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BWR/6 MARK III HYDROGEN CONTROL OWNERS' GROUP

REPORT ON HYDROGEN CONTROL ACCIDENT SCENARIOS, HYDROGEN GENERATION RATES AND EQUIPMENT REQUIREMENTS

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ILLUSTRATIONS

LIST OF SPONSORS

This study was undertaken by the General Electric Company in support of the following electric utilities which are part of the Hydrogen Control Owners Group:

Cleveland Electric Illuminating

Mississippi Power & Light

Illinois Power Company

Tennessee Valley Authority

Gulf States Utilities

Puget Sound Power & Light/Northwest Energy
Services Company

EXECUTIVE SUMMARY

This report was prepared in support of the BWR/6 Mark III Hydrogen Control Owners' Group (HCOG). The objective of this study was to evaluate key parameters and define key criteria that can be utilized to specify a system to prevent loss of containment integrity due to uncontrolled hydrogen combustion. This objective was met by defining a postulated hydrogen generation event (HGE) and performing a detailed analysis to calculate the hydrogen generation rate for this event. The scope of this study was limited to the standard 238 BWR/6 in a Mark III containment. It is General Electric Company's position that no additional hydrogen control system is justified, and that plant risk is not significantly reduced by inclusion of any additional hydrogen control system.

The extent of hydrogen generation in the Three Mile Island (TMI-2) accident prompted renewed interest by the NRC in hydrogen control systems for nuclear power generating plants. Accordingly, the NRC has proposed rules for hydrogen control in BWR/6 Mark III plants (Federal Register, Vol. 46, No. 246, page 62281, December 23, 1981). The hydrogen control requirements set forth by the NRC for Mark III BWR's were derived on a deterministic basis from the experiences of the accident at TMI-2. Because of the fundamental differences between a BWR and TMI-2, safety related rules derived in such a manner do not adequately reflect the safety features of the BWR which inherently prevent substantial hydrogen generation.

The HCOG commissioned GE to perform a study which will provide the hydrogen generation rates for a postulated Hydrogen Generation Event (HGE) as a basis for the specification of a hydrogen control system. The major phases of the study included defining the mechanistic details of the postulated HGE, performing a detailed analysis of core heatup and metal-water reaction (MWR) during the HGE, extrapolating results of the detailed analysis of the HGE to a number of related events, and identifying the vital equipment which must function during and after the operation of the hydrogen control system.

The choice of a HGE can be based on criteria that are deterministic or probabilistic or on some combination of deterministic and probabilistic criteria. In order to provide a rational basis for the design of a hydrogen control system, a HGE was postulated which would result in significant hydrogen generation without substantial core-melt. A probabilistic approach was used to identify the most representative accident initiating event for the HGE, and deterministic considerations were applied in defining a realistic termination for the HGE before occurrence of significant core-melt. In the probabilistic assessment used in this study, any given mechanical system is modeled as either completely available or completely failed with no subsequent recovery (Binary Approach).

Based on probabilistic considerations, the initiating event of the postulated HGE was defined as turbine trip with bypass and temporary failure of all water makeup systems. Because the postulated HGE would be a slowly progressing accident, the operator would be able to concentrate his attention on recovering the water makeup systems. Based on deterministic considerations, it was assumed that the operator would succeed in restoring a water makeup system within a reasonable period of time (within approximately 45 minutes of the time of the accident initiation). A spectrum of possible accident scenarios and the process of defining the postulated HGE are described in Chapter 2.

Early in this study it was attempted to postulate a mechanistically justifiable scenario which would result in a large amount of hydrogen generation from MWR without significant core-melt, and subject to the constraints of binary system failure as used in the probabilistic approach. No such scenario could be found. The scenario with the smallest margin of core cooling adequate to prevent core-melt resulted in no significant hydrogen from MWR.

For the detailed evaluation of the hydrogen generation rate, the HGE was modeled as a reactor isolation event with temporary failure of all water makeup systems. The high pressure core spray (HPCS) was assumed to be restored 15 minutes after verification of failure of all high pressure and low pressure water makeup systems. The resultant total hydrogen generation was determined to be limited to the equivalent of the reaction of 12.5% of the cladding surrounding the active fuel. The analysis was performed using a methodology based on extrapolation of current GE analytical models used for licensing analysis of loss of coolant accidents (LOCA).

The final phase of this study involved identifying the vital equipment which must survive during and after operation of the hydrogen control system. As fission product release to the surrounding environment is the major determinant of plant risk, the only vital equipment that is required to function in conjunction with a hydrogen control system is that equipment which maintains the containment function. For the sake of completeness, the vital equipment needed to maintain containment integrity (by providing core cooling and decay heat removal) following the postulated HGE was identified in this study.

In summary, this study relies on the methodology developed for performing Probabilistic Risk Assessments (PRAs) to conclude that:

a. Constraining the hydrogen generation event (as a basis for design of a hydrogen control system), to be mechanistically based and to result in no core-melt, leads to the choice of an event which also results in no hydrogen generation. An alternate event, which approximates the major risk producing events (that also have hydrogen generation) results in significant hydrogen generation combined with core-melt. For this event, the hydrogen generation is terminated by the delayed availability of makeup water, as a result of operator action to start water injection. The maximum zirconium reacted in this case is limited to the equivalent of 12.5% of the zirconium cladding surrounding the active fuel.

b. For the events with significant hydrogen generation, the only equipment that is necessary to remain operational is that which retains the containment function, i.e., any equipment that mitigates fission product release to the atmosphere from containment by assuring fission product scrubbing and retention in the suppression pool.

Furthermore, this study concludes that (based on References 10 and 11):

c. Hydrogen control systems only reduce the probability of loss of containment integrity due to the consequences of uncontrolled hydrogen combustion.

d. Elimination of loss of containment integrity due to uncontrolled hydrogen combustion reduces the overall plant risk insignificantly, because the BWR suppression pool provides an effective filter for fission products.

1. INTRODUCTION AND BACKGROUND

Following the accident at Three Mile Island (TMI), the Nuclear Regulatory Commission (NRC) and the nuclear industry have reviewed the existing safety features of nuclear power plants. Partially as a result of this review, the NRC has proposed that Boiling Water Reactors (BWR's) should have additional means for control of large amounts of hydrogen following postulated accident. All BWR's have been designed to safely accommodate smaller amounts of hydrogen generation which would result from the postulated Design Basis Loss of Coolant Accident (DB-LOCA) in accordance with Regulatory Guide 1.7.

This report was prepared in support of the BWR/6 Mark III Hydrogen Control Owners Group (HCOG). The HCOG was formed by mark III BWR owner utilities to deal generically with the NRC position relative to the control of hydrogen in the Mark III containment. That NRC position has evolved through a number of different stages. The most recent NRC position regarding the generation of hydrogen in Mark III BWR plants with existing construction permits is contained in the proposed rule on "Interim Requirements Related to Hydrogen Control." Some of the points of that proposed NRC rule are presented here, according to the Federal Register, Volume 46, Number 246, Page 62281, of December 23, 1981.

"It is proposed that boiling water reactor (BWR) facilities with Mark III type containments...., for which construction permits were issued prior to March 28, 1979, be required to install hydrogen control systems capable of accommodating an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding (surrounding the active fuel region) with water, without the loss of containment integrity".

"The 75% is judged to be representative of the maximum amount of hydrogen likely to be generated in an accident in which the threat to containment is limited to the threat posed by the combustion of hydrogen. Events with metal-water reaction in excess of 75% are judged to be associated with core-melt accidents which could pose a threat to containment greater than the combustion of hydrogen. This 75% value also appears to be reasonable because it is sufficiently greater than the fuel cladding-water reaction analyzed to have occurred at TMI-2 to provide a conservative estimate for the cladding reaction that may occur during a TMI type degraded core cooling accident."

As defined in NUREG-0718 (March 1981), the Near Term Construction Permit applicants are to "provide a system for hydrogen control capable of handling hydrogen generated by the equivalent of a 100% fuel-clad metal-water reaction."

These points of the proposed rule were derived on a deterministic basis from the experiences of the accident at TMI-2. Because of the fundamental differences between a BWR and TMI-2, safety related rules derived in such a manner do not adequately reflect the inherent and designed safety features of the BWR which tend to prevent substantial hydrogen generation.

If the owners of Mark III BWR plants are required to install additional hydrogen control systems in those plants inspite of the evidence that substantial hydrogen generation is effectively prevented in a BWR, then some rational approach had to be developed to serve as a basis for the design of those systems. The most objective method to provide such a basis is a probabilistic risk assessment (PRA). This approach effectively accounts for the basic design features of the BWR and also helps to identify the effectiveness of any system in reducing the overall plant risk to the public. The PRA identifies the risk dominant events and the relative advantage of preventing each of these initiating events from occurring. It can be used to quantify the advantage of mitigating the consequences of any of the major risk-producing events.

Based on the low probability of core-melt for the consolidated events, the risk reduction associated with the inclusion of any additional hydrogen control system in a BWR/6 Mark III appears to be insignificant. Recognizing this point, events and scenarios were nevertheless defined for purposes of designing a hydrogen control system. The choice of these events can be based on deterministic, probabilistic, or combined deterministic and probabilistic considerations. Chapter 2 discusses these possibilities and provides a rationale for defining a hydrogen generation event (HGE). Probabilistic considerations were used to select the most representative initiating event, and deterministic considerations were used to define a reasonable termination of the event. The HGE was defined as turbine trip with bypass followed by temporary failure of all water makeup systems.

Chapter 3 of this report provides a description of the detailed analysis of the HGE, including a discussion of the analytical tools used, the event progression, major analytical inputs, and the results of the analysis. The postulated HGE would be expected to result in hydrogen generation equivalent to the reaction of 12.5% of the cladding surrounding the active fuel.

Chapter 4 considers equipment that might be essential or useful under various situations involving hydrogen generation. As the release of fission products to the surrounding environment is the major determinant of plant risk, the only vital equipment that is required to function in conjunction with a hydrogen control system is that equipment which maintains the containment function by assuring fission product scrubbing and retention in the suppression pool. The BWR suppression pool provides an effective filter for the fission products and provides a means of maintaining the containment function even if containment integrity should be lost. Thus, the only equipment which is vital is that which is needed to keep the fission product pathway through the suppression pool intact. However, the equipment that is necessary to maintain containment integrity (by providing core cooling and decay heat removal), has been identified for the sake of completenes. All analyses and calculations presented in this report are based on a standard 238 BWR/6 in a Mark III containment.

2. ACCIDENT SCENARIOS

An important input in the design of a hydrogen control system is the definition of accident scenarios the system must handle. Section 2.1 discusses the criteria that can be used to define a hydrogen generation event (HGE) as a basis for design of a hydrogen control system. Section 2.1.2 discusses the possible initiating events and their relationship to each other. Section 2.2 provides a discussion on the choice of a hydrogen generation event and on how this event is representative of most BWR events. This section demonstrates that for any event without core-melt no significant hydrogen generation is expected.

2.1 CRITERIA FOR CHOOSING A HGE

The choice of a HGE can be based on criteria that are deterministic or probabilistic or on some combination of deterministic and probabilistic criteria. Deterministic criteria have been used in defining the evaluation bases in numerous safety analyses, but in the absence of other considerations they might lead to the choice of a hydrogen control system basis that is neither the most representative nor the one which contributes the most to overall plant risk. Probabilistic analyses are useful for a comparison of alternatives and have the potential for quantifying the relative frequencies of various initiating events, and the relative contribution of different accident scenarios to total plant risk. In order to provide a rational basis for the design of a hydrogen control system, a hydrogen generation event (HGE) was postulated which would result in significant hydrogen generation and only minor core-melt. A probabilistic approach was used to identify the most representative accident initiating event for the HGE, and deterministic considerations were applied in defining a realistic termination for the HGE before occurrence of significant core-melt*. Each of these three approaches: i.e., deterministic, probabilistic, and a combination of deterministic and probabilistic, is discussed in detail in the following sections.

*Core-Melt: In this study it has been assumed that localized incremental melting and slumping of the core occurs as the local cladding temperature reaches the melting point of Zircaloy (3366°F), and the extent of core-melt at a given time is quantified as the fraction of the core for which the local cladding temperature has exceeded the melting point of Zircaloy at any previous time in the accident.

2.1.1 Deterministic Criteria

Deterministic criteria have been used in defining the evaluation bases in numerous safety analyses, e.g., the basis for Loss of Coolant Accident (LOCA) analysis is an instantaneous double ended rupture of the largest pipe in the reactor coolant system combined with a single failure. Another area where deterministic criteria have been used in safety analyses is related to operational transients. The Nuclear Safety Operational Analysis (NSOA) in Final Safety Analysis Reports (FSARs) have been used to demonstrate that single failures following an operational transient will not result in exceeding thermal margins. However, the use of deterministic criteria has some inherent shortcomings. The criteria could lead to the choice of an evaluation basis that is neither the most representative nor the one which contributes the most to the overall plant risk.

Recognizing the shortcomings of a deterministic approach, a deterministic criterion could still be used for hydrogen control. The NRC has proposed criteria in draft rules on hydrogen control primarily to prevent the loss of containment integrity as a result of hydrogen burning or detonation. The designer is required to assume that 75 - 100% of the zirconium cladding surrounding the fuel has oxidized within a certain time period, resulting in hydrogen generation. The percentage of zirconium required to be reacted presumably represents an upper bound estimate of the cladding reacted at Three Mile Island 2. As discussed in the following section (Section 2.1.1.1), this approach tends to ignore the inherent features of the BWR design that prevent substantial hydrogen generation.

Additional deterministic approaches to defining a HGE could include the requirement to consider a certain fraction of the core as uncovered or without significant cooling for a fixed period of time. Alternatively, one could extend the current safety analysis requirements to include an additional failure for operating transients or for small breaks, and then determine the hydrogen control requirements using realistic analytical methods for these multiple failure situations. This latter approach does consider the unique BWR design features of being able to sustain multiple failures, but does not necessarily address the issue of whether all major risk dominating situations have been included.

2.1.1.1 BWR Unique Features

The high level of safety of the Boiling Water Reactor is provided by systems which prevent an accident from producing adverse consequences and by additional systems which effectively mitigate adverse consequences, should an accident occur. A number of the design features unique to the BWR which assure accident prevention and no significant hydrogen generation are discussed in the following paragraphs.

a. The BWR provides the plant operator with a direct indication of reactor vessel water level. Reactor water level is the only parameter needed by the operator to assess the adequacy of core cooling following a transient or accident. Figure 2.1.1-1 provides a schematic representation typical of BWR/6 plants of the range of water level measurement coverage, including the number of level sensors, their relative locations, and the number of indicators and recorders available to the operator.

b. The BWR is designed with highly redundant water delivery systems to ensure core coolability and to prevent core damage and hydrogen generation. The typical BWR cooling network includes six high pressure pumps, seven low pressure pumps, and the Automatic Depressurization System. It is important to note that in most instances, any one of eleven pumps is sufficient to maintain core coverage or prevent potential core damage due to a transient or a small break accident*. Figure 2.1.1-2 schematically illustrates the feedwater, high pressure core spray, reactor core isolation cooling, and control rod drive systems, and provides typical

*For a transient or accident followed operation at less than 65% power plants may also be adequately cooled with one control rod drive (CRD) pump, if the reactor is not depressurized.

pumping capacities. Figure 2.1.1-3 illustrates the low pressure core spray, low pressure coolant injection and condensate booster pump systems and their water delivery capabilities. The high pressure systems which are electrically driven (HPCS and CRD) are also available after the reactor is depressurized, and provide increased flows at lower pressure. The cooling systems provide diverse types of cooling capability through both core flooding and core spray, and diverse injection locations (feedwater spargers, core spray spargers, core bypass injection and control rod drives). The injection systems also have diverse water sources (e.g., the condensate storage tank, the suppression pool, and the feedwater train).

c. The injection systems also have diverse power sources - steam driven and electrically driven. The electrically driven water delivery systems can draw power either from the offsite power grid or from any of three onsite diesel-generator sets. Common mode failure of the three diesel-generator sets is not likely because of the extensive differences between the systems, as outlined in Table 2.1.1-1 for a typical BWR/6.

d. The BWR provides multiple means to rapidly and easily depressurize the primary system, permitting the operation of the condensate, low pressure coolant injection, and low pressure core spray systems. Figure 2.1.1-4 schematically illustrates the depressurization system and provides typical time durations to achieve low pressure, depending on the depressurization scheme employed. Also shown schematically in Figure 2.1.1-4 are the reactor pressure vessel and safety-relief valve vent lines which provide the high point venting of the BWR.

e. The suppression pool provides a large passive heat sink with the capability to handle in excess of 1260×10^6 BTU heat load before the containment pressure is raised to the design limit by steam and air pressure. The suppression pool capability effectively decouples the reactor from the balance of plant for short-term decay heat removal, permitting operations to be fully devoted to reactor inventory maintenance.

f. The BWR is inherently capable of operating under conditions of natural circulation. Figure 2.1.1-5 schematically depicts the normal natural circulation flow path from the downcomer through the jet pumps and into the core shroud region. A second natural circulation loop is shown inside the shroud. This loop will exist if the primary natural circulation flow is reduced to a low value following an abnormal transient or a postulated pipe break. Water will flow downward through the core bypass region and into the bottom of the active fuel region through the normal bypass leakage paths. This inherent BWR design feature prevents significant hydrogen generation by providing cooling whenever water inventory is available.

g. In specifying the amount of metal-water reaction or hydrogen generation, the NRC has not considered important differences between BWR and TMI-2 event sequences and core heatup. In calculations of metal-water reaction, an unlimited amount of steam in the core is generally assumed in order to maximize zirconium-steam reaction rate, but the evaluation neglects the associated steam cooling, a very effective mode of heat transfer, in making this assessment.

In summary, a deterministic approach to define a HGE would need to take into account the inherent characteristics of the BWR to effectively prevent core-melt and hydrogen generation.

2.1.2 Probabilistic Considerations

Probabilistic analyses could be performed to help make judgments on design modifications and to provide information to answer specific design and licensing questions. Probabilistic analyses provide a method of estimating the likelihood of multiple failures which could interfere with the ability of the plant systems to perform the needed mitigating action for an unplanned event. However, there have been no formally established probabilistic acceptance criteria. These analyses are useful for a comparison of alternatives and have the potential for identifying areas in system design and/or operation that could improve overall reliability. Probabilistic risk analyses also provide quantitative risk reductions for various design changes enabling the designer to determine the comparative benefit of solving one problem in preference to another, e.g., hydrogen control versus improved water delivery or prevention systems.

The following sections discuss a simplified methodology that was used to define a probabilistic hydrogen generation event for a hydrogen control system. The steps are similar to that for a PRA and consist of the following:

1. Identification of BWR initiating events;
2. Description of each event including impact of multiple failures;
3. Determination of frequency of occurrence of each initiating event;
4. Determination of core damage probability for the events.

Having performed the above four evaluations, a hydrogen generation event based on highest frequency of the initiating event or on the highest core damage probability is chosen.

In the probabilistic assessment used in this study, most mechanical systems were modeled as either completely available or completely failed with no subsequent recovery (binary approach). In reality, many system failures are only partial and/or temporary (with subsequent recovery). For virtually all accidents the flow capacity of any one of the emergency core cooling systems is sufficient to keep the core cooled. Partial success of one or more systems will significantly lessen the severity of the accident. Thus, basing the probabilistic assessment on a binary approach to model system failures adds conservatism to the probabilistic assessment results.

2.1.2.1 Description of BWR Initiating Events

BWR initiating events can be classified into four general categories as shown in Table 2.1.2-1. They are operational transients, anticipated transients without scram (ATWS), loss of coolant accidents (LOCAs), and manual shutdown requiring decay heat removal. The majority of events that a plant is likely to experience are operational transients. In the short term, the thermal/hydraulic and nuclear response of the BWR to these operational transients is event dependent. The normal control systems are designed to automatically handle the two basic concerns in the short term, i.e., overpressure and over-power protection. In the long term, the system behavior is independent of the initiating event and consists of water inventory control and decay heat removal.

Comparison of the event progression of operational transients (with multiple failures) to small breaks, shows that the long term inventory control is the same in both situations. The primary means of inventory control is the normal high pressure makeup systems (feedwater). In the event these are unavailable, the high pressure safety systems - HPCS and RCIC - provide inventory makeup. Failure of the three high pressure systems requires primary system depressurization to permit the numerous low pressure systems to provide inventory makeup.

Manual shutdown of the reactor followed by loss of water makeup has an insignificant contribution to the probability of core damage. The inability or failure to start decay heat removal in a timely manner can result in containment heat-up but does not involve immediate concerns related to the control of reactor water inventory. Assuming that the shutdown is planned, it is unlikely that loss of core cooling in the long term would occur. Therefore, the following discussions will not include this initiating event in any further consideration.

2.1.2.2 Initiating Event Frequency

Based on a review of BWR performance and incorporation of post-TMI improvements in the BWR/6 design, the expected frequency of occurrence of most initiating events can be defined. Table 2.1.2-2 provides this expected frequency for BWR/6 initiating events (Reference 2).

2.1.2.3 The Probability of Core-Melt Damage for Consolidated Events

As discussed in Section 2.1.2.1, operational transients with multiple failures have essentially the same long term event progression, even though in the short term there are some differences. This basic similarity for long term transient behavior (Reference 1) is summarized in

Table 2.1.2-1, and is the basis for consolidation for all operational transients and LOCAs into five categories. For example, all operational transients have a strong nuclear thermal hydraulic coupling for a few seconds but ultimately will result in reactor isolation (if no water makeup is available), water boiloff and inventory loss through the S/RV's at essentially the same rate. The five categories are:

a. Loss of feedwater/MSIV closure;
b. Inadvertent open relief valve;
c. Loss of offsite power;
d. Small breaks;
e. Large breaks.

Each one of these consolidated events contributes in differing amounts to the overall plant risk. However, one measure of overall plant risk is the probability of core-melt for each of these consolidated events. This criterion can be used to roughly estimate the relative contribution of each event to overall plant risk.

Determination of the core-melt probability for each of these consolidated events involves the following steps:

1. Definition of success, i.e., what combination of systems will result in no loss of containment integrity or no core-melt;

2. Definition of event trees, i.e., a set of logic diagrams describing potential equipment failure modes and other events which could disable a system or group of systems, and ultimately lead to inadequate core cooling or loss of containment integrity.

Based on an evaluation of each of the consolidated events, the core-melt probability was estimated based on a binary probabilistic approach (Reference 2) and is shown in Table 2.1.2-3. The results show that loss of feedwater and loss of offsite power are the dominant events.

2.1.3 Deterministic/Probabilistic Considerations

One of the NRC's proposed bounding conditions is that the hydrogen control system should be able to deal with 75 to 100% MWR but that no significant core-melt should occur. The probabilistic approach discussed above results in the choice of events that are more likely to lead to substantial core-melt. Thus, from the binary probabilistic approach alone, it is not possible to define a HGE that does not result in substantial core-melt. On the other hand, the deterministic approach, discussed in 2.1.1, results in events that do not necessarily address the significant contributors to overall plant risk. In order to define a HGE which is mechanistically justifiable and which results in significant hydrogen generation with minimal core-melt, it is necessary to use a combination of both the probabilistic and deterministic approaches. The former can be used to narrow down the large number of events to be considered and then one can use a deterministic approach which includes one or some combination of the following:

a. Definition of the event as the one where the lowest capacity water makeup system, that can prevent core-melt, is operating.

b. Definition of the event as one where no makeup system is operating, because of failure of the primary inventory determining instruments (i.e., water level indicators), and alternate indications such as radiation monitors alert the operator to initiate injection.

c. Establishment of the initiating event with temporary failure of all water makeup systems based on frequency of occurrence and then using some reasonable criterion for recovery of a system to define the end of the event.

2.2 HYDROGEN GENERATION EVENT

As discussed in Section 2.1.3, the most reasonable method for choosing a hydrogen generation event is a combination of probabilistic and deterministic approaches.

Based on the results in Table 2.1.2-2, the most frequently occurring accident initiating events are the loss-of-feedwater type events. Furthermore, Table 2.1.2-3 shows that the major contribution to plant risk, as measured by core-melt probability, comes from two types of event: loss-of-feedwater (LOF) and loss-of-offsite power (LOOP). Both of these events have the same order of magnitude probability of core-melt. In addition, the expected event progressions and hydrogen generation rates for accidents based on these two initiating events are similar. However, any event involving operator error (e.g., misalignment of a valve in a water makeup system) is more likely to have an event progression which is more similar to the LOF event. Finally, for the purpose of designing a hydrogen control system to mitigate the consequences of hydrogen generating accidents which do not result in substantial core-melt, the LOF event is a more appropriate basis than the LOOP event, since it affords the operator more options for recovery. Based on these considerations - expected frequency of occurrence, probability of core damage, and similarity to a broad class of events - the loss of feedwater event is the most logical choice as the accident initiating event upon which the HGE is based. It is modeled as a turbine trip with bypass valves operating.

Having chosen the initiating event, it is now necessary to review the event progression, assuming multiple failures, to ascertain if it is possible to generate a significant amount of hydrogen without significant core-melt. If the lowest capacity water make-up system - a single CRD pump - is the only system available, the event progression results both in rapid hydrogen generation and rapid core-melt. Similarly, if both of the CRD pumps are available with no other systems, then hydrogen generation with substantial core-melt will result. The next logical step is to assume the simultaneous operation of the next higher capacity water makeup system - the RCIC for one on and off cycle - combined with both of the CRD pumps. The event progression in this case results in no core-melt, assumes only the automatic operation of the lowest capacity water makeup systems, and assumes only that the operator starts the second CRD pump. This event can be used to bound the event progression for all the events that do not lead to core-melt and, hence, represents a possible choice for HGE. But a choice of this event as the HGE results in no hydrogen generation. In conclusion, using a binary probabilistic approach, it is not possible to define an event that results in significant hydrogen generation without any core-melt.

However, it is reasonable to assume that the operator would be able to restore at least one of a large number of water makeup systems within a reasonable time (within about 45 minutes after the accident initiation). Restoration of a water makeup system with subsequent reflooding and recovery of the core would prevent complete core meltdown. Hence, an alternate event involving water makeup system restoration and core recovery has been defined as the HGE and is described in detail below.

Initiating event: Turbine trip with bypass loss of feedwater type event with temporary failure of all water makeup systems.

Event Progression: The reactor would be scrammed and it is assumed that normal feedwater is lost simultaneously. Turbine bypass valves open to control pressure. The assumed failure of the other high pressure water makeup systems - RCIC, CRD and HPCS - will result in an isolation of the pressure vessel from a low-low water level signal. The reactor water level would continue to drop because of loss of inventory through the cycling relief valves with no compensating water makeup. As the water level drops below the top of the active fuel the operator is expected to depressurize the reactor following Contingency 3 of the Emergency Procedure Guidelines (Reference 9). This is based on the assumption that the operator has determined that no low pressure water makeup systems are available. If low pressure systems had been determined to be available, the operator would have rapidly depressurized the reactor earlier and flooded the reactor pressure vessel with no consequent core heatup and hydrogen generation.

Depressurization of the reactor by the operator, according to the guidelines, is deemed highly likely as there are more water makeup systems available to the operator at a lower pressure (below about 200 psia).

Event Recovery: After the reactor is depressurized below the shutoff heads of the numerous low pressure systems, the operator is available to work on correcting the problems causing the unavailability of any injection system, i.e., failure of injection valves, pumps, initiating signals and/or failure of power sources to open the valves or run the pumps. As there are more water sources available at this lower pressure, it is reasonable to assume that the operator will be able to start injection with one of these sources within 10-15 minutes after the pressure drops below the shutoff head of the low pressure systems. But as the operator has had a longer time to restore the high pressure system, the probability of getting the HPCS working again is higher. Hence, the recovery phase of the HGE is defined as the operation of HPCS within a reasonable time after event initiation.

In summary, this section provides the definition of a hydrogen generation event that can be used for designing a hydrogen control system. The initiating event, the event progression, and the recovery phase are defined. The following section (Section 3) discusses the calculation of the hydrogen generation rates for the hydrogen generation event (HGE).

Table 2.1.1-1

DIFFERENCES BETWEEN THE STANDBY POWER SUPPLY SYSTEMS

(DIESEL-GENERATOR SETS)

ITEM	DESCRIPTION	DIVISION I & II	DIVISION III
1	Engine-Manufacturer	DeLaval	GM-EMD
2	Generator-Manufacturing	Elect. Prod.	Ideal Electric Co.
3	Nominal Load Rating	7000 KW	2600 KW
4	Speed (RPM)	450	90C
5	Fuel	Each Division is independent	Dedicated to Division
6	Starting Air System	2 complete and redundant Air Start Systems (each air start has capacity for 5 consecutive starts)	2 complete and redundant Air Start Systems (each air start has capacity for 5 consecutive starts)

From TVA STRIDE Design

Table 2.1.2-1

ACCIDENT INITIATING EVENTS

Event	Characterization
o Loss of Coolant Accidents	o No Significant Nuclear Coupling
- Large Break	- Pressure Rate to -50 psi/sec - Rapid Water Level Decrease - Possible Complete Core Uncovery - Refill/Reflood
- Small Break	- High Pressure - Water Level Maintenance
o Operational Transients	o Strong Nuclear/Thermal Hydraulic Coupling for a Few Seconds
- Pressurizations (e.g., turbine trip)	- Pressure Rate to 15) psi/sec for ~1 second - Water Level Maintenance
- Depressurizations (e.g., pressure regulator failure)	- Depressurization Rate ~15-20 psi/sec - Water Level Maintenance
- Level Transient (e.g., loss of feedwater)	- Water Level Maintenance
- Power Rises (e.g., loss of feedwater heater)	- Caused by Negative Reactivity Coefficients - Water Level Maintenance
o Anticipated Transients Without Scram	o Strong Nuclear/Thermal Hydraulic Coupling until Boron Shutdown ~20 minutes
o Manual Shutdown	o Requires Manual Decay Heat Removal Initiation

Table 2.1.2-2

EXPECTED FREQUENCY OF ACCIDENT INITIATING EVENTS

Event	Expected Annual Frequency
1. MSIV/Loss of Feedwater Closure	4.4
2. Loss of Offsite Power (>15 minutes)	0.05
3. Inadvertent Open Relief Valve	0.03
4. Loss of Coolant Accidents	
(a) Small Break + Intermediate Break	4×10^{-4}
(b) Large Break	1×10^{-4}
5. Anticipated Transient without Scram	5×10^{-6}

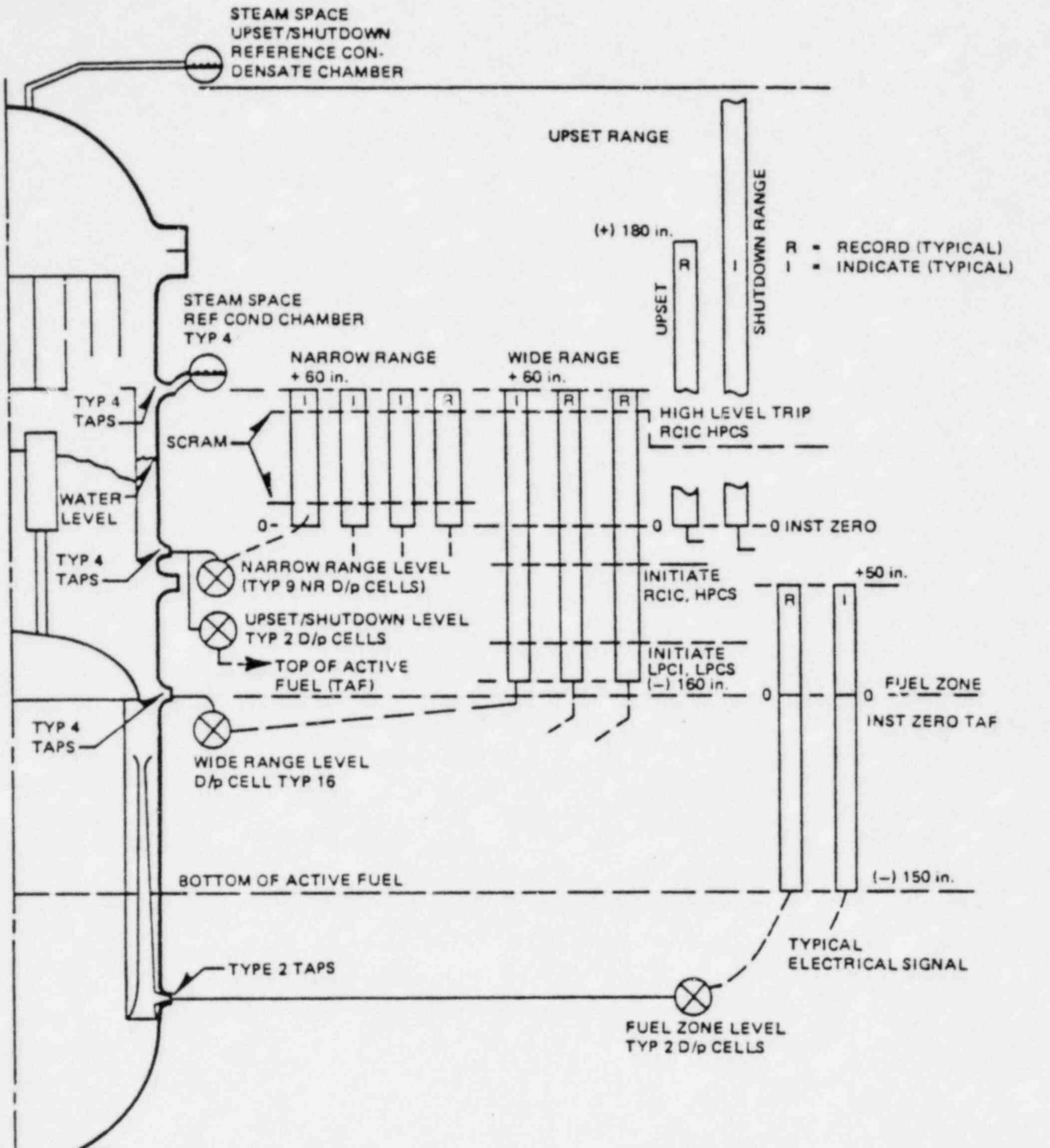
Table 2.1.2-3

ESTIMATE OF PROBABILITY OF CORE-MELT FOR CONSOLIDATED EVENTS

Consolidated Event	Annual Probability
Loss of Feedwater	2×10^{-7}
Loss of Offsite Power	8×10^{-7}
Inadvertent Open Relief Valve	1×10^{-8}
Anticipated Transient Without Scram	$<1 \times 10^{-7}$
Small Break and Intermediate Break LOCA's	$<5 \times 10^{-8}$
Large Break LOCA	$<1 \times 10^{-8}$

Figure 2.1.1-1

Typical Reactor Level Indicators on Reactor Control Panels - BWR/6



NOTES: 1. SEPARATE D/p CELLS ARE USED FOR NARROW RANGE INDICATION AND TRIP UNITS
2. INDICATION/RECORD AND TRIP UNITS FOR WIDE RANGE USE COMMON D/p CELLS
3. WATER LEVEL FOR INITIATION OF RCIC AND HPCS IS ALSO PERMISSIVE FOR ADS INITIATION

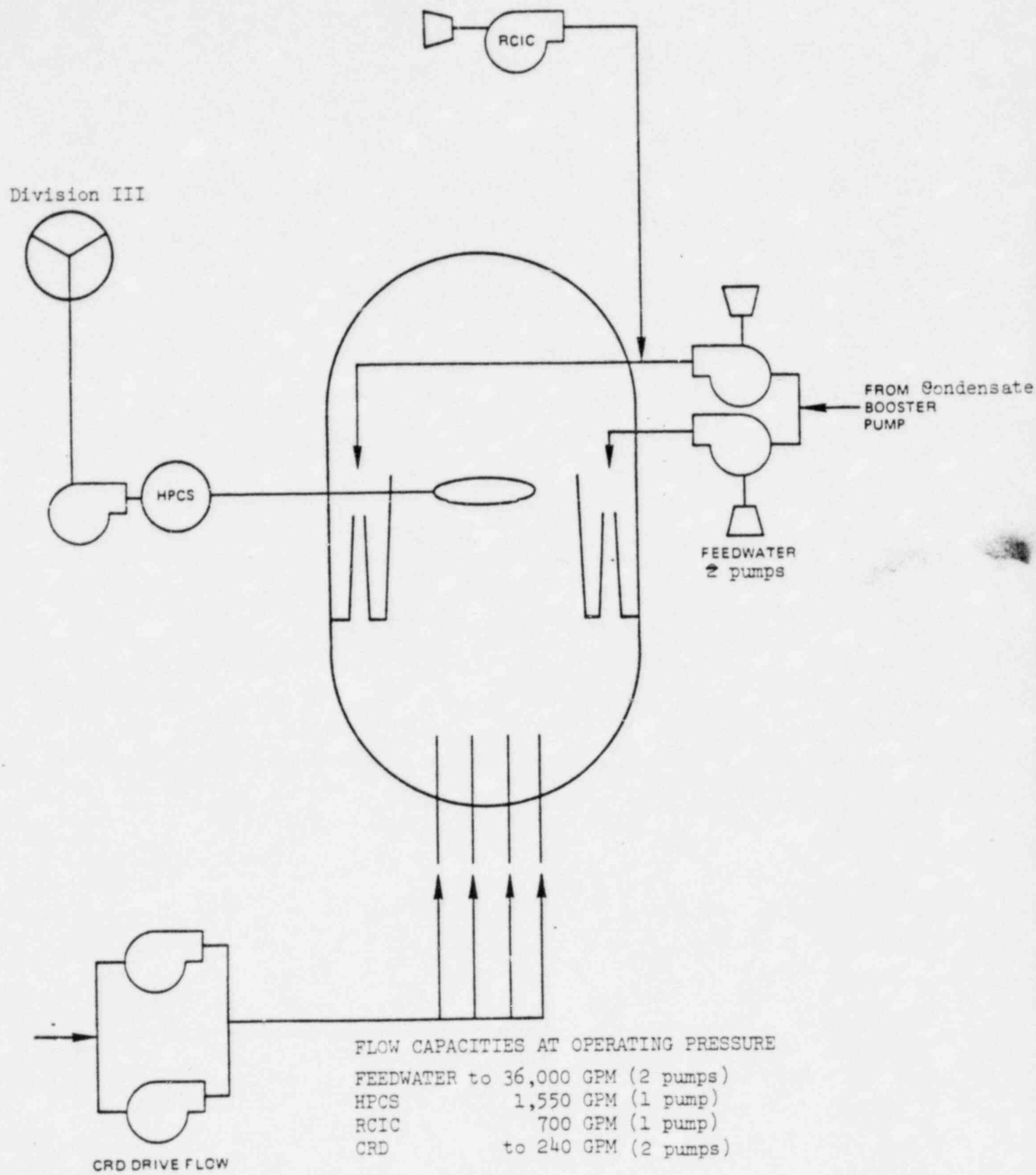
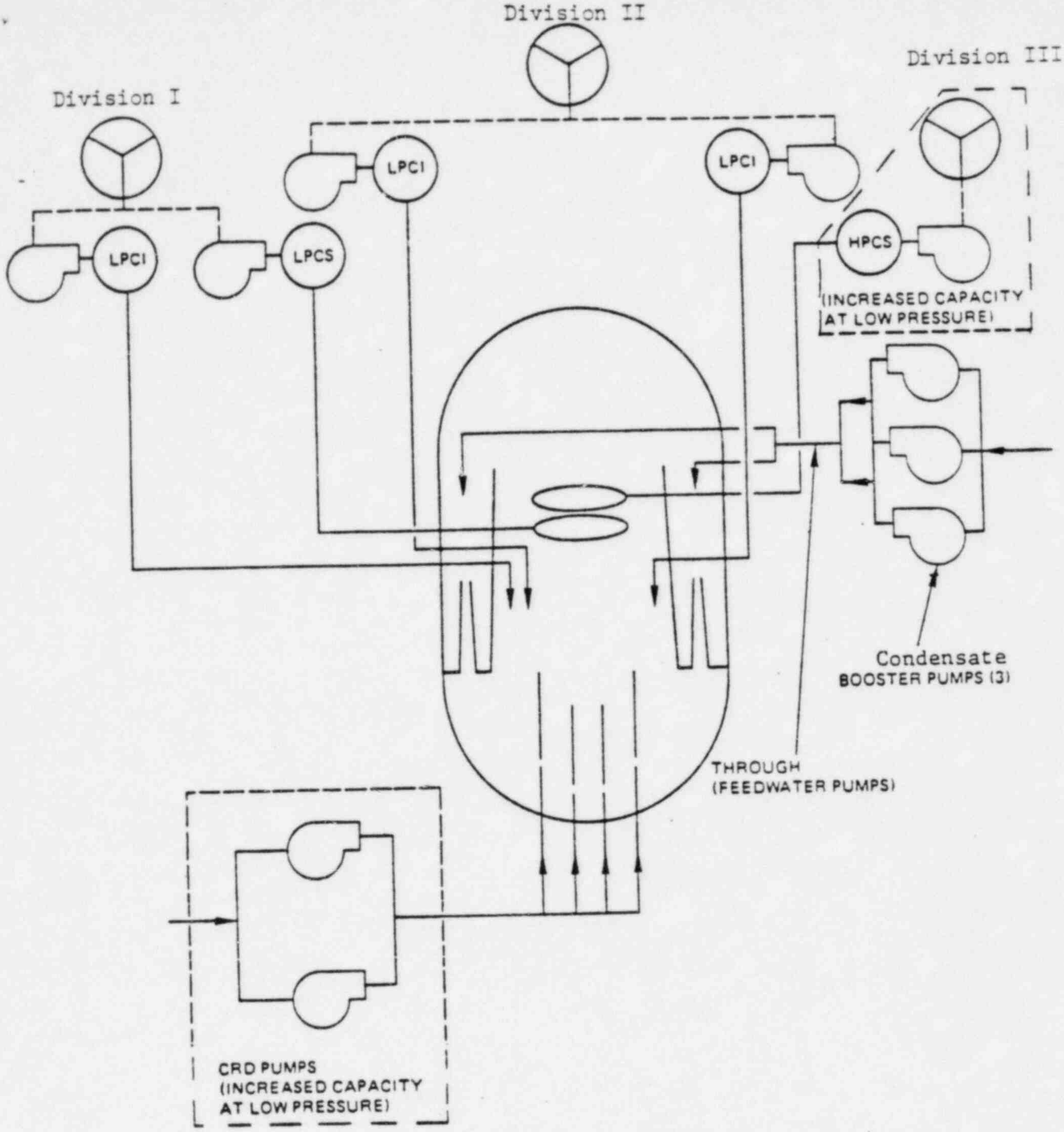


Figure 2.1.1-2. Typical 238 BWR/6 High Pressure Water Delivery Sources



Rated Capacities - Low Pressure Systems

LPCS	6000 GPM	(1 pump)
LPCI	19,500 GPM	(3 pumps)
Condensate Booster	30,500 GPM	(3 pumps)

High Pressure Systems with Increased Capacity at Lower Pressure

HPCS	6000 GPM	(1 pump)
CRD	340 GPM	(2 pumps)

Figure 2.1.1-3 Typical 238 BWR/6 Low Pressure Water Delivery Sources.

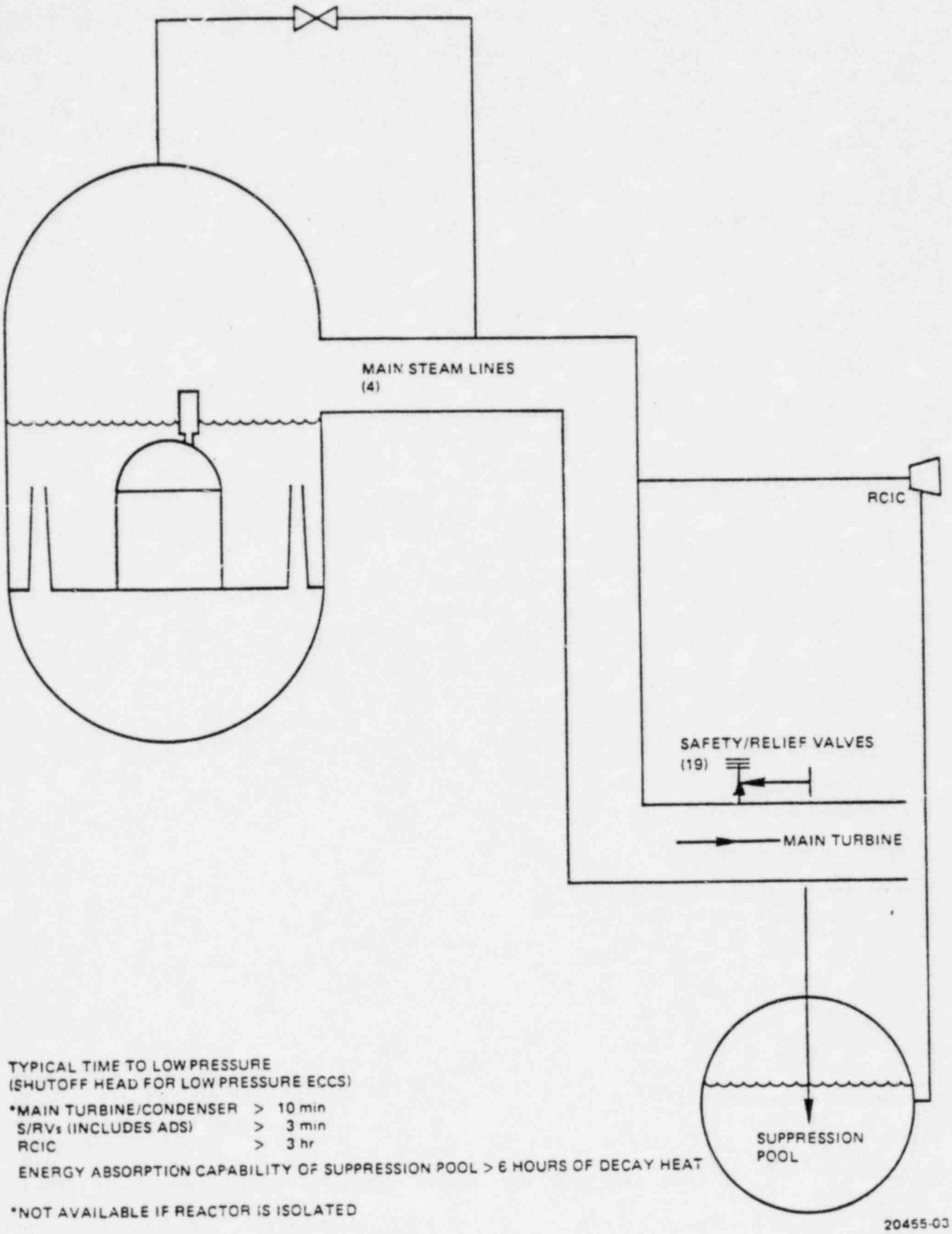
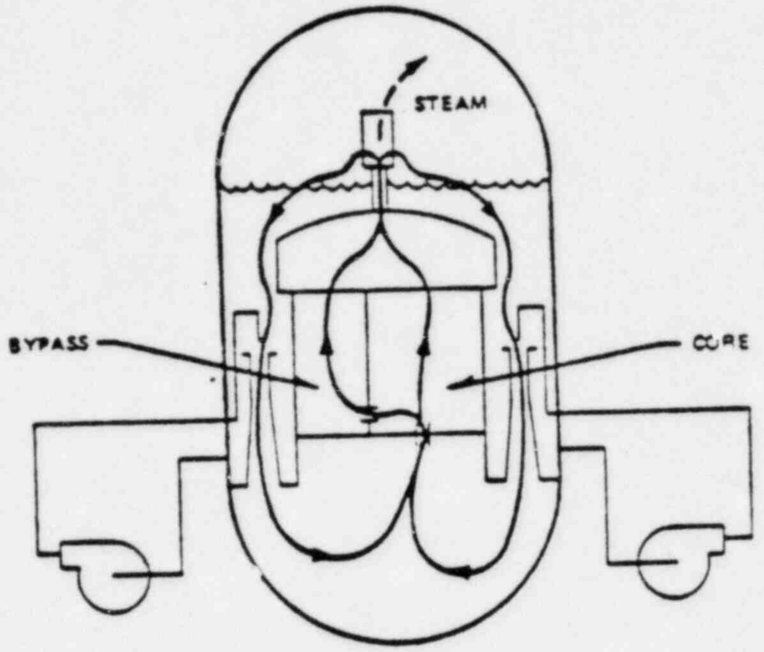


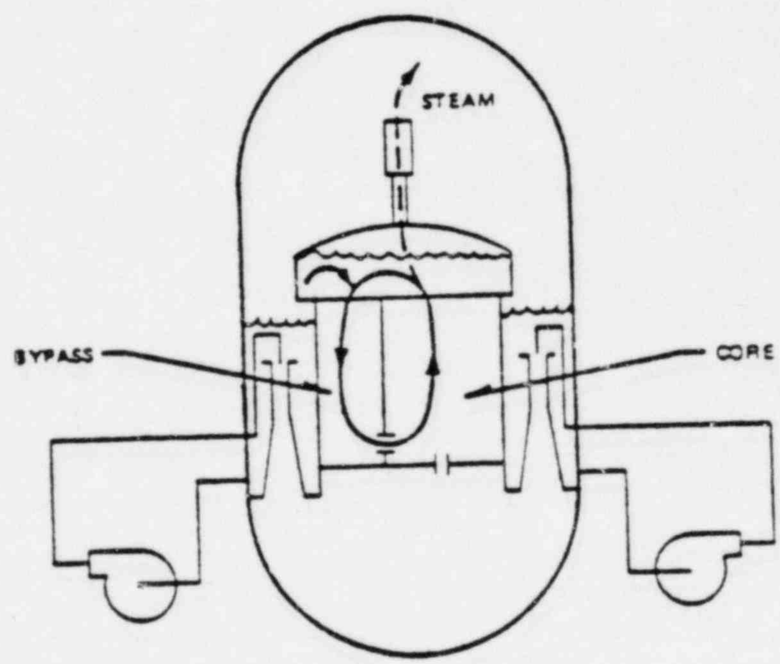
Figure 2.1.1-4. Typical 238 BWR/6 Venting and Depressurization Schemes

Figure 2.1.1-5

Typical BWR Natural Circulation



A. NORMAL NATURAL CIRCULATION



B. INTERNAL LOOP CIRCULATION

3. HYDROGEN GENERATION RATES

The calculation of hydrogen generation rates for various events requires knowledge of the core heatup behavior. This calculation can be performed using the analytic methodology described in Section 3.1. A detailed description of the analysis of the hydrogen generation event (HGE) is given in Section 3.2, including event progression, major analysis inputs, and results. A sensitivity study was performed to determine the effect of further delaying the recovery of makeup water until all the water in the RPV lower plenum has been depleted, as discussed in Section 3.3. Hydrogen generation rates for other scenarios are briefly discussed in Section 3.4, including transient with stuck open relief valve, small liquid line break, and anticipated transient without scram (ATWS). The non-mechanistic upper bound for hydrogen generation is also discussed in Section 3.4.

3.1 METHODOLOGY DESCRIPTION

Evaluation of the hydrogen generation rate is based on an extrapolation from GE's current loss of coolant accident (LOCA) analysis methods. An approach to extrapolate current LOCA models is needed to determine the hydrogen generation rate for the postulated hydrogen generation event (HGE) as defined in Chapter 2. The postulated HGE was defined as being initiated by loss of feedwater with temporary failure of all water delivery systems. This event would progress slowly, and would initially require minimal operator attention. It is assumed that the operator would concentrate his attention on restoring the water delivery function and would succeed within a reasonable period. Under these conditions the core would slowly uncover, would begin to heat up from a temperature near the saturation temperature of the water, and might eventually reach temperatures at which the rate of metal-water reaction becomes significant. It is assumed that the operator succeeds in restoring the water delivery function at a reasonable time after determining that all the systems have failed. Restoring any one of the water delivery systems (other than CRD) will result in rapid refilling of the RPV lower plenum, and rapid reflooding and quenching of the core. The metal-water reaction (MWR) is terminated for the HGE when the hottest part of the core has been cooled below about 1800°F.

The methodology used in calculating the hydrogen source term can be divided into five steps:

1. The initial blowdown calculation is performed by using a RPV blowdown model (the SAFE code). This model determines the RPV pressure, water level, water inventory, steam flow rate, and core uncovery time.

2. After the core has uncovered, the steam cooling calculation, which accounts for the steam superheating, provides the necessary information on the steam cooling heat transfer coefficient (HTC) from the fuel rod to the surrounding steam. These calculations are performed during the transient when the core is partially or fully uncovered and both convective and radiative heat transfer from the rod to the steam are important.

3. The core heatup calculations are performed by using a core heatup model (the CHASTE code). The core is divided into several nodes and the model computes the nodal temperature transient and the metal water reaction of cladding and channels for surface temperatures up to 3366°F, which is the melting point of Zircaloy.

4. For cladding temperatures above 3366°F, calculations are performed to model the core melt pattern and further generation of hydrogen and steam as the molten parts of the core slump into the water in the lower plenum.

5. For the time period when water makeup is restored (called the recovery phase), calculations are started at the point in the event when the water delivery system is made available. The profiles of core temperature and core oxidation fraction are taken from Step 3 results at that time. The recovery phase calculation involves solution of the coupled equations for mass and energy balances for the core (fuel, cladding, and channel) and for the liquid and steam within the RPV. The core is divided into several nodes; the liquid and steam are modeled as single nodes. The model computes the nodal temperature transient, metal-water reaction on cladding and

channels, steam cooling and evaporative cooling of the core, and water level rise or fall in the core. The calculation is terminated when the temperature of the hottest unslumped part of the core drops below 1800°F.

Additional descriptions of the various models used are given in Appendix A, and the use of these models for the calculation of hydrogen generation rates is dicussed in Sections 3.2.2 and 3.3.

3.2 HYDROGEN GENERATION EVENT

Earlier in this study, attempts were made to define a mechanistic hydrogen generation event (HGE) which could result in significant hydrogen generation with insignificant core-melt. This section discusses the analysis and provides the results of calculations for the scenarios which were considered in defining the HGE. Two separate scenarios were individually investigated in depth. For both scenarios, the initiating event was selected to be turbine trip with bypass valves operating, based on the frequency of occurrence and the probability of the initiating event leading to core-melt. The two scenarios differ in that the first scenario is based on the assumption that system failures are binary (either completely failed or completely available) while the second scenario is based on the assumption that system failures can be temporary.

The first scenario was postulated by determining which system or combination of systems would provide the smallest flow rate of makeup water adequate to prevent significant core-melt. For this scenario, it was assumed that the Reactor Core Isolation Cooling (RCIC) system initiates at Level 2, trips off at Level 8, and fails to restart when the RPV water level falls to Level 2 again. All other water makeup systems except the two Control Rod Drive (CRD) pumps are assumed unavailable. The operator is assumed to depressurize the reactor by opening 8 Safety/Relief valves when the water level reaches the top of active fuel (TAF).

System pressure and water level as functions of time for the first scenario are shown in Figures 3.2-1 and 3.2-2. Since over half of the core is covered with water at all times in the transient (Figure 3.2-2), the steam generated in the covered part of the core is adequate in providing the cooling to the uncovered part of the core, keeping the peak cladding temperatures below the threshold temperature (~1800°F) at which metal-water reaction is important. The flow from two CRD pumps with optimized valve settings will eventually balance the water evaporated by the decay heat and the core will be reflooded completely. The hydrogen generation for this scenario is thus negligible. This analysis demonstrates that even with assumed multiple failures of makeup systems, little or no hydrogen generation results.

The second scenario was postulated by assuming the temporary failure of all water makeup systems. It was assumed that, within a reasonable period of time after the accident initiation, the operator would succeed in restoring one of the numerous water makeup systems. This enables the operator to reflood the core, thus terminating hydrogen generation and preventing significant core-melt. For this reason, the second scenario was selected as the HGE, for use in designing a hydrogen control system.

3.2.1 Hydrogen Generation Event (Event Progression)

The HGE was analyzed by conservatively modeling it as a reactor isolation event with temporary failure of all water delivery systems. At time zero the reactor water level is at normal operating water level, the RPV is scrammed, and closure of the main steamline isolation valves (MSIV's) is initiated. MSIV closure is completed and the RPV is isolated in about 5 seconds. After isolation the decay power in the fuel continues to generate steam from the water inside the RPV. This steam accumulates in the upper part of the RPV until it is vented to the suppression pool by the Safety/Relief Valves (S/RV's). As the water inside the RPV is boiled off by decay power and pressurizes the RPV, the S/RV's cycle open and closed to maintain RPV pressure near the normal operating value, as shown in Figure 3.2.1-1. Assuming that no makeup water is initially available, then the water level inside the RPV drops slowly, as shown in Figure 3.2.1-2, uncovering the top of active fuel (TAF) at about 15.5 minutes after scram, and dropping to the middle of active fuel (MAF) at about 23 minutes after scram. When the water level has reached MAF, the operator is assumed to begin RPV depressurization by opening one of the S/RV's. The operator is assumed to open seven more valves as the RPV pressure drops. Depressurization of the RPV generates a large amount of steam, due to flashing and boil-off, most of which flows through the core providing significant steam cooling for a period of time. Depressurization also causes the water level to fall rapidly and the entire core becomes uncovered. Finally, depressurization of the RPV must be at least partially completed before the operator can verify the successful operation of one or more of the low pressure water delivery systems. For the HGE, it is assumed that after the RPV has depressurized to below the shutoff head for the low pressure systems (at 31.0 minutes after scram), the operator attempts to verify operation of each of these systems, but without immediate success. After a delay of reasonable duration, it is assumed that the operator succeeds in restoring the high pressure core spray (HPCS) system. As a result, the core is rapidly reflooded and cooled to below the lower threshold temperature for MWR.

During the period after the core is completely uncovered and before recovery of the water delivery system, convective steam cooling of the core becomes increasingly less than adequate to prevent core heat up and subsequent MWR as shown. The rate of MWR of fuel cladding is insignificant while the cladding temperatures are below about 1800°F. The rate of MWR increases with increasing temperature. If the cladding at a given location in the core reaches the melting point of Zircaloy before being reflooded, that particular core node is assumed to slump and fall into the RPV lower plenum water. During quenching of the slumped nodes, additional MWR occurs and substantial steam is generated.

For the HGE it is assumed that the operator attempts to initiate the high pressure and low pressure water makeup systems at the appropriate times but is initially unsuccessful. It is further assumed that by concentrating his attention on restoring the water delivery function, the operator will succeed in a reasonable period of time in recovering a water delivery system. For this initiating event, the high pressure water makeup system should have been initiated on a low water level signal approximately 3.5 minutes after the start of the accident. Flow from the low pressure systems into the reactor should have commenced at about 31. minutes (when RPV pressure fell below the shutoff head of the low pressure systems) if they had functioned as designed. For the analysis it was assumed that the operator succeeded in restoring one of the water delivery systems at 46. minutes. By that time, the operator would have been aware of the unavailability of all water makeup systems (high pressure and low pressure) for at least 15 minutes and would have had a reasonable amount of time to correct most systems problems. Furthermore, by 46 minutes the operator would have been aware of unavailability of the high pressure ECC makeup water system for over 40 minutes.

By the time of makeup water initiation, decay heat (and some MWR heat) would have raised the core temperatures to the extent that the hottest parts of the core would have reached the melting point of Zr (3366°F). As the water level rises into the hot core, substantial steam would be generated providing steam cooling for the unsubmerged parts of the core. Local steam blanketing around the submerged portions of the rods would persist for only a matter of seconds.

3.2.2 Hydrogen Generation Event (Major Analysis Inputs)

In using the methodology for calculating steam cooling and the core heatup models (CHASTE) the reactor core was conservatively divided into two radial zones. One is the peak power zone with the radial power peaking factor (PPF) of 1.4, and the other is the average power zone with the PPF of 1.0. In the core heatup modeling using the CHASTE code, each radial power zone calculation is represented by a peak and an average power bundle. The peak power bundle CHASTE calculation gives a conservative prediction of the starting time for hydrogen generation since the higher the power of a bundle, the earlier the hydrogen starting time. The average power bundle CHASTE calculation gives a good representation of the core-wide hydrogen generation behavior since about two thirds of the core radial PPFs are between 0.8 and 1.20. Also in the steam cooling and the CHASTE calculations, there are five axial nodes. Each axial node represents a 2.5 foot length of a bundle. The axial PPFs of 0.127, 0.230, 0.275, 0.227, and 0.141, are incorporated in both the steam cooling and core heatup calculations.

The portion of the fuel cladding above the top of the active fuel (about 1 foot in length) is not expected to contribute to the core heatup and MWR calculations. The core heatup rate in this part of the cladding is expected to be low due to a lack of internal generation and relatively more mass of materials (mass of internal springs, end plugs, etc.) to be heated up. The MWR from this position of the cladding is considered negligible. The total initial mass of active Zircaloy cladding modeled in these MWR calculations is 75,000 lbs (for the standard 238 BWR/6 plant).

For this analysis, the water makeup system which was assumed to have been made operational by the operator is the high pressure core spray (HPCS) system. The RPV pressure used for the recovery phase calculation is conservatively taken to be 400 psia with the HPCS flow corresponding to this pressure being approximately 5000 gpm. This assumption accounts for the reactor pressurization from steam generated during quenching of hot core.

3.2.3 Hydrogen Generation Event (Results)

The results of the MWR calculation described above are plotted in Figure 3.2.3-1 for the hydrogen generation event. These results provide the necessary information on the time of start of substantial MWR and hydrogen generation, the metal-water reaction rate and the maximum possible MWR. Plots of local cladding temperature (at the core mid plane) versus time are shown in Figures 3.2.3-2 and 3.2.3-3 for the average and peak power bundles respectively. The mass generation rates of hydrogen and steam are given in Figures 3.2.3-4 and 3.2.3-5, respectively. Hydrogen and steam generated inside the reactor are released through the S/RV's to the suppression pool.

The HPCS initiation time corresponds to about 46. minutes after the event is initiated. At the HPCS initiation time, approximately 5% of the core was calculated to have melted. The maximum zirconium reacted is limited to the equivalent of 12.5% of the zirconium cladding surrounding the active fuel. The hydrogen generation rate as a result of this MWR can be determined as follows: one percent of equivalent MWR of the cladding surrounding the active fuel corresponds to 33. lbs of hydrogen.

3.3 SENSITIVITY STUDIES:

The recovery phase of the HGE discussed in detail in Section 3.2 was based on the assumption that the operator would succeed in restoring an ECC system 46 minutes after the accident initiation. As indicated in Section 3.2, the HGE is expected to result in hydrogen generation equivalent to the reaction of 12.5% of the cladding surrounding the active fuel without substantial core melt. A sensitivity study is presented here to assess the effect on the overall hydrogen generation of recovering a water makeup system at a later time into the accident than 46 minutes, as was considered in the HGE recovery phase. The scope of this sensitivity study was to consider the additional hydrogen generation which would result from restoring a 5000 gpm water makeup system after all of the water in the RPV has been depleted by boiloff, depressurization, and quenching of slumped core fractions. This sensitivity study is presented here with the recognition that the HGE is the appropriate basis for design of a hydrogen control system, since it leads to significant hydrogen generation without substantial core melt. Accidents with longer delay in restoring water makeup are expected to result in substantial core melt and are, therefore, not appropriate as a basis for design of a hydrogen control system.

3.3.1 RESTORING WATER MAKEUP AT TIME OF DEPLETION OF RPV WATER

This event is similar to the HGE except that recovery of the water makeup system is delayed much longer than 46 minutes. The core heatup, metal-water reaction, and core slumping would continue at essentially the same rate as for the HGE until a water makeup system becomes available. The mass of water which remained in the RPV after depressurization would be completely boiled off at about 57 minutes, after quenching the fraction of the core (~70%) which would have slumped into the lower plenum by this time. At this time, a large mass of dry, relatively cool core debris might be expected to reside in the lower plenum, and a smaller mass to remain intact in the core at various temperatures below the melting point of Zircaloy. If a water makeup system were restored at this point, the water would begin filling the RPV lower plenum, with a low steam generation rate due only to the decay power of the core debris in the lower plenum. (For this study, the core spray heat transfer coefficient was taken as zero. This is conservative with regard to hydrogen generation.) At such a steam flow rate the steam cooling effect on the remaining intact portion of the core is small, but sufficient steam is provided for hydrogen generation to continue until the water level reaches the core and quenches it.

For this case, the amount of hydrogen generation after recovery of the water makeup system was calculated based on the analytical methodology developed for the HGE recovery phase (described in Appendix A.5). The physical parameters needed for this analysis were derived from the thermal conditions in the core and RPV lower plenum at the time of depletion of the initial inventory of water in the RPV. The results of this sensitivity study are given in terms of the additional metal-water reaction expected to occur after startup of the water makeup system for various restoration times of the makeup water system. If the water makeup system were restored at 57 minutes, the time at which initial RPV inventory of water would be depleted, the additional hydrogen generated after startup of the water makeup system would be equivalent to the reaction of 2.2% of the cladding surrounding the active fuel. For comparison, the additional amount of hydrogen generated after startup of the water makeup system for the HGE (see Section 3.2) would be equivalent to the reaction of 7.1% of the cladding surrounding the active fuel, out of a total metal-water reaction of 12.5%.

3.3.2 RESTORING WATER MAKEUP LATER THAN TIME OF DEPLETION OF RPV WATER

If the water makeup system is not restored until several minutes after depletion of the initial RPV inventory of water (at 57 minutes), the sources of hydrogen generation which should be considered include:

(a) metal-water reaction which could be supported by the stagnant steam in the RPV during the period between depletion of the initial inventory of RPV water and startup of the restored water makeup system.

(b) metal-water reaction resulting from pouring water on top of the hot, slumped core debris in the RPV lower plenum, and

(c) metal-water reaction in the intact remainder of the core during reflooding of the RPV.

If the water makeup system is not restored until several minutes after 57 minutes, then the stagnant steam in the RPV would continue to support hydrogen generation for a short period of time, until either a water makeup system is restored or all of the stagnant steam is consumed by MWR. The remaining intact portion of the core would continue to heat up (by decay power and by stagnant steam MWR) and some of it might be expected to slump into the RPV lower plenum, adding hot dry core debris on top of the relatively cool, previously quenched, dry core debris already there. Adding water under these conditions would give rise to vigorous boiling and additional metal-water reaction in the hot slumped core debris. Very little additional metal-water reaction is expected in the larger fraction of core debris which had been cooled by quenching before depletion of the original inventory of RPV water. Under these circumstances, the duration of rapid metal-water reaction in the intact remainder of the core would be limited to a few seconds after startup of the water makeup system since the steam cooling associated with the very high steam flowrate immediately provides very effective cooling to the core region and the core temperatures quickly fall below 1800°F, terminating any further MWR in the core.

In considering stagnant steam MWR, it was estimated that all of the stagnant steam would be consumed before 70 minutes, and would provide for additional hydrogen generation equivalent to the reaction of 3.8% of the cladding surrounding the active fuel, if a water makeup system were not recovered before that time.

As noted above, MWR would also take place both in the slumped core debris in the RPV lower plenum and in the intact portion of the core during the time from restoration of a water makeup system to reflooding of the core. For this sensitivity study, the extent of MWR resulting from pouring water over hot, slumped core debris was estimated based on conservatively assuming the complete reaction of the unreacted Zircaloy in the fraction of core slumped during the period from 57 minutes (when the initial RPV inventory of water is depleted) to the assumed time of restoration of a water makeup system. The extent of MWR in the intact core during the recovery phase is expected to be insignificant (<.1%) for assumed times of restoration of water makeup later than 57 minutes, because of the substantial steam cooling which would be provided by pouring water over the hot, slumped core debris. If the water makeup system were started at 70 minutes after the accident initiation, the additional hydrogen generation after startup of the water makeup system is conservatively estimated to be limited to the equivalent of reaction of 19% of the cladding surrounding the active fuel. Similarly, if the water makeup system were started at 80 minutes after accident initiation, then the additional hydrogen generation after startup of the water makeup system is conservatively estimated to be limited to the equivalent of reaction of 25% of the cladding surrounding the active fuel. Stagnant steam metal-water reaction is not included in these figures.

3.3.3 SENSITIVITY STUDY SUMMARY

This sensitivity study was undertaken to assess the impact on hydrogen generation with respect to the time at which the water makeup system is restored. For the HGE (discussed in Section 3.2), the water makeup system is recovered at 46 minutes, and an additional hydrogen generation equivalent to the metal-water reaction of 7.1% of the cladding surrounding the active fuel (out of a total MWR of 12.5%) results after initiation of the water makeup system. If the water makeup system is restored at 57 minutes (just when the initial liquid inventory of water in the RPV is depleted), less hydrogen is generated during the post-injection recovery phase than during the corresponding period for the HGE. In this case, additional hydrogen generation equivalent to the metal-water reaction of 2.2% of the cladding surrounding the active fuel occurs after the initiation of the water makeup system. If the water makeup system is not restored until somewhat later than 57 minutes, then substantial hydrogen generation may result from pouring water over the hot slumped core debris. Accidents with longer delay in restoring a water makeup system are expected to result in substantial core-melt and are, therefore, not appropriate as a basis for design of a hydrogen control system.

3.4 DISCUSSION OF HYDROGEN GENERATION RATES FOR OTHER SCENARIOS

The hydrogen generation rate for the hydrogen generation event (HGE) was presented in Section 3.2.3. This section presents a qualitative assessment of the hydrogen starting time and the hydrogen generation rate expected for other events such as transient with stuck open relief valve (SORV) small liquid break (0.1 ft^2), and anticipated transient without scram (ATWS).

For scenarios which are initiated by transients with temporary failure of water makeup systems, the hydrogen generation rates are well represented by the HGE. These transient initiating events include turbine trip (TT) and loss of offsite power (LOOP). However, for a transient with SORV or small liquid break or ATWS as the initiating event, the hydrogen generation rate is expected to be somewhat different from that of the HGE due to the fact that for these three initiating events, the loss of the reactor water inventory occurs somewhat faster than that for the HGE.

The hydrogen generation estimates given in this section are based on extrapolations from the HGE. For each of the scenarios discussed in this section (with the exception of the non-mechanistic upper bound presented in Section 3.4.4), an abbreviated analysis and extrapolation was performed. The reactor water inventory was calculated using SAFE, and the core heatup (up to the temperature at which MWR becomes significant) was calculated using the steam cooling calculation methodology. No detailed core heatup calculations were performed for the extrapolation cases. The starting time of hydrogen generation and the average hydrogen generation rate were extrapolated from the HGE, based on the core heatup rate, as approximated from the results of the steam cooling calculation methodology.

3.4.1 Transient With Stuck Open Relief Valve

The sequence of events for a transient scenario with SORV is similar to that of the HGE (see Section 3.2.2). The difference between the two cases is that after the S/RV's open for the first time during the transient, one S/RV fails to close as the RPV pressure drops below the closing setpoint (Figures 3.4.1-1 and 3.4.1-2). This S/RV remains open for the rest of the transient. In this event, the RPV is depressurized through one SORV relatively early in the transient. The RPV is further depressurized by the operator as discussed in Section 3.2.1. Due to a more rapid loss of the reactor water inventory for this case, compared with the HGE, core uncovery and heat up starts sooner. Using the extrapolation approach described above, it is estimated that the hydrogen start time is about 10 minutes sooner and the hydrogen generation rate is slightly higher (approximately by 10%) than the HGE.

3.4.2 Small Liquid Break (0.1 ft²)

For a scenario with a small liquid break as the initiating event, the sequence of events is also similar to that of the HGE (Section 3.2.1). The difference between these two cases is that a liquid break of the size of 0.1 ft² at the suction side of the jet pump recirculation line takes place at time zero. For a liquid break, the rate of loss of the reactor water inventory is much higher (3 to 4 times) than that through a steam break of the same break size and at the same RPV pressure (Figures 3.4.2-1 and 3.4.2-2). The RPV is depressurized through the break and is further depressurized by the operator as discussed in Section 3.2.1. In this case, the loss of the reactor water inventory is much faster than that of the HGE. Using the extrapolation approach described above, it is estimated that the hydrogen start time is about 20 minutes sooner and the hydrogen generation rate is higher (approximately by 30%) than the HGE.

3.4.3 Anticipated Transient Without Scram (ATWS)

The class of accident initiating events identified as ATWS (Anticipated Transient Without Scram) is characterized by failure of the control rods to insert upon demand. The Standby Liquid Control System (SLC) provides an alternative means for controlling the core power by injecting a solution of sodium pentaborate into the reactor, and the reactor can be safely shut down (with no MWR) following an ATWS event by using the SLC along with controlled flow of ECCS.

ATWS was considered to be an inappropriate choice for HGE for several reasons. The expected frequency of ATWS and the probability of an ATWS event leading to core-melt are very low, as shown in Tables 2.1.2-2 and 2.1.2-3, respectively. Furthermore, an ATWS event can lead to accident conditions which are pertinent to hydrogen control considerations only if the accident sequence includes improper function of all the major water makeup systems, resulting in core damage prior to loss of containment integrity. The combined probability of failure to scram plus improper function of all water makeup systems is even lower than the probability of failure to scram only. Also, if failure of the SLC is postulated concurrent with ATWS, then the properly controlled ECCS will be adequate to maintain core cooling but the capacity of the Residual Heat Removal System (RHR) for removing heat from the containment will not be adequate. In this case, the containment will lose its integrity (by steam over-pressurization) prior to core damage, regardless of whether or not there is a hydrogen control system installed in the plant.

For this study, based on the relative frequencies of various initiating events (Table 2.1.2-2), the ATWS sequence was assumed to be initiated by an operational transient such as loss-of-feedwater or turbine trip, and was modeled as an MSIV closure with failure to scram and failure of all water makeup systems. As noted above, this scenario is expected to have an extremely low probability. For this ATWS sequence, the rate of boiloff of reactor water inventory is higher (Figure 3.4.3-1) than for the HGE due to higher power in the early part of the accident. This results in the start time for hydrogen generation occurring about 30 minutes sooner than for the HGE. Based on extrapolation of core heatup rates, the average hydrogen generation rate for this case is expected to be higher (approximately by 30%) than for the HGE.

3.4.4 Non-Mechanistic Upper Bound

In the development of a hydrogen source term for the base case HGE presented in Section 3.2, it was explained that the MWR is mechanistically terminated when the core has been reflooded and the highest local core temperature has fallen below 1800°F. The maximum MWR for the HGE was demonstrated to be equivalent to reaction of 12.5% of the cladding surrounding the active fuel. As 75% or 100% MWR is not possible mechanistically, the MWR to 75% or 100% can only be obtained by extrapolating from the calculation for the HGE assuming the same rates as in Figure 3.2.3-1. The extrapolation is shown in Figure 3.4.4-1.

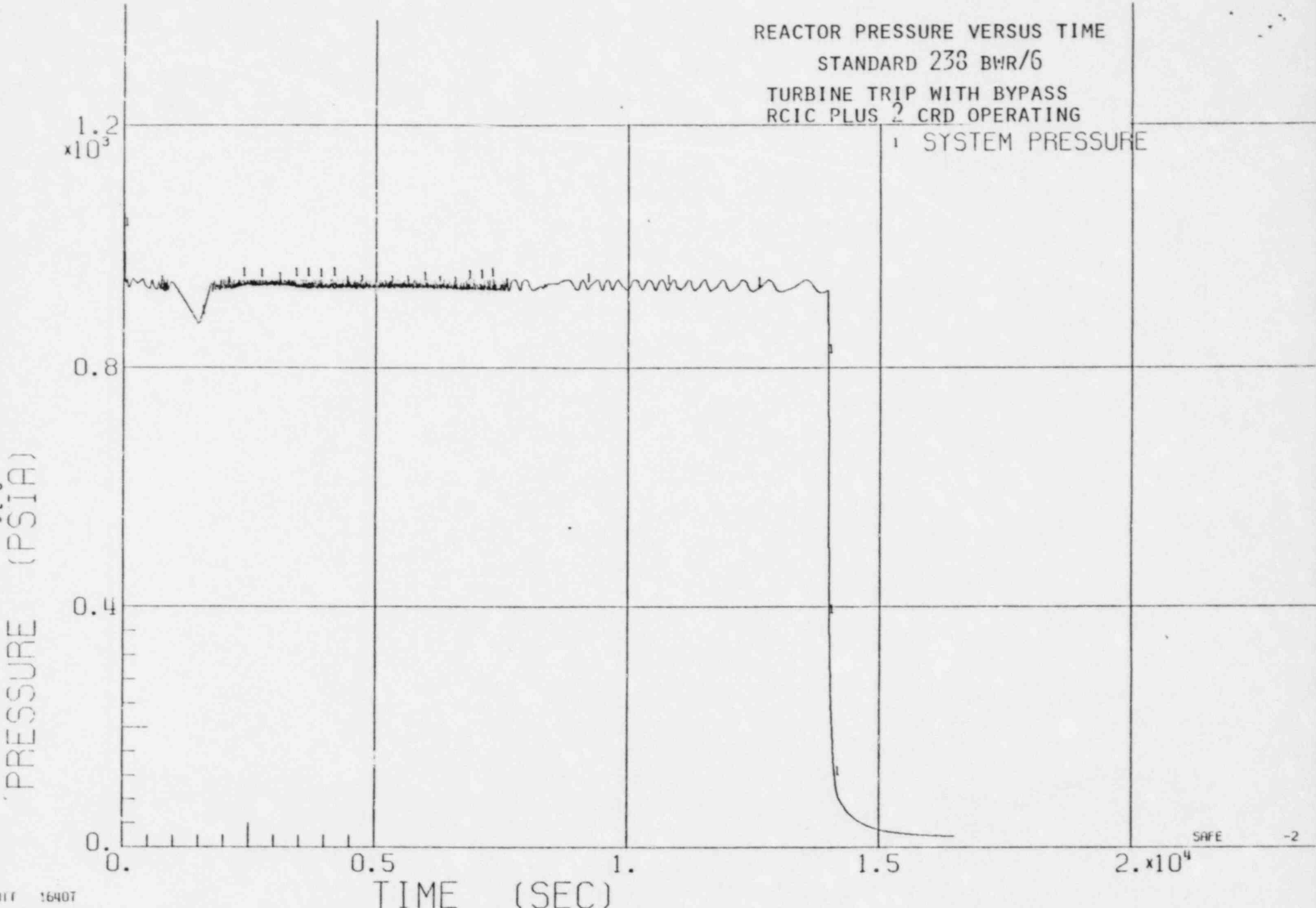


Figure 3.2-1 Reactor Pressure Transient for Turbine Trip With RCIC/CRD

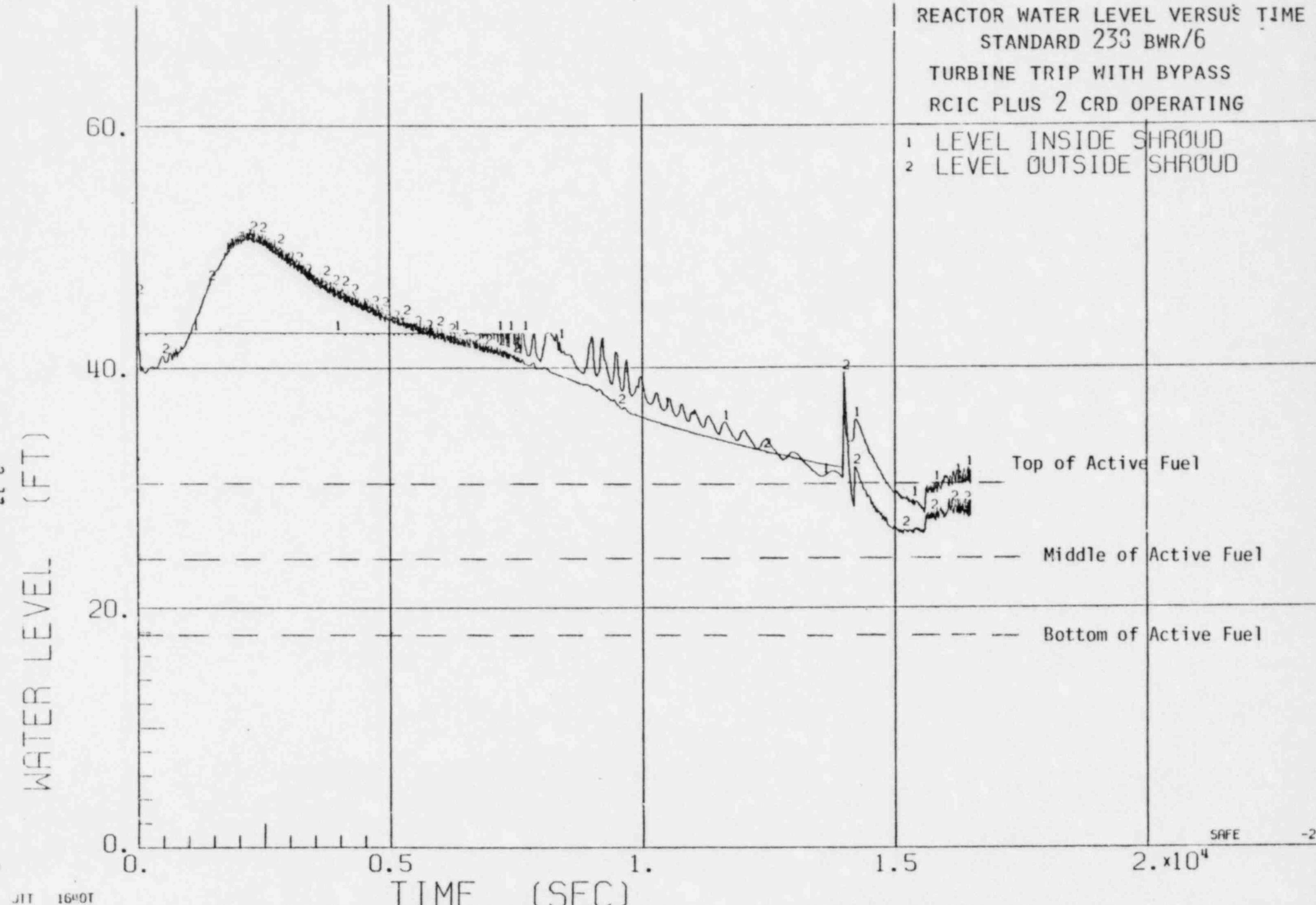


Figure 3.2-2 Reactor Water Level Transient for Turbine Trip with RCIC/CRD

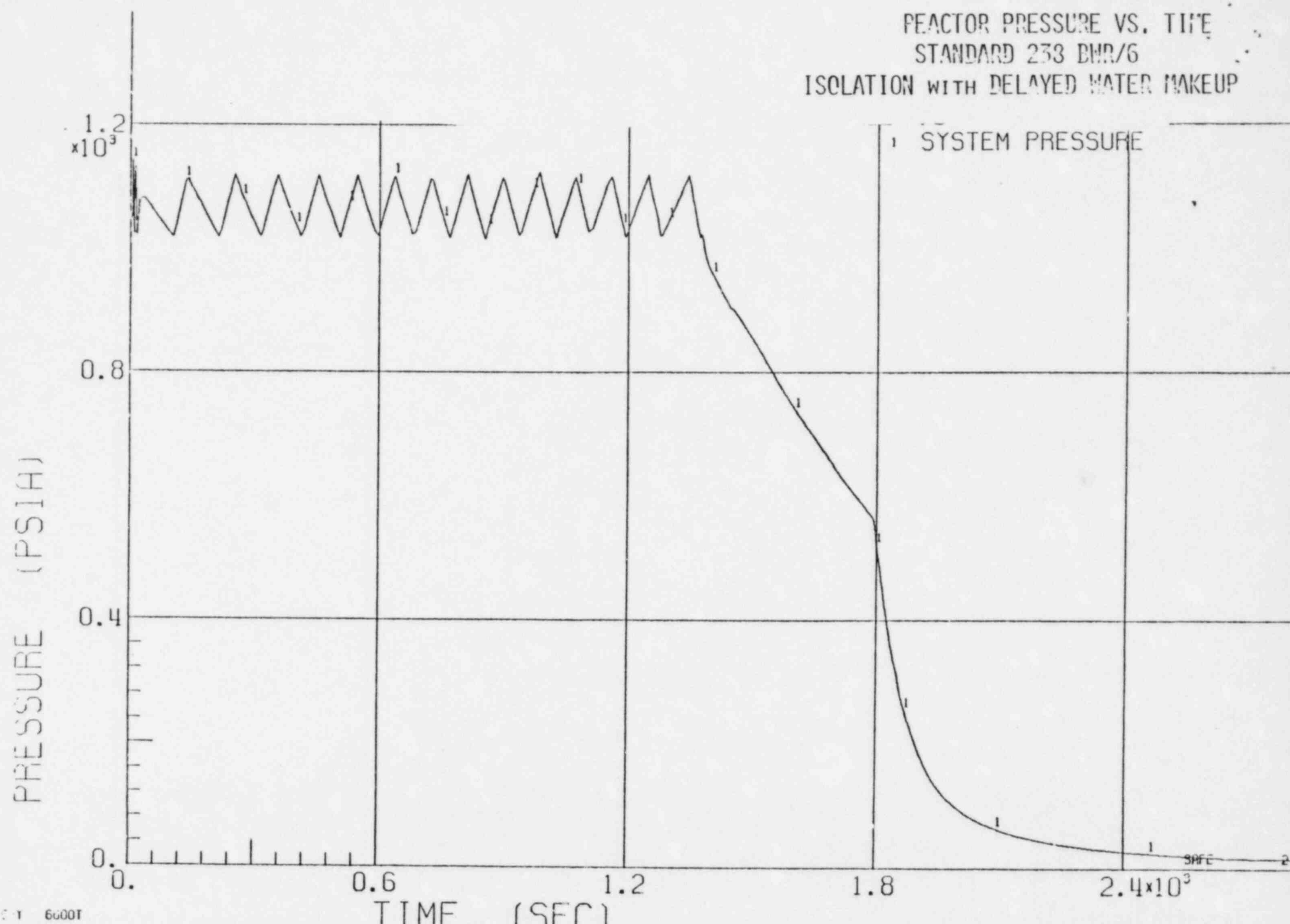


Figure 3.2.1-1 Reactor Pressure Transient for Isolation with Delayed Water Makeup

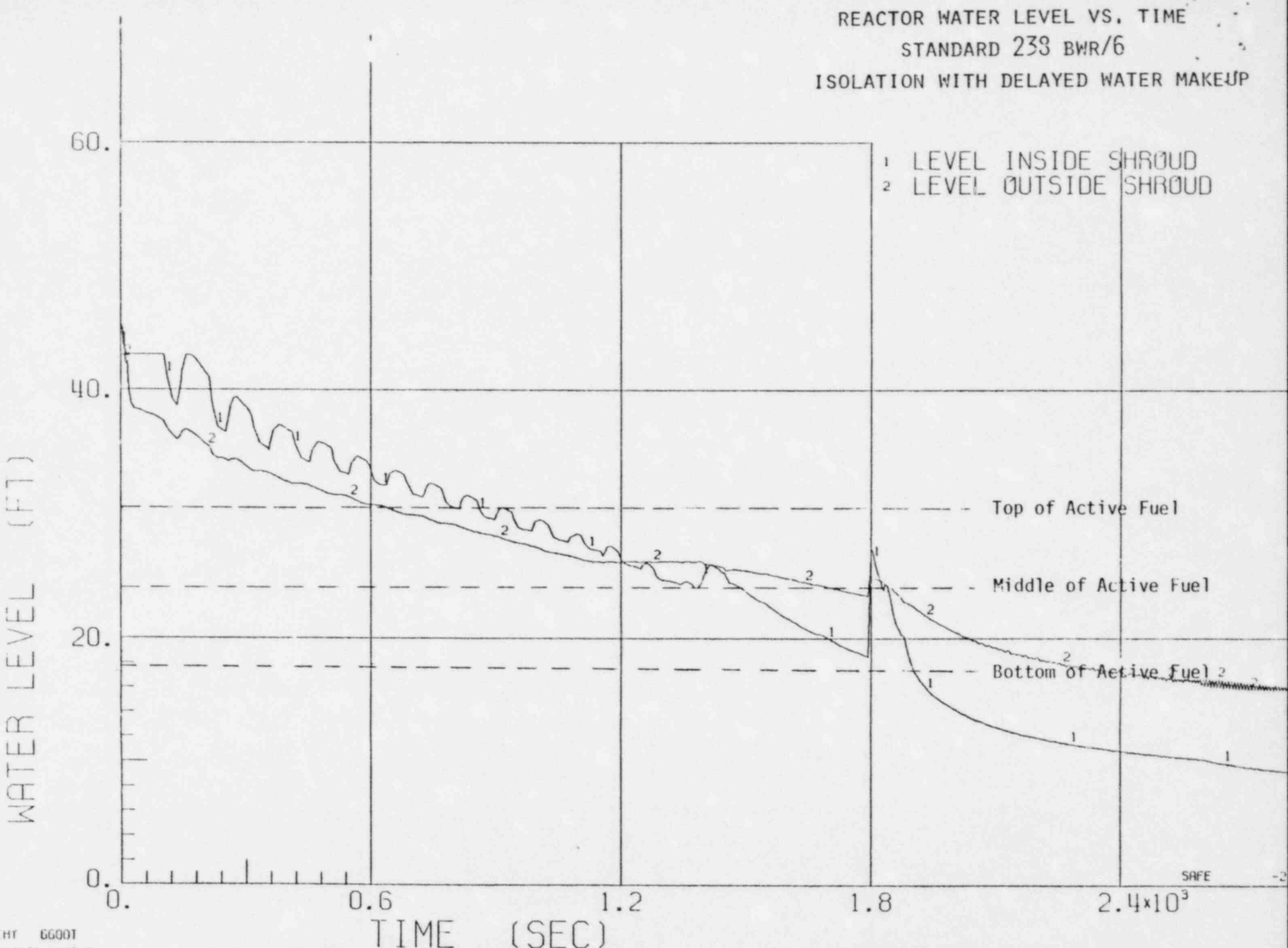


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Figure 3.2.1-2 Reactor Water Level Transient for Isolation With Delayed Water Makeup

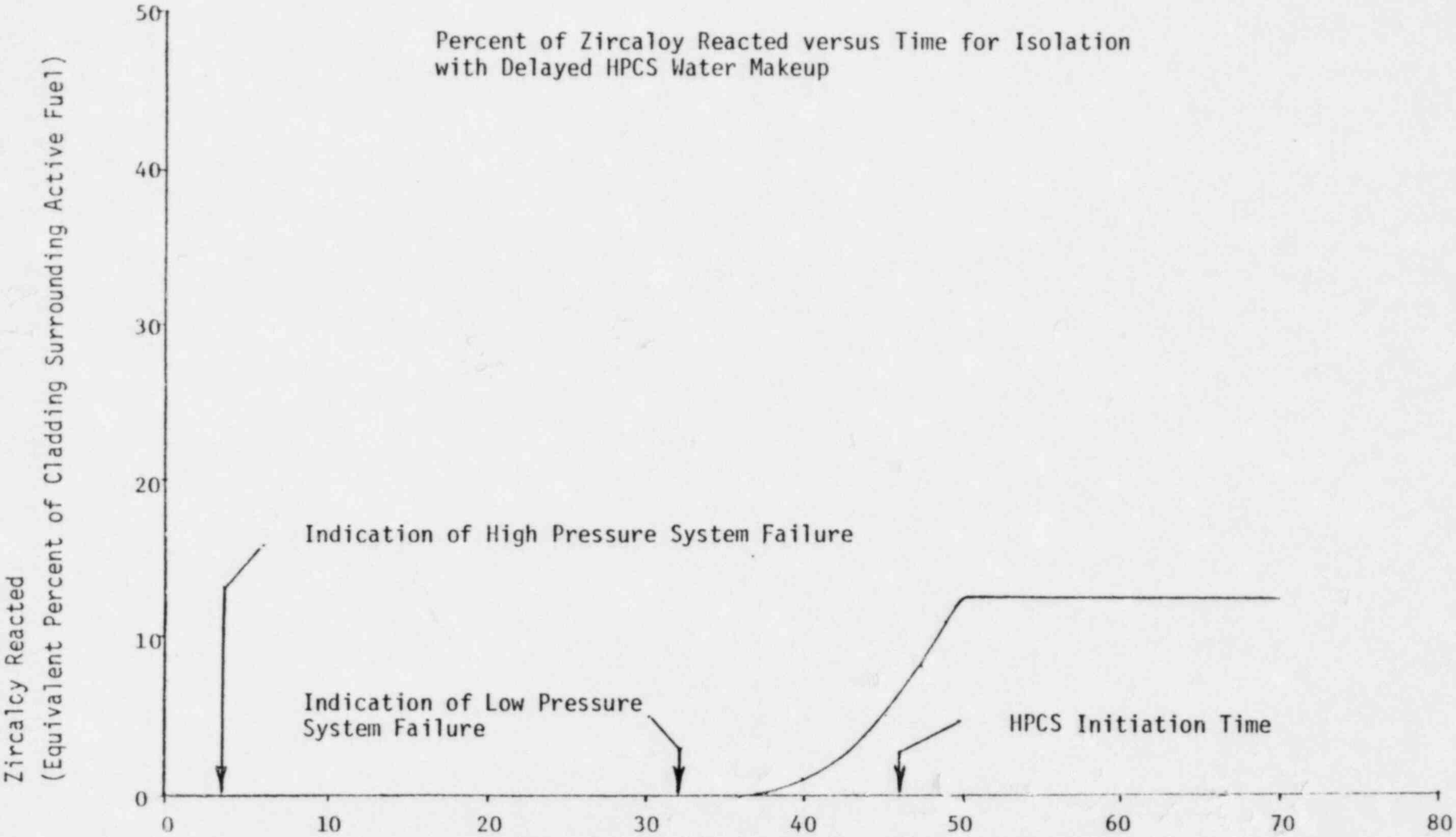

Figure 3.2.3-1 Percent of Zircaloy Reacted versus Time for Isolation with Delayed HPCS Water Makeup

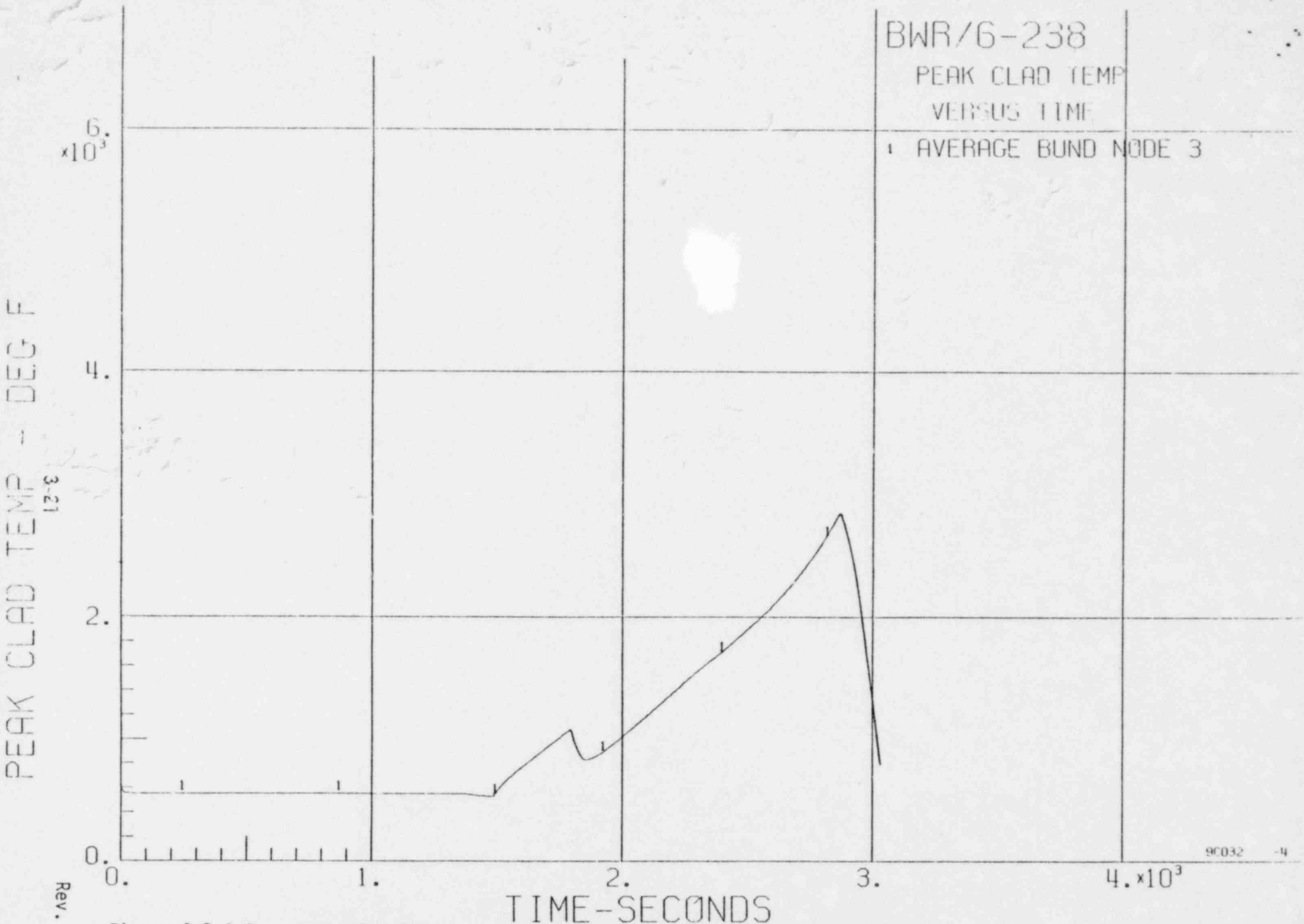

Figure 3.2.3-2 Peak Clad Temperature for Average Bundle Peak Power Axial Node (Isolation with Delayed Water Makeup)

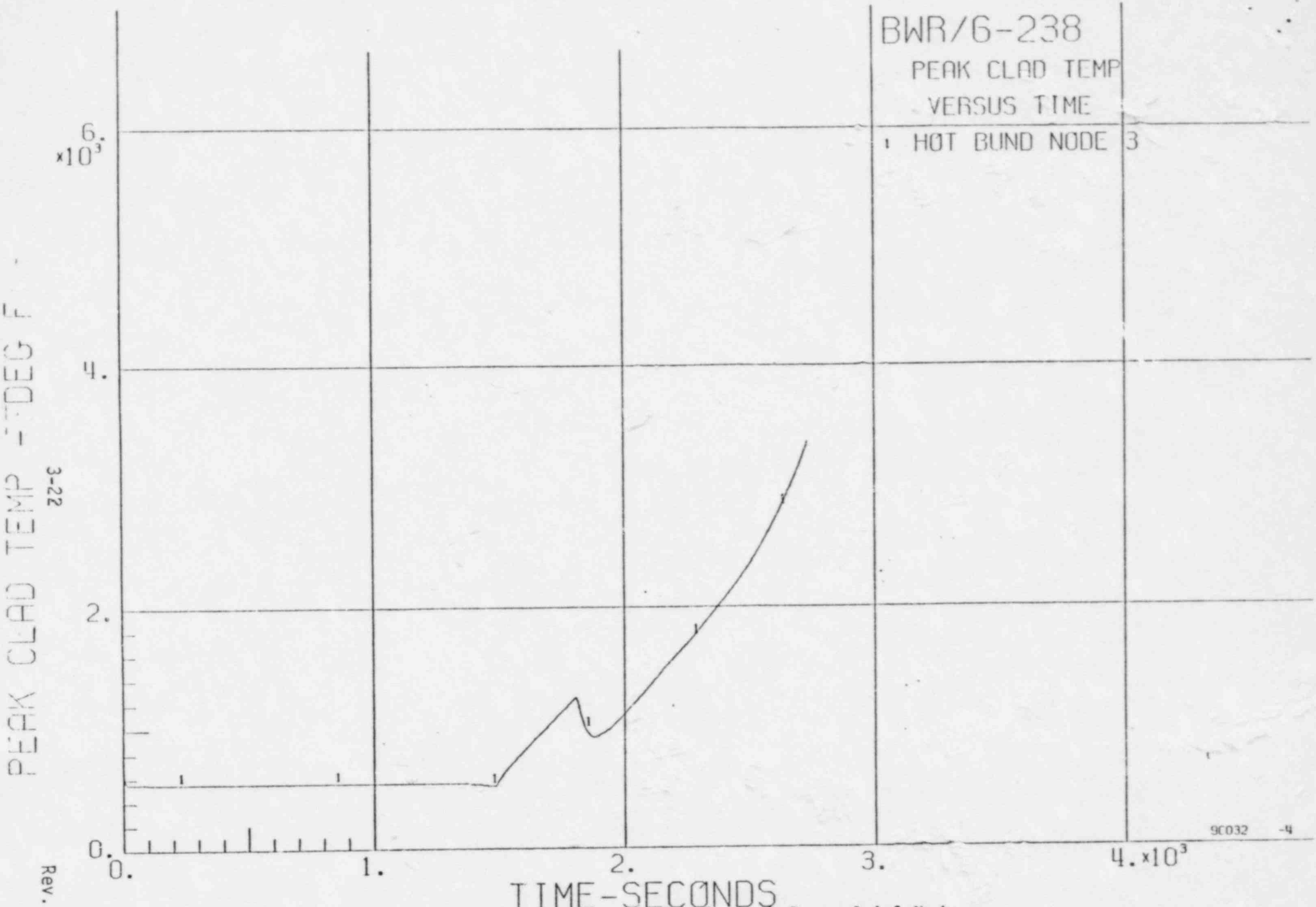

Figure 3.2.3-3 Peak Clad Temperature for Hot Bundle Peak Power Axial Node (Isolation with Delayed Water Makeup)

HYDROGEN MASS GENERATION RATE versus TIME

STANDARD 238 BWR/6

ISOLATION with DELAYED WATER MAKEUP

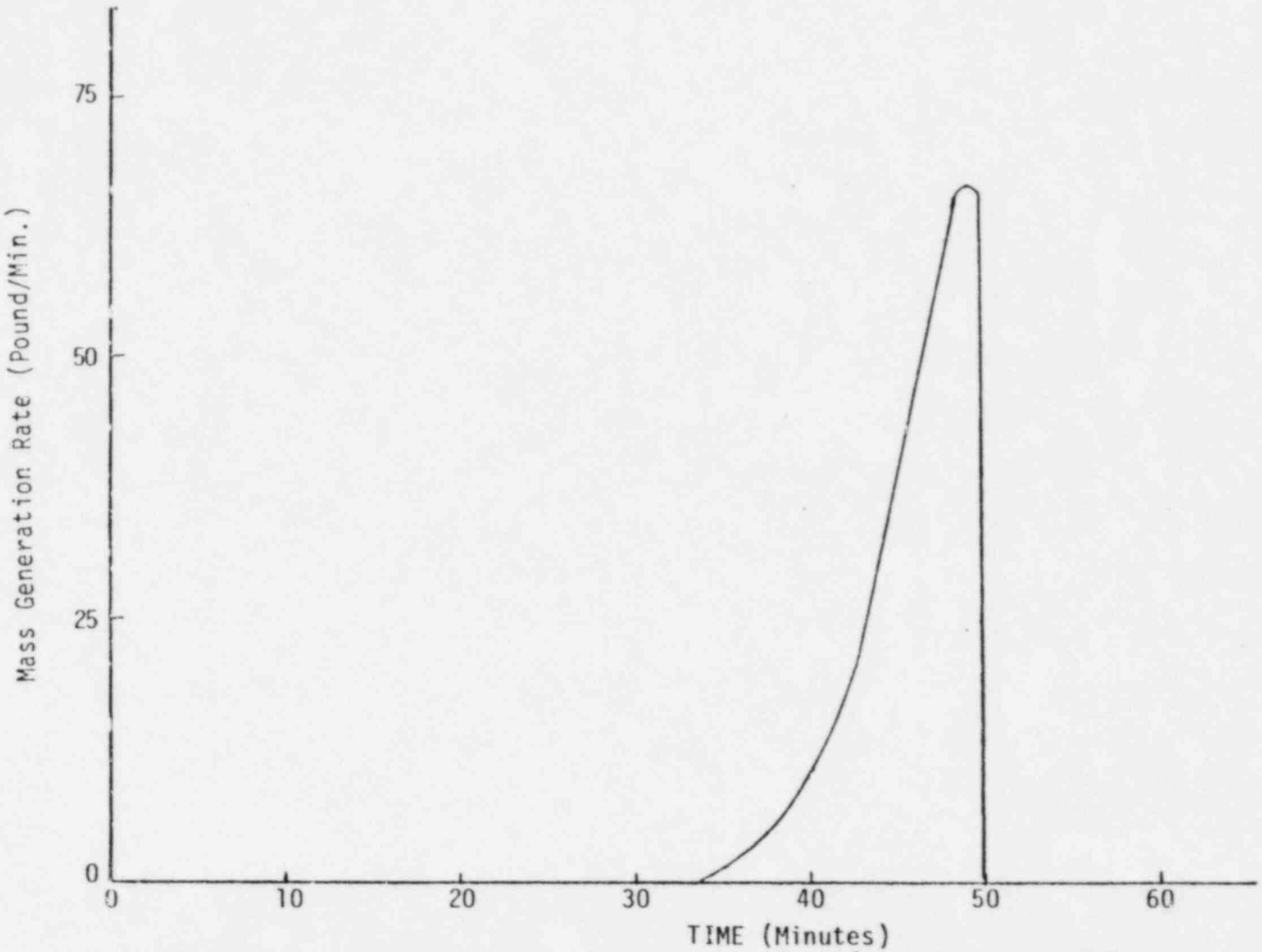

Figure 3.2.3-4 Hydrogen Mass Generation Rate vs. Time (Isolation with Delayed Water Makeup).

STEAM MASS GENERATION RAT' versus TIME
STANDARD 238 B.k/6
ISOLATION with DELAYED WATER MAKEUP.

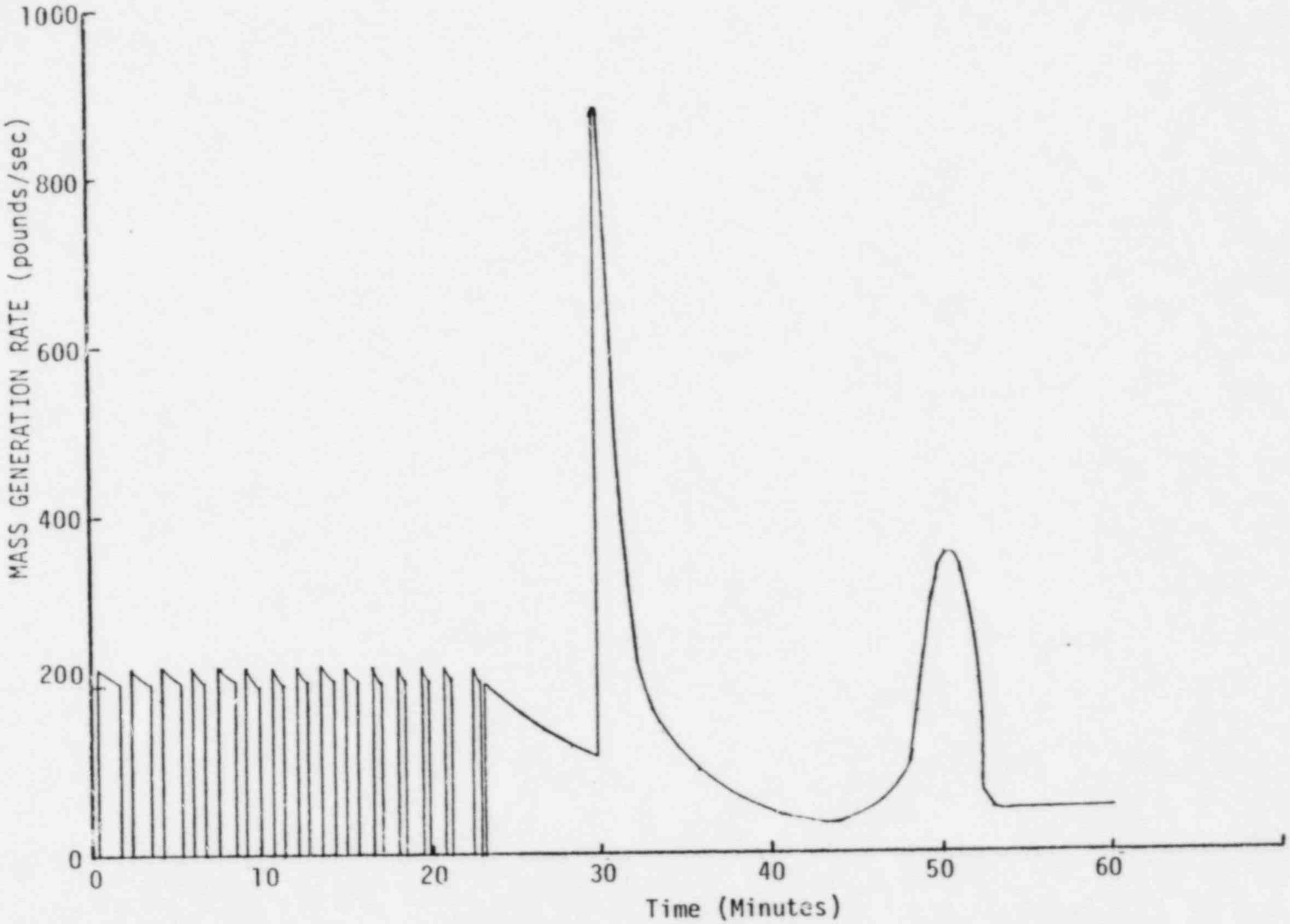

Figure 3.2.3-5 Steam Mass Generation Rate vs. Time (Isolation with Delayed Water Makeup).

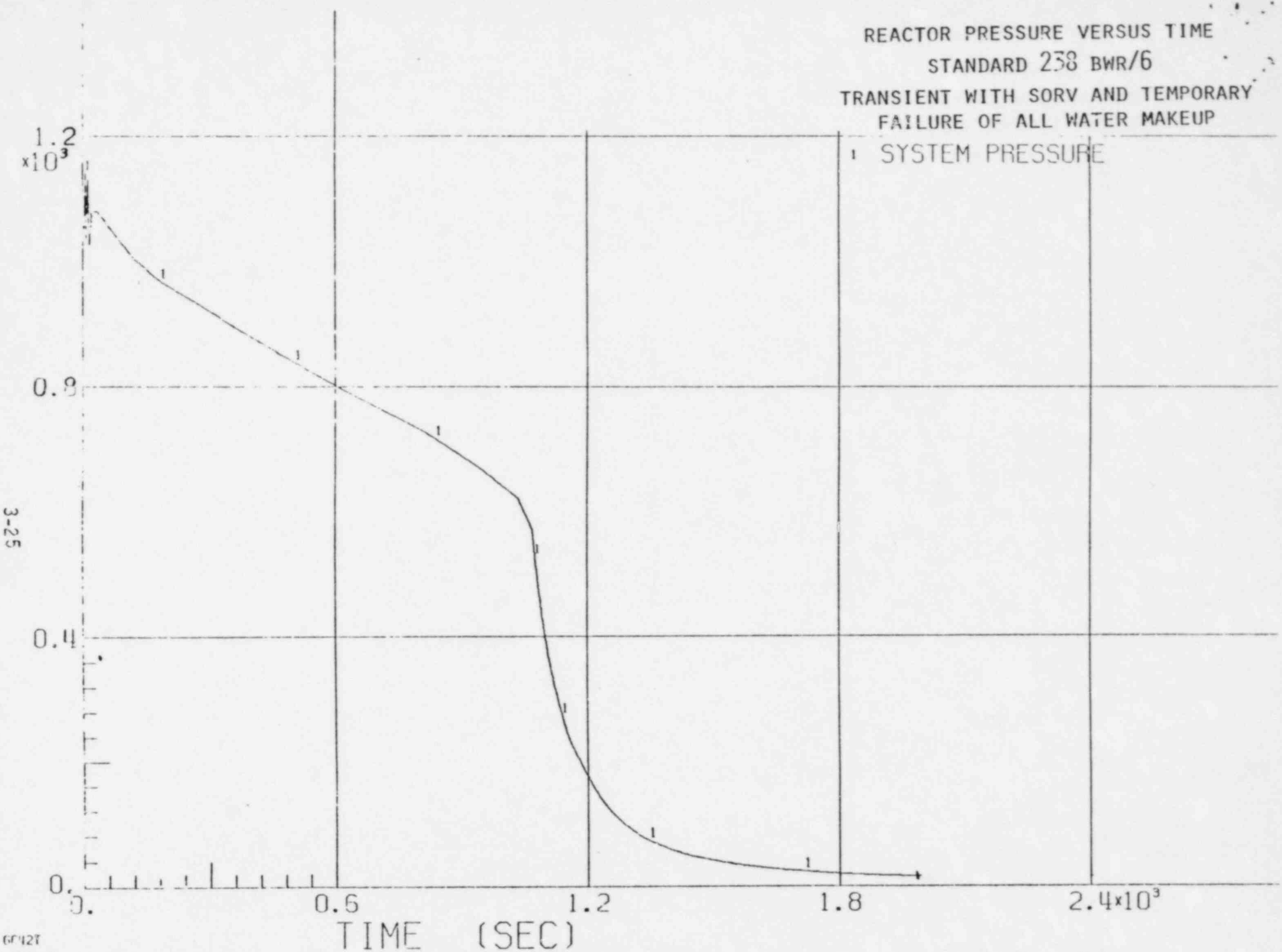

Figure 3.4.1-1 Reactor Pressure for Transient with SORV and Temporary Failure of all Water Makeup.

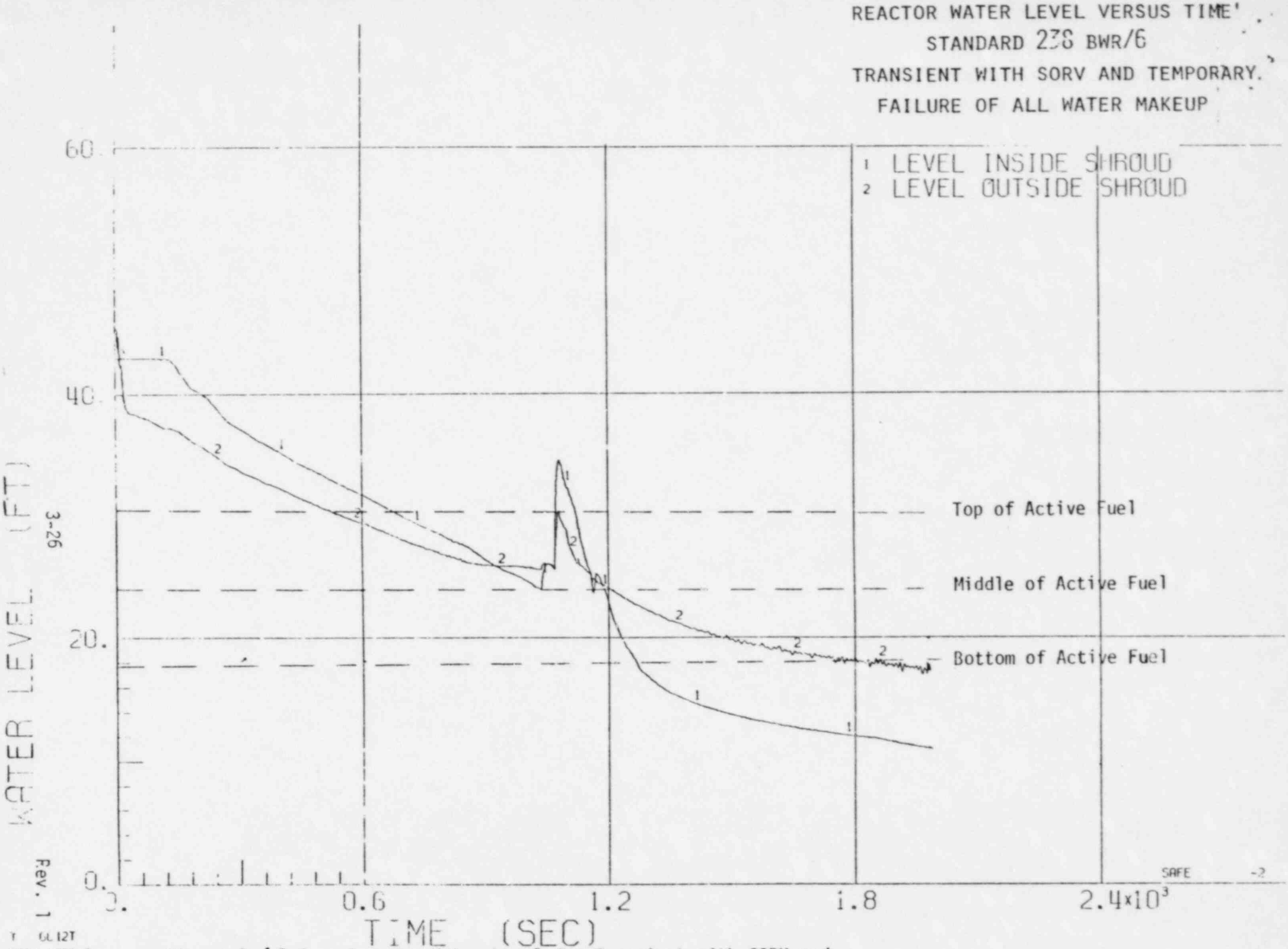

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Figure 3.4.1-2 Reactor Water Level for Transient with SORV and Temporary Failure of all Water Makeup.

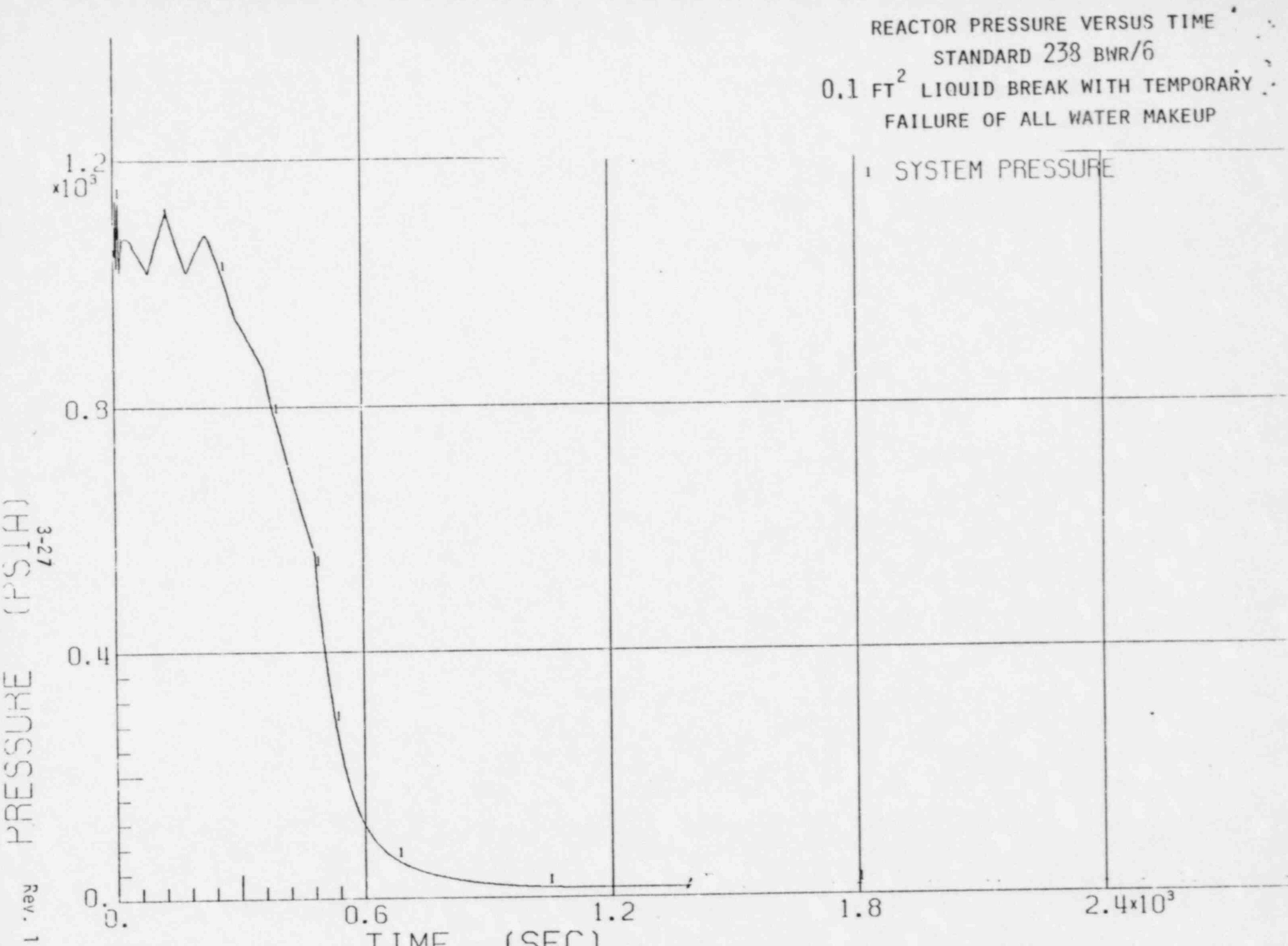

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Figure 3.4.2-1 Reactor Pressure Transient for 0.1 Ft^2 Liquid Break with Temporary Failure of all Water Makeup.

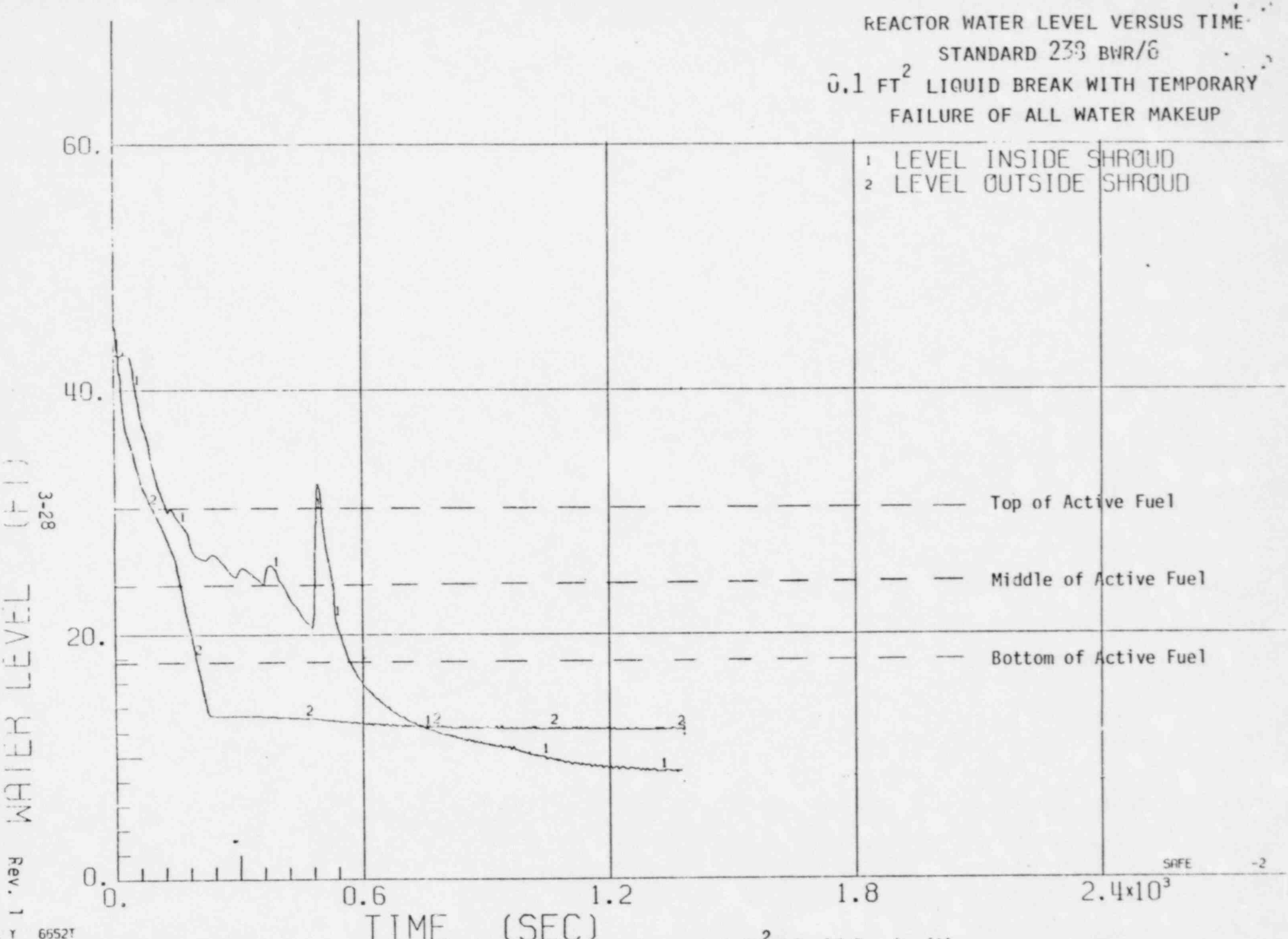

Figure 3.4.2-2 Reactor Water Level Transient for 0.1 Ft^2 Liquid Break with Temporary Failure of All Water Makeup.

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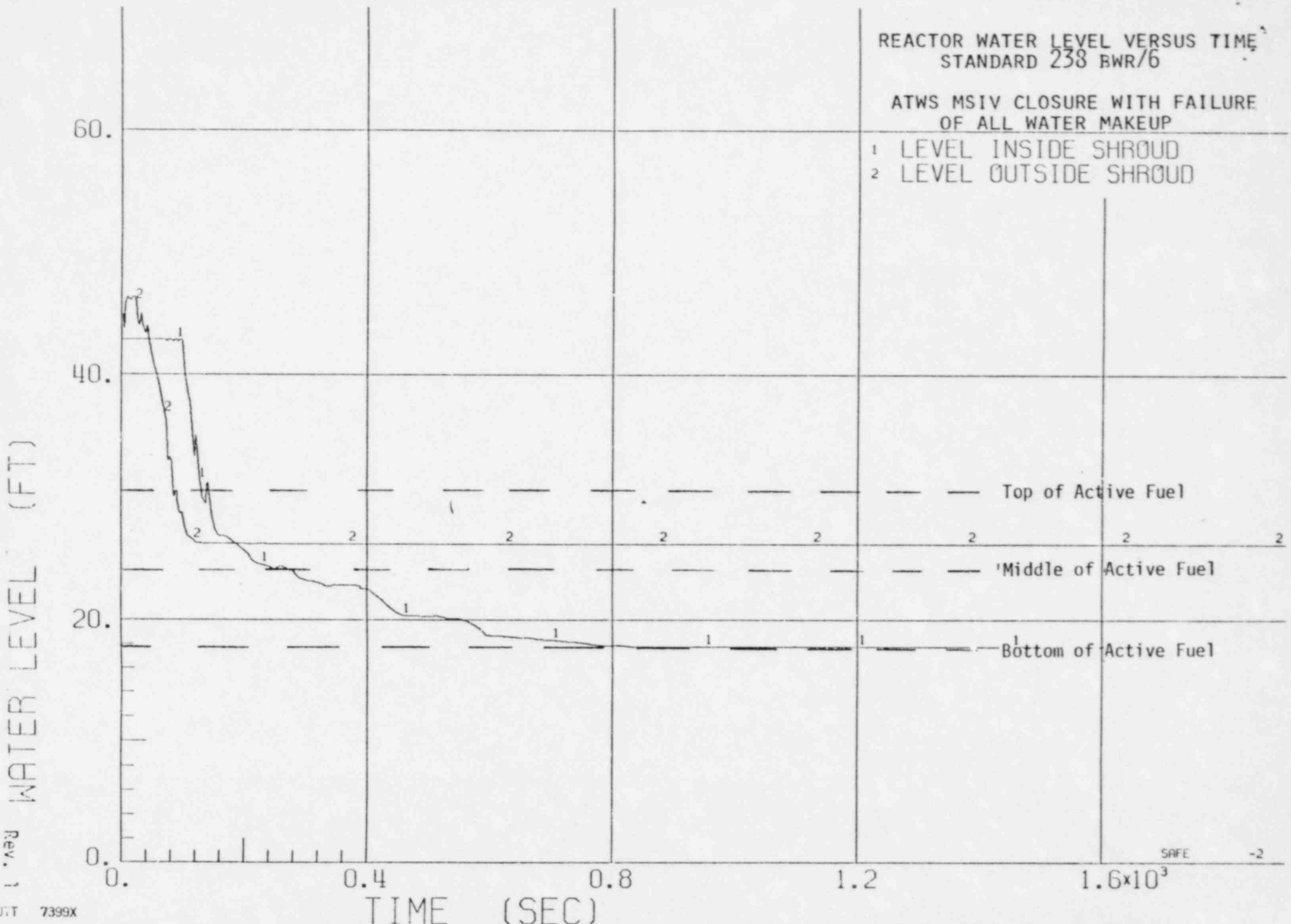

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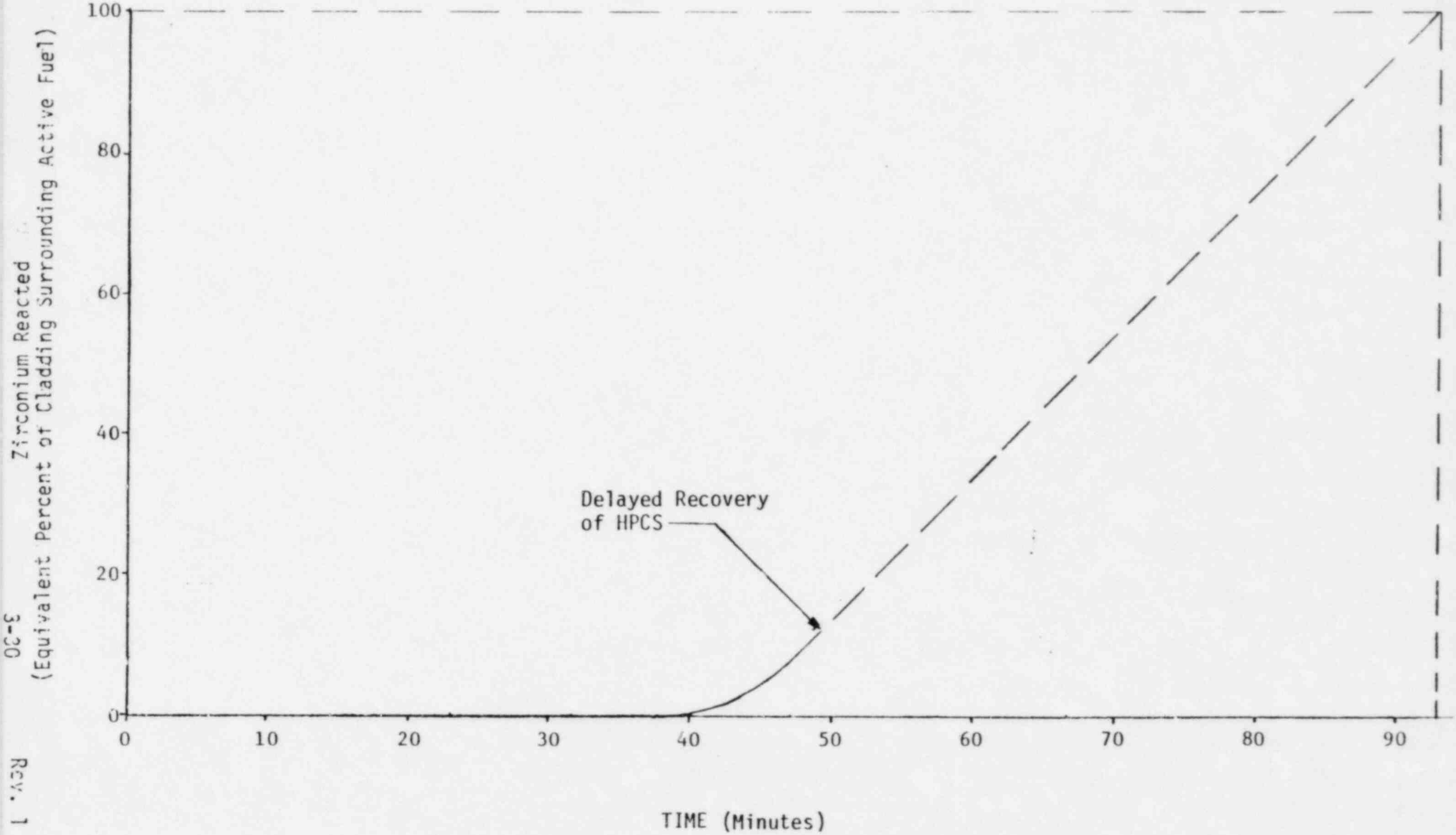
Firure 3.4.3-1 Reactor Water Level Transient for ATWS MSIV Closure with Failure of all Water Makeup.

Figure 3.4.4-1

Non-Mechanistic Extrapolation of Percent of Zirconium Reacted vs. Time for Isolation with Temporary Failure of All Water Makeup.



4. VITAL EQUIPMENT LIST

Using a probabilistic risk assessment approach, one finds that a hydrogen control system can reduce the probability of loss of containment integrity due to the consequences of uncontrolled hydrogen combustion. However, on an overall plant risk basis this is a small consideration as the hydrogen control system does not eliminate other modes of loss of containment integrity. Conditions which lead to these other modes of loss of containment integrity are as likely as the conditions that lead to significant hydrogen generation, i.e., hydrogen generation and core-melt are only likely to occur simultaneously. In order to prevent release of fission products (as a consequence of core-melt) to the atmosphere, the vital equipment that one would require to function in conjunction with a hydrogen control system is that which keeps the containment isolated. This criterion is selected for the purposes of the report because the NRC has specified the requirement (in the proposed rule) to maintain containment integrity. However, it is General Electric Company's view that the containment function is the key feature to be retained. The BWR suppression pool provides an effective filter for fission products and provides a means of maintaining the containment function even if the containment integrity were to be lost. Thus, any equipment that is required to keep the fission product release pathway through the suppression pool intact (e.g., S/RV discharge lines) is considered vital.

For the sake of completeness, the vital equipment needed to maintain contanment integrity following the postulated HGE is also identified in this report. (Any equipment which may be necessary as part of the hydrogen control system design was not considered.)

The systems needed to maintain containment integrity following the HGE are those systems which are necessary to maintain core cooling and to remove decay heat from the containment. Those plant systems are:

a. High Pressure Core Spray (HPCS) System (plus the Emergency Power System for HPCS)

b. Safety/Relief Valve (S/RV) System (Safety-Mode Only)

c. Decay Heat Removal (Residual Heat Removal (RHR) System in the Suppression Pool Cooling Mode)

d. Suppression Pool Makeup System

The HPCS was specified as the most suitable water makeup system in preference over the numerous other systems because of the ability of HPCS to deliver a substantial flow of water at pressures ranging from atmospheric to the upper S/RV pressure setpoint. The S/RV system provides for transport of decay power in the form of steam from the RPV to the suppression pool and ensures that the pressure in the RPV is maintained within the operating limits of the HPCS. The suppression pool cooling mode of the decay heat removal system is sufficient to remove decay power from the containment as all decay power is transported from the RPV to the suppression pool in the form of steam, as long as the core is maintained covered with water. Although loss of water inventory from the suppression pool is expected to be insignificant, the suppression pool makeup system is specified as vital to assure that the proper water level will be maintained in the suppression pool.

The vital equipment list given in Table 4-1 itemizes all the equipment located inside the containment which must function in order to provide core cooling and decay heat removal. To this list must be added any other equipment or structures necessary to assure containment integrity or containment function (e.g., drywell integrity), and to assure proper functioning of the specific hydrogen control system design.

TABLE 4-1

EQUIPMENT NEEDED TO MAINTAIN CONTAINMENT INTEGRITY
(BY PROVIDING CORE COOLING AND DECAY HEAT REMOVAL)

SYSTEM	MPL NO.	COMPONENT	PLANT LOCATION
High Pressure Core Spray System	E21-F006	Valve	DW
	E21-F007	Valve	DW
	E22-F036	Valve	DW
	E22-F005	Valve	DW
High Pressure Core Spray Emergency Power System	-	-	Outside
Safety/Relief Valves	B21-F047	Valve	DW
	B21-F041	Valve	DW
	B21-F051	Valve	DW
Decay Heat Removal (RHR in the Pool Cooling Mode) System	E12-F042	Valve	CT
	E12-F041	Valve	DW
	E12-F009	Valve	DW
	E12-F010	Valve	DW
	E12-F028	Valve	CT
	E12-F037	Valve	CT
	E12-D003	Transmitter	CT
Suppression Pool Makeup System	-	-	Outside

NOTES: DW: Drywell

CT: Containment

Outside: Located outside of Drywell and Containment

5. SUMMARY AND CONCLUSION

This study relies on the methodology for performing Probabilistic Risk Assessments (PRAs) to conclude that:

a. Constraining the hydrogen generation event (as a basis for design of a hydrogen control system), to be mechanistically based and to result in no core-melt, leads to the choice of an event which also results in no hydrogen generation. An alternate event, which approximates the major risk producing events (that also have hydrogen generation) results in significant hydrogen generation combined with core-melt. For this event, the hydrogen generation is limited by the delayed availability of makeup water, as a result of operator action to start water injection. The maximum zirconium reacted in this case, is limited to the equivalent of 12.5% of the zirconium cladding surrounding the active fuel.

b. As fission product release to the surrounding environment is the major determinant of plant risk to the public, the only vital equipment that is required to function in conjunction with a hydrogen control system is that equipment which maintains the containment function. The BWR suppression pool provides an effective filter for fission products and provides a means of maintaining the containment function, even if the containment integrity should be lost. Thus, the only equipment which is vital is that needed to keep the fission product pathway through the suppression pool intact.

Furthermore, this study concludes that (based on References 10 and 11):

c. Hydrogen control systems only reduce the probability of loss of containment integrity due to the consequences of uncontrolled hydrogen burning combustion.

d. Elimination of loss of containment integrity due to uncontrolled hydrogen combustion reduces the overall plant risk insignificantly, because the BWR suppression pool provides an effective filter for fission products.

6. REFERENCES

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4. Bruce C. Slifer, "Loss of Coolant Accident and Engineering Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, April 1971.

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9. "Emergency Procedure Guidelines - BWR 1-6," Revision 1, NEDO-24934, January 1981.

10. A. A. Strod, et.al., "Preliminary BWR/6 Mark III Probabilistic Risk Assessment," ANS Topical Meeting on Probabilistic Risk Assessment, Portchester, New York, September 21-23, 1981.

11. D. A. Hankins, et.al., "Effects of BWR Suppression Pool Scrubbing on Degraded Core Accident Sequences", American Nuclear Society Transactions, Volume 39, page 7

APPENDIX A - MODEL DESCRIPTIONS

This Appendix provides a detailed description of the models used to determine the hydrogen generation rates.

A.1 Description of the SAFE Code (the Vessel Blowdown Analysis)

The SAFE computer program is a vessel blowdown model for the analysis of loss of inventory events. The program is capable of analyzing the complete spectrum of loss of inventory transients including break sizes at different locations with various combinations of water makeup systems. The reactor vessel is modeled using a single spatially uniform pressure and four liquid regions. Heat addition from the core, vessel, and vessel internals is included. Blowdown flow is calculated using Moody's critical flow model (Reference 3). Cooling systems that can be included in the analysis are Core Sprays, Low Pressure Coolant Injection (LPCI), Reactor Core Isolation Cooling (RCIC) System, and Control Rod Drive (CRD) Flow.

Because the SAFE code is intended to predict long term transients, local effects are generally ignored. During the first few seconds after the initiation of a transient, local effects predominate. However, after an initial period of rapid change, pressure traces from SAFE agree very well with those from codes which allow spatial pressure variation based on more detailed models of the internal structure (Reference 1). The core is modeled as a series of one-dimensional heat transfer nodes. The simple core model represents a good approximation of the decay heat delivered to the coolant during the transient.

In the model, the fluid within the reactor vessel is divided into five regions and a system node. There are four nodes containing liquid or liquid-vapor mixture, two inside the shroud and two outside as shown in Figure A-1. A steam dome of saturated vapor fills the remainder of the vessel (References 4 and 5). Within each region the fluid is homogeneous. The system node includes the steam dome and all saturated regions. Before every transient, the reactor is initialized to a steady state. All of the regions are free to expand and contract, based on integrated mass and energy balances. The mass balances allow level tracking. Valves and auxiliary systems can be tripped based on reactor vessel pressure and water levels.

There are two sources of heat, the stored and decay heat from the fuel, and the stored heat from the metal within the vessel. The heat source from the metal of the vessel and the vessel internals is lumped into one-dimensional radial heat transfer nodes. The heat transfer coefficient associated with a node is assumed to be a function of the void fraction of the fluid in contact with it.

A.2 Description of the Methodology for Calculation of Steam Cooling

During these transients steam is generated in the reactor due to reactor depressurization and due to heat transfer to the two-phase mixture in the reactor core. The single phase convective steam cooling mechanism is especially important in the analysis of degraded conditions involving multiple failures of water makeup systems.

To determine the fuel temperature response and its dependence on steam cooling, core heatup calculations were performed using a simple steam cooling heat transfer model. The heat transfer coefficient (HTC) from the fuel rods to steam, needed to quantify steam cooling, is calculated using the Dittus-Boelter correlation. The Dittus-Boelter correlation is applicable for fully developed turbulent flow heat transfer in a tube with constant heat flux as boundary conditions. For laminar flow, a Nusselt number of 4.36 is used as it defines the lower bound of the heat transfer coefficient.

The methodology for the steam cooling calculation has a simple core heatup model from which the approximate cladding temperature is determined. Application of this approximate cladding temperature is limited to the evaluation of the steam cooling HTC only.

The heat transfer coefficients (HTC) obtained from the steam cooling calculation are then used in the core heatup model (CHASTE code) to determine the final cladding temperatures. Since the determination of HTC requires the knowledge of the fuel cladding temperature which in turn depends on the value of HTC used in the CHASTE code, an iterative process is used to obtain the final fuel cladding temperatures and steam HTCs consistent with one another. The fuel cladding temperatures from the CHASTE analysis using the steam HTCs from the first iteration of the steam cooling calculation, and those obtained using steam HTC's from the second iteration of the steam cooling calculation, show very little difference. The cladding temperature calculations are within 2% of each other. This demonstrates the validity of using an independent steam cooling analysis to calculate the steam cooling HTCs consistent with the calculation of fuel cladding temperatures using the core heatup model (CHASTE code). To evaluate the steam cooling heat transfer coefficient, the steam flow area per fuel bundle and the hydraulic diameter of the fuel bundle are determined from the fuel bundle geometry. The core pressure and the steam mass flow rate are obtained from the SAFE analysis.

A.3 Description of the CHASTE Code (the Core Heatup Analysis)

This section gives the general description of the core heatup model (CHASTE code). The specific applicability of this model to analyze the postulated highly degraded core cooling conditions is discussed in Sections 3.2.1 and 3.2.2.

The CHASTE digital computer program is a multi-rod core model whose primary purpose is to analytically determine the transient response of the reactor core to a loss of inventory event. In particular, the core temperatures and the extent of metal-water reaction are calculated. The following phenomena are considered:

1. Inter-rod and rod-to-channel thermal radiation;
2. Metal-water reaction;
3. Temperature dependence of material properties;
4. Heat conduction within the fuel;
5. Decay heat.

The core of a boiling water reactor is composed of several separate fuel rod bundles, each of which has an enclosing channel wall and several fuel rods. The specific power within a bundle varies axially within a rod and locally from rod to rod at a given elevation. The axial power distribution is primarily due to void distributions existing in the coolant at normal operating conditions. The local power distribution is due to fuel enrichment variations in the fuel rods. The core heatup model treats a single axial plane of a single bundle. The bundle is nodalized radially involving the grouping of individual fuel rods according to position and power peaking. All rods within a rod group have the same local power peaking factor and the same radiation grey body factors to other rod groups. Therefore, all rods within a group have the same temperature response to the transient and are considered as a single body. The nodalization within a fuel pellet divides the fuel into a maximum of eight equal volume radial nodes. The node center of each fuel node is at the radius which divides the fuel node into two equal volumes.

The transient thermal response of a core to a loss of inventory event can generally be broken down into three stages: (1) fuel pin temperature redistribution; (b) fuel rod bundle temperature redistribution; (c) metal-water reaction heatup. Phenomena occurring during these stages that are considered in the analysis are described below:

(a) Fuel Pin Temperature Redistribution

Following a reactor shutdown, a large heat source is still available within the core in the form of sensible heat in the fuel as represented by the temperature profile in the fuel rod. Following shutdown, the sensible heat in the fuel will be distributed by thermal conduction within the fuel and cladding and by convection and radiation in the gap between fuel and cladding, with the amount of heat removed being dependent on surface conditions. At the end of three or more fuel time constants (a fuel time constant is about ten seconds), the radial temperature profile in the fuel pin is almost flat, consistent with the low fission product decay power generation.

(b) Fuel Rod Bundle Temperature Redistribution

As the cladding temperature increases and the core coolant void fraction approaches unity, radiant heat transmission between rods and the channel wall tends to equalize the temperatures of all rods at a given axial position. The average core temperature however, continues to increase if the heat removed from the core is less than the decay heat generation.

(c) Metal-Water Reaction Heatup

The fuel pin cladding is made of an alloy of Zirconium, which reacts with steam at high temperatures. The Zircaloy-steam chemical reaction rate is exothermic and strongly dependent upon the reaction temperature. The temperature dependence is exponential and the rate of reaction becomes significant at cladding temperatures in the range of 2200°F. The standard Baker-Just equation, with the constants modified to those reported by Electric Power Research Institute (EPRI), is used to calculate the rate of metal-water reaction (Reference 6).

A.4 Approach to Extrapolate Current LOCA Models

Sections A.1 to A.3 describe the analytical models used to evaluate the core heatup and associated metal-water reaction (MWR) prior to the meltdown of the fuel cladding. During this phase of a highly degraded core cooling event, standard GE LOCA models are adequate to determine the core behavior. When the core is heated above the zirconium melting temperature (3366°F), the core geometry may no longer be maintained in its original configuration. For cladding temperature above the zircaloy melting temperature, an approach to extrapolate current LOCA models is needed to extend the analysis to cover the meltdown phase during a highly degraded core cooling event. Since the knowledge of the behavior of core meltdown and core slumping is not well defined, some assumptions have to be made to model the core melt patterns and the slumping of molten core into the lower plenum. The extrapolation approach along with the necessary assumptions made to model the core melt phenomenon is described as follows:

1. Evaluation of Corewide MWR Fraction

In a BWR core design, the fuel rod cladding and the bundle channel walls are made of Zircaloy, a zirconium alloy. Since the wall thicknesses and heatup rates are different in the cladding and the channel wall regions, the MWR fraction for each region is also expected to be different. At any given time, the total MWR of each region is added to achieve an equivalent corewide MWR fraction based on the entire mass of the zirconium surrounding the active fuel.

2. MWR During Quenching of Slumped Core Increments

The two major assumptions made to calculate the MWR of the slumped part of the core are as follows:

(a) When the cladding in one increment or node reaches the melt temperature of 3366°F, the entire node is assumed to slump and fall into the water in the lower plenum of the reactor pressure vessel (RPV). In the lower plenum, the molten cladding and the UO_2 pellets are quenched and produce more steam and hydrogen during the quenching phase.

(b) It is assumed that the molten material fragments into small particles upon contact with water in the lower plenum. For the purpose of calculating the MWR during quenching, a particle of ~3000 μm in diameter falls into a pool of water and is quenched. Consistent with experimental results (References 7 and 8) the corewide MWR fraction includes an additional 10% reaction of unreacted zirconium in every node that slumps and quenches in the lower plenum. Since the molten core, upon contact with water, fragments into small particles, the effective surface area for heat transfer from the molten core to the water in the lower plenum is large. It is therefore reasonable to assume that quenching of the slumping node occurs instantaneously.

3. Liquid Inventory Depletion During Quenching of Slumped Core Increments

The computer code SAFE is used to calculate RPV liquid inventory during the blowdown phase and up to the time when the first core increment (node) slumps. Results of the SAFE analysis indicate that by the time the first core node slumps, the RPV has already been depressurized to almost atmospheric pressure (the operator is assumed to follow the Emergency Procedure Guidelines (Reference 9) for highly degraded core cooling conditions). The steam generation during the quenching phase is primarily due to heat transfer from the molten corium to water in the lower plenum. The following energy source terms are the inputs used in calculating steam generation during the quenching phase of a slumped core increment (node):

(a) Sensible energy of UO_2 in the slumped node;
(b) Sensible and latent energy of prequench mass of zircaloy in the slumped node;
(c) Sensible energy of prequench ZrO_2 mass in the slumped node;
(d) Heat of reaction from MWR of the slumped node during quenching phase.

During the periods between the slumping of two consecutive core increments, steam generation is calculated based on the decay power of all the core increments already slumped. The MWR and the generation of hydrogen terminate when all the water inventory in the lower plenum is depleted in the process of quenching the slumped part of the core.

A.5 Approach to Evaluate the Recovery Phase

Sections A.1 to A.4 describe the methodology for evaluating the metal-water reaction (MWR) and the associated hydrogen generation during the core heatup, meltdown, and slumping periods of a postulated highly degraded core cooling event. Those calculations, however, do not account for the recovery phase, should a makeup system become available during the transient. This section describes the method for analyzing the recovery phase when a water makeup system becomes available sometime later into the transient and terminates the event with no further metal-water reaction. In such an event, the lower plenum region of the reactor pressure vessel (RPV) is refilled and the core region is reflooded with water. During the core reflood period, the core is quenched by water with some additional metal-water reaction on both the cladding and channel surfaces during the quenching phase. The necessary technical details to model the recovery phase are described below. The model description is divided into three sections, 1) calculations of vessel inventory, 2) calculation of quenching of each node in the core, and 3) metal-water reaction.

1. Calculation of Vessel Inventory

A mass and energy balance to calculate the water inventory inside the RPV at any given time is performed by adding the amount of makeup water inflow to the initial RPV water inventory, and subtracting the amount of RPV water boiloff. Boiloff of the RPV water is due to:

(1) decay power from the submerged nodes,
(2) stored energy of the nodes undergoing quenching.

The RPV water inventory at the initiation time of the recovery phase is obtained from the previous analysis (Section A.1 to A.4, also Section 3.2.1). The RPV pressure used for the recovery phase calculation is conservatively approximated as about 400 psia. At ~400 psia, the flow rate of the delayed water makeup system can be specified and the time needed for the water to refill the lower plenum region of the RPV to the bottom of the active fuel (BAF) can be estimated.

2. Calculation of Node Quenching

A mass and energy balance is written for the fuel node which includes the fuel pellets, the cladding, and the channel as a lumped mass. The energy source and sink terms for the analysis of energy balance at a node include:

(1) the decay heat
(2) the metal-water reaction heat,
(3) the heat transfer of the stored energy from the node during the quenching phase for submerged nodes,
(4) the heat transfer due to steam cooling (Section A.2) for the uncovered nodes.

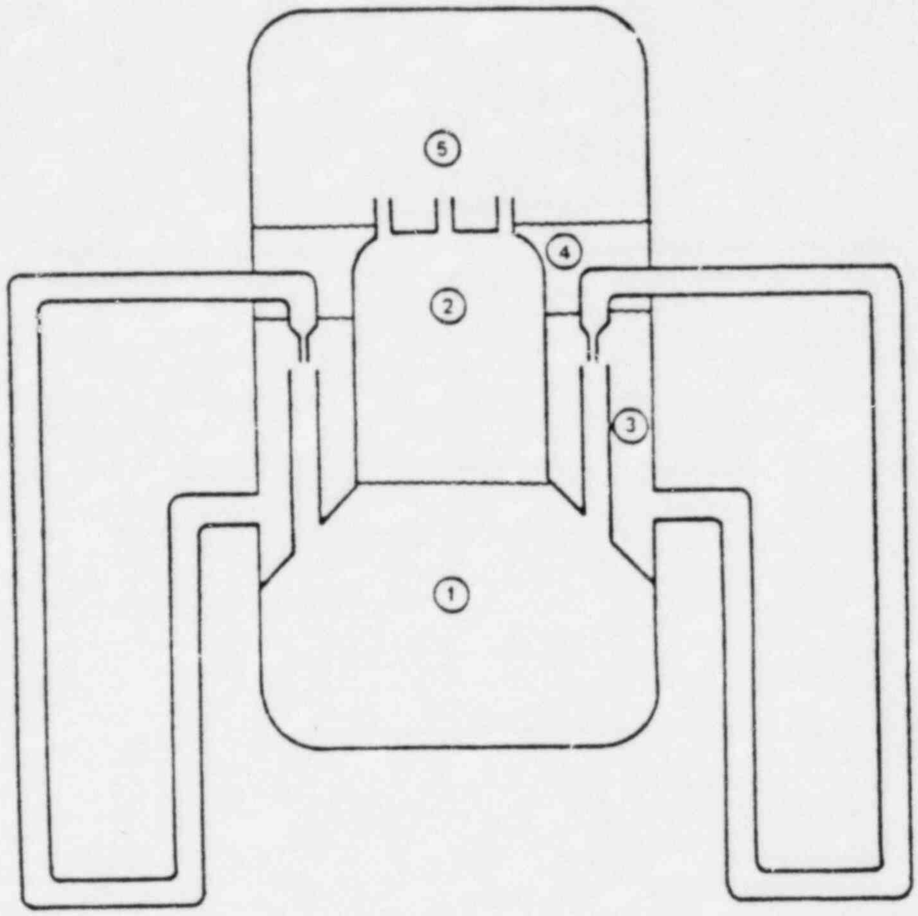
Prior to the core recovery, depending on the axial and radial location of the nodes, the temperature profile for the intact portion of the core ranges from several hundred degrees up to temperatures close to the zirconium melting temperature (3366°F). A large amount of stored energy is contained in the intact core. During the core reflooding and quenching period, this stored energy together with the decay power of the core and the metal-water reaction heat, is released to the water and evaporation of water takes place. A film boiling heat transfer coefficient of 60 BTU/hr-ft^{2}F was assumed during this phase of the transient. A large amount of vapor bubbles will be generated and the void fraction in the core region for the recovery phase calculation can be conservatively estimated to be 0.5. Calculation of node quenching is terminated when the temperature of a node is the same as the saturation temperature of the surrounding water.

3. Calculation of Metal-Water Reaction During Recovery Phase

The temperature profile of the nodes in the core, the initial zirconium oxide layer thicknesses on the cladding and the channel surfaces, and the percentage of the core slumped at the time when the RPV water level is at the BAF are obtained from the previous calculations (Sections A.3 to A.4, also Section 3.2.1). These values are used to specify the initial conditions for the recovery phase core behavior and the MWR calculation.

The calculation of metal-water reaction for the intact portion of the core during the recovery phase is consistent with the method described in Section A.3. The MWR accounts for reaction on both the cladding and the channel surfaces. After a node has reached its melting temperature, the additional MWR during quenching, as described in Section A.4, is added to the node and the MWR calculation for that node terminates.

During these calculations, the MWRs and the associated hydrogen generations on the cladding and the channel surfaces for all nodes in the core, are added to reflect a global MWR behavior of the entire core (Section A.3).



INITIAL CONDITIONS OF REGIONS

1. SUBCOOLED LIQUID INSIDE SHROUD
2. SATURATED MIXTURE INSIDE SHROUD
3. SUBCOOLED LIQUID OUTSIDE SHROUD
4. SATURATED LIQUID OUTSIDE SHROUD
5. SATURATED VAPOR IN STEAM DOME

Figure A-10 Fluid Regions Specified in the SAFE Code