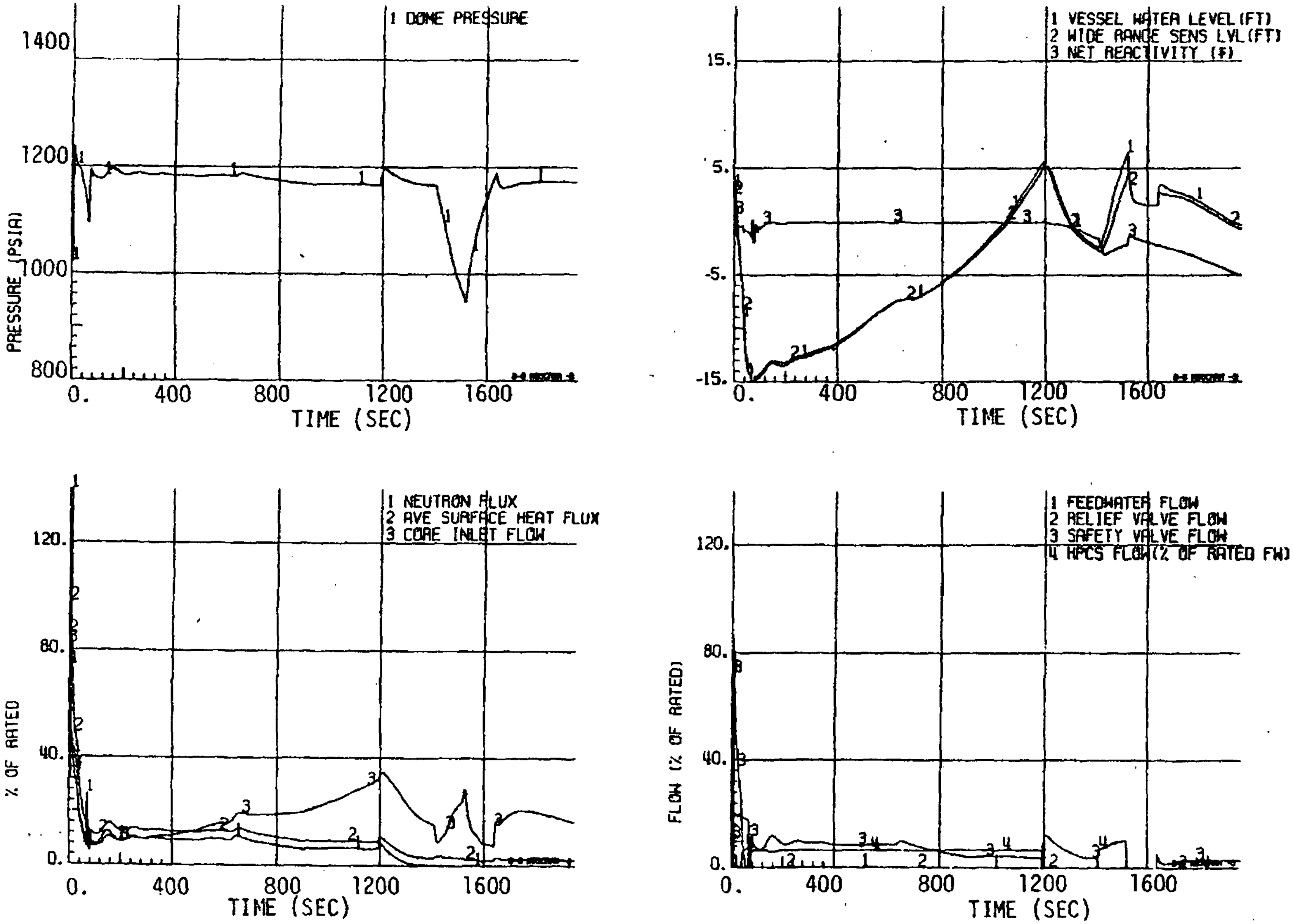


Figure 3.3.11-3. BWR/6 Loss of Normal AC Power, ARI Failure

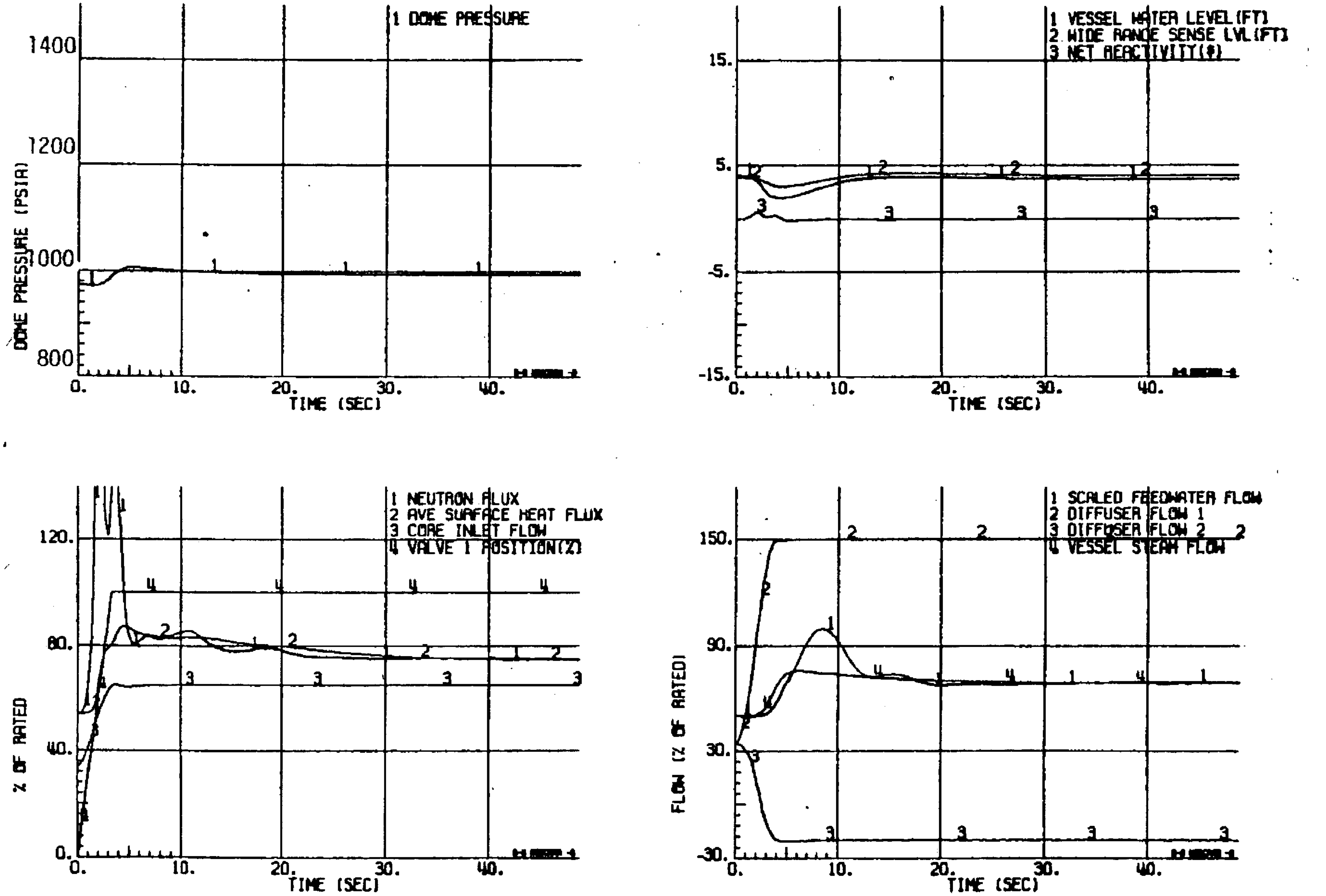


Figure 3.3.12-1. BWR/6 Recirculation Flow Controller Failure, ARI Failure

Table 3.4.1-1

COMPARISON OF PEAK VALUES FROM REDY/ODYN MSIV CLOSURE AND TURBINE TRIP CASES (BWR/6-238/748)

Table 3.4.2-1

COMPARISON OF PEAK VALUES FROM REDY/ODYN TURBINE TRIP
WITH BYPASS FAILURE CASES (BWR/6-238-748)

Figure 3.4.1-1. ATWS MSIV Closure - ARI Failure (REDY)

Figure 3.4.1-2. ATWS MSIV Closure - ARI Failure (ODYN)

Figure 3.4.1-3. ATWS Turbine Trip - ARI Failure

Figure 3.4.1-4. ATWS Turbine Trip - ARI Failure

Figure 3.4.2-1. ATWS Turbine Trip W/O BP - ARI Failure

Figure 3.4.2-2. ATWS Turbine Trip W/O Bypass - ARI Fail-ue

3.5 ANALYSIS OF ATWS TURBINE TRIPS WITH BYPASS FAILURE

In response to NRC staff questions, this infrequent case has also been studied for ATWS conditions. The frequency of occurrence of this event is well below once per plant lifetime and efforts are underway to reclassify the event (in normal SAR documentation) consistent with this experience. For informaation purposes, cases for typical BWR/4, 5 and 6 plants are included here, as requested by the NRC Staff.

3.5.1 BWR/4 Turbine Trip With Bypass Failure

3.5.1.1 Overview of Response Without The Scram

The transient is described in detail in the following sections. Except for the initial 20-30 seconds the characteristics of this improbable event follow the MSIV event very closely. The faster isolation of this case results in a sharp neutron flux peak which momentarily exceeds the MSIV closure case. However, due to the very narrow peak, lower peaks are seen for heat flux and reactor pressures.

Here as in other pressurization cases the power and pressure increases are limited by the action of the S/RV's and RPT. The long term power shutdown is achieved in two ways. ARI employs an alternate design of the protection logic leading to diverse injection of the control rods. In the unlikely event that ARI fails the automated SLCS provides further protection and shutdown capability.

3.5.1.2 Sequence of Events For the BWR/4 Turbine Trip With Bypass Failure

The listing of significant events during this event is provided in Table 3.5.1-1. Results for both cases - with ARI and also assuming its failure are presented.

This highly improbable event starts with an unexpected closure of turbine stop valves (within about 0.1 second) accompanied by failure of the turbine bypass valves to open. The beginning of this event is much like the MSIV closure case in that both result in isolation of the reactor system. However, this event sees a faster isolation because the turbine stop valves are assumed to close within 0.1 seconds as compared to a 4 second nominal MSIV closure time.

This results in a higher neutron flux peak for this event but due to the cushioning effect of the main steamline which absorbs some of the pressure shock the peak pressure and fuel heat flux are lower in this case and they combine to give a lower integrated power peak. Figure 3.5.1-1 and 3.5.1-2 show the initial portions of the event for the more likely plant ATWS transient in which ARI provides a diverse logic path to quickly shut down the unit, and the case in which ARI also fails and the automated SLCS is called upon to shut down the plant.

In each case the initial power and pressure increases are the same neutron flux reaches 655% NBR near 1 second (compared to 527% in MSIV case), fuel average heat flux reaches 138% NBR near 2 seconds (compared to 143% in MSIV case). Some fuel may experience boiling transition. The peak pressure occurs at vessel bottom and is 1267 psig at about 6 seconds (compared to (1296 psig in the MSIV case). The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to the automatic action of RPT which is initiated when vessel dome pressure exceeds 1150 psig and the relieving action of the S/RV's which all open then start reclosing near 23 seconds.

After the first 20-30 seconds the characteristics of this event follow the MSIV case event very closely. Because of similarities between these two events, Section 3.1.1 also represents the long term behavior of the turbine trip with bypass failure transient. Figure 3.5.1-3 shows the long term behavior as predicted for this event.

Peak suppression pool bulk temperature is slightly greater than the MSIV closure case, reaching 191°F for the ARI-failure case about 25 minutes after the start of the event. The corresponding containment pressure is 11.4 psig. With ARI, the results are far less severe. All values are within the containment limits.

3.5.2 BWR/5 Turbine Trip With Bypass Failure

3.5.2.1 Overview of Response Without Scram

The transient is described in detail in the following sections. Except for the initial 20-30 seconds the characteristics of this improbable event follow the MSIV event very closely. The faster isolation of this case results in a sharp neutron flux peak which momentarily exceeds the MSIV closure case. However due to the very narrow peak, lower heat flux, reactor pressures, and integrated power peaks are seen.

Here as in other pressurization cases the power and pressure increases are limited by the action of the S/RV's and RPT. The long term power shutdown is achieved in either of two ways. ARI employs an alternate design of the protection logic leading to diverse insertion of the control rods. In the event that ARI also fails, the automated SLCS provides further protection and shutdown capability.

3.5.2.2 Sequence of Events for BWR/5 Turbine Trip With Bypass Failure

The listing of significant events during this event is provided below. Results for both cases, with ARI and also assuming its failure, are presented.

This highly improbable event starts with an unexpected closure of turbine stop valves (within 0.1 second) accompanied by failure of the turbine bypass valves to open. The beginning of this event is much like the MSIV closure case in that both transients result in isolation of the reactor system. However this event sees a faster isolation because the turbine stop valves are assumed to close within 0.1 second as compared to a 4 second nominal MSIV closure time. This results in a higher neutron flux peak for this event but due to the cushioning effect of the main steamline which absorbs some of the pressure shock the peak pressure and fuel heat flux are lower in this case and thus combine to give a lower integrated power peak. Figure 3.5.2-1 and 3.5.2-2 show the initial portions of the event for the more likely plant ATWS transient in which ARI provides a diverse logic path to quickly shut down the reactor, and the case in which ARI also fails and the automated SLCS is called upon to shut the reactor down.

In both cases, the initial power and pressure increases are the same. Neutron flux reaches 643% NBR near 1 second (compared to 614% in MSIV case), fuel average heat flux reaches 138% NBR near 3 seconds (compared to 143% in MSIV case). Some fuel may experience boiling transition. The peak pressure occurs at vessel bottom and is 1230 psig at about 3 seconds (compared to 1247 psig in MSIV case). The transient pressure is limited within the service Level C overpressure limit of 1500 psig. This is due to the automatic action of RPT (which is initiated when vessel dome pressure exceeds 1150 psig) and the relieving action of the S/RV's which all open then start reclosing near 18 seconds.

After the first 20-30 seconds the characteristics of this event follow the MSIV closure event very closely. Because of similarities between these two events, Section 3.2.1 also represents the long term behavior of turbine trip with bypass failure transient. Figure   shows the long term behavior as predicted for this event.

Peak suppression pool bulk temperature is less than MSIV closure reaching 178°F for the ARI-failure case about 28 minutes after the start of the event. The corresponding containment pressure is 9.1 psig with ARI, the results are far less severe. All values are within containment limits.

3.5.3 BWR/6 Turbine Trip With Bypass Failure

3.5.3.1 Overview of Response Without Scram

Except for the initial 20-30 seconds the characteristics of this improbable event follow the MSIV closure event very closely. The faster isolation of this case results in a sharp neutron flux peak which momentarily exceeds the MSIV closure case. However, due to the very narrow peak, lower heat flux, reactor pressure, and integrated power peaks are seen.

Here, as in other pressurization cases, the power and pressure increases are limited by the action of the S/RV's and RPT. The long term power shutdown is achieved in either of two ways. ARI (employing an alternate design of the protection logic) leads to diverse insertion of the control rods. In the event that ARI also fails, the automated SLCS provides further protection and shutdown capability.

3.5.3.2 Sequence of Events for BWR/6 Turbine Trip With Bypass Failure

The listing of significant events during this event is provided below. Results for both cases, with and without ARI, are presented.

This highly improbable event starts with an unexpected closure of turbine stop valves (within 0.1 second) accompanied by failure of the turbine bypass valves to open. The beginning of this event is much like the MSIV closure case in that both transients result in isolation of the reactor system. However, this event sees a faster isolation because the turbine stop valves are assumed to close within 0.1 seconds as compared to a 4 second nominal MSIV closure time. This results in a higher neutron flux peak for this event but due to the cushioning effect of the main steamline which absorbs some of the pressure shock the peak pressure and fuel heat flux are lower in this case and they combine to give a lower integrated power peak. Figure 3.5.3-1 and 3.5.3-2 show the initial portions of the vent for the more likely plant ATWS transient in which ARI provides a diverse logic path to quickly shut down the reactor and the case in which ARI also fails and the automated SLCS is called upon to shut the reactor down.

In both cases, the initial power and pressure increases are the same. Neutron flux reaches 773% NBR near 1 second (compared to 745% in MSIV case), fuel average heat flux reaches 144% NBR near 2 seconds (compared to 147% in MSIV case). Some fuel may experience boiling transition. The peak pressure occurs at vessel bottom and is 1285 psig at about 5 seconds (compared to 1300 psig in MSIV case). The transient pressure is limited within the Service Level C overpressure limit of 1500 psig. This is due to the automatic action of RPT (which is initiated when vessel dome pressure exceeds 1150 psig) and the relieving action of the S/RV's which all open then start reclosing near 18 seconds.

After the first 20-30 seconds the characteristics of this event follow the MSIV closure event very closely. Because of similarities between these two events, Section 3.3.1 also represents the long term behavior of the turbine trip with bypass failure transient. Figure 3.5.3-3 shows the long term behavior as predicted for this event.

Peak suppression pool bulk temperature is less than MSIV closure, reaching 168°F for the ARI-failure case about 28 minutes after the start of the event. The corresponding containment pressure is 7.0 psig. With ARI, the results are far less severe. All values are within the containment limits.

Table 3.5.1-1

BWR/4 TURBINE TRIP WITH BYPASS FAILURE TRANSIENT

	Sequence of Events	Time: With ARI	Time: With ARI Failure
1.	Turbine trips - bypass fails to open - All normal scrams fail	0	0
2.	Pressure and power begin to rise	0	0
3.	Power peaks	1 Second	1 Second
4.	Relief valves lift	1 Second	1 Second
5.	ATWS high pressure setpoint is reached - ARI is initiated - SLCS timed logic is activated	1 Second	1 Second
6.	Some fuel may experience boiling transition	1 Second	1 Second
7.	Pressure peaks	7 Seconds	7 Seconds
8.	ARI control rod insertion completed, eliminating SLCS initiation and feedwater limit actions	21 Seconds	Fails
9.	ATWS logic - Initiates feedwater limit	N/A	30 Seconds
10.	Reactor water level drops to Level 2 - Initiates containment isolation - Initiates HPCI and RCIC	39 Seconds	42 Seconds
11.	Feedwater flow runback to lower limit value		45 Seconds
12.	HPCI and RCIC flow begins	59 Seconds	62 Seconds
13.	Reactor water level reaches minimum and begins to rise. (Fuel always remains covered)	69 Seconds	5 Minutes

Table 3.5.1-1

BWR/4 TURBINE TRIP WITH BYPASS FAILURE TRANSIENT (Continued)

	Sequence of Events	Time With ARI	Time With ARI Failure
14.	ATWS logic timer completed - Initiates SLCS	N/A	2 Minutes
15.	Liquid control flow reaches core	N/A	3 Minutes
16.	RHR flow begins (pool cooling)	$\geq$11	11 Minutes
17.	Hot shutdown achieved	21 Seconds	18 Minutes
18.	Peak containment pressure and pool bulk temperature occur	>2 Hours	25 Minutes

Table 3.5.2-1

BWR/5 TURBINE TRIP WITH BYPASS FAILURE TRANSIENT

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Turbine trips - bypass fails to open - all normal scrams fail	0	0
2.	Pressure and power begin to rise	0	0
3.	Relief valves lift	1 Second	1 Second
4.	Power peaks	1 Second	1 Second
5.	Some fuel may experience boiling transition	1 Second	1 Second
6.	ATWS high pressure setpoing is reached - ARI is initiated - SLCS time logic is activated	2 Seconds	2 Seconds
7.	Pressure peaks	3 Seconds	3 Seconds
8.	ARI control rod insertion completed, eliminating SLCS initiation and FW limit actions	21 Seconds	Fails
9.	ATWS logic initiates FW limit	N/A	30 Seconds
10.	Reactor water level drops to Level 2 - Initiates containment isolation - Initiates HPCS and RCIC	33 Seconds	40 Seconds
11.	Feedwater flow run back to lower limit value	45 Seconds	45 Seconds
12.	HPCS and RCIC flow begins	53 Seconds	60 Seconds
13.	ATWS logic timer initiates SLCS	N/A	2 Minutes
14.	Liquid control flow	N/A	3 Minutes
15.	Reactor water level reaches minimum and begins to rise	56 Seconds	5.4 Minutes
16.	RHR flow begins (pool cooling)	$\geq$11 Minutes	11 Minutes
17.	Hot shutdown achieved	21 Seconds	22 Minutes
18.	Peak containment pressure and pool bulk temperature occur	>2 Hours	28 Minutes

Table 3.5.3-1

BWR/6 TURBINE TRIP WITH BYPASS FAILURE TRANSIENT

	Sequence of Events	Time With ARI	Time With ARI Failure
1.	Turbine trips - bypass fails to open - All normal scrams fail	0	0
2.	Pressure and power begin to rise	0	0
3.	Relief valves lift	1 Second	1 Second
4.	ATWS high pressure setpoint is reclosed - ARI is initiated - SLCS time logic is activated	1 Second	1 Second
5.	Power peaks	1 Second	1 Second
6.	Some fuel may experience boiling transition	1 Second	1 Second
7.	Pressure peaks	5 Seconds	5 Seconds
8.	ARI control rod insertion completed, eliminating SLCS initiation and feedwater limit actions	21 Seconds	Fails
9.	ATWS logic timer initiates feedwater limit	N/A	30 Seconds
10.	Feedwater flow run back to lower limit value	45 Seconds	45 Seconds
11.	Reactor water level drops to Level 2 - Initiates containment isolation - Initiates HPCS and RCIC	51 Seconds	53 Seconds
12.	HPCS and RCIC flow begins	65 Seconds	65 Seconds
13.	Reactor water level reaches minimum and begins to rise. (Fuel always remains covered).	76 Seconds	128 Seconds
14.	ATWS logic timer initiates SLCS	N/A	2 Minutes

Table 3.5.3-1

BWR/6 TURBINE TRIP WITH BYPASS FAILURE TRANSIENT (Continued)

		Time	
	Sequence of Events	With ARI	With ARI Failure
15.	Liquid control flow reaches core	N/A	3 Minutes
16.	RHR flow begins (pool cooling)	$\geq$11 Minutes	11 Minutes
17.	Hot shutdown achieved	20 Seconds	23 Minutes
18.	Peak containment pressure and pool bulk temperature occur	>2 Hours	28 Minutes

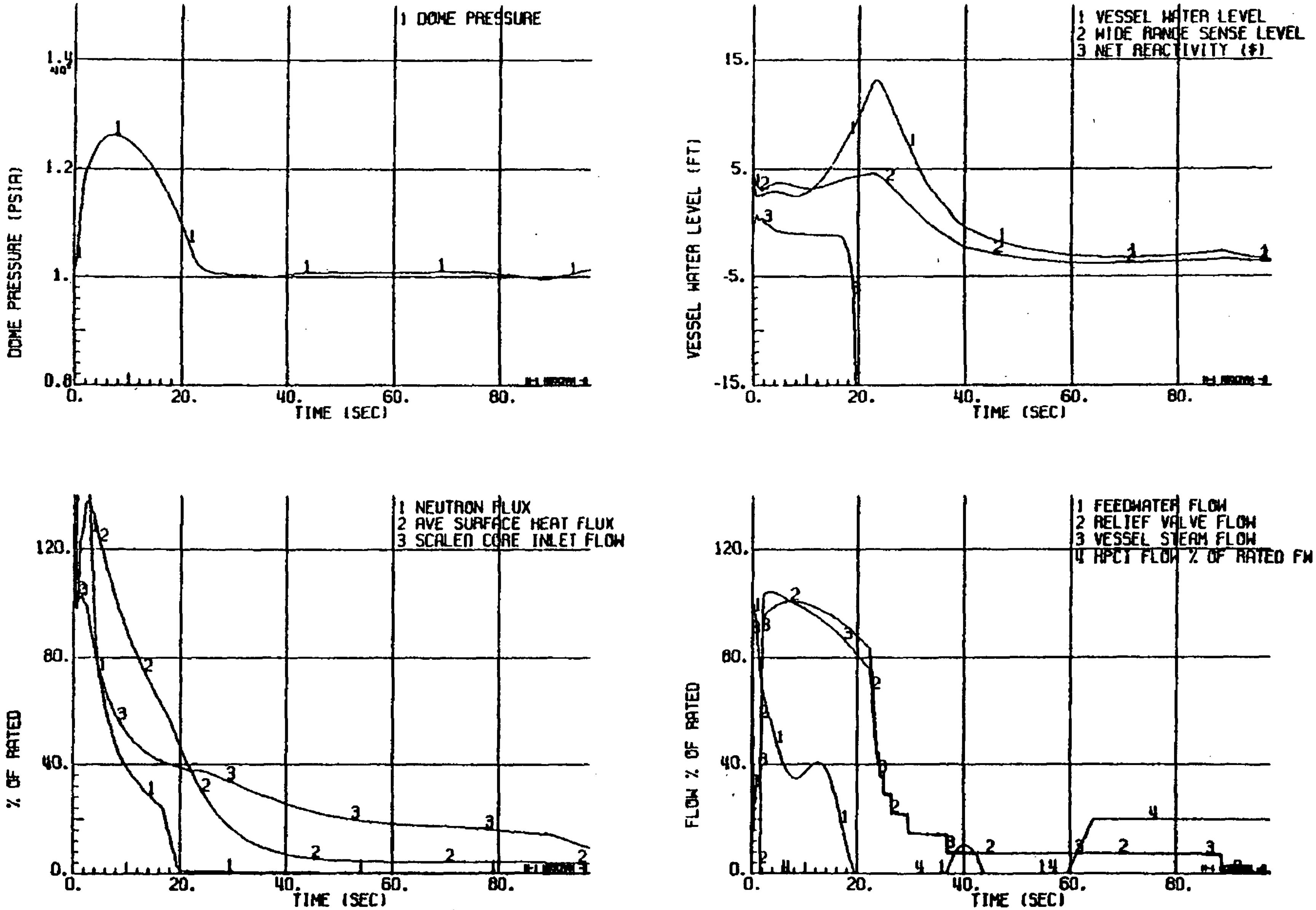


Figure 3.5.1-1. BWR/4 Turbine T) With Bypass Failure - With ARI

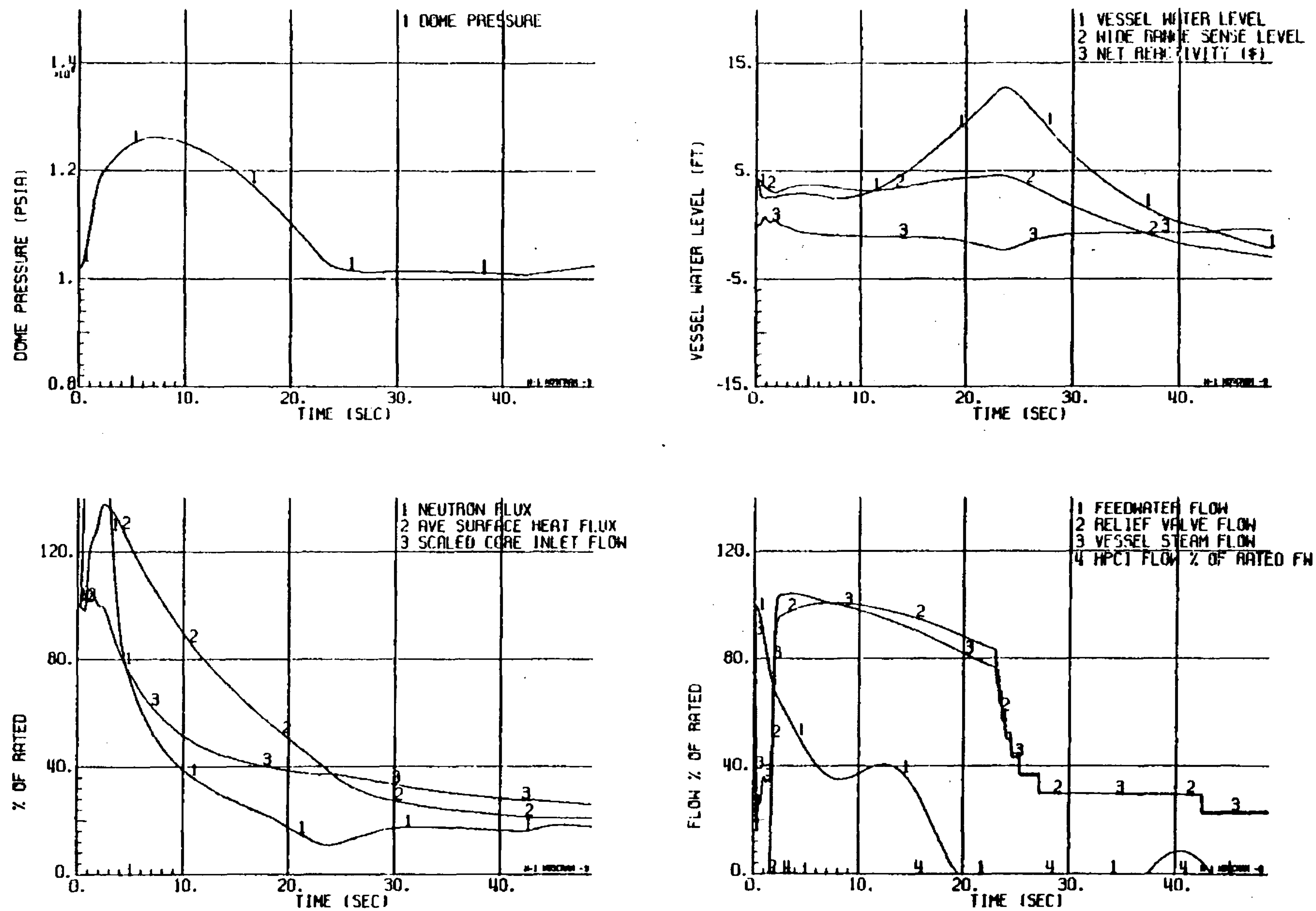


Figure 3.5.1-2. BWR/4 Turbine Trip With Bypass Failure - With ARI Failure

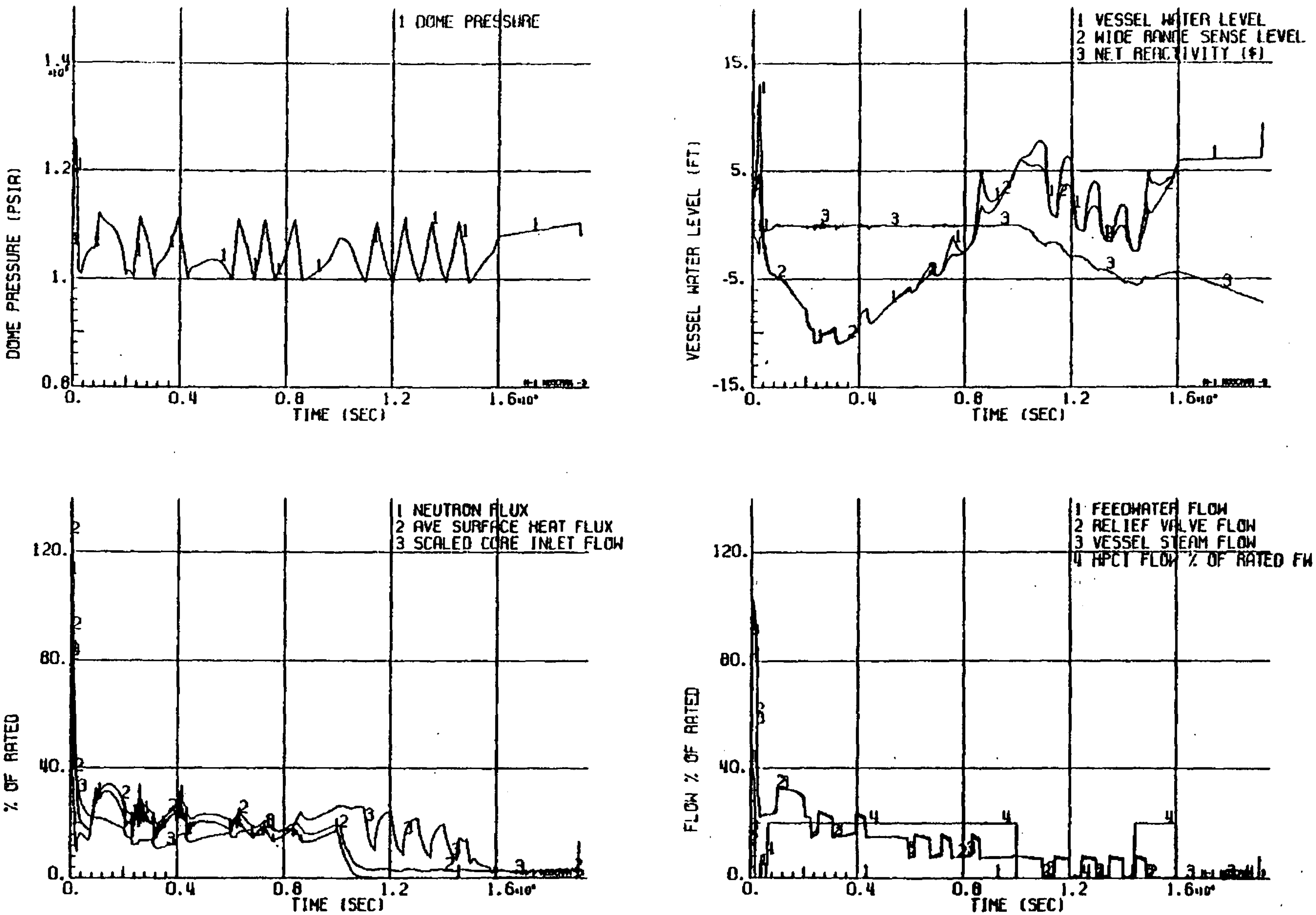


Figure 3.5.1-3. BWR/4 Turbine T..p With Bypass Failure – No ART

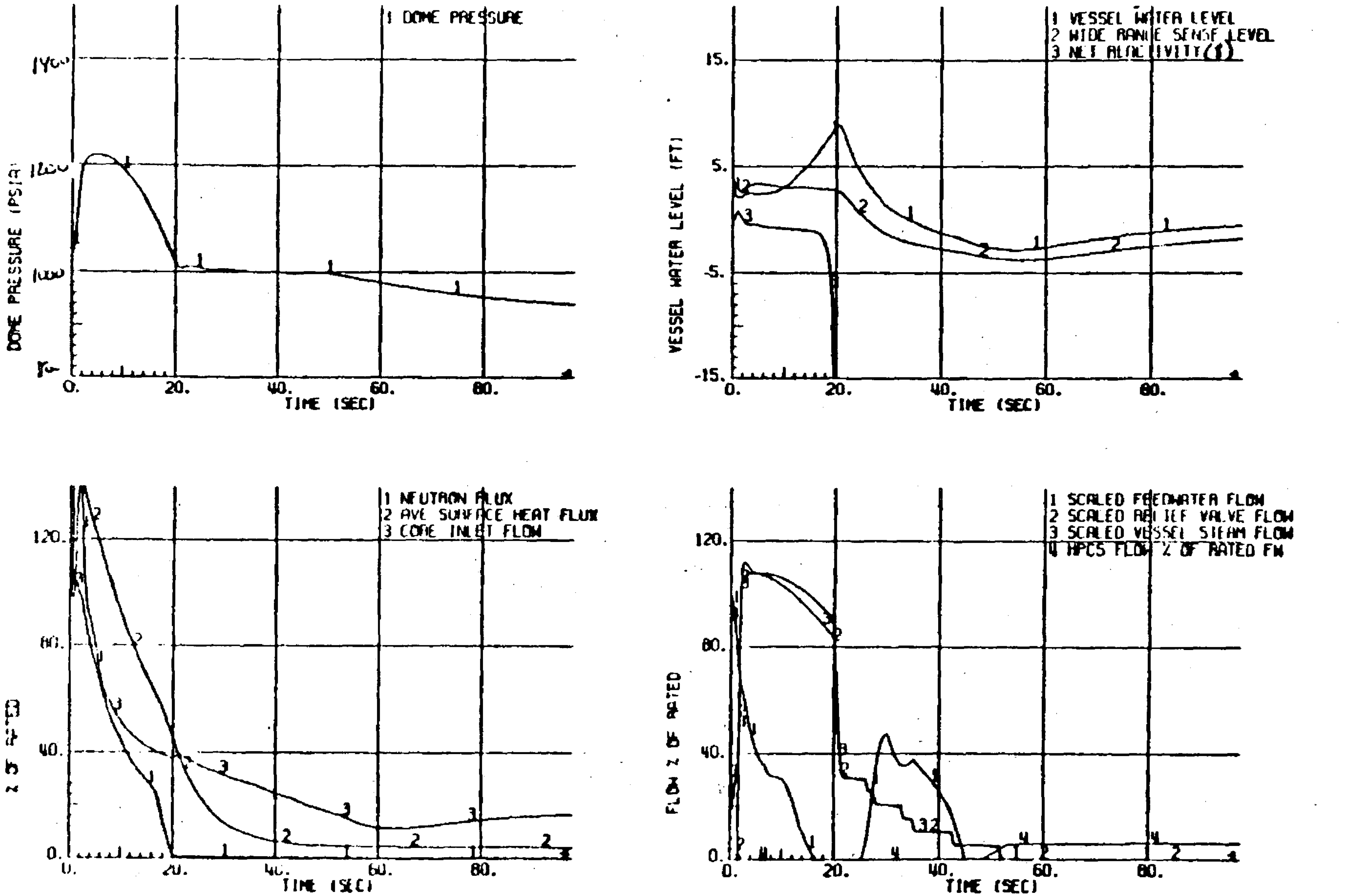


Figure 3.5.2-1. BWR/5 Turbine Trip With Bypass Failure - With ARI

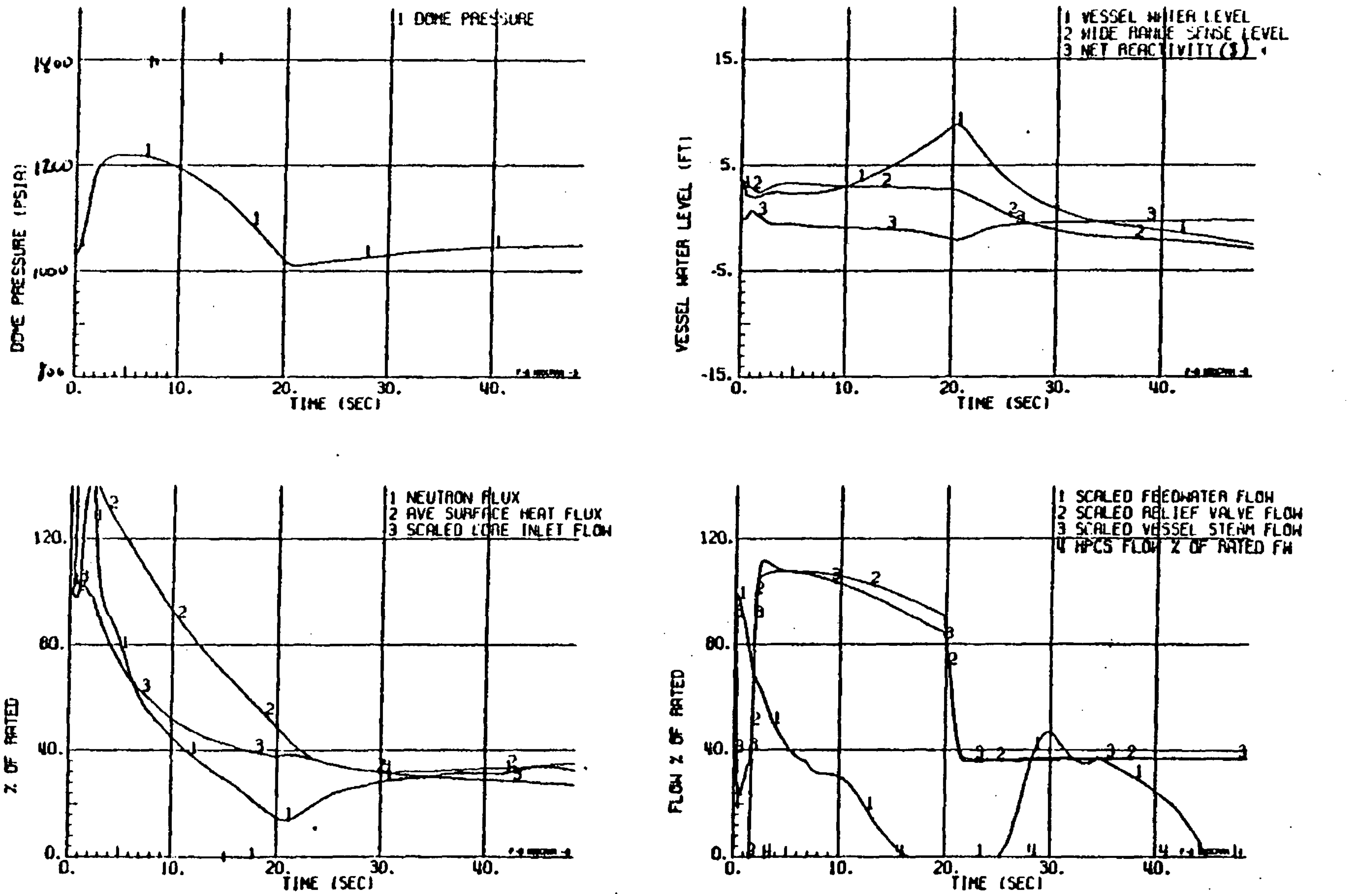


Figure 3.5.2-2. BWR/5 Turbine Trip With Bypass Failure - ARI Failure

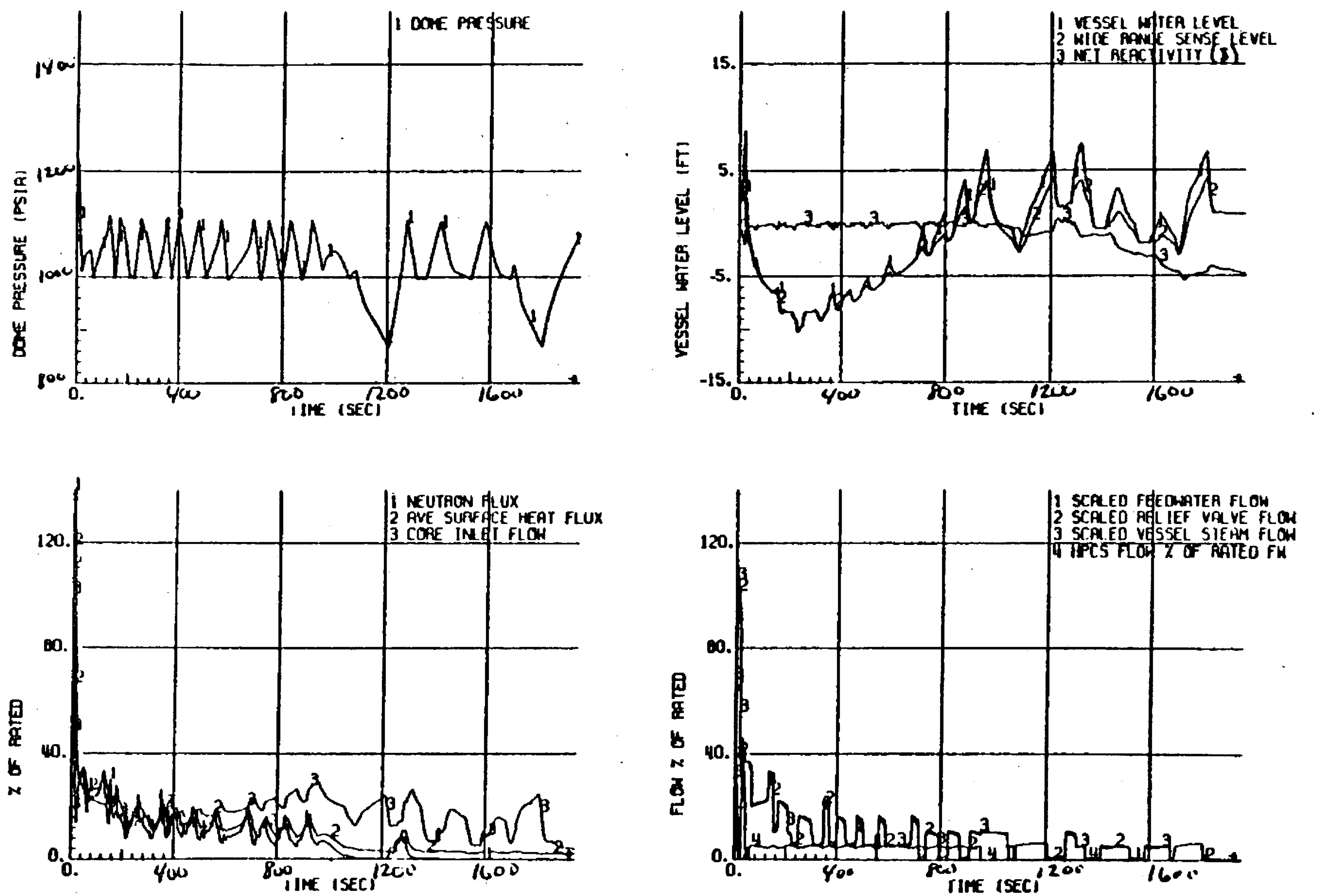


Figure 3.5.2-3. BWR/5 Turbine Trip With Bypass Failure - ARI Failure

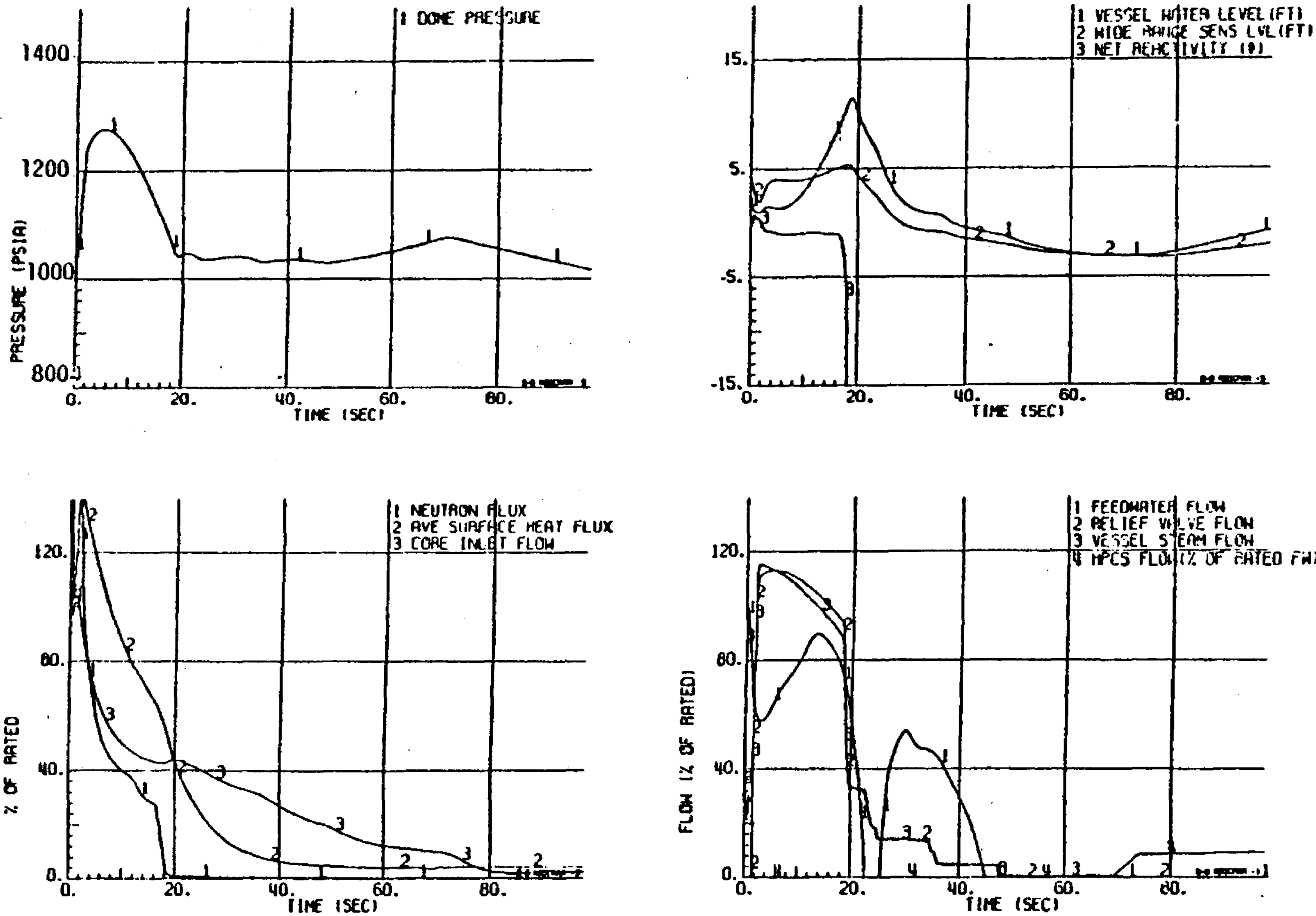


Figure 3.5.3-1. BWR/6 Turbine Trip with Bypass Failure - ARI Failure

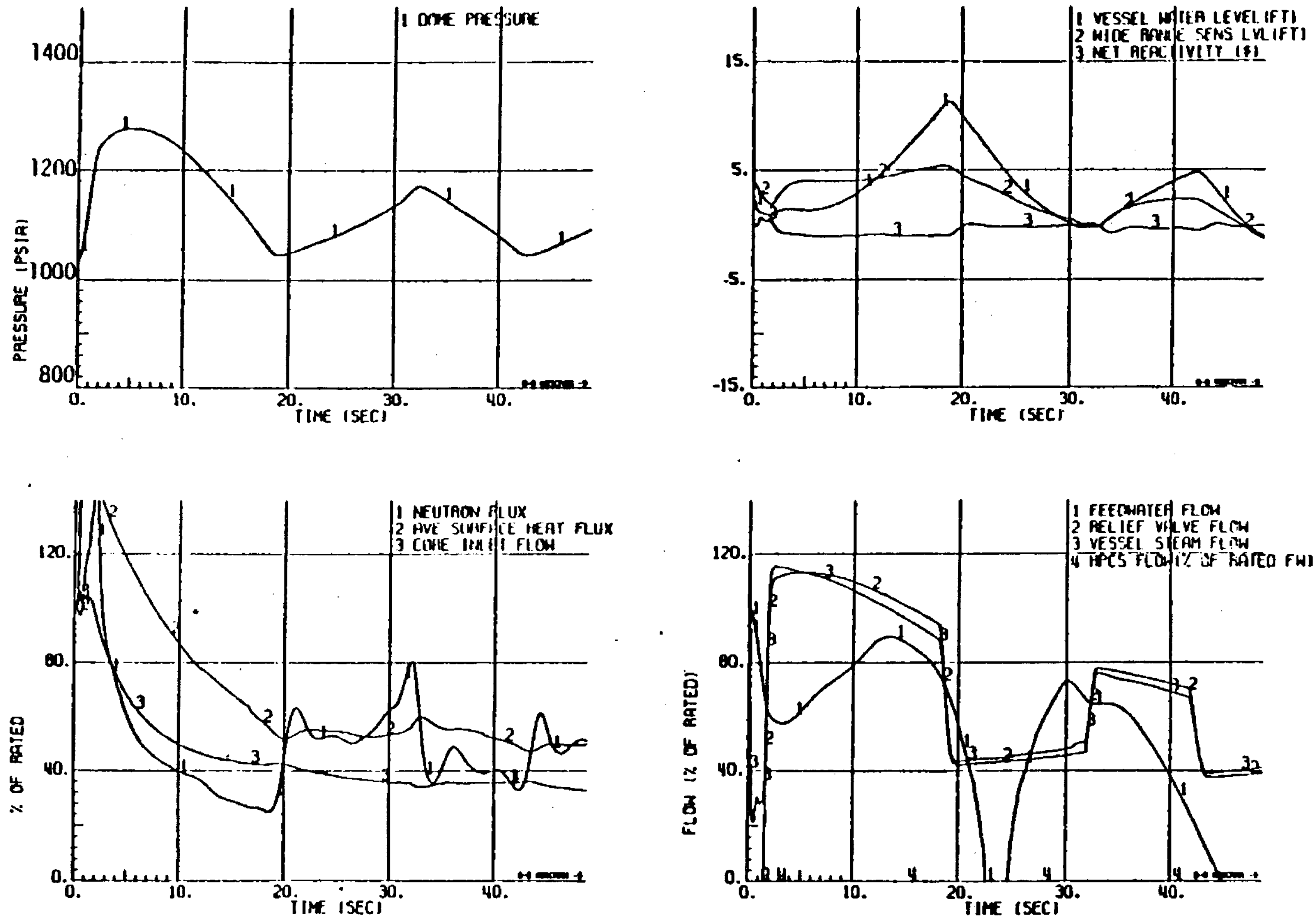


Figure 3.5.3-2. BWR/6 Turbine Trip with Bypass Failure - ARI Failure

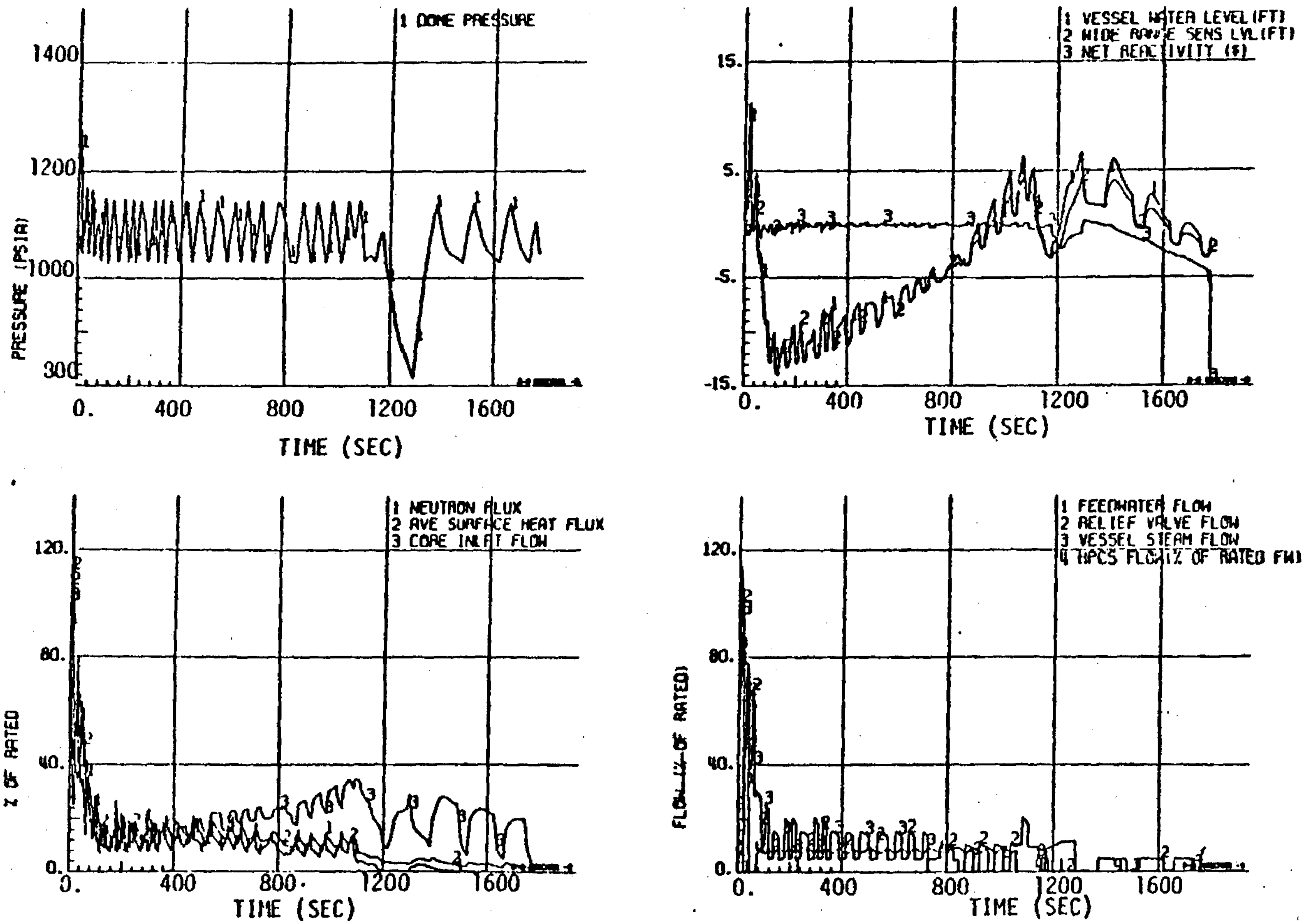


Figure 3.5.3-3. BWR/6 Turbine Trip with Bypass Failure - ARI Failure

4. ACCEPTANCE LIMITS AND CONFORMANCE

4.1 PRIMARY SYSTEM INTEGRITY

RPV primary system peak pressures are under the emergency limit of 1500 psi for all events analyzed.

4.2 CONTAINMENT INTEGRITY

Postulated ATWS events subject the containment structure to static and dynamic loads which are less than those for which the containment has been designed. This section of the report shows that containment structural integrity is maintained by comparison with existing design loads.

4.2.1 Static Pressure and Temperature

For the static pressure and temperature loads, ATWS event consequences are much less severe than the results of loss-of-coolant accidents which form part of the design basis for containments as documented in Section 5.2 of Safety Analysis Reports. The table below shows that, for all containment types, the pressures and temperatures due to ATWS events are within the capability of the containment.

	Peak Pressure (psig)		Bulk Pool Temperature (°F)	
Containment	ATWS	Design Basis	ATWS	Design Basis
Mark I (BWR 4)	11	56	189	281
Mark II (BWR 5)	10	45	185	220*
Mark III (BWR 6)	7	15	170	185

*However, most Mark II plants have a design basis of 240°F.

4.4 FUEL INTEGRITY

A fundamental assumption in the NRC guidelines in NUREG-0460 is that the occurrence of some fuel perforations during an ATWS event is acceptable providing: a) the extent/number of perforations does not cause unacceptable radiological consequences; and, b) the resulting fuel condition does not preclude coolability and ultimate safe shutdown. The safety condition is assured through the application of the fuel damage criteria of 10CFR50 Appendix K, which are used to assure coolable geometry during a loss-of-coolant accident (LOCA).

4.4.1 Fuel Integrity Criteria

The fuel integrity criteria are those used to assure maintenance of a coolable geometry. These criteria must not be violated by an ATWS event. The specific criteria are: 1) the maximum peak cladding temperature must not exceed 2200°F; and, 2) the maximum local cladding oxidation must not exceed 17%.

As in a LOCA event, satisfaction of these criteria will assure maintenance of a coolable geometry in the fuel.

4.4.2 Results of Fuel Integrity Evaluations

4.4.2.1 Peak Cladding Temperature

The peak calculated cladding temperature for all ATWS events analyzed was significantly below the 2200°F requirement from 10CFR50 Appendix K.

4.4.2.2 Localized Cladding Oxidation

The maximum calculated local cladding oxidation for the ATWS events analyzed was found to be significantly below the accepted maximum value (17% of cladding volume). Therefore, no effects due to oxidation are expected to occur.

4.4.3 Conclusions

The foregoing sections indicate that there is substantial margin with respect to assuring coolability of the core and safe reactor shutdown. Perforations assumed to occur in fuel which experiences boiling transition results in relatively small radiological releases (Section 4.5). General Electric therefore concludes that the ATWS rules and requirements specified in NUREG-0460 (Volume 3, Appendix IV) can be fulfilled under the most severe ATWS events.

REFERENCES

4.5 RADIOLOGICAL ANALYSIS

The radiological analysis prepared for this submittal has considered the BWR/4-6 reactors and the Mark I-III containments. While there are significant differences for these plants, analyses have shown that the dose consequences of all ATWS events in a BWR/6-MK III bound the consequences for all three containment types and product lines. A schematic of the containment system and fission product transport pathways used in this analysis is shown in Figure 4-5.1. One of the primary differences between the Mark I-II containments and the Mark III containment is the "open" suppression pool for Mark III. There will be a short period of time prior to containment isolation when fission products may be released directly to the environment from the Mark III containment. Because of the "closed supression pool for the Mark I-II containments this pathway does not exist for these designs, therefore, the radiological consequences for a BWR/6 Mark III bound the consequences for these two product lines. For the Mark III and for some Mark I-II containments forced mixing of the air within the secondary containment is provided. However, this is not a universal feature, therefore one of the assumptions applied to the "conservative assessment" is that zero mixing occurs within secondary containment. The meteorological conditions bound all BWR sites licensed to date.

Two analytical evaluations, which include two cases each, are presented in subsequent sections. These two evaluations are arbitrarily defined as "Realistic Assessment" and "Conservative Assessment."

4.5.1 Assumptions/Conditions of Analysis

The assumptions or conditions considered appropriate for the evaluation of the radiological calculations are presented in Tables 4.5.1 - 4.5.4 Parametric values in the conservative columns in these tables are consistent with the guidance offered in Reference 4.4.4. Where guidance was lacking, assumptions were made which are consistent with previous BWR licensing practice.

Parametric values in the realistic columns are consistent with experimental data obtained from operating BWRs or which are considered conservative if operating data is lacking. For the zero perforation case (case 1) the assumption is made that the activity released from defective fuel rods is proportional to the negative change in reactor vessel pressure.

4.5.2 ATWS Events Evaluated

Radiological evaluations have been performed for nine ATWS events. For two of these events, PREGO and FWCF, the two cases in Table 4.5.1 were evaluated (i.e., zero fuel perforations and 100% fuel perforations) for the BWR 4/5/6. The remaining events were evaluated with conservative assumptions, 100% fuel perforations, and a BWR/6 only. The events evaluated for all three product lines were:

1) Pressure Regulator Failure - Open (PREGO)

2) Feedwater Controller Failure - Maximum Demand (FWCF)

The other events evaluated for 100% fuel perforations and the BWRIG - Mark III were:

3) Loss of Auxiliary Power (LOAP)

4) Loss of Normal Feedwater (LNFW)

5) Turbine Trip without Bypass (TTWOB)

6) Pressure Reg. Failure Closed (PREGC)

7) Loss of Condenser Vacuum (LOCV)

8) Loss of Feedwater Heater (70FW)

9) Recirculation Flow Control Failure - Maximum Demand (FSTOP)

Transients 8 and 9 are not expected to result in any fuel damage or isolation, therefore, the off gases continue to be processed by the off gas system and no additional radiological consequences are expected, therefore these events will not be discussed hereafter.

For blowdown to an "open" suppression pool, such as for a BWR/6-Mark III containment, isolation of the containment ventilation occurs at the following times:

Event	Signal Initiating Closure	Isolation Signal Occurs at (Sec)	Containment Isolated (Sec)* (Relative to 0 Time)
PREGO & LOCV, PREGC	High Radiation Plenum Exhaust	2.5 (a)	7.5
TTWOB & LOAP	High Radiation Plenum Exhaust	4 (a)	9
FWCF & LNFW	Low Water Level Level (2)	23	28

(a) Assume instantaneous homogeneous distribution of Fission Products in containment.

*Includes allowance of 5 seconds to close ventilation valves.

4.5.2.1 Pressure Regulator Failure - Open (PREGO)

An examination of the pressure in the reactor pressure vessel for a BWR/6 shows that the pressure initially goes to about 1200 psi, drops in pressure (due to SRV opening) to about 1113 psia, and cycles between this value and 1150 psi for approximately 240 seconds, and then cycles at lower pressure fluctuations from that time on. As noted previously, the fission product release for defective fuel rods is assumed to be proportional to the negative change in reactor pressure. Therefore in approximately 240 seconds the fuel is assumed to have experienced a pressure reduction equivalent to 1050 psia. For purposes of evaluation the total "spiking" activity in Table 4.5.4 for case 1 is, therefore assumed to be released uniformly to the primary coolant over a 240 second time period. As an example, the release rate of I-131, for the conservative analysis, is assumed to be 14,000/240 = 58 Ci/sec. This release rate occurs for a time period of 240 seconds after which no additional release occurs to the RPV. The activity in the primary coolant is determined as follows:

$$dN_1/dt = - N_1(\lambda + L_1) + S_1 \qquad (1)$$

$$N_1 = \frac{S_1}{\lambda + L_1} \left(1 - e^{-(\lambda + L_1) t}\right) \text{ (Ci)} \qquad (2)$$

where t in Equation 2 is valid between 0 and 240 seconds. For $t > 240$ seconds the activity in the primary coolant is determined as follows:

$$dN_1/dt = -(\lambda + L_1)N_1$$

$$N_1 = \frac{S_1}{\lambda + L_1} \left(1 - e^{-(\lambda + L_1)\,240}\right) \left(e^{-(\lambda + L_1)\,(t - 240)}\right) \qquad (3)$$

where:

λ = radioactive decay constant (sec^{-1})

L_1 = release rate from the RPV and is determined as follows:

$$L_{NG} = \frac{\text{Steam blowdown rate (\#/sec)}}{\text{Mass of Steam in Steam Done (\#)}}$$

$$L_1 = \frac{\text{Steam blowdown rate (\#/sec) (0.02)}}{\text{Mass of Primary Coolant (\#)}}$$

S_1 = source input from defective fuel (Ci/sec)

The 0.02 factor in L_1 is the iodine carryover fraction in steam (i.e., each pound of steam contains 2% of the activity contained in a pound of primary coolant).

The activity being discharged to the condenser or suppression pool is determined by multiplying Equations 2 or 3 by the appropriate value of L, which takes into consideration the type of activity being evaluated and the actual steam blowdown rate to these two areas. For this event all discharges were to the suppression pool.

For case 2, where 100% clad failure is assumed to occur, it is conservatively assumed that failure occurs at t = 0. The activity is the primary coolant is, therefore, defined as follows:

$$dN_1/dt = -(\lambda + L_1)\ N_1 \qquad (4)$$

$$N_1 = N_o e^{-(\lambda + L_1)t}$$

Where Equation 4 is valid for the time periods of interest and the variables are as defined previously.

The pressure transient was evaluated for a BWR/4 - Mark I and a BWR/5 - Mark II using the same analytical approach.

4.5.2.2 Feedwater Controller Failure - Maximum Demand (FWCF):

As for the PREGO transient event, this event has also been quantitatively examined with the conclusion being that from a radiological viewpoint the BWR/6 transient will bound the consequences associated with a BWR/4 or BWR/5. For the BWR/6 after the initial 380 psi depressurization (in 1200 secs) the RPV depressurization rate will depend on operator action. Therefore in the case of 0% fuel failure it is assumed that 100% of the available spiking activity is released within 1200 secs. Equations 2 and 3 from Section 4.5.2.1 are appropriate for this transient with the 240 seconds replaced by 1200 seconds.

For case 2, where 100% clad failure is assumed to occur, it is conservatively assumed that failure occurs at t = 0 and Equation 4 of Section 4.5.2.1 is appropriate for the time periods of interest. For this event, 100% of the decay steam generation rate is conservatively assumed to be discharged to the condenser.

4.5.2.3 All Other ATWS Events:

The remaining 5 events were analyzed assuming 100% fuel failure and using the conservative assumptions in Table 4.5.1 for a BWR 16 - Mark III plant. All doses were less than the TTWB event presented previously.

4.5.2.4 Fission Product Release to the Environment

The fission product activity released to the environment is dependent upon the release pathway from the reactor vessel and the reduction factors and compartmental leakage rates between the RPV and the environment. The activity airborne in the compartments of concern is defined by the following differential equations.

a) Activity in Condenser

$$dN_c/dt = -(\lambda + L_6)\, N_c + L_4\, N_1\, DF \tag{5}$$

b) Activity in Primary Containment

$$dN_{pc}/dt = -(\lambda + L_3 + L_5)\,N_{pc} + L_2\,N_1\,DF$$

c) Activity in Secondary Containment

$$dN_{sc}/dt = -(\lambda + L_1)\,N_{sc} + L_c\,N_{pc}$$

where the above parameters are schematically defined in Figure 4.5.1.

4.5.2.5 Radiological Consequences

Based upon the preceeding discussion, the following radiological exposures are calculated for the seven ATWS events of concern. These consequences can be compared to the guidelines in 10 CFR 100 which are 25 Rem whole body and 300 Rem thyroid inhalation. It can be seen that even if 100% of the rods were to perforate, the guidelines would not be exceeded.

BWR/4

Event	Radiological Consequence (Rem)							
	Site Boundary				Low Population Zone			
	Whole Body		Inhalation		Whole Body		Inhalation	
	Real.	Cons.	Real.	Cons.	Real.	Cons.	Real.	Cons.
(1) PREGO								
• Case 1	1.1-5[a]	2.3-2	7.2-8	5.8-4	3.6-6	5.1-3	1.8-7	3.4-5
• Case 2	2.5-2	2.6-1	5.7-4	5.0-2	6.3-3	7.1-2	1.4-3	3.4-2
(2) FWCF								
• Case 1	2.0-5	2.1-2	2.0-6	2.3-3	4.5-6	4.7-3	5.0-6	1.4-3
• Case 2	2.4-2	1.8-1	6.3-3	4.5-1	5.0-3	4.8-2	1.5-2	3.3-1

Case 1 - 0% perforations Real = Realistic

Case 2 - 100% perforations Cons = Conservative

(a) 1.1-5 = 1.1×10^{-5} Rem

BWR/5

Event	Radiological Consequence (Rem)							
	Site Boundary				Low Population			
	Whole Body		Inhalation		Whole Body		Inhalation	
	Real.	Cons.	Real.	Cons.	Real.	Cons.	Real.	Cons.
(1) PREGO								
• Case 1	1.5-5[a]	2.9-2	1.1-7	2.0-4	6.1-6	6.6-3	2.7-7	1.1-4
• Case 2	3.2-2	2.3-1	8.9-4	2.4-1	8.2-3	6.2-2	2.1-3	1.8-1
(2) FWCF								
• Case 1	2.0-5	1.4-2	6.0-7	6.8-4	4.9-6	3.4-3	1.5-6	4.1-4
• Case 2	3.3-2	2.4-1	4.7-3	3.4-1	6.8-3	6.5-2	1.1-2	2.4-1

Case 1 - 0% perforations　　Real = Realistic

Case 2 - 100% perforations　　Cons = Conservative

(a) 1.5-5 = 1.5×10^{-5} Rem

BWR/6

Event	Radiological Consequence (Rem)							
	Site Boundary				Low Population			
	Whole Body		Inhalation		Whole Body		Inhalation	
	Real.	Cons.	Real.	Cons.	Real.	Cons.	Real.	Cons.
(1) PREGO								
• Case 1	1.3-5[a]	5.9-2	6.3-7	3.1-4	1.1-5	1.3-2	2.1-7	1.8-4
• Case 2	7.7-2	5.8-1	5.9-4	5.8-2	1.3-2	1.2-1	6.8-3	3.1-1
(2) FWCF								
• Case 1	2.3-5	1.4-2	9.4-7	9.7-4	5.7-6	3.5-3	2.3-6	5.8-4
• Case 2	3.9-2	2.9-1	7.0-3	5.0-1	8.1-3	7.8-2	1.7-2	3.6-1

Case 1 - 0% perforations　　Real = Realistic

Case 2 - 100% perforations　　Cons = Conservative

(a) 1.3-5 = 1.3×10^{-5} Rem

BWR/6

Event	Radiological Consequence (Rem)*			
	Site Boundary		Low Population Zone	
	Whole Body	Inhalation	Whole Body	Inhalation
(3) LOAP[b]	<7.3-1[a]	<6.6-1	<1.4-1	<3.5-2
(4) LNFW	5.8-1	8.1-1	7.8-2	2.9-1
(5) TTWOB	7.3-1	6.6-1	1.4-1	3.5-2
(6) PREGC	<5.8-1	<5.8-2	<1.2-1	<3.0-1
(7) LOCV	<5.8-1	<5.8-2	<1.2-1	<3.0-1

*assumed 100% perforations

(a) 7.3-1 = 7.3×10^{-1} Rem

(b) Events (3), (6), (7) were compared to other ATWS analyses with similar blowdown rates, therefore doses represent an upper bound.

4.5.3 Conclusions

Based on the results presented in Section 4.5.2.5, it can be concluded that the radiological exposure for the ATWS events evaluated are well below the guideline in 10CFR100 for all three BWR product lines (BWR/4, 5 and 6), and for all three BWR containment designs (Mark I, II and III).

BWR/6

Event	Radiological Consequence (Rem)*			
	Site Boundary		Low Population Zone	
	Whole Body	Inhalation	Whole Body	Inhalation
(3) LOAP[b]	<7.3-1[a]	<6.6-1	<1.4-1	<3.5-2
(4) LNFW	5.8-1	8.1-1	7.8-2	2.9-1
(5) TTWOB	7.3-1	6.6-1	1.4-1	3.5-2
(6) PREGC	<5.8-1	<5.8-2	<1.2-1	<3.0-1
(7) LOCV	<5.8-1	<5.8-2	<1.2-1	<3.0-1

*assumed 100% perforations

(a) 7.3-1 = 7.3×10^{-1} Rem

(b) Events (3), (6), (7) were compared to other ATWS analyses with similar blowdown rates, therefore doses represent an upper bound.

4.5.3 Conclusions

Based on the results presented in Section 4.5.2.5, it can be concluded that the radiological exposure for the ATWS events evaluated are well below the guideline in 10CFR100 for all three BWR product lines (BWR/4, 5 and 6), and for all three BWR containment designs (Mark I, II and III).

Table 4.5.1

ASSUMPTIONS/CONDITIONS UPON WHICH RADIOLOGICAL ANALYSIS IS BASED

	Parametric Value Assumed	
	Realistic	Conservative
1. Power level : BWR 4/5/6	2558/3323/4146	2558/3323/4146
2. Fuel Type	8 x 8	8 x 8
3. Fuel Rod Perforations (%)		
• Case 1	0	0
• Case 2	100	100
4. Fission Products Rel. to Pri. Coolant (Curies)		
• Case 1		
I-131	1200/1631/1810	1.03+4/1.4+4/1.4+4
Xe-133	6500/8863/9840	1.6+5
• Case 2		
Iodines	2% Rod Act.	2% Rod Act.
Noble Gases	2% Rod Act.	2% Rod Act.
5. Fission Products Rel. to Supp. Pool or Main Turbine Condenser	Proportional to mass blowdown rate, pri. coolant vol., RPV stm. dome vol. and 2% carryover for Iodine	
6. DF[b] in Suppression Pool/Turbine Conden.		
• Noble Gases	1	1
• Iodine	0.01	0.1
7. Pri. Containment Leak Rate (%/day)	0.5/0.5/1.0	0.5/0.5/1.0
8. Sec. Containment Leak Rate (%/day)	100	100
9. Condenser Leak Rate (%/day)	1	1
10. Mixing in Secondary Containment (%)	100	0
11. SGTS Iodine Filter Efficiency (%)	99	95
12. Meteorology[c] (χ/Q - sec/m^3)		
Site Boundary (0-2 hrs.)	2.5×10^{-5}	1.8×10^{-4}
Low Population Zone		
0.2 hrs.	10^{-6}	2.6×10^{-5}
2-8		1.7×10^{-5}
8-24		2.6×10^{-6}
24-96		1.4×10^{-6}
96-720		5.8×10^{-7}

(b) DF = $\frac{\text{what comes out}}{\text{what goes in}}$

(c) Realistic meteorology is avg. annual and conservative meteorology is 10% of the 95% meteorology.

Table 4.5.2

FISSION PRODUCTS RELEASED TO PRIMARY COOLANT FROM NORMALLY FAILED OR ATWS INDUCED FAILED FUEL RODS

BWR/4 (218-560)

Isotope	Activity Released (Ci)		
	Case 1		Case 2
	0 Perforations		100% Perforations
	Realistic	Conservative	
I-131	1.2+3 [a]	1.0+4 [b]	1.3+6
132	1.8+3	1.5+5	1.9+6
133	2.8+3	7.3+4	2.9+6
134	3.0+3	4.2+5	3.4+6
135	2.7+3	1.2+5	2.6+6
K4-83m	5.1+2	6.5+4 [c]	2.1+5
83m	1.2+3	1.2+5	6.6+5
85	2.8+2	2.8+2	2.1+4
87	2.4+3	3.8+5	1.2+6
88	3.4+3	3.8+5	1.6+6
89	4.5+3	2.3+6	2.0+6
Xe-131m	3.7+1	2.8+2	1.3+4
133m	1.8+2	5.5+3	7.1+4
133	6.5+3	1.6+5	2.9+6
135m	1.0+3	5.0+5	1.0+5
135	6.2+3	4.1+5	2.7+6
137	5.9+3	2.8+6	2.6+6
138	5.9+3	1.7+6	2.4+6

a) 1.2+3 = 1.2×10^3 curies.
b) I-131 release based on a release rate from the fuel for 4 hours equal to 250 times the tech. spec. value, where the tech. spec. equals 0.2 μCi/gm dose equivalent I-131.
c) Noble gas values based on 1330 times normal offgas integrated over a 4 hr. period.

Table 4.5.3

FISSION PRODUCTS RELEASED TO PRIMARY

NORMALLY FAILED OR ATWS INDUCED FA

BWR/5 (251-764)

Isotope	Activity Relea Case 1 0 Perforations Realistic	Conservative
I-131	1.6+3[a]	1.4+4[b]
132	2.5+3	2.0+5
133	3.8+3	9.9+4
134	4.2+4	5.7+5
135	3.7+3	1.6+5
Kr-83m	6.9+2	6.5+4[c]
85M	1.7+3	1.2+5
85	3.8+2	2.8+2
87	3.3+3	3.8+5
88	4.7+3	3.8+5
89	6.1+3	2.3+6
Xe-131m	5.0+1	2.8+2
133m	2.5+2	5.5+3
133	8.9+3	1.6+5
135m	1.4+3	5.0+5
135	8.4+3	4.1+5
137	8.0+3	2.8+6
138	8.1+3	1.7+6

a) 1.6+3 = 1.6×10^3 curies.

b) I-131 release based on a release rate from the to 250 times the tech. spec. value, where the 0.2 μCi/gm dose equivalent I-131.

c) Noble gas values based on 1330 times normal 4 hr. period.

Table 4.5.4

FISSION PRODUCTS RELEASED TO PRIMARY COOLANT FROM NORMALLY FAILED OR ATWS INDUCED FAILED FUEL RODS BWR/6 (251-848)

Isotope	Activity Released (Ci)		
	Case 1		Case 2
	0 Perforations		100% Perforations
	Realistic	Conservative	
I-131	1.8+3[a]	1.4+4[b]	2.1+6
132	2.7+3	2.0+5	3.2+6
133	4.3+3	1.0+5	4.7+6
134	4.6+3	5.7+5	5.5+6
135	4.1+3	1.6+5	4.2+6
K4-83m	7.7+2	6.5+4[c]	3.4+5
85m	1.9+3	1.2+5	1.1+6
85	4.2+2	2.8+2	3.4+4
87	3.7+3	3.8+5	1.9+6
88	5.2+3	3.8+5	2.7+6
89	6.8+3	2.3+6	3.3+6
Xe-131m	5.6+1	2.8+2	2.2+4
133m	2.8+2	5.5+3	1.2+5
133	1.0+4	1.6+5	4.7+6
135m	1.5+3	5.0+5	1.3+6
135	9.3+3	4.1+5	4.5+6
137	8.9+3	2.8+6	4.2+6
138	9.0+3	1.7+6	4.0+6

a) 1.8+3 = 1.8×10^3 curies.
b) I-131 release based on a release rate from the fuel for 4 hours equal to 250 times the tech. spec. value, where the tech. spec. equals 0.2 μCi/gm dose equivalent I-131.
c) Noble gas values based on 1330 times normal offgas integrated over a 4 hr. period.

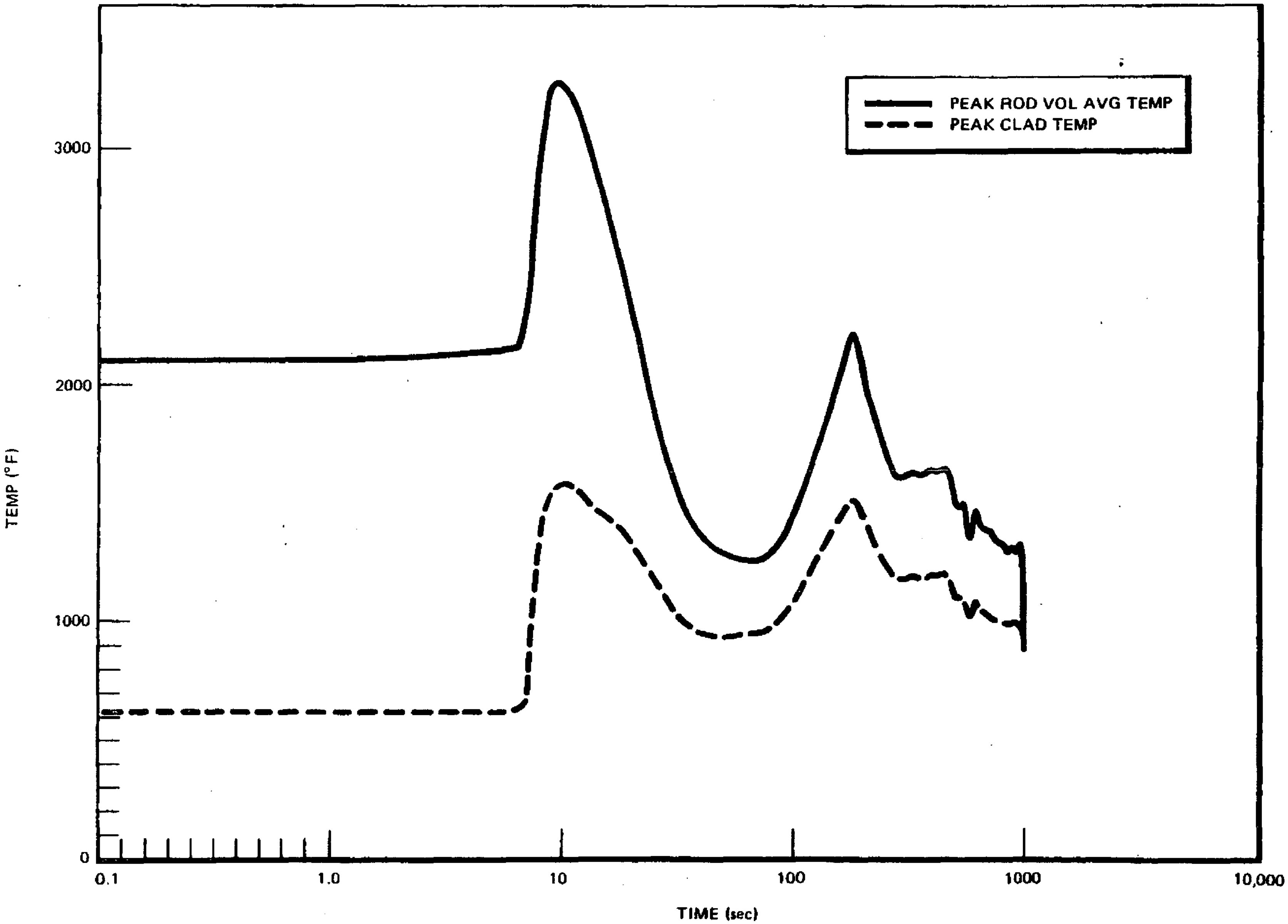


Figure 4.4-1. BWR/4 ATWS Feedwater Controller Failure Fuel Evaluation

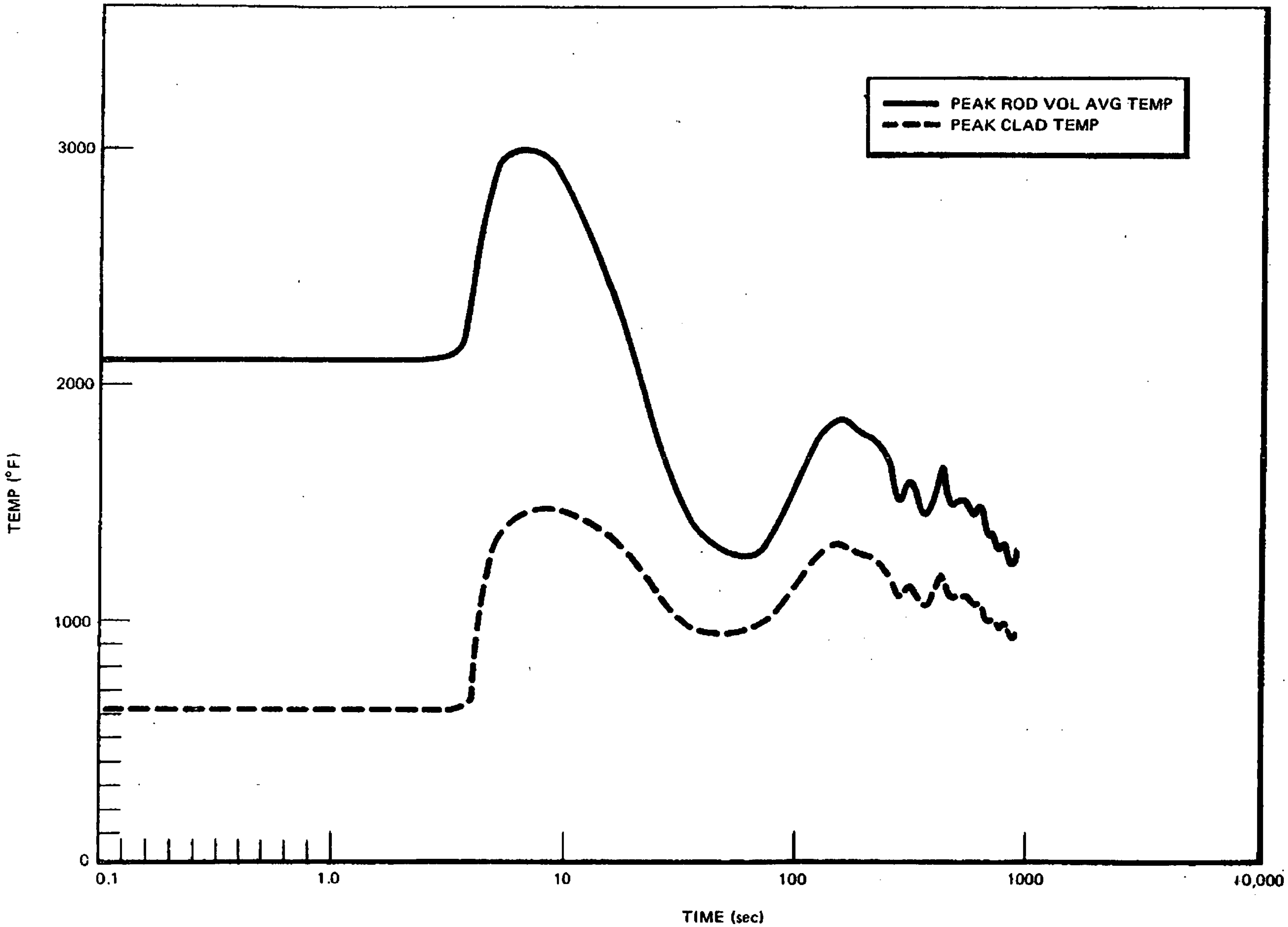


Figure 4.4-2. BWR/4 ATWS MSIV Fuel Evaluation

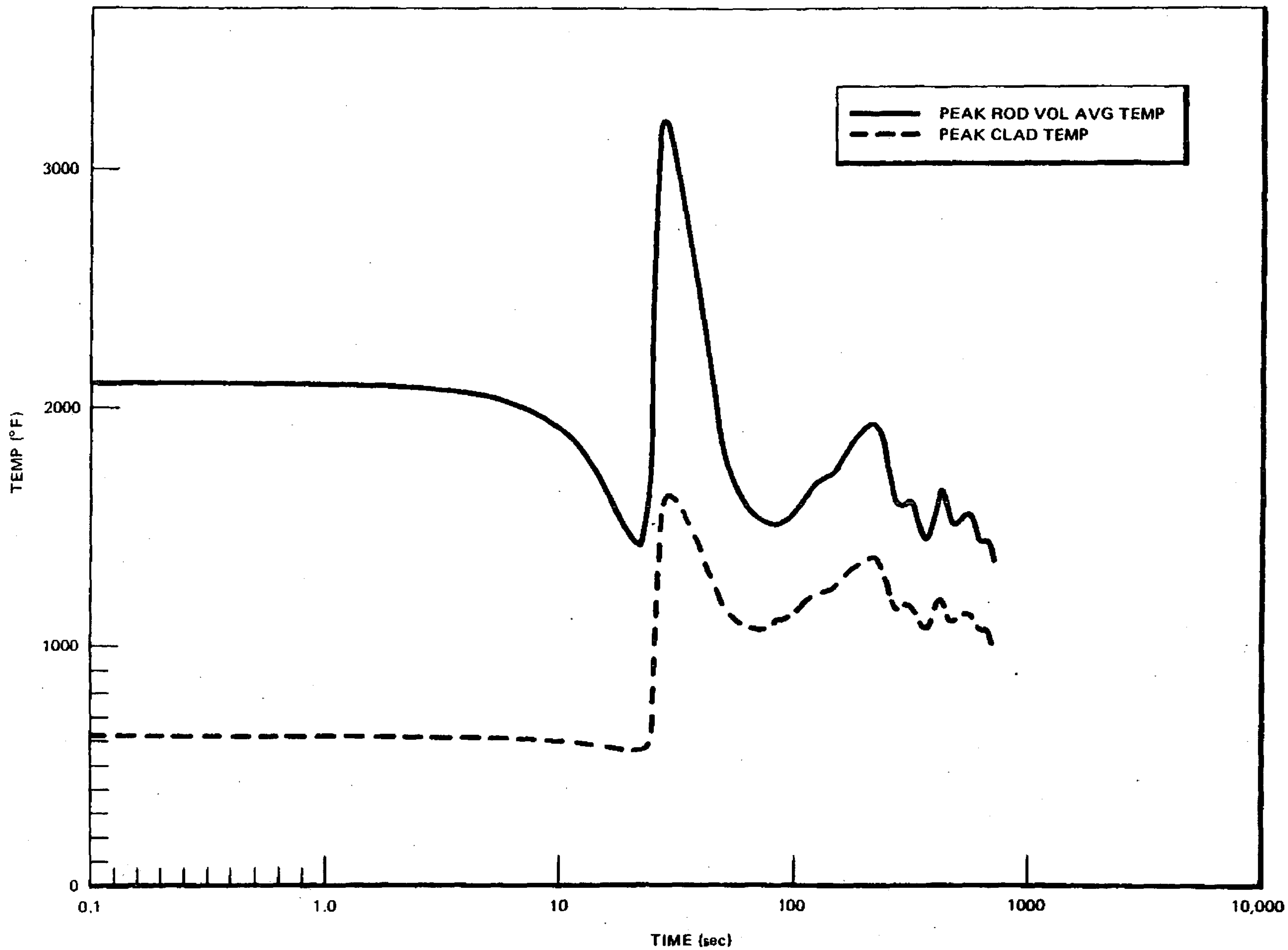


Figure 4.4-3. BWR/4 ATWS Pressure Regulator Failure (Maximum Demand) Fuel Evaluation

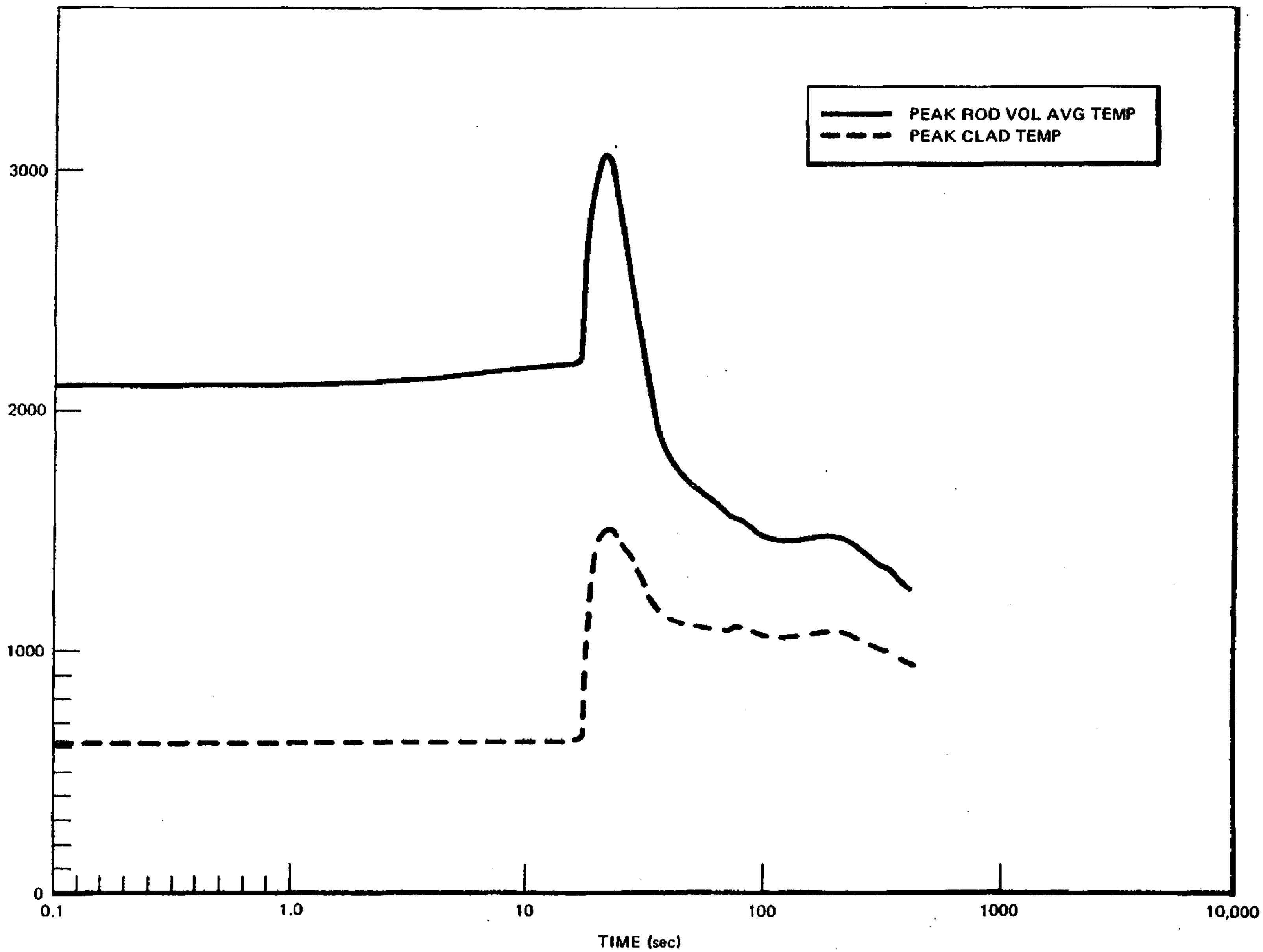


Figure 4.4-4. BWR/5 ATWS Feedwater Controller Failure Fuel Evaluation

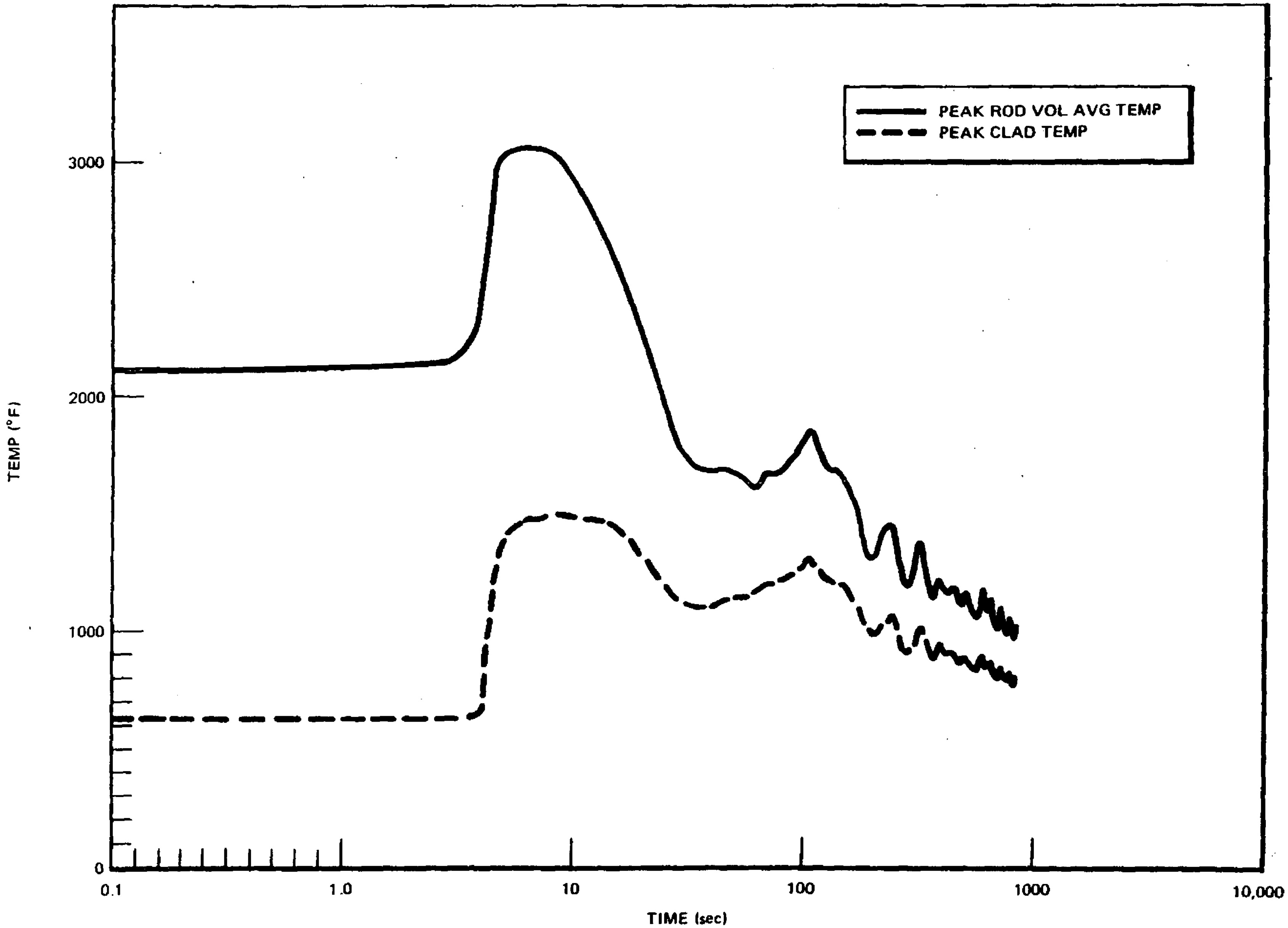


Figure 4.4-5. BWR/5 ATWS MSIV Closure Fuel Evaluation

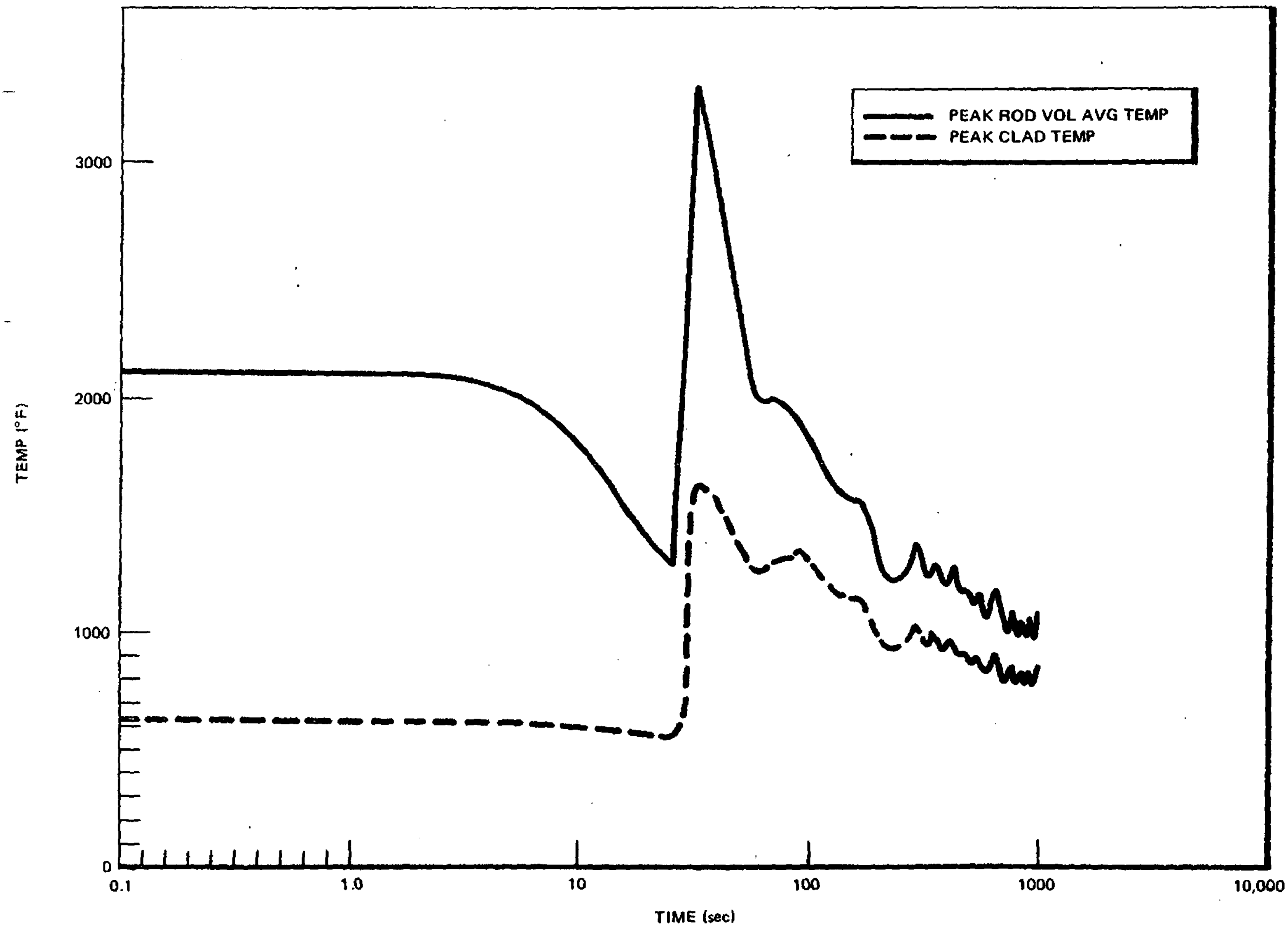


Figure 4.4-6. BWR/5 ATWS Pressure Regulator Failure (Maximum Demand) Fuel Evaluation

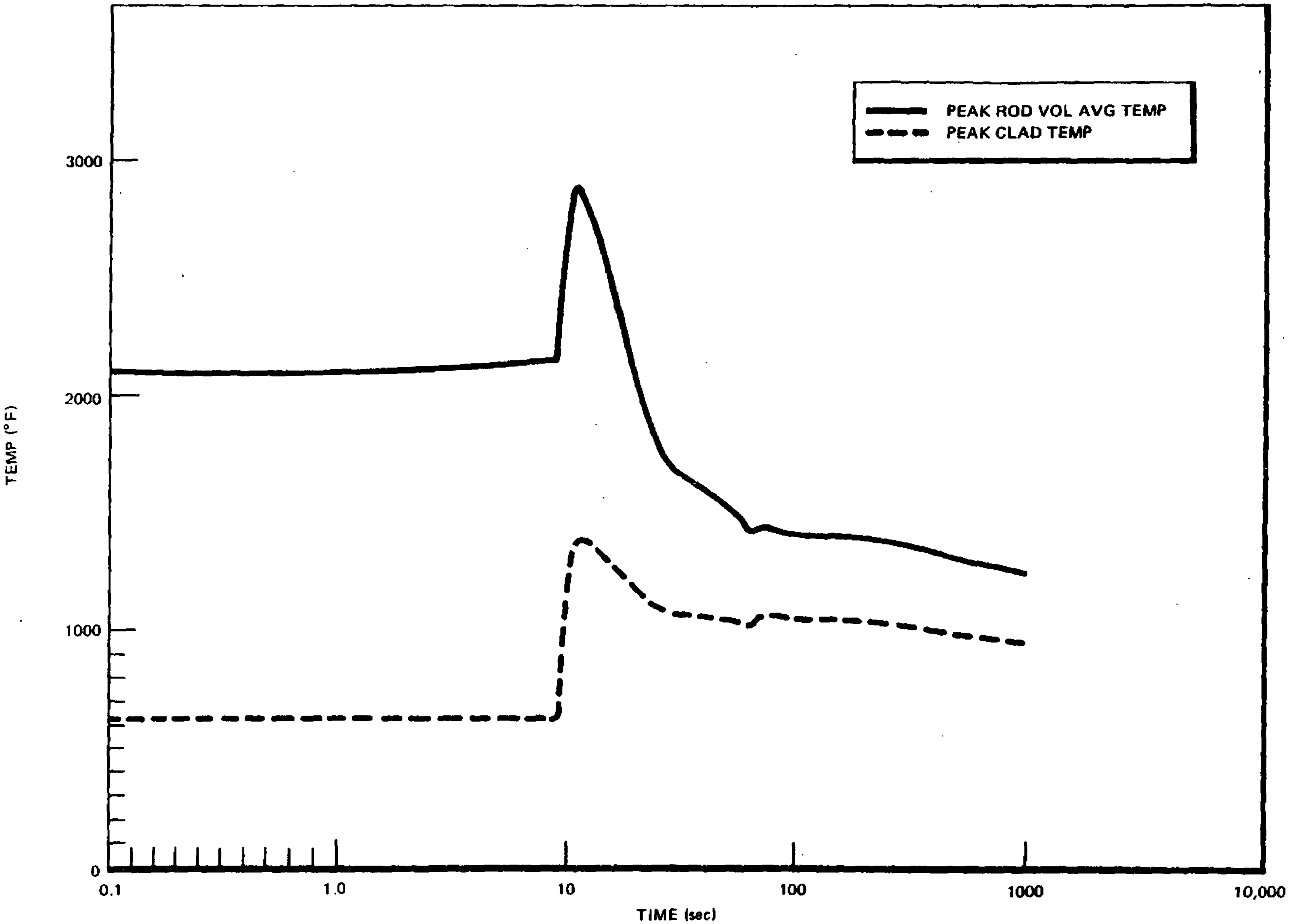


Figure 4.4-7. BWR/6 ATWS Feedwater Controller Failure Fuel Evaluation

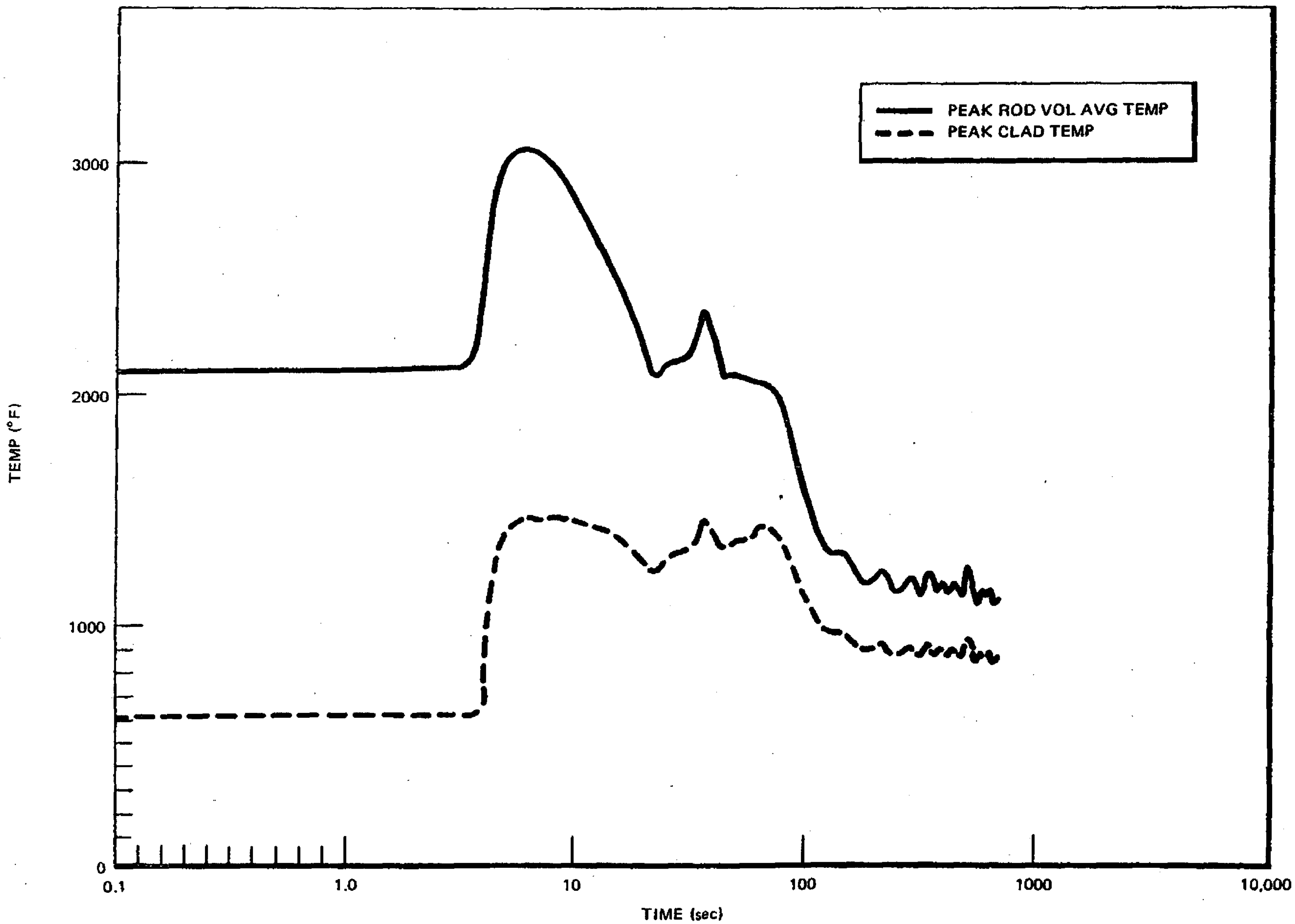


Figure 4.4-8. BWR/6 ATWS MSIV Closure Fuel Evaluation

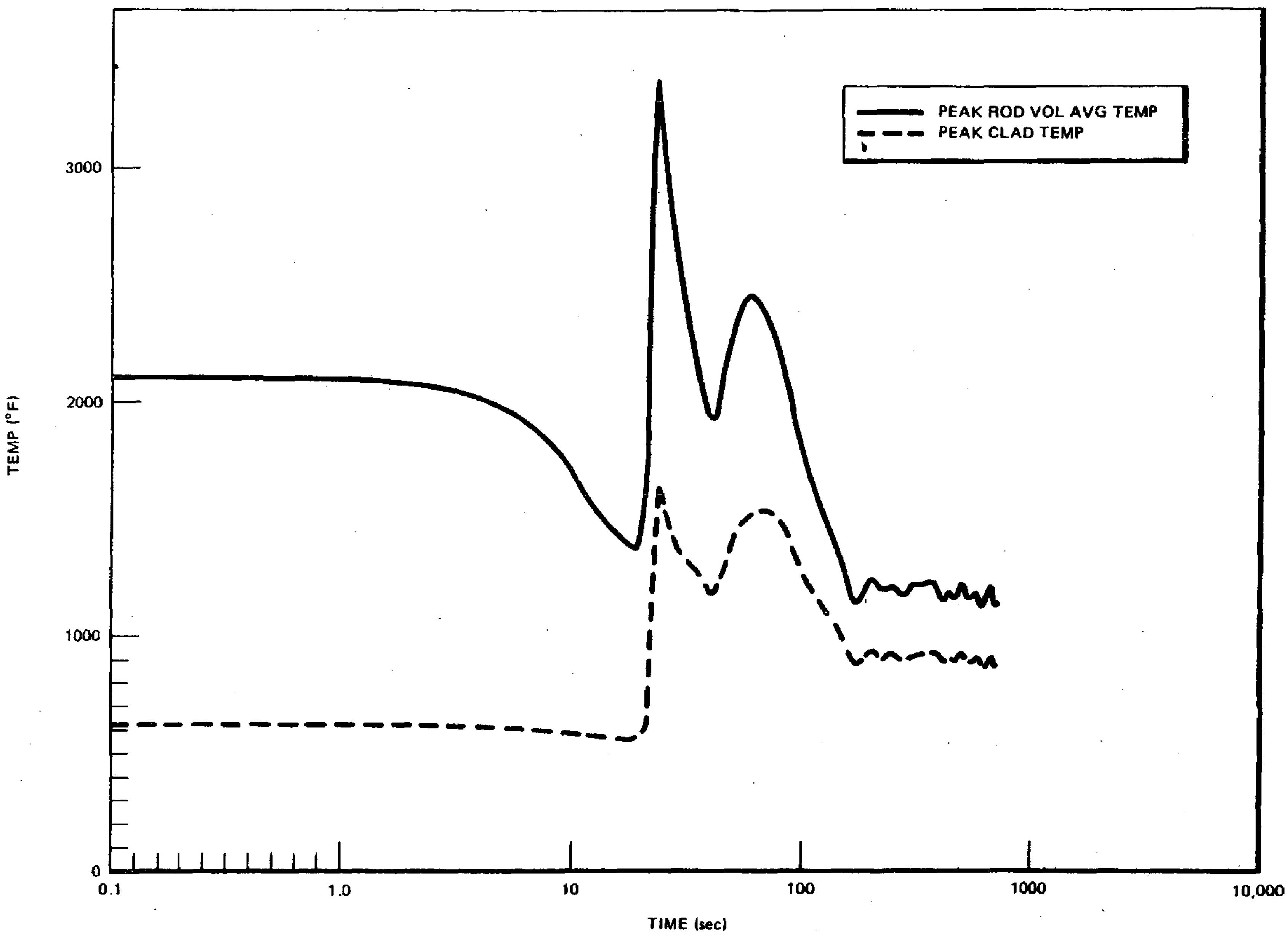


Figure 4.4-9. BWR/6 ATWS Pressure Regulator Failure (Maximum Demand) Fuel Evaluation

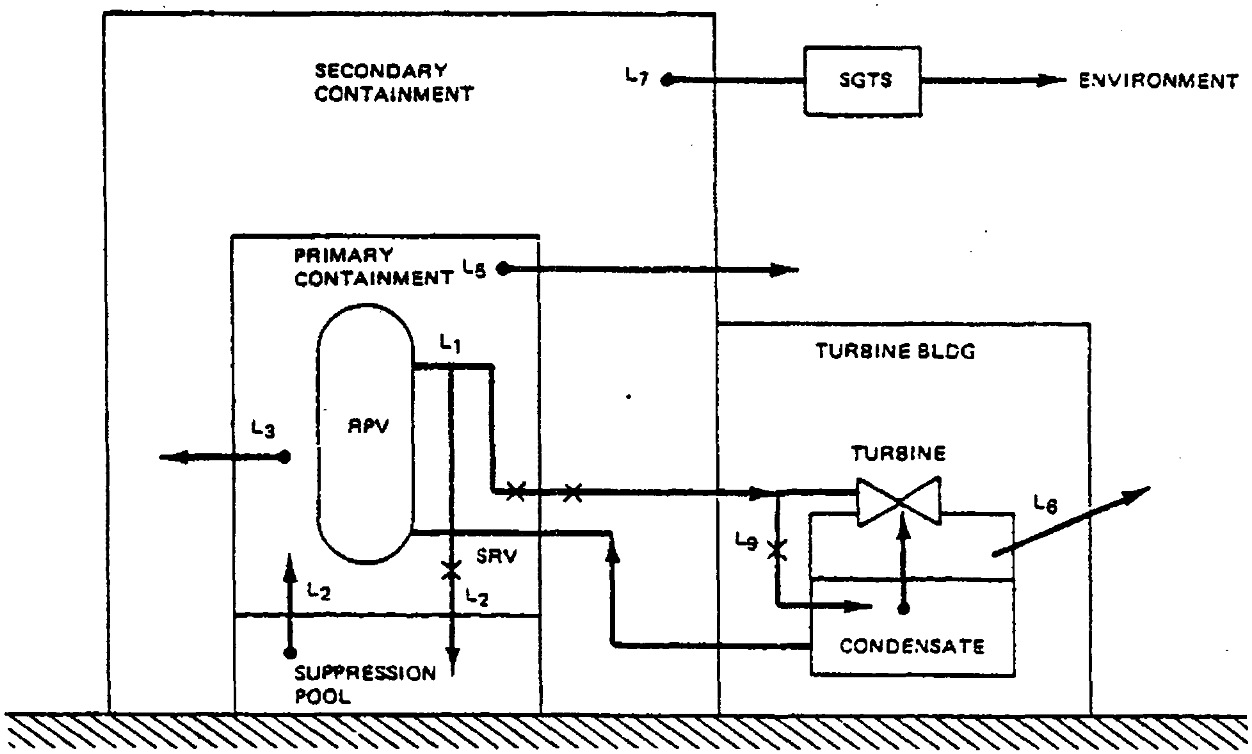


Figure 4.5.1. Fission Product Containment and Leakage Pathways

5. OTHER ATWS CONSIDERATIONS

5.1 DIVERSITY CONSIDERATIONS

The primary diversity provided by ARI is in the use of an "energized-to-trip" circuit versus a "deenergized-to-trip" circuit in the output devices of the current scram system. In addition, the relays in the ARI will be from a different manufacturer than the output relays called scram contactors (scram breakers) which directly deenergizes power to the solenoids in the Hydraulic Control Unit (HCU) in the current scram system.

Functional diversity currently exists in the sensors of the scram system. This diversity in the current scram system is depicted in Table 5.1-1. All plant transients in Table 5.1-1 have at least two diverse means (level/pressure, valve position, or flux/radiation sensors) for initiating a scram signal. Eight out of the nine transients in Table 5.1-1 have three diverse means for initiating a scram signal. The sensor diversity provides protection from failures due to functional deficiency of a sensed scram variable, miscalculation of sensors, and other maintenance errors by a single individual or crew.

5.2 EQUIPMENT OVERPRESSURE PERFORMANCE SUMMARY

A list of components affected by an ATWS was made for plants of the BWR/4 through the BWR/6 product line. The components were divided into logical subsystems, and a conservative estimate of the peak pressure seen by that section was calculated. (Peak pressures were calculated for the vessel-limiting case of 1500 psig vessel bottom pressure, and for currently published peak pressures.) The suggested design pressure of components within the scope of GE designed parts was then listed. Rules for Service Level C conditions (defined by ASME Section III Subsection NB) were applied to the suggested design criteria to show the acceptable limit for the component during an ATWS. The results show that, despite the conservatism used in estimating the peak pressure, no piping exceeds the Service Level C pressure limit. The list of components is given in Appendix A.3.

ATWS was assumed to affect all valves from the vessel to the last closed valve upstream (i.e., vessel-side) of a relief valve, storage tank, or the suppression pool. All piping and valves from pump discharge to vessel inlet were considered (where appropriate). It was assumed that pump discharge piping and valve have been designed so that operability is not impaired by running at pump shut-off head (i.e., only pressures caused by vessel peak pressures were considered).

The peak pressure seen by each region of the reactor was taken from computer simulations of the ATWS event for the given plants. The bounding ATWS event chosen was a simultaneous closure of all MSIV's, with an accompanying failure to scram. Using the peak dome pressure, steam line pressure, and maximum recirculation pump head obtained, auxiliary pump and conservative static heads were added to arrive at the estimated peak pressure for each system.

For the limit case the recirculation suction pressure and the vessel-bottom pressure were assumed to be 1500 psig. Peak recirculation pump discharge pressure was obtained by adding 50% of full load pump head to the peak recirculation suction pressure. The pump has been decreased below this value during the ATWS pump countdown before the peak pressure occurs.

5.3 BORON EFFECT ON BWR COMPONENTS

The systems exposed to some concentration of sodium pentaborate after an ATWS event (or inadvertent injection) can include the following:

a. The reactor vessel, vessel internals, fuel, and the recirculation and RHR systems. The maximum concentration of borate in these systems is expected to be 1000 ppm equivalent boron during hot standby, with the RHR system acting in its steam-condensation mode. At cold shutdown conditions the maximum borate concentration is about 750 ppm boron.

b. The suppression pool. S/RV discharge to the suppression pool during isolation in an ATWS transient gives the pool a concentration of about 3 ppm boron.*

c. The main turbine condenser. The steam bypassed to the main condenser can give the condensate inventory a concentration of up to 30 ppm boron.*

d. Main steam lines, main steam line drains, and MSIV leakage control system. The concentration in different locations can be anywhere between 1 ppm boron (the low estimate for steam boration at reactor temperature) and 1000 ppm boron, the maximum concentration in the entrained moisture. RCIC turbine and steam lines will be exposed to dry steam, containing between 1 and 20 ppm boron (high estimate of steam boration).

e. Radwaste will see at most 750 ppm boron during the borate cleanup, except for the evaporator and the system downstream, which could see at most 25% by weight pentaborate.

f. The reactor water cleanup system will be isolated from all systems containing borated water. It will only be used for the final "polishing" of reactor water, and should see no more than ∿ 1 ppm boron.

g. The offgas system will be exposed to borated steam if steam-jet air ejectors continue to operate after an ATWS transient without isolation. The concentration in the steam will be between 1 and 20 ppm boron.

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*The estimates of boron transfer to the suppression pool and main condenser are based on conservative calculations of water carryover from the reactor. In addition, the worst-case value of 20 ppm boron in steam was used; the actual valve is considered to be closer to the lower estimate of 1 ppm boron. If the latter steam carryover is used, the concentrations in the suppression pool and condenser are cut by a factor of 5.

At the beginning of shutdown after any ATWS transient, the reactor water pH is between 7.5 and 8 and the borate concentration is no more than 750 ppm boron. The borate species present are pentaborate hydrate and other poly-borate hydrates; the exact breakdown is determined solely by the existing shutdown conditions, not by preceding conditions, since known borate chemistry includes no irreversible reactions.

Removal of borate from the reactor water may require a shutdown of as long as several months. GE expects no damage to the exposed BWR systems to occur in such short periods, although the available information on the properties and effects of sodium pentaborate on BWR operation is rather sparse. Only the radwaste system will continue to be exposed to borate after startup, since borate water in storage will remain to be treated.

Relevent data and information are:

a. Corrosion tests conducted with 15000 ppm potassium tetraborate, a similar chemical, at 680°F and pH 10 indicated no detrimental effects on zirconium alloys, stainless steels and nickel alloys. The lower temperature and concentration during an ATWS transient and the subsequent shutdown should further reduce possible corrosiveness to these metals and to carbon steels, on which no information is available.

   In addition, $\sim$ 13% by weight pentaborate decahydrate at pH $\sim$ 7 and 80°F has caused no known cracking or corrosion in the standby liquid control tanks in existing GE BWR systems. These tanks are stainless steel, and have been exposed to borate for many years.

b. Standard manufacturers' information shows that aqueous perborate solutions (pH $\sim$ 10) cause little or no damage to elastomeric gaskets within the temperatures ranges recommended for the materials. The low concentrations of pentaborate, at a more neutral pH, are not expected to affect the exposed elastomers.

c. Except for its effect on reactivity, an aqueous pentaborate solution of 1000 ppm boron has essentially the same properties as water. The presence of pentaborate is not expected to affect fluid flow of heat transfer in any system during ATWS or the subsequent shutdown. The radwaste system is designed to handle relatively concentrated solutions such as the 25% pentaborate solution produced by the evaporator.

d. Precipitation of pentaborate or other borates from these essentially neutral -pH solutions will not occur for temperatures greater than 70°F. No caking on surfaces or clogging of nozzles is expected, even in Radwaste.

e. The offgas recombiner catalyst is not expected to be inactivated by exposure to borated steam. Any borate deposits can be washed away with wet steam, since borates are soluble at steam temperatures.

The final result of an ATWS event will be the production of 2000 to 11000 ft^3 of drummed waste during the total borate cleanup process, including processing of stored borated water after startup. The time required for cleanup, and the amount of waste produced, depend on the type of radwaste and condensate treatment systems. The ATWS waste can be monitored and stored on-site in temporary buildings. No excessive personnel exposure should result.

The available evidence supports the expectation that the injection of sodium pentaborate during an ATWS transient is not expected to have any effect on GE BWR system's functions and materials. Additionally no significant long term effects are expected.

5.4 BORON MIXING

5.4.1 Definition of Mixing Efficiency

Liquid boron injected into the reactor is most effective if all of it stays uniformly distributed in the core. However, since the reactor coolant keeps flowing through the core, the liquid boron will spread into the other regions of the reactor as well. Therefore, for comparison purposes a reference "perfectly mixed" condition is defined as one in which, at a given point in time during the ATWS, all the liquid boron is uniformly mixed throughout the coolant within the reactor coolant pressure boundary. Since non-uniformities in liquid boron concentration are possible and expected, the mixed condition at any time is related to the reference condition through a mixing efficiency defined as follows:

$$\eta(t) = \frac{W(t)}{W_{pm}(t)}$$

where

$\eta(t)$ = Mixing efficiency at time t

$W(t)$ = Amount of liquid boron in thecore at time instant t

$W_{pm}(t)$ = Amount of liquid boron that would be present in the core at time t, if perfectly mixed condition is assumed.

$\eta(t)$ is also equal to the ratio of the average concentration of boron in the liquid part of the coolant in the core to the average concentration throughout the coolant in the RPV (Reactor Pressure Vessel). The actual time-varying mass of liquid in the vessel and recirculation loops was used in calculating the vessel mixed boron concentration. When boron is injected inside or near the core, the average concentration in the core could be greater than the average value in the RPV, and thus the mixing efficiency as defined above could be greater than 100%. However, as mixing and dispersal continues the efficiency will eventually approach unity.

5.4.2 Discussion of Boron Mixing Process for Injection Through the Jet Pump Instrumentation (JPI) Lines

Figure 3.3-2 shows the schematic of the boron injection arrangement using the JPI lines as the points of entry into the RPV. This design is applicable to BWR's which in an ATWS, supply makeup water outside the core shroud by HPCI (generally BWR/4's). The discharge side of the SLC pump is connected to the JPI lines outside the containment. These JPI lines are connected to the upper part of the jet pump diffusers inside the reactor pressure vessel. If the SLC pumps are initiated at time T_0, the liquid boron will first reach the RPV at a time T_1, where (T_1-T_0) is the "transport delay" outside the RPV and is equal to the time taken by the liquid boron flow to displace all the water in the pipeline from the pumps to the jet pump diffusers.

Figure 5.4-1 shows typical schematic details of the connection of the upper instrumentation tap with the top of the jet pump diffuser. When liquid boron is injected through this line, it issues into the diffuser flow through ten 0.09" diameter holes in the wall of the diffuser. The resulting average jet velocity at a total flow rate of 86 GPM is approximately 22 ft/sec. This high velocity combined with the effect of high Reynolds number of the flow (of the order of 10^6 with 10% of the rated flow through the jet pump diffuser) causes thorough mixing of the liquid boron and the diffuser flow. Simulation test observations confirm this fact. The flow and mixing of liquid boron injected through the JPI lines was tested in a transparent simulation model. Some features of the test model are:

a. 1/6 scale, slab geometry representation of the reactor.

b. The geometry of the reactor intervals is simulated in a simplified way.

c. Simulation includes the portion of the reactor from bottom of the lower plenum to the bottom of the separator skirt.

d. Reactor coolant flow is simulated with flow of water at room temperature and pressure.

   The model has provision for simulating forced circulation of the coolant by means of pumps and natural circulation is provided by airlift created by bubbling air through the simulated core region.

e. The liquid boron injection into the JPI is simulated by a solution of sodium bromide which simulates nearly all density difference between liquid boron solution and reactor coolant at 550°F.

Flow and mixing patterns were observed in the test model with the help of a colored dye injected with the liquid boron flow. The observations indicated that at all core flows greater than approximately 5% of the rated value, the following conclusions are applicable.

a. The liquid boron injected through the JPI at the top of the diffuser thoroughly mixes with the diffuser exit flow.

b. The jet pump exit flow fills up the lower plenum from the bottom.

These results are presently being further confirmed by quantitative measurements from the tests. Based on the above experimental observations from the simulation tests, the following simplified picture of the boron mixing process can be constructed. Consider the condition in which the reactor core coolant flow (by natural circulation) is at a constant rate of $\dot{Q}_c$ lbm/sec and liquid boron flow constant at $\dot{Q}_B$ lbm/sec into the jet pump diffusers. After liquid boron enters the JP diffusers, the water flowing out at the exit of the jet pumps will have a boron concentration of $\dot{Q}_B C_B/(\dot{Q}_c+\dot{Q}_B)$ PPM, where C_B is the boron concentration in the sodium pentaborate solution in the SLC tank. Core coolant with this concentration will appear at the core inlet at a time T_2, where (T_2-T_1) is time required by the jet exit flow to "displace" all the water in the flow path from jet pump suction to the bottom of the active fuel. To facilitate further discussion we define (T_2-T_1) as the "first pass delay" which is also the time to delay from the instant at which liquid boron enters the RPV to the time at which it begins its effect on the core power. As the incoming borated water displaces the coolant in the core, core average concentration increases. This increase is modeled linearly from 0 PPM at time $\leq T_2$ to $\dot{Q}_B C_B/(\dot{Q}_C+\dot{Q}_B)$ at time T_3. Where (T_3-T_2), defined as the "core passage time", is the time required for the core inlet flow to displace all the core coolant between the bottom and the top of the active fuel. The borated core exit flow then flows through the jet pumps, and there picks up a higher concentration equal to $2\dot{Q}_B\ C_B/(\dot{Q}_C + \dot{Q}_B)$ due to the liquid boron still being pumped into the JP diffusers. This new concentration will appear at the core inlet after the lower plenum water is again displaced by the jet exit flow. Thus at a time T_4, the boron concentration at the core inlet will be $2\dot{Q}_B\ C_B/(\dot{Q}_c + \dot{Q}_B)$, where $T_4-T_2)$, defined as "loop delay time", is the time taken by a particle of water to traverse the natural circulation flow loop through the core, separators, downcomers, jet pump diffusers and lower plenum. The average concentration in the core will rise to $2\ \dot{Q}_B C_B / (\dot{Q}_c + \dot{Q}_B)$ at T_5, where $(T_5 - T_4)$ is again the "core passage time". This set of events from T_2 to T_4 will repeat as long as the core flow and the liquid boron are maintained at their constant values. In the foregoing discussion, the steam leaving the separators is assumed to be made up by water flow into the reactor (by HPCI or feedwater) keeping water level constant, and boron that would be lost with the steam is assumed to be negligible.

The buildup of boron concentration in the core water by the process discussed above is shown in Figure 5.4-2 by the solid curve 1. Here the values of the different transport and delay times are calculated for a typical 251" reactor at 15% core flow and 86 GPM liquid boron flow, and listed in Table 5.4-1. When core flow and liquid boron are assumed to be constant the boron buildup follows a nearly staircase function as indicated by curve 1. However, even at constant average core flow, the flow loop times would be actually different for different water particles because of flowing through differently located fuel bundles, separator, etc. Further, the effect of turbulence in the reactor flow tends to diffuse the sharp boron concentration front that would otherwise be sustained by pure transport. For these reasons the average boron concentration development in the core water is expected to fall within a band shown by the shaded area which envelops the curve 1.

Curve 2 in Figure 5.4-2 shows the buildup of concentration when 100% is assumed (perfect mixing as discussed in 5.4-1). It is seen that this line lies approximately in the middle of the shaded band of actually expected concentration buildup.

In applying the discussion of Figure 5.4-2 to the BWR's ATWS analysis, the following must be considered:

a. The first pass delay inside the vessel core passage time and the flow loop delay times are inversely proportional to the core flow rate. Similarly the increase in concentration experienced by the core flow in each of its passage through the jet pumps is also inversely proportional to the core flow rate. Therefore, at core flow rates greater than 15%, curve 1 in Figure 3 will be closer to the 100% efficiency curve 2 (and the shaded band will be narrower). At core flows smaller than 15%, the "steps" of the curve 1 will be larger, but the curve 2 will still pass through the middle of the corresponding enveloping shaded area.

b. In a typical ATWS base case for BWR's with HPCI (i.e., generally BWR/4's), the core flow rate at the time liquid boron first enters the RPV is generally near 20%. Further between this time and hot shutdown, the calculated core flows are high enough to sustain a mixing process similar to the one discussed above.

Based on the above discussion, the following assumptions made in the ATWS analysis for injection through the JPI lines, are considered appropriate.

a. The delay (T_2-T_0), between the start of the SLC pump and the time at which boron first becomes effective in the core is 60 seconds.

b. The mixing efficiency is assumed to be 95%. Here although 100% mixing efficiency is appropriate, a 5% margin is allowed for unknown uncertainties. One such uncertainty could be with respect to circumferencial distribution since fewer than all the jet pumps are used for boron injection.

Under these assumptions, the boron concentration buildup in the core water will also be linear. A comparison of this (Curve 3) with the Curve 1 indicates that the assumptions are justifiable.

5.4.3 Discussion of the Boron Mixing Process for Injection in Core Spray Sparger

Figure 5.3-4 and 5.3-6 show the schematic of boron injection arrangement for the BWR's which, in ATWS transients, supply makeup water inside the core exit plenum (all BWR/5's and BWR/6's and some BWR/4's belong to this class).

In this case the liquid boron is injected into the HPCS line very near the RPV downstream of the check valve nearest to the RPV. This arrangement is adopted because of the desirability to borate the cold HPCS water that comes in at the top of the core.

Test data for the flow pattern of the HPCS flow under ATWS core flow conditions is not available, and the mixing pattern of the HPCS jets with the two phase mixture in core exit plenum is very complex. For the purposes of the present discussion, certain "bounding" modes of flow and mixing can be postulated to determine bounding values of initial and loop delay times. Using these delays, the development of boron concentration in the core can be constructed in the same manner as was done for Jet Pump Instrument line (JPI) in Section 3.1.1.3.2. Possible modes of flow considered are:

MODE 1: The borated HPCS flow mixes with the core exist flow uniformly and after passage through the separators, downcomer, jet pumps and lower plenum becomes effective in the core. Subsequent flow passes through the core bringing increased boron concentration to the core.

MODE 2: In this mode the borated HPCS flow is assumed to partially mix with the steam water mixture in the core exit plenum and flow down into the core bypass region (providing negative reactivity), it is assumed to mix with core exit flow and then flow upward similar to Mode 1.

MODE 3: In this mode the borated HPCS flow is assumed to partially mix steam water mixture in the core exist plenum and flow down through core active and bypass regions. After filling up the lower plenum and control rod guide tubes it is assumed to fill up the core bypass and active fuel regions providing negative reactivity. After this, it would mix with the core exit flow and flow upward as in Mode 1.

Table 5.4-2 shows the initial delay and flow loop delay associated with the three different flow modes postulated above. The values shown in the table are for an assumed constant core flow rate of 10% and a HPCS jet entrainment mixing ratio of 1:10. It is seen from the table that the flow Mode 1 causes the largest initial delay and the flow loop delay is the same for all the three modes listed. Thus for boron mixing, postulating the flow Mode 1 would be most limiting. If this flow Mode 1 is assumed, then the expected buildup of average boron concentration in the core water with time is shown in Figure 5.4-3 for 10% core flow. Figure 5.4-3 also shows for comparison the concentration buildup if perfect mixing is assumed.

ATWS transient calculations indicate that a representative value of core flow expected during the period before hot shutdown is approximately 10% of rated. The appropriate bounding value of the initial delay time to be used for liquid boron, therefore appears to be 80 seconds (including the transport delay of ∿20 seconds in the pipeline outside the RPV) and the appropriate mixing efficiency appears to be nearly 100%. In the ATWS analysis for BWR/5's and 6's a 60 second initial delay and a mixing efficiency of 75% have been used. The latter is a traditionally used value for ATWS and is obviously conservative. The initial delay time used is slightly less than the value constructed above. However, parametric studies provided in the sensitivity runs (Section 3.2.1.4 and 3.3.1.4) show that with initial delays assumed as high as 180 seconds and mixing efficiency of 75%, the calculated ATWS consequences for pool temperature are acceptable.

5.4.4 Description of Boron Mixing Simulation Tests for Injection Through JPI Lines

The process of liquid boron mixing in the BWR under ATWS flow condition was simulated and tested in the laboratory using a scaled down model In the following are given the bases of simulation, a description of the test and a discussion of the applicability of the results to the prototype BWR, for the case of injection through JPI lines.

5.4.4.1 Bases of Simulation

Several flow and geometrical factors influence the mixing phenomenon. Important aspects of these are considered for simulation.

Geometry: A scaled down ratio of 1/6th is used. A two foot wide slab section of the prototype reactor is modeled as shown in Figure 5-4. The reactor components simulated include two jet pumps, lower plenum with two rows of control rod guide tubes, core section, three rows of steam separators, downcomers and liquid boron injection points.

Simulation of Reactor Coolant Flow and Liquid Boron Injection: Reactor recirculation flow is simulated with water at room temperature and atmospheric pressure. The injection of sodium pentaborate soluation is simulated by an injection of sodium bromide.

The coolant recirculation in the modeled is achieved by using an external recirculation pump and the action of the two jet pumps. Additionally, the natural circulation created by boiling in the core of the prototype reactor (the core flow during ATWS is mostly by this natural circulation) is simulated by air lift created by bubbling air through the core region of the model. By proper adjustment of velocities and density of sodium bromide solution conservative simulation of the liquid boron mixing in the BWR can be achieved.

Molecular Viscosity and Turbulence: Typical velocity of water in the prototype BWR are of the order of 10 ft/sec. Using the jet pump exit diameter as a characteristic length and fluid properties of saturated water at 550°F, this corresponds to a Reynolds number of the order of 10^6. Thus, the agitation due to boiling and the various obstacles to flow provided by the reactor intervals maintain the flow in the reactor turbulent in most regions. In the model the Reynolds number is lower due to the smaller scale, lower flow velocity and higher viscosity of water at room temperature. However, the flow still remains turbulent. The lower Reynolds number and the corresponding lower level of turbulence in the model would be a conservative simulation of mixing.

Effect Density Difference: In the reactor the sodium pentaborate solution injected is approximately 46% higher in density compared to the saturated water at the reactor pressure (this is mostly due to the lower density of water at 550°F compared to water at normal temperatures). Sodium bromide solution used to simulate the liquid boron in the model is also made concentrated enough to be 46% higher in density than the cold water in the model. In order to properly obtain dynamic similarity with respect to density difference effects, the flow velocity in the model is scaled down such that the modified Froude number (which is the ratio of negative buoyancy forces to the inertia forces) given by

$$Fr = \frac{\Delta\rho}{\rho} \frac{gL}{V2}$$

is the same in the model and also in the prototype reactor. Here ρ is the density of water recirculating in the reactor/model, $\Delta\rho$ is the density difference between the liquid boron/sodium bromide solution and the reactor/model water, g is the acceleration due to gravity, L is a characteristic length such as jet exit diameter in the reactor/model and V is a characteristic velocity such as jet exit velocity in the reactor/model. With the linear scale down ratio of G used in the model, the characteristic velocity of flow in the model is $1/\sqrt{6}$ of the velocity in the prototype reactor.

With the above simulation bąses used the ratio of various physical quantities in the model to those in the prototype reactor are as follows.

Table 5.4-3

Physical Quantity	Ratio Model/Prototype
Length	1/6
Velocity	$1/\sqrt{6}$
Time (Transport time, Fill up Time, etc.)	$\sqrt{6}$
Flow Pressure Drop	1/6th
Concentration (Ratio to injection solution concentration)	1

Measurements: Principal measurements made in the mixing simulation tests are core flow rate and NaBv concentration at various locations in the core. Both these are measured continuously as functions of time. The core flow is obtained by measuring the jet pump exit velocity by means of a pilot tube. Concentration is measured indirectly by measuring temperature by means of RTD sensor installed in the model. The basis for this is discussed below.

In addition to quantitative measurements, visual observation of the mixing process is provided by dye injection at convenient locations. The walls of the model and those of most of the internals are purposely made of transparent plexiglass to facilitate visual observation.

Basis for Using Temperature as a Measure of Concentration: The most dominant mechanisms of mixing both in the model and in the prototype BWR are turbulent dispersion and transport. Under these conditions, heat transfer and mass transfer processes are similar. Therefore, if the distribution of heat and mass sources/links and the boundary condition are identical, then distribution of a nondimensional concentration and a nondimenstional temperature will be identical throughout the model. Considering the mixing of sodium bromide in the model, the only source is due to injection at the JPI lines. Also the walls of the model and those of the internals in the model act perfectly impervious to the transfer of NaBv across them. Now consider the introduction of a heat source at the JPI lines by heating up the NaBr soluation to a higher temperature. The only other source/link of heat is due to the bubbling of air through the core of due to the heat added by the recirculation pumps if they are used for creating flow in the model. Also, the thermal boundary conditions at the model walls and the internals are nearly adiabatic. At any rate, the heating/cooling effect due to pumping/air bubbling and cooling effect due to heat loss through the walls can be measured and separated from heating effect due to the injection of hot NaBr solution. The similarity relationship between the heat and mass transfer can then be written as follows:

$$\frac{T - Ti}{T_S - Ti} = \frac{C - Ci}{C_S - Ci}$$

Ti, Ci = Initial temperature and NaBr concentration in the model water before the start of NaBr injection (Ci is generally zero in most tests).

T_s, C_s = Temperature and concentration of NaBv solution that is injected into the model at the JPI lines.

C = Concentration as a function of spatial location and time in the model during and after the NaBr injection.

(T-Ti) = Temperature rise due to NaBr injection only, as a function of spatial location and time.

Using the above equation, the concentration at any point in the model can be inferred from a temperature measurement at the same point. In the tests such an indirect measurement is verified by comparison with actual measurement of concentration of grab samples taken at a few spatial locations in the model several times during the test.

From a number concentration measurements in the core region of the model, the average concentration in the core is calculated as an approimate weighted average.

5.4.4.2 Description of a Typical Test

All tests are made at constant core flow conditions, although the core flow in a typical ATWS situation gradually changes with time. The following is a brief outline of a test in which core flow is driven by air lift. By controlling the rate of air supply for air lift the core flow (as indicated by the pilot tube velocity measurements at the jet pump diffuser exit) is adjusted to a desired valve. The model is allowed to run at this steady flow condition for about 10 minutes. During this period and throughout the whole test the readings from the various flow sensors and RTD's are monitored and recorded. The injection of NaBr solution at the JPI lines at a steady rate is then started. The injection is continued for approximately 30 minutes and then turned off. The model is allowed to run for an additional 10 minutes during which the air supply for air lift is maintained steady.

Figure 5.4-5 shows the temperature trace from a typical RTD sensor installed in the core region of the model. The line segment AB represents the steady condition before the test. When the core flow is started up at B the RTD response shows a slight gradient due to possible heat loss/gain to the outside and due to air bubbling. During the hot NaBr injection the temperature rise pronounced (line segment BC). After the hot NaBr injection is stopped but the core flow is still maintained by airlift, the RTD response shows a negative gradient representing heat loss from the hot contents of the model (line segment DE). The line CD' represents the predicted heat loss during the hot NaBr injection. The slope of the line CD' is a weighted average of those of the lines DC and DE. The vertical distance between the lines CD and CD' represents the temperature rise at the location of the sensor under consideration due purely to the hot NaBr injection and is a measure of the concentration buildup of NaBr at that location.

Visual and photographic observations are also made in addition to recording the quantitative reading from measuring sensors. A small quantity of a color dye is injected into the NaBr supply pipeline and the dye pattern pictures are photographed by still pictures at regular intervals of time. The whole test is also recorded on a movie film.

5.4.4.3 Applicability of the Test Results to the Prototype BWR

The concentration measurements as a function of time in the core region of the model can be appropriately integrated over the core region to obtain an average core concentration as a function of time. This time varying quantity contains both the transfer delay (time delay between the injection of NaBr and its entry into the core) and the mixing efficiency as defined in Section 5.4.1. Within the limitations of the model simulation bases assumptions this transport delay and mixing efficiency can be translated into those for the prototype BWR with the help of the model/prototype relationships given in Table 5.4-3. Some of the limitations imposed by the assumption in the simulation bases are discussed below.

At the concentration levels expected both in the model and in the prototype BWR, the NaBr solution or sodium pentaborate solute would have properties which are nearly the same as those of water. The simulation of sodium penaborate mixing by NaBr injection thus introduces only a negligible discrepency. However, as indicated before, the Reynolds number in themodel could be couple of order of magnitude smaller than that in the prototype BWR. However, the effect of this is to make the simulation more conservative. Another major difference between the model and the prototype is the 2D slab geometry used. First, the 2D aspect makes the simulation applicable only to those cases in which the injection of liquid boron in the BWR is nearly axisymmetry. Near axisymmetry is obtained when the rates of sodium pentaborate injection into each jet pump diffuser is the same. Secondly the use of slab geometric (rather than pie shaped slice geometry) makes the (non-dimensional) transport delay in the model greater than that in the prototype. But again this works in the direction of conservative simulation. The third difference between the model and the prototype lessens the density difference caused by cold sodium pentaborate injection. Since density difference is the major force which segregates NaBr/sodium pentaborate solution and lessens mixing efficiency the absence core heating in the model makes the simulation conservative again.

The results from the tests that are used for the discussion of section 5.4.2 are two conclusions from visual and photographic observation from several tests in which the model core flow was approximately 5% or greater. The first of the conclusions is that the NaBr solution injected at JPI lines at the top of the jet pump diffuser comes out perfectly mixed with the jet pump flow. Secondly the mixture coming out of the jet pump diffuser sweeps the lower plenum region before it enters the core. The latter is a conservative picture of the transport of NaBr in the lower plenum. The same conclusions would be true in the prototype reactor as long as the core flow is equal to or greater than 5%. Calculated core flow for BWR/4 ATWS indicate that the core flow is indeed stronger than 5% well past the hot shutdown point. Therefore the conclusions from visual observations of the model tests are appropriate for use in the BWR ATWS analyses.

5.5 LIQUID BORON EFFECTIVENESS

In the dynamic analysis of ATWS, the negative reactivity effect of liquid boron is assumed to be proportional to the amount of boron present in the core between the bottom and the top of the active fuel. The negative reactivity due to liquid boron at any time, is calculated from the equation:

$$R(t) = R_{HS} * \frac{W(t)}{W_{HS}(t)}$$

where

$R(t)$ = Liquid boron reactivity at time t ($)

R_{HS} = Liquid boron reactivity at hot shutdown condition ($)

$W(t)$ = The weight of boron present in the core (lbm)

W_{HS} = Amount of boron in the core necessary to maintain hot shutdown condition.

W_{HS} is obtained from a steady state, three dimensional core reactivity calcu calculation assuming the following conditions:

a. No voids

b. Core coolant at 280°C

c. Liquid boron uniformly distributed in the core

d. Critical rod pattern

R_{HS} is chosen to account for the combined void and doppler reactivity difference between the operating and no void-saturated hot shutdown conditions. It is selected based on calculations which have been performed with a three dimensional BWR core model which show that the assumed linear representation is a good approximation to determine core power level down to hot shutdown.

$W_{(t)}$ is calculated by using the boron concentration found for the time-varying mass of liquid in the vessel and recirculation loops, then applying it to the liquid portion of the core only. Thus, the simplified simulation accounts for the effect of displacement of borated core water by voids.

The simplified "displacement" of boron by voids approach described above is assumed in both the transient code formulation and the three-dimensional BWR core model against which it has been cross-checked during the boron injection nuclear shutdown process. The transient model only considers the active core, and accounts for boron which is also building up in the inactive region only through the indirect correlation to the 3-D core model. Dynamic changes in active core void fraction are conservatively assumed to affect all the boron worth as if it were all in the active region. The 3-D core simulator accounts for the boron in both regions.

A conservatism in current boron simulation by both computer tools involves the effect of boiling on boron concentration in the core. Current methods simply assume that void formation displaces borated water in the core (the same as if air took the place of some of the water). Currently the concentration of boron in all "drops" of water is not allowed to change as vaporization takes place. Realistically the concentration of boron in the water actually increases as some of the water is boiled while it moves up the core. For example, if the core inlet flow under quasi-steady-state conditions has 100 ppm boron concentration, and the power conditions are such that 10% exit quality still exists, then at the exit plane of the core the liquid flow has a concentration of 111 ppm since boron mass conservation requires that 100* (Inlet Flow) = 111* (Inlet Flow) (1-Xe). Both models (transient and 3-D steady-state) neglect this factor. Granted, the core is not totally filled with (in the above example) 100 ppm water because the voids associated with the 10% quality condition would reduce the total amount of liquid boron mixture in the core. The concentration of boron within the remaining liquid would not remain altogether at the inlet value - it would be increased as it approaches the core exit to a value determined by the exit quality.

This effect, if properly programmed, would increase somewhat the amount of boron calculated to be in the core and the negative reactivity it contributes in both models. Some shift of power (toward the bottom of the core) would also be expected, tending to increase the void fraction slightly and further decrease power.

The net result is definitely conservative, delaying somewhat the time of nuclear shutdown and making the calculated results of ATWS events more severe than expected. Future simulation techniques may be able to address this process more accurately, however, all analyses presented in this report contain the conservatism produced by the simplified simulation.

Table 5.1-1

SENSOR DIVERSITY FOR MAJOR TRANSIENTS

Scram Signals - Order of Occurrence

	Transient	Inputs from Pressure Differential Pressure Transmitters and Trip Units		Inputs from Position of Micro Switch Contact Opening			Inputs from Radiation Sensors		ARI Variables Reached	
		Reactor Pressure >1065 psig	Reactor Level <Level 3	Turb Cont. Valve Oil Pres. Set Point	Turb Stop Valve Pos. <90% Full Open	MSIV Pos. <90% Full Open	APRM >120%	MSIV HI Rad. >6 x Beck	Reactor Pressure >1150 psig	Reactor Level <Level 2
1	MSIV Closure	3	4			1	2		1	2
2	Turb Trip (with bypass)	3			1		2		1	M
3	Generator Trip (with bypass)	3		1			2		1	M
4	Pres. Regulator Failure (primary pressure decrease)	3	4			1	2		1	2
5	Pres. Regulator Failure (primary pressure increase)	2					1		1	
6	F.W. Flow Control Failure (reactor Water inventory increase)	3			1		2		1	
7	F.W. Flow Control Failure (reactor water inventory decrease)	3	1			2		4	2	1
8	Loss of Condenser Vacuum	3		4	1	5	2		1	M
9	Loss of Normal AC Power	4	5	2	1	6	3		1	M

Table 5.4-1

TYPICAL BORON MIXING TRANSPORT AND DELAY TIMES FOR JPI INJECTION

251" Reactor Vessel   15% Core Flow   86 GPM Liquid Boron Flow

Transport Delay in Pipeline Outside RPV	44 seconds
Initial Delay Inside RPV	26 seconds
Core Passage Time	12 seconds
Loop Delay Time	120 seconds*

*Conservatively assuming water level is in the normal range; lower levels (experienced throughout much of the event) would give shorter loop delay times.

Table 5.4-2

BORON TRANSPORT AND DELAY TIMES FOR INJECTION IN THE HPCS SPARGER

	Flow Mode Assumed		
	Mode 1	Mode 2	Mode 3
Delay Outside RPV*	20	20	20
Initial Delay*	60	25	50
Flow Loop Delay***	170	170	170

*HPCS assumed to be in operation. (3500 GPM)

**Delay calculated using reduced water level present during initial portion of events.

***Conservatively assuming water level is in the normal range; lower levels (experienced throughout much of the event) would give shorter loop delay times.

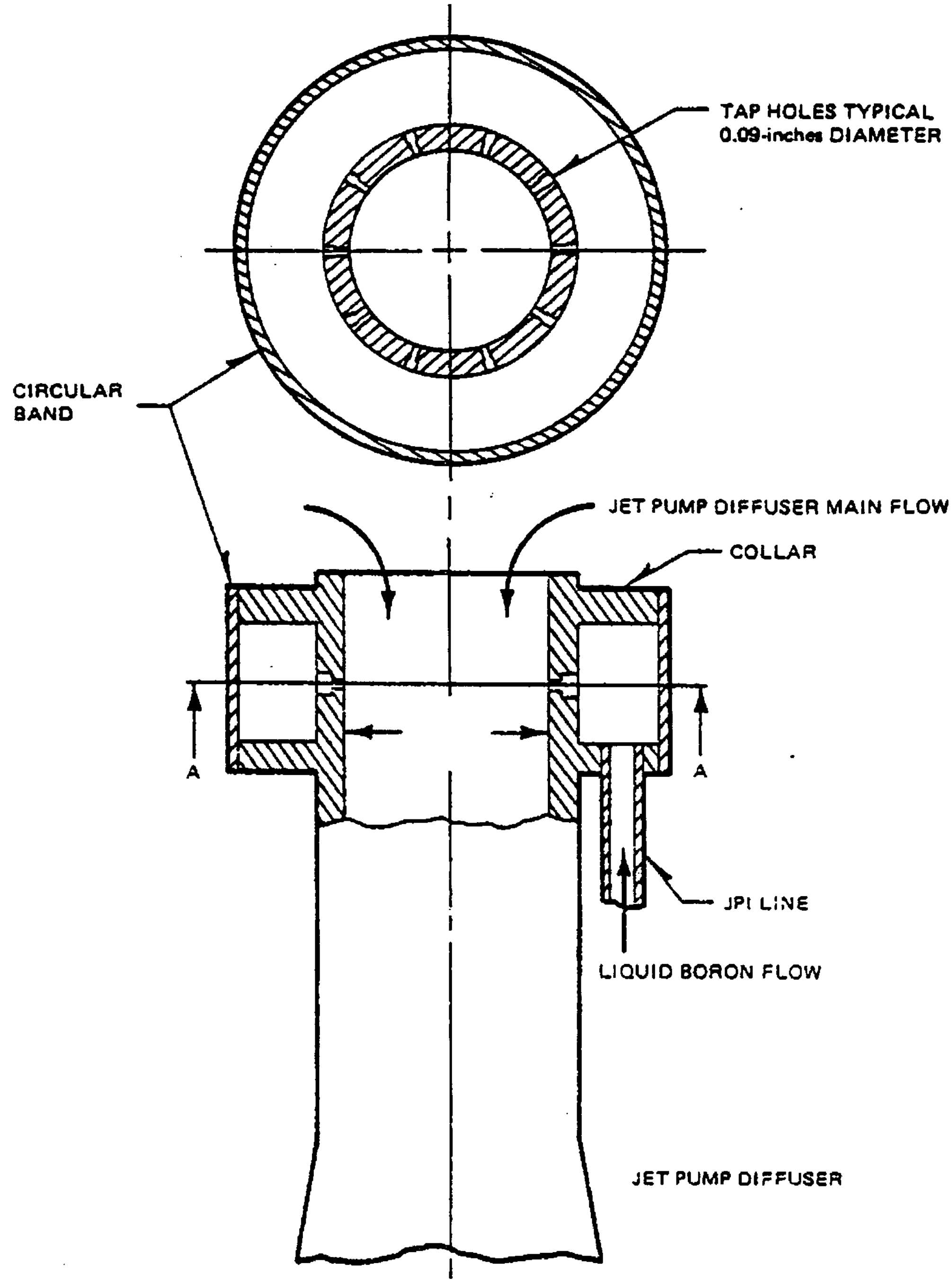

Figure 5.4-1. Schematic of Typical Connection of JPI Line With the Jet Pump Diffuser

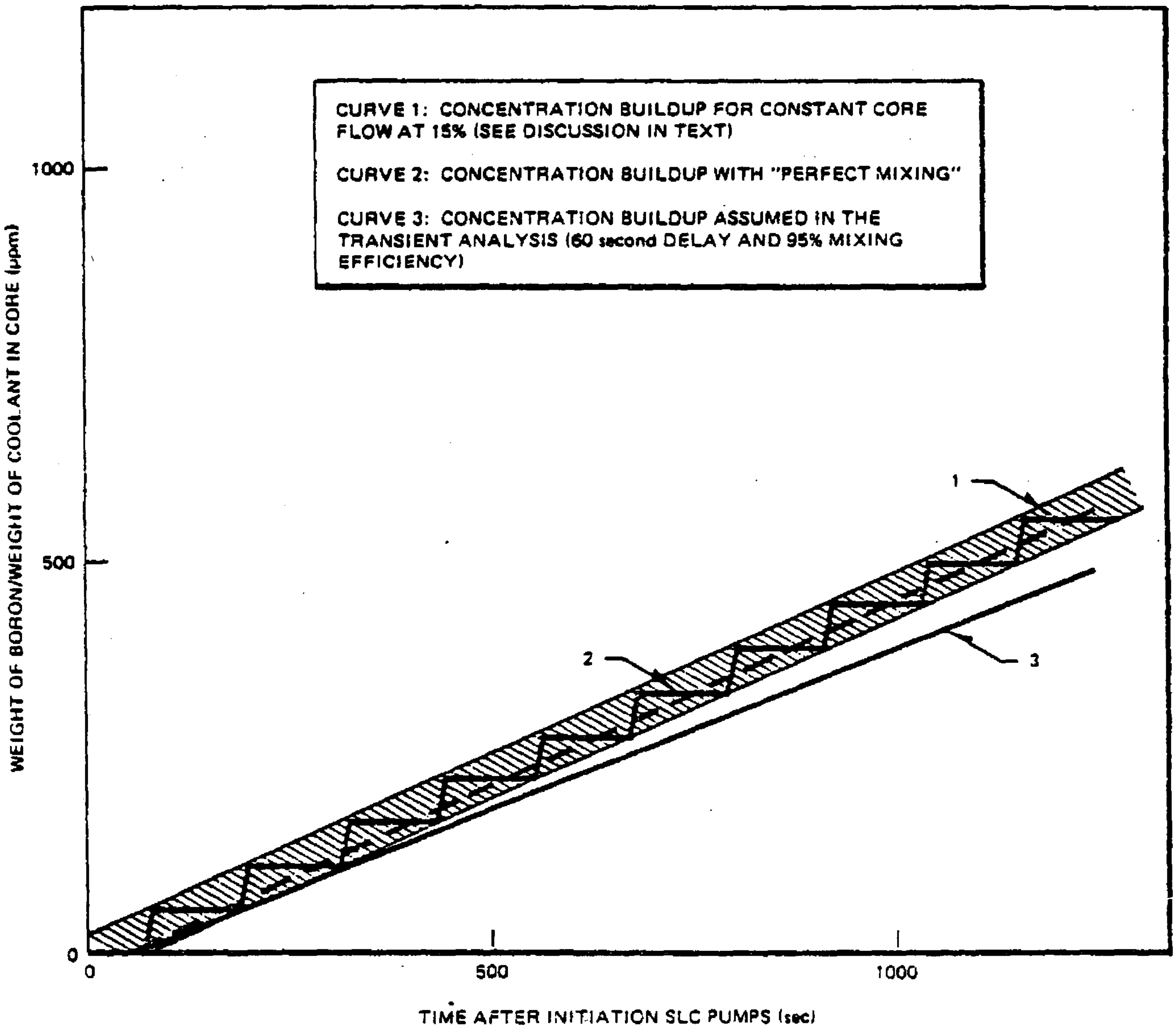

Figure 5.4-2. Boron Concentration Buildup in Core Water for JPI Injection Liquid Boron Flow Equals 86 GPM

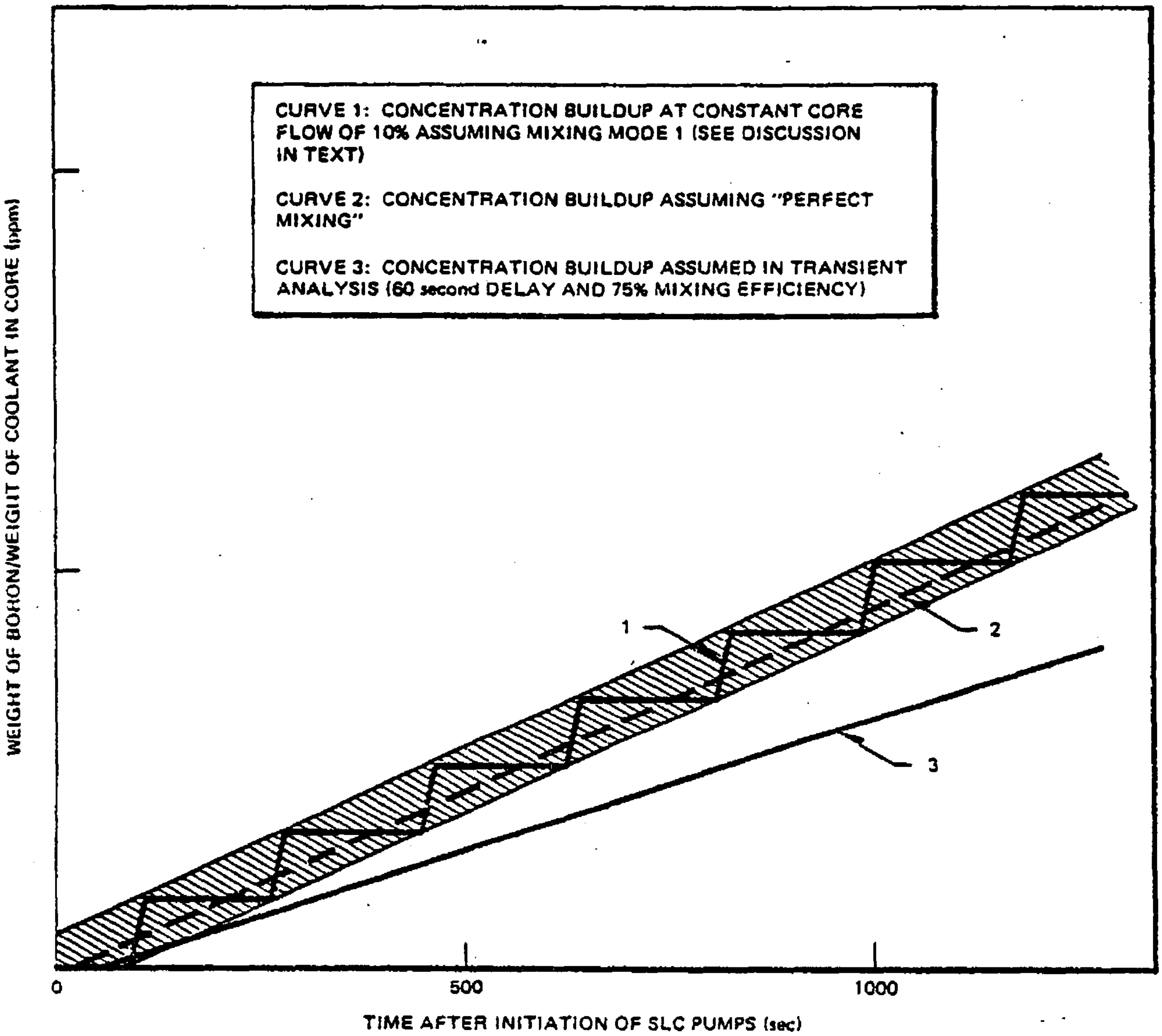

Figure 5.4-3. Boron Concentration Buildup in Core Water for HPCS Injection Liquid Boron Flow Equals 86 GPM

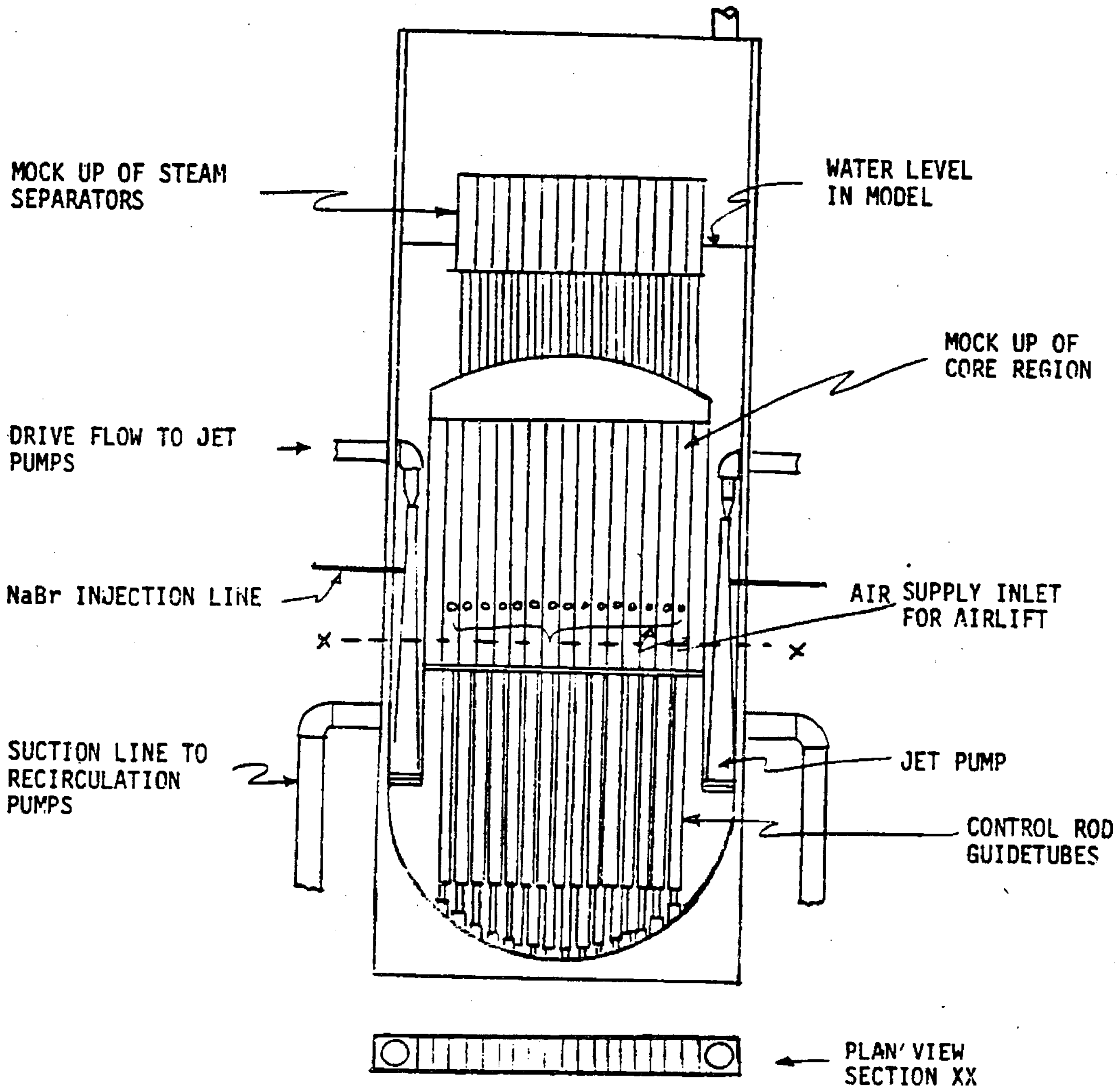

Figure 5.4-4. Schematic of Boron Mixing Simulation Test Setup

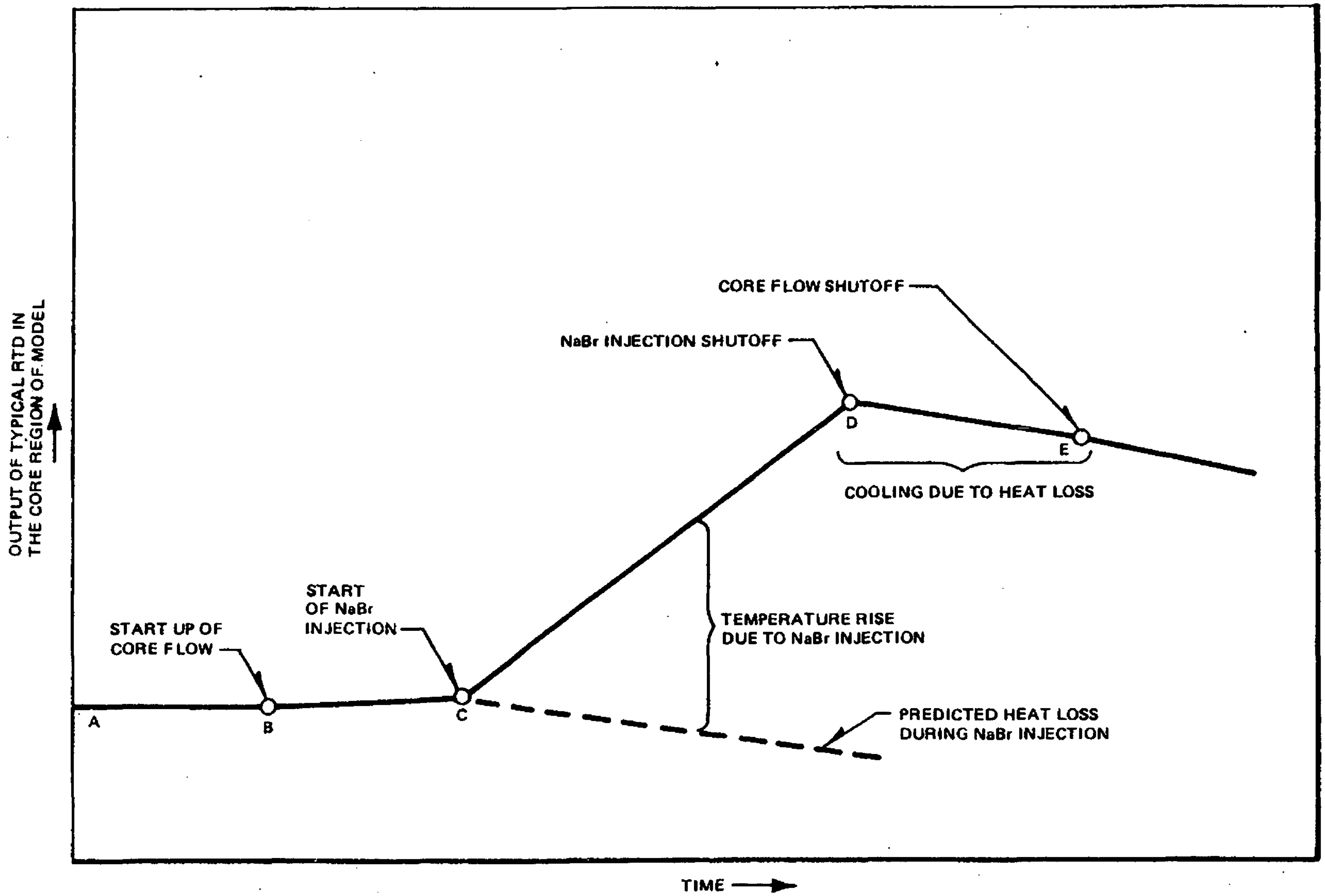

Figure 5.4-5 Typical Response of an RTD in the Model Core Region During Mixing Test

APPENDIX A.1
QUENCHER PERFORMANCE MEMORANDUM

Figure A.1.1. Evolution of the General Electric Quencher Device

Figure A.1.2. Results From Licensee Tests of Various Hole Patterns on Pipe Segment (Company Proprietary)

Figure A.1.3. Licensee Full-Scale Development Test Tank and Specimen Geometries (Company Proprietary)

Figure A.1.3A. Licensee Full-Scale Development Test Tank and Specimen Geometries (Company Proprietary)

Figure A.1.3B Licensee Full Scale Development Test Tank and Specimen Geometries

Figure A.1.4. Hole Sizes and Spacings Used on Licensee Quencher Developmert Tests (Company Proprietary)

Figure A.1.5. Floor Pressure as a Function of Local Subcooling for Licensee Full-Scale Development Tests (Subcooling Based on T_{STAT} = 233°F at 17.9 ft Submergence) (Company Proprietary)

Figure A.1.6. Condensation Regimes Observed in Licensee Full-Scale Development Tests (Subcooling Based on T_{SAT} = 233°F at 17.9 ft Submergence) (Company Proprietary)

Figure A.1.7. Range of Test Conditions for Licensee Full-Scale Quencher Condensation Development Tests (Subcooling Based on T_{SAT} = 233°F at 17.9 ft Submergence) (Company Proprietary)

Figure A.1.8. Licensee Full-Scale In-Plant Test Pool and Quencher Geometries (Company Proprietary)

Figure A.1.9. General Electric Quencher Geometries
(Company Proprietary)

Figure A.1.10. Results from Licensee Test on Segment of Full Scale Array in Small Scale Tank (Company Proprietary)

APPENDIX A.2

LOCAL-TO-BULK SUPPRESSION POOL ΔT

Figure 1. Temperature Histories for Extended S/RV Discharge (With RHR) From First Monticello Test

Figure 2. Temperature Vs. Time Time Caorso S/RV Test Phase 2
Extended Blowdown

Figure 3. BWR/4 Mark I Plants (Analytical Model)

APPENDIX A.3

(See Section 3.5)

The pressure for each subsystem was calculated as follows:

Main steam line

Line pressure (gauge)	given in Table 1
S/RV outlet	40% of main steam line pressure (S/RV discharge is sized to limit back pressure to 40%)
Main steam line drains	main steam line pressure + 10 psi head
Dome vent lines	dome pressure (Table 1)

Feedwater

Feedwater line	assumed losses: Sparger friction loss 25 psi; piping friction loss 20 psi; losses + dome pressure + 10 psi static head.

Recirculation loop

Loop suction	peak dome pressure + 10 psi head
Pump discharge	given in Table 1
Downstream of discharge valves	used pump discharge

Residual heat removal system

Head spray	connected to RCIC head spray
Recirculation suction	same as recirculation loop suction

Recirculation return	same as recirculation downstream of discharge valves
LPCI connection	dome pressure + 10 psi static head
Steam condensing line	same as main steam line

Reactor core isolation cooling system

Main steam to RCIC turbine	same as main steam line
Drain line	Main steam line + 10 psi static head
RCIC pump discharge	dome pressure + head spray nozzle loss, 20 psi + piping function loss 10 psi + static head 10 psi

High pressure coolant injection/core spray

Main steam to HPCI turbine	same as main steam line
HPCI/S pump discharge	dome pressure + sparger friction loss 26 psi + piping function loss 20 psi + static head 10 psi
Drain line	main steam line + 10 psi static head

Standby liquid control system

Main system, test lines, etc	vessel bottom pressure

The maximum allowable pressure for each component was calculated from the design pressure according to ASME Section III Subsection NB. Under emergency conditions (as defined in ASME III NB 3113.2) piping should not exceed 150% of design pressure (ASME III NB 3655.1). Valves are classified in two groups: those which are not required to operate during an ATWS must not exceed 120% of design pressure (ASME III NB 3526a). If a valve is to operate during an ATWS, "emergency will be specified as the normal condition," (ASME III NB 3526b). As many of the valves were designed prior to ASME III, it was felt that the intent of the NRC question could be answered by use of hydrotest data. The logic is that if a component has been hydrotested to a specific pressure (for a minimum of 10 min. - see ASME III NB 6224), that the same component will operate at least up to that pressure during an ATWS event.

Table 1

VALUES USED FOR CALCULATIONS

	BWR/4	BWR/5	BWR/6
Peak core exist pressure, psig	1281	1235	1283
Peak dome pressure, psig	1275	1228	1276
Peak steam line pressure, psig	1268	1192	1264
Recirculation pump head, psi	164	288	258
Peak recirculation pump discharge pressure, psig	1442	1519	1536
Peak vessel bottom pressure, psig	1296	1252	1300

DESCRIPTION OF TABLE 2

Product line

		BWR/4 Design	BWR/4 Maximum	BWR/5 Design	BWR/5 Maximum	peak system pressure (see note 2)
system or subsystem	Recirc. loop					
	suction	1510 (1380)		1510 (1285)		design press.
component	piping	1150	1725	1250	1875	Maximum (emergency) limit (see note 3)
	block	1750[h]		1325	2180	
	2" valves B33F029	0				

NOTES:

1. All pressures are in psig.

2. A conservative estimate of the peak pressure seen by this system or subsystem. This is calculated for an ATWS event in which the peak vessel-bottom pressure would reach 1500 psig. Values in parentheses are the system peak pressures associated with currently published ATWS peak pressures.

3. Pressure boundary emergency limits are defined as follows:

   1.5 x design for piping
   1.2 x design for turbines, pumps, and accumulators
   1.2 x design for valves

DESCRIPTION OF TABLE 2
(continued)

[h]This means that the valve has been hydrotested to the indicated pressure and shown operable subsequently.

[a]This means that the valve has undergone a finite element or similar analysis, and determined to be operable at the indicated pressure.

Legend of symbols used in Table 2

a	value was determined by analysis
h	hydrotest data (see explanation above)
o	specified by others
NA	not applicable to this plant
s	suggested max/1.2
-	information not available
*	At a vessel bottom pressure of 1500 psig, the system pressure exceeds the pressure boundary limit.

Table 2[1]

Component	BWR/4 Design Maximum		BWR/5 Design Maximum		BWR/6 Design Maximum	
Main Steam Line	1481 (1268)		1470 (1193)		1473 (1264)	
Piping	1118	1677	1250	1875	1250	1875
MSIV body		1818[h]		1833[h]		1662[h]
MSIV disc	1250	1500				2137[h]
S/R valve inlet		1811[h]		1811[h]	-	
S/R disc		2139[a]		2221[h]	-	
Safety valve inlet		2261[h]	NA		NA	
Safety valve disc		2705[h]	NA		NA	
S/R Valve Outlet	592 (510)		588 (480)		591 (510)	
Outlet		676[h]		678[h]	-	
Piping	o		o		o	
Steam drains	1491 (1278)		1480 (1203)		1488 (1274)	
Piping	1118	1677	1250	1875	1250	1875
3" valves: F016	o		o		o	
F019	o		o		o	
F020	o		o		o	
F021	o		o		o	
F034	o		o		o	
F035	o		o		o	

[1]Refer to pages A.3-6 and A.3-7 for legend and key to layout of this table.

-, o, NA, h: Refer to pages A.3-6 and A.3-7 for legend and key to layout of this table.

Table 2 (Continued)

Component	BWR/4 Design Maximum		BWR/5 Design Maximum		BWR/6 Design Maximum	
Steam drains (cont'd)	1491 (1278)		1480 (1203)		1488 (1274)	
2" valves: F001	o		o		o	
F002	o		o		o	
F005	o		o		o	
F038	o		NA		NA	
1" valves: F033	o		o		o	
3/4" valves: F017	o		o		o	
F018	o		o		o	
F023	o		NA		NA	
F025	o		NA			
F026	o		NA			
F027	o		o		o	
1/2" Lines	1485 (1275)		1489 (122)		1485 (1276)	
Piping	1118	1677	1250	1875	1250	1875
F003	o		NA		NA	
F004	o		NA		NA	
Feedwater	1540 (1330)		1544 (1283)		1540 (1331)	
Piping to block	1300	1950	1300	1950	1300	1950
Piping beyond block	o		o		o	
18" valves: F010	o		o		o	
F011	o		o		o	
F032	o		o		o	
3/4" valves: F030, 31	o		o		o	
F074, 75	o		-		o	
F076, 77	o		-		o	
F067,70, and 71	o		-		-	

-, o, NA: Refer to pages A.3-6 and A.3-7 for legend and key to layout of this table.

Table 2 (Continued)

Component	BWR/4 Design Maximum		BWR/5 Design Maximum		BWR/6 Design Maximum	
Recirculating loop, downstream of block	1582 (1442)		1644 (1519)		1627 (1536)	
Piping	1325	1987	1525	2288	1550	2325
Sample line piping	1325	1987	1325	1987	1325	1987
3/4" valves: F019, 20, 21, 22	o		o		o	
F059	NA		o		o	
Recirculating loop suction	1500 (1285)		1500 (1238)		1500 (1286)	
Piping	1150	1725	1250	1875	1250	1875
Block		1750[h]	2180			2180[h]
2" valves: F029	o		o		o	
F030	o		o		o	
F051	o		o		o	
F052	o		o		o	
3/4" valves: F024, 25, 25, 27, 28	o		NA		NA	
Discharge	1582 (1442)		1644 (1519)		1629 (1536)	
Piping	1325	1986	1600	2400	1650	2475
Pump	1615	1938	2037	2444	2132	2550
Flow control valve	NA		3071	3685	3071	3685
Block		1835[h]		2876[h]		2876[h]
Bypass		1947[h]		2525[h]		4590[h]
F074	NA		1675	2010	o	

o, NA, h: Refer to pages A.3-6 and A.3-7 for legend and key to layout of this table.

Table 2 (Continued)

Component	BWR/4 Design Maximum	BWR/5 Design Maximum	BWR/6 Design Maximum
Discharge (Cont'd)	1582 (1442)	1644 (1519)	1629 (1536)
3/4" drain and vents F068, 69, 70, 71	o	o	o
F066, 72, 73	NA	NA	NA
RHR Head Spray	1525 (1315)	1529 (1268)	1525 (1316)
Piping	1118[s] 1677	1250 1875	1250 1875
8" valves: F019	o	o	o
F022	o	NA	NA
F023	o	o	o
Recirculation Suction	1500 (1285)	1500 (1238)	1500 (1286)
Piping	1150[s] 1725	1475 2212	1475 2212
20" valves: F008	o	o	o
F009	o	o	o
F020	o	-	o
Recirculation Return	1582 (1442)	1644 (1519)	1629 (1536)
Piping	1325 1987	1475 2212	1475 2212
24" valves, F053	o	o	o
F050	o	o	o
F090	o	-	NA
F099	o	-	NA
Connect to LPCI	NA	1499 (1285)	1495 (1286)
Piping		1250 1875	1250 1875
F041		o	o
F042		o	o
F039		-	o

-, s, NA, o: Refer to pages A.3-6 and A.3-7 for legend and key to layout of this table.

Table 2 (Continued)

Component	BWR/4 Design Maximum		BWR/5 Design Maximum		BWR/6 Design Maximum	
Steam Condense	1481 (1268)		1470 (1193)		1478 (1264)	
Piping	1118	1677	1250	1875	1250	1875
F051	o		o		o	
F052	o		o		o	
F087	NA		o		o	
Vent and drain	o		o		o	
RCIC Turbine inlet	1481 (1268)		1470 (1193)		1478 (1264)	
Piping	1118	1677	1114s	1671	1114s	1671
Turbine	1250	1500	1250	1500	-	
4" valves: F063	o		o		o	
F064	o		o		o	
F045	o		o		o	
F076	NA		o		o	
3/4" valves: F072, 73	o		o		o	
Drain	1491 (1278)		1480 (1203)		1488 (1274)	
Piping	1118	1677	1425	2137	1425	2137
F025, 26, 38, 39	o		o		o	
F052, 53, 54	o		o		o	
F067, 68	o		o		NA	
Pump Discharge	1525 (1315)		1529 (1268)		1525 (1316)	
Piping	1280	1920	1400s	2100	1114s	1671
Pump	1500	1800	-		1500	1800

-, s, NA, o: Refer to pages A.3-6 and A.3-7 for legend and key to layout of table.

Table 2 (Continued)

Component	BWR/4 Design Maximum	BWR/5 Design Maximum	BWR/6 Design Maximum
Pump Discharge (Cont'd)	1525 (1315)	1529 (1268)	1525 (1316)
6" valves: F012	o	o	o
F013	o	o	NA
F065	o	o	o
F066	NA	o	o
F046	o	o	o
F019	o	o	o
F021	o	o	o
3/4" valves, F006, 7	NA	-	NA
F034, 35	o	o	o
RCIC Test	1525 (1315)	1529 (1268)	1525 (1316)
Piping	o	1400 2100	1290 1935
F022	o	o	o
F059	o	o	o
3/4" vent: F057, 58	o	o	o
HPCI/S Turbine Inlet	1481 (1268)	NA	NA
Turbine	1250 1500		
Piping	1118 1677		
8" valves: F001	o		
F002	o		
F003	o		
Pump	1535 (1325)	1539 (1278)	1535 (1326)
Pump	1500 1800	1376 1651	1575 1326
Piping	o	1575 2362	1250 1875

-, o, NA: Refer to pages A.3-6 and A.3-7 for legend and key to layout of this table.

Table 2 (Continued)

Component		BWR/4 Design Maximum		BWR/5 Design Maximum		BWR/6 Design Maximum	
Pump (cont'd)		1535 (1325)		1539 (1278)		1535 (1326)	
14" valves:	F004	o	1980	1820	2184 1820	o	
	F005	1650	1650	o		o	
	F024	o		o		o	
	F026	NA		o		o	
	F035	NA		o		o	
	F036	NA		o		o	
3/4" valves:	F031, 03	o		o		o	
	F021, 22	NA		o		o	
Test Line		1535 (1325)		1539 (1278)		1535 (1326)	
Piping		o		1575[s]	2362	1575	2362
10" valves:	F010	o		1170	1404*	o	
	F011	o		1170	1404*	o	
	F011	1650	1980	NA		NA	
	F012	1510	1812	NA		NA	
	F046	o		NA		NA	
	F023	NA		1170	1404*	o	
	F006	NA		o		o	
	F007	NA		o		NA	
	F012	NA		1170	1404*	o	
3/4" valves:	F064, 65	o		NA		NA	
Drain Line		1491 (1278)		NA		NA	
Piping		1118	1677				
F028, 29, 36, 37, 54, 55 and 56		o					
F071		o					

-, o, NA: Refer to pages A.3-6 and A.3-7 for legend and key to layout of this table.

Table 2 (Continued)

Component	BWR/4 Design	BWR/4 Maximum	BWR/5 Design	BWR/5 Maximum	BWR/6 Design	BWR/6 Maximum
SLC Main	1500 (1296)		1500 (1252)		1500 (13)	
Piping	1150	1725	1250	1875	1250	1875
Pump	1400	1680	1400	1680	-	
F003	o	1680	o	1680	1	1680
F004	1400	1400*	1400	1400*	1400	1400*
F006	o		o		o	
F007	o		o		o	
F008	o		o		o	
3/4" valves: F026, 27	o		o		o	
Test	1500 (1296)		1500 (1252)		1500 (1300)	
Piping	1400	2100	1325	1987	1325	1987
Accumulative	1500	1500	NA		NA	
F029	o		o		o	
F016	o		o		o	
F017	o		o		o	
F033	o		o		o	
F021, 24, 25	o		o		o	

-, o, NA: Refer to pages A.3-6 and A.3-7 for legend and key to layout of this table.

APPENDIX A.4
SCRAM DISCHARGE VOLUME CHANGES

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ASSESSMENT OF BWR MITIGATION OF ATWS, VOLUME I (NUREG 0460 ALTERNATE NO. 3)

GENERAL ELECTRIC

NEDO-24222
80NED021
Class I
February 1981

ASSESSMENT OF BWR MITIGATION OF ATWS, VOLUME I (NUREG 0460 ALTERNATE NO. 3)

Approved: R Gridley for
G. G. Sherwood, Manager
Safety and Licensing Operation

Approved: DC Ditmore
D. C. Ditmore, Manager
Projects Engineering

NUCLEAR POWER SYSTEMS DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

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1. INTRODUCTION

1.1 SUMMARY

The NRC Staff issued their technical report NUREG-0460, "Anticipated Transients without Scram for Light Water Reactors," Volumes 1, 2 and 3 in 1978. Volume 3 describes recommendations for mitigation systems for the various plant categories. In February 1979, the Staff requested General Electric to document the response of the BWR to proposed mitigation systems (February 15, 1979 letter from R. J. Mattson to G. G. Sherwood).

General Electric considers the NRC proposed ATWS mitigation systems to be unwarranted in light of the high reliability of the current BWR shutdown system. Nevertheless, General Electric has performed the assessment illustrating the capability of the timed, two-pump standby liquid control system (SLCS) to mitigate the consequences of the hypothetical ATWS event within limits imposed by the Staff. The transients evaluated in this report are based on generic analyses representative of a BWR/4 Mark I, a BWR/5 Mark II, and a BWR/6 Mark III. They are not intended to bound all plants in each class.

1.2 APPROACH USED IN THIS STUDY

The approach used in this study was to confirm the analyses given in NEDO-10349[1], and NEDO-20626[2], and to use these as a guide to identify the most limiting type of transients when failure to scram is considered. The conclusions from this step were then checked with updated calculations for the limiting cases involved for all reactor product lines. Although more extensive comparisons will be made and submitted on a later schedule, the current analyses are believed to be representative of the most limiting events from the set of anticipated transients requiring ATWS consideration.

[1]Analysis of Anticipated Transients Without Scram, March 1971 (NEDO-10349).

[2]Studies of BWR Designs for Mitigation of Anticipated Transients Without Scram, October 1974 (NEDO-20626).

Power densities, pressure rate characteristics, core performance parameters, recirculation characteristics, protection system capabilities, and pressure relieving capacities are very similar within each BWR product line. Hence, one plant was chosen to represent the basic behavior of all units within each product line.

By prior agreement with the Staff, sensitivity studies are provided only for the key parameters affecting suppression pool temperature. Additional sensitivity studies will be performed and submitted at a later date. Since the analyses of this report are not intended to bound all BWRs, any significant unique features which might exist on a particular plant will be considered later. Small differences in hardware selection are not expected to produce significant changes in the analysis presented here. Additionally, since NUREG-0460, Volume 3 suggests an implementation schedule of at least two years after rulemaking, this analysis whenever possible utilizes parameters expected to be present at that time, such as all 8x8 fuel.

Due to the extremely low probability of the occurrence of an ATWS, nominal parameters and initial conditions have been used in these analyses. This is consistent with the NRC Staff request. In spite of this approach, many conservatisms remain which may make the analysis conservative in light of the unlikely nature of the ATWS event. The major emphasis in this report was on the short term mitigation capability of the BWR. The actions necessary to achieve cold shutdown are also addressed.

The basis of these analyses is that the systems used to mitigate the postulated ATWS events would be designed such that the consequences do not result in a threat to public health and safety. Specifically, the systems are based upon meeting the following criteria:

1. The reactor coolant pressure boundary shall remain below emergency pressure limits.

2. The containment pressure shall remain below design limits. The containment temperature shall remain below local saturation temperature limits.

3. A coolable core geometry shall be maintained.

4. Radiological releases shall be maintained within 10CFR100 allowable limits.

5. All equipment necessary to mitigate the postulated ATWS event shall function in the environment (pressure, temperature, humidity) predicted to occur as a result of the ATWS event.

1.3 CONCLUSIONS

In response to the requirements of Alternate 3, set forth in NUREG0460, Volume 3, mitigation of the consequences of a postulated ATWS event have been assessed for the representative BWR/4/5/6 plants chosen. The conclusions drawn from this assessment are:

a. Recirculation Pump Trip (RPT) on high vessel pressure or low water level maintains vessel pressure within emergency limits, and quickly reduces power well below rated.

b. Alternate Rod Insertion (ARI), utilizing diverse logic and sensors on high vessel pressure or low water level, in conjunction with RPT, results in a scram after a delay of approximately 15 seconds. The combination of RPT and ARI results in low suppression pool temperatures, assures core coverage and maintains core temperatures well within acceptable limits, with no expected fuel failure.

c. Two-pump SLCS initiated on high vessel pressure or low water level in conjunction with RPT and after confirmation of an ATWS condition results in acceptably low suppression pool temperatures, core coverage and acceptable core temperatures for all initiating ATWS events.

d. The radiological analysis demonstrates that the limits of 10 CFR 100 are not exceeded for any ATWS event, and would not be exceeded, even if 100% of the fuel cladding were failed.

2. ASSESSMENT OF ATWS ISSUE

2.1 BACKGROUND

The NRC Staff in Volume 3 of NUREG-0460 has recommended a number of system design modifications for BWRs to reduce the likelihood and consequences of Anticipated Transients Without Scram (ATWS). As a result of extensive studies of transient frequency, scram system reliability and consequences of ATWS, General Electric believes that current BWRs with recirculation pump trip maintain a sufficiently low likelihood of ATWS with severe consequences that no additional design modifications are necessary.

The results of the General Electric study submitted in September 1976 in a report to the NRC indicate that the current likelihood of an ATWS event is less than $3x10^{-6}$ events/reactor year. With this low expected frequency of occurrence, the justifiable expenditure for avoidence based upon a value impact assessment by General Electric is not more than $100,000 per plant. The basis for this assessment was presented at the October 1978 ACRS meeting, and is consistent with the NRC Staff assessment in NUREG-0460 when adjusted for event frequency and likelihood of core melt.

The General Electric studies of scram system reliability have identified some relatively simple modifications to the BWR scram system that would increase its reliability by several orders of magnitude. It is General Electric's belief that no more than these additional ATWS prevention measures should ever be contemplated for resolution of the ATWS issue.

2.2 VALUE-IMPACT ASSESSMENT

Volume 3 of NUREG-0460 (Appendix F) contains a risk analysis which includes many unrealistic conservatisms. A few of these conservatisms are:

a. All ATWS events lead to core melt.

b. Unreliability of the scram system is $3x10^{-5}$/demand.

c. The number of transients with potential for severe ATWS consequence is eight/reactor year.

d. Alternate Rod Insertion yields only a 50% reduction in scram system unavailability; i.e., mechanical and electrical portions of the scram are assumed to be equally reliable.

However, even using the above conservative assumptions, Appendix F of NUREG-0460, Volume 3, shows that for no modifications (Alternate 1) the probability of core melt due to ATWS in all BWRs in 1992 is only a factor of two greater than the WASH-1400 estimate of non-ATWS core melt probability for all LWRs in 1992.

If Alternate 2 (RPT and ARI) is implemented for all BWRs, the Staff states in Appendix F of NUREG-0460 (using the same set of conservative assumptions) that the probability of core melt due to ATWS in all BWRs in 1992 is reduced to approximately one half of the WASH-1400 estimate of non-ATWS core melt for all LWR's in 1992.

Therefore it can be concluded that even using the conservative NRC assumptions stated above, implementation of Alternate 2 as defined by the Staff is all that would be necessary to make the ATWS risk acceptable based on the Staff's own statement in NUREG-0460, Volume 3 "...the present likelihood of severe consequences arising from an ATWS event is acceptably small and presently there is no undue risk to the public from ATWS." The value, therefore, of implementing any modifications beyond Alternate 2 is diminishingly small.

2.2.1 <u>NRC Core Melt Assumption</u>

Contrary to the assumptions in NUREG-0460, not all ATWS events lead to core melt. In fact, with RPT, there are not any ATWS events that lead to vessel pressure exceeding emergency limits. There are also a number of manual actions which an operator can take, given a failure to scram, to avoid core uncovery and suppression pool overheating. They include the following: 1) the operator can manually scram control rods, 2) the operator can selectively insert control rods in the normal mode, and 3) the operator can reopen the main steam line isolation valves to gain access to the main condenser as a heat sink and the feedwater system as an additional coolant source.

2.2.2 Scram System Availability

An extensive study* of BWR scram system reliability has been performed by General Electric including both relay type and solid state type logic systems. This study involved the performance of detailed failure mode and effects analysis (FMEA), event trees, and fault trees for all scram system components including the potential for common mode failure. Study results were documented to the Staff in September 1976. Section 7.3 is a summary of the BWR scram system reliability study and a description of its mechanical and instrumentation design.

This reliability study demonstrated that the current scram system unreliability is on the order of 0.8×10^{-6}/demand as compared to the value of 3×10^{-5} used by the Staff in NUREG-0460. This difference alone would reduce the Staff's estimate of ATWS core melt probability by a factor of 40 giving a 1992 BWR ATWS core melt probability of a factor of approximately 20 less than the non-ATWS core melt probability for all LWRs in 1992 in WASH-1400.

2.2.3 Transient Frequency

The above General Electric study and a similar Electric Power Research Institute (EPRI) study** of transient data from operating reactors showed that the frequency of isolation transients, i.e., those transients which have the greatest capability to lead to serious consequences occur about two times per reactor year for BWRs, as opposed to the eight as assumed by the NRC Staff. This difference reduces the Staff's estimated frequency of ATWS with potential for severe consequences by an additional factor of 4.

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*Transmitted via letter from E. A. Hughes, General Electric, to D. F. Ross, NRC, dated 9/30/76, **General Electric Company ATWS Reliability Report.**

R. C. Erdmann, et. al., **ATWS: A Reappraisal, Part II: Evaluation of Societal Risks due to Reactor Protection System Failure; Vol., 2: BWR Risk Analysis, EPRI NP-265, Part II, Vol. 2, August 1976. (Ltr from G. Lellouche (EPRI) to B. Rusche (NRC) dated September 30, 1976.)

Since the ATWS event frequency is composed of the sum of the products of each initiating transient frequency and the probability of failure to scram for that transient, General Electric's assessment of ATWS frequency is lower than NUREG-0460 by a significant amount.

2.2.4 Benefit for Alternate Rod Insertion (ARI)

The General Electric scram system reliability study has demonstrated that the mechanical portion of the scram system is roughly two orders of magnitude more reliable than the electrical portion, i.e., sensors and logic. This study has led to the conclusion that scram system reliability can be most effectively improved by utilizing diverse sensors and logic to actuate air header exhaust valves redundant to existing backup scram exhaust valves (i.e., add ARI). This system has been analyzed to show a total scram system unreliability of 0.7×10^{-8}/demand or a factor of approximately 125 reduction in scram system unreliability over the original scram system.

2.2.5 Value-Impact Conclusions

NUREG-0460 Volume 3, Appendix E documents the information provided to the Staff by industry on impact of various modifications. In summary, for BWR plants under construction, the proposed Alternate 2 requirements in NUREG-0460, Vol. 3, cost $700,000 to $1,200,000 per plant for direct costs and a factor of two times this in indirect costs giving a total cost (direct plus indirect costs) range of 2 to 3 million dollars per plant, excluding costs due to spurious boron injection. The proposed Alternate 3 requirements add $700,000 to $900,000 additional direct cost or 2 to 3 million dollars of additional total costs per plant beyond Alternate 2. This excludes costs due to spurious boron injection which would be approximately 5 million dollars per plant. Thus, the combined total cost for Alternate 3 modification for BWR plants under construction is in the 10 to 15 million dollars/plant range due to the additional instrumentation and control elements. This cost is not justified by the General Electric assessed maximum justifiable expenditure of $100,000 for ATWS-related scram improvements.

2.3 PREVENTION VERSUS MITIGATION

The General Electric studies of scram system reliability have identified the areas that limit scram system performance. In considering ways to reduce the risk of ATWS, it is prudent to consider additional areas of low cost prevention before expensive full mitigation measures are specified.

A major concern with requiring full ATWS mitigation capability is that high reliability scram system goals will no longer receive the priority that is currently given in the regulatory process. As pointed out by the General Electric study, if scram system reliability is a primary concern, then the first level of effort should be to improve its reliability and credit for its improvement should be recognized in the regulatory area.

3. DESCRIPTION OF ATWS ANALYSES

This section defines the considerations made in defining the scope and content of this submittal. Specifically, the classification of plants chosen for analysis is discussed, the reasons for choosing the particular plant transients analyzed are given, the plant conditions utilized and the assumptions employed are listed, the equipment and systems required to be operative are specified and discussed, and the analytical models used in the evaluation are defined.

3.1 CLASSIFICATION OF PLANTS ANALYZED

Analyses of ATWS events are provided for three classes of BWRs, consistent with General Electric product line designations. The three classes contained in this report, the BWR/4 (Mark I), BWR/5 (Mark II) and BWR/6 (Mark III), are described in Table 3.1-1. BWR/3 plants are not included in this report.

3.2 ATWS EVENTS ANALYZED

The first major GE study of ATWS was presented in NEDO-10349[1], which presents results of failure to scram for transients when the recirculation pump trip feature is implemented. This report evaluated peak pressures and peak heat flux during the early portion of the transients without scram until primary coolant conditions had returned to normal. Beyond this point, manual initiation of the standby liquid control system was shown to adequately shut down the reactor so long as the normal heat removal systems remained available. NEDO-10349 demonstrated that the most limiting transient with failure to scram is the MSIV closure event. A list of all anticipated operational transients which result in a scram demand is given in Appendix 7.1.

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[1]Analysis of Anticipated Transients Without Scram, Licensing Topical Report, March 1971 (NEDO-10349).

A later study was reported in NEDO-20626[1], which presents the results of the ATWS analysis performed in response to WASH-1270. In this report, General Electric confirmed the conclusion reached in NEDO-10349 that the MSIV transient is the most limiting ATWS event. This conclusion is applicable to all BWR/4, 5 and 6 product lines. NEDO-20626 presented detailed results for BWR/4, 5 and 6 product lines of the MSIV ATWS event assuming recirculation pump trip and automatic initiation of the standby liquid control system. The MSIV closure event is also the limiting transient for containment temperature conditions because this transient releases the largest amount of steam into the suppression pool and, therefore, results in the highest suppression pool temperature.

Inadvertent opening of a safety/relief valve (depending on the operator action taken following failure of manual scram) can result in high suppression pool temperatures. For this reason, the inadvertent opening of safety/relief valves is also included in the current study.

Finally, to demonstrate performance during most other types of ATWS events, the turbine trip/load rejection transient with scram failure is presented. Lower pool temperatures are reached during this ATWS event due to the accessibility of the condenser heat sink and the ability of the feedwater system to make up coolant inventory in the reactor vessel. This event also results in moderately high core power for the first few minutes into the transient, and hence represents additional fuel duty considerations.

Recent studies have shown that failures of the steam bypass system with turbine trip have a frequency of less than one per plant lifetime. Because of this, General Electric does not believe that the bypass failure event should be included in any required ATWS study. On the other hand, the consequences of the ATWS event are not expected to be significantly different than the MSIV closure event, thus limited information on the turbine trip without bypass ATWS event will also be presented.

[1]Studies of BWR Designs for Mitigation of Anticipated Transients Without Scram, General Electric Licensing Topical Report, October 1974 (NEDO-20626).

In summary, this report contains analyses of:

a. The MSIV closure ATWS event as the most limiting event.

b. Inadvertent opening of a safety/relief valve ATWS event to examine suppression pool temperatures.

c. Turbine trip/load rejection with bypass system operation ATWS to demonstrate other typical performance results.

d. Turbine trip with bypass failure as requested by the Staff.

3.3 PLANT CONDITIONS

Initial operating conditions for the typical plants used to represent each BWR Product Line are listed in Table 3.3.1. They are consistent with NUREG-0460 guidelines and represent a conservative nominal operating condition. The listing shows most parameters which are expected to influence the course of the ATWS events. Wherever possible, these parameters are normalized to the rating of the unit so that most effective generic use of this report can be made by units of all sizes within a product line, e.g., initial suppression pool volume is given in full-flow-minutes of rated feedwater.

Only Alternate 2 and Alternate 3 modifications are assumed to be implemented in these analyses. The features of Alternate 3 include all Alternate 2 features plus the implementation of automated SLCS repiped to new injection locations and to allow operation with both pumps. The automatic initiation of SLCS includes a two minute delayed actuation to allow for operator interruption in the event of spurious initiation after an actual scram has been confirmed.

The analysis for acceptable ATWS performance assumes the use of quenchers on the safety relief valve discharge piping on each plant.

Table 3.3.1
TYPICAL INITIAL OPERATING CONDITIONS

*

*Bars in right-hand margins indicate General Electric Company Proprietary Information deleted.

3.4 OPERATIVE EQUIPMENT AND SYSTEMS

3.4.1 Systems Utilized

The systems which perform the major functions during ATWS events and a brief description of system functional requirements are given below.

3.4.1.1 Recirculation Pump Trip (RPT)

The recirculation pumps trip in ATWS events as a result of either high vessel dome pressure (1150 psig) or low level (Level 2).

3.4.1.2 Safety/Relief Valves

The safety/relief valves open at their relief pressure set point and reclose at their closure pressure set point.

3.4.1.3 Control Rod Drives

The control rod drives insert the control rods in response to ARI logic actuations.

3.4.1.4 ARI Valves

The valves on the scram air header open on the same signals that initiate RPT to reduce the air pressure in the header so that the air-operated scram discharge valves will open, initiating reactor scram.

3.4.1.5 HPCI, HPCS and RCIC

RCIC plus one other high pressure water makeup system pump water into the vessel when the water level reaches or goes below the low level (Level 2) point. They turn off when the water level reaches the high level (Level 8) point.

3.4.1.6 **Standby Liquid Control System**

In the event that neither normal scram nor ARI have inserted the control rods into the core, the Standby Liquid Control System will pump the sodium pentaborate solution into the vessel to shut down the nuclear reaction.

3.4.1.7 **Suppression Pool and Containment**

The suppression pool absorbs the energy discharged into it from the relief valves. For Mark III containment automatic isolation occurs.

3.4.1.8 **Residual Heat Removal (RHR) System**

The RHR is capable of being placed into the pool cooling mode of operation. Both loops are needed for this. Later in the event, after the pool has been properly cooled and the reactor partially cooled down, it shall be capable of operating in the shutdown cooling mode. For those plants which have steam condensing mode, this mode is available to remove steam directly from the reactor vessel.

3.4.1.9 **Main Condenser**

For those events which do not isolate the reactor vessel, the main condenser is available for removing steam from the primary system.

3.4.1.10 **Main Steam Isolation Valves (MSIV)**

The MSIV's are able to close at some time during the event if their closure was not the cause of the event in the first place. They are also capable of being reopened following their closure.

3.4.1.11 **Feedwater System**

The feedwater system runs back to a lower flow rate upon receipt of an ATWS signal.

3.4.1.12 Turbine Pressure Controls and Bypass System

The turbine pressure controls and steam bypass system operate normally during those ATWS events which do not isolate the reactor vessel.

3.4.1.13 Condensate Storage Tank (CST)

This is the primary source of water for the HPCI, HPCS and RCIC.

3.4.1.14 Reactor Water Cleanup System

The isolation valves in the reactor water cleanup system close to prevent that system from diluting and removing the sodium pentaborate solution.

3.4.1.15 Standby Gas Treatment System

The standby gas treatment system is available for processing the air in the containment.

3.4.1.16 Diesel Generator

For that case where the loss of normal a-c power is the initiating event, the diesel generator sets are available to provide emergency a-c power to the plant.

3.4.1.17 Instrumentation

The following functions will be monitored during an ATWS event:

a. Reactor Power - LPRM/APRM
b. Control Rod Position Indication
c. Dome Pressure
d. Vessel Water Level
e. Suppression Pool Temperature

The following secondary functions will be monitored during an ATWS event:

a. SLC Storage Tank Level
b. Condensate Storage Tank Level
c. Core Flow
d. HPCI (or HPCS) Flow
e. Feedwater Flow
f. Drywell/Containment Pressure
g. Radiation Monitoring
h. MSIV Position

3.4.2 Equipment Performance Assumed in Analysis

Characteristics of the important pieces of equipment used to mitigate the consequences of failure to scram are listed in Tables 3.4.1 to 3.4.3. The BWR/4 characteristics are given in Table 3.4.1, the BWR/5 characteristics are given in Table 3.4.2, and the BWR/6 characteristics are given in Table 3.4.3.

3.4.3 ATWS Functional Logic

The functional logic for potential automated standby liquid control injection and other BWR ATWS mitigation and prevention features has been evolving since the early topical reports introduced high pressure and low level recirculation pump trip (NEDO-10349). Figures 3.4.3-1 to 3.4.3-6 show the current status of that functional logic development and the standby liquid injection path arrangements anticipated for BWR/4,5 and 6 plants if assigned Alternate 3 requirements from NUREG-0460, Volume 3.

Table 3.4.1

BWR/4

EQUIPMENT PERFORMANCE CHARACTERISTICS

Table 3.4.2

BWR/5

EQUIPMENT PERFORMANCE CHARACTERISTICS

Table 3.4.3

BWR/6

EQUIPMENT PERFORMANCE CHARACTERISTICS

Figure 3.4.3-1. Simplified ATWS Mitigation Logic BWR/4

Figure 3.4.3-2. Arrangement Schematic of SLCS BWR/4

Figure 3.4.3-3. Simplified ATWS Mitigation Logic BWR/5

Figure 3.4.3-4. Arrangement Schematic of SLCS, BWR/5

Figure 3.4.3-5. Simplified ATWS Mitigation Logic BWR/6

Figure 3.4.3-6. Arrangement Schematic SLCS, BWR/6

3.5 COMPUTATIONAL MODELS

3.5.1 REDY/ODYN Comparison

3.5.2 REDY Prediction of ATWS Stability

4. RESULTS OF ATWS EVENT ANALYSES

4.1 RESULTS OF ATWS EVENTS - BWR/4 MARK I

4.1.1 MSIV Closure Event

4.1.1.1 Overview of Response Without Scram

A detailed description of the sequence of events for MSIV closure is given below. The behavior of the plant is basically separable into an early or short term transient involving a sharp pressure rise and power peak, and a longer term portion that requires evaluation of coolant and containment conditions as the reactor is ultimately brought to shutdown.

The effectiveness of the recirculation pump trip (RPT) feature presented in NEDO-10349 and NEDO-20626 are reconfirmed by this analysis. It assists the relief valves in limiting the pressure disturbance acceptably and allows the establishment of a relatively low power generation rate for the long term portion of the transient. Figure 4.1.1 illustrates this first period.

Ultimate solution to the lack of scram situation must involve insertion of negative reactivity into the reactor, thereby bringing the reactor to a fully shutdown condition. The ARI is provided as an effective way to mitigate common-cause failures in the logic of the scram system. In the very remote case of ARI ineffectiveness, the automated SLCS provides further protection and shutdown capability. Coolant inventory is adequately maintained by using HPCI and RCIC available in each BWR to replace the coolant loss as steam flow leaves the primary system through the relief valves. Simply adding more water without inserting negative reactivity has the effect of raising the power generation rate and the amount of inventory leaving the system as steam.

The steam reaching the suppression pool continues to heat it and pressurize the containment until the power generation/steam flow can be reduced and finally terminated. The RHR system ultimately cools the pool and eventually the reactor also, if the Main Steam Isolation Valves cannot be reopened establishing flow to the main condenser, the preferred method of cooldown.

The detailed sequences of events and the results of analyses are described below.

4.1.1.2 Sequence of Events for MSIV Closure Transient

The MSIV Closure transient provides the most severe conditions following a postulated failure to scram. Listed in Table 4.1.1 is the sequence of occurrence and significant points of the transient with representative times when each event occurs.

The sequence of events begins with the nominal 4 second closure of the Main Steam Isolation Valves. With motion of the MSIV's, the pressure begins to rise which results in a reduction in void fraction and rapid increase in power. This sequence of events is shown in Table 4.1.1. For the BWR/4, this power reaches a maximum of 572% of the initial value at 4 seconds into the event and rapidly decreases again. In just under 4 seconds, the set point pressure of the relief valves is reached and they begin to lift and arrest the pressure rise. At about the same time that the relief valves are opening it is expected some of the fuel will experience transition boiling. Shortly after 4 seconds, the vessel dome pressure reaches 1150 psig, the maximum recirculation pump trip point, and both of the recirculation pumps trip.

A delay of 530 milliseconds is used from the time the 1150 psig is reached until the time that recirculation pump trip is effected. This delay time (500 milliseconds delay in the sensor and 30 milliseconds in the logic and trip) is consistent with industry experience. At the same time that the recirculation pump trip occurs, the logic chain is activated to start ARI and if necessary make the decision that an ATWS may have occurred and provide appropriate mitigation.

Pressure continues to rise for a short period of time until, at approximately 9 seconds into the event, it reaches its peak and begins to decline. The maximum pressure in the vessel is 1297 psig at 8.9 seconds. Those plants which have turbine-driven feed pumps, will begin to coast down as soon as the MSIVs are closed and, for this analysis, have lost their ability to overcome vessel pressure head at 20 seconds. The relief valves begin to close shortly after 20 seconds and pressure is then stabilized at the relief valve set point pressure, this part of the transient is shown in Figure 4.1.1. Some peak pressures at other points in the system are given in Table 4.1.4 as well as a

Table 4.1.1

BWR/4 MARK I MSIV CLOSURE WITHOUT ARI

Sequence of Events

1.	Nominal 4 second MSIV Closure - Scram Fails	0
2.	Pressure Rise Begins	1 Second
3.	Relief Valves Lift	4 Seconds
4.	Some Fuel Experiences Transition Boiling	5 Seconds
5.	Recirculation Pumps Trip on High Pressure, ARI is Initiated, and Timed SLCS Logic is Triggered	5 Seconds
6.	Vessel Pressure Peaks	9 Seconds
7.	Feedwater Flow Coasts Down to Lower Limit	20 Seconds
8.	ARI Fails	30 Seconds
9.	HPCI and RCIC Flow Starts after Level 2 Initiation	57 Seconds
10.	ATWS Logic Timer Complete, SLCS Starts	2 Minutes
11.	Liquid Control Flow Reaches Core	3 Minutes
12.	Water Level Reaches Minimum and Begins to Rise	5 Minutes
13.	RHR Flow Begins (Pool Cooling)	11 Minutes
14.	Hot Shutdown is Achieved	18 Minutes
15.	Containment Bulk Temperature and Pressure Peak	27 Minutes

summary of key parameters for this and other events. The same pressure signal (1150 psig) that initiated the recirculation pump trip will cause the opening of valves on the scram air header which will allow the air pressure in the header to bleed down. In the improbable event that scram has not already occurred from any of the several available signals, this reduced pressure will allow the scram discharge valves to open and the control rods to insert.

For this condition (ARI), tests have shown that the pressure in the header will have been reduced sufficiently in 15 seconds to allow the control rods to insert. The control rods will all be expected to be fully into the core in 5 additional seconds. ARI mitigates the ATWS situation and 25 seconds after the event began, it is essentially over. Water level will continue to drift downward as decay energy generates small amounts of steam, and when Level 2 is reached, the HPCI and RCIC will automatically start and replenish the vessel water inventory, from the condensate storage tank, to the high level trip. They will then reset themselves and continue to supply water to the vessel inventory as necessary.

If for some reason the ARI is also not effective, the BWR/4 is still able to mitigate the event. Without ARI and feedwater flow at zero, the level of the water in the vessel will go down and pass Level 2, the level at which HPCI and RCIC are initiated, at 37 seconds. Twenty seconds later, water from these systems will begin to enter the reactor vessel.

With confirmation from the flux monitoring system and the rod position indicating system that scram has not taken place, the SLCS will be activated.

Automatic SLCS will be started 2 minutes after the event begins. There will be one minute of transport time in the lines and the vessel. So nuclear shutdown begins at 3 minutes into the event if it is necessary to use the SLCS.

Using both of the SLCS pumps in most BWR/4 plants, a volumetric flow of 86 gpm is available. With this flow rate of sodium pentaborate, the reactor will be brought to hot shutdown in approximately 18 minutes from the beginning of the event. This can be seen in the lower left hand graph of Figure 4.1.2. The reaction of several other parameters is also depicted in Figure 4.1.2.

Even though the HPCI and RCIC are pumping water into the vessel from approximately 1 minute the water level within the vessel continues to decrease until approximately 5 minutes. At this time it reaches its lowest level and begins to rise. As the level is increasing, core flow is increased, thereby reducing the average void fraction. The various contributors to reactivity insertion and power production (boron, voids, etc.), must always be in balance with the power production. Water level is completely restored and the HPCI and RCIC are turned off at approximately 16 minutes. A large-scale plot of water level is shown on Figure 4.1.3.

Following hot shutdown, the decay power will continue to generate a small amount of steam which will continue to cycle the relief valves. At 27 minutes the suppression pool bulk temperature will reach its maximum value of 189°F. The maximum containment pressure with this much energy transferred to the suppression pool is 11.1 psig. Figure 4.1.4 provides plots of suppression pool bulk temperature and containment pressure. For the case with the ARI, the maximum suppression pool temperature is 141°F and it occurs approximately 5 hours after the event. Either way, all parameters are maintained within the limiting criteria for this event.

4.1.2 Turbine Trip Event

4.1.2.1 Overview of Response Without Scram

The overview which was given for the MSIV Closure Event, Section 4.1.1 in general also applies for the Turbine Trip event. A key difference is that, with this event, the main condenser is still available. This means that steam will be discharged to the suppression pool for only a short time at the beginning of the event and that from the time steam flow is within the bypass capacity, the main condenser will be used to remove the steam from the vessel.

4.1.2.2 Sequence of Events for Turbine Trip Transient

The Turbine Trip event begins with the rapid closure of the turbine stop valves and the resultant opening of the turbine bypass valves. The stop valves close in 0.1 second. The pressure immediately begins to rise which results in a reduction in void fraction and rapid increase in power. The sequence of events is shown in Table 4.1.2. For the BWR/4, this power reaches a maximum of 420% of the initial value at 0.93 second into the event and rapidly decreases again. At approximately 1.5 seconds, the set point pressure of the relief valves is reached and they begin to lift and arrest the pressure rise. Shortly after 2 seconds, it is expected that some of the fuel will see heat flux conditions such that it will experience transition boiling. At about the same time, the vessel dome pressure reaches 1150 psig, the maximum RPT point, and both of the recirculation pumps trip.

At the same time that RPT occurs the logic is activated to start ARI and if necessary to make the decision that an ATWS may have occurred and provide appropriate mitigating action.

Pressure will continue to rise briefly until at 3 seconds it has passed its peak and begins to decline. The maximum pressure in the vessel is 1195 psig at 2.7 seconds. Although the feedwater pumps remain available for the turbine trip case, in order to reduce the amount of power produced, the feedwater flow will be limited with this design and has been chosen to be zero. This minimizes power generation. The relief valves begin to close very early in this transient (about 9 seconds) and are all closed for the last time in less than 4 minutes, even without ARI, and the remainder of the generated steam went through the bypass to the main condenser. The first portion of this transient is shown in Figure 4.1.5. Peak values of other key variables in the system are given in Table 4.1.4.

The same pressure signal (1150 psig) that initiated the recirculation pump trip will cause the opening of valves on the scram air header which will allow the air pressure in the header to bleed down. In the improbable event that scram has not already occurred from any of the several available signals, this reduced pressure will allow the scram discharge valves to open and the control rods to insert. For this condition, tests have shown that the pressure in the header will have been reduced sufficiently in 15 seconds to allow the control rods to insert. They will all be expected to be fully into the core in 5 additional seconds. This condition is known as Alternate Rod Insertion (ARI). It completely mitigates the ATWS situation and 25 seconds after the event begins, it is over. Since in this event feedwater is not lost and the runback of feedwater

Table 4.1.2
BWR/4 MARK I TURBINE TRIP WITHOUT ARI

Sequence of Events

1.	Turbine Trips - Assumes Scram Fails	0
2.	Pressure Rise Begins	0
3.	Relief Valves Lift	2 Seconds
4.	Some Fuel Experiences Transition Boiling	2 Seconds
5.	Recirculation Pumps Trip on High Pressure, ARI is Initiated, and Timed SLCS Logic is Triggered	2 Seconds
6.	Vessel Pressure Peaks	3 Seconds
7.	Assumes ARI Fails	30 Seconds
8.	Feedwater Flow Runs Back to Zero	45 Seconds
9.	HPCI and RCIC Flow Starts on Level 2 Initiation	85 Seconds
10.	ATWS Logic Timer Complete, SLCS Starts	2 Minutes
11.	Liquid Boron Flow Reaches Core	3 Minutes
12.	Containment Bulk Temperature and Pressure Peak when Relief Valves all Close	4 Minutes
13.	Water Level Reaches Minimum and Begins to Rise	5 Minutes
14.	RHR Flow Begins (Pool Cooling)	11 Minutes
15.	Hot Shutdown is Achieved	21 Minutes

does not occur until the failure of both normal scram and ARI are confirmed, the feedwater system will continue to function and provide water to the reactor.

If for some reason the ARI is not effective, the BWR/4 is still able to mitigate the event. Without ARI and feedwater flow now having gone to zero, the level of the water in the vessel will go down and pass Level 2, the level at which HPCI and RCIC are initiated, at 65 seconds. Twenty seconds later water from these systems will begin to enter the reactor vessel.

With confirmation from the flux monitoring system and the rod position indicating system that scram has not taken place, the SLCS will now be activated. This system will be started 2 minutes after the event begins and will reach the core after an additional 1 minute of transport time in the lines and the vessel. Thus nuclear shutdown begins at 3 minutes into the event if it is necessary. For most BWR/4 plants, a volumetric flow of 86 gpm is available. With this flow rate of sodium pentaborate, the reactor will be brought to hot shutdown approximately 21 minutes from the beginning of the event. This can be seen in the lower left hand graph of Figure 4.1.6. The behavior of several other parameters is also depicted in Figure 4.1.6.

In a few events, oscillations of neutron flux, pressure, core flow, and steam flow are calculated with REDY as shown in Figure 4.1.6. These result from a combination of inherent thermal, hydraulic, and nuclear characteristics in the core and recirculation system and interactions with the relief valves and/or water level pressure controls. In this event, the oscillations in neutron flux have a characteristic expected of a limit cycle for a short period of time with an average power level of about 25% and peak generated power near 80%. Fuel thermal power swings are less than 10%, no fuel damage is expected and coolable geometry is constantly maintained for all fuel. The limit cycle is stopped as boron becomes more effective near 600 seconds and the event proceeds to cold shutdown. From calculations of this type it has been determined that stability will not be a problem during the ATWS event.

Even though the HPCI and RCIC are providing water into the vessel from approximately 1 minute, the water level within the vessel continues to decrease until approximately 5 minutes. At this time it reaches its lowest level and begins to rise. As the level is increasing, core flow is increased, thereby reducing the average void fraction. The various contributors to reactivity insertion and power production (boron, voids, etc.), must always be in balance with the power production. Water level is completely restored and the HPCI and RCIC are turned off at approximately 19 minutes. A large-scale plot of water level is shown in Figure 4.1.7.

Following hot shutdown, the decay power will continue to generate a small amount of steam which will go through the bypass to the main condenser. Since the major portion of the steam generated in this event goes to the main condenser, the temperature rise in the suppression pool will be minimal. The maximum suppression pool temperature calculated in this case is 102°F which results in a maximum containment pressure of 0.6 psig.

4.1.3 Inadvertent Open Relief Valve (IORV) Event

4.1.3.1 Overview of Response Without Scram

A detailed description of the sequence for this event is given below as it has been simulated in this analysis. This event has no rapid excursions as the previous two events but is merely a long term depressurization. The recirculation pump trip feature does not occur until late in the event after hot shutdown is achieved.

4.1.3.2 Sequence of Events for Inadvertent Open Relief Valve

This event begins when one of the primary relief valves on the main steam lines volunteers to open without influence from any other portion of the system. All pressure levels in the reactor coolant pressure boundary are at a nominal value prior to the event. This sequence of events is shown in Table 4.1.3.

At the time that the relief valve opens, there is a momentary depressurization (a few seconds) until the turbine pressure control valve senses it and closes slightly to control the pressure. After slightly less than two minutes, the suppression pool temperature, which was initially at 90°F, has risen to the alarm point of 95°F. The operator at this point will turn on the RHR system in the pool cooling mode to maintain low suppression pool temperature. The temperature will continue to rise and at 7.5 minutes will reach 110°F at which point the operator is required to manually scram the plant. Manual scram and ARI are activated and the logic chain will make the decision that an ATWS has occurred, and provide appropriate mitigating action.

If for some reason neither normal manual scram nor the ARI are effective, the BWR/4 is still able to mitigate the event. The ATWS logic will have determined that the control rods are not inserting and at 9 minutes into the event will activate the standby liquid control system. For this case, because the recirculation pumps have not been tripped, the delay time inside of the vessel is small and 0.5 minute of transport time is sufficient. At 9.5 minutes into the event the standby liquid control pumps start and at 10 minutes the control liquid reaches the core and shutdown begins. Within 24 minutes the power has been reduced to the point that the amount of steam generated is less than the relief valve capability flow to the turbine stops and the pressure now begins to fall more rapidly. The turbine control valves have closed completely. These events are depicted in Figure 4.1.8. By 28 minutes, the pressure will have dropped to the low pressure isolation point of 850 psig and the main steam isolation valves will close. For plants with turbine-driven feedwater pumps, the feedwater was assumed to be lost 20 seconds later. This causes the water level in the vessel to decrease, and at 33 minutes the low level point was reached where the recirculation pumps are automatically tripped and the high pressure coolant injection and reactor core isolation cooling systems were activated. These systems will continue to cycle on at low level and off at high level to maintain water inventory in the vessel. The depressurization of the vessel will continue with the relief valve discharging into the suppression pool, and the maximum pool temperature of 184°F will occur at 87 minutes. The suppression pool temperature trace is shown in Figure 4.1.9. The values given here are representative values. Each specific plant size has some features which may alter the results of this event.

In the case where the ARI will have the plant shut down at 8 minutes, the maximum pool temperature would be 163°F.

4.1.4 Sensitivity Results

Sensitivity studies have been done to determine the manner in which peak suppression pool temperature will vary given variations in system performance or initial conditions.

Table 4.1.3

BWR/4 MARK I INADVERTENT OPENING OF A RELIEF VALVE WITHOUT ARI

Sequence of Events

1.	Relief Valves Opens Inadvertently and Fails to Close	0
2.	Alarm Sounds at 95°F and Operator Initiates Pool Cooling	2 Minutes
3.	Suppression Pool Bulk Temperature Reaches 110°F, Operator Attempts Manual Scram, ARI and Timed SLC Logic Initiated, Assumes Scram Fails	7.5 Minutes
4.	Assumes ARI Fails	8 Minutes
5.	Standby Liquid Control System Automatically Starts	9.5 Minutes
6.	Liquid Boron Reaches Core	10 Minutes
7.	Power is less Than Relief Valve Capacity	24 Minutes
8.	Isolation on Low Line Pressure	28 Minutes
9.	Hot Shutdown is Achieved	~30 Minutes
10.	Peak Suppression Pool Bulk Temperature and Pressure are Reached	87 Minutes

Table 4.1.4
SUMMARY BWR/4 MARK I

Without ARI:

	66* gpm Boron, 95% Mixing Eff., Two Minute Timer		
	MSIV Closure	Turbine Trip	IORV
Maximum Neutron Flux (%)	572 at 3.96 sec	420 at 0.93 sec	100 at 0
Maximum Core Pressure (psig)	1282 at 9.15 sec	1167 at 2.54 sec	1015 at 0
Maximum Vessel Bottom Pressure (psig)	1297 at 8.93 sec	1195 at 2.70 sec	1044 at 0
Maximum Steamline Pressure (psig)	1269 at 9.24 sec	1144 at 2.33 sec	995 at 0
Maximum Average Heat Flux (%)	144 at 5.11 sec	134 at 2.71 sec	100 at 0
Maximum Suppression Pool Bulk Temperature (°F)	189 at 27 min	102 at 4 min	184 at 87 min
Containment Pressure (psig)	11.1	0.6	10.1

With ARI:**

Maximum Suppression Pool Bulk Temperature (°F)	141 at 5 hours
Containment Pressure, psig	4.3

*See Note 3, Table 3.4.3.

**With ARI all events occurring prior to 30 seconds remain unchanged.

The sensitivity shown in Table 4.1.5 shows the effect of plant size and boron pumping rate on peak suppression pool temperature. The 112 gpm boron pumping capacity in a 251 inch diameter vessel is equivalent to an 86 gpm system pumping into a 218 inch diameter vessel. Since the goal of these analyses was to determine the maximum suppression pool temperature of a plant with a 218 inch diameter vessel, the HPCI and RCIC flow rates, as a percent of rated feedwater flow, were used. From this table, it may be concluded that a plant with a 218 inch diameter vessel would typically experience a maximum suppression pool temperature of 185°F.

Table 4.1.6 shows variation in peak suppression pool temperature as a function of HPCI and RCIC flow rate expressed as a percentage of rated feedwater flow. The standby liquid control system is an 86 gpm system pumping into a 251 inch diameter vessel. On the large BWR/4 the HPCI plus RCIC flow is 19.8% of rated feedwater. This is exactly equivalent to the base case (BWR4-218) reported in Section 4.1.1 which results in a maximum suppression pool temperature of 189°F.

The 197°F peak suppression pool bulk temperature shown in Table 4.1.7 represents a hypothetical plant with a 251 inch diameter vessel but with an HPCI plus RCIC only large enough to pump this same percentage (22.9%) of rated feedwater as a plant with a 218 inch diameter vessel. For that hypothetical 251 inch diameter plant, the effect of reducing the boron system delay times is shown. In Table 4.1.7, the speed with which the boron system gets to the core is varied from 1 to 3 minutes.

The previous sensitivity studies have been for the MSIV Closure case. One sensitivity case was run for the IORV. In Section 4.1.3, it was reported that with ARI, the maximum suppression pool bulk temperature was 163°F. If the ARI were delayed until 10 minutes after the suppression pool bulk temperature reached 110°F, the maximum temperature would be 176°F.

Table 4.1.5

BWR/4 MARK I VESSEL MSIV CLOSURE

HPCI & RCIC CAPACITY IS 22.9% OF RATED FEEDWATER FLOW

	SLCS Capacity, Two Minute Timer, 95% Mixing Efficiency		
218 in. Vessel	66 gpm	77 gpm*	86 gpm**
251 in. Vessel	86 gpm	100 gpm	112 gpm
Maximum Bulk Suppression Pool Temperature (°F)	197 at 20 min	189 at 27 min	185 at 32 min

*Note that this sensitivity case coincidentally gives the same result as shown in Table 4.1.4. It is not, however, the reference case.

**Note that this is the typical expected result for BWR4-218, but is not the reference case. The reference case given in Table 4.1.4 is the BWR4-218, 66 gpm Boron, with 19.8% HPCI and RCIC capacity to represent all plant sizes.

Table 4.1.6

BWR/4 MARK I 251-INCH VESSEL MSIV CLOSURE

HPCI and RCIC Capacity (% NBR Feedwater)

	86 gpm Boron, 95% Mixing Efficiency 2-Minute and 1-Minute Delay		
	17.7	19.8*	22.9
Maximum Bulk Suppression Pool Temperature (°F)	184 at 32 min	189 at 27 min**	197 at 20 min

*Typical HPCI + RCIC capacity for 251-in. Vessel (5600 gpm).

**This result is equivalent to the reference case (BWR4-218 - 66 gpm Boron).

Table 4.1.7
SENSITIVITY STUDY
BWR/4 MARK I 251 INCH SIZE MSIV CLOSURE
HPCI + RCIC CAPACITY IS 22.9% OF RATED FEEDWATER*

	86 gpm Boron at 95% Mixing Eff. Total Delay Time (Minutes)		
	1	2	3*
Maximum Bulk Suppression Pool Temperature (°F)	186 at 30 min	192 at 25 min	197 at 20 min*

*These results are for generic HPCI + RCIC capacity using a 218-in. vessel. For a 251-in. vessel, the peak pool bulk temperature is 189°F (with the 3-minute delay - 2 minutes for logic and 1 minute transport time).

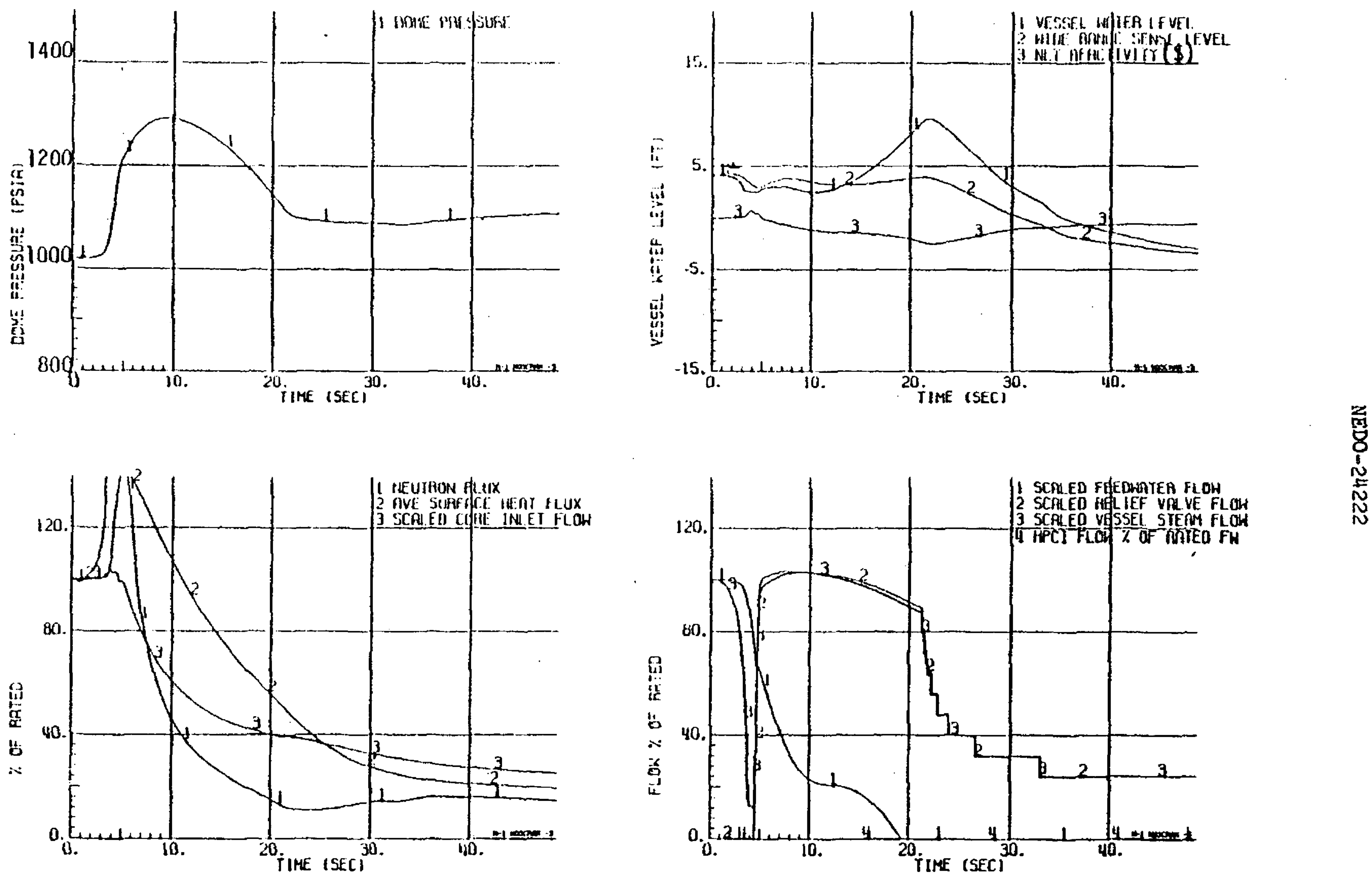

Figure 4.1.1. BWR/4 MSIV Closure, 2 min, Without ARI

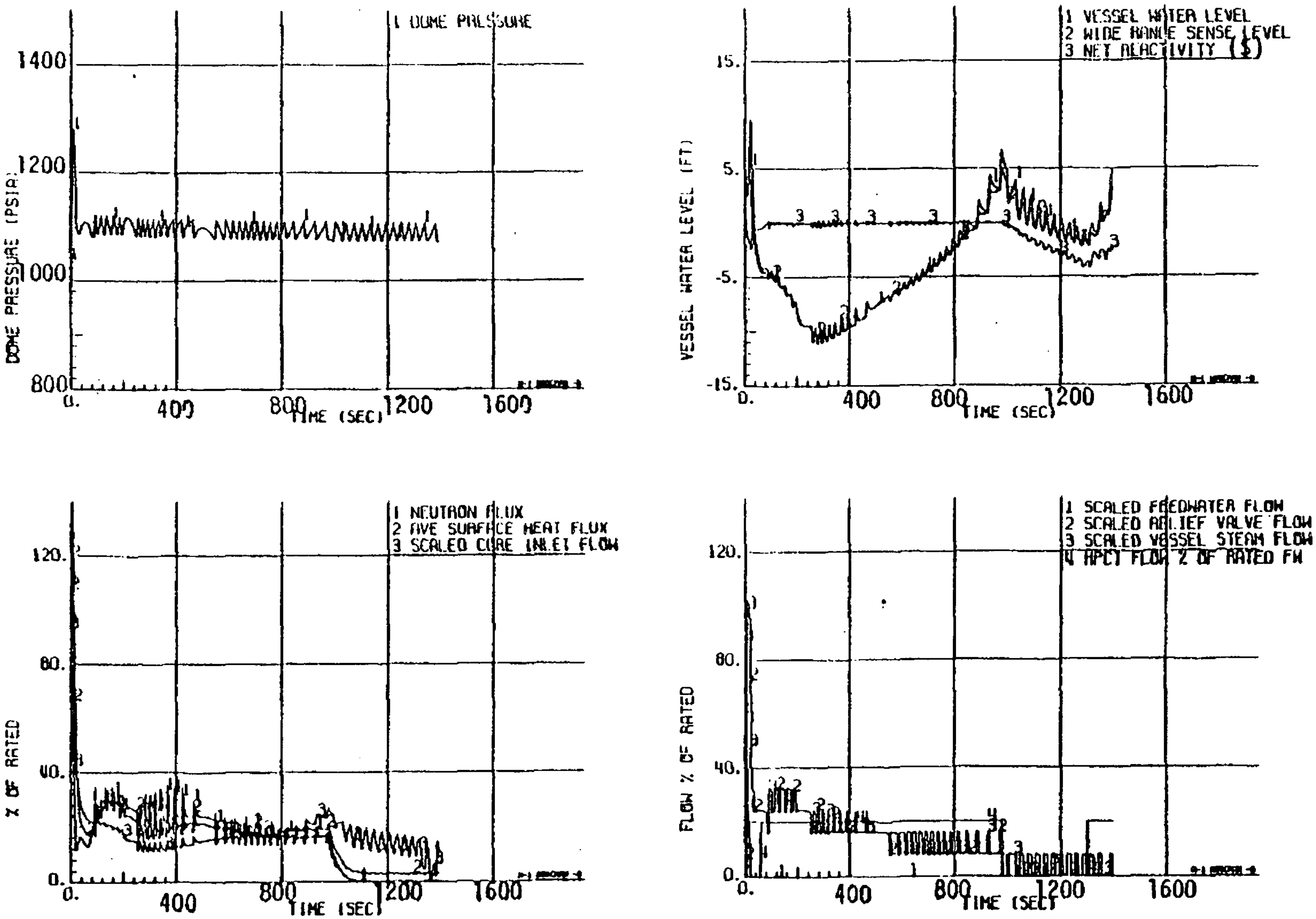

Figure 4.1.2. BWR/4 MSIV Closure, Without ARI, 2 min

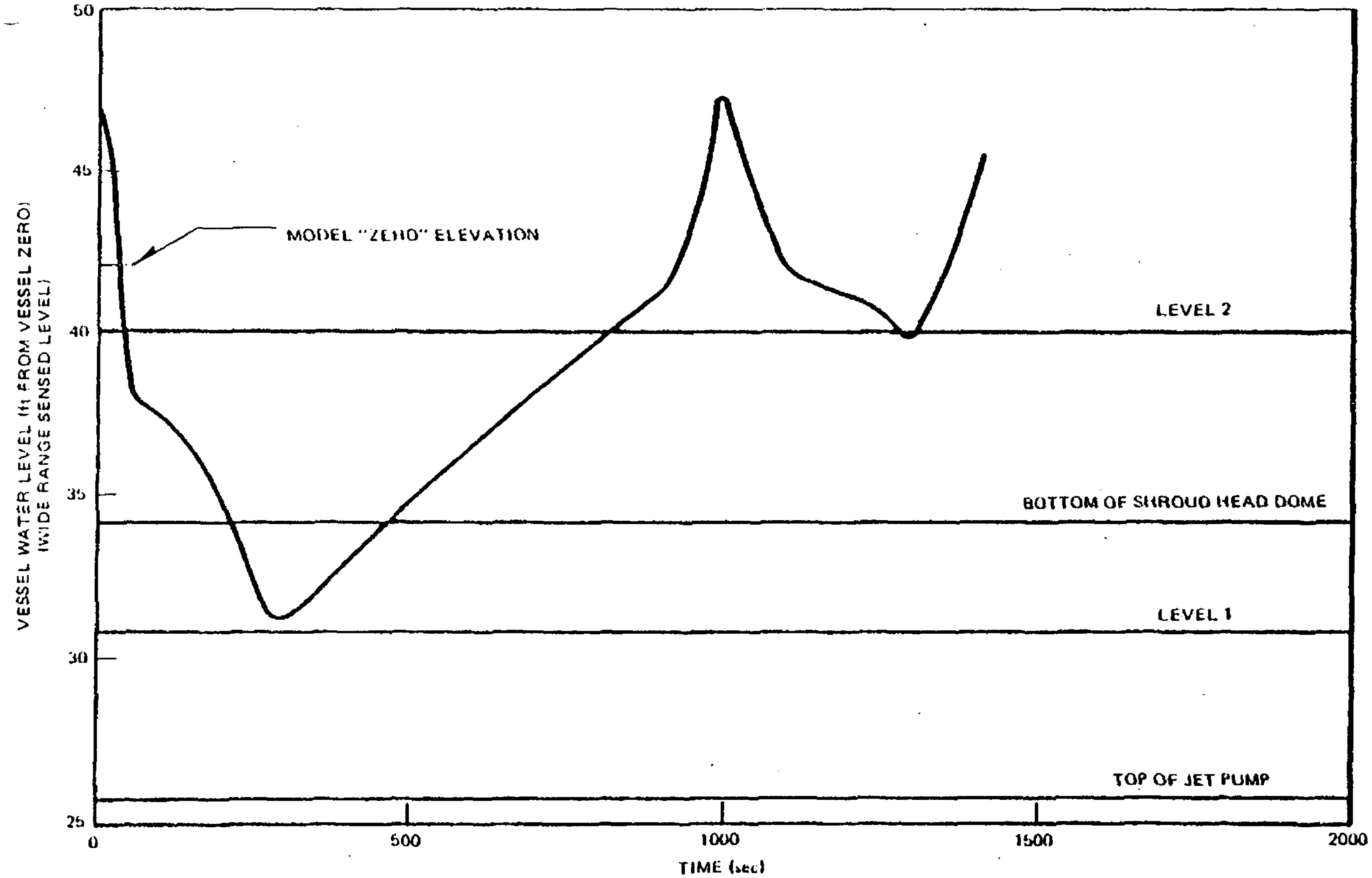

Figure 4.1.3. BWR/4 MSIV Closure with 2 min Boron Injection without ARI

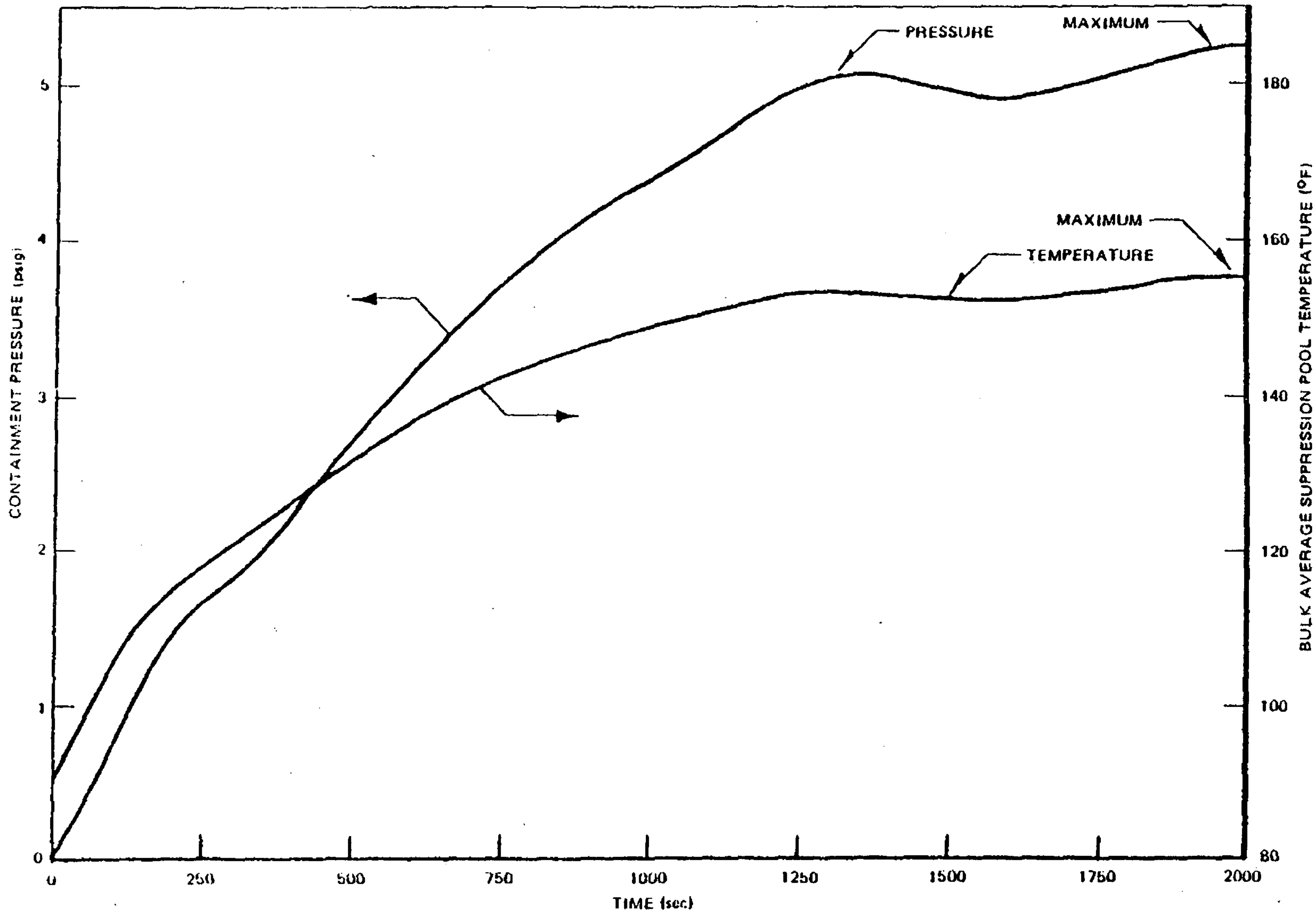

Figure 4.1.4. BWR/4 MSIV Closure with 2 min Boron Injection without ARI

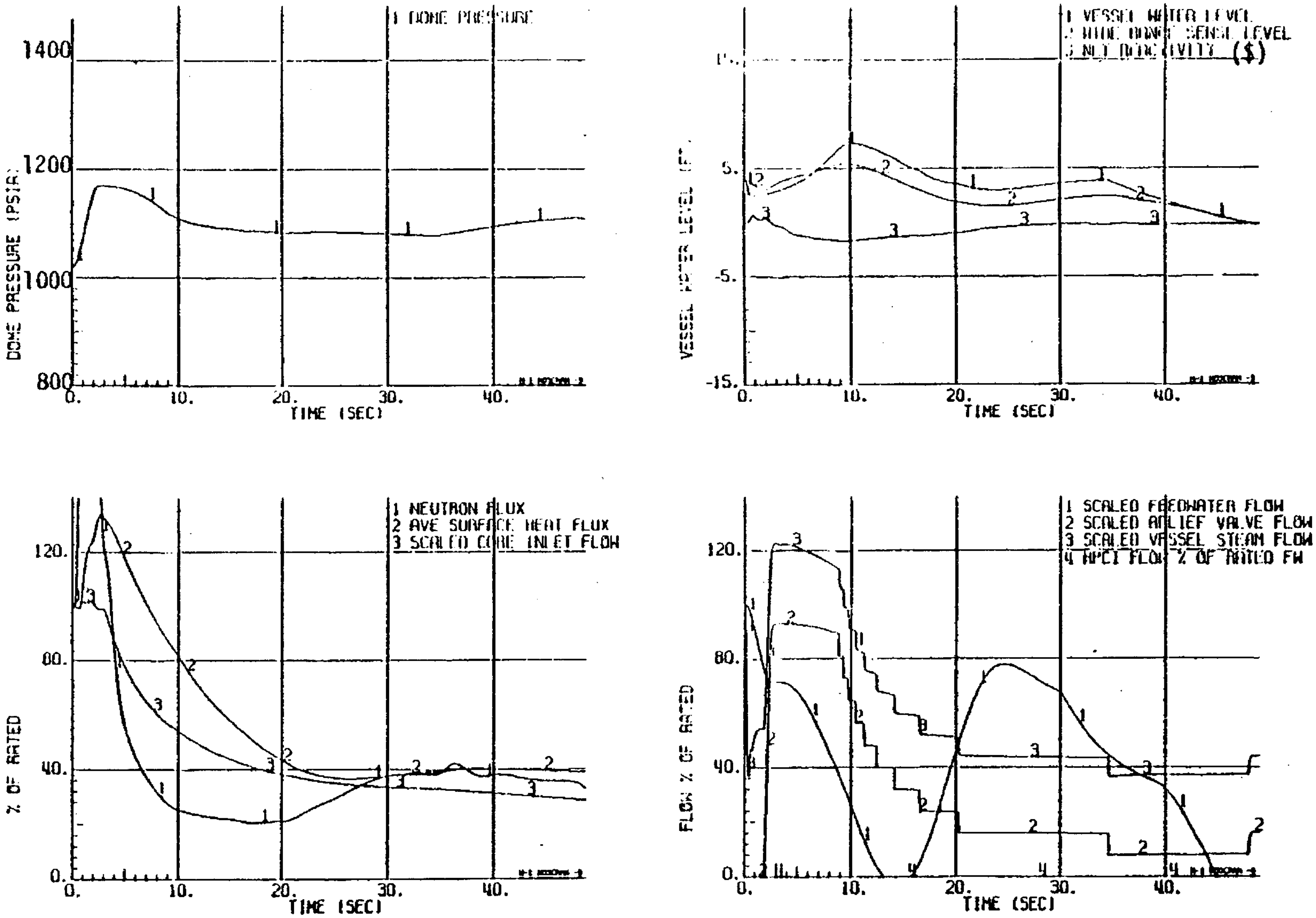

Figure 4.1.5. BWR/4 Turbine Trip without ARI, 2 min

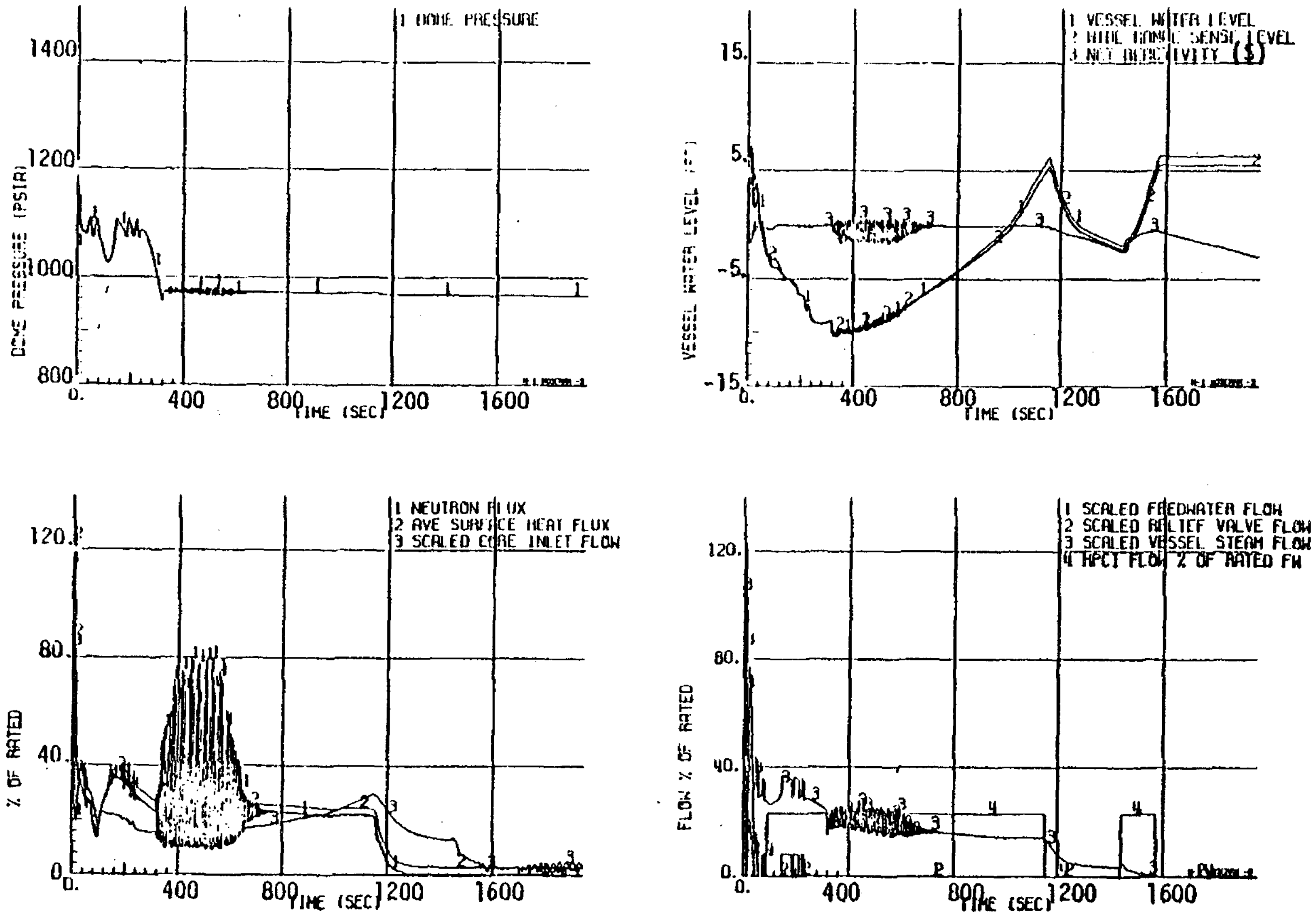

Figure 4.1.6. BWR/4 Turbine Trip without ARI, 2 min

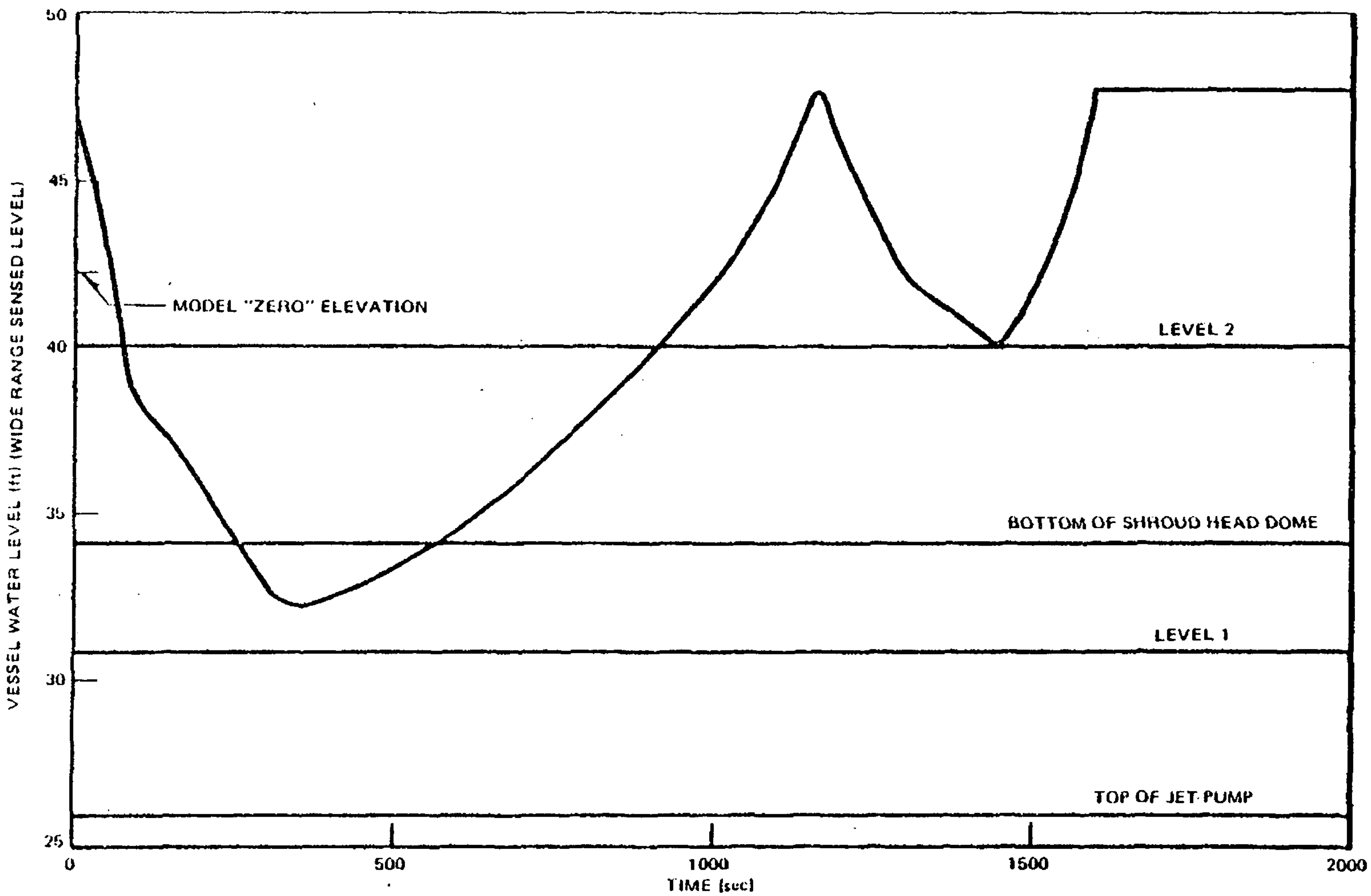

Figure 4.1.7. BWR/4 Turbine Trip with 2 min Boron Injection without ARI

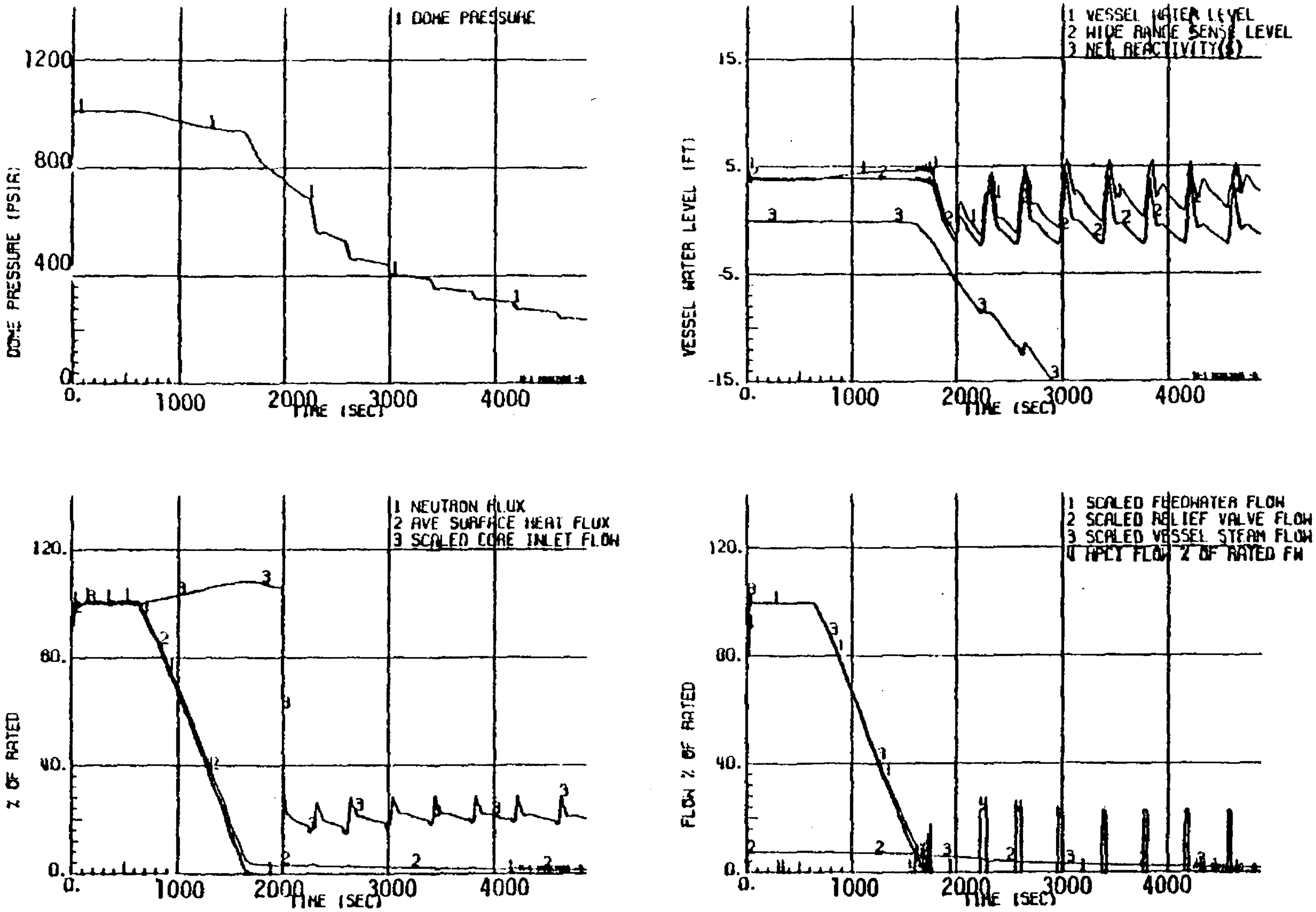

Figure 4.1.8. BWR/4 IORV - ATWS, 2 min After Manual Scram

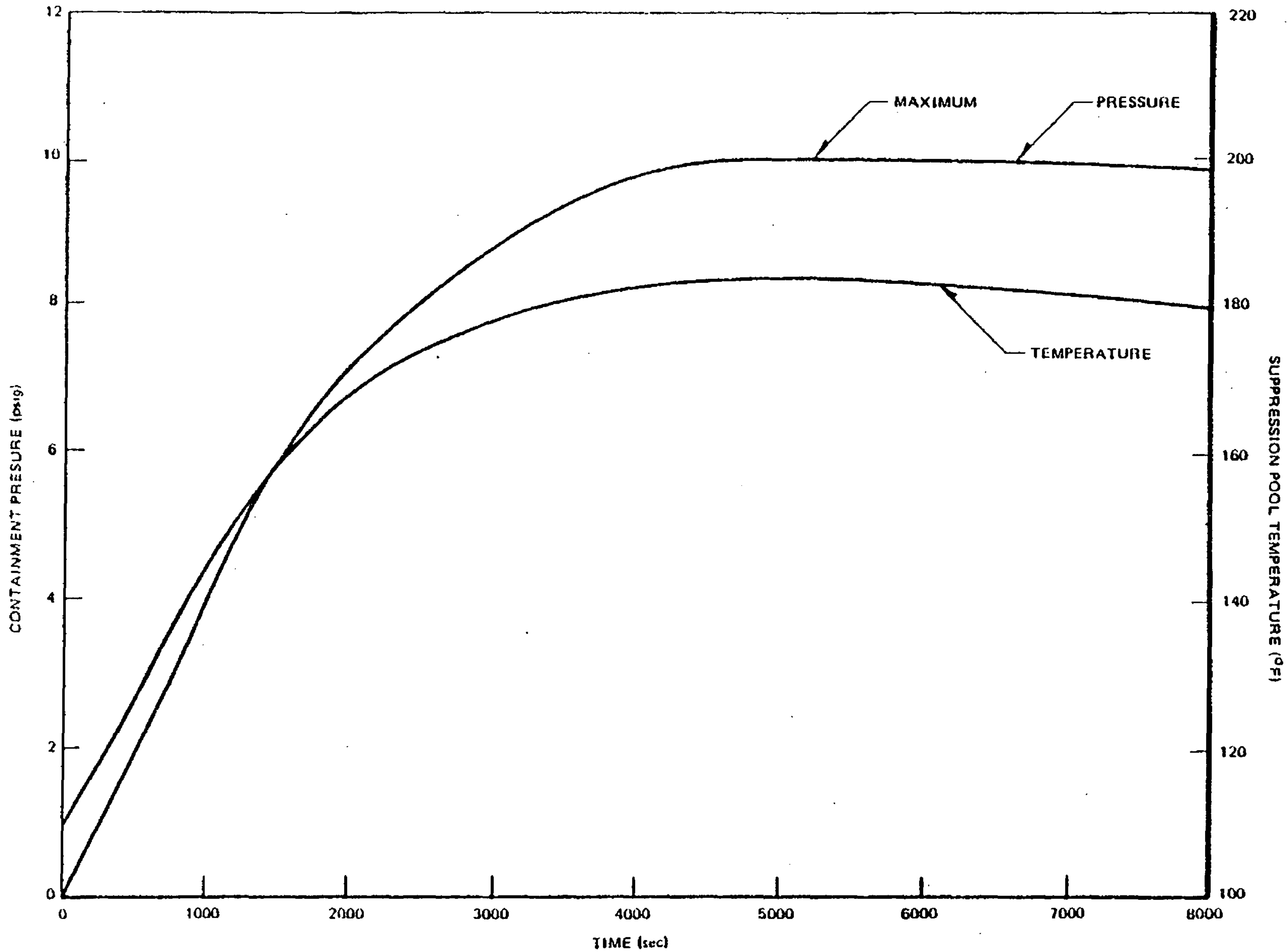

Figure 4.1.9. BWR/4 IORV with 2 min Boron Injection without ARI

4.2 RESULTS OF ATWS EVENTS - BWR 5/MARK II

4.2.1 MSIV Closure Event

4.2.1.1 Overview of Response Without Scram

A detailed description of the sequence of events for this transient is given below. The behavior of the plant is basically separable into an early or short term transient involving a sharp pressure rise and power peak, and a longer term portion that requires evaluation of coolant and containment conditions as the reactor is ultimately brought to shutdown.

The effectiveness of the recirculation pump trip (RPT) feature presented in NEDO-10349 and NEDO-20626 are reconfirmed by this analysis. It assists the relief valves in limiting the pressure disturbance acceptably and allows the establishment of a relatively low power generation rate for the long term portion of the transient. Figure 4.2.1 illustrates this first point.

Ultimate solution to the lack of scram situation must involve insertion of negative reactivity into the reactor, thereby bringing the reactor to a fully shutdown condition terminating the long term aspects of the event. The ARI feature is provided as an effective way to mitigate common cause failures in the logic of the scram system. In the very remote case of ARI ineffectiveness, the automated SLCS provides further protection and shutdown capability. Coolant inventory is adequately maintained by using the HPCS system and RCIC system available on each BWR/5 to replace the coolant loss as steam flow leaves the primary system through the relief valves. Simply adding more water without inserting negative reactivity has the effect of raising the power generation rate and the amount of inventory leaving the system as steam. The steam reaching the suppression pool continues to heat it and pressurize the containment until the power generation/steam flow can be reduced and finally terminated. The RHR system ultimately cools the pool and eventually the reactor also (in the shutdown cooling mode) if the MSIVs cannot be reopened establishing flow to the main condenser (the preferred method of cooldown).

4.2.1.2 Sequence of Events for MSIV Closure Transient

The MSIV closure transient provides the most severe conditions following a postulated failure to scram. Listed in Table 4.2.1 in sequence of occurrence are significant points of the transient with representative times when each event occurs.

The sequence of events begins with the nominal 4 second closure of the main steam isolation valves. With motion of the MISVs, the pressure begins to rise which results in a reduction in void fraction and a rapid increase in power. This sequence of events is shown in Table 4.2.1. For the BWR/5, this power (neutron flux) reaches a maximum of 613% of the initial value at 4 seconds into the event and rapidly decreases again. In just under 4 seconds, the set point pressure of the relief valves is reached and they begin to lift and arrest the pressure rise. Shortly after 4 seconds, the vessel dome pressure reaches 1150 psig, the maximum RPT point, and both of the recirculation pumps trip. A delay of 530 milliseconds is used from the time the 1150 psig is reached until the time that RPT is effected. This delay time (500 milliseconds delay in the sensor and 30 milliseconds in the logic and trip) is consistent with industry experience. At the same time that the RPT occurs, the logic chain is activated to start ARI and if necessary make the decision that an ATWS may have occurred and provide appropriate mitigating action.

Table 4.2.1

BWR/5 MARK II

MSIV CLOSURE - WITHOUT ARI

Sequence of Events

1.	Nominal-4 Second MSIV Closure - Assume Scram Fails	0
2.	Pressure Rise Begins	1 second
3.	Relief Valves Lift	4 seconds
4.	Recirculation Pumps Trip on High Pressure, ARI is initiated and timed SLCS logic is triggered	5 seconds
5.	Vessel Pressure Peaks	7 seconds
6.	Feedwater Flow Coasts Down to Lower Limit	20* to 45** seconds
7.	Assume ARI Fails	30 seconds
8.	HPCS and RCIC Flow Starts after Level 2 Initiation	1 minute
9.	ATWS Logic Timer Complete, SLCS Starts	2 minutes
10.	Liquid Boron Flow Reaches Core	3 minutes
11.	Water Level Reaches Minimum and Begins to Rise	3 minutes
12.	RHR Flow Begins (Pool Cooling)	11 minutes
13.	Hot Shutdown is Achieved	18 minutes
14.	Containment Bulk Temperature and Pressure Peak	28 minutes

*Feedwater turbine coastdown assumed after loss of steam.

**Motor-driven feed pumps.

Reactor pressure continues to rise for a short period of time until, at approximately 7 seconds into the event, it reaches its peak and begins to decline. The maximum pressure in the vessel is 1260 psig at 7 seconds. Those plants which have turbine-driven feed pumps will begin to coast down as soon as the MSIVs are closed and will have lost their ability to overcome vessel pressure head at 20 seconds(1). The relief valves begin to close shortly after 20 seconds and pressure is then stabilized at the relief valve set point pressure. This part of the transient is shown in Figure 4.2.1. Some peak pressures at other points in the system are given in Table 4.2.4.

The same pressure signal (1150 psig) that initiated the RPT will cause the opening of valves on the scram air header which will allow the air pressure in the header to blow down. In the improbable event that scram has not already occurred from any of the several available signals, this reduced pressure will allow the scram discharge valves to open and the control rods to insert. For this condition, tests have shown that the pressure in the header will have been reduced sufficiently in 15 seconds to allow the control rods to insert. They will all be expected to be fully inserted into the core in 4 additional seconds. ARI completely mitigates the ATWS situation and 25 seconds after the event begins, it is essentially over. Figure 4.2.2 shows the expected course of the event. Water level drifts downward as decay energy generates small amounts of steam, and when level 2 is reached, the HPCS and RCIC automatically start and replenish the vessel water inventory, from the condensate storage tank. When water level is restored to the high level trip, they will then reset themselves and continue to supply water to the vessel inventory as necessary.

If for some reason the ARI is not effective, the BWR/5 is still able to mitigate the event. With the ARI failed and feedwater flow at zero, the level of the water in the vessel will go down and pass level 2, the level at which HPCS and RCIC are initiated, at 46 seconds. Fifteen seconds later, water from these systems will begin to enter the reactor vessel.

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(1)For this analysis, it was assumed that a motor driven feedwater (FW) system was available and the FW shutoff occurred near 45 seconds as the ATWS FW limiter was activated.

With confirmation from the flux monitoring system and the rod position indicating system that scram has not taken place, the SLCS will now be automatically activated. This system will be started 2 minutes after the event begins and there will be 1 minute of transport time in the lines and the vessel. Therefore, nuclear shutdown begins at 3 minutes into the event if it is necessary to use the SLCS. Using both of the SLCS pumps in most BWR/5 plants, a volumetric flow of 86 gpm is available. With this flow rate of sodium pentaborate, the reactor will be brought to hot shutdown in approximately 18 minutes from the beginning of the event. This can be seen in the lower left hand graph of the long-term plot of this event, Figure 4.2.3. The values of several other parameters are also depicted in Figure 4.2.3.

Even though the HPCS and RCIC are pumping water into the vessel after approximately 1 minute, the water level within the vessel continues to decrease until approximately 3 minutes. At this time it reaches its lowest level and begins to rise. As the level is increased, core flow is increased, thereby reducing the average void fraction. The various contributors to reactivity insertion and power production (boron, voids, etc.) must always be in balance with the power production. Water level is completely restored and the HPCS and RCIC are turned off at approximately 17 minutes. A larger-scale plot of water level is shown in Figure 4.2.4.

Following hot shutdown, the decay power will continue to generate a small amount of steam which will continue to cycle the relief valves. At 28 minutes the suppression pool temperature will reach its maximum value of 177°F. The maximum containment pressure with this much energy transferred to the suppression pool is 8.5 psig. For the case where the ARI functions as expected, the maximum suppression pool bulk temperature is 138°F and it occurs approximately 4 hours after the event. Figures 4.2.5 and 4.2.6 show long-term containment conditions for these cases. All parameters are maintained within the limiting criteria for this event.

4.2.2 Turbine Trip Event

4.2.2.1 Overview of Response Without Scram

The overview which was given above for the MSIV closure event is also essentially applicable for the turbine trip event. A key difference is that, with this event, the main condenser remains available. This means that steam will be discharged to the suppression pool for only a short time at the beginning of the event and that from the time that steam flow is within the bypass capacity, the main condenser will be used to remove the steam from the vessel.

4.2.2.2 Sequence of Turbine Trip Transient Event

The turbine trip event begins with the rapid closure of the turbine stop valves and the resultant opening of the turbine bypass valves. The stop valves close in 0.1 second. The pressure immediately begins to rise which results in a reduction in void fraction and rapid increase in power. The sequence of events is shown in Table 4.2.2. For the BWR/5, this power reaches a maximum of 380% of the initial value at one second into the event and rapidly decreases again. At approximately 1.5 seconds, the set point pressure of the relief valves is reached and they begin to lift and arrest the pressure rise. Shortly after 2 seconds, it is expected that some of the fuel will experience transition boiling. At about the same time, the vessel dome pressure reaches 1150 psig, the maximum RPT point, and both of the recirculation pumps trip. For this analysis, the earlier trip of recirculation pumps directly from the stop valve closure was conservatively neglected. It makes the early event results even milder. At the same time that RPT occurs the logic chain is activated to start ARI and if necessary to make the decision that an ATWS may have occurred and provide appropriate mitigating action.

Table 4.2.2

BWR/5 MARK II

TURBINE TRIP (TT) - WITHOUT ARI

Sequence of Events

1.	Turbine Trips - Assume Scram Fails	0
2.	Pressure Rise Begins	1 second
3.	Relief Valve Lift	2 seconds
4.	Some Fuel Experiences Transition Boiling	2 seconds
5.	Recirculation Pumps Trip on High Pressure, ARI is Initiated and Timed SLCS Logic is Triggered	3 seconds
6.	Vessel Pressure Peaks	3 seconds
7.	Assume ARI Fails	30 seconds
8.	Feedwater Flow Runs Back to Zero	45 seconds
9.	HPCI and RCIC Flow Starts on Level 2 Initiation	75 seconds
10.	ATWS Logic Timer Complete, SLCS Starts	2 minutes
11.	Containment Bulk Temperature and Pressure Peak	2 minutes
12.	Liquid Boron Flow Reaches Core	3 minutes
13.	Water Level Reaches Minimum and Begins to Rise	4 minutes
14.	RHR Flow Begins (Pool Cooling)	11 minutes
15.	Hot Shutdown is Achieved	18 minutes

Reactor pressure will continue to rise briefly until at 3 seconds it has passed its peak and begins to decline. The maximum pressure in the vessel is 1192 psig at 2 seconds. Though the feedwater pumps remain available for the turbine trip case, in order to reduce the amount of power produced, the feedwater flow will be terminated. This minimizes power generation and steam passed to the suppression pool. The relief valves begin to close very early in this transient (about 9 seconds) and are all closed for the last time in less than 2 minutes even without ARI, and the remainder of the generated steam will go through the bypass to the main condenser. The first portion of this transient with and without ARI is shown in Figures 4.2.7 and 4.2.8. Some peak pressures at other points in the system are given in Table 4.2.4.

The same pressure signal (1150 psig) that initiated the recirculation pump trip will cause the opening of valves on the scram air header which will allow the air pressure in the header to blow down. In the improbable event that scram has not already occurred from any of the several available signals, this reduced pressure will allow the scram discharge valves to open and the control rods to insert. For this condition, tests have shown that the pressure in the header will have been reduced sufficiently in 15 seconds to allow the control rods to insert. They will all be expected to be fully into the core in 5 additional seconds. ARI completely mitigates the ATWS situation and 25 seconds after the event begins, it is over. Since in this event feedwater is not lost and the runback of feedwater discussed earlier does not occur until the failure of both normal scram and ARI are confirmed, the feedwater system will continue to function and provide water to the reactor.

If for some reason the ARI is not effective, the BWR/5 is still able to mitigate the event. Without ARI and feedwater flow now having been limited to zero, the level of the water in the vessel will go down and pass level 2, the level at which HPCS and RCIC are initiated, at 55 seconds. Twenty seconds later, water from these systems begins to enter the reactor vessel (as shown in the figures).

With confirmation from the flux monitoring system and the rod position indicating system that scram has not taken place the SLCS will now be activated. This system will be started 2 minutes after the event begins and 1 minute of transport time is assumed for the lines and the vessel. Therefore, nuclear shutdown begins at 3 minutes into the event if it is necessary to use the SLCS. Using both of the SLCS pumps in most BWR/5 plants, a volumetric flow of 86 gpm is available. With this flow rate of sodium pentaborate, the reactor will be brought to hot shutdown approximately 18 minutes from the beginning of the event. The reaction of several parameters is depicted in Figure 4.2.9 for the long-term event.

Even though the HPCS and RCIC are providing water into the vessel from approximately one minute, the water level within the vessel continues to decrease until approximately 4 minutes. At this time it reaches its lowest level and begins to rise. As the level is increased, core flow is increased, thereby reducing the average void fraction. The various contributors to reactivity insertion and power production (boron, voids, etc.) will always be in balance with the power production. Water level is completely restored and the HPCS and RCIC are turned off at approximately 14 minutes. A larger-scale plot of water level is shown on Figure 4.2.10.

Following hot shutdown, the decay power will continue to generate a small amount of steam which will go through the bypass to the main condenser. Since the major portion of the steam generated in this event goes to the main condenser, the temperature rise in the suppression pool will be minimal. The maximum suppression pool temperature calculated in this case is 105°F at 2 minutes, which results in a maximum containment pressure of 0.8 psig.

4.2.3 Inadvertent Open Relief Valve Event

4.2.3.1 Overview of Response Without Scram

A detailed description of the sequence of events is given below. This event has no rapid excursion as the previous two events but is merely a long term depressurization. The recirculation pump trip feature does not occur until late in the event after hot shutdown is achieved.

4.2.3.2 Sequence of Events During Inadvertent Open Relief Valve (IORV) ATWS Transient

This event begins when one of the primary relief valves on the main steam lines inadvertently opens without influence from any other portion of the system. All pressure levels in the reactor coolant pressure boundary are at a nominal value prior to the event. The resulting sequence of events is shown in Table 4.2.3.

At the time that the relief valve opens, there is a momentary depressurization (a few seconds) until the turbine control valve senses it and closes slightly (dropping unit electrical output) to control pressure. Section 4.2.4 contains a valve sensitivity analysis. After about 2 minutes, the suppression pool temperature, which was initially assumed to be at 90°F, has risen to the alarm point of 95°F. If attempts to reclose the valve are unsuccessful, the operator at this point will turn on the RHR system in the pool cooling mode to maintain low pool temperature. If attempts to close the valve continue to be unsuccessful, the temperature will continue to rise and at 7.5 minutes will reach 110°F at which point the operator is required to manually scram the plant. For this example case, the manual scram also activates the ATWS protection paths to ARI and the logic chain which monitors if scram really occured, and if necessary will make the decision that further ATWS mitigating action is needed. The logic paths shown in Section 3.4 are utilized, although discussion below shows that the largest BWR/5 plants do not require the automatic logic path from manual scram to ATWS mitigation protection.

If for some reason neither normal manual scram nor the ARI are effective, the BWR/5 is still able to mitigate the event. The ATWS logic will have determined that the control rods are not inserted and at 9.5 minutes into the event will automatically activate the SLCS. For this case, because the recirculation flow is maintained a boron mixing efficiency of 95% is assumed, and the delay time inside of the vessel is small so that at 10 minutes the liquid boron reaches the core and shutdown begins. By 18 minutes the power has been reduced to the point that the amount of steam generated is less than the relief valve

Table 4.2.3

BWR/5 MARK II

INADVERTENT OPENING OF A RELIEF VALVE (IORV)

WITHOUT ARI, 8.3% NBR VALVE

Sequence of Events

1.	Relief Valve Opens Inadvertently and Attempts to Close it are Unsuccessful	0
2.	Alarm Sounds at 95°F and Operator Initiates Pool Cooling	2 minutes
3.	Suppression Pool Bulk Temperature Reaches 110°F Operator Attempts Manual Scram; Assume Scram Fails	7.5 minutes
4.	Assume ARI Fails	8 minutes
5.	Liquid Boron Control System Automatically Starts	9.5 minutes
6.	Control Liquid Reaches Core	10 minutes
7.	Power is Less Than Relief Valve Capacity	18 minutes
8.	Isolation on Low Steam Line Pressure (850 psi)	20 minutes
9.	Hot Shutdown is Achieved	30 minutes
10.	Peak Suppression Pool Bulk Temperature and Pressure are Reached	60 minutes

capability and the pressure now begins to fall more rapidly. The turbine control valves have closed completely. These events are depicted in Figure 4.2.11. By 20 minutes, the pressure has dropped to the low line pressure isolation point of 850 psig and the main steam isolation valves close. Simulating plants with turbine-driven feedwater pumps, the feedwater was assumed to be lost within 20 seconds of the isolation. This causes the water level in the vessel to decrease and at 23 minutes the low level point (L2) was reached where the recirculation pumps are automatically tripped and the HPCS and RCIC systems were activated. These systems are shown to automatically cycle on at low level (L2) and off at high level (L8) as specified to maintain water inventory in the vessel, although manual action is expected to maintain level with the RCIC alone. The depressurization of the vessel will continue with the relief valve discharging into the suppression pool. The maximum pool bulk temperature of 185°F will occur at about 1 hour. The peak containment pressure of 10.3 psig occurs at the same time, well below the 46 psig design pressure for the Mark II containments.

In the case where the ARI will have brought the plant to shutdown at 8 minutes, the maximum pool temperature would be about 166°F.

4.2.4 Sensitivity Results

Sensitivity studies have been done to determine the manner in which peak suppression pool temperature will vary, given variations in system performance or initial conditions.

For the MSIV closure event, the sensitivity to changing boron system delay times was studied. In Table 4.2.5 and Figure 4.2.12 time when the initial boron flow gets to the core is varied from 3 to 7 minutes. The 3 minute values represent the base case delay times.

Sensitivity cases were also run for the IORV transient. Variation of the relative size of the relief valve is shown in Figure 4.2.13. Base case assumptions for all other parameters were maintained.

The time of initiation of the standby liquid control injection was also studied for the IORV event as shown on Figure 4.2.14. This study used the relief valve capacity typical of the 251-inch vessel BWR/5 units. It shows that boron injection could be initiated more than 10 minutes after the manual scram was attempted. It is anticipated, therefore, that these largest plants will not include the automatic logic link from the reactor trip system manual scram to the ARI and automated SLCS initiation logic as shown in the general logic description in Section 3.4. Smaller plants will require added, size-dependent analyses before they can make the same simplification.

Table 4.2.4
SUMMARY BWR/5 MARK II

Without ARI	86 gpm Boron, Two Minute Timer 2-Min + 1-Min Delay		
	MSIV Closure	TT W/BP	IORV
Maximum Neutron Flux (%)	613 at 4.1 sec	380 at 1.0 sec	100 at 0
Maximum Core Pressure (psig)	1242 at 7.5 sec	1168 at 2.6 sec	1031 at 0
Maximum Vessel Bottom Pressure (psig)	1260 at 7.4 sec	1192 at 2.2 sec	1058 at 0
Maximum Steamline Pressure (psig)	1200 at 7.8 sec	1128 at 1.9 sec	986 at 0
Maximum Average Heat Flux (%)	150 at 5.0 sec	129 at 2.6 sec	100 at 0
Maximum Suppression Bulk Pool Temperature (°F)	177 at 28 min	105 at 2 min	185 at 1 hour
Containment Pressure (psig)	8.5	0.8	10.3

With ARI		
(All events occurring prior to 30 seconds remain unchanged)		
Maximum Suppression Pool Bulk Temperature (°F)	138 at 4 hours	98 at 2 min
Containment Pressure (psig)	3.2	0.4

Table 4.2.5
BWR/5 MARK II
MSIV CLOSURE WITHOUT ARI
SENSITIVITY STUDY

	86 gpm Boron at 75% Mixing Eff Total Delay Time (Minutes)		
	3*	5	7
Maximum Suppression Pool Bulk Temperature (°F)	177 at 28 min	188 at 35 min	194 at 38 min
Maximum Containment Pressure (psig)**	8.5	10.7	12.0

*Base case has 2 minute logic delay plus 1 minute transport time allowance.
**Mark II containment design pressure is 46 psig.

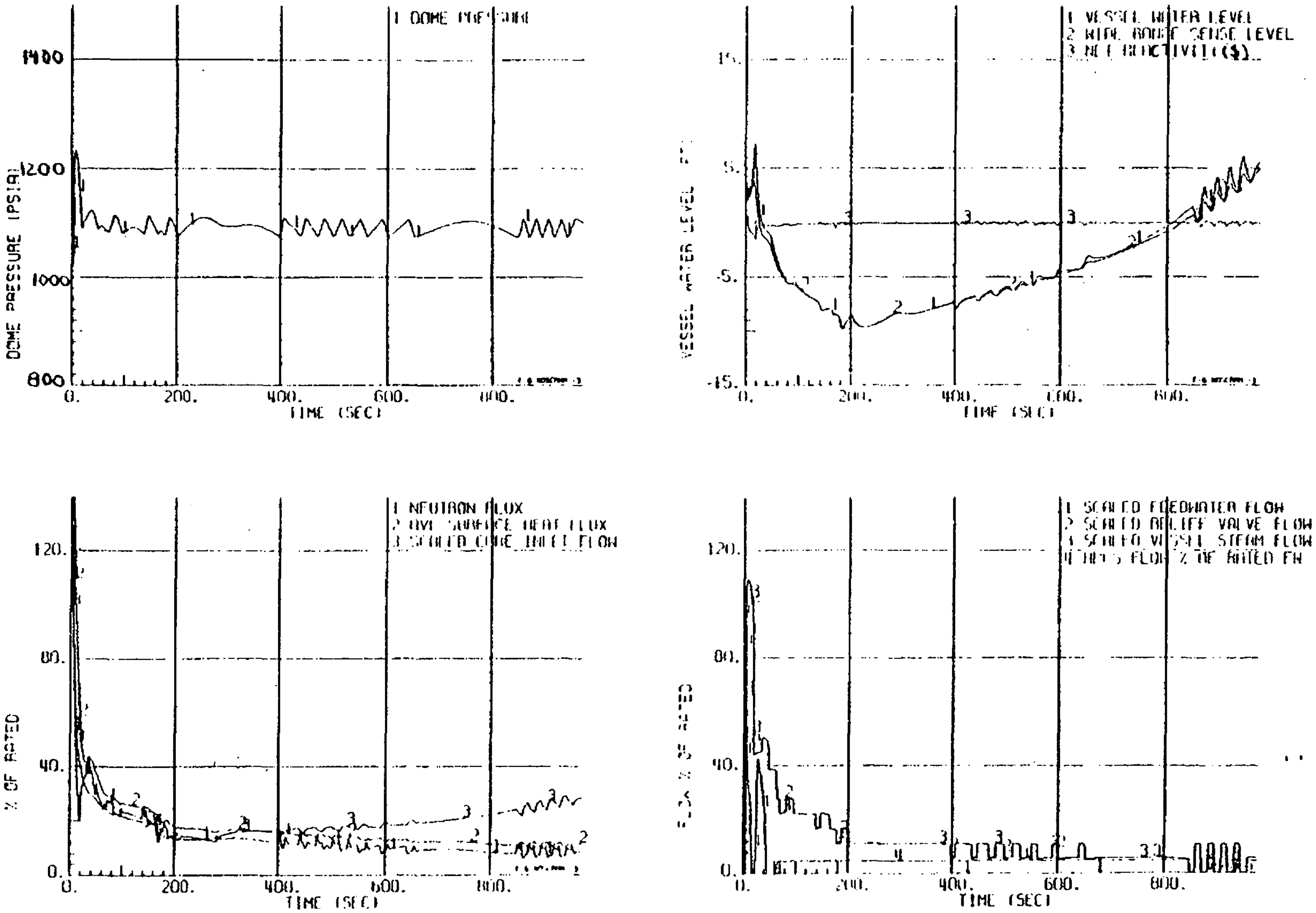

Figure 4.2.1. BWR/5 MSIV Closure, Without ARI, 75% Boron Mixing Efficiency, 2 min - 86 gpm Boron

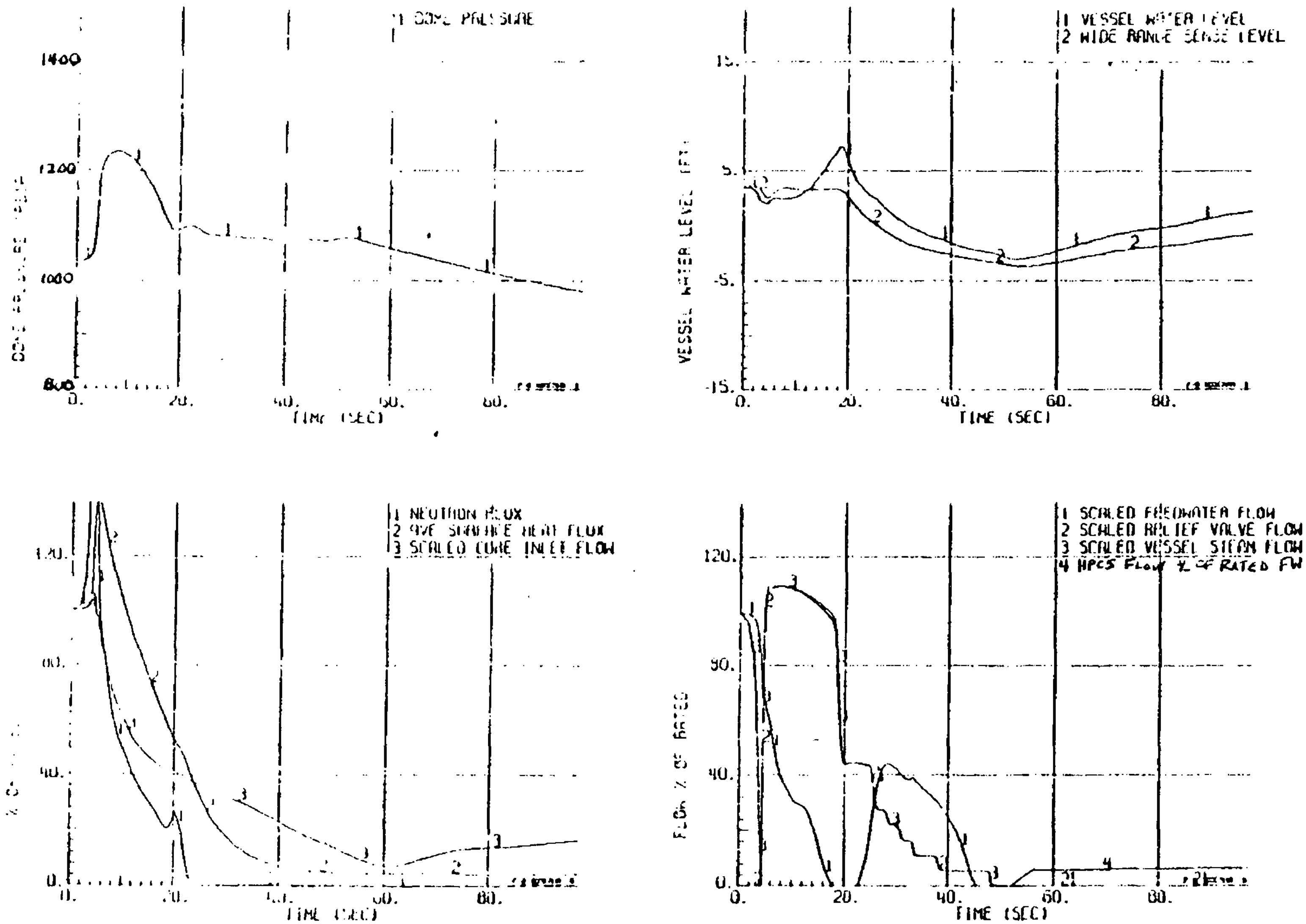

Figure 4.2.2. BWR/5 MSIV Closure, ARI at 20 sec

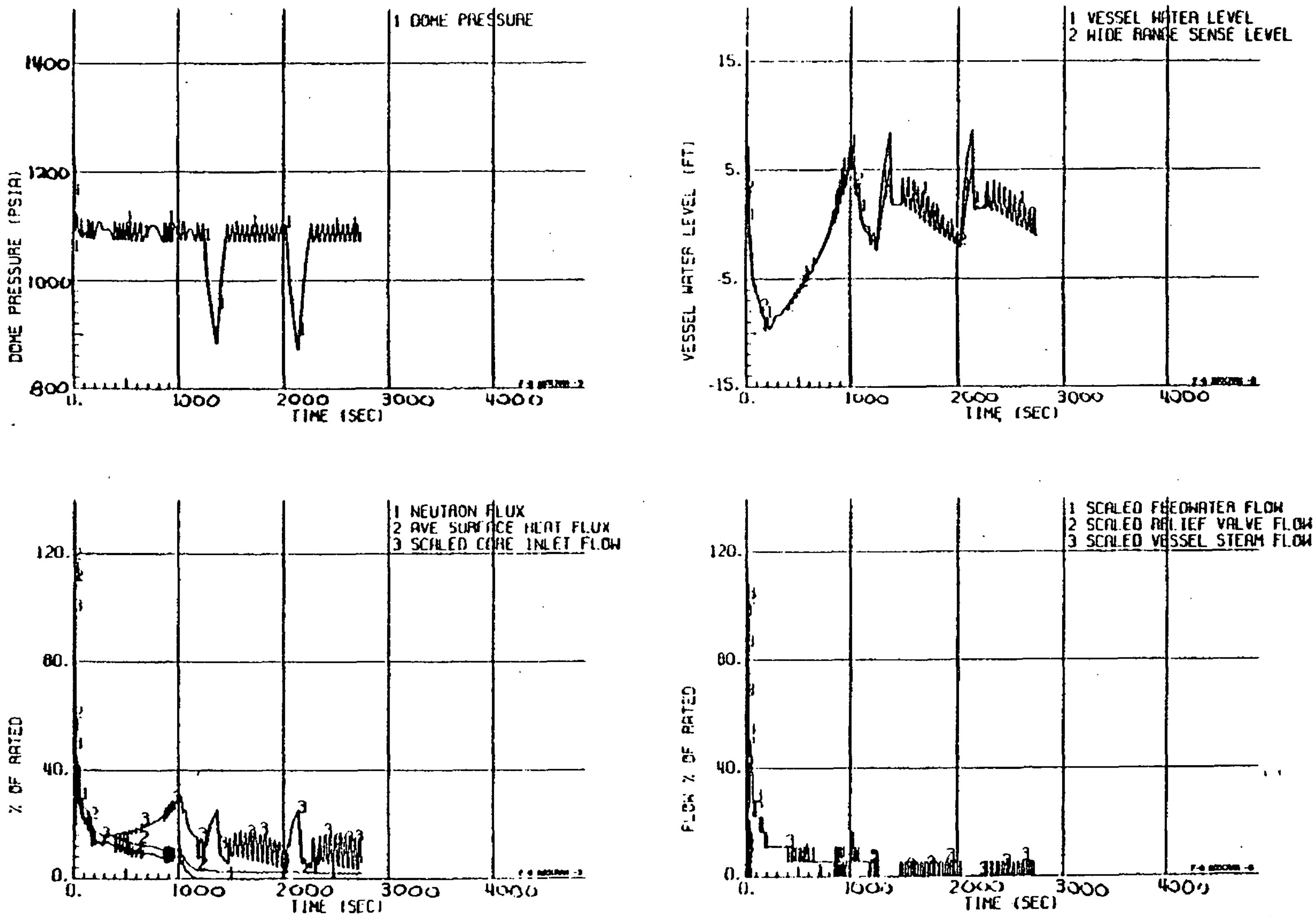

Figure 4.2.3. BWR/5 MSIV Closure, Without ARI, 2 min - 86 gpm Boron, 75% Mixing Efficiency

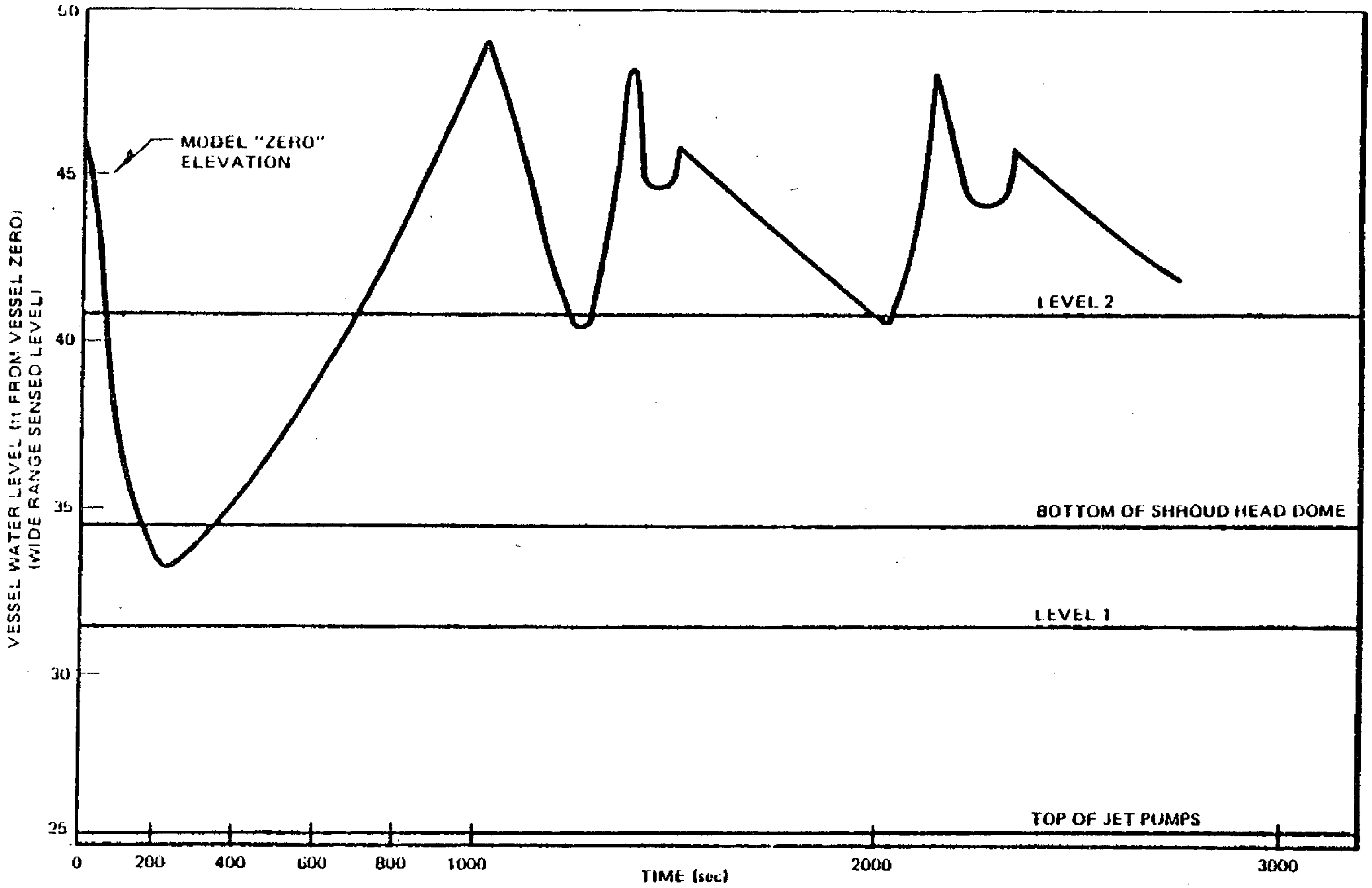

Figure 4.2.4. BWR/5 MSIV Closure with 2 min Boron Injection without ARI

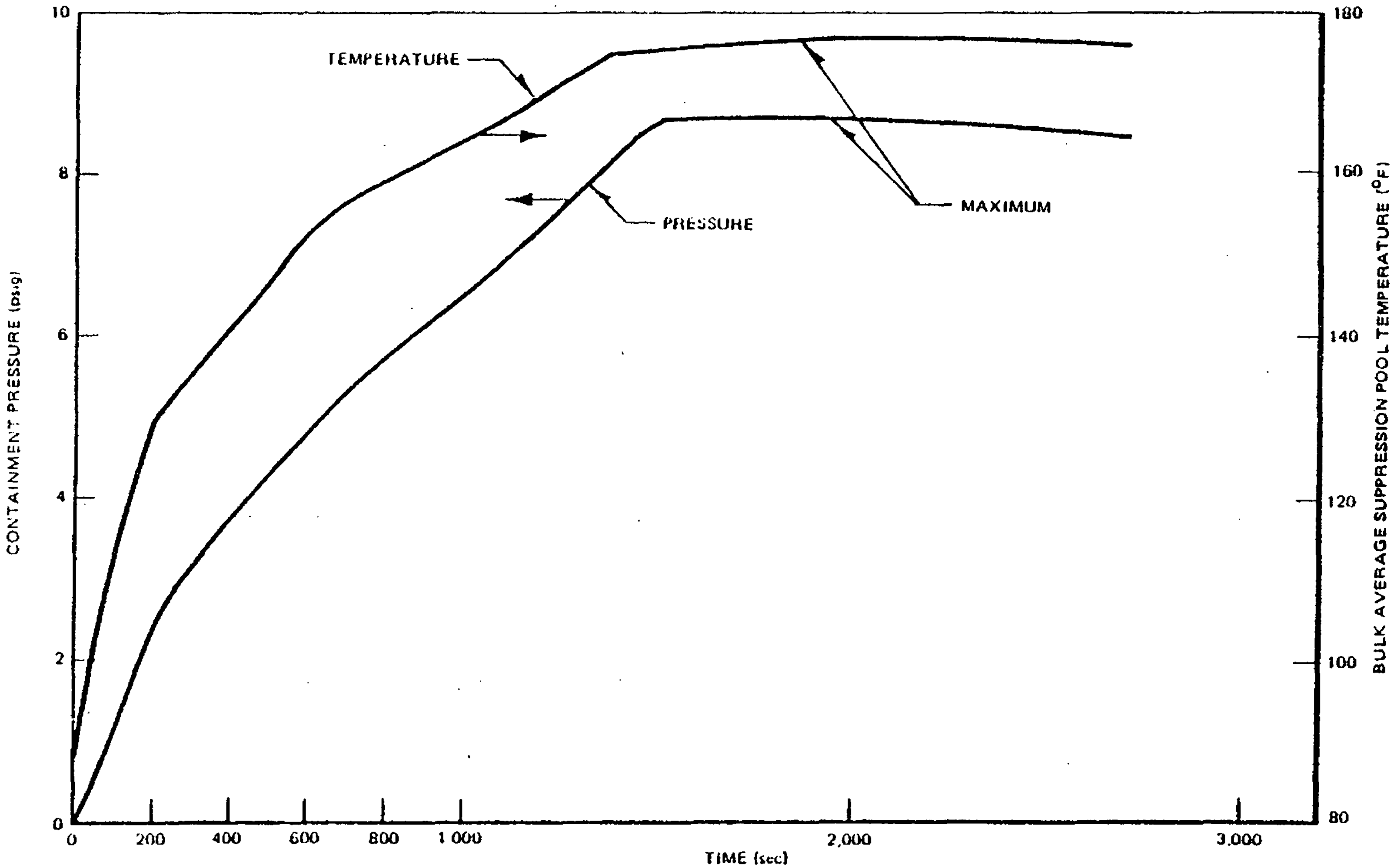

Figure 4.2.5. BWR/5 MSIV Closure with 2 min Boron Injection without ARI

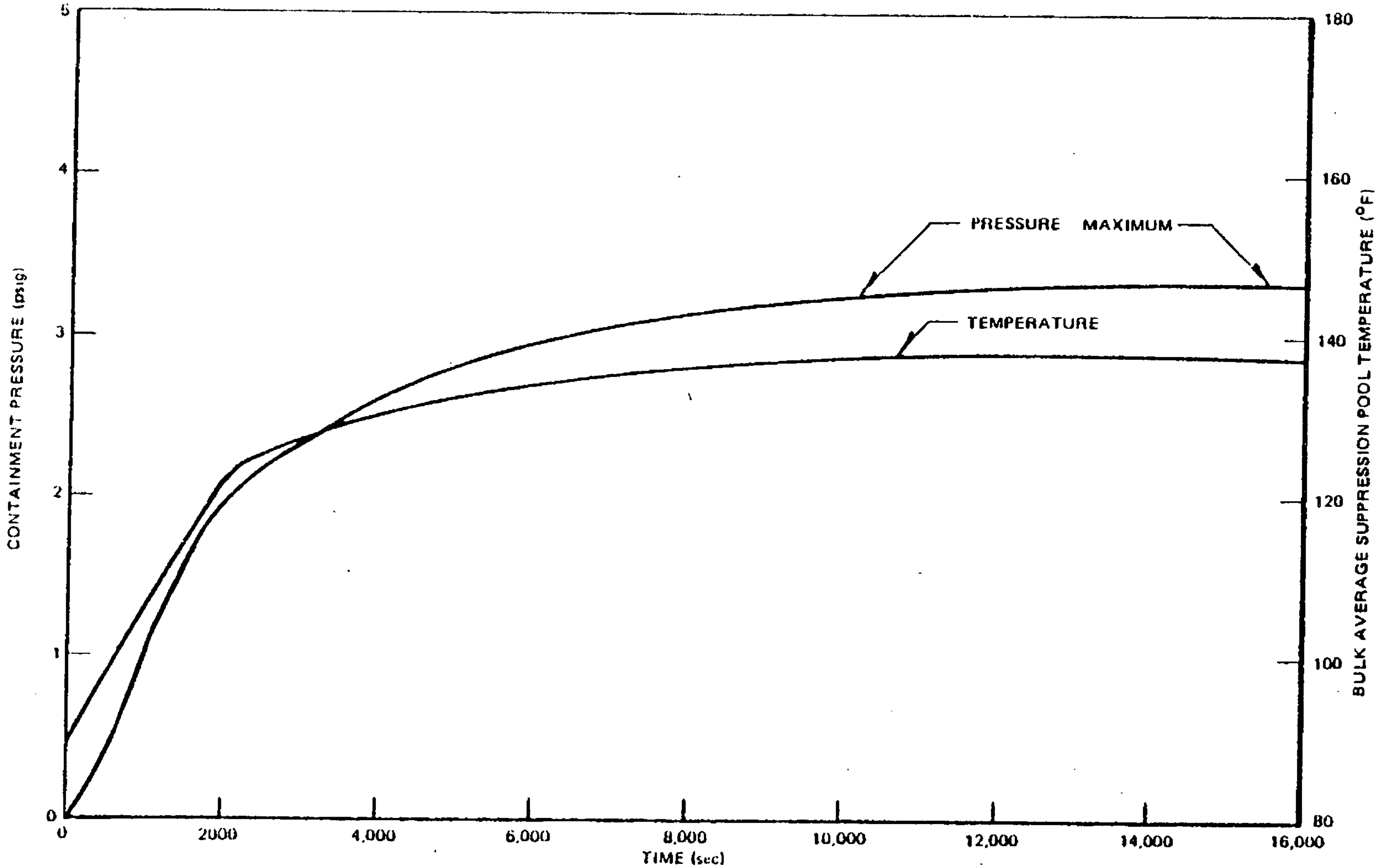

Figure 4.2.6. BWR/5 MSIV Closure with ARI

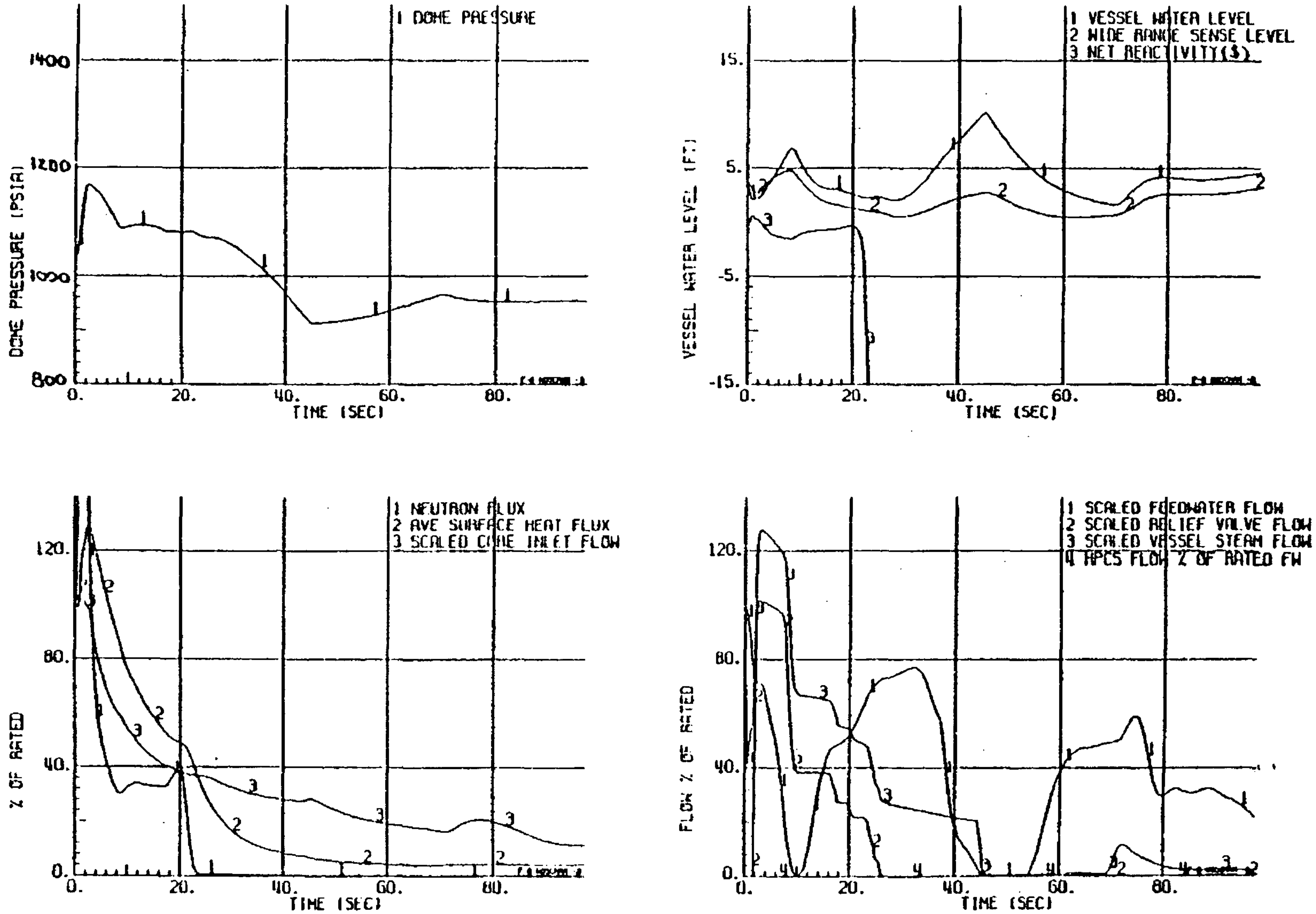

Figure 4.2.7. BWR/5 Turbine Trip, ARI at 20 sec

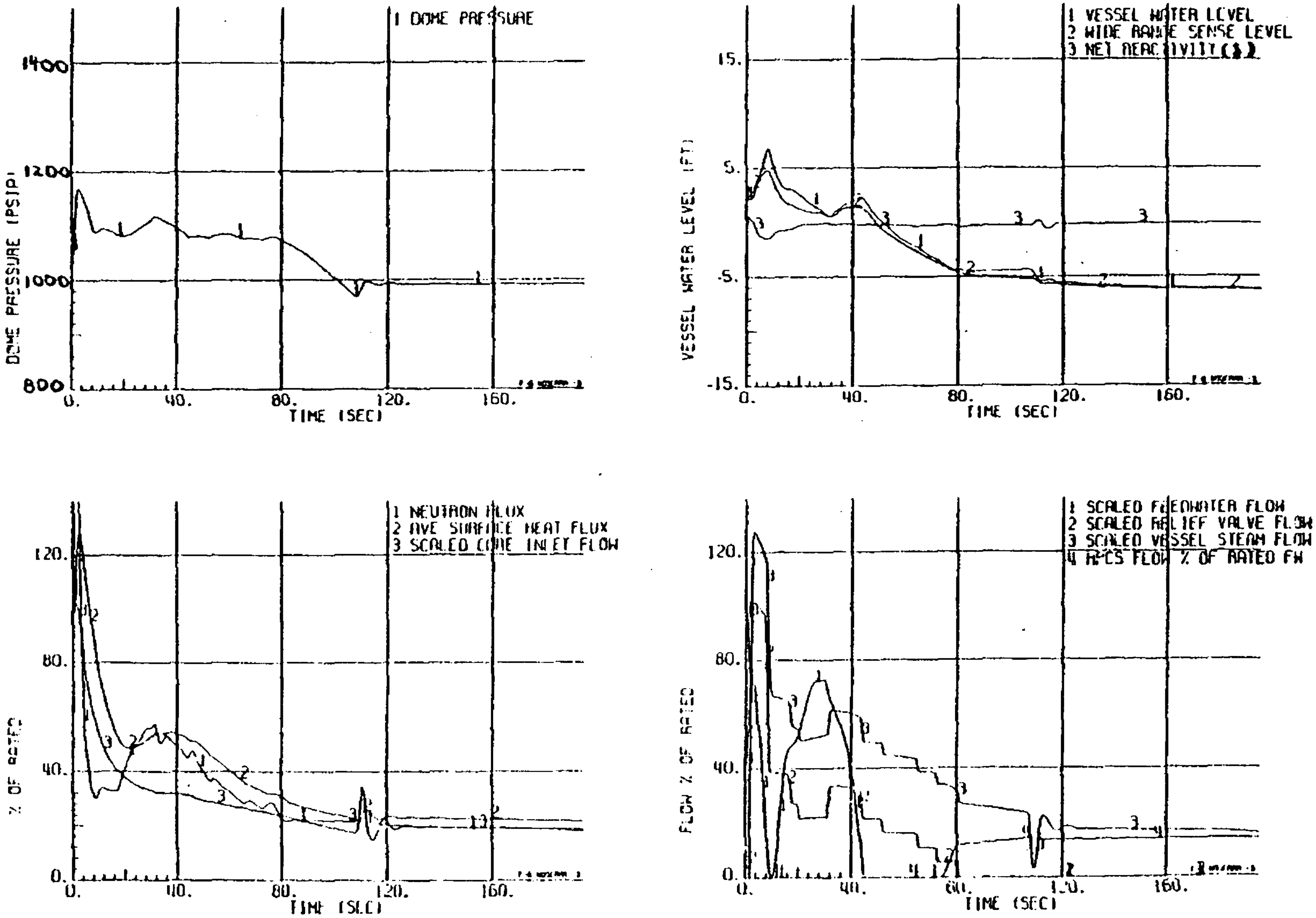

Figure 4.2.8. BWR/5 Turbine Trip, FW Limited, Without ARI, 2 min - 86 gpm Boron

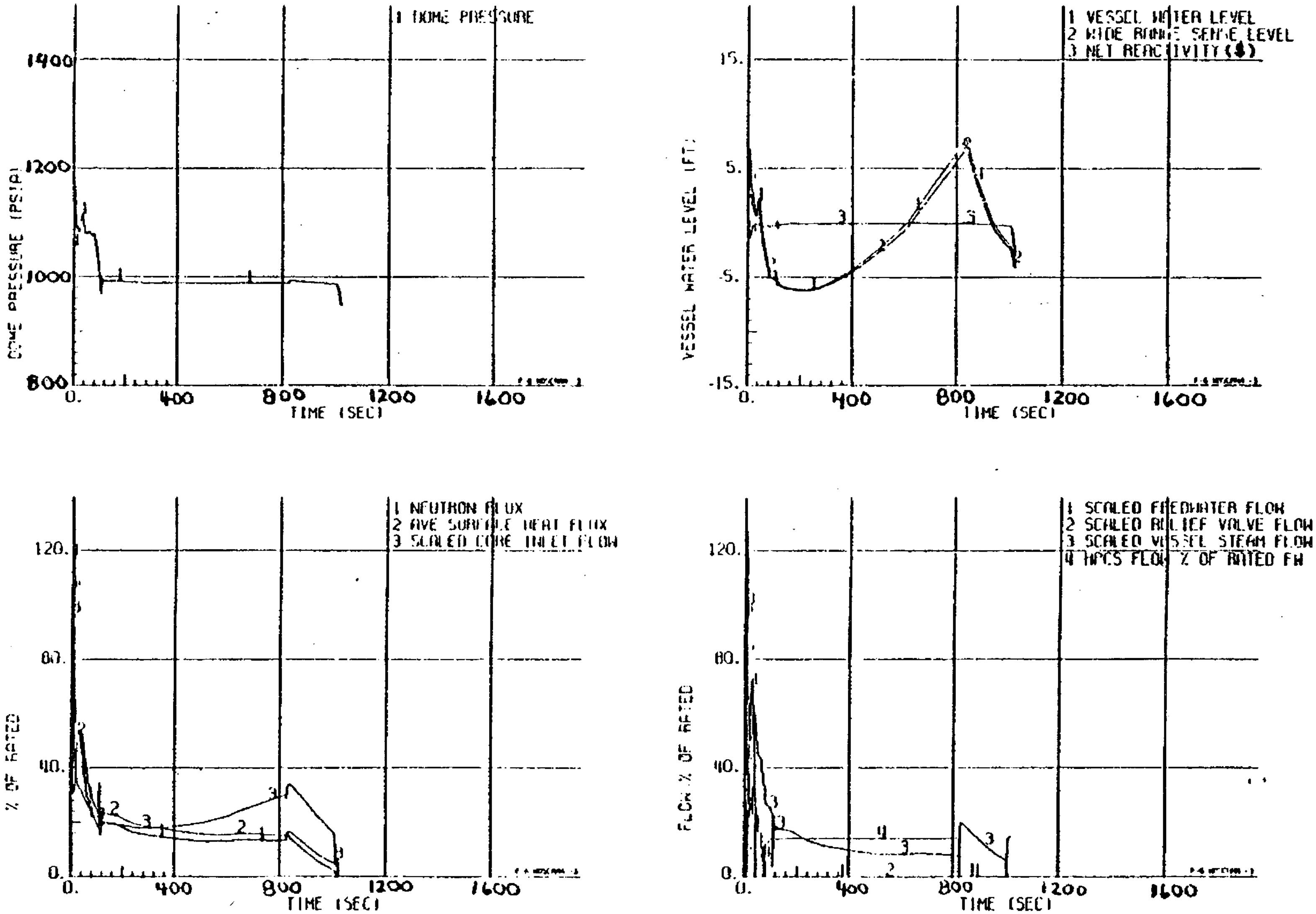

Figure 4.2.9. BWR/5 Turbine Trip, FW Limited, Without ARI, 2 min - 86 gpm Boron

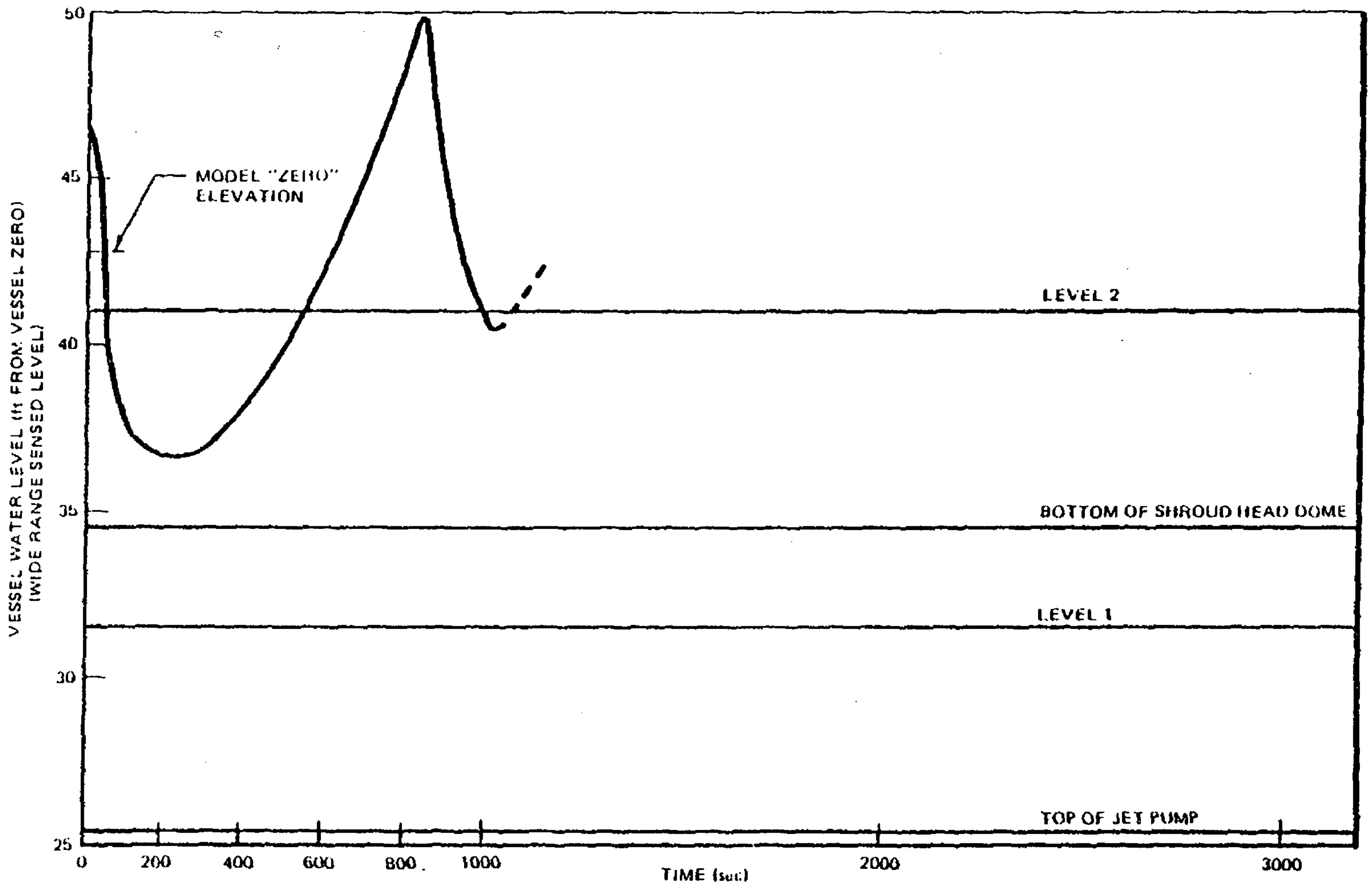

Figure 4.2-10. BWR/5 Turbine Trip with 2 min Boron Injection without ARI

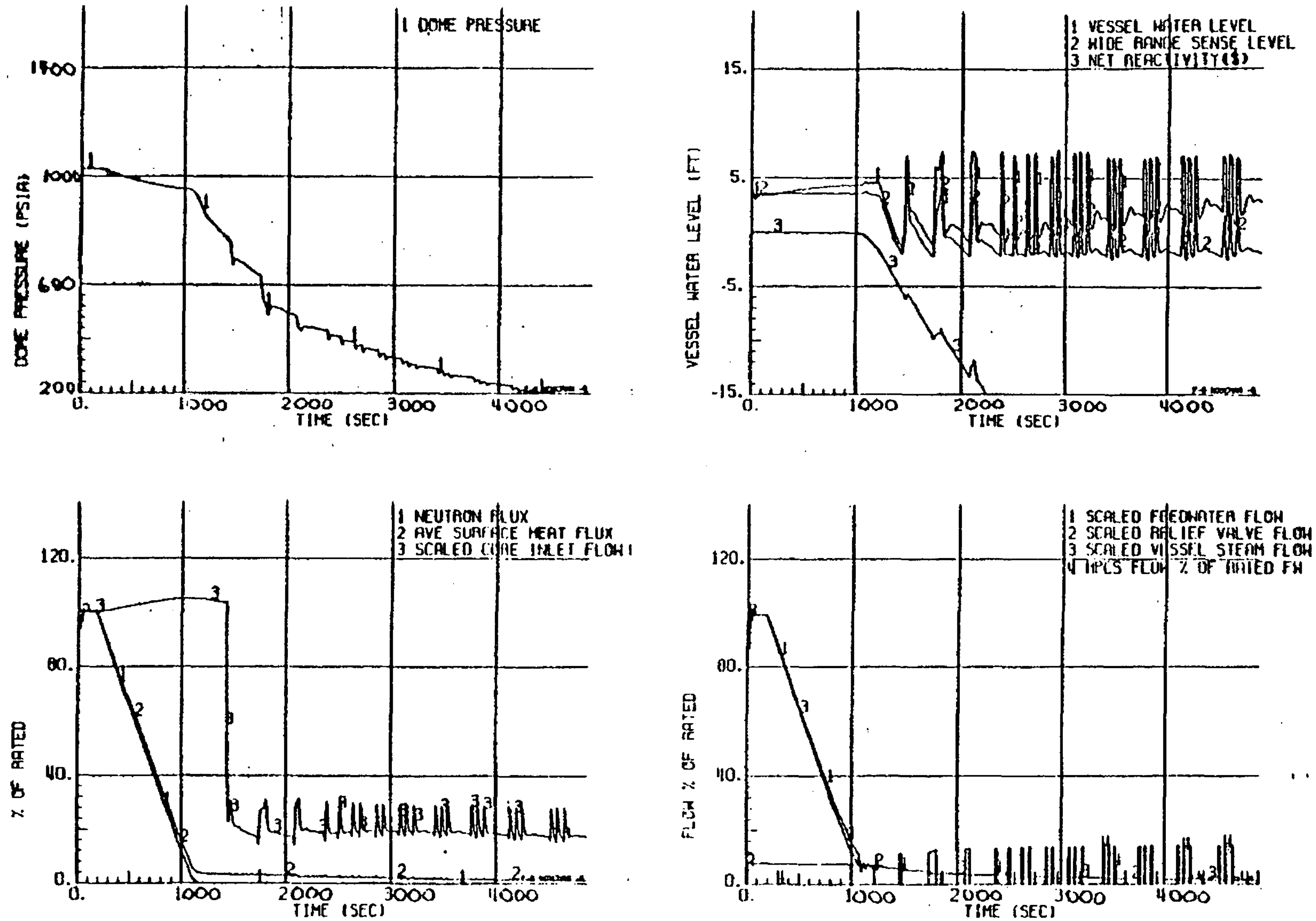

Figure 4.2.11. BWR/5 IORV 2 min - 86 gpm Boron, After Manual Scram Fails, 95% Mixing Efficiency

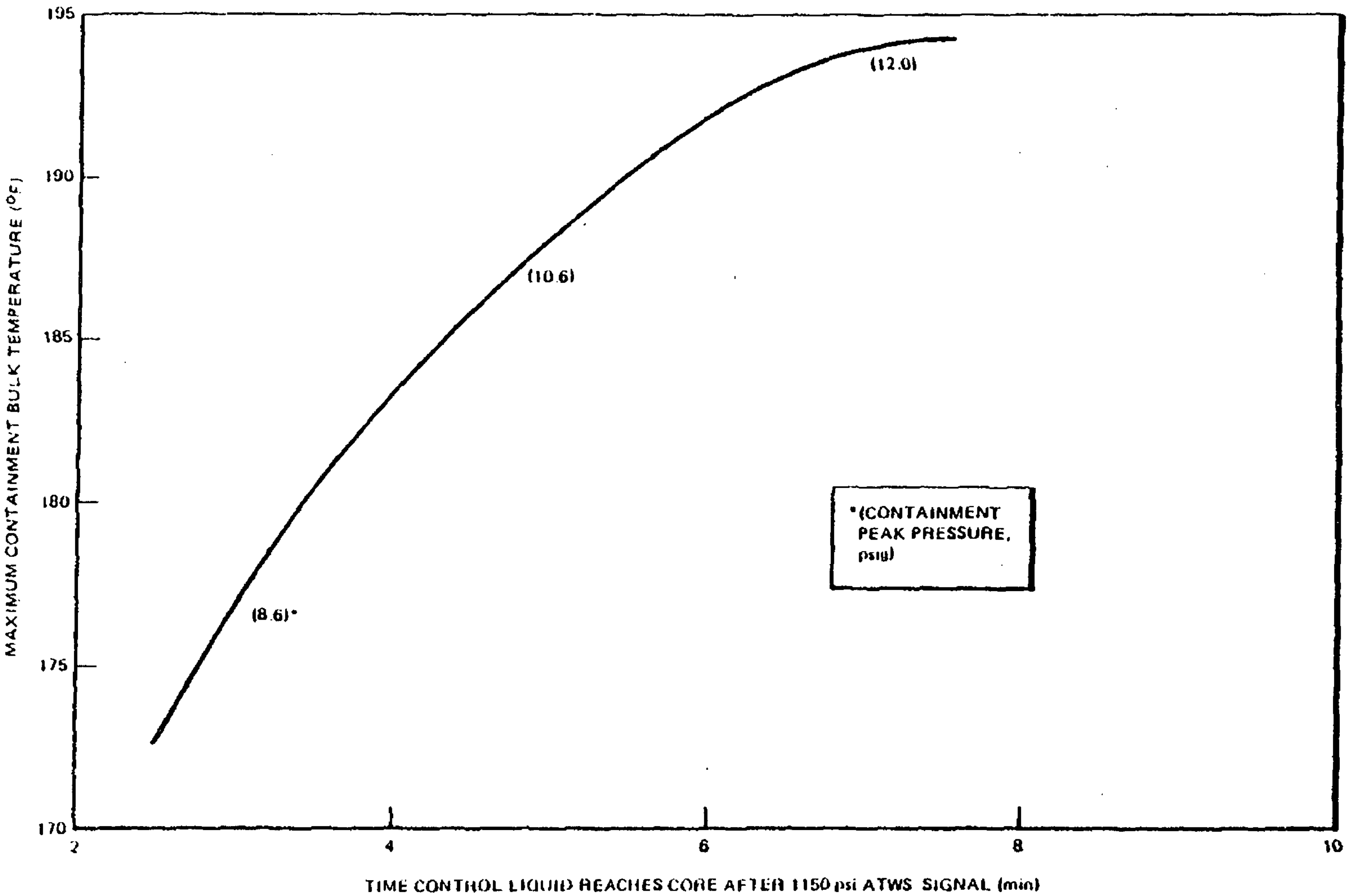

Figure 4.2.12. BWR/5 MSIV Closure, 86 gpm Boron, Without ARI

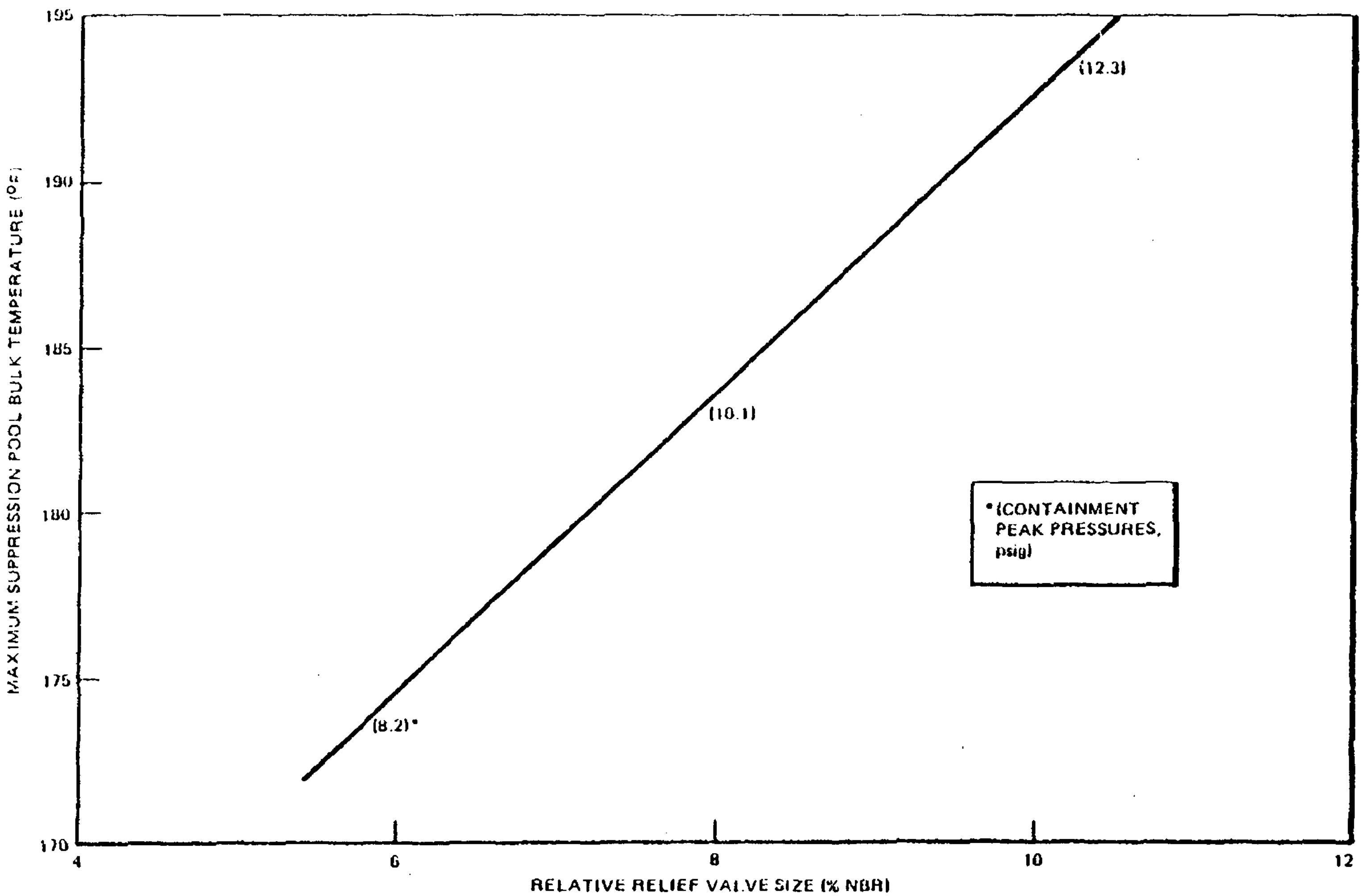

Figure 4.2.13. BWR/5 IORV, Without ARI, 2 min - 86 gpm Boron

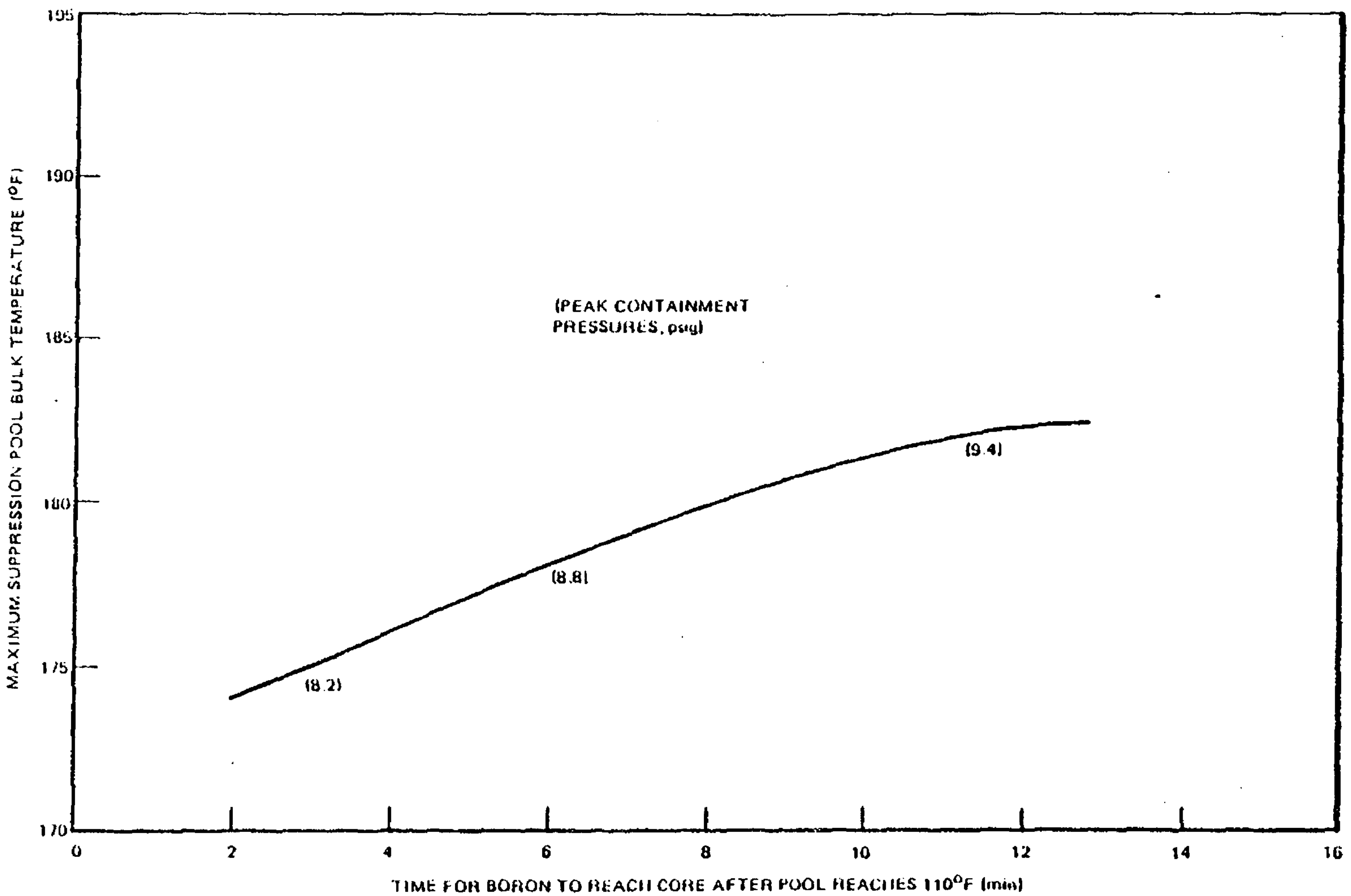

Figure 4.2.14. BWR/5 IORV, Without ARI, 86 gpm Boron.

4.3 RESULTS OF ATWS EVENTS - BWR/6 MARK III

4.3.1 MSIV Closure Event

4.3.1.1 Overview of Response Without Scram

A detailed description of all aspects of this event is given below. The behavior of the plant is basically separable into an early or short term transient involving a sharp pressure rise and power peak, and a longer term portion that requires evaluation of coolant and containment conditions as the reactor is ultimately brought to shutdown.

The effectiveness of the recirculation pump trip feature presented in NEDO-10349 and NEDO-20626 are reconfirmed by this analysis. It permits the relief valves to limit the pressure disturbance acceptably, reduces the power peak which is created early in the transient, and establishes a relatively low power generation rate for the long-term portion of the transient.

Ultimate solution of the lack-of-scram situation must involve insertion of negative reactivity into the reactor, thereby bringing the reactor to a fully shutdown condition terminating the long-term aspects of the event. The ARI feature is provided as an effective way to mitigate for common-cause failures in the logic of the scram system. In the very remote case of its ineffectiveness, the automated SLCS provides further protection and shutdown capability. Coolant inventory must be adequately maintained by using the high pressure coolant supply systems available on each BWR to replace the coolant loss as steam flow leaves the primary system through the relief valves. Simply adding more water without inserting negative reactivity has the effect of raising the power generation rate and the amount of inventory leaving the system to the containment. The steam reaching the suppression pool continues to heat it and pressurize the containment until the power generation/steam flow can be reduced and finally terminated. The RHR ultimately recools the pool and eventually the reactor also (in shutdown cooling mode) if the isolation valves cannot be reopened (the preferred method of cooling down).

4.3.1.2 Sequence of Events During the BWR/6 MSIV Closure Transient

The MSIV closure transient provides some of the most severe conditions following a postulated failure to scram. Listed below in the sequence of occurrence are significant points from the transient with representative times when each event occurs. Results for both cases - with ARI and also assuming its failure are presented in Table 4.3.1.

Table 4.3.1
BWR/6 MARK III
MSIV CLOSURE
SEQUENCE OF EVENTS

		With ARI	Without ARI
1.	Nominal (4-sec) MSIV Closure Begins - All Normal Scrams Fail	0	0
2.	Pressure and Power Rise Begins	0-2 seconds	0-2 seconds
3.	Relief Valves Lift	4 seconds	4 seconds
4.	ATWS High Pressure Set Point is Reached (1150 psig): Recirculation pumps are tripped, ARI is initiated, and SLCS timed logic is activated	5 seconds	5 seconds
5.	Some Fuel Experiences Transition Boiling	5 seconds	5 seconds
6.	Vessel Pressure Peaks	8 seconds	8 seconds
7.	Reactor Water Level Drops to Level 2 and Initiates RCIC and HPCS	24 seconds	24 seconds
8.	ARI Control Rod Injection Completed, Eliminating SLCS and FW Limit Actions	25 seconds	Fails
9.	ATWS Logic Timer Completed - Initiate FW Limit	N/A	35 seconds
10.	Feedwater Flow Stops or Coasts Down (or limited)	20 seconds*	20 seconds*

*FW turbine coastdown assumed after loss of steam. Motor-driven FW plants have slightly different sequence for level and inventory supply actions.

Table 4.3.1 (Continued)

		With ARI	Without ARI
11.	Reactor Water Level Drops to Level 3 and Initiates Containment Isolation	20 seconds	20 seconds
12.	HPCS and RCIC Flow Starts	43 seconds	43 seconds
13.	Water Level Reaches Minimum and Begins to Rise	50 seconds	100 seconds
14.	Final ATWS Logic Timer Completed - Initiate SLCS	N/A	2 minutes
15.	Liquid Boron Flow Reaches Core	N/A	3 minutes
16.	RHR Flow Begins (Pool Cooling)	≥11 minutes	11 minutes
17.	Hot Shutdown is Achieved	25 minutes	21 minutes
18.	Peak Containment Temperature and Pressure	130 minutes	33 minutes

4.3.1.3 **MSIV Closure Event Discussion**

In this event, all main steam lines are assumed to isolate from rated power condition with nominal valve closure speed (4 seconds). Figures 4.3.1 and 4.3.2 show the initial portions of the event for the more likely plant ATWS transient in which ARI quickly shuts down the unit, and the case in which ARI also fails and the automated SLCS is called upon to shut down the plant.

In each case, the initial power and pressure increases are the same, with neutron flux reaching 790% NBR near 4 seconds, fuel average surface heat flux reaches 151% at 5 seconds, some fuel may experience boiling transition, and the peak pressure is limited to 1329 psig near 8 seconds. The normal reactor scrams from position switches on the MSIVs, high neutron flux, and high vessel pressure are totally ignored in this analysis. The transient is limited well within the Service Level C (Emergency) overpressure limit of 1500 psig through the automatic action of the ATWS high pressure recirculation pump trip which is initiated when vessel dome pressure exceeds 1150 psig and the relieving action of the safety/relief valves which all open, then start reclosing near 20 seconds. The action of the recirculation pump trip and the diminishing reactor pressure reduces neutron flux until near 20 seconds it is less than

20% NBR. Peak fuel conditions are quickly reduced with the power reduction and no fuel damage is expected. This neglects for now, rewetting of the fuel which should occur in cases of this type after the initial power transient has subsided.

By 25 seconds the high pressure logic which began the ATWS protection will have accomplished the ARI function, inserting the rods and shutting off the generated power. This deactivates the automatic boron injection and feedwater limit and turns the remainder of the event into an essentially normal isolated shutdown as shown in Figure 4.3.1. Some relief valve cycling will occur to handle steam generated by decay heat, but peak suppression pool temperature will be only 130°F (at 2 hours and 10 minutes) assuming the RHR loops are turned on in the pool cooling mode after 11 minutes into the event. The water level in the reactor drops to the Level 3 setpoint (another scram signal plus initiation of containment isolation) at about 20 seconds, and to the Level 2 set point (about -2 ft on Figure 4.3.1) near 25 seconds, starting the RCIC and HPCS systems. They replace the main feedwater system which was assumed to coast down to zero flow (near 20 seconds) due to loss of steam to the turbine driven pumps. If the plant had motor-driven pumps, normal feedwater flow would remain available and reduce the subsequent level swings. The minimum level for the simulated case is reached near 50 seconds as shown in Figure 4.3.3, about 15 inches above the Level 1 set point. The HPCS and RCIC then restore level to its normal range and an essentially normal shutdown can be accomplished.

If the ARI function is arbitrarily assumed to fail as well as all other attempts to insert enough control rods within the two-minute timed period, the ATWS logic will continue to sense that the APRM signals are not downscale and not enough rods are in their full-in positions, and the automatic start of boron injection will begin. Instead of shutting off immediately, the power is predicted to remain in the 10-30% range beyond 20 seconds as shown in Figure 4.3.2 and extend through the long term transient in Figure 4.3.4. The significant features and peak values during the early part of the event are the same as the previous case, however, the key difference here is the continuing reduction of water level outside the core shroud until it reaches a minimum between the top of the jet jumps and the Level 1 set point at about 2 minutes. Figure 4.3.5 shows level transient with more detail. The steam-water mixture inside the core shroud remains well above the core and up into the steam separator standpipes as RCIC and HPCS flow provide inventory. Most of the relief valves have reclosed by this time with about 3 valves handling the generated steam and pressure, cycling near their setpoints.

Boron injection is started by the two SLCS pumps near 2 minutes and it reaches the core about 1 minute later. During the following 15 minute period (out to about 1100 seconds on these plots), the key result is that power is suppressed slightly, reducing the steaming rate and allowing water level to be restored. This also induces higher natural circulation core flow which follows the water level behavior closely. The level has reached the high level turn-off (Level 8) of the HPCS and RCIC at about 1150 seconds and an off-on-off cycle of these systems is shown here as level swings down to Level 2 and back up to Level 8 (between 1200 and 1400 seconds) in a fully automatic simulation. By this time, manual operator action using the RCIC to modulate level in the normal range would be recommended and expected.

Near 1295 seconds the generated power approaches zero as net reactivity (shown in the upper right plot of Figure 4.3.4) becomes negative and continues thereafter to be forced negative by the accumulation of boron in the reactor. At that time the water in the vessel has 345 ppm boron even assuming only 75% mixing of the injected poison. This accomplishes hot shutdown, and the remainder of the steam flow to the pool is simply due to decay heat.

The bulk average temperature of the suppression pool and pressure in the containment are shown in Figure 4.3.6. They rise gradually to peaks of 155°F and 5.3 psig, respectively, after 33 minutes. Beyond this time the pool cooling capability of the RHR exceeds the steaming rate generated by decay heat and containment conditions are gradually reduced. The peaks are well within the design limits: 185°F and 15 psig, and containment integrity is maintained. Boron continues to be injected into the vessel for about 50 minutes. At this point controlled reactor cooldown can be initiated. The total concentration is specified to be enough to maintain cold, nuclear shutdown conditions even when the RHR system is eventually switched to the reactor shutdown mode, bringing the plant successfully all the way down in temperature by normal procedures.

4.3.2 Turbine Trip Event

4.3.2.1 Overview of System Response Without Scram

This important type of transient is described in detail in the following sections. Its initial characteristics are much like the main steam isolation event described above with a rapid steam shutoff and pressure and power increases which are mitigated by the action of the safety/relief valves and the high pressure ATWS RPT. As the event progresses, however, the availability of the main condenser makes it possible for the relief valves to close after about a minute. If subsequent isolation can be avoided, this terminates steam flow to the pool. However, BWR/6 water level is close to the Level 1 isolation setpoint. If initiated, the final portion of the event is similar to the MSIV closure event.

4.3.2.2 Sequence of Events During the BWR/6 Turbine Trip Transient

The listing of significant events during this ATWS event is provided below. Results for both cases - with ARI and also assuming its failure - are presented in Table 4.3.2.

Table 4.3.2
BWR/6 MARK III
TURBINE TRIP EVENT
SEQUENCE OF EVENTS

	With ARI	Without ARI
1. Turbine Trips, Bypass Opens - Assume all Normal Scrams Fails	0	0
2. Pressure and Power Rise Begins	0	0
3. Relief Valves Lift	2	2 seconds
4. ATWS High Pressure Setpoint is Reached (1150 psig): Recirculation Pumps are Tripped,* ARI is Initiated, and SLCS Timed Logic is Activated	2	2 seconds

*Direct recirculation pump trip from turbine stop valve closure was conservatively neglected in this series of simulations. Later analyses will include its effect.

Table 4.3.2 (Continued)

		With ARI	Without ARI
5.	Vessel Pressure Peaks	2-3	2-3 seconds
6.	Some Fuel Experiences Transition Boiling	3	3 seconds
7.	ARI Control Rod Injection Completed, Eliminating SLCS and FW Limiting Actions	22 seconds	Fails
8.	ATWS Logic Timer Completed-Initiate Feedwater Flow Limit	N/A	32 seconds
9.	Reactor Water Level Drops to Level 3 and Initiates Containment Isolation		50 seconds
10.	Reactor Water Level Drops to Level 2 and Initiates HPCS and RCIC		60 seconds
11.	HPCS and RCIC Flow Begins		78 seconds
12.	MSIV's Close on Low Level (L1)	N/A	100 seconds
13.	Final ATWS Logic Timer Completed - Initiate SLCS	N/A	2 minutes
14.	Liquid Control Flow Reaches Core	N/A	3 minutes
15.	Reactor Water Level Reaches Minimum and Begins to Rise		3-4 minutes
16.	RHR Flow Begins (Pool Cooling)	$\gtrsim$11 minutes	11 minutes
17.	Hot Shutdown is Achieved	22 seconds	20 minutes
18.	Containment Temperature and Pressure Peak		33 minutes

4.3.2.3 Turbine Trip Event Discussion

This abnormal transient event starts with an unexpected closure of all turbine stop valves (within about 0.1 second). Figure 4.3.7 and 4.3.8 show the initial portions of the event for the more likely plant ATWS transient in which ARI provides a diverse logic path to quickly shut down the unit, and the case without ARI and the automated standby liquid control system (SLCS) is called upon to shut down the plant.

In each case, the initial power and pressure increases are the same, with neutron flux reaching 410% NBR near one second. Fuel average heat flux reaches 138%

NBR at about 3 seconds, some fuel may experience boiling transition, and the peak pressure is limited to 1230 psig between 2-3 seconds. The normal reactor scram signal from position switches on the valves, high neutron flux, and high vessel pressure are totally ignored in this analysis. Even so, the transient pressure is limited well within the Service Level C overpressure limit of 1500 psig through the automatic action of the ATWS high pressure recirculation pump trip (which is initiated when vessel dome pressure exceeds 1150 psig) and the relieving action of the safety/relief valves which all open, then start reclosing near 7 seconds. The plots show both the steam flow leaving the vessel and the relief valve flow - the difference is the flow through the bypass valves to the condenser.

By about 22 seconds, the high pressure logic which began the ATWS protection will have accomplished the ARI function, inserting the rods and shutting off the generated power. This deactivates the automatic boron injection and FW limit, and turns the remainder of the event into an essentially normal turbine-generator trip shutdown. No additional relief valve flow will occur as the bypass/pressure control system will handle steam generated by decay heat. Peak suppression pool bulk temperature will occur at the time of the last relief actuation and will be only 96°F. The RHR can be activated in the pool cooling mode whenever convenient to reduce the pool temperature. Reactor water level remains in the normal range throughout the event by the feedwater system and no RCIC or HPCS initiation is expected.

If the ARI function is arbitrarily assumed to fail as well as all other attempts to insert enough control rods within the two minute timed period, the ATWS logic will continue to sense that the APRM signals are not downscale and not enough rods are in their full-in positions, and the automatic start of boron injection will begin. The long-term behavior predicted for this event is shown in Figure 4.3.9. Near 95 seconds, the pressure regulator and bypass valves close down to regain pressure control near the regulator setpoint. However, the water level has dropped to the main steam isolation setpoint (Level 1) which occurs at about 100 seconds. Avoidance of this isolation is strongly desired and can be accomplished by optimization of the feedwater limiter (shown here to shut feedwater totally off). This case represents a bounding event where the isolation might occur and maximizes the resulting pool temperature. Pressure is raised to the relief setpoints and the transient takes on the form of the MSIV closure shown in Figure 4.3.4. Introduction of boron to the core at 3 minutes gradually restores level and core flow before dropping the power near 20 minutes when hot shutdown is achieved. Thereafter, only decay heat is boiled off to the pool, giving the peak pool temperature of 150°F (4.6 psig) at about 33 minutes. These values remain well within the containment design requirements of 185°F and 15 psig. Figure 4.3.10 and 4.3.11 show detailed plots of the water level outside the core shroud, the pool temperature, and the containment pressure through the peak portion of the event. Water level inside the core shroud is a two-phase mixture which remains well above the core and up into the steam separator standpipes as RCIC and HPCS flow provide coolant inventory. The boron will continue to build the poison concentration in the vessel until it is all injected near 50 minutes making it possible for a controlled reactor cooldown. The total concentration is specified to be enough to maintain cold nuclear shutdown conditions even when the RHR system is eventually switched to the reactor shutdown cooling mode, bringing the plant successfully all the way down in temperature by normal procedures. In the non-isolated case, the main condenser is available to make this shutdown even easier.

4.3.3 Inadvertent Open Relief Valve Event

4.3.3.1 Overview of Response Without Scram

A detailed description of the sequence of events for this event is given below as it has been simulated. This event has no rapid excursions as in the previous two events but is merely a long term depressurization. The recirculation pump trip feature does not occur until late in the event after hot shutdown is achieved.

4.3.3.2 Sequence of Events During Inadvertent Open Relief Valve (IORV) ATWS Transient

This event begins when one of the primary relief valves on the main steamlines inadvertently opens without influence from any other portion of the system. The reactor pressure at a nominal value prior to the event. The resulting sequence of events is shown in Table 4.3.3. At the time that the relief valve opens, there is a momentary depressurization (a few seconds) until the turbine control valves sense it and close slightly (dropping unit electrical output) to control the pressure. For general application of this analysis, a relief valve capacity of 7.1% net boiler rated was utilized (the nominal flow of a valve on a BWR/6-218 inch vessel plant). Larger plants should be less severe. After about two minutes, the suppression pool temperature, which was initially assumed to be at 90°F, has risen to the alarm point of 95°F. If attempts to reclose the valve are unsuccessful, the operator at this point will turn on the RHR system in the pool cooling mode to maintain low suppression pool temperature. If attempts to close the valve continue to be unsuccessful, the temperature will continue to rise and at 9 minutes will reach 110°F at which point the operator is required to manually scram the plant. Should scram fail to occur at this point, it should be obvious to the operator who would be watching the panels for positive feedback from the manual scram attempt. If no rods go into the core, manual ATWS protection is to be taken within 10 minutes. The manual ATWS "button" available to the operator will initiate ARI as well as start the SLCS timed logic.

If for some reason neither normal manual scram nor the ARI are effective, the BWR/6 is still able to mitigate the event at this time. The ATWS logic will have determined that even ARI was unsuccessful and the control rods are still not inserted, and at 21 minutes into the event will activate the standby liquid control system. For this case, because the recirculation flow is maintained, a boron mixing efficiency of 95% is assumed and the delay time inside of the vessel is small so that at 22 minutes the control liquid reaches the core and shutdown begins. The power has been reduced by 38 minutes to the point that the amount of steam generated is less than the relief valve capacity and the pressure now begins to fall more rapidly. The turbine control valves have closed completely. These events are depicted in Figure 4.3.12. By 41 minutes, the pressure has dropped to the low line pressure isolation point of 850 psig and the main steam isolation valves close. Simulating plants with turbine-driven feedwater pumps, the feedwater was assumed to be lost within 20 seconds of the isolation. This caused the water level in the vessel to decrease and at 34 minutes the low level point (L2) was reached where the recirculation pumps were automatically tripped and the HPCS and RCIC systems were activated. These systems are shown to automatically actuate on at low level (L2) and off at high level (L8) as specified to maintain water inventory in the vessel, although manual action is expected to maintain level using only the RCIC. The depressurization of the vessel will continue with the relief valve discharging into the suppression pool, and the maximum bulk pool temperature of 170°F will occur at 72 minutes. The peak containment pressure of 7.4 psig occurs at the same time. Both values are well below the criteria values of 185°F and 15 psig.

Table 4.3.3
BWR/6 MARK III
INADVERTENT OPENING OF A
RELIEF VALVE (IORV), WITHOUT ARI

1.	Relief Valve Opens Inadvertently and Fails to Close	0
2.	Alarm Sounds at 95°F and Operator Initiates Pool Cooling	2 minutes
3.	Suppression Pool Bulk Temperature Reaches 110°F, Operator Attempts Manual Scram, Assume Scram Fails	9 minutes
4.	Operator Manually Initiates ATWS Protection (ARI and SLCS)	19 minutes
5.	Assume ARI Fails	19.5 minutes
6.	Standby Liquid Control System Starts	21 minutes
7.	Liquid Boron Reaches Core	22 minutes
8.	Power is less than Relief Valve Capacity	38 minutes
9.	Isolation on Low Steamline Pressure (850 psi)	41 minutes
10.	Hot Shutdown is Achieved	
11.	Peak Suppression Pool Bulk Temperature and Pressure are Reached	72 minutes

4.3.4 Sensitivity Results

Although suppression pool temperatures for the BWR/6 Mark III designs are predicted to be well below the 185°F containment temperature criterion in all cases, some sensitivity studies have been done to determine the manner in which peak suppression pool temperature will vary given variations in system performance.

For the MSIV closure event, the sensitivity to changing the boron system delay times was studied. In Table 4.3.5 the time when the initial boron flow gets to the core is varied from 1 to 5 minutes. The center column of 3 minutes is for the base case delay times and specifically represents an 86 gpm boron system pumping into a 251-inch vessel. Smaller plants which may also have 86 gpm capability may have even greater margins. Clearly, the BWR/6 plants are well below all criteria even when the boron delay variation is considered. A summary for all events is shown in Table 4.3.6.

Table 4.3.5
BWR/6 MARK III
MSIV CLOSURE WITHOUT ARI
SENSITIVITY STUDY

	86 gpm Boron at 75% Mixing EFF Total Delay Time (Minutes)		
	1	3*	5
Maximum Suppression Pool Bulk Temperature (°F)	151 @ 27 min	155 @ 27 min	159 @ 25 min
Maximum Containment Pressure (psig)**	4.8	5.3	5.9

*Base case has 2 minute logic delay plus 1 minute transport time allowance.
**Mark III containment design pressure is 15 psig.

Table 4.3.6

SUMMARY BWR/6 MARK III

Without ARI:

	MSIV Closure	Turbine Trip	IORV
Maximum Neutron Flux (%)	790 @ 4.0 sec	410 @ 0.9 sec	100 @ 0
Maximum Vessel Bottom Pressure (psig)	1329 @ 4.0 sec	1230 @ 2.4 sec	1025
Maximum Average Heat Flux (%)	151 @ 5.0 sec	138 @ 2.6 sec	100 @ 0
Maximum Suppression Pool Bulk Temperature (°F)	155 @ 33 min	150 @ 33 min	170 @ 72 min
Associated Containment Pressure (psig)	5.3	4.6	7.4

With ARI:

	MSIV Closure	Turbine Trip	IORV
Maximum Suppression Pool Bulk Temperature (°F)	130 @ 33 min	96 @ 45 sec	
Associated Containment Pressure (psig)	2.7	0.4	

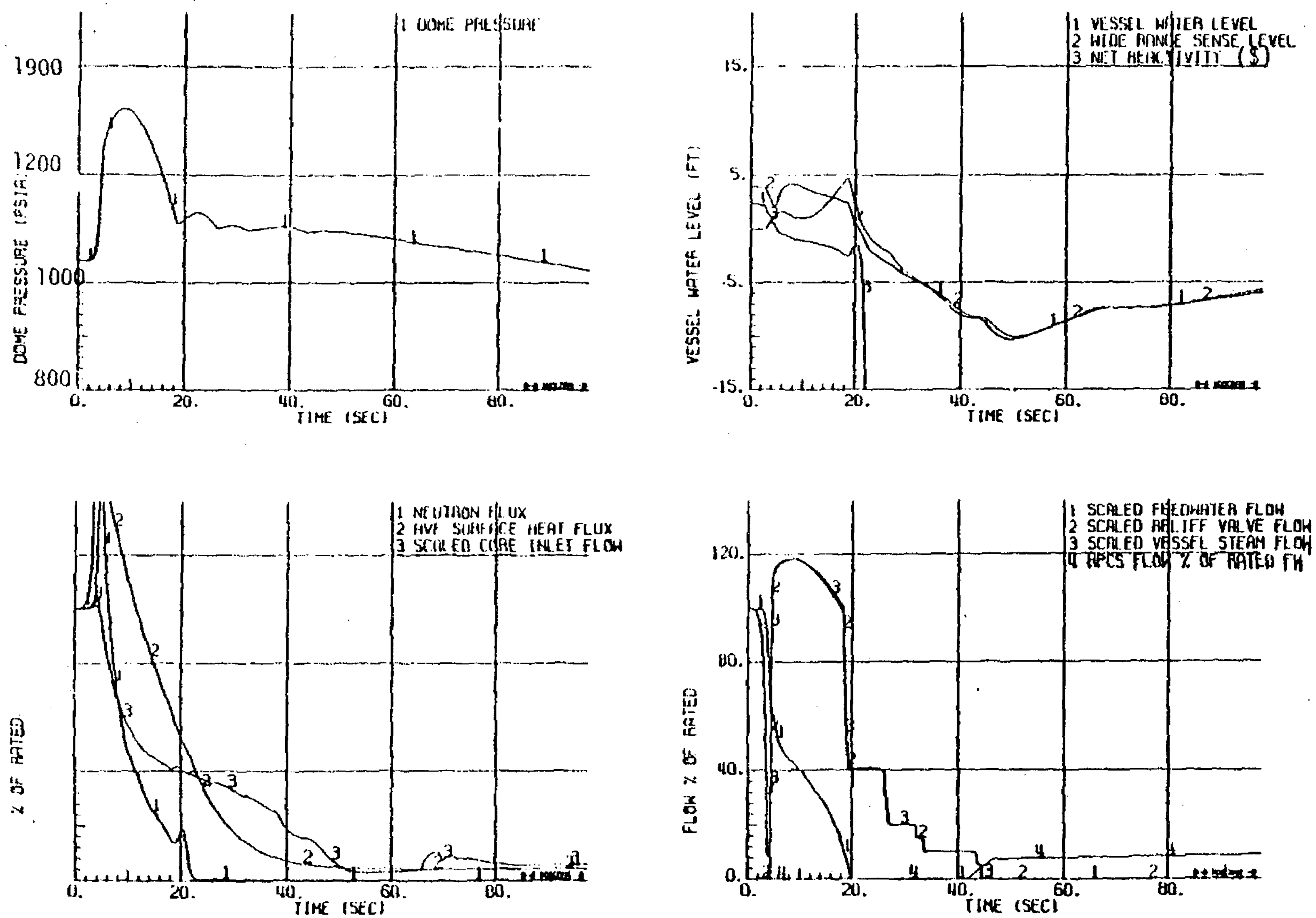

Figure 4.3.1. BWR/6 MSIV Closure, With ARI at 20 sec

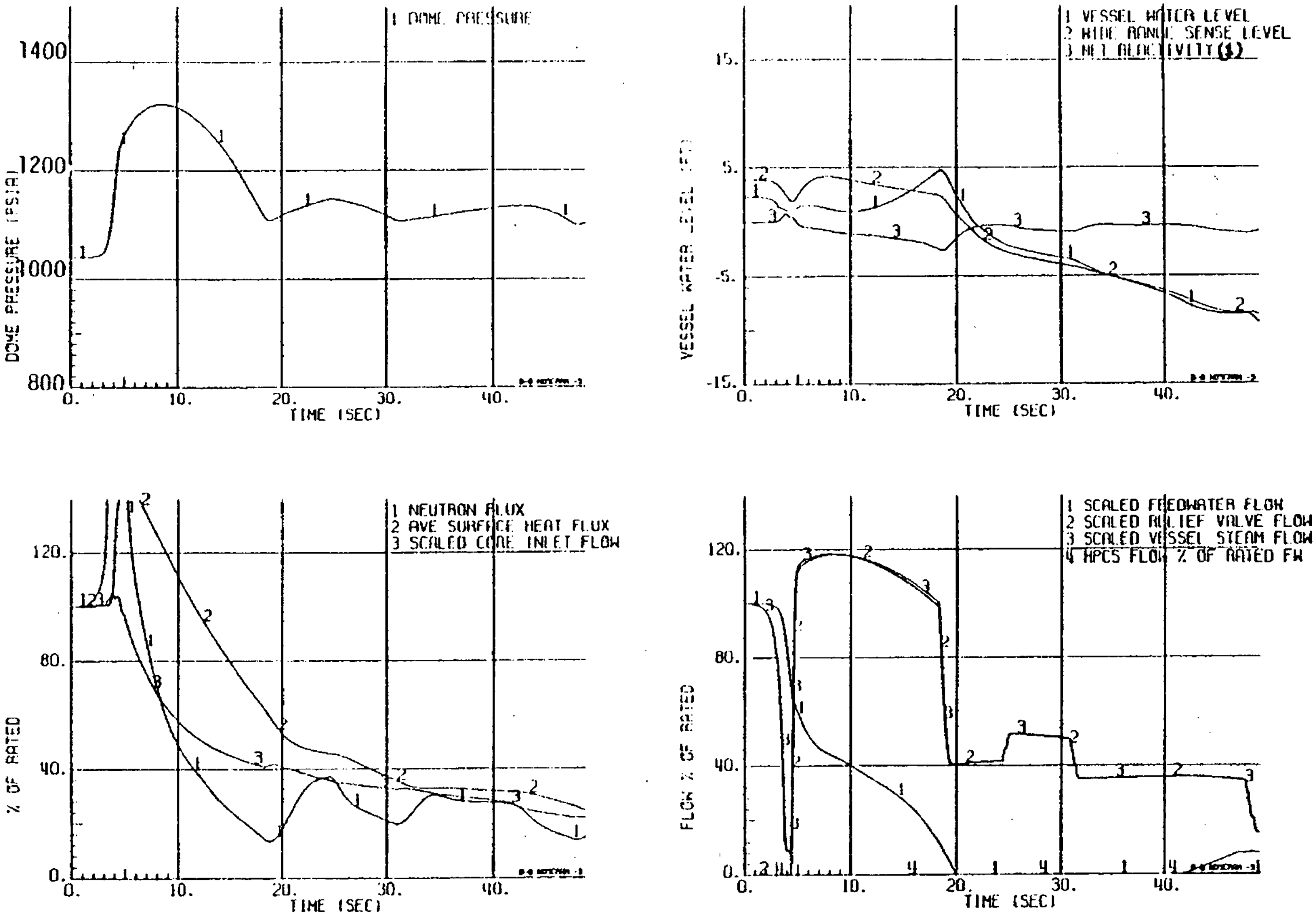

Figure 4.3.2. BWR/6 MSIV Closure, 2 min - 86 gpm Boron, Without ARI, 75% Mixing Efficiency

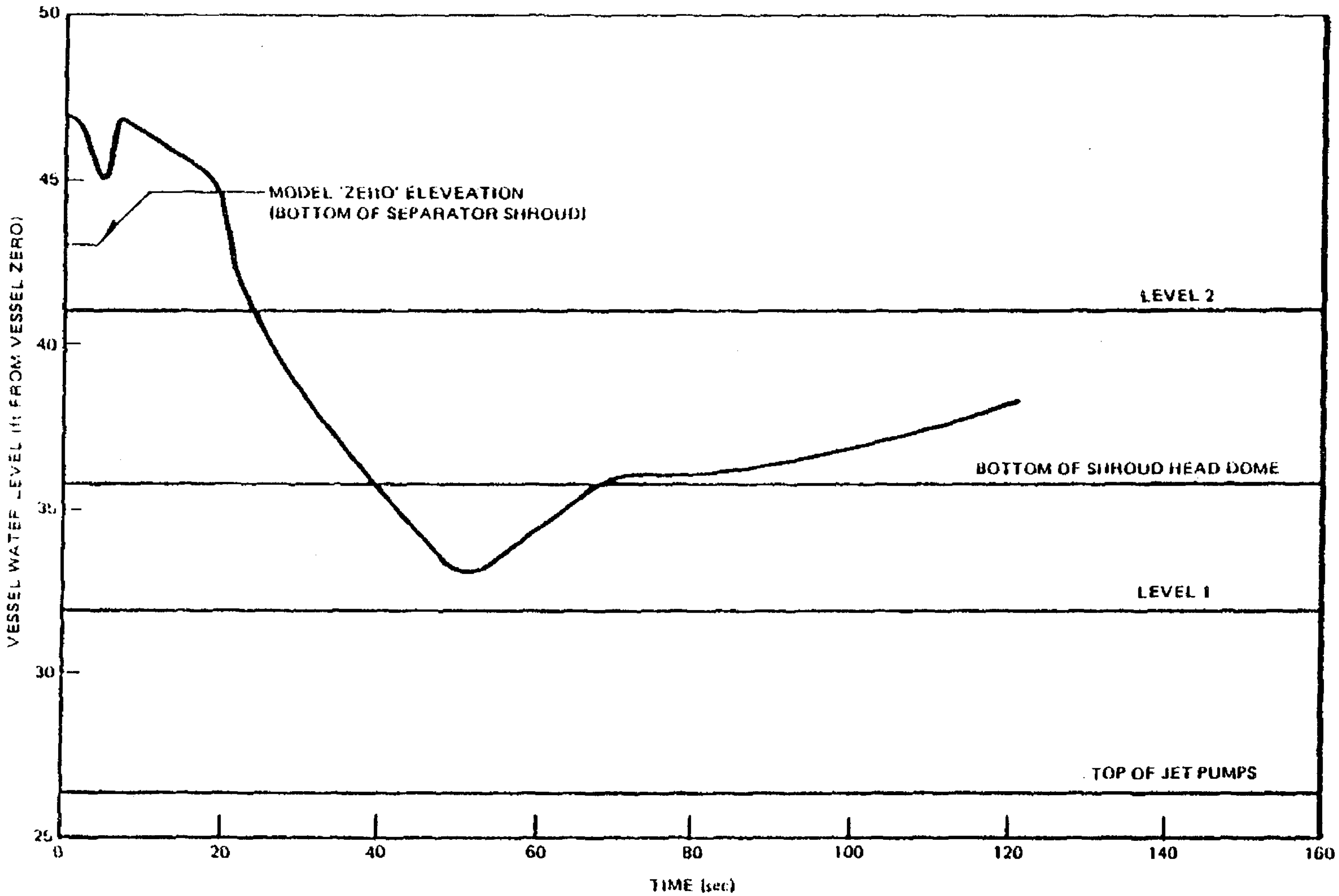

Figure 4.3.3. BWR/6 MSIV Closure with ARI

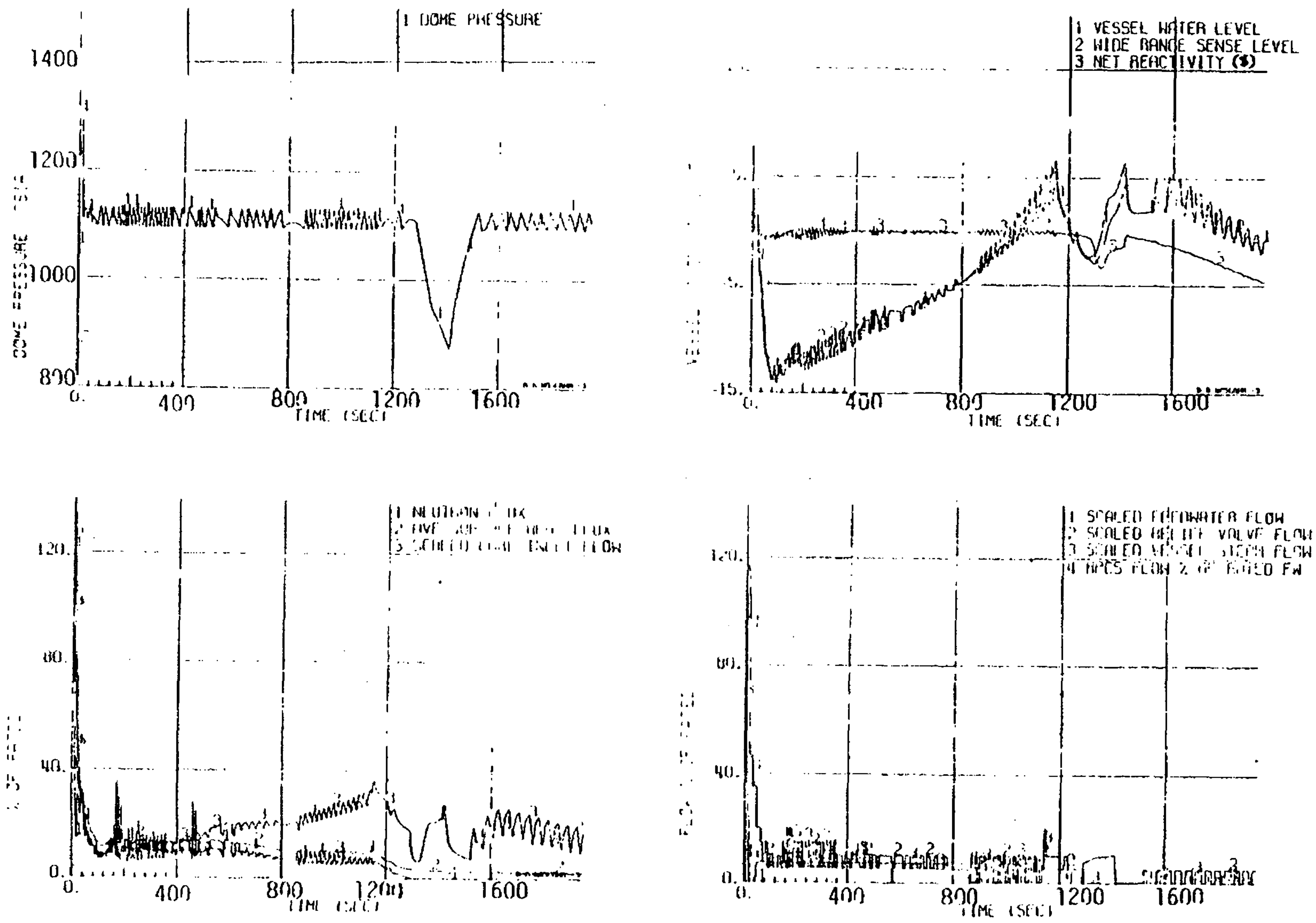

Figure 4.3.4. BWR/6 MSIV Closure, Without ARI, 2 min - 86 gpm Boron, 75% Mixing Efficiency

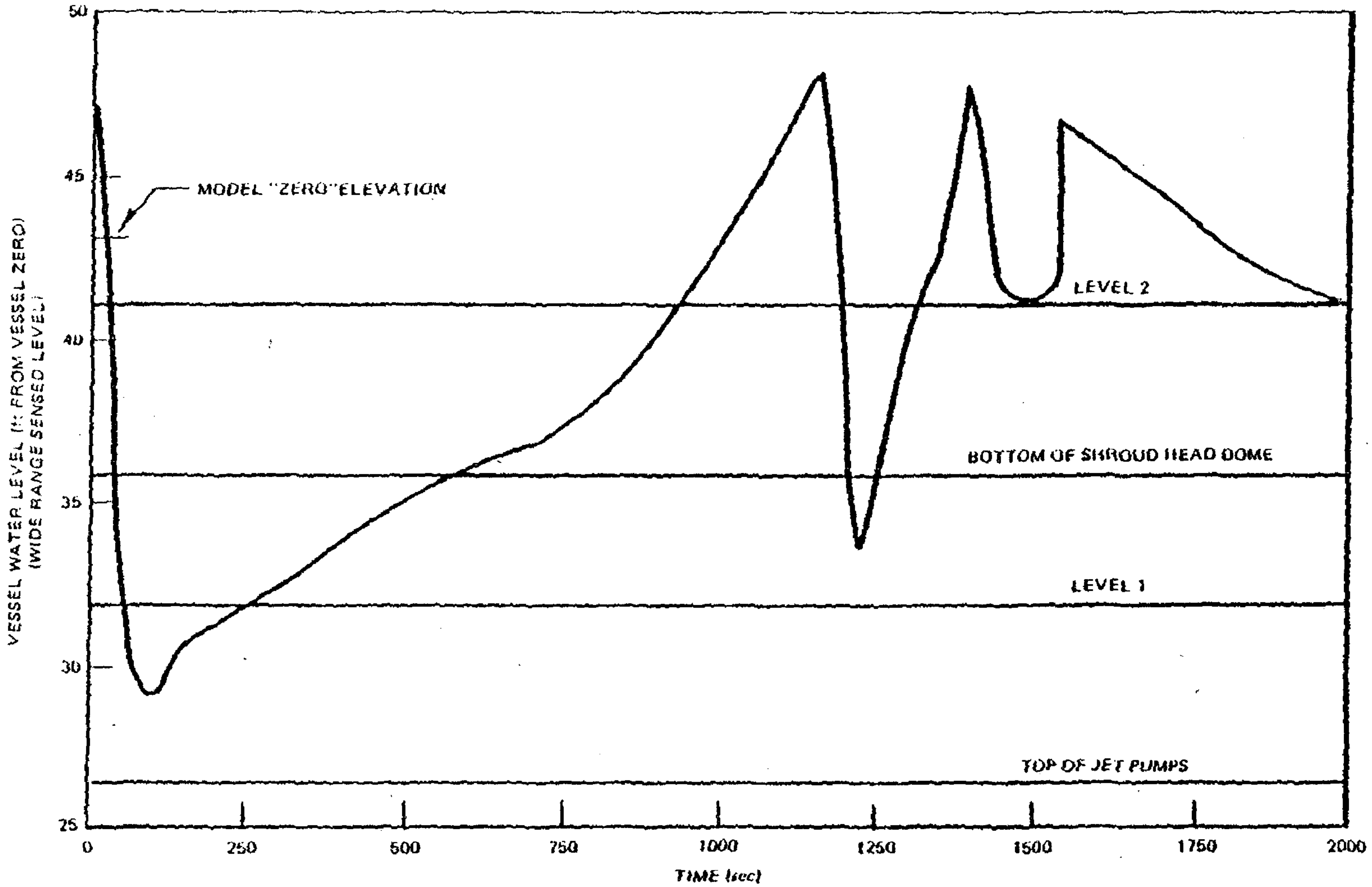

Figure 4.3.5. BWR/6 MSIV Closure with 2 min Boron Injection Without ARI

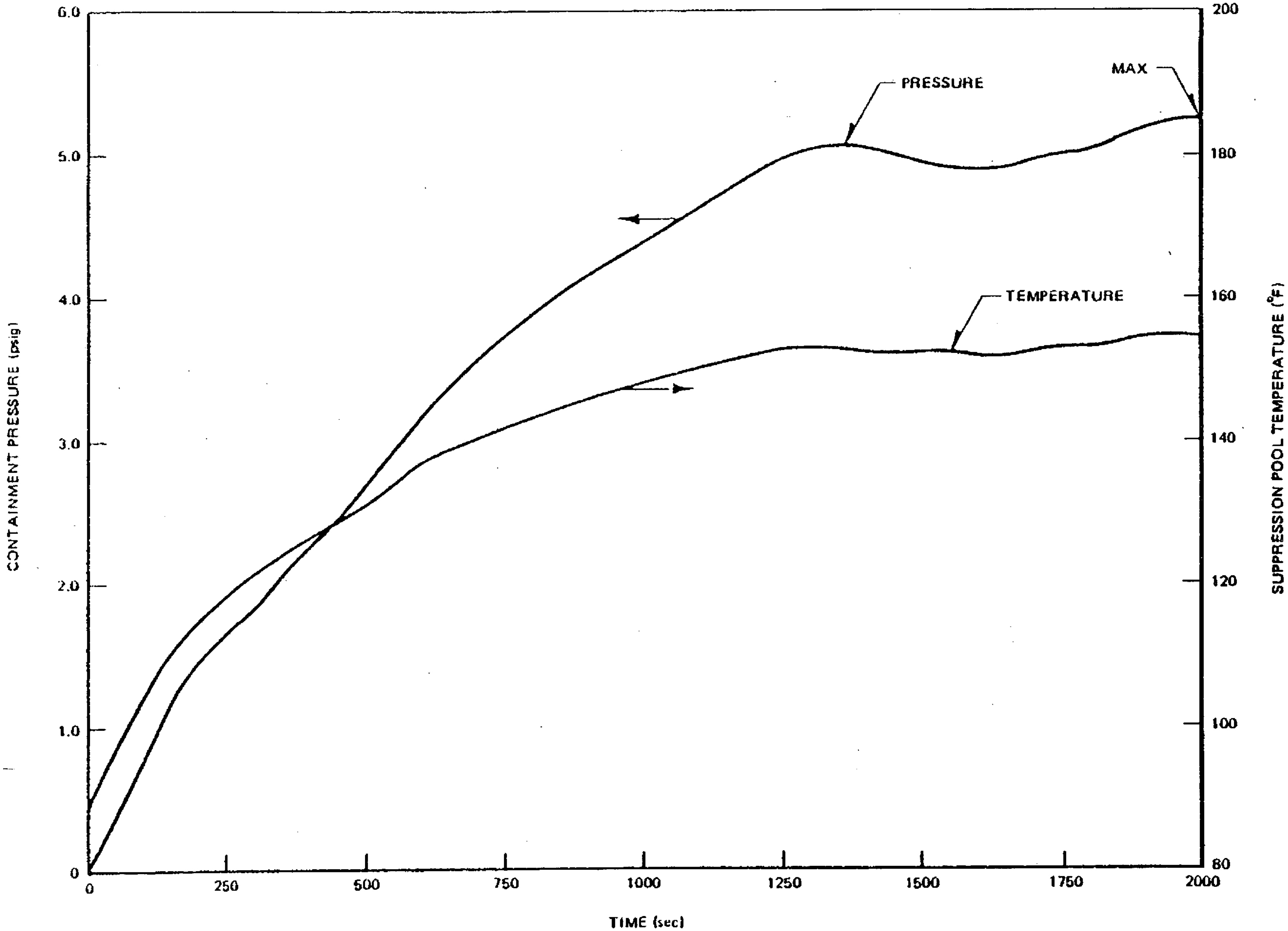

Figure 4.3.6 BWR/6 MSIV Closure with 2 min Boron Injection without ARI

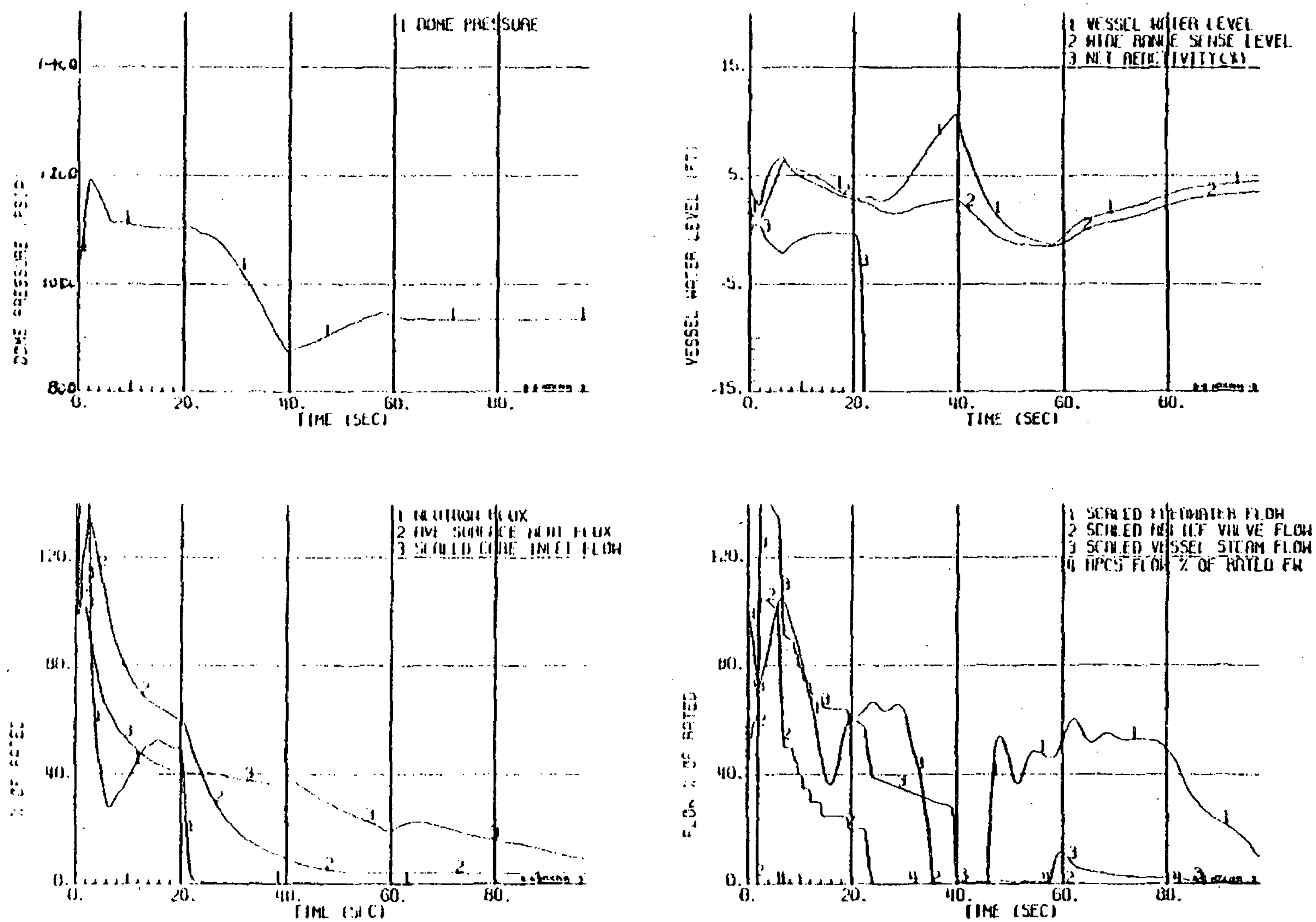

Figure 4.3.7. BWR/6 Turbine Trip, ARI at 20 sec

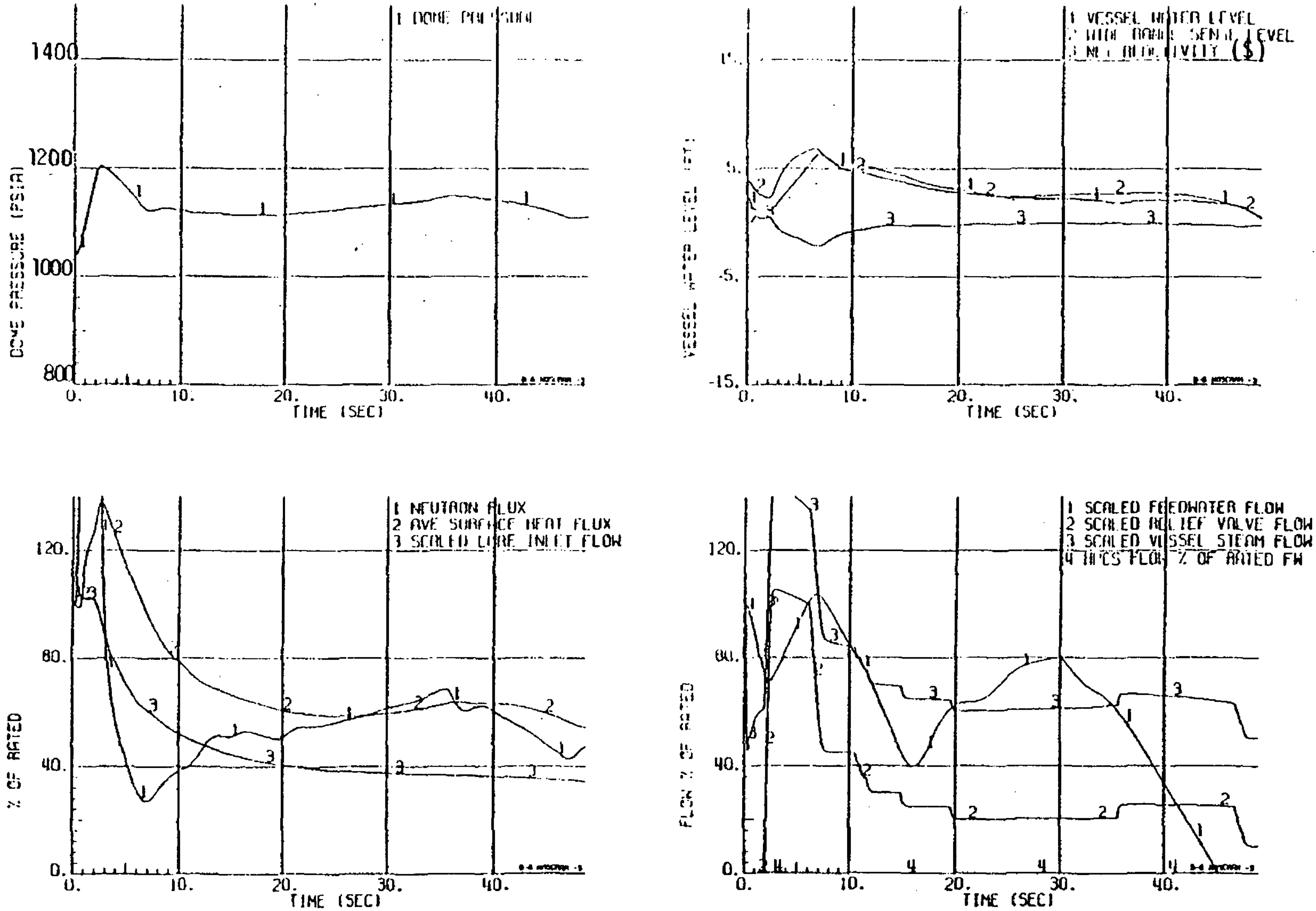

Figure 4.3.8. BWR/6 Turbine Trip, Without ARI, 2 min - 86 gpm Boron

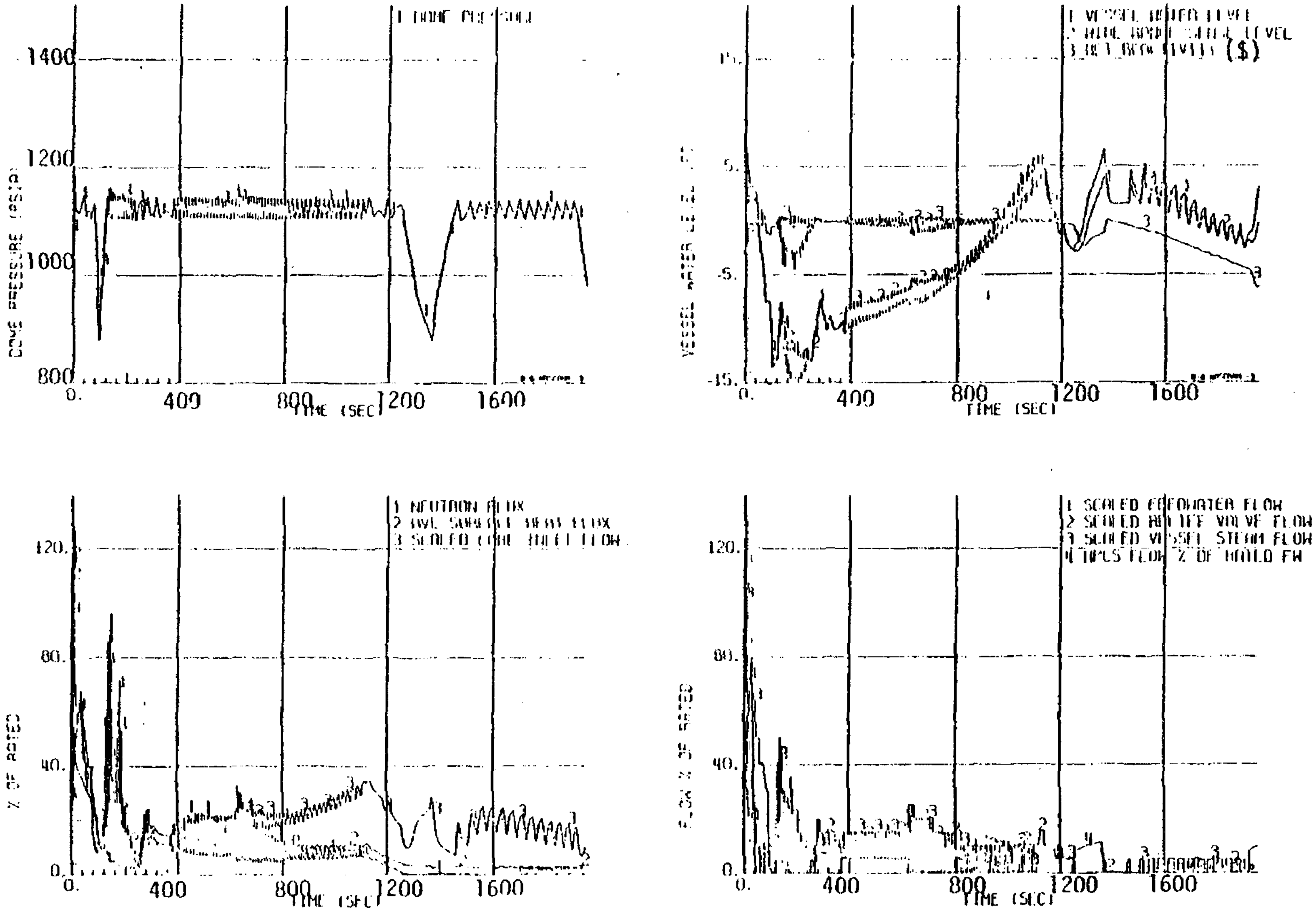

Figure 4.3.9. BWR/6 Turbine Trip, Without ARI, 2 min - 86 gpm Boron

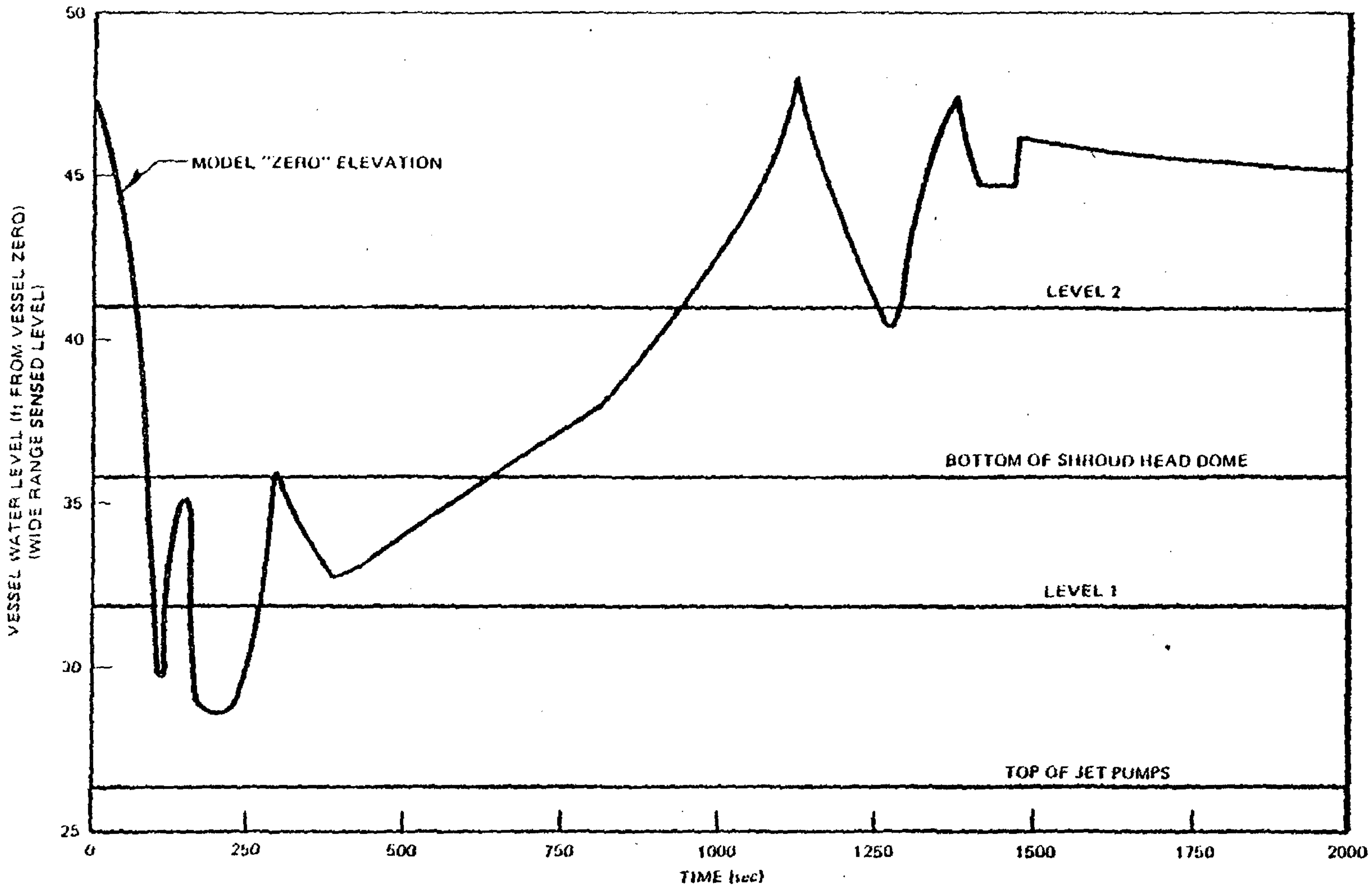

Figure 4.3.10. BWR/6 Turbine Trip with 2 min Boron Injection without ARI

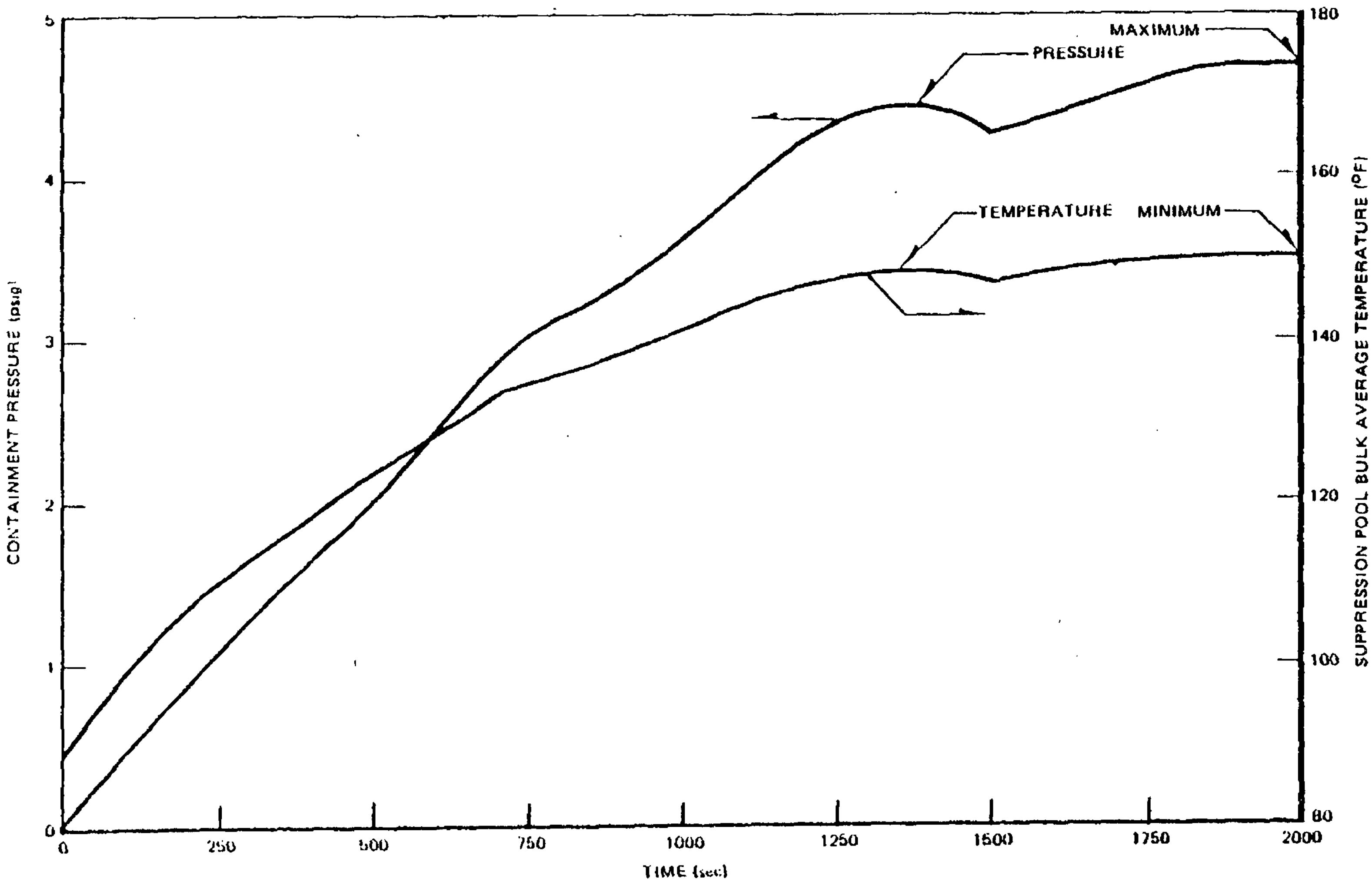

Figure 4.3.11. BWR/6 Turbine Trip with 2 min Boron Injection without ARI

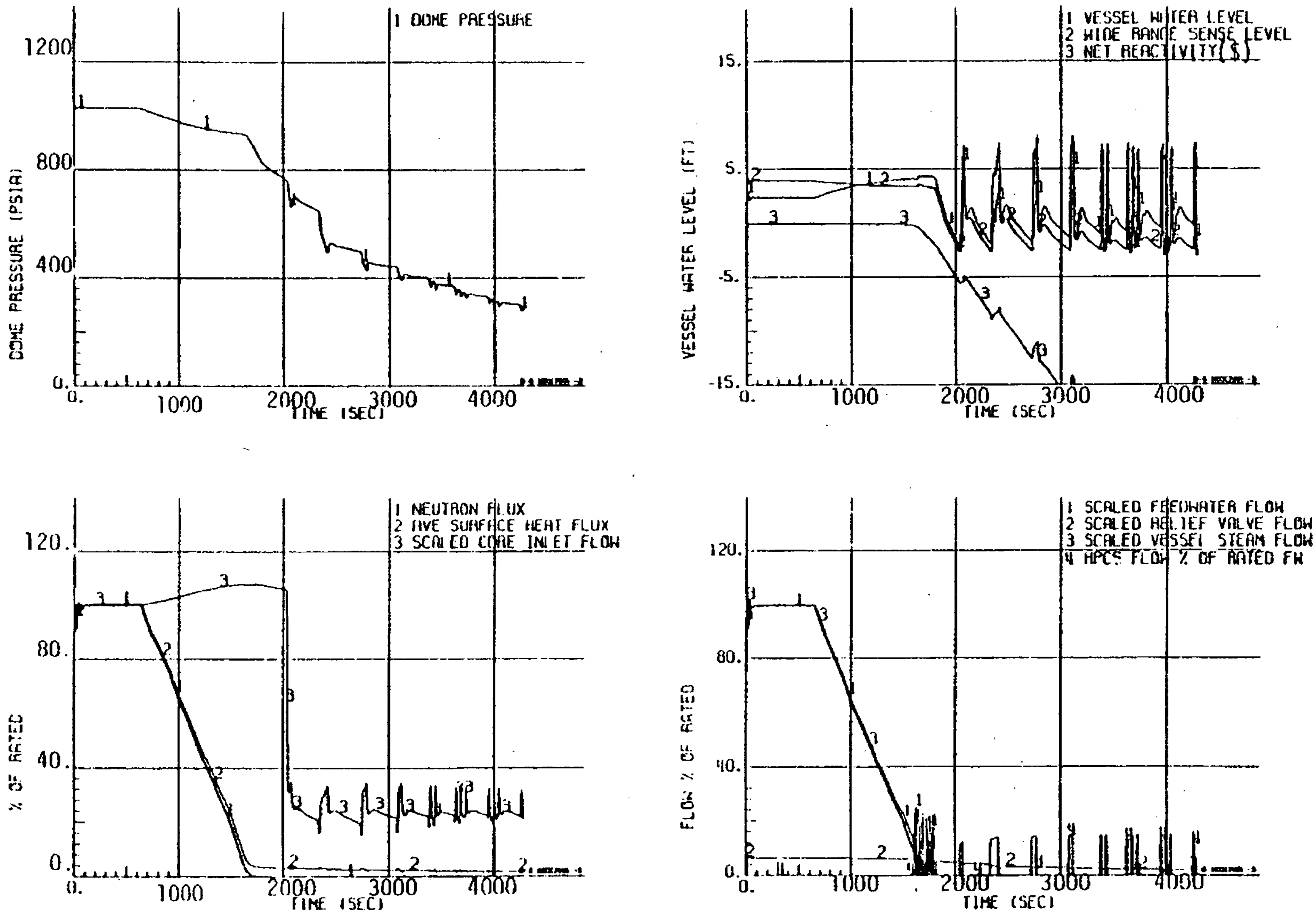

Figure 4.3.12. BWR/6 IORV, Manual SLCS After 10 min, Without ARI

4.4 TRANSIENT MODEL COMPARISON

[1]Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

[2]Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, October 1978 (NEDO-24154).

Table 4.4.1
COMPARISON OF PEAK VALUES FROM
REDY/ODYN CASES
(BWR/6 238/748)

Figure 4.4.1. BWR/6 REDY - MSIV Closure, 2 min - 86 gpm Boron, Without ARI

Figure 4.4.2. BWR/6 ODYN - MSIV Closure, 2 min - 86 gpm Boron, Without ARI

Figure 4.4.3. BWR/6 REDY - Turbine Trip, Without ARI

Figure 4.4.4. BWR/6 ODYN - Turbine Trip, Without ARI

4.5 TURBINE GENERATOR TRIP WITH BYPASS FAILURE

In response to NRC Staff questions, this improbable case has also been studied for ATWS conditions. The frequency of occurrence of this event is well below once per plant lifetime and efforts are under way to reclassify it consistent with this experience. However, for information purposes, one case is included here.

Figure 4.5.1 shows the detailed traces of the most important variables during the initial portion of this event as simulated for a BWR/6 plant (using REDY). The long term portion of this event is essentially the same as the MSIV event. Peak values are:

Neutron Flux:	860%
Average Fuel Surface Heat Flux:	139%
Dome Pressure:	1236 psig
Pressure at Bottom of Vessel:	1262 psig

This case was analyzed with the same input values as the BWR/6 238/748 analyses provided above. Although the sharp neutron flux peak exceeds the MSIV closure case, the integrated power peak is less as shown by the lower heat flux and peak pressures. No unusual anomalies are seen when this unlikely event is compared to the other pressurization cases. Characteristics of the turbine trip without bypass event beyond 20 seconds follows the MSIV event very closely.

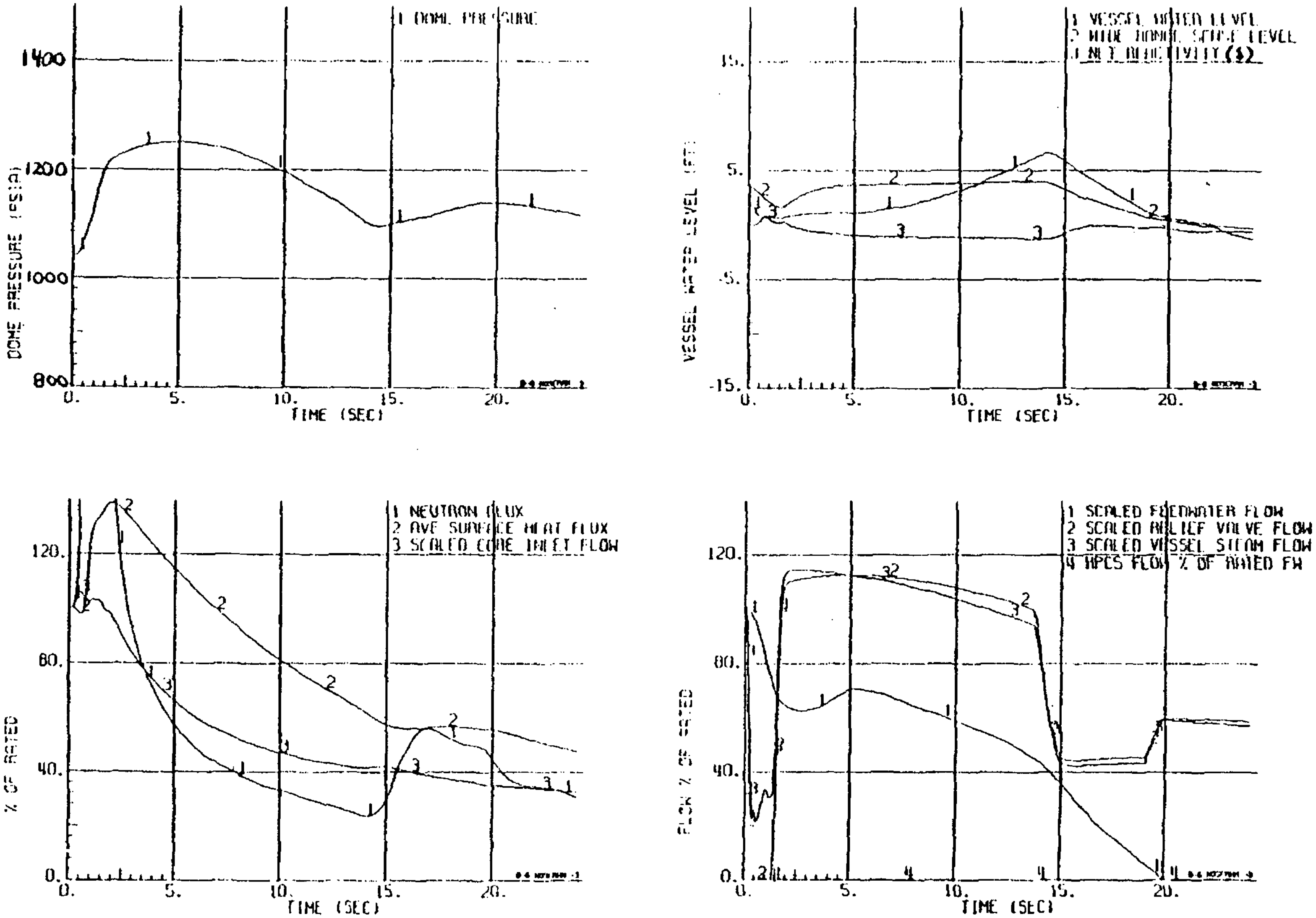

Figure 4.5.1. BWR/6 - Turbine Trip without Bypass, 2 min - 86 gpm Boron

5. ACCEPTANCE LIMITS AND CONFORMANCE

5.1 PRIMARY SYSTEM INTEGRITY

RPT primary system peak pressures are well under the emergency limit of 1500 psi for all events analyzed.

5.2 CONTAINMENT INTEGRITY

Postulated ATWS events subject the containment structure to static and dynamic loads which are less than those for which the containment has been designed. This section of the report shows that containment structural integrity is maintained by comparison with existing design loads.

5.2.1 Static Pressure and Temperature

For the static pressure and temperature loads, ATWS event consequences are much less severe than the results of loss-of-coolant accidents which form part of the design basis for containments as documented in Section 6.2 of Safety Analysis Reports. The table below shows that, for all containment types, the pressures and temperatures due to ATWS events are within the capability of the containment.

	Peak Pressure (psig)		Bulk Pool Temperature (°F)	
Containment	ATWS	Design Basis	ATWS	Design Basis
Mark I (BWR/4)	11	56	189	281
Mark II (BWR/5)	10	45	185	220
Mark III (BWR/6)	7	15	170	185

5.2.2 **S/RV Air Clearing Loads**

5.4 FUEL INTEGRITY

A fundamental assumption in the NRC guidelines in NUREG-0460 is that the occurrence of some fuel perforations during an ATWS event is acceptable providing: a) the extent/number of perforations does not cause unacceptable radiological consequences; and, b) the resulting fuel condition does not preclude coolability and ultimate safe shutdown. The safety condition is assured through the application of the fuel damage criteria of 10CFR50 Appendix K, which are used to assure coolable geometry during a loss-of-coolant accident (LOCA).

5.4.1 Fuel Integrity Criteria

The fuel integrity criteria are those used to assure maintenance of a coolable geometry. These criteria must not be violated by an ATWS event. The specific criteria are: 1) the maximum peak cladding temperature must not exceed 2200°F; and 2) the maximum local cladding oxidation must not exceed 17%.

As in a LOCA event, satisfaction of these criteria will assure maintenance of a coolable geometry in the fuel.

5.4.2 Results of Fuel Integrity Evaluations

5.4.2.1 Peak Cladding Temperature

The peak calculated cladding temperature for all ATWS events analyzed was significantly below the 2200°F requirement from 10CFR50 Appendix K.

5.4.2.2 Localized Cladding Oxidation

The maximum calculated local cladding oxidation for the ATWS events analyzed was found to be significantly below the accepted maximum value (17% of cladding volume). Therefore, no effects due to oxidation are expected to occur.

5.4.3 Conclusions

The foregoing sections indicate that there is substantial margin with respect to assuring coolability of the core and safe reactor shutdown. Perforations assumed to occur in fuel which experiences boiling transition results in relatively small radiological releases (Section 4.5). General Electric therefore concludes that the ATWS rules and requirements specified in NUREG-0460 (Volume 3, Appendix IV) can be fulfilled under the most severe ATWS events.

5.5 RADIOLOGICAL ANALYSIS

The radiological analysis prepared for this submittal has considered the BWR/4-6 reactors and the Mark I-III containments. While there are significant differences for these plants, only one analysis for each ATWS event has been performed and is shown herein to be representative of the consequences for all three containment types and product lines. A schematic of the containment system and fission product transport pathways used in this analysis is shown in Figure 5-5.1. One of the primary differences between the Mark I-II containments and the Mark III containment is the "open" suppression pool for Mark III. There will be a short period of time prior to containment isolation when fission products may be released directly to the environment from the Mark III containment. Because of the "closed" suppression pool for the Mark I-II containments this pathway does not exist for these designs, therefore, the radiological consequences included herein will overestimate the consequences for these two product lines. For the Mark III and for some Mark I-II containments forced mixing of the air within the secondary containment is provided. However, this is not a universal feature, therefore one of the assumptions applied to the "conservative assessment" is that zero mixing occurs within secondary containment. The meteorological conditions bound all BWR sites licensed to date.

Two analytical evaluations, which include two cases each, are presented in subsequent sections. These two evaluations are arbitrarily defined as "Realistic Assessment" and "Conservative Assessment."

5.5.1 Assumptions/Conditions of Analysis

The assumptions or conditions considered appropriate for the evaluation of the radiological calculations are presented in Tables 5.5.1 and 5.5.2. Parametric values in the conservative columns in these tables are consistent with the guidance offered in Reference 5.4.4. Where guidance was lacking, assumptions were made which are consistent with previous BWR licensing practice.

Parametric values in the realistic columns are consistent with experimental data obtained from operating BWRs or which are considered conservative if operating data is lacking. For the zero perforation case (case 1) the assumption is made that the activity released from defective fuel rods is proportional to the negative change in reactor vessel pressure.

5.5.2 ATWS Events Evaluated

Radiological evaluations have been performed for three ATWS events. For each of these events, except the IORV case, the two cases in Table 5.5.1 were evaluated (i.e., zero fuel perforations and 17% fuel preforations). For the IORV case only the zero perforation case (case 1) was evaluated because no rods go into boiling transition. The events evaluated were

a. turbine trip with bypass (TTWB),

b. inadvertent opening of a safety relief valve (IORV) and

c. main steam line isolation valve closure (MSIV).

For blowdown to an "open" suppression pool, such as for a BWR/6-Mark III containment, isolation of the containment ventilation occurs at the following times:

Event	Signal Initiating Closure	Isolation Signal Occurs At (Sec)	Containment Isolated (Sec)* (Relative to 0 Time)
TTWB	High Drywell Pressure	333	338
IORV	High Drywell Pressure	400	405
MSIV	Low Water Level Level (2)	23	28

*Includes allowance of 5 seconds to close ventilation valves.

5.5.2.1 Turbine Trip with Bypass (TTWB)

An examination of the pressure in the reactor pressure vessel for a BWR/6 shows that the pressure initially goes to about 1200 psi, drops in pressure (due to SRV opening) to about 1113 psia, and cycles between this value and 1150 psi for approximately 570 seconds, and then cycles at lower pressure fluctuations from that time on. As noted previously, the fission product release for defective fuel rods is assumed to be proportional to the negative change in reactor pressure. Therefore in approximately 900 seconds the fuel is assumed to have experienced a pressure reduction equivalent to 1050 psia. For purposes of evaluation the total "spiking" activity in Table 5.5.2 for case 1 is, therefore assumed to be released uniformly to the primary coolant over a 900 second time period. As an example, the release rate of I-131, for the conservative analysis, is assumed to be 16,000/900 = 18 Ci/sec. This release rate occurs for a time period of 900 seconds after which no additional release occurs to the RPV. The activity in the primary coolant is determined as follows:

$$dN_1/dt = - N_1(\lambda + L_1) + S_1 \qquad (1)$$

$$N_1 = \frac{S_1}{\lambda + L_1}\left(1 - e^{-(\lambda + L_1)\,t}\right) \quad (Ci) \qquad (2)$$

where t in Equation 2 is valid between 0 and 900 seconds. For $t > 900$ seconds the activity in the primary coolant is determined as follows:

$$dN_1/dt = -(\lambda + L_1)N_1$$

$$N_1 = \frac{S_1}{\lambda + L_1}\left(1 - e^{-(\lambda + L_1)\,900}\right)\left(e^{-(\lambda + L_1)\,(t - 900)}\right) \qquad (3)$$

where:

λ = radioactive decay constant (sec^{-1})

L_1 = release rate from the RPV and is determined as follows:

$$L_{NG} = \frac{\text{Steam blowdown rate (\#/sec)}}{\text{Mass of Steam in Steam Dome (\#)}}$$

$$L_1 = \frac{\text{Steam blowdown rate (\#/sec)}}{\text{Mass of Liquid in Air Coolant (\#) (0.02)}}$$

S_1 = source input from defective fuel (Ci/sec)

The 0.02 factor in L_1 is the iodine carryover fraction in steam (i.e., each pound of steam contains 2% of the activity contained in a pound of primary coolant).

The activity being discharged to the condenser or suppression pool is determined by multiplying Equations 2 or 3 by the appropriate value of L, which takes into consideration the type of activity being evaluated and the actual steam blowdown rate to these two areas.

The pressure transient was also evaluated for a BWR/4 and BWR/5. It was determined that it took 1400 seconds to have an integrated change in RPV pressure, due to SRV cycling, of 1031 psi, therefore, the pressure trace for the BWR/6 will be used to evaluate the consequences of this event for all classes of BWRs.

The approximate steam partitioning, as a function of time, from the RPV and between the suppression pool and the condenser is as follows:

Time	Percent of Decay Steam Flow to	
(sec)	Suppression Pool	Condenser
0-300	30	70
300-600	20	80
>600	0	100

For case 2, where 17% clad failure is assumed to occur, it is conservatively assumed that failure occurs at t = 0. The activity in the primary coolant is, therefore, defined as follows:

$$dN_1/dt = -(\lambda + L_1)\ N_1 \tag{4}$$

$$N_1 = N_0 e^{-(\lambda + L_1)t}$$

Where Equation 4 is valid for the time periods of interest and the variable is as defined previously.

5.5.2.2 <u>Inadvertent Opening of the Safety Relief Valve (IORV)</u>

This event has been qualitatively evaluated for the BWR 4-6 plants and is found to result in approximately the same depressurization rate and volumetric steaming fractions for each plant. Therefore the radiological consequences are presented for only the BWR/6 plant. An examination of the RPV depressurization rate shows a ΔP/sec of $\sim$0.2 psi/second. For case 1 it is assumed that the fission product release rate from the normally defective fuel rods is at a constant rate for 6000 seconds. Equations 1 through 3 in Section 5.5.2.1 are therefore appropriate for this evaluation with the 900 second value being replaced by 6000 seconds.

The approximate steam partitioning, as a function of time, from the RPV and between the suppression pool and condenser is as follows:

Time (sec)	Percent of Decay Steam Flow to Suppression Pool	Condenser
0-800	6	94
800-1000	10	90
1000-1400	20	80
1400-1500	30	70
1500-1600	60	40
>1600	100	0

5.5.2.3 Main Steam Isolation Valve Closure (MSIV)

As for the other two transient events, this event has also been qualitatively examined with the conclusion being that from a radiologicial viewpoint the BWR/6 transient will bound the consequences associated with a BWR/4 or BWR/5. For the BWR/6 after the initial 200 psi depressurization the RPV depressurization rate is approximately 2 psi/sec, therefore approximately 400 seconds will be required to have an equivalent 1030 psi pressure reduction. Equations 2 and 3 from Section 5.5.2.1 are appropriate for this transient with the 900 seconds replaced by 400 seconds.

For case 2, where 17% clad failure is assumed to occur, it is conservatively assumed that failure occurs at t = 0 and Equation 4 of Section 5.5.2.1 is appropriate for the time periods of interest. For this event, 100% of the decay steam generation rate is discharged to the suppression pool.

5.5.2.4 Fission Product Release to the Environment

The fission product activity released to the environment is dependent upon the release pathway from the reactor vessel and the reduction factors and compartmental leakage rates between the RPV and the environment. The activity airborne in the compartments of concern is defined by the following differential equations.

a) Activity in Condenser

$$dN_c/dt = -(\lambda + L_6)\, N_c + L_4\, N_1\, DF \qquad (5)$$

b) Activity in Primary Containment

$$dN_{pc}/dt = -(\lambda + L_3 + L_5)\, N_{pc} + L_2\, N_1\, DF$$

c) Activity in Secondary Containment

$$dN_{sc}/dt = -(\lambda + L_1)\, N_{sc} + L_c\, N_{pc}$$

where the above parameters are schematically defined in Figure 5.5.1.

5.5.2.5 Radiological Consequences

Based upon the preceeding discussion, the following radiological exposures are calculated for the three ATWS events of concern. These consequences can be compared to the guidelines in 10 CFR 100 which are 25 Rem whole body and 300 Rem thyroid inhalation. It can be seen that even if 100% of the rods were to perforate, the guidelines would not be exceeded.

Event	Radiological Consequence (Rem)							
	Site Boundary				Low Population Zone			
	Whole Body		Inhalation		Whole Body		Inhalation	
	Real.	Cons.	Real.	Cons.	Real.	Cons.	Real.	Cons.
(a) TTWB								
• Case 1	4.7-5[a]	2.6-1	5.8-5	8.2-2	7.3-6	3.8-2	6.6-4	1.4-1
• Case 2	1.1-1	6.7-1	1.7-2	1.2+0	4.9-3	1.1-1	1.3-1	2.0+0
(b) IORV								
• Case 1	3.3-6	3.0-3	4.3-6	6.0-3	1.2-6	7.8-4	1.3-5	3.6-3
(c) MSIV								
• Case 1	2.1-5	8.8-2	2.2-8	7.6-4	9.8-6	1.8-2	5.1-6	2.8-3
• Case 2	3.2-2	2.8-1	7.7-5	1.4-2	2.0-3	4.7-2	9.9-4	5.1-2

*Case 1 - 0% perforations
Case 2 - 17% perforations
(a) 4.7-5 = 4.7×10^{-5} Rem

Real = Realistic
Cons = Conservative

5.5.3 Conclusions

Based on the results presented in Section 5.5.2.5, it can be concluded that the radiological exposure for the ATWS events evaluated are well below the guideline in 10CFR100 for all three BWR product lines (BWR/4, 5 and 6), and for all three BWR containment designs (Mark I, II and III).

Table 5.5.1
ASSUMPTIONS/CONDITIONS UPON WHICH
RADIOLOGICAL ANALYSIS IS BASED

Variable	Parametric Value Assumed Realistic	Parametric Value Assumed Conservative
1. Power (each MWt)	4353	4353
2. Fuel Type	8x8	8x8
3. Fuel Rod Perforations (%)		
• Case 1	0	0
• Case 2	17	17
4. Fission Products Released to Primary Coolant[a]		
• Case 1		
I-131	1900 Ci	1.6×10^4 Ci
Xe-133	10^4 Ci	2×10^5 Ci
• Case 2		
I-131	2% Rod Act	2% Rod Act
Xe-133	2% Rod Act	2% Rod Act
5. Fission Products Released to Suppression Pool or Main Turbine Condenser	Proportional to mass blowdown rate, primary coolant volume, RPV Steam Dome Volume and 2% carryover for Iodine	
6. DF in Suppression Pool/Turbine Condenser[b]		
• Noble Gases	1	1
• Iodine	0.01	0.1
7. Primary Containment leak rate (%/day)	1	1
8. Secondary Containment leak rate (%/day	100	100
9. Condenser leak rate (%/day)	1	1
10. Mixing in Secondary Containment (%)	100	0
11. SGTS Iodine filter efficiency (%)	99	95

(a) See Table 5.5.2

(b) $DF = \frac{\text{What comes out}}{\text{What goes in}}$

Table 5.5.1 (Continued)

Variable	Parametric Value Assumed Realistic	Parametric Value Assumed Conservative
12. Meteorology (X/Q - sec/m^3)[c]		
Site Boundary (0-2 hr)	2.5×10^{-5}	1.8×10^{-4}
Low Population Zone		
0-2 hr	10^{-6}	2.6×10^{-5}
2-8 hr	10^{-6}	1.7×10^{-5}
8-24 hr	10^{-6}	2.6×10^{-6}
24-96 hr	10^{-6}	1.4×10^{-6}
96-720 hr	10^{-6}	5.8×10^{-7}

[c] Realistic meteorology is average annual and conservative meteorology is 10% of the 95% meteorology.

Table 5.5.2
FISSION PRODUCTS RELEASED TO PRIMARY COOLANT

Isotope	Activity Released (Ci)		
	Case 1		Case 2
	0 Perforations		17% Perforations
	Realisitic	Conservative	
I-131	1.9+3[a]	1.6+4[b]	3.7+5
I-132	2.8+3	2.3+5	5.6+5
I-133	4.4+3	1.2+5	8.3+5
I-134	4.8+4	6.6+5	1.7+5
I-135	4.2+3	1.9+5	7.6+5
Kr-83m	7.9+3	7.6+4[c]	6.2+4
Kr-85m	1.9+4	1.4+5	1.9+5
Kr-85	4.3+2	3.3+2	6.1+3
Kr-87	3.7+3	4.5+5	3.5+5
Kr-88	5.4+3	4.5+5	4.7+5
Kr-89	7.0+3	2.7+6	5.9+5
Xe-131m	5.5+1	3.3+2	3.8+3
Xe-133M	2.8+2	6.5+3	2.1+4
Xe-133	1.0+4	2.0+5	8.3+5
Xe-135m	1.6+3	5.9+5	2.3+5
Xe-135	9.5+3	4.9+5	7.9+5
Xe-137	9.2+3	3.3+6	7.6+5
Xe-138	9.4+3	2.0+6	7.1+5

(a) 1.9+3 = 1.9 x 10^3 curies

(b) I-131 release based on a release rate from the fuel for 4 hours equal to 250 times the tech spec value, where the tech spec equals 0.2 μCi/gm dose equivalent I-131.

(c) Noble gas values based on 1600 times normal offgas integrated over a 4-hour period.

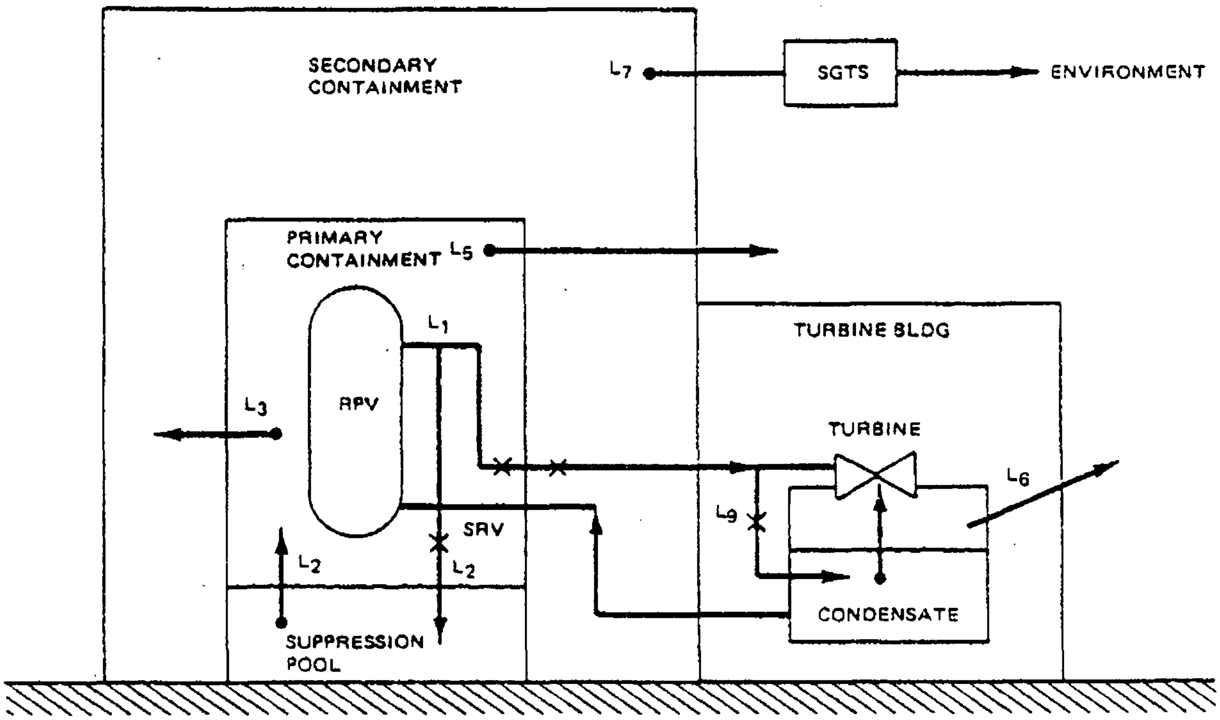

Figure 5.5.1. Fission Product Containment and Leakage Pathways

5.6 EQUIPMENT ENVIRONMENTAL QUALIFICATION

Equipment relied upon to mitigate the ATWS transient must be capable fo performing its required function under expected ATWS environmental conditions. In evaluating equipment capability, consideration is given to the relative timing of the equipment function and the development of various environmental conditions. In this way, not all equipment has to be assessed against the most severe environment.

The systems utilized for ATWS mitigation are listed in Section 3.4.1. From this list, the major systems and equipment required to operate under conditions unique to ATWS are as follows:

RPT
HPCI/HPCS
RCIC
Standby Liquid Control System
Suppression Pool and Containment
Residual Heat Removal System
MSIV
S/RV
Control Rod Drive System
Instrumentation:
- APRM (Neutron Flux)
- RPIS (Rod Position)
- Dome Pressure
- Reactor Vessel Water Level
- Condensate Storage Tank Level

In assessing the required capability for the above system and euipment, an ATWS MSIV closure event was selected as being most limiting.

For an MSIV closure event the entire reactor coolant pressure boundary is subject to a maximum pressure below 1500 psig. This maximum pressure peak will occur early in the event (about 25 seconds following initiation) after which the pressure will be reduced to the relief valve set points and remain there until depressurization is initiated.

Depressurization is not expected to occur before approximately 4 hours into the event. The temperature will be at saturation for the given pressure which is approximately 600°F for 1500 psig. The Standby Liquid Control System will have emptied the storage tank into the vessel within 2 hours and the reactor water will contain sodium pentaborate at a maximum concentration of approximately 0.1% by weight.

The capability of the primary system to withstand these ATWS pressurization conditions is addressed in Section 5.1.

It is required that the recirculation pump trip breakers operate properly. Since this happens at approximately 5 seconds into the event, the environment and duty for this function will be at normal conditions.

As a result of the discharge through the S/RV's the suppression pool temperature will increase which in turn will cause a general heating and pressurization of the containment. These maximum conditions are addressed in Section 5.2 for the three representative reactor types. These conditions are repeated below:

Reactor/Containment	Peak ATWS Pressure	Peak ATWS Bulk Pool Temperature
BWR/4 - Mark I	11 psig	189°F
BWR/5 - Mark II	10 psig	185°F
BWR/6 - Mark III	7 psig	170°F

The capability of the Suppression Pool and containment to withstand these ATWS pressure and temperature conditions is addressed in Section 5.2. The above pressures and temperatures also represent the maximum conditions to which systems and equipment required to function thoughout the ATWS event will be subjected. The operating conditions expected to be imposed on each system and their conformance are discussed in the following sections.

5.6.1 Operating Conditions

5.6.1.1 HPCI/HPCS and RCIC

For the MSIV closure case, these are the systems which will be relied upon to supply water inventory to the reactor. It is not mandatory that the RCIC be available if the HPCI or HPCS is operating. The RCIC by itself cannot maintain inventory until hot shutdown is attained, which is approximately 28 minutes after the event begins. After that, any of these systems, can supply, sufficient inventory. The HPCI/HPCS and RCIC pumps and valves are capable of operating under containment design conditions which bound the ATWS conditions (Section 5.2). Moisture and steam going to the HPCI or RCIC turbine will contain a maximum of 0.01% sodium pentaborate by weight. This concentration is not known to present any equipment or material problems.

Although ATWS peak pressures are higher than the RCIC/HPCI/HPCS equipment operating pressures, these occur at the earliest stages of the transient. The RCIC/HPCI/HPCS systems are not initated until the return to near normal pressure conditions.

5.6.1.2 Standby Liquid Control System

Those parts of this system which are part of the reactor coolant pressure boundary will withstand the peak conditions given above. At two minutes the valves must open and the pumps operate for approximately 50 minutes. In a Mark III containment, this system is located within the containment and is designed to operate through maximum Mark III containment conditions (15 psig/185°F). The temperature of the suppression pool at the 2 minute actuation time, when the explosive valves must open will be approxiamtely 110°F. The reactor water cleanup system isolation valves are designed to close on actuation of the Standby Liquid Control System to prevent dilution and cleanup of the boron which was pumped into the vessel. In the case of the Mark I and Mark II containment designs the SLCS is located outside the containment and will not see any abnormal ATWS environmental conditions.

5.6.1.3 Residual Heat Removal System

Because of the isolation, all of the energy generated during this ATWS event is deposited in the suppression pool. The RHR system is designed to remove this energy and transfer it to the service water. Both loops of the RHR will be needed in pool cooling mode until it is possible to change to steam condensing mode (for those plants which have it) and then later to shutdown cooling mode. The suppression pool temperature may be as high as 189°F in Mark I containments. The RHR system pumps and valves are capable of operating under containment design conditions which bound ATWS conditions (see Section 5.2). While the steam condensing mode is in use, the moisture and steam will contain no more than 0.01% sodium pentaborate by weight. When the RHR system is in the shutdown cooling mode, the reactor water will contain approximately 0.08% sodium pentaborate by weight. This concentration is not known to present any equipment or material problems.

5.6.1.4 Standby Gas Treatment System

The Standby Gas Treatment System is designed to function during the LOCA event. The LOCA gives a much more severe environment than ATWS.

5.6.1.5 Main Steamline Isolation Valves

All MSIV's are judged acceptable when subjected to ATWS conditions. Each MSIV is production tested prior to release for shipping from the valve manufacturer's shop. This testing follows a sequence of hydrostatic test on the valve in the open position, hydrostatic test on the seat in the closed position, main seat leakage test, and cyclic test. Thus, operation of the valve is verified from both the open and closed position following the hydrostatic test. The hydrostatic test of the valve is in the 2175 to 2400 psi range, while the seat test has been in the 1450 psi to 2400 psi range. Those tested only to 1450 psi can be justified for use at higher pressures based on their similarity to the valves actually tested to the higher pressures.

5.6.1.6 Safety/Relief Valves

The S/R valves are required to open at their set point during the initial pressure rise. They must stay open through the peak pressure and temperature and then reclose at their closure pressure set point. Several of the valves will be required to go through multiple open/close cycles. The steam passing through the valve will be saturated as previously described while the environment external to the valve will be nearly the same as that for the containment which in a BWR/4 Mark I may be 189°F and 11 psig. Moisture and steam will contain sodium pentaborate to a maximum concentration of 0.01% by weight. The external environment is not any more severe than the design conditions for which these valves must function. The sodium pentaborate concentration is not known to present any equipment or material problems.

5.6.1.7 Control Rod Drive System

The design pressure of the CRDs is 1250 psig. The ASME code allows occasional pressures 20% greater than the design pressure (i.e., 1500 psig). The rest of the CRD system which would be subjected to the postulated 1500 psig have a design pressure of 1750 psig (2000 psig for BWR/6).

The CRDs will be capable of scram even at 1500 psig vessel pressure. However, the time required to complete scram may be increased slightly since the ΔP between the scram accumulator pressure and the RPV pressure will be reduced.

The 0.1% (by weight) concentration of sodium pentaborate in the RPV should not affect the scram function. Therefore, based on this preliminary investigation the CRD system could be qualified to operate during and follow the postulated ATWS event.

5.6.1.8 Instrumentation

The neutron flux and rod position indications are required during the first few minutes of the ATWS event. Their environmental capabilities are shown in Table 5.6-1. Dome pressure is required in the first few seconds of the ATWS event and is not exposed to any temperature or pressure transients during this time. The reactor vessel water level and CST level required throughout the ATWS event and their environmental capabilities shown in Table 5.6-1.

5.6.2 OBE Requirements

With the exception of the Condensate Storage and Supply System,all other systems have been designed to meet seismic requirements as indicated in the system design descriptions.

The condensate storage and supply system will probably withstand the results of the OBE level seismic event even though the analysis has not been done to verify it. This equipment is usually designed to at least ASME Section III Class III requirements. An available operating data point is the June 12, 1978 Japanese earthquake near the Fukushima site. This earthquake resulted in 0.13g free field ground acceleration at the site and no damage was observed. These two plants continued to operate through the event.

The Japanese experience serves as a useful demonstration of ability of systems to withstand the OBE, since the 0.13g ground acceleration is near or above the OBE level for power plants in service today.

Fianlly, the condensate storage tank is not strictly needed to accommodate the ATWS event. The HPCS, HPCI and RCIC can take suction from the suppression pool upon failure of the condensate storage tank.

5.6.3 Equipment Environmental Qualification

All equipment required to function during an ATWS event have been examined for the environment in which they need to operate. The primary parameters of concern are temperature and pressure in addition to chemistry discussed above. Those systems located inside the containment building will be exposed to suppression pool temperature levels and would be most critically affected.

Because most safety type components within the containment building are qualified to operate in the LOCA environment, they are easily capable of taking the ATWS environment. The following table summarizes the qualification of the critical ATWS components.

Table 5.6-1
ATWS EQUIPMENT QUALIFICATION

Equipment Required for ATWS	Containment Type Mark I	Mark II	Mark III
Reactor Coolant Pressure Boundary	1500 psi*	1500 psi*	1500 psi*
Recirculation Pump Trip Breakers and ARI	(Actuate before Abnormal Conditions Occur)	(Actuate before Abnormal Conditions Occur)	(Actuate before Abnormal Conditions Occur)
Safety/Relief Valves (External Environment)	340°F 56 psi	340°F 45 psi	330°F 15 psig
Control Rod System (Function with 1500 psi Vessel Pressure?)	Yes	Yes	Yes
Standby Liquid Control System and Pumps	Outside Containment	Outside Containment	185°F 15 psig
Reactor Water Cleanup System Isolation Valves	340°F 56 psig	340°F 45 psig	330°F 15 psig
HPCS/I Valves	340°F 56 psig	340°F 45 psig	330°F 15 psig
Condensate Storage Tank and Level Indicators/Cabling	Outside Containment Level is Class 1E	Outside Containment Level is 1E	Outside Containment Level is Class 1E
Feedwater Control System	Outside Containment	Oustide Containment	Oustide Containment
Main Steamline Isolation Valves	340°F 56 psig	340°F 45 psig	330°F 15 psig
RHR Injection Valves and Check Valves	340°F 56 psig	340°F 45 psig	330°F 15 psig
Vessel Pressure and Level Transmitters and Cabling	Transmitters Oustide of Containment	Transmitters Oustide of Containment	185°F, 15 psig (In Containment Outside of Drywell)
Neutron Flux Sensors and Cabling	230°F Continuous**	230°F Continuous**	230°F Continuous**

*Service Level C
**Qualified for greater than 340°F for short periods of time

Table 5.6-1 (Continued)
ATWS EQUIPMENT QUALIFICATION (Continued)

Equipment Required for ATWS	Mark I	Containment Type Mark II	Mark III
Rod Position Indication System Cabling	230°F Continuous	230°F Continuous	330°F 15 psig 185°F 15 psig To 6 hrs (Containment)
Rod Position Indication System Multiplexer	N/A	N/A	Qualification requirements for ATWS service are currently under review
Bulk Pool Temperature Limit for RHR/HPCS(NPSH)	212°F	212°F	212°F
Temperature for HPCI/RCIC Water Supply	Greater Than 170°F***	Greater Than 170°F***	Greater Than 170°F***

***Preferred source for these systems is ambient CST water. Qualification requirements for ATWS service are currently under review.

6. OTHER ATWS CONSIDERATIONS

6.1 DIVERSITY CONSIDERATIONS

6.1.1 Alternate Rod Insertion

ARI has been presented as a means for improving the reliability of the existing BWR shutdown system. ARI adds improved independence and diversity of hardware in the part of the scram system which is most vulnerable to a common cause failure which could lead to scram failure. A detailed analysis of the common cause failure potential in the existing BWR scram system and ARI is provided in the GE Scram System Reliability Analysis submitted to NRC during 1976.*

The primary diversity provided by ARI is in the use of an "energized-to-trip" circuit versus a "deenergized-to-trip" circuit in the output devices of the current scram system. ARI can be designed to be independent of the circuitry of the current scram system. ARI diversity primarily provides protection against common cause failure that affects redundant components of the same type and manufacturer. It is unlikely that common cause failures due to environmental or manufacturing processes would cause the loss-of-scram function simultaneously in the current trip circuit and the ARI circuit. For example, fires that occur in the logic control cabinets are likely to cause open circuits in the existing trip circuit to deenergize and result in a scram. On the other hand, a postulated failure mechanism during normal plant operation which prevents the existing trip circuit from deenergizing would be complemented by the diverse function of the ARI circuit. ARI would perform its function during a demand by the "energized-to-trip" signal thereby accomplishing reactor scram.

Common cause failures due to potential operating and maintenance errors or functional deficiencies are primarily protected by designed diversity in the current trip system. Approximately 98% of all plant transients requiring a reactor scram have at least three diverse means (level/pressure, valve position, and flux/radiation sensors) for inititiating a scram signal. The sensor diversity provides protection from failures due to functional deficiency of

*loc cit page 2-3

a sensed scram variable, miscalibration of sensors, and other maintenance errors by a single individual or crew. ARI design provides another independent shutdown system sensor/logic circuit which reduces the potential of failure due to a miscalibration or maintenance error, the most likely common cause failure.

The independent ARI logic circuit provides additional protection against common cause failures affecting the current relay backup scram circuit, i.e., circuitry for the actuation of redundant pilot valves on the scram air header. Although the backup scram circuit is an "energized-to-trip" circuit, the logic uses the same output relays as the primary "deenergized-to-trip" circuit. Therefore, common cause failure in the existing output relays can affect both the prompt and backup scram functions. The ARI circuit provides a separate set of output relays which are not dependent on failure in either the primary or backup scram circuit.

Successful reactor shutdown by ARI relies on the operation of recirculation pump trip and the mechanical portion of the scram system. Operability of ARI and other equipment required for reactor shutdown given loss of the prompt scram function is discussed under Section 3.4. This discussion on diversity assumes that this equipment will operate under the environment imposed by loss of the prompt scram function.

6.1.2 Recirculation Pump Trip

The recirculation pump trip provides a prompt negative reactivity effect during the initial part of an ATWS event. This recirculation pump trip function is required to account for the ARI delay in blowing down the scram air header. The trip function is also required for the boron injection system.

The recirculation pumps are tripped by the same sensors/logic which initiate the ARI circuit. Therefore, the discussion of the ARI diversity is applicable to the recirculation pump trip. Either of the two divisional initiation signals will trip both recirculation pumps. The recirculation pump circuit is an "energized-to-trip" versus the "deenergized-to-trip" circuit for the prompt scram initiation function. Operability of the recirculation pump trip circuit given a loss of the prompt scram function is discussed in Section 3.4.

6.1.3 Automated Standby Liquid Control System (SLCS)

The prompt scram and ARI functions rely on the insertion of control blades for negative reactivity for reactor shutdown. The auto SLCS relies on recirculation pump trip (short term) and the injection of liquid poison (long term) for negative reactivity. The SLCS is therefore functionally diverse from the reactor scram system.

The auto-SLCS system is initiated by the same circuit which initiates the ARI and recirculation pump trip. A discussion of the diversity of the initiating signals is given under the ARI discussion. An additional circuit is provided to confirm an unsuccessful control rod insertion. This circuit uses a timer, the neutron flux signal (APRM - Average Power Range Monitors) and the control rod position indication to prevent poison injection following a scram or successful ARI function event. The neutron flux signal permissive is bypassed for a loss of normal a-c power event since the loss of power to the APRM's would inhibit the poison injection initiation.

Successful reactor shutdown by the poison injection function relies on the successful operation of the following auxiliary systems:

- Safety/Relief Valves
- Containment Isolation (BWR/6 Plants)
- Reactor Water Cleanup System Isolation
- High Pressure Coolant Injection (HPCI) or High Pressure Core Spray (HPCS) System
- Feedwater Runback
- Residual Heat Removal (RHR) System
- Suppression Pool and Containment

A detailed systematic review of the common cause failure potential between the poison injection system (SLCS, permissive logic, and auxiliary systems) and the scram system has not been completed. Initial analysis of the diversity/independence between the two reactor shutdown functions indicates adequate protection against the potential for a common cause failure. With the exception of containment isolation for BWR/6, automatic initiation of the auxiliary systems is performed by a diverse and independent energized-to-trip circuit. The use of the auto SLCS logic for initiation of the BWR/6 containment isolation circuit is being reviewed in order to achieve diversity and independence with the scram system. The majority of the auxiliary systems are initiated by motor operated or squib valves versus air operated valves in the scram system. Although the APRMs are used for scram initiation and the automatic SLCS permissive, their failure will not disable both the scram and automatic SLCS function. Operability of the SLCS (including the auxiliary systems) given an unsuccessful control rod insertion is discussed under Section 3.4.

6.2 DESCRIPTION OF BRINGING PLANT TO POST-ATWS COLD SHUTDOWN

APPENDIX 7.1
EVALUATION OF ANTICIPATED OPERATIONAL TRANSIENTS

This appendix discusses the various categories of transients anticipated in Boiling Water Reactors, the characteristics of each transient and a qualitative discussion in some detail in order to understand the relative severity of each event with failure to scram. This material is an update of the transient event description originally presented in NEDO-10349.

7.1.1 Identification of Anticipated Operational Transients

Anticipated operational occurrences are those conditions of operation which are expected to occur one or more times during the life of the nuclear power plant. From the list of all anticipated operational transients which could occur, those for which scram occurs are identified in Table 7.1.1 along with the initiating signals in the order in which they occur in time. All signals shown after the first signal for each event are redundant to the first signal.

7.1.2 Evaluation of Anticipated Operational Transients

7.1.2.1 Events Resulting in a Nuclear System Pressure Increase

7.1.2.1.1 Loss of Main Condenser Vacuum

A loss of condenser vacuum causes turbine stop valve closure and, at a lower vacuum set point, turbine bypass valve closure. Once initiated, all of the turbine stop valves achieve full closure within about 0.1 second. Closure of multiple stop valves of more than 10% initiates the first reactor scram signal. Following the stop valve closures, the reactor pressure rises causing the collapse of voids in the core. This results in increased neutron flux and initiates a second reactor scram signal. As the pressure in the vessel continues to rise a third reactor scram signal is initiated due to high vessel pressure.

This event produces results similar to the turbine trip transient and gives the most limiting results for the transient cases evaluated with trip scram. With scram, this case gives more limiting results than the main steam isolation valve closure because of the rapid valve closure prior to scram. With failure to scram, however, the MSIV closure event produces more limiting results due to the reduced steam volume available to buffer the valve closure.

7.1.2.1.2 Closure of All main Steam Line Isolation Valves

Closure of one isolation valve at power less than rated is permitted for testing purposes without initiating a scram signal. However, if three steam lines are closed in excess of 10%, it is interpreted as the beginning of a system isolation and a reactor scram signal is initiated. If the reactor successfully scrams on the first signal, no further scram signals will be initiated because neutron flux and vessel pressure will not reach the scram set points. This event results in scram at all power levels.

The short term consequences of the main steam line isolation valve events with trip scram are less severe than the corresponding turbine trip events because the isolation valves closure times are slower (3-5 seconds) than the turbine stop valve closure times. In the case of a failure to scram the long term consequences are more severe for the MSIV as discussed above.

7.1.2.1.3 Closure of One Main Steam Line Isolation Valve, High Power

Normally the operator should reduce power to about 75-90% of rated to avoid scram. If he does test at design power, and scram is not available, the consequences are less severe than for a complete isolation because of the lower pressurization rate.

7.1.2.1.4 Turbine Trip/Load Rejection, High Power

A turbine trip/load rejection will have the same transient sequence as a loss of condenser vacuum except that for a turbine trip/load rejection the turbine bypass valves would remain open.

Turbine trip/load rejection from lower initial power levels decrease in severity to the point where scram may even be avoided within the bypass capacity if auxiliary power is available from an external source.

7.4.2.1.4.1 Bypass Valves Failure Following Turbine Trip/Load Rejection High Power

This event is included to assess the consequences of the turbine bypass valve failing to open in conjunction with a turbine trip. However, this unlikely event would produce a transient similar to, but no more severe than the MSIV closure event as discussed above. A sample case was presented in 4.5.

7.1.2.1.4.2 Bypass Valves Failure Following Turbine Trip/Load Rejection Low Power

This abnormal operational transient is of interest because turbine stop valve closure and turbine control valve fast closure scrams are automatically bypassed when the reactor power level is low. Turbine first-stage pressure is used to initiate this bypass. The highest power level for which these scrams remain bypassed is about 25% of rated power. Reactor scram occurs in less than 2 seconds after initiation of the event and the consequences are less severe than the MSIV closure transient.

7.1.2.1.5 Generator Trip, High Power

A generator trip is a loss of generator electrical load which results in a speed up of the turbine-generator. The turbine-generator acceleration protection devices trip to initiate the control valve fast closure and a reactor scram signal.

With trip scram, this transient is similar to the turbine trip. At power levels below bypass capacity the bypass system will transfer steam around the turbine and avoid scram. Above bypass capacity, high pressure scram will result unless operator action can reduce power to within the bypass capacity.

For failure to scram, this event is similar to the turbine trip event and is bounded by the MSIV closure ATWS event.

7.1.2.1.6 Pressure Regulator Failure-Increasing Pressure, High Power

If the regulator fails, the backup regulator will function automatically producing a slight 10 psi pressure change. Pressure regulator malfunctions that result in the turbine steam flow shutoff and a nuclear system pressure increase are similar to, but of milder consequence, than the generator trip decribed previously. Turbine control valve closure is slower than the fast closure time of this valve.

7.1.2.2 Events Resulting in a Reactor Moderator Temperature Decrease

7.1.2.2.1 Loss of a Feedwater Heater-Manual Recirculation Flow Control

In the event of the loss of a feedwater heater (70°F), the reactor vessel receives cooler feedwater which produces increased reactivity, and an increase in core power results. If the reactor is not in the automatic control mode, and if the reactor is operating near full power and flow, a high neutron flux reactor scram signal may occur. If no shutdown action ensues, the mismatch results in an increase in the vessel pressure. However, pressure remains below the scram set point.

This transient is less severe from lower power levels for two main reasons:

a. lower initial power levels will have greater thermal power fuel limit margins; and

b. The magnitude of the power rise decreases with the initial power condition.

Therefore, transients from other reactor operating states or lower power levels will be less severe.

7.1.2.2.2 Feedwater Controller Malfunction-Maximum Demand

Failure of the feedwater controller in the direction of increased feedwater flow results in a moderator temperature decrease causing a reactor power increase through the effect of the negative void reactivity coefficient. This initial power increase is not sufficient to initiate a high flux scram signal and the reactor will continue to operate at a slightly increased power level while water level increases. Under severe conditions, the mismatch between the steam line mass flow rate and the feedwater mass flow rate will cause the water level in the reactor vessel to rise at the rate of approximately 3 inches per second. A high water level turbine trip will be initiated when the sensed level has been increased by approximately 1 foot. This transient then reverts to that of a turbine trip.

7.1.2.2.3 Shutdown Cooling (RHR'S) Malfunction-Decreasing Temperature

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for the RHR heat exchangers. The resulting temperature decrease causes a slow insertion of poitive reactivity into the core. If the reactor were critical or near critical, a very slow reactor power increase could result. If no operator action were taken to control the power level, a high neutron Intermediate Range Monitor (IRM) flux reactor scram (12% of rated) would terminate the transient without fuel damage and without any measurable nuclear system pressure increase.

7.1.2.3 Events Resulting in a Positive Reactivity Insertion

7.1.2.3.1 Continuous Rod Withdrawal During Reactor Startup

Control rod withdrawal errors are considered when the reactor is at power levels below the power range. The most severe case occurs when the reactor is just critical at room temperature and an out-of-sequence rod is continuously withdrawn. The rod worth minimizer would nomally prevent withdrawal of such a rod. It is assumed that the IRM channels are in the worst conditions of allowed bypass. The scaling arrangement of the IRMs is such that for unbypassed IRM channels a scram signal is generated before the detected neutron flux has increased by more than a factor of ten. In addition, a high neutron flux scram is generated by the APRMs at 120% of rated power or at a lower set point as determined by the flow reference scram.

7.1.2.4 Events Resulting in a Reactor Vessel Coolant Inventory Decrease

7.1.2.4.1 Pressure Regulator Malfunction-Decreasing Pressure, High, Medium, Low Power

If either the operating pressure regulator or the backup pressure regulator fails in an open direction, the turbine admission valves can be fully opened, and the turbine bypass valves can be partially opened. This action initially results in decreasing coolant inventory in the reactor vessel as the mass flow rate of steam leaving the vessel exceeds the mass flow rate of water entering the vessel. This depressurization results in the formation of voids which causes nuetron power to decrease. The main steam line isolation valves automatically close when the pressure at the turbine decreases by approximately 100 psi. After the isolation valves begin to close, this transients reverts to the transient for closure of all main steam line isolation valves except with a reduced initial power level. The sequence of actions varies for this event depending upon the initial power level.

7.1.2.4.2 Loss of Feedwater Flow

A loss of feedwater flow results in a situation where the mass flow rate of steam leaving the reactor vessel exceeds the mass flow rate of water entering the vessel, resulting in a net decrease in the vessel coolant inventory. After the water level in the vessel drops to the low water level scram set point a reactor scram signal is initiated, and after an additional drop of 48 inches, a signal to close the isolation valves is initiated. After the isolation valves close 10%, a second reactor scram signal is initiated. This transient then reverts to a case similar to that of an isolation valve closure.

7.1.2.4.3 Inadvertent Opening of a Safety or Relief Valve

A mild depressurization transient is introduced by the inadvertent opening of a valve on the main steam line. Opening of a saftey valve will initiate a scram only on drywell pressure exceeding 2 psig since neutron flux, reactor water level, and vessel pressure are not significantly affected. Opening of a relief valve will take longer to initiate a scram since the pressure increase in the drywell is through the wetwell. In the meantime a mild depressurization transient is introduced. The turbine pressure regulator senses this pressure decrease and drops turbine flow to maintain pressure control The reactor settles to nearly the initial power until scram is initiated. Operator intervention can avoid scram should he be able to correct the condition in time. Generally scram is required for the event only if suppression pool temperature reaches a prescribed limit. At that time, manual scram is required.

7.1.2.4.4 Inadvertent Opening of All Bypass Valves

This event diverts some of the turbine steam flow directly to the main condenser causing a decrease in turbine pressure. The pressure decrease is detected by the pressure regulator which initiates closure of the turbine control valves in an attempt to control steam flow.

When steam flow is lower than bypass capacity the pressure regulator will rapidly close the control valves. Bypass flow will continuously be greater than steam line flow and depressurization will occur. Should turbine inlet pressure decrease to the low steam line pressure set point the main steam line valves will be closed initiating a scram signal.

When the power level is higher than bypass capacity the pressure regulator would potentially close the control valves to maintain vessel steam flow. Turbine flow would be decreased by an amount equal to bypass flow. The transient would be mild and a scram signal would not be initiated.

7.1.2.4.5 Loss of Auxiliary Power

A complex sequence of events occurs when the plant loses all auxiliary power. This transient is classified as an event resulting in a reactor vessel coolant inventory decrease. The loss of power to all electrical pumps initiates several types of transients. The loss of feedwater pumps causes the water inventory in the reactor to decrease. The loss of recirculation pumps causes core flow to drop resulting in increased voiding and a decrease in neutron power. The loss of the main condenser circulating water pumps causes condenser vacuum to drop to the turbine trip setting in approximately 6 seconds. The protection system motor-generator (MG) sets coast down to the point of scram and main steam line isolation in approximately 5 seconds. The transient reverts to that of closure of all main steam line isolation valves except it has a reduced initial power level.

7.1.2.5 Events Resulting in a Core Coolant Flow Increase

7.1.2.5.1 Recirculation Flow Controller Malfunction-Increase Flow

Failure of the master controller can result in a speed increase of both recirculation pumps so that flow increase would cause the neutron flux to increase beyond initial values. The most severe case, however, is the failure of the speed controller of one of the motor-generator sets, since the speed controller rate limits are adjusted to keep the effect of master flow controller failure less severe than that of single speed controller failure. As a result the high neutron flux scram set point may be reached. If this set point is not reached, no system limits are exceeded. The bypass system can adquately handle the increase in steam flow. The increased recirculation flow will not cause system damage.

7.1.2.5.2 Startup of Idle Recirculation Pump Between 60 and 65% Power

The event will not raise power sufficiently to initiate scram if initial power is below 60%. Initial power cannot be above 65% because this is the capability limit of one recirculation pump. If the idle loop water is sufficiently low in temperature, a scram might be required, but even in the worst case, this transient is less severe than 7.1.8.1 above.

Table 7.1.1
ABNORMAL OPERATIONAL TRANSIENTS
RESULTING IN REACTOR SCRAM

Class of Transients and Initiating Events	Scram Signals Initiated By
Nuclear System Pressure Increase	
Loss of Condenser Vacuum	Stop Valves, Flux, Vessel Pressure
Closure of All Main Steam Line Isolation Valves	Isolation Valves, Flux, Vessel Pressure
Turbine Trip, High Power	Stop Valves, Flux, Vessel Pressure
Generator Trip, High Power	Control Valves, Flux, Vessel Pressure
Pressure Regulator Failure-Increasing Pressure, High Power	Flux, Vessel Pressure[1]
Moderator Temperature Decrease	
Loss of Feedwater Heater-Manual Recirculation Flow Control	Flux
Feedwater Controller Malfunction-Maximum Demand	Stop Valves, Flux, Vessel Pressure
Shutdown Cooling (RHRS) Malfunction Decreasing Temperature	Flux
Rectivity Insertion	
Continuous Rod Withdrawal During Reactor Startup	IRM Flux, APRM Flux

[1]Scram level may not be reached.

Table 7.1.1 (Continued)

Class of Transients and Initiating Events	Scram Signals Initiated By
<u>Decrease of Coolant Inventory</u>	
Pressure Regulator Failure-Decreasing Pressure, High Power	Isolation Valves, Low Water Level, Vessel Pressure
Pressure Regulator Failure-Decreasing Pressure, Medium Power	Stop Valves, Flux, Vessel Pressure
Pressure Regulator Failure-Decreasing Pressure, Low Power	Isolation Valves, Flux, Vessel Pressure
Loss of Feedwater Flow	Low Water Level, Isolation Valves, Flux, Vessel Pressure
Inadvertent Opening of a Safety or Relief Valve	Containment Pressure of Manual Scram following Suppression Pool High Temperature Alarm
Inadvertent Opening of All Bypass Valves	Flux[1]
Loss of Auxiliary Power	Stop Valve, Flux, Vessel Pressure
<u>Core Coolant Flow Increase</u>	
Recirculation Flow Controller Malfunction-Increasing Flow	Flux
Startup of Idle Recirculation Pump, 60-65% Power	Flux[1]

[1]Scram level may not be reached.

APPENDIX 7.2
SAR DESIGN BASIS OF EQUIPMENT

The SAR Design Bases given below are from typical SAR's and do not necessarily reflect systems on all plants.

7.2.1 Reactor Core Isolation Cooling System (RCIC)

7.2.1.1 Safety Design Basis

The RCIC system is designed to:

Ensure that adequate core cooling takes place to prevent the reactor fuel from overheating in the event that reactor isolation is accompanied by loss of flow from the reactor feedwater system.

7.2.1.2 Power Generation Design Basis

The RCIC system is designed to:

a. Operate automatically in time to maintain sufficient coolant in the RPV so that the low pressure emergency core cooling systems (ADS, LPCI and core spray systems) are not actuated.

b. Provide for remote-manual operation of the system by an operator.

c. Provide a high degree of assurance that the system will operate when necessary.

d. Have the power supply for the system from an immediately available energy source of high reliability.

e. Provide for periodic testing during plant operation.

7.2.1.3 System Description

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the RPV.

The steam supply to the turbine comes from the reactor vessel. The steam exhaust from the turbine dumps to the suppression pool. The pump can take suction from the demineralized water in the condensate storage tank or from the suppression pool.

The pump discharges either to the feedwater line (head spray nozzle for BWR/6) or to a full-flow return test line to the condensate storage tank. A minimum-flow bypass line to the suppression pool is provided to protect the pump during startup and shutdown. The makeup water is delivered into the RPV through the feedwater line (head spray nozzle for BWR/5 and BWR/6) and is distributed within the reactor vessel through the feedwater sparger. Cooling water for the RCIC turbine lube oil cooler and barometric condenser is supplied from the discharge of the pump.

Following any reactor shutdown, steam generation continues because of heat produced by the radioactive decay of fission products. Initially, the rate of steam generation can be as much as approximately 6% of rated flow and is augmented during the first few seconds by delayed neutrons and some of the residual energy stored in the fuel. Steam normally flows to the main condenser through the turbine bypass or, if the condenser is isolated, to the suppression pool or to the RHR heat exchangers which function as steam condensers. The fluid removed from the RPV is normally made up by the feedwater pumps supplemented by leakage from the CRD system. If makeup water is required to supplement these primary sources of water, the RCIC turbine-pump unit starts automatically on receipt of an RPV low-water-level signal or is started by the operator from the main control room. The RCIC delivers its design flow within 30 seconds after actuation. To limit the amount of fluid leaving the RPV, the RPV low-water-level signal also actuates the closure of the MSIVs.

The RCIC makeup capacity is sufficient to avoid the need for the low-pressure ECCS. Pump suction is normally lined up to the condensate storage tank. The volume of water stored for the RCIC is sufficient to allow operation for eight hours after shutdown, assuming that none of the steam generated in the RPV is returned to the RPV as condensate. Other systems that use the same reservoir and could jeopardize the availability of this quantity of water can be isolated. A low-level alarm is energized when the level in the storage volume fails to the minimum required to meet the design requirements of the RCIC system.

The RCIC system is sized to prevent actuation of the triple low level signal for RPV isolation incidents. Prevention of this signal ensures core cooling and prevents ADS actuation, thus preventing inadvertent blowdown of the RPV for this situation.

The backup supply of cooling water for the RCIC is the suppression pool. The turbine-pump assembly is located below the level of the condensate storage tank and below the minimum water level in the suppression pool to ensure positive suction head to the pump. NPSH requirements are satisfied by providing adequate suction head and adequate suction line size.

All components required for initiating the RCIC are completely independent of auxiliary ac power, plant service air, and external cooling water systems. These components require only power derived from the station battery to operate the valves and logic. The power source for the turbine-pump unit is the steam generated in the RPV by the decay heat in the core. The steam is piped directly to the turbine, and the turbine exhaust is piped to the suppression pool.

If for any reason the RPV is isolated from the main condenser, pressure in the RPV increases but is limited by automatic or remote-manual actuation of the safety/relief valves. After approximately 30 minutes of safety/relief valve operation, it becomes necessary to limit any further temperature rise in the suppression pool. Therefore, as soon as possible after isolation, the operator directs reactor steam to the RHR heat exchangers where the steam is condensed and subcooled. At this time, the RCIC pump suction is manually changed to the RHR heat exchanger, where it pumps the condensate back to the RPV.

This mode of operation provides a closed-circuit cooling loop, which conserves reactor coolant and limits the increase in temperature and volume of the suppression pool. In this mode, the nuclear system pressure can be reduced by increasing the rate of steam condensation. This mode of operation must be initiated deliberately to put the nuclear system in a hot standby condition.

Throughout the period of RCIC operation, exhaust from the RCIC turbine is condensed in the suppression pool which results in a slow temperature rise of approximately 3°F per hour in the pool. If necessary, one RHR heat exchanger can be used to cool the suppression pool after approximately 1.5 hours. If for any reason the RCIC is unable to supply sufficient flow for core cooling, the ECCS provides the required boundary protection.

Long-term heat removal capability may be provided by the RCIC during scram, pressure relief, core cooling, RPV isolation, and restoration to ac power. The RHR system may be used for long-term heat removal during any long-term isolation. These events are all situations in which the RPV is isolated from the main condenser.

The HPCI and RCIC systems are located in separate rooms in different corners of the reactor building. Piping runs are separated and the water delivered from each system enters the RPV via different nozzles.

7.2.1.4 Safety Evaluation

To ensure that the RCIC operates when necessary and in time to provide adequate core cooling, the power supply for the system is taken from immediately available energy sources of high reliability. Added assurance is given by the capability for periodic testing during station operation. Evaluation of reliability of the instrumentation for the RCIC shows that no failure of a single initiating sensor either prevents or falsely starts the system.

The RCIC system components within the drywell, up to and including the outer isolation valve, are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 1. For BWR/6, the RCIC is also designed as seismic Category I equipment.

7.2.1.5 Tests and Inspections

A design flow functional test of the RCIC system is performed during plant operation by taking a suction from the condensate storage tank and discharging through the full-flow test return line back to the condensate storage tank.

The discharge valve to the feedwater line remains closed during the test. Discharge is obtained by first closing the upstream discharge valve. Control system design provides automatic return from the test to the operating mode when system operation is required during testing of individual components.

Periodic inspections and maintenance of the turbine-pump unit are conducted in accordance with the manufacturer's instructions. Valve position indication and instrumentation alarms are displayed in the main control room.

7.2.1.6 Initiating Circuits

RPV low water level is monitored by four indicating-type level switches which sense the difference between the pressure of a constant reference leg of water and the pressure resulting from the actual height of water in the vessel. Each level switch contains four sets of switch contacts; two high and two low. Two pipelines, attached to taps above and below the water level on the reactor vessel, are required for each of the two reference legs used with the RCIC. Pipelines are physically separated from each other and tap off the reactor vessel at widely separated points. Two pairs of differential pressure sensing lines from the two reference legs terminate outside the primary containment and inside the reactor building.

The RCIC system is initiated only by low water level. The signals are derived from relays that are part of the RHR system. The RCIC initiation circuit is arranged in a "one-out-of-two taken twice" logic.

The RCIC system is automatically initiated after the receipt of a RPV low water level signal and produces the design flow rate within 30 seconds. The controls then function to provide a flow of design makeup water to the RPV until the amount of water delivered to the RPV is adequate to restore vessel level. At this time, the RCIC system automatically shuts down. The controls are arranged to allow remote-manual startup, operation, and shutdown.

High water level in the RPV indicates that the RCIC system has performed satisfactorily in providing makeup water to the RPV. Further increase in level could result in RCIC system turbine damage caused by gross carryover of moisture. The reactor vessel high water level setting which trips the turbine is near the top of the steam separators and is sufficient to prevent gross moisture carryover to the turbine. Two level switches that sense differential pressure are arranged so that both switches are required to trip to initiate a turbine shutdown.

7.2.1.6.1 Redundancy, Diversity, and Separation. Four water level sensors in a "one-out-of-two taken twice" circuit supply the start signal.

As in the ECCS, the RCIC system is separated into divisions designated I and II. The RCIC is a Division I system, but the inside steamline valve is in Division II: therefore, part of the RCIC logic is treated as Division II. The inside valve is an ac-logic powered valve. The rest of the valves are dc-powered valves. Division I logic is powered by 125 volt dc bus 2A and the Division II logic is powered by 125 volt a-c bus 2B. In order to maintain the required separation, RCIC logic relays, cabling, instruments, and manual separation from Division II is maintained.

7.2.1.7 Actuated Devices

All automatic valves in the RCIC are equipped with remote-manual test capability so that the entire system can be operated from the main control room. Motor-operated valves are provided with appropriate limit switches to turn off the motors when the fully open or fully closed positions are reached. Logic circuitry that controls valves which are automatically closed on isolation or turbine trip signals is equipped with manual reset devices so that the valves can be reopened by operator action. All required components of the RCIC controls operate independent of ac power.

To ensure that the RCIC system can be brought to design flow rate within 30 seconds from the receipt of the initiation signal, the following maximum operating times for essential RCIC valves are provided by the valve operation mechanisms:

a. RCIC turbine steam supply valve - 15 seconds

b. RCIC pump discharge valves - 15 seconds

c. RCIC pump minimum flow bypass valve - 5 seconds

Three pump suction valves are provided in the RCIC. One valve lines up pump suction from the condensate storage tank; the other two from the suppression chamber. The condensate storage tank is the preferred source. All three valves are operated by dc motors. Upon receipt of a RCIC initiation signal, the condensate storage tank suction valve automatically opens.

7.2.1.8 Environmental Considerations

The only RCIC control component located inside the primary containment that must remain functional is the control mechanism for the inside isolation valve. The RCIC instrumentation and controls equipment located outside the primary containment is selected in consideration of the normal and accident environments in which it must operate.

7.2.2 High Pressure Core Spray System (HPCS) for BWR 5/6

7.2.2.1 System Description

The High Pressure Core Spray System (HPCS) consists of a single motor-driven pump located outside the containment and associated system piping, valves, controls, and instrumentation. The system is designed to operate from normal offsite auxiliary power or from a standby diesel generator supply if offsite power is not available.

The principal HPCS equipment is located outside the containment. Suction piping is provided from the condensate storage tank and the suppression pool. Such an arrangement provides the capability to use reactor grade water from the condensate storage tank when the HPCS system functions to backup the RCIC system. In the event that the condensate storage water supply becomes exhausted or is not available, automatic switchover to the suppression pool water source will assure a closed cooling water supply for extended operation of the HPCS system.

HPCS pump suction is also automatically transferred to the suppression pool if the suppression pool water level exceeds a prescribed value.

After the HPCS injection piping enters the vessel, it divides and enters the shroud. A semicircular sparger is attached to each outlet. Nozzles are spaced around the spargers to spray the water radially over the core and into the fuel assemblies. The HPCS injection piping is provided with an isolation valve on each side of the containment barrier. Remote controls for operating the valves and diesel generator are provided in the plant control room.

The HPCS system is designed to cool the reactor core sufficiently to prevent fuel cladding tempratures from exceeding the 10CFR50 limit of 2200°F following any break in the nuclear system piping. The system is designed to pump water into the reactor vessel over a wide range of pressures. For small breaks that do not result in rapid reactor depressurization, the system maintains reactor water level and depressurizes the vessel. For large breaks the HPCS system cools the core by a spray.

If a loss-of-coolant accident should occur, a low water level signal or a high containment pressure signal initiates a reactor scram, the HPCS and its supporting equipment. The HPCS flow automatically stops when a high water level in the reactor vessel is signaled. The HPCS system also serves as a backup to the RCIC system in the event the reactor becomes isolated from the main condenser during operation and feedwater flow is low.

If normal auxiliary power is not available, the HPCS pump motor is driven by its own on-site power source.

When the system is started, initial flow rate is established by primary system pressure. As vessel pressure decreases, flow will increase. When vessel pressure reaches 200 psid* the system reaches rated core spray flow. The HPCS motor size is based on peak horsepower requirements.

*psid = differential pressure between the reactor vessel and the suction source. The elevation of the HPCS pump is such that a flooded pump suction is assured.

Pump NPSH requirements are met even with the containment at atmospheric pressure. NPSH will be calculated in accordance with Regulatory Guide 1.1.

A motor-operated valve is provided to isolate the High Pressure Core Spray System from the nuclear system when the HPCS system is not required for core cooling or RCIC backup. This valve, installed outside the containment, is normally closed as a backup to the inside testable check valve for containment integrity purposes. A drain line is provided between the two valves. The test connection line is normally closed with two valves to assure containment integrity.

If the HPCS line should break outside the containment, a check valve in the line prevents loss of reactor water outside the containment. The HPCS pump and piping are positioned to avoid damage from the physical effects of design-basis accidents, such as pipe whip, missiles, high temperature, pressure, and humidity.

To assure continuous core cooling, signals to isolate the containment do not operate any HPCS valves.

The HPCS equipment and support structures are designed in accordance with seismic Category I criteria.

7.2.2.2 Applicable Codes and Classification

All piping systems and components (pumps, valves, etc.) for the HPCS will comply with the applicable codes, addenda, code cases and errata in effect at the time the equipment is procured.

The piping and components of the HPCS system within the containment and out to and including the pressure retaining injection valve are Class I. All other piping and components are Class II except the HPCS condensate storage tank test return line downstream of the shutoff valves.

The equipment and piping of the HPCS are designed to the requirements of seismic - Class I.

7.2.2.3 Materials Specifications and Compatibility

Nonmetallic materials such as lubricants, seals, packings, paints and primers, insulation, as well as metallic materials, etc., are selected for compatibility with other materials in the system and the surroundings with concern for chemical, radiolytic, mechanical and nuclear effects.

7.2.2.4 Provisions for Performance Testing

a. A full flow test line is provided to route water from and to the condensate storage tank without entering the reactor pressure vessel. The suction line from the condensate tank also provides reactor grade water to treat the RPV during shutdown.

b. A full flow test line is provided to route water from and to the suppression pool without entering the reactor pressure vessel.

c. Instrumentation is provided to indicate system performance during normal and test operations.

d. All motor-operated valves are capable of manual operation either locally or remotely for test purposes.

e. System relief valves are removable for bench testing.

f. Drains are provided to leak test the major system valves.

7.2.3 High Pressure Coolant Injection (HPCI) System for BWR/4

The HPCI system consists of a steam turbine which drives a constant-flow pump, system piping, valves, controls, and instrumentation.

The principal HPCI system equipment is installed in the reactor building. The turbine-pump assembly is located in a shielded area to ensure that personnel access to adjacent areas is not restricted during operation of the HPCI system. Suction piping comes from the condensate storage tank and the suppression pool. Injection water is piped to the reactor feedwater pipe at a "T" connection.

Steam supply for the turbine is piped from a main steam header in the primary containment. This piping is provided with an isolation valve on each side of the drywell barrier. Remote controls for valve and turbine operation are provided in the main control room.

The HPCI system is provided to ensure that the reactor is adequately cooled to limit fuel clad temperature in the event of a small break in the nuclear system and a loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the nuclear plant to be shut down while maintaining sufficient reactor vessel water inventory until the reactor vessel is depressurized. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which either LPCI operation or core spray system operation maintains core cooling.

If a LOCA occurs, the reactor scrams upon receipt of a low-water-level signal from the reactor or a high-pressure signal from the drywell. The HPCI system starts when the water level reaches a preselected height above the core or if high pressure exists in the primary containment. The HPCI system automatically stops when it receives a signal of high water level in the reactor vessel.

The HPCI system is designed to pump water into the reactor vessel for a wide range of pressures in the reactor vessel. Two sources of water are available. Initially, the system uses demineralized water from the condensate storage tank. approximately 100,000 gallons of the 500,000-gallon condensate storage tank are held in reserve for the HPCI system and reactor core isolation cooling (RCIC) system. The value of 100,000 gallons is based on (1) the megawatt thermal rating of the plant, (2) 100°F makeup water available from the condensate storage tank (HPCI/RCIC system design criteria), and (3) the inventory loss due to the boil-off rate in the reactor for an eight-hour integrated decay heat factor. System demands on the condensate storage tank other than the HPCI systems and RCIC system will draw from a tank internal standpipe. The inlet to this standpipe will be set at a level so that approximately 100,000 gallons will be below the intake and unavailable to these other systems except through a locked valve crossover that could be opened during refueling. Both the HPCI system and RCIC system will connect separately to the condensate storage tank near the bottom. In addition the condensate storage tank will have a backup capacity from the 100,000-gallon demineralized water storage tank. Should the condensate storage tank be drawn down to a low level, automatic transfer to the suppression pool occurs. Water from either source is pumped into the reactor vessel via a feedwater line. Flow is distributed within the reactor vessel through the feedwater spargers, causing mixing with the hot water or steam in the reactor pressure vessel.

To ensure positive suction head to the pump, the pump is located below the level of the condensate storage tank and below the water level in the suppression pool. Pump NPSH requirements are met by providing adequate suction head and adequate suction line size. Available NPSH is calculated using the assumptions of Regulatory Guide 1.1 (November 1970).

The HPCI turbine-pump assembly and piping are located so as to be protected from the physical effects of design basis accidents such as pipe whip, flooding, and high temperature. The equipment is located outside the primary containment.

Steam from the reactor vessel drives the HPCI turbine. Decay heat and stored heat generate steam which is extracted from a main steam header upstream of the main steamline isolation valves. The two HPCI system isolation valves in the steam line to the HPCI turbine are normally open to keep piping to the turbine at elevated temperatures and to permit rapid startup of the HPCI system. Signals from the HPCI control system open or close the turbine stop valve.

To prevent the HPCI steam supply line from filling with water, a condensate drain pot is provided upstream of the turbine stop valve. The drain pot normally routes condensate to the main condenser, but upon receipt of an HPCI system initiation signal or a loss of control air pressure signal, isolation valves on the condensate line close automatically.

Two devices control turbine power: 1) a speed governor limits turbine speed to its maximum operating level and 2) a control governor with an automatic speed set point control is positioned by a demand signal from a flow controller to maintain constant flow over the pressure range of HPCI system operation. When the governor is in the test mode, it can be operated manually; however, it is automatically repositioned by the demand signal from the flow controller if system initiation is required.

As reactor steam pressure decreases, the HPCI turbine throttle valve opens wider to permit passage of the steam flow required to provide sufficient core cooling to prevent clad melting while the pressure in the reactor vessel exceeds that at which core spray and LPCI become effective.

Exhaust steam from the HPCI turbine is discharged to the suppression pool. A drain pot at the low point in the exhaust line collects condensate which is discharged through a steam trap to the suppression pool or automatically bypassed to the gland seal condenser.

The pump is designed and tested in accordance with the standards of the Hydraulic Institute.

Startup of the HPCI system is completely independent of ac power. For startup to occur, only dc power from the plant batteries and steam extracted from the nuclear system are required.

Various operations of the HPCI system components are summarized below.

The HPCI system controls automatically start the system and bring it to design flow rate within 25 seconds from receipt of a low-water-level signal from the reactor vessel or a high-pressure signal from the primary containment (drywell).

The HPCI system turbine is shutdown automatically by any of the following signals:

a. Turbine overspeed. This prevents damage to the turbine and turbine casing.

b. Reactor vessel high water level. This prevents flooding of steamlines.

c. HPCI pump low suction pressure. This prevents damage to the pump due to loss of flow.

d. HPCI turbine exhaust high pressure. This indicates a turbine or turbine control malfunction.

If an initiation signal is received after the turbine is shutdown, the system is capable of automatic restart if no shutdown signals exist.

Because the steam supply line to the HPCI system turbine is part of the nuclear system process barrier, certain signals automatically isolate this line, causing shutdown of the HPCI turbine.

In addition to the automatic operational features of the system, it also provides for remote manual startup, operation, and shutdown (provided automatic initiation or shutdown signals do not exist). All automatically operated valves are equipped with a remote manual functional test feature.

HPCI system initiation automatically actuates the following valves:

a. HPCI system pump discharge shutoff valve

b. HPCI system steam supply shutoff valve

c. HPCI system turbine stop valve

d. HPCI system turbine control valve

e. HPCI system steam supply line drain isolation valves

f. HPCI control loop valve

The hydraulic oil pump must be started and the hydraulic control system must be functioning properly before the turbine valves can be opened. The gland seal condenser components must be operating to prevent outleakage from the turbine shaft seals. Startup of the equipment is automatic, but its failure does not prevent the HPCI system from fulfilling its core cooling objective. When rated flow is established, the flow controller signal adjusts the setting of the control governor to maintain rated flow as nuclear system pressure decreases.

A minimum-flow bypass is provided for pump protection. The bypass valve automatically opens on a low-flow signal and automatically closes on a high-flow signal. When the bypass is open, the flow is directed to the suppression pool.

A system test line provides recirculation on the condensate storage tank during system test. Shutoff valves are provided with proper interlocks which automatically close the test line upon receipt of an HPCI system initiation signal.

7.2.3.2 Applicable Codes and Classification

The HPCI piping, components, and system designs comply, as a minimum, with applicable codes, addenda, code cases, and errata in effect at the time the equipment was procured. These systems are designed and constructed in accordance with seismic Category I criteria.

The HPCI is divided into two classes. The Class I portion includes all piping and components which are part of the reactor system boundary out to and including the second isolation valve.

The Class I portions of the HPCI system is designed and constructed in accordance with Subsection NB of the ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components.

The remaining portions of the HPCI system is designated Class 2 and are designed and constructed in accordance with Subsection NC of the ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components.

7.2.3.4 Materials Specifications and Compatibility

Nonmetallic materials such as lubricants, seals, packings, paints, primers, and insulation, as well as metallic materials, are selected for compabitility with other materials in the system and surroundings with concern for chemical, radiolytic, mechanical, nuclear radiation, and temperature effects.

HPCI is protected against the effects of pipe whip, which might result from piping failures up to and including the LOCA, by separation barriers, pipe-whip restraints, or energy absorbing materials. One or more of these three methods will be applied to provide protection against cascading damage to the piping and components of the ECCS which could otherwise result in a reduction of ECCS effectiveness to an unacceptable level.

ECCS piping and components located outside the containment are protected from internally and externally generated missiles by the reinforced concrete structure of the ECCS pump rooms. In addition, the watertight construction of the ECCS pump rooms below grade level protects against damage by flooding.

7.2.3.4 Provisions for Performance Testing

a. A full-flow test line is provided to route water from and to condensate storage tank without entering the RPV.

b. A bypass-flow test line is provided to route water from and to the suppression pool without entering the RPV.

c. Instrumentation is provided to indicate system performance during normal and test operations.

d. All motor-operated valves are capable of manual operation, either local or remote, for test purposes.

e. Drains are provided to leak test the major system valves.

7.2.4 Residual Heat Removal System for BWR/4 (RHR)

7.2.4.1 Safety Design Bases

The RHR system is designed to:

a. act automatically, in the LPCI mode, in combination with other ECCS systems, to restore and maintain the coolant inventory in the RPV so that the core is adequately cooled to preclude fuel-clad perforation and subsequent energy release due to a metal-water reaction.

b. give such diversity and redundancy, in conjunction with other ECCS systems, that only a highly improbable combination of events could result in its inability to provide adequate core cooling.

c. provide a source of water for restoration of reactor vessel coolant inventory located within the primary containment in such a manner that a closed cooling water path is established.

d. provide a high degree of assurance that the RHR system operates satisfactorily during a LOCA and that each active component is capable of being tested during operation of the nuclear system.

e. satisfy seismic Category I requirements.

f. permit RHR service water to be pumped directly into the RHR system.

g. provide heat exchangers with a heat-removal capability for long-term containment cooling.

7.2.4.2 Power Generation Design Bases

The RHR system is designed:

a. to have enough heat-removal capacity to cooldown the reactor to 125°F within 20 hours after shutdown.

b. to have fuel pool connections so that the RHR heat exchangers can be used to supplement the fuel pool cooling capacity.

c. to be able to condense reactor steam generated by decay heat and direct the condensate to the suction side of the RCIC pumps.

d. so that closed loop flow path between the suppression pool and the RHR heat exchangers can be established so that the heat-removal capability of these heat exchangers can be used to cool the suppression pool.

7.2.4.3 System Description

7.2.4.3.1 <u>Summary</u>. The RHR system is designed for seven modes of operation to satisfy all the objectives and bases. The modes are summarized as follows:

Mode*	Action	Function
Low-pressure coolant injection (LPCI)	Accident safety	Restore and maintain reactor vessel water level after a LOCA.
Containment spray	Post-accident safety	Limit temperature and pressure in the torus and drywell after a LOCA.
Condensing*	Abnormal operation	Condense steam while the reactor is isolated from the main condenser and level is being maintained by RCIC.
Pool cooling*	Abnormal operation	Remove heat from the suppression pool water.
Shutdown cooling*	Planned operation	Remove decay and residual heat from the reactor core to achieve and maintain a cold shutdown condition.
Minimum flow	Equipment protection	Prevent pump damage when operating against closed discharge valve.
Test	System Test	Test RHR system during plant operation.

The major equipment of the RHR system consists of two heat exchangers and four RHR pumps. The RHR service water system provides cooling water to the heat exchangers. The equipment is connected by associated valves and piping, while controls and instrumentation are provided for proper system operation.

*Containment cooling occurs when RHR service water and LPCI water (with or without containment spray water) is flowing through the RHR heat exchangers.

The RHR pumps are sized for the flow required during LPCI operation, which is the subsystem that requires the maximum flow rate. The pumps are arranged and located so that adequate suction head is ensured for all operating conditions. The pump motor is air cooled.

The heat exchangers are sized on the basis of their required duty for the shutdown cooling function. The heat exchanger shell and tube sides are provided with drain connections. The shell side is provided with a vent to remove noncondensable gases. Thermal relief valves on the heat exchanger shell side, a relief valve on the RHR pump discharge, and a relief valve on the HPCI steam supply line to the RHR heat exchangers protect the heat exchanger from overpressure.

The most limiting duty is that duty associated with cooling the reactor to 125°F in the normal shutdown cooling mode. The performance of this type of heat exchanger operating in the normal shutdown cooling mode (water to water) is well established in currently operating BWR facilities.

The steam condensing mode poses a less severe duty requirement, and the design is very similar to an upright feedwater heater which has proven its reliability through many years of service.

One loop, consisting of a heat exchanger, two RHR pumps in parallel, and associated piping, is located in one area of the reactor building. The remaining heat exchanger, pumps, and piping, all of which form a secoind loop, are located in another area of the reactor buiilding to minimize the possibility of a single physical event causing the loss of the entire system.

7.2.4.4 Steam Condensing Mode

During RCIC operation, there is a limit to the amount of decay heat that can be dumped to the suppression pool without providing suppression pool cooling. This limit is fixed by requiring that the allowable temperature does not exceed 170°F immediately after the design basis LOCA. After a period of steam condensing, one RHR heat exchanger must be placed in pool cooling operation to ensure that pool temperature limits are met.

During the reactor condensing mode, decay heat is transferred to the RHR service water instead of the pool by using the RHR system heat exchangers as direct steam condensers. The relatively cool condensate from the heat exchangers is either dumped to the suppression pool or returned to the suction side of the RCIC system pump and then pumped back into the RPV via a connection to the feed-water line.

When the RHR system heat exchangers are operating in the steam condensing mode, steam at reactor pressure and temperature is taken from the RPV via a connection to the steam supply line to the HPCI system turbine. The steam is then reduced to the desired operating pressure upstream of the heat exchanger. The steam is condensed and subcooled by the RHR service water passing through the tubes of the heat exchanger. The heat transfer across the tubes controls the level of condensate in the heat exchanger. The water level in the heat exchanger in turn controls the flow of condensate to the suppression pool by control of a throttle valve or controls the return flow to the reactor vessel by controlling the RCIC system turbine speed. The operator has the option to select the control mode.

One or two hours after reactor shutdown, one RHR heat exchanger operating as a steam condenser has sufficient heat-transfer capacity to handle decay heat. At this time, the other heat exchanger is operated in the pool cooling mode to cool the suppression pool directly.

7.2.4.5 Safety Evaluation

An interlock exists in the logic for the RHR shutdown cooling valves, which are normally closed during power operation, to prevent opening of the valves above a preset pressure set point. This set point is selected to assure that pressure integrity of the RHR system is maintained. Administrative operating procedures require the operator to close these shutdown cooling valves prior to pressure operation. However, as a backup, the interlock will automatically close these valves when the pressure set point is reached. Double indicating lights are provided in the main control room for valve-position indication.

The RHR pump piping, controls, and instrumentation are separated and protected so that any single physical event or missile cannot make both RHR loops inoperable.

The RHR system piping cannot be overpressurized from a single failure for the following reasons:

a. The suction piping may not be connected to the recirculation piping until the pressure has decayed to 135 psig. Also, the suction piping outside the suppression pool piping is classed as 300 lb rated.

b. The discharge piping is not overpressurized whenever the LPCI injection valve is open because a check valve between the system and the vessel blocks pressure. Leakage past the closed check valve is accommodated by relief valves. In addition, the injection valve may not be opened for testing unless the upstream valve, rated for full pressure, is also closed.

c. The heat exchanger and its piping are protected against failure of the steam pressure control valves by relief valve.

7.2.4.6 Tests and Inspections

A design flow functional test of the RHR pumps is performed for each pair of pumps during normal plant operation by taking suction from the suppression pool. The discharge valves to the RHR loops remain closed during this test, and reactor operation is undisturbed.

An operational test of the discharge valves is performed by shutting the downstream valve after it has been satisfactorily tested, thereby establishing the RHR at the downstream valve, and then operating the upstream valve. The discharge valves to the containment spray headers are checked in a similar manner by operating the upstream and downstream valves individually. All these valves can be actuated from the main control room by using remote manual switches. Control system design provides automatic return from the test to the operating mode if LPCI initiation is required during testing.

Testing of the sequencing of the LPCI mode of operation is performed after the reactor is shutdown and the RHR system has been drained and flushed. Testing of the operation of the valves required for the remaining modes of operation for the RHR is performed at this time.

7.2.5 Residual Heat Removal System for BWR/5 and BWR/6 (RHR)

7.2.5.1 Safety Design Bases

The safety related functions of the RHR system are:

a. Low Pressure Coolant Injection Mode

b. Containment Heat Removal Mode

7.2.5.2 Power Generation Design Bases

The RHR system shall be designed to meet the following power generation design bases:

a. The system shall have enough heat removal capacity to cool down the reactor to 125°F within approximately 20 hours after shutdown.

b. Fuel pool connections shall be provided so that RHR heat exchangers can be used to supplement the fuel pool cooling capacity.

c. The system shall be able to condense reactor steam generated by decay heat and direct the condensate to the suction side of the RCIC pumps.

7.2.5.3 Description

7.2.5.3.1 Summary. The RHR system combined four subsystems. The major equipment of the RHR system consists of three independent closed loops, two heat exchangers, three main system pumps, and service water pumps. The equipment is connected by associated valves and piping. Control and instrumentation are provided for correct system operation.

The main system pumps are sized for the flow required during low pressure coolant injection (LPCI) operation. The pumps are arranged and located so that adequate suction head is assued for all operating conditions. The pump motor is air cooled by the ventilation and heating system.

The heat exchangers are sized on the basis of their required duty for the post LOCA function. The heat exchanger shell and tube sides are provided with drain connections. The shell side is provided with a vent to remove noncondensable gases. Relief valves on the heat exchanger shell side, on the RHR pump discharge, and on the RCIC steam supply line protect the heat exchanger from overpressure.

The most limiting duty is that associated with the post LOCA mode. The performance of this type of heat exchanger operating in the post LOCA mode (water to water) is well established by currently operating BWR facilities.

The steam condensing mode poses a less severe duty requirement and the design is very similar to an upright feedwater service heater which has proven its reliability through many years of service.

Two loops, each consisting of a heat exchanger, main system pump, and associated piping, are located in separate protected areas of the auxiliary building. The third loop, made up of a pump and associated piping, is also located in a separate area of the auxiliary building to minimize the possibility of a single physical event causing the loss of the entire system.

7.2.5.4 Steam Condensing Mode in RCIC System

Decay heat is transferred to the service water directly in the steam condensing mode by using the RHR heat exchangers as steam condensers. The relatively cool condensate from the heat exchangers is either dumped to the suppression pool or returned to the suction side of the RCIC pump. The condensate is usually dumped in the suppression pool to prevent the less desirable water from entering the reactor core during the initial stage of operation as RCIC condensers.

When operating the RHR heat exchangers in the steam condensing mode, steam at reactor pressure and temperature is taken from the reactor vessel through a connection to the main steam supply line. The steam is then reduced to the operating pressure upstream by a pressure control valve upstream of the heat exchanger. Heat transferred to the service water is controlled by controlling the level of condensate in the heat exchanger. The water level in the heat exchanger in turn controls the flow of condensate either to the suppression pool, by a throttle valve, or to the reactor vessel by the RCIC turbine speed.

About 1-1/2 hours after reactor shutdown, one RHR heat exchanger can handle decay heat. At this time, the other RHR heat exchanger is operated to cool the suppression pool. If pool cooling is not conducted, the RCIC turbine, which continues to dump exhaust steam to the suppression pool, continues to increase the suppression pool temperature (approximately 3°F/hr).

An evaluation was made to assess whether the consequences of a single valve malfunction or operator error could result in possible damage to a heat exchanger of the RHR system while in the steam condensing mode. The evaluation examined the consequences from two aspects: 1) overpressurization and 2) hydrodynamic and concluded that the systems will respond in an acceptable manner and limit the pressure rise and loadings resulting from hydrodynamic forces to within the RHR and RCIC design limits.

7.2.5.5 Inspection and Testing

A design flow functional test of the RHR main system pumps is separately performed for each pump during normal plant operation by taking suction from the suppression pool and discharging through the test line back to the suppression pool. All other discharge valves remain closed during this test; reactor operation is undisturbed.

All motor- and air-operated valves required to operate for safety reasons, are capable of being exercised periodically during normal power operation. The layout and arrangement of critical equipment, such as drywell wall penetrations, piping, and valves, is designed to permit access for appropriate equipment used in testing and inspection system integrity.

Sequencing of the LPCI subsystems operation is tested after the reactor is shutdown and the RHR system has been drained and flushed. Valves required for remaining subsystems may be tested at this time.

Drains are provided outside the drywell wall in the piping between the isolation valves for reactor process system leakage testing. Relief valves on the low pressure lines are removable for testing. A line is provided on the pump discharge line to take water samples.

Periodic inspection and maintenance of the main system pumps, pump motors, and heat exchangers are conducted in accordance with the manufacturer's instructions.

Preoperational tests are conducted during the final stages of plant construction prior to initial startup. These tests assure correct functioning of all controls, instrumentation, pumps, piping and valves. System reference characteristics such as pressure differentials and flow rates are documented during the preoperational testing and are used as base points for measurements obtained in subsequent operational tests.

For the suppression pool cooling system the preoperational tests verify that the RHR heat exchanger shell side design flow rate can be obtained while circulating water from the suppression pool, through the RHR pump, the RHR heat exchanger and back to the suppression pool. During the test, head versus flow curves are developed for reference in evaluating the future performance of the suppression pool cooling mode and the RHR pumps.

During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the containment can be inspected visually at any time. Components inside the containment can be inspected only when it is open for access. Testing frequencies are correlated with testing frequences of the associated controls and instrumentation. When a pump or valve control is tested, the operability of that pump or valve and its associated instrumentation is tested by the same action. When a system is tested, operation of the component is indicated by installed instrumentation. Relief valves are removed as scheduled at refueling outages for bench tests and setting adjustments.

7.2.6 Condensate and Feedwater System

The purpose of the condensate and feedwater system is to deliver condensate from the condenser to the reactor. This subsection discusses the condensate and feedwater system from the condenser to the outermost feedwater shutoff valve.

7.2.6.1 Design Bases

7.2.6.1.1 Safety Design Bases. The condensate and feedwater system is not required to affect or support safe shutdown of the reactor or to perform in the operation of reactor safety features.

The condensate and feedwater system is designed with necessary shielding and controlled access to protect plant personnel.

7.2.6.1.2 Power-Generation Design Bases. The condensate and feedwater system provides a dependable supply of high quality feedwater to the reactor. The system provides the required flow at the required pressure and temperature to the reactor, allowing sufficient margin to allow continued flow under anticipated transient conditions.

The feedwater system supplies the reactor with feedwater at a minimum pressure of 1000 psia from the reactor feed pumps. This system has sufficient capacity to provide at least 115% of the feedwater required for reactor rated flow. The feedwater heaters provide the required temperature of feedwater to the reactor with six stages of closed feedwater heating. The final feedwater temperature is 420°F.

Pumped-forward heater drains are sufficiently deaerated in the shells of the pumped (third stage) feedwater heaters to maintain a level of 70 ppb (or less) oxygen content in the final feedwater supplied to the reactor during normal full load operation.

To minimize the corrosion product input to the reactor, a unit startup line is provided from the reactor feedwater supply lines, downstream of the high-pressure feedwater heaters, to the condenser.

All components of the condensate and feedwater system that contain the system pressure are designed and constructed in accordance with the applicable codes.

7.2.6.2 System Description

The condensate and feedwater system consists of the piping, valves, pumps, heat exchangers, controls, instrumentation, and the associated equipment and subsystems that supply the reactor with heated feedwater in a closed steam cycle using regenerative feedwater heating.

The condensate and feedwater system is a six-heater regenerative feedwater heating cycle.

The low pressure feedwater heaters are divided into three 1/3-capacity parallel systems. The high pressure heaters are divided into two 1/2-capacity parallel systems.

The final feedwater temperature is 420°F at rated unit output. The two lowest pressure heaters are located in the condenser exhaust neck.

Condensate from the condenser hotwell is pumped by four motor-driven condensate pumps (one spare). The condensate is prepared through the steam jet air ejector, the gland steam packing exhauster, the offgas condenser, and the condensate cleanup system, and then to the suction of the condensate booster pumps.

Four motor-driven condensate booster pumps are provided (one spare). The booster pumps provide the required head to pump the condensate through the five low pressure heaters and to provide sufficient excess head to assure proper suction head on the reactor feedwater pumps.

Two turbine-driven reactor feedwater pumps are provided. Minimum flow through the reactor feed pumps is controlled by utilizing a recirculation control valve located in the pump discharge lines to permit a recirculation of feedwater to the condenser.

7.2.6.5 Instrumentation Application

Feedwater flow control instrumentation measures the feedwater flow rate from the condensate and feedwater system. This measurement is used by the feedwater control system that regulates the feedwater flow to the reactor to meet system demands.

Instrumentation and controls regulate pump recirculation flow rate for condensation pumps, condensate booster pumps, and reactor feedwater pumps.

7.2.7 Condensate and Storage Transfer System

7.2.7.1 Design Bases

The condensate storage and transfer system is designed:

a. to store condensate for the RCIC and HPCI systems

b. to maintain the level of condensate in the condenser hotwell

c. to provide condensate to other plant systems where required

7.2.7.2 System Description

The condensate storage system consists of a 500,000 gallon stainless steel storage tank, two 500 gpm condensate transfer pumps, and the necessary piping and instrumentation to convey and monitor the water to various systems.

The condensate storage tank is a covered atmospheric storage tank located outdoors and built to the requirements of ASME Section III Class 3. With the exception of small instrument connections, a drain line, which is normally closed by a valve and a blind flange, and RCIC and HPCI suction connections to the tank, all other lines terminate inside the tank above the 100,000 gallon level to ensure that RCIC and HPCI systems are not deprived of their minimum reserve storage requirements by other less essential systems. An overflow connection on the tank is piped to the radwaste system waste surge tank.

The single condensate transfer pump will furnish condensate water to various equipment in the reactor and radwaste building except for the RCIC, HPCI, CRD, core spray, and condenser hotwell transfer lines which draw directly from the tank. The introduction of a low-pump-discharge-pressure signal will automatically start the standby pump and simultaneoiusly initiate an alarm in the main control room. To accelerate the filling of the reactor well and drywell-separator pool during refueling, both transfer pumps are operated in parallel. A level controller automatically maintains the level in the tank from 38 feet to 40 feet above the tank bottom through the addition of demineralized water makeup. High tank level (43 feet above the tank bottom) will alarm in the main control room and water level in the tank is continuously recorded. Should the level in the tank be depleted far below the 100,000 gallons minimum required by the HPCI system, a low level signal (set at 12 inches above the tank bottom) automatically switches the HPCI pump suction to the suppression pool. Pressure gauges are located at various points in the condensate transfer system for convenience in checking the operating conditions.

7.2.7.3 Safety Evaluation

The operation of the condensate transfer system is not a safety-related system.

The condensate storage tank is the initial source of water for the RCIC and HPCI systems. By providing standpipes inside the tank for outlet lines designated for other systems, the RCIC and HPCI systems are assured of a 100,000 gallon reserve. Should the water supply in the tank be depleted far below the minimum 100,000 gallons, through operation of the HPCI or RCIC systems, or through leakage, a low-level signal corresponding to 12 inches above the tank bottom automatically shifts the HPCI pump suction path to the suppression pool.

Plant administrative control will limit the radioactivity level in the tank to 10^{-3} Ci/cc. With this level of activity, the dose at the site boundary to an individual due to direct radiation from the tank will not exceed the requirements of 10CFR20.

7.2.7.4 Tests and Inspections

The condensate transfer pumps shall be proven operable by virtue of being in service during normal plant operations and by rotating the operation of the pumps periodically. The tank can be visually inspected for leakage after filling. Routine visual inspection and checking of components, instrumentation, and alarms are adequate to verify systems operability.

7.2.8 Main Steam Line Isolation Valves (MSIV)

7.2.8.1 Safety Design Basis

The MSIVs, individually or collectively, are designed to:

a. close the main steamlines within the time established by design basis accident analysis to limit the release of reactor coolant.

b. close the main steamlines slowly enough that simultaneous (inadvertent) closure of all steamlines will not exceed NSSS design limits.

c. close the main steamlines when required, despite single failure in either valve or in the associated controls, to provide a high level of reliability for the safety function.

d. use separate energy sources as the motive force to independently close the redundant isolation valves in the individual steamlines.

e. use local stored energy (compressed air and springs) to close at least one isolation valve in each steam pipeline without relying on the continuity of any variety of electrical power to furnish the motive force to achieve closure.

f. be able to close the steamlines, either during or after seismic loadings, to ensure isolation if the nuclear steam boundary is breached.

g. have the capability for being tested, during normal operating conditions, to demonstrate that the valves will function.

7.2.8.2 Description

Two isolation valves are welded in a horizontal run of each of the four main steam pipes. One valve is as close as possible to the primary containment barrier and inside it, and the other is just outside the barrier. When closed, the valves form part of the nuclear system process barrier for openings outside the containment and part of the pressure barrier for nuclear system breaks inside the containment.

Figure 5.5-6 shows an MSIV. Each is a 24-inch, Y-pattern globe valve. Design steam flow rate through each valve is 2.74×10^6 lb/hr. The main disc or poppet is attached to the lower end of the stem and moves in guides at a 45-degree angle from the inlet pipe. Normal steam flow tends to close the valve, and higher inlet pressure tends to hold the valve closed.

Operating air is supplied to the valves from the plant compressed-air system or nitrogen supply through a check valve. An accumulator tank between the control valve and the check valve provides backup operating air.

In the worst case conditions of the main steam line rupturing downstream of the valve, steam flow would quickly increase to 200% of rated flow. Further increase is prevented by the venturi flow restrictor upstream of the valves.

During appproximately the first 75% of closing, the valve has little affect on flow reduction because the flow is choked by the venturi restrictor upstream of the valves. After the valve is approximately 75% closed, flow is reduced as a function of the valve area versus travel characteristic.

Design specification ambient conditions for normal plant operation are 135°F normal temperature, 150°F maximum temperature, 100% humidity, in a radiation field of 15 R/hr due to radiation gamma and 25 R/hr due to neutron-plus-gamma radiation, continuous for design life. The inside valves are not continuously exposed to maximum conditions, particularly during reactor shutdown, and valves outside the primary containment and shielding are in ambient conditions that are considerably less severe.

The MSIVs are designed to function under the following environmental conditions:

a. 340°F for 1 min at 65 psig.

b. 340°F for 3 hours at 45 psig.

c. 320°F for an additional 3 hours at 45 psig.

d. 250°F for an additional 24 hours at 25 psig.

e. 200°F for an additional 100 days at 20 psig.

The valve is operated by pneumatic pressure and by the action of compressed springs. The control unit is attached to the air cylinder. This unit contains three types of control valves: pneumatic, ac control system A, and ac control system B. These control valves open and close the main valve and exercise it at slow and fast speed. Remote manual switches in the main control room enable the operator to operate the valves.

After the valves are installed in the NSSS, each valve is tested several times in accordance with the preoperational and startup test procedures. Two isolation valves provide redundancy in each steamline so that either can perform the isolation function, and either can be tested for leakage after the other is closed. The inside valve and outside valve and their respective control systems are separated physically.

The isolation valves and their installation are designed as seismic Category I equipment.

Electrical equipment that is associated with the isolation valves and that operates in an accident environment is limited to the wiring, solenoid valves, and position switches on the isolation valves.

7.2.8.4 Tests and Inspections

The MSIVs can be functionally tested for operability during plant operation and refueling outages. During a refueling outage the MSIVs can be functionally tested, leak tested, and visually inspected. The MSIVs can be tested and exercised individually to the 90% open position because the valves still pass rated steam flow when 90% open.

The MSIVs can be tested and exercised individually to the fully closed position if reactor power is reduced sufficiently to avoid scram from reactor overpressure or high flow through the steamline flow restrictors.

7.2.9 Safety Relief Valve System (S/RV)

7.2.9.1 Safety/Relief Valve Sizing

Sizing of the safety/relief valve capacity is based on establishing an adequate margin from the peak vessel pressure to the vessel code limit (1375 psig) in response to a specified transient. General Electric design practice and ASME Code requirements are satisfied with the closure of all MSIVs with scram tripped by a high-neutron flux signal as the reference transient. The minimum capacity determined according to the specified criteria is translated into a discrete valve requirement and compared with the total number of valves required to meet the availability index criterion.

The safety/relief valve capacity required to provide overpressure protection at all levels of indirect scram is derived from an evaluation of the MSIV-pressure scram transient.

7.2.9.2 Availability Index (I_A)

The availability index is based upon the number of safety/relief valves required to provide an acceptable margin to the vessel code limit (1375 psig) for the MSIV-flux scram transient. The data employed in the derivation of the availability index is outlined as follows:

a.	Safety/relief valves total installed	11
b.	Safety/relief valves (MSIV-flux scram)	7
c.	Valve failure rate* *failures/10^6 operating hours)	1.1
d.	Testing interval (years)	$\leq$2.2
e.	Availability index	0.999999

7.2.9.3 **Safety/Relief Valve Characteristics**

7.2.9.3.1 Pressure Drop in Inlet and Discharge. Pressure drop on the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures reported above.

Pressure drop with ASME-rated flow in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent back pressure on each safety/relief valve from exceeding 40% of the valve inlet pressure; thus ensuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping. Each safety/relief valve has its own separate discharge line.

7.2.9.4 **Safety/Relief Valve Description**

These valves comply with ASME III, Paragraph N911.4(a)(1) for pilot-operated valves.

Representative quantities and set points are as follows:

Quantity	Set Point psig	ASME Rated Capacity at 103% of Set Pressure lb/hr minimum
4	1090	869,000
4	1100	876,800
3	1110	884,700

7.2.10 Standby Liquid-Control System

7.2.10.1 Design Bases

The SLCS shall meet the following safety design bases:

a. Backup capability for reactivity control shall be provided, independent of normal reactivity control provisions in the nuclear reactor, to be able to shut down the reactor if normal control ever becomes inoperative.

b. The backup system shall have the capacity for controlling the reactivity difference between the steady-state rated operating condition of the reactor with voids and the cold shutdown condition, including shutdown margin, to ensure complete shutdown from the most reactive condition at any time in core life.

c. The time required for actuation and effectiveness of the backup control shall be consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions.

d. Means shall be provided by which the functional performance capability of the backup control system components can be verified periodically under conditions approaching actual use requirements. Demineralized water, rather than the actual neutron absorber solution, can be injected into the reactor to test the operation of all components of the redundant control system.

e. The neutron absorber shall be dispersed within the reactor core in sufficient quantity to provide a reasonable margin for leakage or imperfect mixing.

f. The system shall be reliable to a degree consistent with its role as a special safety system; the possibility of unintentional or accidental shutdown or the reactor by this system shall be minimized.

7.2.10.2 Description

The SLCS is manually initiated from the main control room to pump a boron neutron absorber solution into the reactor if the operator believes the reactor cannot be shut down or kept shut down with the control rods.

The SLCS is required only to shut down the reactor and to keep the reactor from going critical again as it cools.

The SLCS is used only in the highly improbable event that not enough control rods can be inserted in the reactor core to accomplish shutdown and cooldown in the normal manner.

The boron solution tank, test water tank, the two positive-displacement pumps, the two explosive valves, and associated local valves and controls are mounted in the reactor building. The solution is piped into the RPV and discharged near the bottom of the core shroud so it mixes with the cooling water rising through the core.

The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the uranium fuel.

The specified neutron absorber solution is sodium pentaborate ($Na_2B_{10}O_{16}10H_2O$). It is prepared by dissolving stoichiometric quantities of borax and boric acid in demineralized water. A sparger is provided in the tank for mixing, using air. To prevent system plugging, the tank outlet is located above the bottom of the tank.

Whenever it is possible to make the reactor critical, the SLCS shall be able to deliver enough sodium pentaborate solution into the reactor to ensure reactor shutdown. This is accomplished by placing sodium pentaborate in the standby liquid control tank and filling with demineralized water to at least the low-level-alarm point. The solution is at design concentration at the low-level-alarm point and may be diluted with water up to within 6 inches of the overflow level volume to allow for evaporation losses or to lower the saturation temperature.

Heat tracing is run along the pipe from the tank to the pump suction to maintain the solution temperature within 75 to 85°F. The heating element is controlled by a thermostat. A temperature switch in the suction line will also actuate the low-solution-temperature alarm in the main control room. A heater system in the tank maintains the solution temperature at 75 to 85°F to prevent precipitation of the sodium pentaborate from the solution during storage. High or low temperature or high or low liquid level causes an alarm in the main control room.

Each positive-displacement pump is sized to inject the solution into the reactor in 50 to 125 minutes depending on the amount of solution in the tank. The pump and system design pressure between the explosive valves and the pump discharge is 1400 psig. The two relief valves are set slightly under 1400 psig to exceed the reactor operating pressure by a sufficient margin to avoid valve leakage. The relief valves are installed with the discharge lines flooded to prevent evaporation and precipitation within the valve. To prevent bypass flow from one pump in case of relief valve failure in the line from the other pump, check valve is installed downstream of each relief valve line in the pump discharge pipe.

The two explosive-actuated injection valves provide assurance of opening when needed and ensure that boron does not leak into the reactor even when the pumps are being tested.

Each explosive valve is closed by a plug in the inlet chamber. The plug is circumscribed with a deep groove so the end readily shears off when pushed with the valve plunger. This opens the inlet hole through the plug. The sheared end is pushed out of the way in the chamber; it is shaped so it does not block the ports after release.

The shearing plunger is actuated by an explosive charge with dual ignition primers inserted in the side chamber of the valve. Ignition circuit continuity is monitored by a trickle current, and an alarm occurs in the main control room if either circuit opens. Indicator lights show which primer circuit opened. To service a valve after firing, a six-inch length of pipe (spool piece) must be removed immediately upstream of the valve to gain access to the shear plug.

The SLCS is actuated by a three-position keylocked switch on the main control room console. This ensures that switching from the off position is a deliberate act. Switching to either side starts an injection pump, actuates both of the explosive valves, and closes the reactor cleanup system outboard isolation valve to prevent loss of dilution of the boron.

A green light in the main control room indicates that power is available to the pump motor contactor and that the contactor is open (pump not running). A red light indicates that the contactor is closed (pump running).

Instrumentation consisting of solution temperature indication and control, tank level, and heater system status is provided locally at the storage tank.

7.2.10.3 Safety Evaluation

The SLCS is a reactivity control system and is maintained in a standby operational status whenever it is permissible for the reactor to be critical. The system is expected never to be needed for safety reasons because of the large number of independent control rods available to shut down the reactor.

However, to ensure availability of the SLCS, two sets of the components required to actuate the system (pumps and explosive valves) are provided in parallel redundancy.

The specified minimum average concentration of natural boron in the reactor to provide the specified shutdown margin, after operation of the SLCS, is 660 ppm. Calculation of the minimum quantity of sodium pentaborate to be injected into the reactor is based on the required 660 ppm average concentration in the reactor coolant, including recirculation loops, the RHR system in the shutdown cooling mode at 70°F, and reactor normal water level. The result is increased by 25 percent to allow for imperfect mixing and leakage and to account for the volume in other small piping connected to the reactor. An additional 250 ppm is provided to accommodate dilution by the RHR system in the shutdown cooling mode. This concentration will be achieved if the solution is prepared as defined above and maintained above saturation temperature as defined by Figure 4.2-25.

The SLCS equipment essential for injection of neutron absorber solution into the reactor is designed as seismic Category I for withstanding the specified earthquake loadings. Nonprocess equipment such as the test tank is not seismic Category I.

The SLCS is required to be operable in the event of an offsite power failure. Therefore, the pumps, heaters, valves, and controls are powered from the standby ac power supply. The pumps and valves are powered and controlled from separate buses and circuits so that a single active failure will not prevent system operation.

The SLCS and pumps have sufficient pressure margin, up to the system relief valve setting of approximately 1400 psig, to ensure solution injection into the reactor above the normal pressure in the bottom of the reactor. The nuclear system relief and safety valves begin to relieve pressure above approximately 1100 psig. Therefore, the SLCS positive-displacement pumps cannot overpressurize the nuclear system.

7.2.10.4 Tests and Inspections

Operational testing of the SLCS is performed in at least two parts to avoid inadvertently injecting boron into the reactor.

With the valves from the storage tank and to the reactor closed and the three valves to and from the test tank opened, demineralized water in the test tank can be recirculated by locally starting either pump.

During a refueling or maintenance outage the injection portion of the system can be functionally tested by valving the suction lines to the test tank and actuating the system from the main control room. Both injection valves open on actuation. System operation is indicated in the main control room.

After functional tests, the injection valve shear plugs and explosive charges must be replaced and all the valves returned to their normal positions as indicated in Figure 4.2-24.

After closing a local locked-open valve to the reactor, leakage through the injection valves can be determined by opening valves at a test connection in the line between the containment isolation check valves. Position-indicator lights in the main control room indicate that the local valve is closed for test or open and ready for operation. Leakage from the reactor through the first check valve can be detected by opening the same test connection when the reactor is pressurized.

The test tank contains demineralized water for approximately 3 minutes of pump operation. Demineralized water from the makeup system or the condensate storage system is available for refilling or flushing the system.

Should the boron solution ever be injected into the reactor, either intentionally or inadvertently, after making certain that the normal reactivity controls will keep the reactor subcritical, the boron is removed from the reactor coolant system by flushing for gross dilution followed by operating the reactor water cleanup (RWCU) system.

The concentation of the sodium pentaborate in the solution tank is determined periodically by chemical analysis.

7.2.10.5 Instrumentation

The instrumentation and control system for the SLCS is designed to allow the injection of liquid poison into the reactor and the maintenance of the liquid poison solution well above the saturation temperature.

7.2.10.6 Logic and Sequencing

When the SLCS is initiated, both the explosive valves fire and the pump that has been selected for injection starts.

There are no bypasses. When the SLCS is initiated to inject soluble neutron absorber into the reactor, the outboard isolation valve of the RWCU is automatically closed.

7.2.10.7 Redundancy and Diversity

The redundancy exists in duplicated pumps, explosive valves, check valves, relief valves, and power supplies.

7.2.10.8 Actuated Devices

When the SLCS is initiated to inject soluble neutron absorber into the reactor, one of the two injection pumps, and both the explosive valves are actuated.

7.2.10.9 Supporting Systems

The power supply to explosive valve F004A and injection pump C001A is from essential 600 volt bus 2C. The power supply to explosive valve F004B and injection pump C001B is from essential 600 volt bus 2D. The power supply to the tank heaters and heater controls is connected to the essential 600 volt buses 2C and 2D.

7.2.10.10 Operational Considerations

The control scheme for the SLCS can be found in Figure 3.4.3-2. The standby liquid control is manually initiated in the main control room by inserting the proper key into the keylocking switch and turning it to either system A or system B. When the injection is completed, the system is manually turned off by returning the keylocking switch to the OFF position.

The provisions taken in accordance with General Design Criterion 19 of 10 CFR 50 Appendix A to provide the required equipment outside the main control room for hot and cold shutdown is described in Section 7.5.

7.2.10.11 Operator Information

The SLCS indicators are as follows:

a. The system pressure is indicated with an indicator that has a range of 0 to 1800 psig in the main control room.

b. The storage tank level is indicated with an indicator that has a range of near empty to near full, calibrated to read in gallons of liquid storage in the main control room.

c. The continuity of the explosive valve dual primer ignition circuit is monitored by measuring a trickle current through the primers. If either of the dual primer or the primer ignition circuits become open circuited, the continuity meter reads downscale.

d. Indicator lights in the main control room show if either of the pumps is running, stopped, or tripped.

e. Indicator lights in the main control room show if either of the explosive valves' firing circuit has an open circuit or not.

f. Indicator lights in the main control room show if service valve F008 is open or closed.

g. Indicator lights on the local panel show if the manually-controlled storage tank heater for solution mixing is on or off.

h. Indicator lights on the local panel show if the thermostatically controlled storage tank heater for maintaining solution temperature is on or off.

The SLCS main control room annunciators annunciate when:

a. The loss of continuity of either explosive valve primers activates a main control room annunciator.

b. The standby liquid storage temperature becomes too hot or too cold.

c. The standby liquid tank level is too high or too low.

APPENDIX 7.3
BWR SCRAM SYSTEM RELIABILITY SUMMARY*

7.3.1 BWR Scram System Summary

The scram system for General Electric BWRs is composed of the following elements:

a. Sensors of reactor parameters

b. Reactor Protection System (RPS) logic which processes the input from the sensor

c. Hydraulic Control Units (HCUs) (one per control rod drive) with pneumatically controlled valves which open to cause the control rod drives to scram

d. Control Rod Drives whose pistons provide the motive force for inserting the control blades into the reactor

e. Air supply header which provides the air pressure which keeps the scram inlet and discharge valves in the HCUs closed until a scram signal isolates it from the HCUs

f. Scram discharge volume which receives the water discharged by the control rod drives during a scram

g. Air supply exhaust valves, which open for all scram signals to depressurize the air supply header, and provide a backup means of opening the scram valves.

The normal sequence of events during a BWR scram are as follows:

a. The Reactor Protection System logic processes a signal from a set of sensors which indicate a scram is required. This signal deenergizes the solenoids on the three way pilot valves (139a) on all the HCUs

*loc cit page 2-3

(see Figure 7.3.1) which isolates the HCUs from the scram air header and depressurizes the pneumatic lines in all HCUs.

b. The three-way valves in the scram air header (F11A and F110B) are energized which isolates the scram air header from plant instrument air and depressurizes the scram header.

c. Depressurization of the pneumatic lines on the HCUs open exhaust valve 135b which accelerates the depressurization of the pneumatic line to the scram inlet and discharge valves (No. 126 and No. 127 on Figure 7.3.1).

d. When the pneumatic lines to valves No. 126 and No. 127 are depressurized, the valves open, allowing the high pressure water from the HCU accumulators to push up the piston in the control rod drives, while discharging water to the scram discharge volume.

If the HCU pilot valves (No. 139a) were to fail, the depressurization of the scram air header would eventually cause valve No. 139b to open, triggering actuation of the two scram valves (No. 126 and No. 127). If exhaust valve (No. 139b) were to fail also, the depressurization of the pneumatic lines would still take place through the scram air header, but at a slower rate. If the scram inlet valve (No. 126) were to also fail, reactor pressure on the bottom side of the CRD piston is capable of driving the control rod into the reactor. Thus, it can be seen that there is considerable redundancy and diversity in the scram mechanism, especially when it is recognized that each HCU Control rod drive pair is independent from the others.

7.3.2 Scram System Reliability Study Results Summary

1. The existing BWR scram system has an unreliability of 0.9×10^{-6}/demand (including common cause failure).

2. A limiting factor in the reliability of the existing scram system is the common miscalibration of sensors which initiate scram. Having three different types of sensors protects against miscalibration through diversity (see Table 1).

3. The control rod drives each have an unreliability of $3X10^{-9}$/demand for preventing reactor shutdown.

4. Fifty percent of the control rod drives can fail to insert in a checkerboard pattern and the reactor will still shut down. Seventy five percent of the control rod drives can fail to insert and the reactor will be brought to near-decay heat levels.

5. Thirty one percent of the control rod drives can fail to insert in a random pattern and the reactor will still shut down.

6. Five or more control rod drives must fail to insert in a cluster before reactor shutdown capability is lost.

7. The probability is less than 10^{-8}/year that blockage in the scram discharge headers or HCU discharge lines, will be sufficient to prevent reactor shutdown.

8. Periodic overhaul of drives, rod motion tests and scram insertion time tests provide adequate early warning of an impending failure mechanism which could potentially result in failure to scram.

9. Common cause failures do not usually incapacitate an entire system.

10. Common cause failures are detectable.

11. Common cause failures can be analyzed quantitatively.

12. Common cause failures do not usually occur simultaneously; rather, they occur progressively over a finite period of time.

13. There are many licensing requirements already in existence which are directed toward reducing common cause failures in nuclear power plants.

In developing numerical results, random and common mode failure probabilities were combined by taking the log normal mean as in WASH-1400. Subsequent review of this approach has raised questions as to its justification when independent and dependent failure modes differ by many orders of magnitude. General Electric is aware of this criticism but believes that results generated are supported by engineering judgment. The 12 man-year effort to determine scram system unreliability was not simply a numerical analysis of the statistics but included a thorough failure mode and effects analysis, common cause analysis and a thorough review of operating experience, examining all documented component failures. These served as bases for the quantified study using fault trees and event trees. Each quantitative assessment was examined against prudent engineering judgment to assure the reasonableness of the results. The operating experience also does not preclude the assessed probabilities given in the analysis. Therefore, a change in methodology would not substantially modify the conclusions listed above.

7.3.3 RPS Summary

The RPS is a shutdown actuating system which prevents the reactor from operating under unsafe or potentially unsafe conditions by automatically initiating rapid insertion (scram) of the control rods. The RPS is designed to be fail-safe, highly redundant, and provides the highest practical degree of plant safety.

Some of the input parameters to the scram logic are reactor vessel level, vessel pressure, neutron flux, main steam line high radiation, main steam line isolation valve position, turbine control valve fast closure, and turbine stop valve position. There are four redundant sensor input signals from each of these parameters.

The input signals are processed through the appropriate signal conditioning equipment for either the relay or solid state logic designs. The conditioned signal outputs are used to automatically initiate scram of the control rods. The control rods within the reactor core are arranged into four groups to provide a "checkerboard" pattern. The control rod insertion pattern is near optimum for reactor physics control in event one or more control rod groups fail to insert.

The redundancy and diversity provided by the prompt scram logic and backup scram logic provide a high degree of assurance that the probability of a single point failure which disables the scram function is extremely remote. The RPS is a fail-safe system both for random mode and common cause failures.

7.3.4 Diversity Within RPS Signals

Table 7.3.1 is a list of transient events which are expected to occur more frequently than once in 40 years that require scram for the consequences to be acceptable. This table illustrates the diversity of the signals used to actuate the normal scram system. This diversity provides protection against total failure by common design defect, manufacturing error or calibration error. Table 1 also shows the transients and the order of the scram signals for ARI generated for each transient. It can be seen in Table 7.3.1 that for every transient, except number 5, there is at least one scram trip from each diverse category. Transient number 5 will receive two diverse inputs out of the three possible. Therefore, diversity of the RPS sensor output is achieved because of the diverse input device trips that operate in diverse environments, and calibration procedures and calibration standards. Table 7.3.1 also shows all the significant transients that will generate an ARI sensor trip given that normal scram does not occur. There are three fundamental generic types of sensor inputs into the normal scram system that are of interest for these transients. These generic types of inputs are:

Pressure and Differential Pressure Sensors

Position and Micro-Switches
- -MSIV
- -Turbine control valve
- -Turbine stop valve

Radiation Sensors
- -Average Power Range Monitor (APRM)
- -Main steam line radiation monitor

7.3.5 RPS Power Supply Diversity (Solid State)

Each of the four separate division's input cabinets contain a-c to d-c power supplies needed to operate that division's RPS circuits. Should power fail in one division, only that division will be affected. Its consequences will be half scram caused by deenergizing one out of the four logic strings which will deenergize one of the two scram solenoids at each HCU. Before entering the logic circuits all input signals are isolated and buffered to help insure noise-free operation. The power supplies are operated well below their 100% capability for all modes of operation. The input power to the power supply originates from the station batteries that drive a d-c/a-c power inverter in each division. This scheme provides nearly infinite isolation between divisions so a common occurrence, such as lightning, loss of the station normal a-c power or a complete failure of one RPS division will not affect the other division.

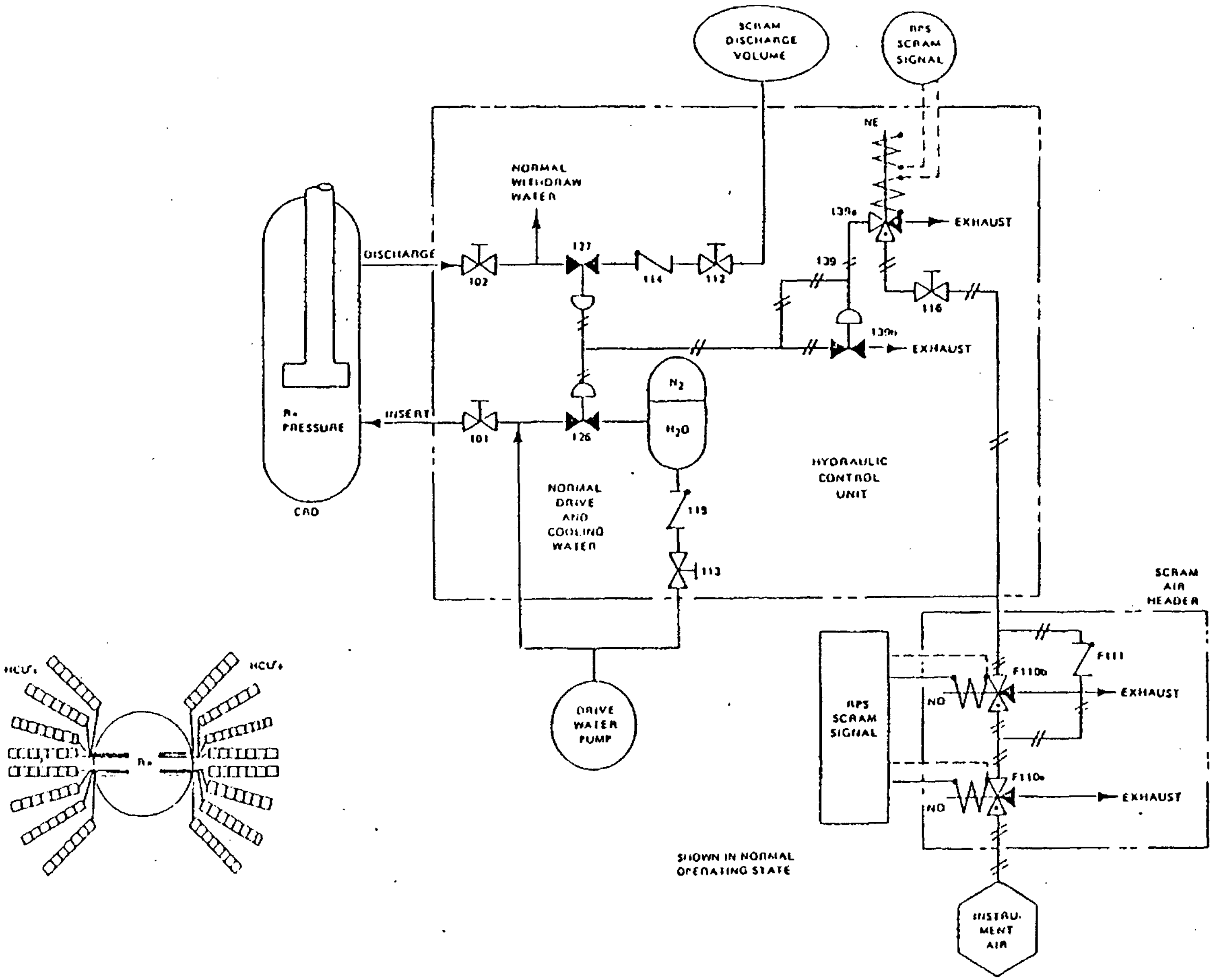

Figure 7.3.1. BWR Hydraulic Control Unit

Table 7.3.1

SENSOR DIVERSITY FOR MAJOR TRANSIENTS

		Scram Signals - Order of Occurrence								
		Inputs from Pressure or Differential Pressure Transmitters and Trip Units		Inputs from Position or Micro Switch Contact Opening			Inputs from Radiation Sensors		ARI Variables Reached	
	Transient	Rx Pressure >1065 psig	Rx Level <Level 3	Turb Cont. Valve Oil Pres. Set Point	Turb Stop Valve Pos. <90% Full Open	MSIV Pos. <90% Full Open	APRM >120%	MSIV Hi Rad. >6 x Back	Rx Pressure >1150 psig	Rx Level <Level 2
1	MSIV Closure	3	4			1	2		1	2
2	Turb Trip (with bypass)	3			1		2		1	M
3	Generator Trip (with bypass)	3		1			2		1	M
4	Pres. Regulator Failure (primary pressure decrease)	3	4			1	2		1	2
5	Pres. Regulator Failure (primary pressure increase)	2					1		1	
6	F.W. Flow Control Failure (reactor Water inventory increase)	3			1		2		1	
7	F.W. Flow Control Failure (reactor water inventory decrease)	3	1			2		4	2	1
8	Loss of Condenser Vacuum	3		4	1	5	2		1	M
9	Loss of Normal AC Power	4	5	2	1	6	3		1	M

M = The Trip Value may be reached.

APPENDIX 7.4

QUENCHER PERFORMANCE MEMORANDUM

Figure 7.4.1. Evolution of the General Electric Quencher Device

Figure 7.4.2. Results from Licensee Tests of Various Hole Patterns on Pipe Segment (Company Proprietary)

Figure 7.4.3. Licensee Full-Scale Development Test Tank and Specimen Geometries
(Company Proprietary)

Figure 7.4.4. Hole Sizes and Spacings Used on Licensee Quencher Development Tests (Company Proprietary)

Figure 7.4.5. Floor Pressure As a Function of Local Subcooling For Licensee Full-Scale Development Tests (Subcooling Based on T_{SAT} = 233°F at 17.9 ft. Submergence) (Company Proprietary)

Figure 7.4.6. Condensation Regimes Observed in Licensee Full-Scale Development Tests (Subcooling Based on T_{SAT} = 233°F at 17.9 ft. Submergence) (Company Proprietary)

Figure 7.4.7. Range of Test Conditions for Licensee Full-Scale Quencher Condensation Development Tests (Subcooling Based on T_{SAT} = 233°F at 17.9 ft. Submergence) (Company Proprietary)

Figure 7.4.8. Licensee Full-Scale In-Plant Test Pool and Quencher Geometries
(Company Proprietary)

Figure 7.4.9. General Electric Quencher Geometries

(Company Proprietary)

Figure 7.4.10. Results from Licensee Test on Segment of Full Scale Hole Array in Small Scale Tank (Company Proprietary)

APPENDIX 7.5
RELATIONSHIP BETWEEN PCI AND BOILING TRANSITION

7.5.1 NRC Assessment of GE Position

An evaluation by NRC of BWR ATWS events in NUREG-0460 generally agreed with the damage limit presented by General Electric in the main body of this report. However the NRC staff disagrees with General Electric's assessment of pellet-cladding interaction. They believe that as ATWS events involve reactivity initiated power increases, fuel rod failures may result from pellet cladding interaction (Reference 7.5.1).

The NRC is currently attempting to develop a generic PCI model for safety analyses. However, this model is not complete and therefore they do not at this time have an applicable licensing acceptance criteria for PCI. In order to overcome this limitation the NRC Staff has attempted to use a completely different phenomena, boiling transition, to encompass the number of rods estimated to fail as a result of both boiling transition and PCI combined (Reference 7.5.1).

General Electric is in disagreement with this combined failure criterion and in this appendix will demonstrate with a considerable amount of experimental evidence that neither PCI nor boiling transition should be considered as failure mechanisms for an ATWS event.

7.5.2 Effect of PCI and Boiling Transition

In addition to the mechanical effects, the possibility that stress dependent fuel rod internal environmental effects may be a contributing factor to PCI has long been suspected.

General Electric failure mechanism tests and comparisons of PCI failure surface morphologies with out-of-pile stress corrosion cracking and liquid metal embrittlement tests yield undeniable evidence that these environmental effects do play a necessary role in PCI failures. GETR tests and field experience have identified important parameters in determining fuel rod failure susceptibility and the relative importance of mechanical localized strains and environmental effects on fuel rod fractures. These parameters include magnitude of power increase, rate of power increase, power after the increase, hold time at the increased power level and exposure.

To assess the effect on PCI of abnormal operating transients (AOTs) both with and without scram, the events have been examined in some detail to determine if they possess the conditions defined by the previously mentioned important parameters requisite for PCI fuel failure. Based on such an assessment, it is concluded that the ATWS events analyzed are not calculated and are not expected to result in any significant number of PCI failures. Available data from experimental and commercial operating reactors indicate that fuel failures due to PCI are likely to occur after a rapid power increase only if the fuel remains at the higher power for a relatively long period of time (many minutes to many hours). Most of the defined AOTs, however, are of very short duration (3 to 5 seconds at the overpower conditions), and therefore, do not fulfill the hold-time condition which is associated with PCI induced failures. The most quantitative source of data on the hold-time requirement is the Canadian experience which identifies four parameters of failure: maximum power achieved, size of power increase, fuel burnup at the time of the power increase, and duration of the power increase. Other sources suggesting the hold-time characteristic include the CIRENE overpower tests where failures were delayed 1/2 to 5 hours in six of seven tests; GETR tests where failures were delayed from the precipitating ramp typically from many minutes to many hours at the peak power condition; and offgas spikes which have been observed to be delayed in time from the initiating events in operating reactors.

Figure 7.5.1 (from Reference 7.5.2) provides additional information to demonstrate that PCI failures due to hold time at an increased power level are not expected to occur during an ATWS event. This figure shows that short duration overpower transients, such as an ATWS event, do not produce any significant failures below ~10 minutes. The maximum overpower duration for the ATWS events considered has been identified as 5 seconds.

One of the main NRC concerns was that PCI failures are more likely to occur during power increase events than reduction in flow events (Reference 7.5.1). In the former case the fuel pellets heat up and expand more rapidly than the cladding. In the latter type of event the cladding expands more rapidly than the fuel pellets. The recent tests carried out under the Thermal Fuels Behavior Program (References 7.5.3 and 7.5.4) is supportive of GEs position that the PCI would not be expected for the ATWS events analyzed. These tests (References 7.5.3 and 7.5.4) which were carried out on both fresh and preirradiated rods, were performed in a manner which would introduce PCI stresses by a rapid ramping of the test rods. This rapid increase in power was performed at the end of the preconditioning period to power levels of approximately double the highest power levels which the rods have been exposed to in any previous operation. These PCI stresses were maintained, except for relaxation effects, during the final hour of preconditioning. Even though these rods were operated in a boiling transition regime with peak axial powers up to 66 kW/m no failures were detected (Reference 7.5.5). Therefore, pellet cladding interaction should not be considered as a failure criteria for an ATWS event.

Considerable experimental evidence also exists that indicates fuel rods can operate in transition boiling for long periods of time without failure (Reference 7.5.1) Boiling transition tests have been carried out in the United States (Reference 7.5.6), Canada (References 7.5.7 and 7.5.8), Norway (Reference 7.5.9), and England (Reference 7.5.10). Table 7.5.1 (from Reference 7.5.11) summarizes this data. Briefly the highlights of Table 7.5.1 are as follows:

In a test performed in the General Electric Test Reactor (GETR) a fuel rod was operated well beyond the onset of boiling transition for a total period of about 1 hour and 15 minutes. Post irradiation examination revealed that perforation of the cladding did not occur.

The NRU Chalk River rods operated between 6-1/2 and 10 hours past the onset of boiling transition. The only failures occurring after multiple cycles were due to high temperature oxidation.

In tests carried out in the Halden Reactor, one bundle experienced boiling transition 33 times with no cladding degradation. While another experienced boiling transition 60 times over a period of 2-1/2 years reaching a peak burnup of 18,000 MWd/t with no fuel failure.

In the Winfrith SGHWR a 36 rod fuel cluster was operated with greater than 120 boiling transition excursions with no fuel failure.

The EG&G tests, which are probably the most detailed and closely monitored to date, indicate the capability to operate up to 8 minutes with PCT as high as 2700°F. The only rod failures were attributable to gross amounts of cladding oxidation due to the extremely high temperatures, which are not expected in an ATWS event.

The foregoing provides a strong basis to support the position that both stress corrosion cracking (PCI) and boiling transition should not be a concern for the limiting ATWS events analyzed.

7.5 REFERENCES

7.5.1 Letter, R. J. Mattson to G. G. Sherwood, with Enclosure 1 - Generic Questions, February 15, 1979.

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7.5.3 W. F. Domenico, et. al., Fuel Rod Failure During Film Boiling, Paper presented at the 1978 ANS Winter Meeting, 1978 (TANSAO-30 1-814).

7.5.4 Response of Unirradiated and Irradiated PWR Fuel Rods Tested Under Power Cooling Mismatch Conditions, January 1978, (TREE-NUREG-1196).

7.5.5 S. Levine to E. A. Case, Letter #17, Research Information - Power Burst Facility (PBF) Single Rod Power - Cooling Mismatch (PCM) Test Results, May 5, 1978.

7.5.6 T. Sorlie, Consequences of Operating a Zr-2 Clad Fuel Rod Above the Central Heat Flux, October 1965 (APED-4986).

7.5.7 R. D. Page, Engineering and performance of Canada's UO_2 Fuel Assemblies for Heavy-Water Power Reactors, Prog Symp on Heavy Water Power Reactors, Vienna, IAEA, September 1967, STI/Pub/163 Paper SM-99/48, pp. 749-771.

7.5.8 A. S. Bain, A. W. L. Segal, and J. Novak, Examination of Fuel Bundles Irradiated in Intermittent Dryout, Proc, ANS Topical Meeting on Water Reactor Fuel Performance, St. Charles, Illinois, May 9 through 11, 1977.

7.5.9 G. Kjaerheim, Heat Transfer in Water Cooled Nuclear Reactors, Nuclear Engineering and Design, Vol. 21, 1972, No. 2, pp. 279-301.

7.5.10 H. Redpath, Winfrith SGHWR In-Reactor Dry-Out Tests, J. Br. Nucl. Engrg Ser, 1974, Vol. 13, No. 1, pp, 87-97.

7.5.11 K. W. Holtzclaw and A. V. Sheshadri, General Electric BWR Fuel Integrity Limits and Criteria, Paper presented at 1978 Winter Meeting of the American Nuclear Society, Washington, D.C., November 1978.

Table 7.5.1

Summary of Characteristics of Zircaloy-Clad Rods Operating in Boiling Transition

TEST[a]	Number of Rods/Bundles	Fuel Rod o.d. (in)	Dryout Events	Integrated Boiling Transition Time (min)	PCT (Thermocouple) (°F)	PCT (Metallographic) (°F)	ZrO_2 Thickness Typical/Max (mil)	Remarks
GE (3)	Single rod	.565	1	75	b	1780-1820	Max-5	No cladding perforations. Some cladding creepdown.
NRU Chalk River (4)	180/10	.542	1	180	b	842-932	b	No evidence of rod deterioration.
NRU Chalk River (5)	126/6	.542	19	390	1292	2912	b	ThO_2-UO_2 fuel rods. Rod failure occured only after 19'th BT cycle due to heavy oxidation.
NRU Chalk River (5)	216/6	.542	74 6 post dryout	618	1292	1832	.04/39	Rod failed after 74'th BT cycle due to high temp caused by rod oxidation.
Halden (6)	7/1	.557	Several	4-5	b	1562	b	Put back into core after BT - Removed after 5 years, 20 GWd/MT. Assy intact.
Halden (6)	7/1	.55	33	b	b	b	b	No cladding failures.
Halden (6)	9/1	.482	60	b	b	b	b	Dryout events over 2' yr period. Peak burnup 18 GWd/MT. No cladding failures.
Winfrith SGHR (7,8)	36/1	.626	120 9 post dryout	>15	1112	>1022	<.04	Cladding softening but no cladding failures.
EG&G PCM (9)	16[c]	.421	94	32	1137-1907	1250-2186	.08-3.6/7.9	5 rods failed due to brittle cracking of highly oxidized cladding after shutdown and 1 rod failed during post-irradiation handling.
EG&G IE (9)	21[c]	.421	19	27	1377-1898	2016-2916	.79-4.33/9.1	Some significant fuel melting, cladding creepdown.

[a] Numbers in parentheses correspond to reference number temperatures and oxide thicknesses provided where measured.

[b] Not measured and/or reported.

[c] Single rod or 4 rod/bundle.

Table 7.5.2
SUMMARY OF FUEL INTEGRITY ANALYSIS
RESULTS FOR ATWS CASE 1

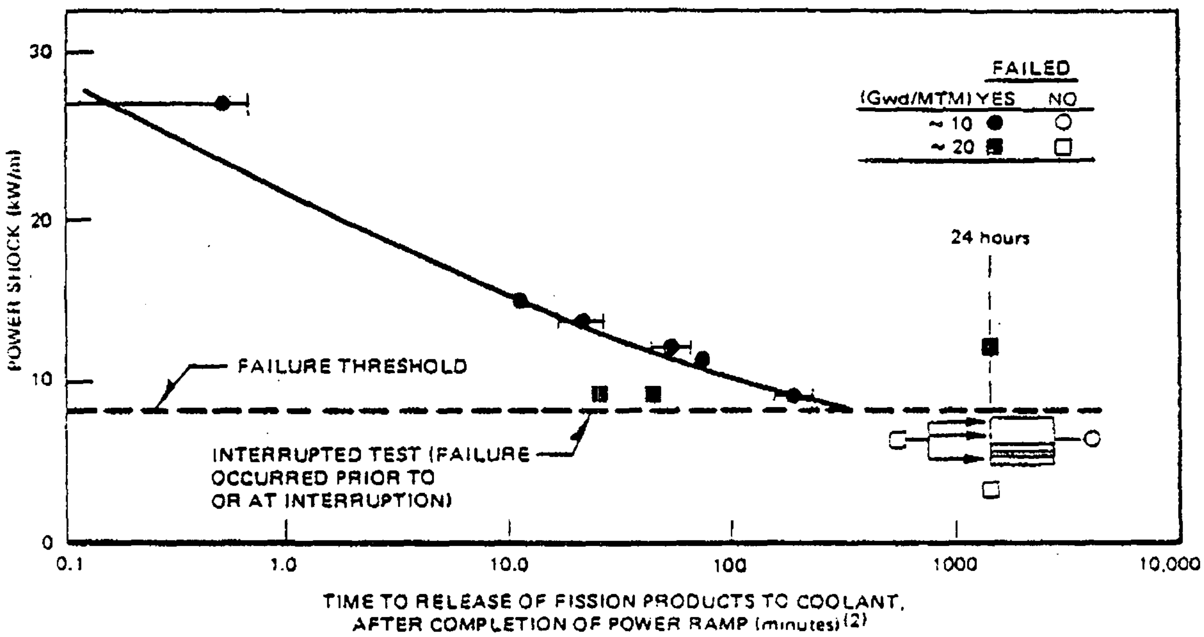

Figure 7.3.1. Studsvik Inter-Ramp (BWR Power Ramp) Project - Power Shock-Failure Correlation Based on Average Power During First High-Power Cycle of Extended Irradiation

APPENDIX 7.6
JUSTIFICATION OF BORON MIXING MODEL ASSUMPTIONS

7.6.1 Definition of Mixing Efficiency

Liquid boron injected into the reactor is most effective if it stays uniformly distributed in the core. However, since the reactor coolant keeps flowing through the core, the liquid boron will spread into the other regions of the reactor as well. Therefore, for comparison purposes a reference "perfectly mixed" condition is defined in which, at a given point in time during the ATWS, all the liquid boron is uniformly mixed throughout the reactor coolant within the reactor coolant pressure boundary. Since nonuniformities in liquid boron concentration are possible and expected, the mixed condition at any time is related to the reference condition through a mixing efficiency equal to the ratio of the average concentration of boron in the liquid part of the coolant in the core to the average concentration throughout the coolant in the RPV. The actual, time varying mass of liquid in the vessel and recirculation loops was used in calculating the vessel mixed boron concentration. When liquid boron is injected inside or near the core, the average concentration in the core could be greater than the average value in the RPV, and thus the mixing efficiency as defined above could be greater than 100%. However, as mixing and dispersal continues the efficiency will eventually approach unity.

7.6.2 Discussion of Boron Mixing Process for Injection Through the Jet Pump Instrumentation (JPI) Lines

Figure 3.4.3-2 from the main report shows the schematic of the boron injection arrangement using the JPI lines as the points of entry into the RPV. This design is applicable to BWR's which in an ATWS, supply makeup water outside the core shroud by HPCI (generally BWR/4's). The discharge side of the SLC pump is connected to the JPI lines outside the containment. These JPI lines are connected to the upper part of the jet pump diffusers inside the reactor pressure vessel. If the SLC pumps are initiated at time T_0, the liquid boron will first reach the RPV at a time T_1, where (T_1-T_0) is the "transport delay" outside the RPV and is equal to the time taken by the liquid boron flow to displace all the water in the pipeline from the pumps to the jet pump diffusers.

Figure 7.6.1 shows typical schematic details of the connection of the upper instrumentation tap with the top of the jet pump diffuser. When liquid boron is injected through this line, it issues into the diffuser flow through ten 0.09" diameter holes in the wall of the diffuser. The resulting average jet velocity (at a total flow rate of 86 GPM through all appropriate jet pumps is approximately 22 ft/sec. This high velocity combined with the effect of high Reynolds number of the flow (of the order of 10^6 with 10% of the rated flow through the jet pump diffuser) causes thorough mixing of the liquid boron and the diffuser flow. Simulation test observations confirm this fact. The flow and mixing of liquid boron injected through the JPI lines was tested in a transparent simulation model. Some features of the test model are:

1. 1/6 scale, slab geometry representation of the reactor.

2. The geometry of the reactor intervals is simulated in a simplified way.

3. Simulation includes the portion of the reactor from bottom of the lower plenum to the bottom of the separator skirt.

4. Reactor coolant flow is simulated with flow of water at room temperature and pressure. The model has provision for simulating forced circulation of the coolant by means of pumps and natural circulation is provided by airlift created by bubbling air through the simulated core region.

5. The liquid boron injection into the JPI is simulated by a solution of sodium bromide which simulates nearly all density difference between liquid boron solution and reactor coolant at 550°F.

Flow and mixing patterns were observed in the test model with the help of a color dye injected with the liquid boron flow. The observations indicated that at all core flows greater than 5% of the rated value, the following conclusions are applicable.

1. The liquid boron injected through the JPI at the top of the diffuser thoroughly mixes with the diffuser exit flow.

2. The jet pump exit flow fills up the lower plenum from the bottom.

These results are presently being further confirmed by quantitative measurements from the tests. Based on the above experimental observations from the simulation tests, the following simplified picture of the boron mixing process can be constructed. Consider the condition in which the reactor core coolant flow (by natural circulation) is at a constant rate of $\dot{Q}_c$ lbm/sec and liquid boron flow constant at $\dot{Q}_B$ lbm/sec into the jet pump diffusers. After liquid boron enters the JP diffusers, the water flowing out at the exit of the jet pumps will have a boron concentration of $\dot{Q}_B C_B/\dot{Q}_c+\dot{Q}_B)$ PPM, where C_B is the boron concentration in the sodium pentaborate solution entering the jet pump diffusers. Core coolant with this concentration will appear at the core inlet at a time T_2, where (T_2-T_1) is time required by the jet exit flow to "displace" all the water in the flow path from jet pump suction to the bottom of the active fuel. To facilitate further discussion we define (T_2-T_1) as the "first pass delay" which is also the time to delay from the instant at which liquid boron enters the RPV to the time at which it begins its effect on the core power. As the incoming borated water displaces the coolant in the core, core average concen-tration increases. This increase is modeled linearly from 0 PPM at time $\leq T_2$ to $\dot{Q}_B C_B/(\dot{Q}_C+\dot{Q}_B)$ at time T_3. Where (T_3-T_2), defined as the "core passage time", is the time required for tne core inlet flow to displace all the core coolant between the bottom and the top of the active fuel. The borated core exit flow then flows through the jet pumps, and there picks up a higher concentration equal to $\frac{2Q_B\ C_B}{(\dot{Q}_C + \dot{Q}_B)}$ due to the liquid boron still being pumped into the JP diffusers. This new concentration will appear at the core inlet after the lower plenum water is again displaced by the jet exit flow. Thus at a time T_4, the boron concentration at the core inlet will be $2\dot{Q}_B\ C_B/(\dot{Q}_c+\dot{Q}_B)$, where (T_4-T_2), defined as "loop delay time", is the time taken by a particle of water to traverse the natural circulation flow loop through the core, separators, downcomers, jet pump diffusers and lower plenum. The average concentration in the core will rise to $2\ \dot{Q}_B C_B/(\dot{Q}_c+\dot{Q}_B)$ at T_5, where (T_5-T_4) is again the "core passage time". This set of events from T_2 to T_4 will repeat as long as the core flow and the liquid boron are main-tained at their constant values. In the foregoing discussion, the steam leaving the separators is assumed to be made up by water flow into the reactor (by HPCI or feedwater) keeping water level content, and boron that would be lost with the steam is assumed to be negligible.

The buildup of boron concentration in the core water by the process discussed above is shown in Figure 7.6.2 by the solid curve 1. Here the values of the different transport and delay times are calculated for a typical 251" reactor at 15% core flow and 86 GPM liquid boron flow, and listed in Table 7.6.1. When core flow and liquid boron are assumed to be constant the boron buildup follows a nearly staircase function as indicated by curve 1. However, even at constant average core flow, the flow loop times would be actually different for different water particles because of flowing through differently located fuel bundles, separator, etc. Further, the effect of turbulence in the reactor flow tends to diffuse the sharp boron concentration front that would otherwise be sustained by pure transport. For these reasons the average boron concentration development in the core water is expected to fall within a band shown by the shaded area which envelops the curve 1.

Curve 2 in Figure 7.6.3 shows the buildup of concentration when 100% mixing is assumed (perfect mixing as discussed in 7.6.1). It is seen that this line lies approximately in the middle of the shaded band of actually expected concentration buildup.

In applying the discussion of Figure 7.6.3 to the BWR ATWS analysis, the following must be considered:

Table 7.6.1

TYPICAL BORON MIXING TRANSPORT AND DELAY TIMES FOR JPI INJECTION

251" Reactor Vessel 15% Core Flow 86 GPM Liquid Boron Flow

Transport Delay in Pipeline Outside RPV	44 seconds
Initial Delay Inside RPV	26 seconds
Core Passage Time	12 seconds
Loop Delay TIme	120 seconds*

*Conservatively assuming water level is in the normal range lower level present in the events give shorter delay time.

1. The first pass delay time inside the vessel core passage time and the flow loop delay times are inversely proportional to the core flow rate. Similarly the increase in concentration experienced by the core flow in each of its passage through the jet pumps is also inversely proportional to the core flow rate. Therefore, at core flow rates greater than 15% and curve 1 in Figure 3 will be closer to the 100% efficiency curve 2 (and the shaded band will be narrower). At core flows smaller than 15%, the "steps" of the curve 1 will be larger, but the curve 2 will still pass through the middle of the corresponding enveloping shaded area.

2. In a typical ATWS base case for BWR's with HPCI (i.e., generally BWR/4's), the core flow rate at the time liquid boron first enters the RPV is generally near 20%. Further between this time and hot shutdown, the calculated core flows are high enough to sustain a mixing process similar to the one discussed above.

Based on the above discussion, the following assumptions made in the ATWS analysis for injection through the JPI lines, are considered appropriate.

1. The delay (T_2-T_0), between the start of the SLC pump and the time at which boron first becomes effective in the core is 60 seconds.

2. The mixing efficiency is assumed to be 95%. Here although 100% mixing efficiency is appropriate, a 5% margin is allowed for unknown uncertainties. One such uncertainty could be with respect to circumferencial distribution since fewer than all the jet pumps are used for boron injection.

Under these assumptions, the boron concentration buildup in the core water will also be linear. A comparison of this (Curve 3) with the Curve 1 indicates that the assumptions are justifiable.

7.6.3 Discussion of the Boron Mixing Process for Injection in Core Spray Sparger

Figure 3.4.3-4 and 3.4.3-6 show the schematic of boron injection arrangement for the BWR's which, in ATWS transients, supply makeup water inside the core exist plenum (all BWR/5's and BWR/6's and some BWR/4's belong to this class). In this case the liquid boron is injected into the HPCS line very near the RPV downstream of the check valve nearest to the RPV. This arrangement is adopted because of the desirability to borate the cold HPCS water that comes in at the top of the core.

Test data for the flow pattern of the HPCS flow under ATWS core flow conditions is not available, and the mixing pattern of the HPCS jets with the two phase mixture in core exit plenum is very complex. However, for the purposes of the present discussion certain "bounding" modes of flow and mixing can be postulated to determine bounding values of initial and loop delay times. Using these delays, the development of boron concentra-tion in the core can be constructed in the same manner as was done for Jet Pump Instrument line (JPI) in Section 7.6.2. Possible modes of flow considered are:

MODE 1: The borated HPCS flow mixes with the core exit flow uniformly and after passage through the separators, downcomer, jet pumps and lower plenum becomes effective in the core. Subsequent flow passes through the core bringing increased boron concentration to the core.

MODE 2: In this mode the borated HPCS flow is assumed to partially mix with the steam water mixture in the core exit plenum and flow down into the core bypass region. After filling up the control guide tubes and the core bypass region (providing negative reactivity), it is assumed to mix with core exit flow and then flow upward similar to Mode 1.

MODE 3: In this mode the borated HPCS flow is assumed to partially mix steam water mixture in the core exit plenum and flow down through core active and bypass regions. After filling up the lower plenum and control rod guide tubes, it is assumed to fill up the core bypass and active fuel regions providing negative reactivity. After this, it would mix with the core exit flow and flow upward as in Mode 1.

Table 7.6.2 shows the initial delay and flow loop delay associated with the three different flow modes postulated above. The values shown in the table are for an assumed constant core flow rate of 10% and a HPCS jet entrainment mixing ratio of 1:10. It is seen from the table that the flow Mode 1 causes the largest initial delay and the flow loop delay is the same for all the three modes listed. Thus for boron mixing, postulating the flow Mode 1 would be most limiting. If this flow Mode 1 is assumed, then the expected buildup of average boron concentration in the core water with time is shown in Figure 7.6.3 for 10% core flow. Figure 7.6.3 also shows for comparison the concentration buildup if perfect mixing is assumed.

ATWS transient calculations indicate that a representative value of core flow expected during the period before hot shutdown is approximately 10% of rated. The appropriate bounding value of the initial delay time to be used for liquid boron, therefore appears to be 80 seconds (including the transport delay of ~20 seconds in the pipeline outside the RPV) and the appropriate mixing efficiency appears to be nearly 100%. In the ATWS analysis for BWR/5's and 6's a 60 second initial delay and a mixing efficiency of 75% have been used. The latter is a traditionally used value for ATWS and is obviously conservative. The initial delay time used is slightly less than the value constructed above. However, parametric studies provided in the sensitivity runs (Section 4) indicate that even with initial delays assumed as high as 180 seconds and mixing efficiency of 75%, the calculated ATWS consequences for pool temperature are acceptable.

7.6.4 Liquid Boron Effectiveness

In the dynamic analysis of ATWS, the negative reactivity effect of liquid boron is assumed to be proportional to the amount of boron present in the core between the bottom and the top of the active fuel. The negative reactivity due to liquid boron at any time, is calculated from the equation:

Table 7.6.2

BORON TRANSPORT AND DELAY TIMES FOR INJECTION IN THE HPCS SPARGER

	Flow Mode Assumed		
	Mode 1	Mode 2	Mode 3
Delay Outside RPV*	20	20	20
Initial Delay**	60	25	50
Flow Loop Delay***	170	170	170

*HPCS assumed to be in operation (3500 gpm).

**Delay calulated using decreased water level present durning initial borom injection potion of events.

***Conservatively assuming water level is in the normal range, lower levels experienced thoughout much of the event would give shorter loop delay times.

$$R(t) = RHS \quad \frac{W(t)}{W_{HS}} \qquad\qquad W_{HS}(t)$$

where:

$R(t)$ = Liquid boron reactivity at time t ($)

R_{HS} = Liquid boron reactivity at hot shutdown condition ($)

$W(t)$ = The weight of boron present in the core (lbm)

W_{HS} = Amount of boron in the core necessary to maintain hot shutdown condition

W_{HS} is obtained from a steady state, three dimensional core reactivity calculation asuming the following conditions:

a. No voids
b. Core coolant and 280°C
c. Liquid boron uniformly distributed in the core
d. Critical rod patern

R_{HS} is assumed to compensate for the combined void and doppler reactivity difference between the operating and no void-saturated hot shutdown conditions.

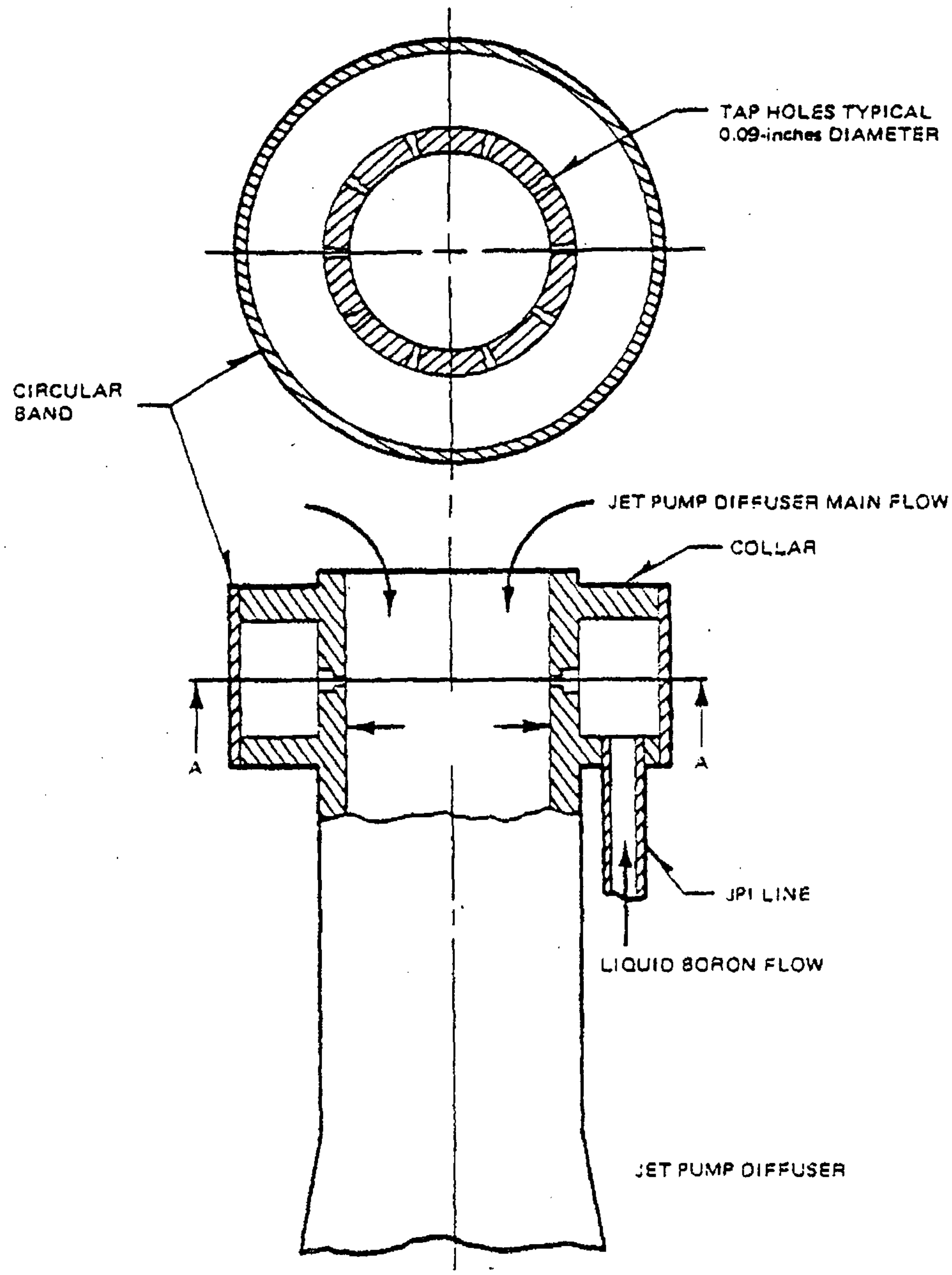

Figure 7.6.1. Schematic of Typical Connection of JPI Line With the Jet Pump Diffuser

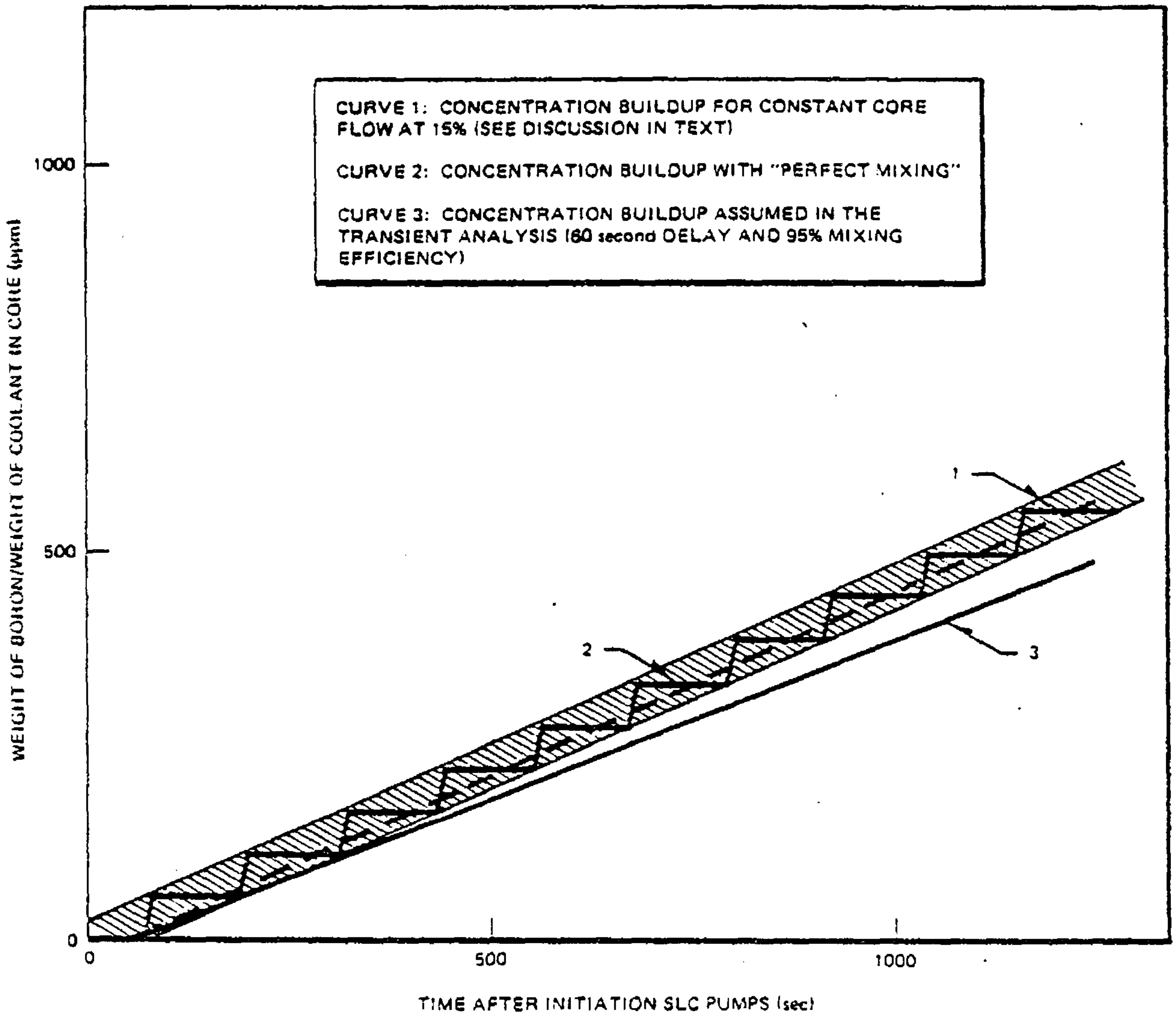

Figure 7.6.2. Boron Concentration Buildup in Core Water for JPI Injection Liquid Boron Flow Equals 86 gpm

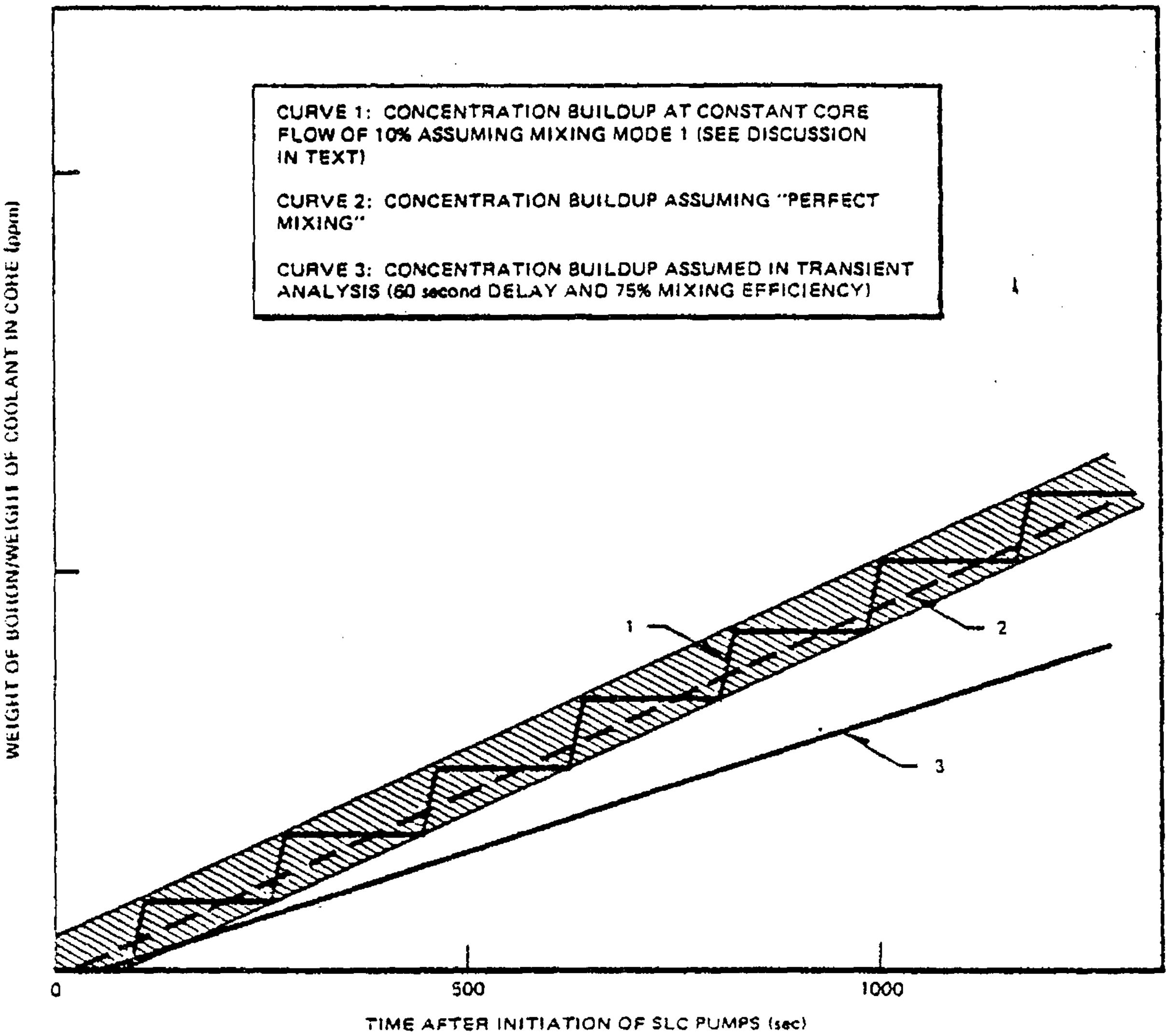

Figure 7.6.3. Boron Concentration Buildup in Core Water for HPCS Injection Liquid Boron Flow Equals 86 gpm

APPENDIX 7.7
CONCEPTUAL DEFINITION FOR ATWS MITIGATION SYSTEMS

Figure 7.7-1. Simplified ATWS Mitigation Logic for BWR/4

Figure 7.7-2. Simplified ATWS Mitigation Logic for BWR/5

Figure 7.7-3. Simplified ATWS Mitigation Logic for BWR/6

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