NUREG/CR–4537
SAND86–0451
RV
Printed September 1986

Summary Report: Electrical Equipment Performance Under Severe Accident Conditions (BWR/Mark I Plant Analysis)

P. R. Bennett, A. M. Kolaczkowski, G. T. Medford

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185 and Livermore, California 94550
for the United States Department of Energy
under Contract DE-AC04-76DP00789

Prepared for
U. S. NUCLEAR REGULATORY COMMISSION

Available from
GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

and

National Technical Information Service
Springfield, Virginia 22161

NUREG/CR-4537
SAND86-0451
RV

SUMMARY REPORT:

ELECTRICAL EQUIPMENT PERFORMANCE UNDER SEVERE ACCIDENT CONDITIONS (BWR/MARK I PLANT ANALYSIS)

P. R. Bennett
Sandia National Laboratories

A. M. Kolaczkowski
G. T. Medford
Science Applications International Corporation

September 1986

Sandia National Laboratories
Albuquerque, NM 87185
Operated by
Sandia Corporation
for the
U.S. Department of Energy

Prepared for
Division of Engineering Technology
Electrical Engineering and Instrumentation Control Branch
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
Under Memorandum of Understanding DOE 40-550-75
NRC FIN No. A1382

ABSTRACT

The purpose of the Performance Evaluation of Electrical Equipment during Severe Accident States Program is to determine the performance of electrical equipment, important to safety, under severe accident conditions. In FY85, a method was devised to identify important electrical equipment and the severe accident environments in which the equipment was likely to fail. This method was used to evaluate the equipment and severe accident environments for Browns Ferry Unit 1, a BWR/Mark I. Following this work, a test plan was written in FY86 to experimentally determine the performance of one selected component to two severe accident environments.

Specifically, equipment important to safety for a BWR was identified--equipment which could mitigate a severe accident or provide monitoring information on plant status. Of this list of equipment, only that located in the primary containment or reactor vessel of Browns Ferry Unit 1 was analyzed further. For five selected BWR severe accident sequences (TB, TC, TW, TQUV, and AE), environmental conditions within containment reached temperatures and pressures exceeding the current equipment qualification testing requirements prior to or during the time the equipment was needed. The results of this analysis suggest the need for testing equipment important to safety to assess performance under severe accident conditions. In particular, the performance of the pneu- matic control manifold assembly (part of the main steam isolation valve equipment assembly) should be tested in the severe accident environments resulting from the TC and TW accident sequences.

In addition to writing a test plan for the pneumatic control manifold assembly, a number of important insights are discussed in the areas of accident management, emergency planning, probabilistic risk assessments, probability and risk reduction, and current equipment qualification requirements. These insights help illustrate how the environmentally-induced failure of certain equipment during a severe accident may adversely impact the ability of a nuclear power plant to cope with severe-accident conditions. However, without testing to confirm the actual limits of equipment survivability, the safety importance of the insights cannot be assessed or addressed.

CONTENTS

CONTENTS (Continued)

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

The purpose of the Performance Evaluation of Electrical Equipment during Severe Accident States (PEEESAS) Program is to determine the performance of important electrical equipment under severe accident conditions. Important electrical equipment is defined as electrical equipment that is important to safety. This includes equipment used to mitigate an accident or provide information on the status of the plant. Specifically, this program will

1. Devise a method to identify important electrical components and the severe accident environments in which they are likely to fail,

2. Use the method to analyze equipment performance for nuclear power plants,

3. Test the performance of selected components to the severe accident environments to determine performance, and

4. Provide the results of equipment performance to operators, emergency planning teams, probabilistic risk assessment analysts, and the Nuclear Regulatory Commission to influence actions and decisions.

During FY85 and FY86, a method was devised to answer questions on the performance of electrical equipment under severe accident conditions. This method provided the means to identify important electrical components and the severe accident environments in which they are likely to fail. Also during FY85, Browns Ferry Unit 1 (BWR/Mark I) was chosen to be the first nuclear power plant analyzed. For this plant, the following areas were investigated: (1) accident sequences (including operator actions that are likely to occur during those sequences), (2) important electrical equipment in the primary containment, (3) environmental profiles, (4) important electrical equipment that will be subjected to environments beyond their current qualification levels, and (5) test plan for the selected equipment.

Accident Sequences and Likely Scenarios

The five accident sequences chosen for this study are as follows: TB (station blackout including loss of all AC power), TC (anticipated transient without scram), TW (transient with loss of long-term heat removal), TQUV (transient with early loss of core cooling), and AE (large-break LOCAs with early loss of core cooling). The selection of BWR/Mark I accident sequences was based on the following criteria: (1) sequences with high probability, (2) sequences with high risk, (3) sequences with the potential for extreme environments, and (4) sequences with operator action required.

Likely scenarios are series of events that are most likely to happen during an accident sequence based on operator actions, timing of system failures, and automatic system actuation. These likely scenarios were used to identify (1) failed equipment by accident sequence definition, (2) equipment assessed to succeed and additional equipment needed to mitigate or provide plant status, and (3) boundary conditions for determining the environmental profile for each accident sequence. Fourteen likely scenarios resulted from the five selected accident sequences. The likely scenarios included the following variations: operator depressurization of the reactor pressure vessel, no operator action, and stuck-open relief valve.

Equipment

From an initial list of BWR equipment important to safety and from a review of the Browns Ferry Unit 1 design, equipment was identified that was important to safety and was located in the primary containment or reactor vessel. (This equipment is generally in the most severe environment.) From qualitative arguments considering the possible importance of the equipment in mitigating or assessing the status of the plant for each selected accident sequence, the following equipment was identified: inboard main steam isolation valves (MSIV); inboard high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) isolation valves; safety relief valve (SRV) pilot and service air solenoid valves; residual heat removal (RHR) shutdown cooling valve; in-core and reactor vessel surface thermocouples; drywell temperature element (RTD); drywell pressure monitor; and drywell hydrogen and radiation monitors.

Environmental Profiles

Environmental profiles of the primary containment were developed for each selected accident sequence. The following parameters were considered: humidity, submersion, spray, radiation/aerosols, vibration, pressure, and temperature. Pressures and temperatures were determined, for each likely scenario, from MARCH and LTAS computer codes. These severe accident environmental profiles were compared to typical equipment qualification profiles to identify areas where the severe accident environmental profile exceeded the equipment qualification profile of IEEE 323-1974, Appendix A.

Reduction of Equipment and Environments

The list of equipment and environments was reduced in two steps. Step 1 identified the time that equipment was demanded and whether the severe accident environments exceeded the equipment qualification levels prior to or during that time. Step 2 determined the relative functional importance of the equipment. In Step 1, equipment was eliminated if the severe accident environments were below that of the typical qualification environment. Then the severe accident environments were further reduced by retaining only those profiles with (1) maximum pressure or temperature or (2) maximum time above the maximum pressure or temperature for the typical equipment qualification profile. (These results represent profiles where the equipment must operate under the most severe conditions for the five selected accident sequences.) In Step 2, the relative functional importance of the equipment was based on the following criteria: redundancy, backup systems, non-complexity, electrical independence, fail-safe position appropriate, plant status indication only, and separation. The equipment is less functionally important if the equipment meets these criteria.

The equipment remaining after Steps 1 and 2 is the main steam isolation valves (MSIV) and the safety relief valves (SRV). This equipment was required to operate during the TC and TW accident sequences.

To choose the first test candidate, the effect of an environmentally-induced failure of the MSIV or SRV (for the TC and TW accident sequences) on probabilistic risk assessments was determined. From a PRA perspective, both the MSIV and the SRV are good test candidates. But for the first test candidate, the MSIV equipment assembly was chosen because (1) failure of the MSIV may increase the core melt probability as well as increase the risk and (2) the performance of the MSIV may be tested in more than one accident environment. Several pieces of equipment are associated with the MSIV equipment assembly. The pneumatic control manifold assembly was chosen to be the first test candidate because it is required to operate the MSIV globe valve, it provides a large heat rejection path, and it is a complex electrical component.

Testing

The performance of the pneumatic manifold assemblies will be evaluated for both the TC (with the MSIV initially open) and the TW accident sequences. (Two manifold assemblies will be tested--one for each accident profile.) Moisture intrusion, due to a combination of moisture and high temperature or pressure, is the dominant failure mechanism for the manifold assembly.

The manifold assembly must perform its required safety function throughout the accident exposure and the acceptance criteria is based on this operational performance.

Both manifold assemblies will be exposed to simultaneous radiation and thermal aging with the solenoids energized. Then each manifold assembly will be exposed to an accident profile. In Test #1, the TC (MSIV open) accident sequence profile will be followed until containment failure at 4.5 hours. If the valve remains open, the chamber pressure and temperature will be increased to determine the fragility level of the manifold assembly. The valve must be energized throughout Test #1. At the conclusion of the test, the valve will be closed (if necessary). The valve must close and remain closed, at that time, to perform its required safety function. In Test #2, the TW accident sequence profile will be followed until containment failure at 35 hours. During this time, the valve will be cycled every 2 hours. If the valve can still be cycled from the closed position to the open position at containment failure, the chamber pressure and temperature will be increased to determine the fragility level of the test specimen. The valve will be cycled open at each fragility plateau. At the conclusion of the test, the valve will be closed (if necessary). The valve must close and remain closed, at that time, in order to perform its required safety function.

However for the TC and TW accident sequences, the MSIV will only be required to open prior to containment failure. Since core melt occurs after containment failure, the manifold assemblies need not be exposed to severe accident radiation. In addition, the containment spray system is not used in the TC and TW sequences; therefore, the manifold assemblies will not be exposed to spray.

Conclusions and Insights

As described above, the primary purpose of this project during FY85 and FY86 was to develop a test plan to evaluate the performance of electrical equipment in severe accident environments. This involved the selection of accident sequences, identification of important electrical equipment, determination of environmental profiles, selection of a test candidate, and development of a test plan for the MSIV manifold assembly in the TC (MSIV open) and TW accident sequences. In addition to these tasks for the test plan, insights from the analysis portion of this study were identified to illustrate how the environmentally-induced failure of certain equipment during a severe accident may adversely impact the ability of a nuclear power plant to cope with severe accident conditions. These insights involve: accident management, emergency planning, probabilistic risk assessments, probability and risk reduction, and current equipment qualification requirements. Without testing, however, to confirm the actual limits of equipment survivability, the safety importance of the insights cannot be assessed or addressed.

In summary, the insights are:

1. Potential environmentally-induced failures of electrical equipment, after equipment qualification limits are exceeded, may render the current Emergency Procedure Guidelines and operator training ineffective.

2. Accident management and emergency planning procedures may need to reflect the effects on equipment operability of those accident conditions which are expected to exceed equipment qualification limits.

3. Probabilistic risk assessments may not adequately address the effects of environmentally-induced equipment failures.

4. Depending on the results of equipment testing under severe accident conditions, current equipment qualification requirements may need to be reviewed for adequacy.

1. INTRODUCTION

This report is a summary report; additional details are located in the appendices. During FY85 and FY86, the first steps were taken to answer questions regarding the performance of electrical equipment under severe accident conditions. These steps included (1) devising a method to identify important electrical components and the severe accident environments in which they are likely to fail, (2) using the method to analyze equipment performance for one nuclear power plant, and (3) writing a test plan to test the performance of a selected component to the severe accident environments. This work was done for the Performance Evaluation of Electrical Equipment during Severe Accident States (PEEESAS) Program.

1.1 Purpose of the PEEESAS Program

The Reactor Safety Study (WASH 1400) and subsequent probabilistic risk assessments have predicted that severe accidents dominate the risk. The Office of Nuclear Reactor Research established a Severe Accident Research Plan to provide an experimental and analytical basis for more accurate assessments of severe accident risks in nuclear power plants. An important part of the severe accident effort is to reduce the many substantial uncertainties in severe accident analyses. One significant source of uncertainty is the lack of data on component performance during a severe accident.

Severe accidents are defined as those which lead to either vessel breach or containment failure and which include the potential for core melt and/or release of radioactivity. (The resulting environment may or may not be more severe than the design basis LOCA environment.)

The purpose of the PEEESAS Program is to determine the performance of important electrical equipment under severe accident conditions. (Important electrical equipment is defined as electrical equipment that is important to safety.) This includes equipment which would be used to mitigate an accident or provide information on the status of the plant. Specifically, this program will

1. Devise a method to identify important electrical components and the severe accident environments in which they are likely to fail,

2. Use the method to analyze equipment performance for nuclear power plants,

3. Test the performance of selected components to the severe accident environments to determine performance, and

4. Provide the results of equipment performance to operators, emergency planning teams, probabilistic risk assessment analysts, and the Nuclear Regulatory Commission to influence actions and decisions.

The results of this program may influence operators, emergency planning teams and probabilistic risk assessment (PRA) analysts in the following ways. First, by knowing the chance that a given piece of electrical equipment will survive the severe accident environment, the operator may effectively deal with the accident by choosing a strategy to mitigate the accident as well as use plant status instrumentation that is not likely to fail. In addition, the results of this program may also influence the timing for evacuation and sheltering. If environmentally-induced failures are likely, core melt or containment failure may occur sooner than previously expected. Therefore, evacuation and sheltering should occur earlier. Furthermore, this program will provide PRA analysts with information on environmentally-induced equipment failures to incorporate into the PRA. (Currently, PRAs implicitly assume some level of performance capability. Environmentally-induced equipment failures are either assumed "negligible" or "certain" with less justification than is desirable.)

1.2 Prior Efforts, FY85 Work, and Plant Choice

A considerable amount of work has been done in the severe accident area by other programs. Concurrent with the development of NUREG-0900, many severe accident research programs have been started. These efforts include PRA analyses and phenomenological models which focus on identifying the dominant severe accident sequences, predicting environments and consequences, and suggesting possible arresting and/or mitigating strategies. The current analytical models predict severe accident sequence progressions, melt of the core, formation of debris beds, interaction of molten debris and concrete, containment environments, and the timing of severe accident events. Therefore, these models provide a good starting point to assess the performance of electrical components during a severe accident sequence.

During FY85, a method was devised to answer questions on the performance of electrical equipment under severe accident conditions. This method provided the means to identify important electrical components and the severe accident environments in which they are likely to fail. Also, this method was used, in FY85, to analyze the equipment and environments for one nuclear power plant and, in FY86, to develop a test plan to experimentally determine the performance of a selected component to the severe accident environments. Browns Ferry Unit 1, a BWR/Mark I, was the nuclear power plant chosen. For this plant, the following areas were investigated: (1) accident sequences (including operator actions that are likely to occur during those sequences), (2) important electrical equipment in the primary containment, (3) environmental profiles, (4) important electrical equipment that will be subjected to environments beyond their current qualification status, and (5) test plan for the selected equipment. This work is summarized in Sections 2.0 through 8.0, together with a discussion to illustrate how environmentally-induced failure of certain equipment during a severe accident may adversely impact the ability of a nuclear power plant to cope with severe accident conditions.

2. ACCIDENT SEQUENCES

2.1 Accident Sequence Selection

The selection of BWR/Mark I accident sequences was based on the following criteria: (1) sequences with high probability, (2) sequences with high risk, (3) sequences with the potential for extreme environments, and (4) sequences with operator action required. The high probability and risk sequences were chosen based on results from the Accident Sequence Precursor Study, Accident Sequence Evaluation Program, Industry Degraded Core Rulemaking Program, and Severe Accident Risk Reduction Program. The results of these studies are summarized in Table 1. From this table, the dominant sequences are TB (station blackout including loss of all AC power), TW (transient with loss of long-term heat removal), and TC (anticipated transient without scram).

These sequences appear to be dominant based on current knowledge. However, other accident sequences which were not considered to be as dominant were also included since environmentally-induced failures might cause a significant increase in their probability. Since time and money did not permit a review of every possible sequence, only two non-dominant sequences were examined: TQUV and AE. The accident sequence TQUV (transient with early loss of core cooling) was of moderate interest, even though it has a lower risk, due to the probability of the sequence. AE, a large LOCA with early loss of core cooling, was of interest due to the rapid steam environment.

The five accident sequences, described in detail in Appendix A, are defined by initiator, functions successful or failed, systems successful or failed including the cause of system failure, and likely scenario until core melt or containment failure occurs. (Functions include reactor subcriticality, reactor coolant system overpressure protection, core heat removal, containment heat removal, containment overpressure protection, and radioactivity removal.) The likely scenarios will be described in Section 2.2.

2.2 Likely Scenarios for the Five Accident Sequences

Likely scenarios, up to core melt or containment failure, were determined from the Severe Accident Sequence Analysis reports (References 1 through 4), Battelle Columbus report (Reference 5), and Emergency Procedure Guidelines (Reference 6). Likely scenarios are series of events that are most likely to happen during an accident sequence based on operator actions, timing of system failures, and automatic system actuation. More than one likely scenario may exist for a sequence due to insufficient information to choose

Table 1

Summary of BWR Accident Sequences

ACCIDENT SEQUENCES	PRECURSOR STUDY (AMONG TOP 90% CONTRIBUTION TO CORE MELT)	CURRENT ASEP CORE MELT <---------ESTIMATES--------->			IDCOR COMMITTED CORE MELT FREQUENCY ESTIMATES		AMONG HIGHEST RISK PER IDCOR	PRELIMINARY DOMINANT CONTAINMENT FAILURE MODE PER SARRP
		PEACH BOTTOM	LIMERICK	GRAND GULF	PEACH BOTTOM	GRAND GULF		
TQUV: transient with loss of core cooling	(TQUV and TB together) - yes	10^{-8}	10^{-7}	10^{-7}	10^{-8}	10^{-8}	No	Leak or late overpressure
TB (form of TQUV): station blackout	(TQUV and TB together) - yes	10^{-6}	10^{-6}	10^{-6}	10^{-7}	10^{-7}	Yes	Leak or late overpressure
TW (or TQW): transient with loss of long-term heat removal	Yes	10^{-7}	10^{-8}	10^{-7}	10^{-7}	10^{-7}	Yes	Core melt in failed containment
TC: ATWS	Yes	10^{-7}	10^{-8}	10^{-7}	10^{-6}	10^{-6}	Yes	Core melt in failed containment
TPQUV (or TPQE): like TQUV but with stuck-open relief valve	No	Not Assessed	10^{-8}	10^{-8}	?	$<10^{-8}$	No	--
TPW (or TPQ): like TW but with stuck-open relief valve	No	Not Assessed	Not Assessed	10^{-8}	?	10^{-8}	No	Core melt in failed containment
SI (including SHI and SJ): small LOCA with loss of long-term heat removal	No	Not Assessed	Not Assessed	10^{-7}	$<10^{-8}$	$<10^{-8}$	No	--
SE: Small LOCA with loss of core cooling	No	Not Assessed	Not Assessed	$<10^{-8}$	$<10^{-8}$	?	No	--
AI: Large LOCA with loss of long-term heat removal	No	Not Assessed	Not Assessed	Not Assessed	?	$<10^{-8}$	No	--
AE: Large LOCA with loss of core cooling	No	10^{-8}	Not Assessed	Not Assessed	10^{-8}	$<10^{-8}$	No	Leak or late overpressure

one path over another. The likely scenarios, with a description of the sequence and a description of the distinguishing features of each scenario, are shown in Table 2. For the description of the TB sequence, two cases were distinguished: a short-term scenario and long-term scenario involving battery depletion in four to seven hours.

Likely scenarios were used instead of worst-case (conservative and bounding) scenarios. The worst-case scenarios would not be as representative of the actual conditions in the plant during the accident and would place too much emphasis on unrealistically high environmental conditions and result in a minimum (and possibly insufficient) amount of operable equipment remaining to mitigate or prevent the accident. This would diminish the value of the results to both industry and the Nuclear Regulatory Commission.

The likely scenarios for each selected sequence identified (1) failed equipment by accident sequence definition, (2) equipment assessed to succeed and additional equipment needed to mitigate or provide plant status, and (3) boundary conditions for determining the environmental profile for each accident sequence up to core melt or containment failure.

Table 2

Likely Accident Scenarios

ACCIDENT CATEGORY	SEQUENCE DESCRIPTION	DISTINGUISHING FEATURE WITH SEQUENCE	SCENARIO NUMBER
TB	TOTAL BLACKOUT WITH HPCI/RCIC FAILURE.	OPERATOR DEPRESSURIZES RPV (IF DC POWER AVAILABLE)	1
		NO OPERATOR ACTION	2
		STUCK-OPEN RELIEF VALVE	3
	TOTAL BLACKOUT WITH HPCI/RCIC AVAILABLE UNTIL BATTERY FAILURE	OPERATOR DEPRESSURIZES RPV (UNTIL DC POWER FAILURE)	4
		STUCK-OPEN RELIEF VALVE	5
TW	TRANSIENT WITH NO RHR AVAILABLE FOR SUPPRESSION POOL COOLING. DRYWELL COOLER FAILURE AT 17 HOURS. SRVs FAIL AFTER 24 HOURS.	OPERATOR DEPRESSURIZES RPV	6
		STUCK-OPEN RELIEF VALVE	7
TC	TRANSIENT WITHOUT SCRAM. MANUAL ROD INSERTION AND LIQUID BORATION ARE FAILED LEVEL CONTROL, DRYWELL COOLER OPERATOR, AND MANUAL SRV CONTROL EVENTUALLY FAIL.	OPERATOR PERFORMS SUPPRESSION POOL COOLING OPERATOR DEPRESSURIZES RPV MANUAL LEVEL CONTROL (MSIVs CLOSED)	8
		AS ABOVE, FOR MSIV OPEN EVENT	8A
		OPERATOR PERFORMS SUPPRESSION POOL COOLING STUCK-OPEN RELIEF VALVE/OPERATOR DEPRESSURIZES RPV MANUAL LEVEL CONTROL (MSIVs CLOSED)	9
TQUV	TRANSIENT WITHOUT HPCI/RCIC/LPCS/LPCI AVAILABLE. CRD FLOW IS AVAILABLE BUT NOT SUFFICIENT TO PREVENT CORE DAMAGE	OPERATOR DEPRESSURIZES RPV	10
		STUCK-OPEN RELIEF VALVE	11
		OPERATOR LEAVES RPV AT OPERATING PRESSURE	12
AE	LARGE LOCA WITH NO HPCI/RCIC/LPCS/LPCI AVAILABLE	NO OPERATOR ACTION	13

3. ELECTRICAL COMPONENTS

An initial list of equipment important to safety was compiled from regulatory and industry documents and a general knowledge of which equipment is important to safety for BWR/Mark I dominant accident sequences (References 7 through 17). This initial equipment list included equipment needed for required operator actions, equipment for monitoring plant and containment conditions, equipment needed to provide information to make emergency response decisions, and equipment in safety systems required to manage dominant accident sequences. The initial list is found in Appendix B, Table B-2.

Based on a review of the Browns Ferry Unit 1 design, equipment from the above list was evaluated for equipment with electrical components located in the primary containment or reactor vessel. (Equipment in the primary containment or reactor vessel is generally in the most severe environment.) For example, since the instrumentation for determining reactor vessel water level is not located in the primary containment or reactor vessel, this instrumentation was eliminated from further consideration. The resulting list of equipment that is important to safety and is located in the primary containment or reactor vessel is found in Appendix B, Table B-3.

For the above equipment, qualitative arguments were made as to the relative importance of the individual components. The qualitative arguments considered the possible importance of the component in mitigating or assessing the status of the plant for each selected accident sequence. (For equipment used to mitigate the selected accident sequences, the equipment must provide core heat removal, reactor coolant injection, or containment heat removal.)

In Appendix B, each component is identified by manufacturer and model number (from Browns Ferry Unit 1 equipment qualification information) and judged to be of high, low, or moderate importance for the selected accident sequences. The importance of the component to the sequence is based on the function provided by the component, any positioning requirements, amount of time a component is useful during each sequence, and any backup systems which perform the same function. Components with moderate or high relative importance were retained. For these nine components, shown in Table 3, operability during a severe accident is most important as these components may be needed during that time. For each component, Table 3 also includes applicable sequences, maximum time component is required, and the required function to be performed.

Table 3

Components Recommended for Further Examination

COMPONENT	SEQUENCE*	TIME PERIOD*	USEFULNESS
Inboard MSIV Solenoid Valves	TW, TC TQUV, TB (after AC restored)	- Containment failure - Vessel breach	Reopening of MSIVs will restore a heat rejection path to avoid containment failure in TW or TC and possibly a core melt in TQUV or TB (if feedwater is also supplied).
HPCI, RCIC inboard isolation Valves	TW, TC, TQUV, TB (after AC restored)	Vessel breach	Reopening or sustained opening of the valves provides core cooling and hence prevents core melt.
Pilot Valves and Service Air Solenoid Valves For SRVs	TW, TC, TQUV, TB (after AC restored)	Vessel breach	Sustained functionality allows for low reactor vessel pressure operation to sustain or restore low pressure injection to the core, thus preventing core melt.
RHR Inboard Shutdown Cooling Valve	TW, TC	Containment failure	Opening of this valve to provide a RHR cooling path could be important in preventing containment overpressurization and failure.
In-Core Thermocouples and Reactor Vessel Surface Thermocouples	TW, TC, TQUV, TB, AE	Vessel breach	Provide status of core cooling adequacy.
Drywell Temperature Element (RTD)	TB, TQUV, TW	Containment failure	Provides status of drywell cooler operation and an indication of margin to containment failure.
Drywell Pressure Monitor	TB, TQUV, TW	Containment failure	Provides indication of containment venting effectiveness and margin to containment failure.
Drywell Hydrogen and Radiation Monitor	TB,TW,TC,TQUV, AE	Beyond containment failure	Provides estimate of core condition and release of fission products.
Cabling, Connectors/Splices, Terminal Blocks for Above Components	See Above	See Above	See Above

*Listed are those sequences for which survivability of the component could be most important for mitigating the accident or providing plant status information up to the time period indicated.

4. ENVIRONMENTAL PROFILES FOR EACH SCENARIO

In the following sections, environmental profiles of the primary containment will be discussed for each selected accident sequence. The severe accident environmental profiles will then be compared to typical equipment qualification profiles. Areas where the severe accident environmental profile exceeds the equipment qualification profile will be identified. Further details and the graphs for drywell temperature, suppression pool temperature, and containment pressure are found in Appendix C.

4.1 Environmental Parameters Considered

This section identifies the environmental parameters that should be considered when defining a severe accident environment. These parameters include humidity, submersion, spray, radiation/aerosols, vibration, pressure, and temperature. Except for pressure and temperature, each parameter is addressed briefly in the following paragraphs. (Section 4.2 will address pressure and temperature profiles.)

Humidity: Electrical equipment within the Browns Ferry Unit 1 containment has been qualified to 100 percent humidity conditions. Therefore, humidity has been adequately addressed by the design basis accident.

Submergence: The selected equipment is located above the possible flooding level based on plant qualification reports and equipment location information. It is therefore assumed that submergence represents little or no hazard to the selected equipment. However, information regarding the basis for the flooding calculations, for Browns Ferry Unit 1, could change this conclusion and cause the submergence issue to have to be reevaluated.

Spray: For accident sequences which use the drywell spray system to cool the containment or prevent containment overpressure, equipment may be exposed to water spray. Since direct spray impingement may present worse conditions than the 100 percent humidity case, water spray should be included for those sequences.

Radiation and Aerosols: Current equipment qualification testing requires exposing equipment to radiation dose levels derived from assuming that 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of other fission products are released to the containment environment. This leads to a maximum dose of 150 Mrads. Current information suggests that this dose is appropriate for severe accident sequences.

However, there are uncertainties associated with aerosol and other fission product dispersal patterns, such as direct plateout on the equipment or preferential radiation shine. In addition, for sequences with vessel breach prior to containment failure (TB and TQUV), aerosols are generated from the core/concrete interaction. Aerosol generation from concrete attack was not considered in this study due to the wide variability of concrete types.

Vibration: For those sequences where containment failure occurs before vessel breach (TW, TC, and AE), the resulting blowdown forces could cause vibration of equipment. Components required between containment failure and vessel breach may need to be examined under blowdown forces.

4.2 Pressure and Temperature Profiles for Each Scenario

Profiles of pressure and temperature, as a function of time, were generated for each likely scenario. These profiles were constructed using two computer codes: LTAS and MARCH. The LTAS code (Reference 18) was developed for Browns Ferry and models thermohydraulic behavior up to core damage. Because LTAS permits simulation of a variety of plant parameters and operator actions, the code was rerun for each selected scenario. MARCH (Meltdown Accident Response CHaracteristics) code results, from past studies, were used to simulate plant response for pressures and temperatures beyond core damage. Since the code was not rerun for each likely scenario, the MARCH results that did exist were examined to find a similar scenario.

A brief description of the trends in the pressure and temperature curves will be described below.

4.2.1 TB Short Term

No Operator Action Case or Late Depressurization by the Operator (if DC power available) Case

These cases can be combined because most of the boiloff occurred before the operator depressurized the reactor.

The initial increase in drywell temperature is due to the loss of the drywell coolers at the start of the accident and heat up of the primary system due to an immediate loss of core coolant injection. The loss of the drywell coolers also caused the containment pressure to increase. During the boiloff period, decay heat is removed by the safety relief valves to the suppression pool. The suppression pool serves as a heat sink until the reactor vessel is without water. However, the rate of temperature rise in the suppression pool decreases as the fixed water inventory is depleted through the safety relief valves. The boiloff period leads to core uncovery. With core uncovery, core melting occurs and vessel breach follows. Vessel breach occurs at 2.1 hours into the accident.

After vessel breach, the decay heat has a direct path to the drywell atmosphere; therefore, the drywell temperature and containment pressure increase. Also, the suppression pool temperature rises due to radiative heating from the containment atmosphere. At 3.2 hours into the accident, the containment fails due to leakage through the electrical penetration assembly seals at 500°F. The containment pressure just prior to containment failure was 100 psia.

<u>Early Depressurization from Stuck-Open Safety Relief Valve Case</u>

This case is similar to the case described above; only, for this case, depressurization occurs earlier. The same trends occur; although, vessel breach is now predicted to occur at 2.3 hours and containment failure at 2.9 hours. The containment failed due to leakage of the electrical penetration seals at 500°F. The containment pressure, just prior to containment failure, is 75 psia.

4.2.2 TB Long Term

<u>Depressurization by the Operator (until DC power failure) or from Stuck-Open Safety Relief Valve Cases</u>

The reactor vessel repressurization after battery failure, for the operator depressurization case, was not found to have a significant effect.

This case is similar to TB Short Term but with a seven hour delay due to coolant injection being available initially. Since coolant injection is available for four hours (until the batteries fail), more decay heat energy is removed to the suppression pool. Therefore, the suppression pool temperature reaches a higher peak value in TB Long Term than in TB Short Term. Vessel breach occurred at 9.0 hours and the containment failed at 10.0 hours into the accident. Once again, the containment failed due to leakage from the electrical penetration seals and the containment pressure, just prior to containment failure, was 100 psia.

4.2.3 TW

<u>Depressurization by the Operator or from Stuck-Open Safety Relief Valve Cases</u>

Both cases led to similar results. Continued steam removal from the reactor vessel to the suppression pool, during the stuck-open case, produced only minor differences in the profiles.

Initially, the drywell temperature decreased because the reactor scrammed, the drywell coolers are operating, and decay heat is removed through the safety relief valves to the suppression pool. At ten hours into the accident, the suppression pool has reached the boiling point for the containment pressure. Steam energy from the wetwell is transferred to the drywell; this caused the drywell temperature and containment pressure to increase. At seventeen hours, the drywell coolers fail due to high temperatures. Therefore, the drywell temperature increased until a new equilibrium was reached which accounted for the loss of the drywell coolers. (The loss of the drywell coolers had little effect on the containment pressure and suppression pool temperature.) Decay heat continues to be removed to the suppression pool. (For the case where the operator depressurizes the vessel, the operator loses control of the safety relief valves at 23 hours into the accident. However, after the vessel repressurizes, the safety relief valves will periodically reopen to maintain reactor vessel pressure at approximately 1000 psig.) As the suppression pool temperature increased, the drywell temperature and containment pressure increased until containment failure at 35 hours. Containment failure was due to the combined effect of drywell temperature (400°F) and containment pressure (120 psia). It was assumed that coolant injection was lost following containment failure. At first, the drywell temperature decreased as energy was released to the environment. Then, the drywell temperature increased as the core uncovered. Vessel breach occurred about 39 hours into the accident.

4.2.4 TC

With the MSIV Closed: Depressurization by the Operator or from Stuck-Open Safety Relief Valve Cases

Because of the failure of the control rods to insert, the reactor power level may be as high as 30 percent. This heat is dumped to the suppression pool through the safety relief valves. The drywell temperature remains constant for the first 1400 seconds; the reactor is dumping the majority of its energy to the suppression pool and the drywell coolers are operating. However, the suppression pool cannot continue to remove enough heat. After 1400 seconds, the suppression pool temperature has reached the saturation point. Once the suppression pool starts to boil, the energy from the pool causes an increase in drywell temperature and containment pressure. By the time of containment failure at 0.9 hours into the accident, the drywell temperature and pressure has increased to 360°F and 132 psia. (Containment failure is due to the high containment pressure.) After containment failure, it was assumed that coolant injection fails. The reactor does shut down due to a lack of moderator, but the core uncovery leads to vessel breach at 3.8 hours into the accident.

With the MSIV Open: Depressurization by the Operator Case

This case is similar to the MSIV closed case for TC. Because of the failure of the control rods to insert, the reactor power level may be as high as 30 percent. However in this case, the power conversion system dissipates 20-25 percent of the heat through the condenser. The remaining heat (5-10 percent) is dumped to the suppression pool through the safety relief valves. This results in a longer time for the suppression pool to reach saturated conditions and containment failure occurs at 3.9 hours into the accident (drywell temperature = 345°F and containment pressure = 132 psia). Containment failure is due to the high containment pressure. Vessel breach is estimated to occur at 6.7 hours into the accident.

4.2.5 TQUV

No Operator Action Case

Since TB is a specialized case of TQUV, the trends in the drywell temperature, suppression pool temperature, and containment pressure profiles are similar. (Both sequences have similar boiloff calculations.) As in the TB sequence, the suppression pool temperature was assumed to change insignificantly. The drywell temperature and containment pressure were also assumed to change insignificantly until the point of vessel breach. Vessel breach occurred at 4.9 hours into the accident and the containment failed at 7.0 hours into the accident. The containment failure was due to leakage of the electrical penetration seals at a drywell temperature of 400-500°F.

Depressurization by the Operator or from Stuck-Open Safety Relief Valve Cases

This case is similar to that described above for TQUV with no operator action. Since the reactor is depressurized, water is available at a higher flow rate from the control rod drive system. This postpones vessel breach and containment failure. Vessel breach occurs at 7.0 hours and containment failure occurs at 8.2 hours into the accident.

4.2.6 AE

No Operator Action Case

The drywell temperature and containment pressure increase due to direct exposure to the superheated steam/water mixture from the large break. Throughout this sequence, the suppression pool remains subcooled. However since no coolant injection is available, the core uncovers. The drywell temperature and containment pressure remain constant until decay heat causes the core to slump. At this point, the drywell temperature and containment pressure experience a tremendous rise due the production of hydrogen and the transport of noncondensible gases to the suppression pool. The containment fails at 0.7 hours with a peak temperature predicted to be in excess of 2000°F and containment pressure of 138 psia. Vessel breach occurs at 2.1 hours into the accident.

4.3 Typical Equipment Qualification Profiles

IEEE 323-1974 addresses the qualification of Class 1E equipment for nuclear power plants. This document states that testing is the preferred method to prove qualification and that equipment must be tested according to the environmental profile of the specific plant. (The environmental profile is based on the postulated design basis event (large LOCA).) Furthermore, the actual test profile must include both the environmental profile and margin to account for variations in manufacturing and the uncertainty in defining satisfactory performance. This margin includes (1) additional peak transient, (2) increasing the temperature by 15°F, (3) increasing the pressure by 10 percent of the gauge pressure, and (4) increasing the time (that equipment must operate following the design basis event) by 10 percent.

Although actual test profiles must be based on plant-specific calculations, a representative test profile is presented in IEEE 323-1974, Appendix A. These temperatures and pressures, as a function of time, were calculated for a large LOCA in a pressurized water reactor and a boiling water reactor. The larger value of temperature or pressure, at time, was used to develop the typical equipment qualification profile used in this study. In addition, IEEE 323-1974 Appendix A also gives an accident dose of 150 Mrad and a demineralized water spray rate, for a boiling water reactor, of 0.15 (gal./min.)/sq. ft.

For this study, it was assumed that the selected equipment had been qualified to these levels. This may not be true in all cases. For cases where equipment has been qualified to levels below the typical equipment qualification profile, the results of this study may have to be modified to include additional equipment.

4.4 Comparison of the Pressure and Temperature Profiles for Each Scenario to the Typical Equipment Qualification Profile

Environmental profiles, for each scenario, were compared with the typical equipment qualification profile. Since severe accident environments in excess of the equipment qualification levels may result in equipment failure, equipment survivability is questioned during these scenarios.

While the equipment qualification profile envelopes some of the environmental conditions (drywell temperature, suppression pool temperature, or containment pressure) which occur for the severe accidents examined, some severe accident sequence environmental conditions have higher temperatures and pressures than current equipment qualification requirements. These profiles are identified in Table 4. Additionally, accident sequences involving long-term containment failure produce conditions different from those of current equipment qualification requirements. These sequences yield relatively low environmental conditions early in the sequence and exceed the current equipment qualification conditions later in the sequence. Therefore, the performance of important electrical equipment may need to be determined for profiles of both general types of sequences--those with fast rising environmental conditions and those with slow rising pressure and temperature profiles. While it may be impractical to expect to achieve equipment qualification at the maximum pressures and temperatures seen in a severe accident, any increase in the current qualification limit may increase the potential for equipment survivability during the severe accident.

Table 4

Comparison of Severe Accident Profiles
to Typical Equipment Qualification Profiles

PROFILE NO.	ACCIDENT	TIME OF VESSEL BREACH (1)	TIME OF CONTAINMENT FAILURE (1)	ENVIR. VARIABLES	TIME ABOVE QUAL LVL (1)	PEAK AMP./%> QUAL LVL (2,3)	TIME OF PEAK AMP. (1)	TIME FIRST EXCEEDS QUAL LVL (1)
1	TB-Short: No/Late Oper. Action	2.1	3.2	D.W. Temp.	.2	500/142%	3.2	3.0
				S.P. Temp.	0	180/51%	3.2	---
				D.W. Press.	.2	100/118%	3.2	3.0
2	TB-Short: Vessel Depress. Early	2.3	2.9	D.W. Temp.	.2	500/142%	2.9	2.7
				S.P. Temp.	0	170/49%	2.9	---
				D.W. Press.	0	75/88%	2.9	---
3	TB-Long	9.0	10.0	D.W. Temp.	2.0	500/142%	10.0	8.0
				S.P. Temp.	0	212/60%	10.0	---
				D.W. Press.	.5	100/118%	10.0	9.5
4	TW	39	35	D.W. Temp.	11	500/142%	39	28
				S.P. Temp.	8	350/100%	35	34
				D.W. Press.	7	125/147%	35	28
5	TC*	3.8(6.7)	0.9(3.9)	D.W. Temp.	<.1 (-)	360/103% (345/98%)	.9(3.9)	.9(-)
				S.P. Temp.	0 (-)	340/97% (340/97%)	.9(3.9)	---(-)
				D.W. Press	.75(.5)	132/155% (132/155%)	.9(3.9)	.6(3.4)
6	TQUV: Vessel At Pressure	4.9	7.0	D.W. Temp.	.5	500/142%	7.0	6.5
				S.P. Temp.	0	170/49	7.0	---
				D.W. Press.	.8	100/129%	7.0	6.2
7	TQUV: Vessel Depress.	7.0	8.2	D.W. Temp	.4	500/142%	8.2	7.8
				S.P. Temp.	0	200/57%	7.0	---
				D.W. Press.	.3	110/129%	8.2	7.9
8	AE	2.1	.66	D.W. Temp.	1.44	2000/571%	.7	.6
				S.P. Temp.	0	270/77%	.66	---
				D.W. Press	.1	138/162%	.7	.6

Notes:

(1) All Times in Hours
(2) All Temperatures in °F/All Pressures in psia
(3) Percentage Based on Exceeding Qualification Profile Maximum Value

*Information in parenthesis is for MSIV open case.

5. REDUCTION OF EQUIPMENT AND ENVIRONMENTS TO SELECT TEST CANDIDATES

5.1 Screening and Ranking

A screening and ranking process was used to select the best test candidates and test profiles by reducing the number of possible equipment and environments. Further details are found in Appendix D.

5.1.1 Time Equipment Demanded and Environments Exceeding Qualification Levels

In Section 3, Table 3, electrical equipment was selected and the time period when the equipment may be used, during each accident sequence, was identified. The time during which the equipment may be used and the severe accident environment, prior to or during that time, was compared. (Although some equipment may be used after both vessel breach and containment failure, this equipment would have less impact than equipment needed before vessel breach and/or containment failure. Therefore, the equipment will only be evaluated until both containment failure and vessel breach have occurred.)

The pressure and temperature environments were examined in great detail. If the equipment was needed during the accident sequence and if the accident profile was above the typical qualification profile prior to or during the time that the equipment was needed, the equipment and profile were retained. This process eliminated equipment with severe accident environments below that of the typical qualification environment. Then, the severe accident environments were further reduced by retaining only those profiles with (1) maximum pressure or temperature or (2) maximum time above the maximum pressure or temperature for the typical equipment qualification profile. These results are shown in Table 5 and represent profiles where the equipment must operate under the most severe conditions for the five selected accident sequences. The pressure and temperature parameters were categorized using (1) "high" for equipment that is required during or after the time the severe accident profile is greater than 40 percent above the maximum equipment qualification level, (2) "medium" for equipment where the severe accident profile is between the maximum equipment qualification level and 40 percent above the equipment qualification level, and (3) "low" for equipment where the severe accident profile is less than the maximum equipment qualification level.

The remaining environmental parameters were ranked as follows. Except for the AE sequence, humidity, spray, submergence, and radiation were categorized as "low" prior

Table 5

Profiles Where Equipment Must Operate Under the Most Severe Conditions

PROFILE	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMOCOUPLE	DRYWELL TEMP. RTD	DRYWELL PRESSURE	HYDROGEN/RADIATION MONITOR
TB3: drywell temperature						142% 2.0 hrs/500°F	142% 2.0 hrs/500°F	
TW4: drywell temperature		142% 11 hrs/500°F		142% 11 hrs/500°F	142% 11 hrs/500°F			142% 11 hrs/500°F
TW4: drywell pressure	147% 7 hrs/125 psia	147% 7 hrs/125 psia	147% 7 hrs/125 psia	147% 7 hrs/125 psia	147% 7 hrs/125 psia	147% 7 hrs/125 psia	147% 7 hrs/125 psia	147% 7 hrs/125 psia
TC5: drywell pressure*	155% .75 hrs/ 132 psia	155% .75 hrs/ 132 psia	155% .75 hrs/ 132 psia	155% .75 hrs/ 132 psia				
AE8: drywell temperature					571% 1.4 hrs/2000°F			571% 1.4 hrs/2000°F
AE8: drywell pressure					162% .1 hr/138 psia			162% .1 hr/138 psia

* Since the TC pressure profile for the MSIV open case is the same magnitude and shorter duration, the profile will be represented in further analysis by the TC MSIV closed profile.

Notes:
(1) All percentages indicate projected environmental level relative to the maximum qualification level.
(2) All times indicate how long the projected environment is in excess of maximum qualification level.
(3) Listed temperatures (°F) and pressures (psia) indicate maximum projected environmental level.

to vessel breach and "medium" after vessel breach. For AE, these parameters were considered to be "medium". The vibration parameter was "low" prior to containment failure and "medium" after containment failure. In addition, the following combinations of parameters were considered: pressure/moisture (humidity or steam), temperature/moisture (humidity or steam), and temperature/radiation. The combined environments were given the combined ranking of each individual parameter. The results for an MSIV are shown in Table 6.

The results described above were tabulated for each sequence using a point system. (For a description of the point system, see Appendix D.) These tabulated results, shown in Table 7, are listed for each piece of equipment. For example, the worst environments for the MSIV are found in the TW and TC accident sequences.

5.1.2 Functional Importance

The relative functional importance of the equipment, based on each accident sequence where the equipment may be needed, was determined for the equipment identified in Section 3, Table 3. Seven criteria were used to measure equipment importance. These criteria describe conditions which imply lower functional importance. The criteria are defined below.

Redundancy: More than one component to perform the equipment function (such as four MSIV valves or two drywell temperature devices).

Backup Systems: Totally different systems able to perform the same function.

Noncomplexity: Few or simple (mechanical or electrical) parts and functions.

Electrical Independence: More than one electrical bus to supply power to the equipment.

Fail-safe Position Appropriate: The deenergized or failed state allows the equipment to operate as required for the sequence. (For example, the fail-safe position of the MSIV is closed, but for the sequences of concern, the MSIV must open. Therefore, the MSIV is not considered to be in an appropriate fail-safe position).

Plant Status Indication Only: Passive equipment which only provides the status of the system, but cannot by itself actively influence the accident sequence outcome.

Separation: Physical distance between redundant components in the equipment.

Table 6

Ranking of Environmental Parameters

MAJOR EQUIPMENT CATEGORY: MSIV

SEQUENCE: ENVIRONMENT	TB	TW	TC	TQUV	AE
Temperature	medium	medium	medium	low	---
Pressure	low	high	high	low	---
Humidity	low	low	low	low	---
Spray/Submergence	low	low	low	low	---
Radiation	low	low	low	low	---
Vibration	low	low	low	low	---
Pressure/Humidity	low	high/low	high/low	low	---
Temperature/Humidity	medium/low	medium/low	medium/low	low	---
Temperature/Radiation	medium/low	medium/low	medium/low	low	---
TIME OF APPLICABILITY:	Vessel Breach	Containment Failure	Containment Failure	Vessel Breach	---

Table 7

Environmental Importance: Tabulated Results for Each Piece of Equipment

SEQUENCE	EQUIPMENT							
	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMO-COUPLE	DRYWELL TEMP RTD	DRYWELL PRESSURE	HYDROGEN/ RADIATION MONITOR
TB	18	18	--	18	18	40	40	40
TW	26	34	26	34	34	26	26	34
TC	26	28	26	28	28	--	--	28
TQUV	12	12	--	12	12	37	37	40
AE	--	--	--	--	46	--	--	46

The equipment was evaluated against each criterion using a point scheme such that the lower the total points for the equipment, the more functionally important the equipment. Equipment information was based on Browns Ferry Unit 1 schematic drawings and engineering knowledge of other "typical" BWR designs.

It is important to realize that all of the equipment has been judged, in Section 3, to be functionally important to the severe accidents. This ranking is simply used to judge the relative importance of the equipment. As shown in Table 8, the main steam isolation valves and safety relief valves are functionally more important than the other equipment.

5.1.3 Ranking

The results, from Tables 7 and 8, were evaluated by a ranking process. The equipment was assigned a value of high, medium, or low by dividing the points that were given in the environmental and functional importance screens into thirds. These results are shown in Table 9. Equipment with high functional importance and with high or medium environmental conditions were retained as possible test candidates. The high and medium environmental conditions include environments above the maximum equipment qualification levels.

The resulting potential test candidates included the main steam isolation valves and the safety relief valves. Both test candidates were required to operate during two accident sequences: TC and TW. These four cases were judged to be equivalent in terms of possible test candidates and test profiles.

5.2 Importance of Selected Test Candidates to Probabilistic Risk Assessments (PRA)

This section describes possible changes to current PRA estimates of accident sequence probabilities and risk, if environmentally-induced equipment failures occur. This analysis is important for two reasons. Currently, the probability of equipment failure is based on operator actions, test and maintenance activities, and past performance--the failure of equipment due to a severe accident environment generally has not been considered. One of the goals of this program is to determine the impact of the environment on equipment and to provide this data incorporation into a PRA. The second reason to use a PRA analysis is that it serves as a convenient method, other considerations being equal, to choose a first test candidate. Therefore, the effect of environmentally-induced equipment failures on probabilistic risk assessments was examined for the four remaining cases. This was done from a relative point of

Table 8

Functional Ranking Results

EQUIPMENT: FUNCTIONAL CATEGORY	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMO-COUPLE	DRYWELL TEMP RTD	DRYWELL PRESSURE MONITOR	HYDROGEN/ RADIATION MONITOR
DEGREE OF REDUNDANCY	1	1(0 for TC Sequence)	1	1	1	1	0	1
BACK-UP SYSTEMS	1(0 for TC Sequence)	1	1	0	1	1	1	1
NON-COMPLEXITY	0	0	0	0	1	1	1	1
ELECTRIC INDEPENDENCE	0 or 1	1	1	0 or 1	0 or 1	0 or 1	1	0 or 1
FAIL SAFE	0	1	0	0	1	0	0	0
PLANT STATUS	0	0	0	0	1	1	1	1
PHYSICAL SEPARATION	0	0	0	0	0	0	0	0
TOTAL SCORE	1-3	3-4	3	1-2	5-6	4-5	4	4-5

Table 9

Functional/Environmental Ranking Comparisons

EQUIPMENT:	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMO-COUPLE	DRYWELL TEMP RTD	DRYWELL PRESSURE MONITOR	HYDROGEN/ RADIATION MONITOR
FUNCTIONAL IMPORTANCE:	(H)	(M)	(M)	(H)	(L)	(L)	(M)	(L)
SEQUENCES								
TB	L	L	-	L	L	H	H	H
TW	M	H	M	H	H	M	M	H
TC	M	M	M	M	M	-	-	M
TQUV	L	L	-	L	L	H	H	H
AE	-	-	-	-	H	-	-	H

H = HIGH

M = MODERATE

L = LOW

view by concentrating on the degree of change in probability or risk rather than absolute numbers.

In order to understand the purpose of the MSIVs and SRVs in TW and TC, it is important to understand the definition of the sequences. Typically, transients are grouped into three major categories: (1) loss of off-site power, (2) initial loss of the power conversion system (PCS), and (3) PCS initially available but subsequently lost as a result of perturbations to the PCS following a reactor trip. These three transient categories contribute about 1 percent, 10 percent, and 90 percent, respectively, to the overall frequency of transients at nuclear power plants (References 19 and 20).

Plant performance data has shown that the probability of recovering off-site power, and hence the ability to restore the use of the PCS or other AC-driven systems as a primary heat removal path, is greater than 50 percent in about one-half hour and exceeds 90 percent by approximately four to five hours after the initial power loss. Events involving loss of the PCS are similar in that there is an estimated 90 percent chance of restoring the PCS by approximately four to five hours after its initial loss (Reference 21). Furthermore, for cases where the PCS has been lost due to perturbations in the system (not hardware faults), recovery of the PCS is even more likely.

Therefore, there is a high chance of recovering the PCS following the initiating transient. In addition, the Emergency Procedure Guidelines often stress using the PCS as the preferred source of heat removal since operators are familiar with the PCS. With PCS recovery likely, especially during the long time prior to containment failure in the TW scenario, the ability of the MSIVs to be reopened following exposure to the severe accident environment becomes important. In addition, because the Residual Heat Removal (RHR) system is of little use in a TC scenario (power level too high) and the RHR system may have failed (and hence recovery is uncertain) in the TW scenario, MSIVs are important in these two sequences.

However, if the PCS cannot be restored, then the operability of the primary system SRVs, in TW and TC, becomes more important--particularly since the high pressure injection systems will eventually fail due to the high temperature of the suppression pool water which provides the water source for the high pressure injection systems. Therefore, using the SRVs would permit the low pressure injection system to operate.

5.2.1 Specific Effects of Environmentally-Induced Equipment Failure on Accident Sequences

The specific effects of environmentally-induced equipment failure on sequence probability or risk are described below.

<u>MSIV and the TC (MSIV open) Sequence</u>

There is a 50 percent chance of the MSIV closing in this sequence (Reference 22). Current PRA estimates do not give credit for reopening the MSIVs if the MSIVs close at or shortly following sequence initiation because of (1) the time necessary to equalize pressure around the valves and open the valves, (2) the short time to restore failed portions of the PCS, and (3) the many other operator actions needed to manage the accident. Since the current probability of reopening the MSIVs is 0.0, any environmentally-induced failure of the MSIVs will not alter the sequence probability.

However, if the MSIVs initially stay open, there is more time to recover from the accident before containment failure and vessel breach. If the MSIVs should subsequently shut because of a previously unconsidered environmentally-induced failure, the chance of the MSIVs closing becomes 100 percent. This results in the same sequence probability, but increases the risk associated with the TC sequence.

<u>MSIV and the TW Sequence</u>

Current PRA estimates assume that as the sequence progresses, the probability of failing to restore the PCS decreases exponentially with time (Reference 21). This assumes that the MSIVs are able to operate throughout the sequence. However, the valves may be unavailable since the drywell temperature and containment pressure exceed equipment qualification profiles 18 hours into the accident and the maximum equipment qualification level about 27 hours into the accident. If an environmentally-induced failure of the MSIV assembly occurs during this time, then there may be only 18-27 hours to recover the PCS instead of the 35 hours presently used in PRA analyses. The nonrecovery probability at 18 hours is 0.02 and the nonrecovery probability at 35 hours is 0.004. Therefore, the sequence probability could change by as much as a factor of 5 if environmentally-induced failure of the MSIVs occurred after 18 hours into the sequence.

<u>SRV and the TC (MSIV closed) Sequence</u>

As explained in the paragraphs entitled "MSIV and the TC Sequence," PRAs distinguish between MSIV open and closed cases. SRV operation is important in the MSIV closed case. After the suppression pool temperature reaches about 200°F, pumps for the high pressure injection system may fail due to high lube oil temperatures or low suction head. At this time, the operator must be able to operate the SRVs to permit operation of the low pressure injection systems. Current PRAs estimate the probability to fail to depressurize at approximately 0.1 (Reference 22). Since the TC sequence has containment pressures exceeding equipment qualification limits, an environmentally-induced failure of the SRV may occur. The environmentally-induced failure may change the depressurization failure to a probability of 1.0. This results in a factor of 10 increase in the TC probability.

SRV and the TW Sequence

For this sequence, the need for low pressure injection systems is relatively low. Even if the high pressure injection systems fail due to suppression pool temperature, the control rod drive (CRD) system is available and should be adequate to maintain reactor vessel water level. Once the containment has been vented or containment failure occurs, the probability of continued operation of the CRD system is currently estimated at approximately 0.9 (Reference 22). Should CRD failure occur, then the SRVs would be needed to depressurize the reactor vessel so that low pressure injection systems could be used to maintain coolant level. Although the TW environment will exceed the equipment qualification levels in this sequence, since the probability of the TW sequence coupled with CRD failure is relatively low, the effect of environmentally-induced SRV failures on the TW sequence are relatively small.

5.2.2 Effect of Environmentally-Induced Equipment Failures on the Total Core Melt Probability

Based on past IDCOR and ASEP work, the total core melt probability per reactor year for some BWR-4s is approximately 10^{-5}. The TC sequence generally accounts for about 50 percent of this total probability, TW sequences 10 percent, and TB sequences 40 percent. Since environmentally-induced failures may cause the TC sequence probability to change by a factor of 10, the overall core melt probability would increase by a factor of 5. Likewise, if environmentally-induced failures in the TW sequence result in an increase in the sequence probability by a factor of 5, the total core melt probability would increase by a factor of 1.5.

5.2.3 Resulting Test Candidates and Environments

Therefore, the recommended test candidates and environments are the MSIVs for the TW or TC (MSIV open) accident sequences and the SRVs for the TC (MSIV closed) accident sequence.

5.2.4 PRA and Emergency Preparedness Insights

The effects of the environmentally-induced failure of the main steam isolation valve and the safety relief valve were discussed above. Environmentally-induced failures of electrical equipment can cause an increase in the current core melt probability and risk estimates. However, changing the equipment to better withstand the severe accident environments may reduce the probability of equipment failure and risk. Until the equipment is demonstrated to withstand the severe accident environments, it may be important for PRAs to include the effects of environmentally-induced failures in their scope.

With regard to emergency preparedness, current evacuation plans require evacuation once vessel breach or containment failure has either occurred or is deemed imminent (Reference 23). However, the results of this study indicate that the basis for evacuation should be reexamined. For example, although containment failure for the TW sequence is currently predicted to occur at 35 hours into the sequence, environmental conditions have exceeded current equipment qualification levels after 18 to 24 hours. Therefore, containment failure may occur sooner than currently expected if environmentally-induced failure of critical indicators or systems is considered (in this example, perhaps as much as 17 hours sooner). This insight may have bearing on current emergency planning for this and other sequences which exceed equipment qualification limits.

6. TEST PLAN INPUT

6.1 Choosing the Test Candidate

From a PRA perspective, both the MSIV and the SRV are good test candidates. But for the first test candidate, the MSIV equipment assembly was chosen because failure of the MSIV may increase the core melt probability as well as increase the risk, and the performance of the MSIV may be tested in more than one accident environment.

Several pieces of equipment are associated with the MSIV equipment assembly: pneumatic control manifold assembly, position switch, main steam drain valve actuator, and globe valve. As discussed in Appendix C, the position switch is less important than the pneumatic control manifold assembly since the pneumatic control manifold assembly is required to operate the MSIV globe valve. The main steam drain valve actuator may be needed to equalize the pressure across the MSIV, but since the MSIV is a globe valve, the globe valve may open even if the pressure across the valve is not equalized. In addition, the heat rejection path associated with the main steam drain valve actuator is smaller than the heat rejection path associated with the pneumatic control manifold assembly. Furthermore the pneumatic control manifold assembly, a complex electrical component, is more susceptible to failure than the globe valve.

Therefore, the MSIV pneumatic control manifold assembly was chosen to be the FY86 test candidate. Both the TC (MSIV open) and TW accident sequence profiles will be used.

6.2 Expected Failure Modes

Possible failure modes of the test candidate due to different environmental parameters are given in Table 10. The failure modes were based on information from manufacturing data, Licensee Event Reports, operation and maintenance records, I & E Information Notices, Qualification Testing Evaluation Program reports, vendor reports, and TMI-2 reports. This literature review identified one major cause of failure for electrical equipment, within containment, was moisture intrusion: (1) humidity and temperature, and (2) humidity and pressure. With this in mind, it may be that equipment longevity in severe accident environments can be enhanced by removing vulnerable components from containment or protecting the vulnerable components from moisture intrusion. However, these options for coping with moisture effects will not be studied in this current program. Instead, moisture will be investigated as one environmental parameter in the test plan. In addition,

Table 10

Failure Modes for the Test Candidate

ENVIRONMENTS	SOLENOID ASSEMBLY	CABLING	JUNCTION BOXES/ CONNECTORS
TEMPERATURE	WINDING INSULATION MELT/HOT SPOT MECHANICAL BINDING	INSULATION MELT/ FAULT PROPAGATION	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY
PRESSURE	SEAL MISALIGNMENT /FAILURE	FAULT PROPAGATION	GASKET SEAL MOVEMENT/ MISALIGNMENT
HUMIDITY	ELECTRICAL SHORT THROUGH EXISTING FAULT	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRA- TION ELECTRICAL SHORT
SPRAY OR SUBMERGENCE	ELECTRICAL SHORT THROUGH EXISTING FAULT	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRA- TION ELECTRICAL SHORT
RADIATION	SEAL/WINDING INSULATION EMBRITTLEMENT	EMBRITTLEMENT/ SHRINKAGE LEADING TO BREAKS & CRACKS	SEAL EMBRITTLE- MENT
VIBRATION	MECHANICAL BINDING LOSS OF WINDING CONTINUITY	RAPID FAULT PROPAGATION MECHANICAL SEVERANCE	SEAL FAILURE CABLE TO CONNECTOR MECHANICAL SEVERANCE
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS
COMBINATION 2	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS MECHANICAL BINDING	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENETRATION	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENETRATION
COMBINATION 3	WINDING EMBRITTLEMENT /HOT SPOTS	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION	JOINT EXPANSION/ SEAL EMBRITTLEMENT LEADING TO LOSS

COMBINATION 1 = PRESSURE/HUMIDITY
COMBINATION 2 = TEMPERATURE/HUMIDITY
COMBINATION 3 = RADIATION/TEMPERATURE

recent testing has identified a synergistic effect when cables are simultaneously exposed to radiation and a LOCA environment. These environmental parameters are also considered in the test plan.

7. SUMMARY OF THE TEST PLAN

The test plan for the pneumatic control manifold assembly is summarized below. Further details are given in Appendix E.

7.1 Sample Description and Mounting

The pneumatic control manifold assembly described below is in place in many licensed BWR plants. Two identical test specimens will be purchased.

Each test specimen has three solenoids (1-125 Vdc and 2-120 Vac) to operate three valves (4-way, 3-way, and 2-way). The solenoids have Class H insulation and the valves have Viton seals. The valves are lubricated with Parker Super-O-Lube.

Each test specimen operates in the following fashion: either the 120 Vac or 125 Vdc main control solenoid activates the 4-way valve; if either main control solenoid fails, the 4-way valve may be operated by the other main control solenoid; and if the 4-way valve fails to cause the MSIV to close, the 2-way valve may be used to close the MSIV. The remaining 120 Vac exercise control solenoid operates the 3-way valve. The 3-way valve is normally used to slowly close the MSIV, during normal plant operation, to determine if the MSIV will shut. Although the 3-way valve and exercise control solenoid can only slowly close the MSIV, they may be used if all other valves and solenoids fail.

As shown in Figure 1, each test specimen will be mounted at a forty-five degree angle with the solenoids upside down during the accident exposure. This is the usual installed configuration at nuclear power plants.

7.2 Test Strategy

The performance of the test specimens is to be evaluated under conditions simulating the TC (MSIV open) and TW accident sequences. During these sequences, the MSIV need only be opened prior to containment failure. Since core melt occurs after containment failure, the test specimens need not be exposed to severe accident radiation levels. In addition, the containment spray system is not operated in the TC and TW sequences; therefore, the test specimens will not be exposed to spray.

7.2.1 Aging Simulation

Both test specimens will be exposed to simultaneous radiation and thermal aging with the solenoids energized. Since the manufacturer recommends replacing some organic materials

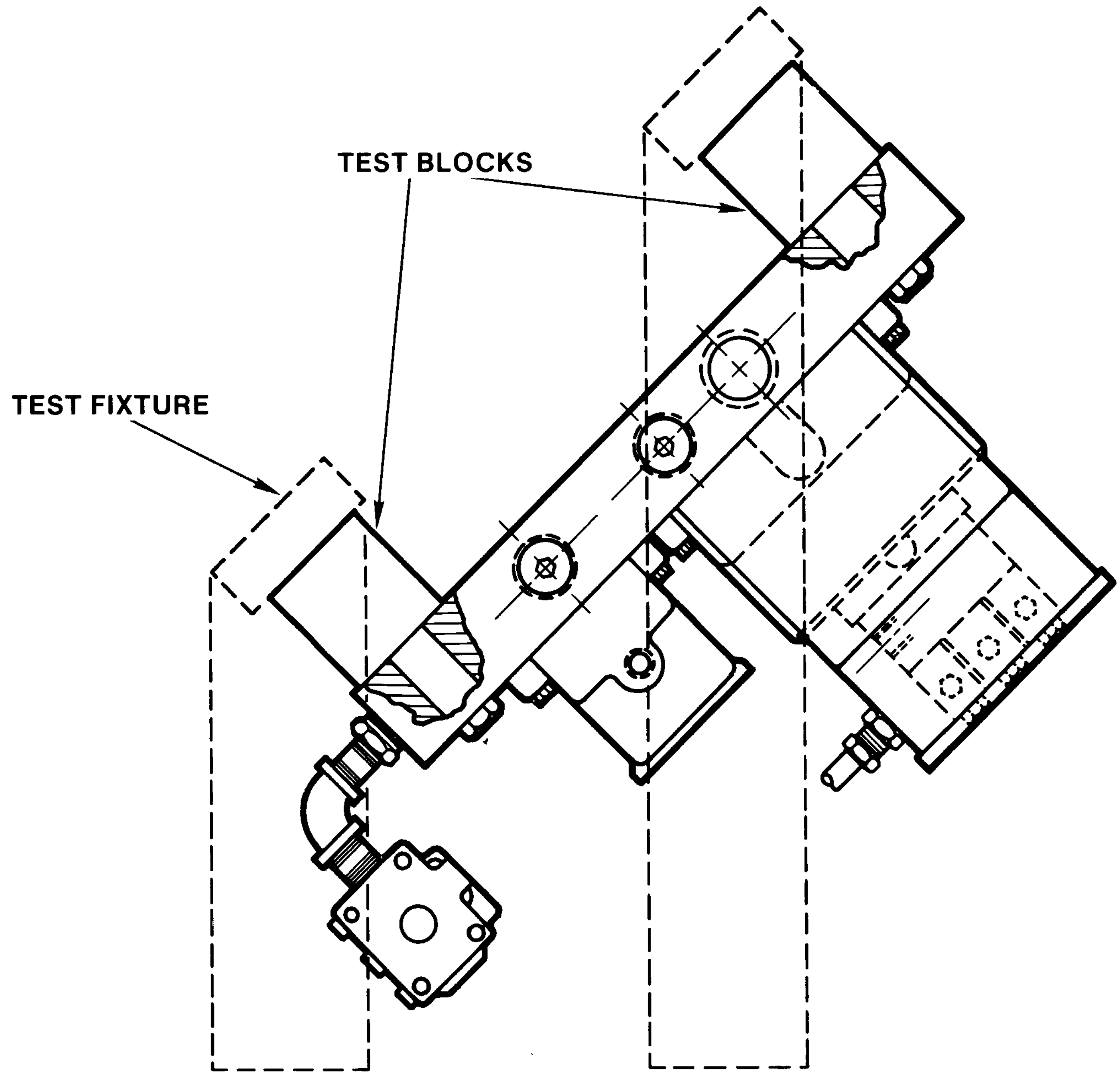


Figure 1. Typical Mounting Configuration

after 15 months, the valve will be aged to an equivalent of 15 months at a service temperature of 185°F. The accelerated aging temperature is 266°F with a total dose of 1.25 Mrads. To avoid overaging the elastomeric materials, the aging will be done in two steps: (1) solenoids for 4 days and (2) the entire assembly for 12.2 days. (Self-heating of the coil has been accounted for.)

7.2.2 Accident Simulation

Each test specimen will be exposed to an accident profile. The solenoids will be energized (rated voltage ± 10 percent) and the valves will be pressurized with instrument air (150 ± 10 psig), as needed.

Test #1

In Test #1, the TC (MSIV open) accident sequence profile will be used. These temperature and pressure profiles are shown in Figures 2 and 3. The profile will be followed until containment failure at 4.5 hours. If the valve remains open, the chamber pressure and temperature will be increased to determine the fragility level of the test specimen.

To determine the fragility level of the test specimen, the chamber temperature will be increased in 25°F increments (and held at that temperature until the valve has stabilized at the chamber temperature for ten minutes) and the pressure will be increased in 5 psig increments. The temperature will continue to be increased until the valve fails to remain open; however, the pressure will only be increased to a maximum pressure of 132 psig due to differential pressure requirements.

The valve must be energized throughout Test #1. At the conclusion of the test, the valve will be closed (if necessary). The valve must close and remain closed, at that time, to perform its required safety function.

Test #2

In Test #2, the TW accident sequence profile will be used. These temperature and pressure profiles are shown in Figures 4 and 5. The profile will be followed until containment failure at 35 hours. During this time, the valve will be cycled every 2 hours. If the valve can still be cycled from the closed position to the open position at containment failure, the chamber pressure and temperature will be increased to determine the fragility level of the test specimen.

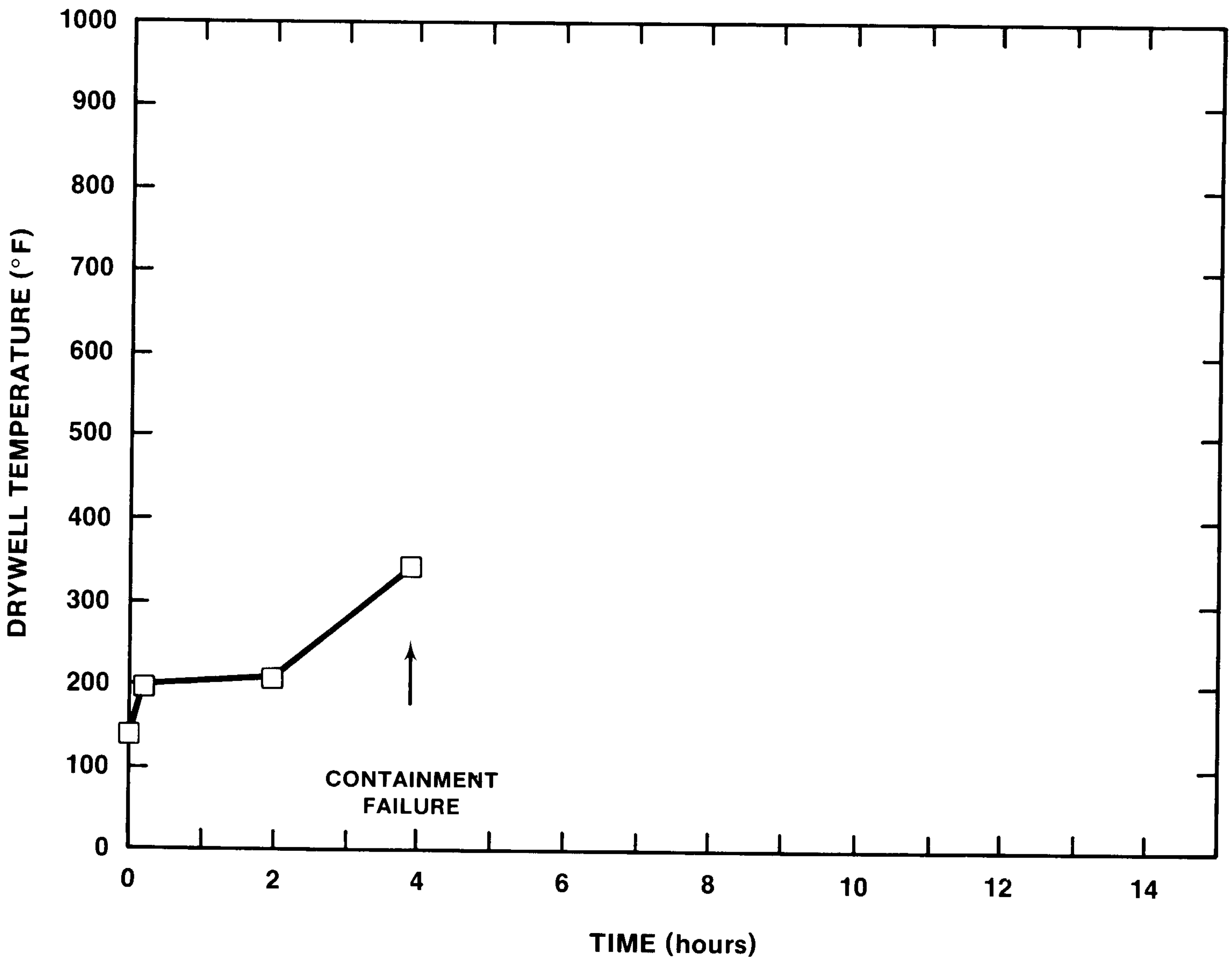


Figure 2. Drywell Temperature Profile for TC (MSIV Open)

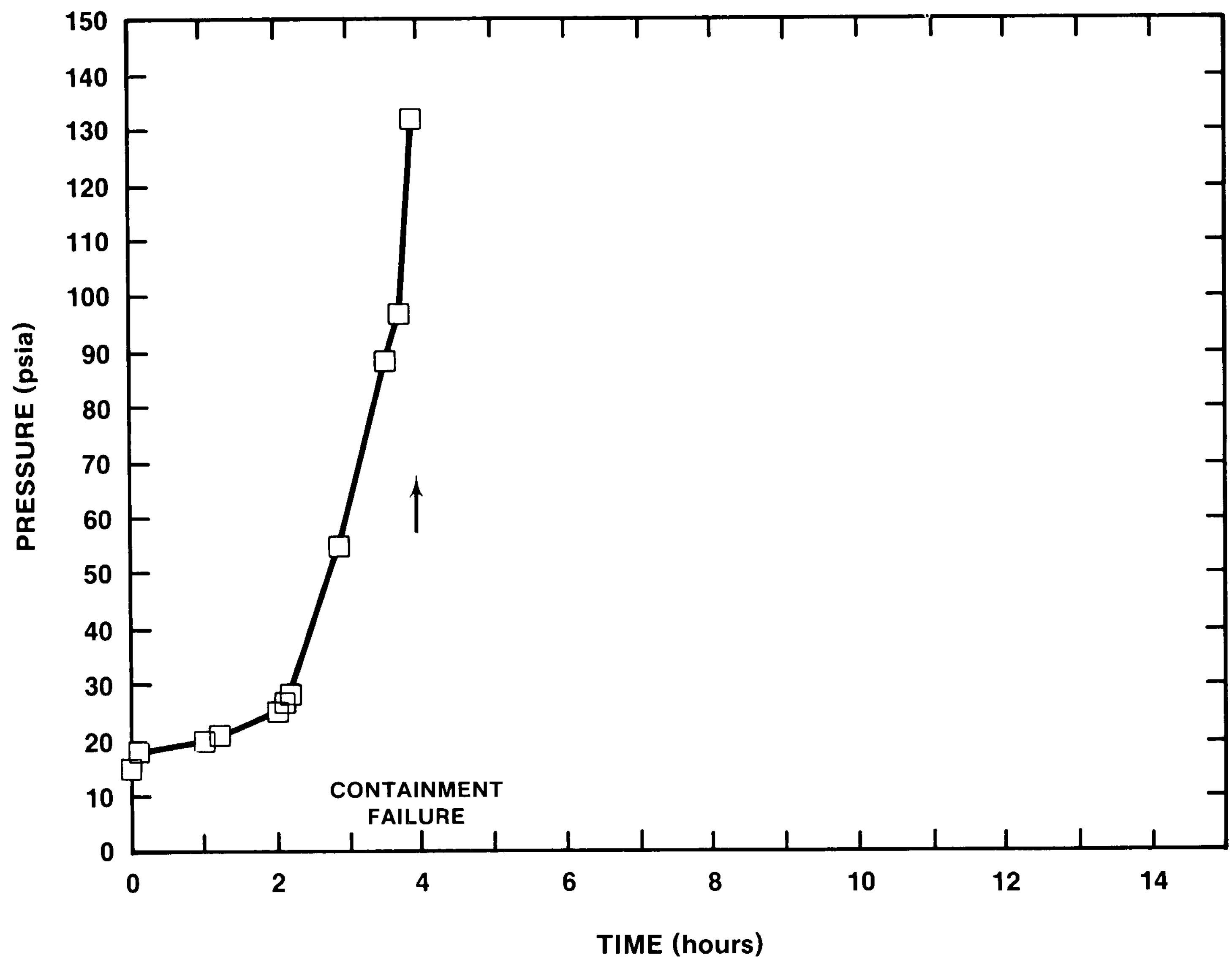


Figure 3. Containment Pressure Profile for TC (MSIV Open)

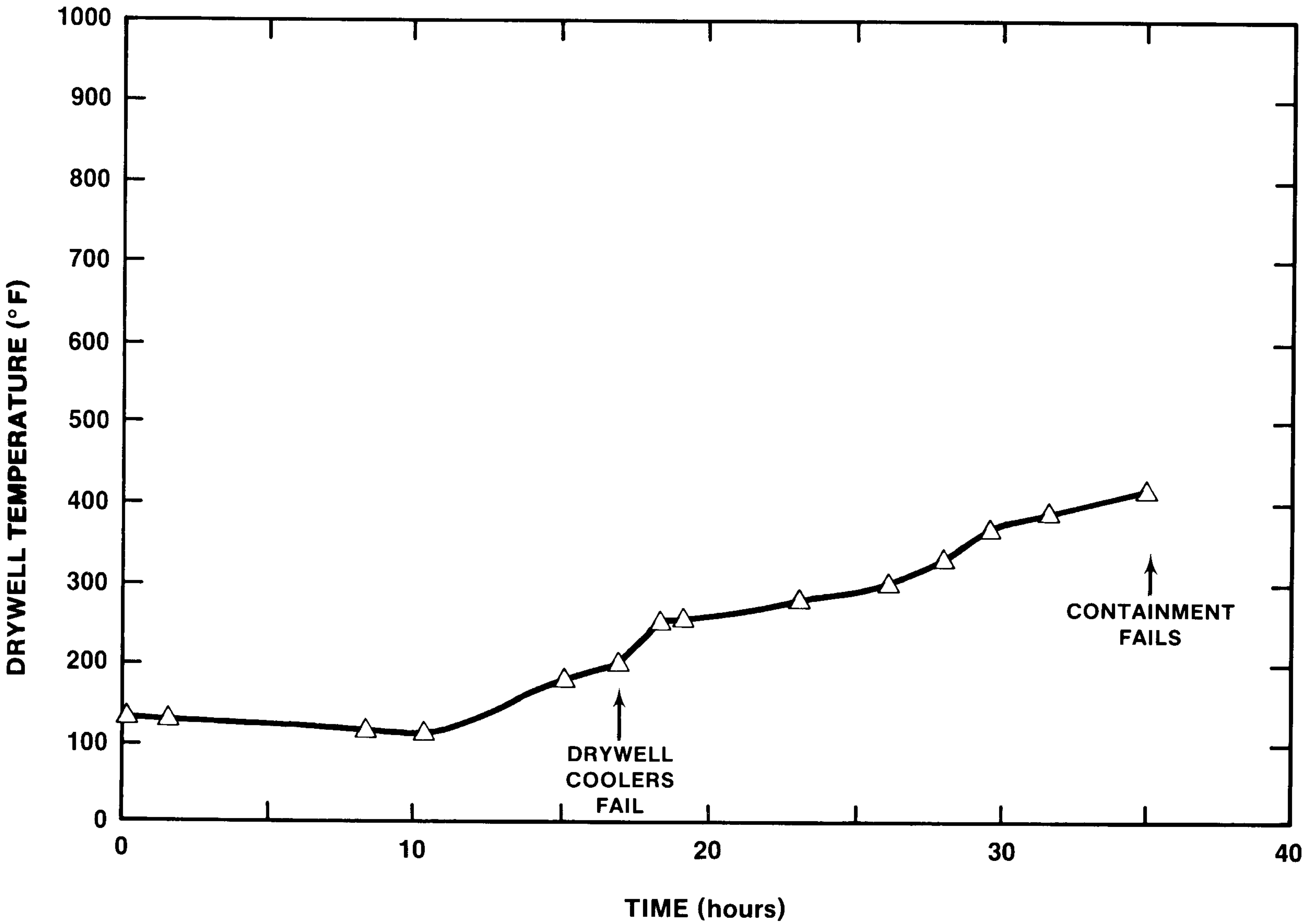


Figure 4. Drywell Temperature Profile for TW

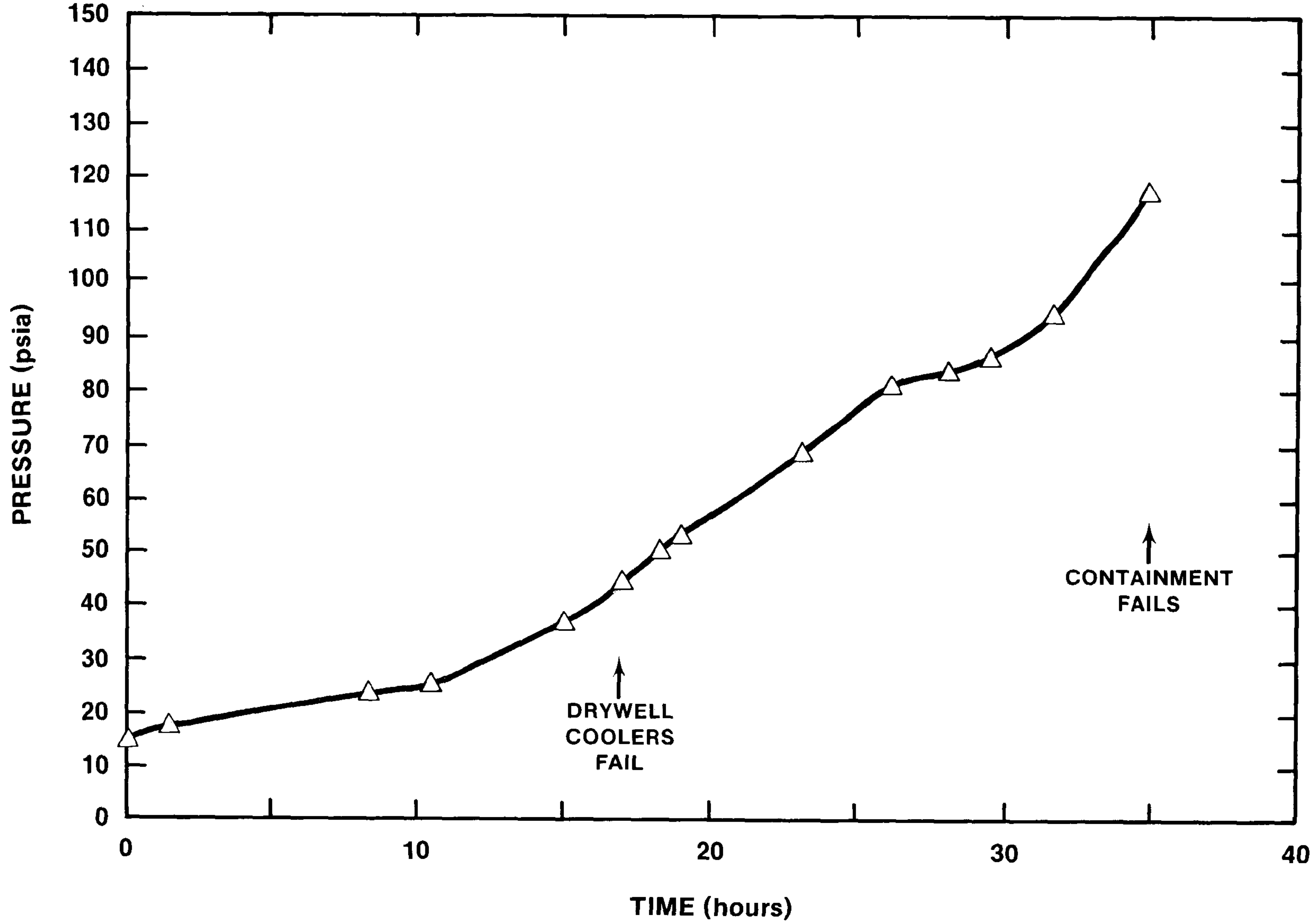


Figure 5. Containment Pressure Profile for TW

To determine the fragility level of the test specimen, the chamber temperature will be increased in 25°F increments (and held at that temperature until the valve has stabilized at the chamber temperature for ten minutes) and the pressure will be increased in 5 psig increments. The temperature will continue to be increased until the valve fails to open; however, the pressure will only be increased to a maximum pressure of 132 psig due to differential pressure requirements. The valve will be cycled open at each fragility plateau.

At the conclusion of the test, the valve will be closed (if necessary). The valve must close and remain closed, at that time, in order to perform its required safety function.

7.3 <u>Acceptance Criteria</u>

The test specimen must perform its required safety function throughout the accident exposure. The acceptance criteria is based on this operational performance.

<u>Test #1</u>

The valve must be maintained in the open position throughout the accident exposure. At the conclusion of the test, the valve must reclose and remain in the closed position.

<u>Test #2</u>

The valve will be closed and must be able to open upon demand during the accident exposure. At the conclusion of the test, the valve must reclose and remain in the closed position.

7.4 <u>Test Facilities</u>

The simultaneous aging exposure will take place in the High Intensity Adjustable Cobalt Array (HIACA). The accident simulation will be conducted using the steam system, at Sandia, which was designed to accommodate severe accident testing. Further details, regarding the test facilities, are given in Appendix E.

8. CONCLUSIONS AND INSIGHTS

8.1 Conclusions

In FY85, a method was devised to identify important electrical equipment and the severe accident environments in which the equipment was likely to fail. This method was used to evaluate the equipment and severe accident environments for Browns Ferry Unit 1, a BWR/Mark I. In addition, a test plan was written to experimentally determine the performance of one selected component to two severe accident environments.

Specifically, equipment was identified that was important to safety for a BWR--equipment which would mitigate severe accident sequences or provide plant status. For this list of equipment, only that equipment located in the primary containment or reactor vessel of Browns Ferry Unit 1 was analyzed further. For the five selected BWR severe accident sequences (TB, TC, TW, TQUV, and AE), environmental conditions within containment reached temperatures and pressures exceeding the current equipment qualification testing requirements prior to or during the time the equipment was needed. The results of this analysis suggest the need for testing the performance of the pneumatic control manifold assembly (part of the main steam isolation valve equipment assembly) during the TC and TW accident sequences.

8.2 Insights

As described in Section 8.1, the primary purpose of this project during FY85 and FY86 was to develop a test plan to evaluate the performance of electrical equipment in severe accident environments. Beyond this, a number of important insights were cited throughout this report in areas of accident management, emergency planning, probabilistic risk assessments, probability and risk reduction, and current equipment qualification requirements. These insights helped illustrate how the environmentally-induced failure of certain equipment during a severe accident may adversely impact the ability of a nuclear power plant to cope with severe accident conditions.

In summary, the insights were:

1. Potential environmentally-induced failures of electrical equipment, after equipment qualification limits are exceeded, may render the current Emergency Procedure Guidelines and operator training ineffective. (Appendix A)

2. Accident management and emergency planning procedures may need to reflect the effects on equipment operability of those accident conditions which are expected to exceed equipment qualification limits. (Section 5.2.4 and Appendix A)

3. Probabilistic risk assessments may not adequately address the effects of environmentally-induced equipment failures. (Section 5.2.4)

4. Depending on the results of equipment testing under severe accident conditions, current equipment qualification requirements may need to be reviewed for adequacy. (Section 4.4)

In general, the basis for these insights was:

1. Some electrical equipment, located in a typical BWR Mark I containment, may be needed to mitigate severe accident sequences or provide plant status. (See Table 3.)

2. During severe accident sequences, environmental conditions within containment may reach temperatures and pressures exceeding the current equipment qualification testing requirements prior to or during the time the equipment is needed. (See Table 4.)

3. A review of electrical equipment failure modes indicated that combinations of temperature with moisture, pressure with moisture, and temperature with radiation are the most likely environments to induce failure of electrical equipment.

In order to judge the safety importance of these insights, tests to confirm the actual survival limits of equipment during severe accidents need to be performed.

9. REFERENCES

1. Harrington, R. M. and L. J. Ott, The Effect of Small Capacity, High-Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One, NUREG/CR-3179, ORNL/TM-8635, September 1983.

2. Cook, D. H. et al., Loss of DHR Sequences at Browns Ferry Unit One Accident Sequence Analysis, NUREG/CR-2973, ORNL/TM-8532, May 1983.

3. Cook, D. H. et al., Station Blackout at Browns Ferry Unit One, Accident Sequence Analysis, NUREG/CR-2182, ORNL/NUREG/TM-455/VI, November 1981, Appendix A.

4. Harrington, R. M. (ORNL), S. A. Hodge (ORNL), ATWS at Browns Ferry Unit One--Accident Sequence Analysis, NUREG/CR-3470, ORNL/TM-8902, July 1984.

5. Gieseke, J. A. et al., Radionuclide Release Under Specific LWR Accident Conditions, BMI-2104 Volume II, July 1984.

6. Emergency Procedure Guidelines for Browns Ferry Unit One, June 1984.

7. BWR Emergency Procedure Guidelines, Revision 3.

8. Regulatory Guide 1.97, Revision 3.

9. Assessment of Generic Instrumentation Systems Used to Meet Provisions of Regulatory Guide 1.97, EGG-EE-6154, February 1983.

10. BWR Status Monitoring During Accident Conditions, NUREG/CR-2100, April 1981.

11. Light Water Reactor Engineered Safety Features Status Monitoring Final Report, NUREG/CR-2278, August 1981.

12. Fundamental Safety Parameter Set For Boiling Water Reactors, NSAC/21, December 1980.

13. Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, NUREG-0654, FEMA-REP-1, Rev. 1, November 1980.

14. Equipment Survivability in a Degraded Core Environment, NUS Corporation for IDCOR, Technical Report 17, July 1983.

15. Preliminary Interface Specification for NDL Data Acquisition System and NRC Terminal, NUREG/CR-2025, June 1982.

16. TMI Nuclear Power Plant Accident Hearings Before the Subcommittee on Nuclear Regulation of the Committee on Environment and Public Works, U.S. Senate, 96th Congress, First Session, October 2-3, 1979.

17. Station Blackout at Browns Ferry Unit One--Accident Sequence Analysis, NUREG/CR-2182, November 1981.

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

19. ATWS: A Reappraisal - Part 3; Frequency of Anticipated Transients, Science Applications International Corporation for EPRI, EPRI-NP2230, January 1982.

20. Evaluation of Station Blackout Accidents at Nuclear Power Plants, U.S. Nuclear Regulatory Commission, NUREG-1032, May 1985.

21. "Accident Sequence Evaluation Program (ASEP) Generic Data Base," Sandia National Laboratories, Draft.

22. Discussions with A. M. Kolaczkowski on the ongoing ASEP analysis of Peach Bottom - 2 (a BWR-4, Mark I) for NUREG-1150, Fall 1985.

23. Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, U.S. Nuclear Regulatory Commission and Federal Emergency Management Agency, NUREG-0654, FEMA-REP-1, Revision 1, November 1980.

Appendix A

LIKELY SCENARIOS FOR FIVE ACCIDENT SEQUENCES

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

1.0 OBJECTIVES

This appendix presents the results of the accident sequence selection process. Key events during accident sequences leading up to core damage or containment failure dictate what equipment has failed by definition and cannot be relied upon, and which systems have been assumed to succeed in certain accident sequences. In addition, the key events provide information that allows the derivation of the environmental profile for each accident sequence up to core melt or containment failure. These environmental conditions may affect the reliability or availability of equipment to mitigate the accident. This document addresses the key events of certain accident sequences of interest and provides a start for defining the needed environmental profiles.

---

*For some sequences, more than one "likely" scenario is identified since determination could not be made that one scenario was significantly more probable than another. In addition, some stuck-open valve scenarios are identified as worthy of examination for Performance Evaluation of Electrical Equipment during Severe Accident States (PEEESAS).

2.0 PLANT AND ACCIDENT SEQUENCES OF INTEREST

Following the lead of the Accident Management Program, Browns Ferry Unit 1 (a BWR-4, Mark I design) was selected as the first model plant of interest. This leads us to a discussion about which accident sequences are important for such a design and therefore worthy of review for addressing equipment survivability and accident management strategies.

A number of criteria exist for choosing the sequences of interest. In general, the sequence should:

(a) be among the most probable to occur,
(b) be potentially a high risk sequence,
(c) have a potential for extreme environments, and
(d) allow for "interesting" operator interaction potential for formulating accident management strategies.

Table A-1 summarizes information pertaining to the first two criteria for accident sequences either found to be dominant in past PRAs or of particular interest due to unique timing, environment, or other aspects. First, a qualitative comparison is provided regarding the relative frequencies of the candidate sequences based on actual precursors to these sequences as analyzed in NRC's Accident Sequence Precursor Program. Second, order of magnitude estimates are provided for the sequences as determined by reanalysis of certain plant PRAs by the Accident Sequence Evaluation Program (ASEP) and Industry Degraded Core Rulemaking (IDCOR) program; NRC and industry-sponsored programs, respectively. These two programs serve as major inputs on accident sequence frequencies for the study of severe accidents by NRC and industry. Third, a risk perspective is provided by both the Severe Accident Risk Reduction Program (SARRP) and IDCOR program; two primary programs for assessing the risks of accident sequences for the NRC and industry, respectively. (1,2,5,11,12)

Based on this information, TB, TW, and TC are generally found to be among the most likely and risk dominant sequences for BWRs in general, including the BWR4-Mark I design. In addition, the potential primary containment

Table A-1. Summary of BWR Accident Sequences Potentially Worthy of Review

ACCIDENT SEQUENCES	PRECURSOR STUDY (AMONG TOP 90% CONTRIBUTION TO CORE MELT)	CURRENT ASEP CORE MELT ESTIMATES			IDCOR COMMITTED CORE MELT FREQUENCY ESTIMATES		AMONG HIGHEST RISK PER IDCOR	PRELIMINARY DOMINANT CONTAINMENT FAILURE MODE PER SARRP
		PEACH BOTTOM	LIMERICK	GRAND GULF	PEACH BOTTOM	GRAND GULF		
TQUV - transient loss of core cooling	(TQUV and TB together) - yes	10^{-8}	10^{-7}	10^{-7}	10^{-8}	10^{-8}	No	Leak or late overpressure
TB (form of TQUV) station blackout	(TQUV and TB together) - yes	10^{-6}	10^{-6}	10^{-6}	10^{-7}	10^{-7}	Yes	Leak or late overpressure
TW (or TQW) transient-loss of long-term heat removal	Yes	10^{-7}	10^{-8}	10^{-7}	10^{-7}	10^{-7}	Yes	Core melt in failed containment
TC-ATWS	Yes	10^{-7}	10^{-8}	10^{-7}	10^{-6}	10^{-6}	Yes	Core melt in failed containment
TPQUV (or TPQE) like TQUV but with stuck-open relief valve	No	Not Assessed	10^{-8}	10^{-8}	?	$<10^{-8}$	No	--
TPW (or TPQI) like TW but with stuck-open relief valve	No	Not Assessed	Not Assessed	10^{-8}	?	10^{-8}	No	Core melt in failed containment
SI (including SHI and SJ) small LOCA and loss of long-term heat removal	No	Not Assessed	Not Assessed	10^{-7}	$<10^{-8}$	$<10^{-8}$	No	--
SE-Small LOCA and loss of core cooling	No	Not Assessed	Not Assessed	$<10^{-8}$	$<10^{-8}$	?	No	--
AI-Large LOCA and loss of long-term heat removal	No	Not Assessed	Not Assessed	Not Assessed	?	$<10^{-8}$	No	--
AE-Large LOCA and loss of core cooling	NO	10^{-8}	Not Assessed	Not Assessed	10^{-8}	$<10^{-8}$	No	Leak or late overpressure

environments for these sequences may be of interest due to such problems as loss of area cooling, high heat loads, etc. While TB has limited potential for operator ingenuity because of no AC power, TW and TC allow for considerable operator interaction. Clearly these sequences should be reviewed in the PEEESAS. TQUV is of moderate interest since the frequency estimates are still of some concern even though its risk potential is generally not as high as the previous three sequences. All the other sequences are generally not believed to be as important as the previously mentioned sequences. TPQE and TPQI-type sequences can be handled as variations of TQUV or TW. The LOCA sequences are of unique interest due to the rapid steam environments that can occur and can be handled by the limiting AE sequence in which the environment is established quickly with an early core melt.

In summary then, it is suggested that the TB, TW and TC sequences definitely be studied in the PEEESAS Program. The other sequences do not appear as important based on current knowledge. However, one must remember that the other accident sequences (those listed in Table A-1 and all other possible sequences) are not important, in part, due to the belief that the accident environment would not cause a significant increase in the sequence frequencies. It is not within the time and money constraints of this program to review every possible sequence. As a result, only two relatively nondominant sequences, TQUV and AE, will also be examined. There remains a possibility that environmental failures may cause a nondominant sequence, which was not selected, to become dominant.

While these five sequences are chosen for study, no conclusion is intended that these sequences are necessarily dominant sequences for Browns Ferry Unit 1. However, these are sequences of general interest for the study of BWRs and hence will be reviewed using the Browns Ferry plant as a model.

Sections 3.0 through 7.0 contain information impacting the definition of the more likely scenarios for each selected sequence. This information is used to define the best estimate or "likely" scenarios that will be used in later tasks to define the environmental profiles for these same sequences.

3.0 TB - STATION BLACKOUT (LOSS OF ALL AC)

3.1 SHORT-TERM SEQUENCE

- Initiated by a loss of all offsite power

- Functions* successful - Reactor subcriticality, RCS overpressure protection

  Functions* failed - Core and Containment Heat Removal, Containment Overpressure Protection, Radioactivity Removal**

- Systems successful - RPS, SRVs/ADS

- Systems failed - HPCI, RCIC, all AC power (each system failed due to hardware faults or test and maintenance unavailabilities)

- Severe Accident Sequence Analysis (SASA) program studies and the following information support certain likely scenario paths. Table 9.5 from NUREG/CR-2182 (ATTACHMENT 1) is representative of the sequence of events in such an accident for the "operator doesn't depressurize the RPV" case.

  - Most likely, some time before 625 sec., operators will have determined HPCI/RCIC loss due to attempted start of these systems per EPG-RC/L-2.*** Guideline says to then follow contingency #1.

  - At 10-15 min. when low-low reactor water level reached (-146 in), ADS timer will initiate.

  - Contingency #1 says to prevent ADS actuation until it is clear that coolant injection can't be restored; go to Contingency #2 and #4 when water level reaches - 164 in (top of active fuel).

---

*Function definitions from IREP Procedures Guide, NUREG/CR-2728. (10)
**Depending on the location and timing of containment failure, the suppression pool can perform radioactivity removal for awhile, but no active spraying is possible.
***Emergency Procedure Guidelines for Browns Ferry, June 1984. (9)

- Contingencies #2, #4 - both call for emergency depressurization with ADS/SRVs (will be needed at ∿ 30-40 min). [Concerns have been raised by the PEEESAS and Accident Management program staffs as to the "likeliness" of operators to depressurize when they know that no low pressure makeup is available due to no AC power. The depressurization process will shorten the time to core uncovery without core coolant makeup thus perhaps affecting this operator decision point.]

- Reactor vessel pressure will most likely be maintained at ∿ 100 psig by the operator (if he follows Contingencies # 2 and # 4) with manual operation of SRVs in accordance with standard BWR practices and as required so as to avoid various limits per EPGs - SP/T-4, SP/L-3, DW/T-3, SP/L-3.2, and PC/P-4 (these deal with pool temperature, pool level, drywell temperature, and suppression chamber pressure).

- When suppression chamber pressure exceeds the primary containment pressure limit, EPG-PC/P-7 calls for venting the primary containment (equipment to be later specified and a note is added that isolation interlocks may need to be defeated).

o Key operator actions up to the point of reactor vessel failure include:

  - Operator emergency depressurization of reactor pressure vessel (RPV) with ADS/SRVs

  - Operator maintaining low RPV pressure (∿ 100 psig) with SRVs

  - Restore AC power and subsequent core and containment cooling systems

o A variation of this sequence, though not as likely, includes the addition of a stuck-open relief valve. Since depressurization of the RPV is thus performed early, the pressure and temperature profiles may be significantly altered and thus require additional analysis. In this case, Reactor Coolant System (RCS) Overpressure Control would have failed. In addition, the operator would not need to depressurize the vessel and maintain a low RPV pressure since this would already occur because of the stuck-open valve.

- "Likely" core-damage scenarios, therefore, appear to be as shown in Figure A-1 based on the above information. Environmental profiles should be established on the basis of these scenarios.

3.2 LONG-TERM SEQUENCE

- Initiated by a loss of all offsite power

- Functions successful - Reactor subcriticality, RCS overpressure protection, Core heat removal (initially)

  Functions failed (if blackout continues) - Core heat removal (eventually), Containment heat removal, Containment overpressure protection, Radioactivity removal

- Systems successful - RPS, SRVs/ADS (temporarily)
  RCIC/HPCI (temporarily)

  Systems failed - all AC (hardware/T&M)
  DC (battery depletion in ∿ 4-7 hours)
  HPCI/RCIC (due to no DC power at 4-7 hours)
  Manual SRVs/ADS (due to no DC power at 4-7 hours)

- SASA program studies and the following information support certain likely scenario paths. Table 9.3 from NUREG/CR-2182 (ATTACHMENT 2) is representative of the sequence of events in such an accident. (See "likely" scenario in Figure A-2). (7)

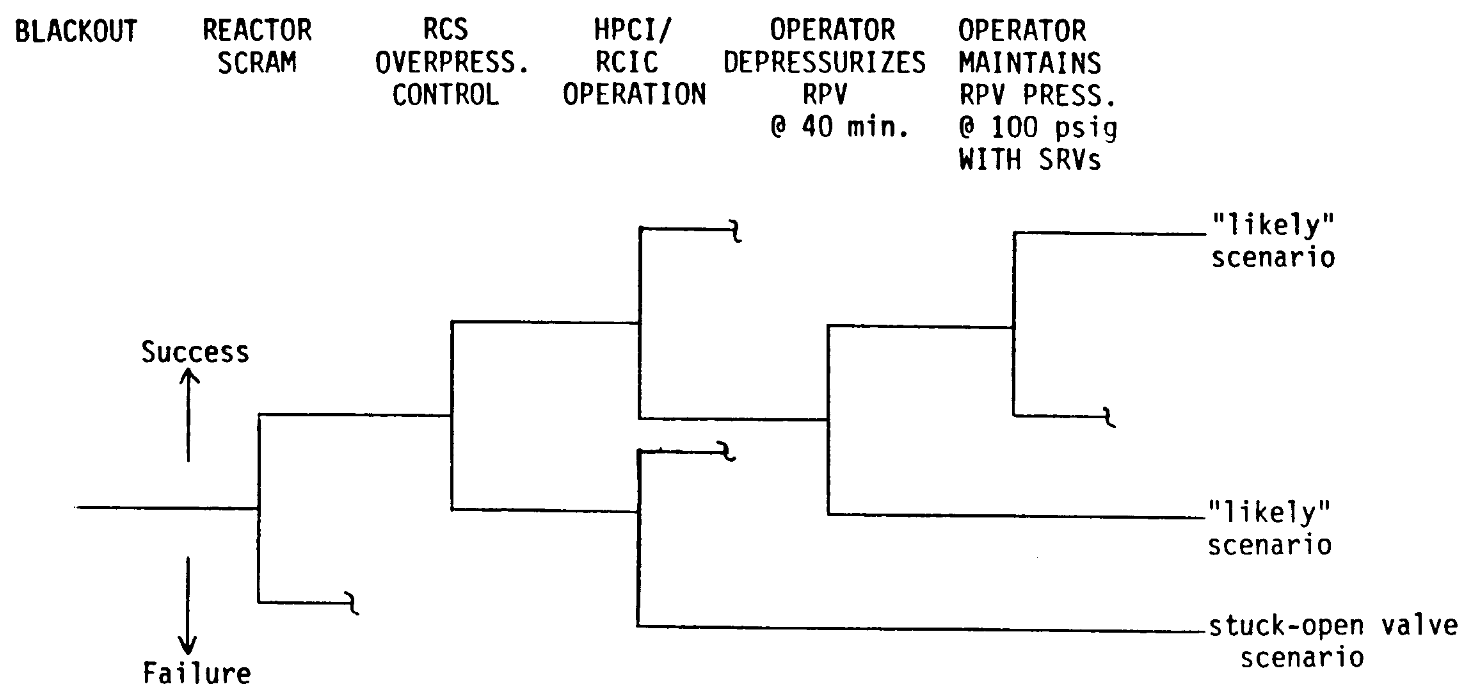


Figure A-1. "Likely" Short-Term TB Scenarios
(up to reactor vessel failure)

- Manual opening of a SRV at 15 min is part of proceeding to cold shutdown per EPG-RC/L-3 particularly with satisfactory core coolant makeup taking place.

- At about the same time, a suppression pool temperature limit requires RPV depressurization anyway per EPG-SP/T-4.

- Reactor vessel pressure will most likely be maintained at ∿ 100 psig by the operator with manual operation of SRVs in accordance with standard BWR practices and as required per EPGs - SP/L-3, SP/L-3.2, DW/T-3, PC/P-4.

- Possible exception to Table 9.3 from NUREG/CR-2182; HPCI would initiate at ∿ 15-20 min at level 2 to restore RPV level thus avoiding temporary core uncovery at 21 min. Such action should also prevent any auto ADS signal.

- When DC power depletes at ∿ 4 hours, it is assumed to terminate RCIC and HPCI operation. Also, manipulation of SRVs is no longer possible, hence repressurization of the RPV takes place.

o Key operator actions up to the point of reactor vessel failure appear to be:

  - Operator controls RPV level with RCIC (or HPCI if necessary)

  - Operator performs controlled depressurization

  - Operator maintains low RPV pressure (∿ 100 psig) with SRVs

  - Restore AC power and subsequent systems

  - Eliminate unnecessary DC loads to make DC power last as long as possible.

- A less likely but perhaps an important variation of this sequence includes the addition of a stuck-open relief valve. Although depressurization of the RPV is performed within $\sim$ 15 min anyways by the operator, additional analysis of the stuck-open valve case is warranted since it would prevent repressurization of the RPV when DC power fails. Note that RCS Overpressure Control would have failed and the operator would not need to purposefully depressurize the vessel or maintain a low pressure. Depending on the pressure level achieved, HPCI operation may be precluded due to a low steam pressure setpoint.

- "Likely" core-damage scenarios, therefore, appear to follow those shown in Figure A-2 up to reactor vessel failure. Environmental profiles should be established on the basis of these scenarios.

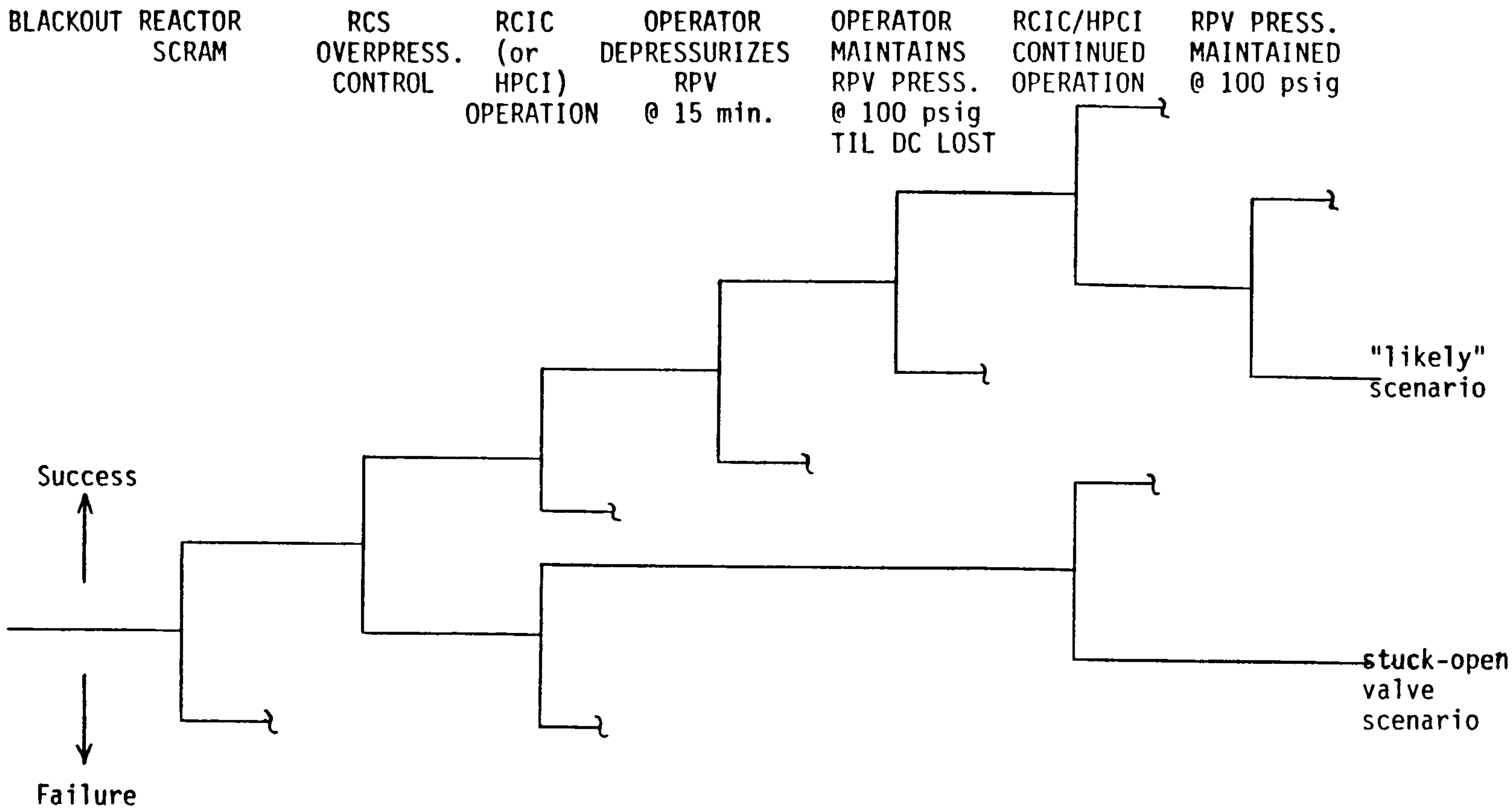


Figure A-2. "Likely" Long-Term TB Scenario
(up to reactor vessel failure)

4.0 TW - TRANSIENT WITH LOSS OF LONG-TERM HEAT REMOVAL

- o Initiated by a variety of transients ultimately causing loss of the PCS (MSIV closure)

- o Functions successful - Reactor subcriticality, RCS overpressure protection, Core heat removal

- o Functions failed - Containment heat removal, Containment overpressure protection,* Radioactivity removal*

- o Systems successful - All but RHR and/or service water for RHR (or diesels for loss of offsite power initiator). Some systems are eventually isolated or otherwise potentially made unavailable (see sequence of events and accompanying notes).

- o Systems failed - RHR and/or service water for RHR (or diesels for loss of offsite power initiator) due to hardware faults, T&M. Also PCS is not restored.

- o SASA program studies and the following information support certain likely scenario paths. Table 3.1 from NUREG/CR-2973 (ATTACHEMENT 3) is representative of some of the key events in this accident sequence.(6)

  - Initial RPV level control via manual RCIC per EPG-RC/L-2.

  - Attempt to restore the PCS and maintain RPV level and pressure control with the PCS per EPGs-RC/L-2 and RC/P-1,2.

  - With PCS not restored, high drywell pressure scram point at ∿ 1 hour can't be reset so CRD pump flow increases; CRD system can handle level control at ∿ 4 hours; at ∿ 8.6 hours must throttle CRD or run intermittently to avoid flood of RPV. Note that oper-

*Depending on the particular equipment failures ator action is required to restart CRD on an emergency bus if the initiator is loss of offsite power.

- Operators valving in station control air at 1 hour is to comply with guidelines to operate drywell cooling per EPG-DW/T-1. This also affects air supply to SRVs.

- Controlled depressurization at 1 hour and maintaining low pressure is per guidelines provided by EPGs-RC/L-2, SP/T-4, SP/L-3, SP/L-3.2, DW/T-3, PC/P-4.

- NUREG/CR-2973 raises questions as to whether emergency procedures cover restart of drywell coolers at 2 hours for loss of offsite power case resulting in two possible scenarios: (a) drywell coolers restarted and run until fail (assumed at 17 hours) (b) drywell coolers not restarted. PEEESAS and Accident Management Program technical staffs raised the fact that operators are usually very aware of drywell cooler operation and try to maintain it (based on training material and simulator observations). Besides, offsite power case is relatively infrequent compared with transients due to other causes.ε

- HPCI switches irreversibly to suppression pool suction at ∿ 3 hours when pool level exceeds + 7 inches. Hot pool water could fail system.

- RCIC should continue to use the condensate tank for suction thus avoiding early failure due to hot suppression pool water unless operator switches this system too. However, if RCIC and HPCI should eventually fail, CRD system can handle makeup requirements.

- Note HPCI/RCIC isolate automatically at ∿ 13 hours per NUREG/CR-2973 material. Not crucial if CRD, SLC, or any low pressure injection systems are available.

- For case when drywell coolers continue to run (most transients) or are restarted at 2 hours after potential trip (loss of offsite power), NPSH on RHR Pumps (if available) may be such that operator throttling of RHR flow into the pool (to avoid pool thermal stratification even if service water is failed) must be performed to avoid possible pump damage or motor failure (see caution #8 of EPGs). Note that no EPGs suggest using RHR to avoid pool thermal stratification.

- Per EPG-DW/T-3, operator should initiate primary containment sprays (if available) using a somewhat difficult procedure (see App A.3 of NUREG/CR-2973) when drywell pressure reaches $\sim$ 20 psig at $\sim$ 10-15 hours. The time difference depends on whether the drywell coolers were restarted, if necessary, at 2 hours. RHR pumps provide containment spray and so the same NPSH warnings mentioned above apply here.

- At $\sim$ 24 hours, drywell pressure exceeds 65 psig (unless the step below is taken) and the SRVs are probably no longer available for manual operation since they need + 25 psig air pressure above drywell pressure and the maximum air pressure is $\sim$ 90 psig. RPV repressurization therefore takes place.

- Venting of containment through the Standby Gas Treatment System (see EPG-PC/P-7) or feed and bleed of containment are offered as possible mitigating operations in NUREG/CR-2973 (see Section 4.3 of the NUREG).

o Key operator actions up to the point of containment failure include:

  - Operator controls RPV level with RCIC, CRD, etc.

  - Recovery of PCS.

  - Operator performs controlled depressurization and maintains low pressure as long as possible.

- Operator restarts drywell coolers at 1 and 2 hours as required.

- Operator assures long term core cooling in likely event that RCIC/HPCI eventually become unavailable.

- Operator throttles RHR pumps if being used.

- Operator initiates containment sprays (if available) at $\sim$ 10-15 hours.

- Operator vents containment to avoid failure.

o A less likely but perhaps important variation of this sequence includes the addition of a stuck-open relief valve. Primary effects of such a scenario include quicker depressurization early in the sequence and the fact that repressurization of the RPV will not occur after loss of SRV control at $\sim$ 24 hours due to the stuck-open valve. Overall timing of containment failure is not significantly affected. However, environmental profiles should be reviewed for this variation.

o The "likely" core-damage scenarios, on the basis of the above information and the likely failure modes of containment cooling appear to be as shown in Figure A-3 up to containment failure. Containment failure is considered to likely occur before core damage. Environmental profiles should be established on the basis of these scenarios.

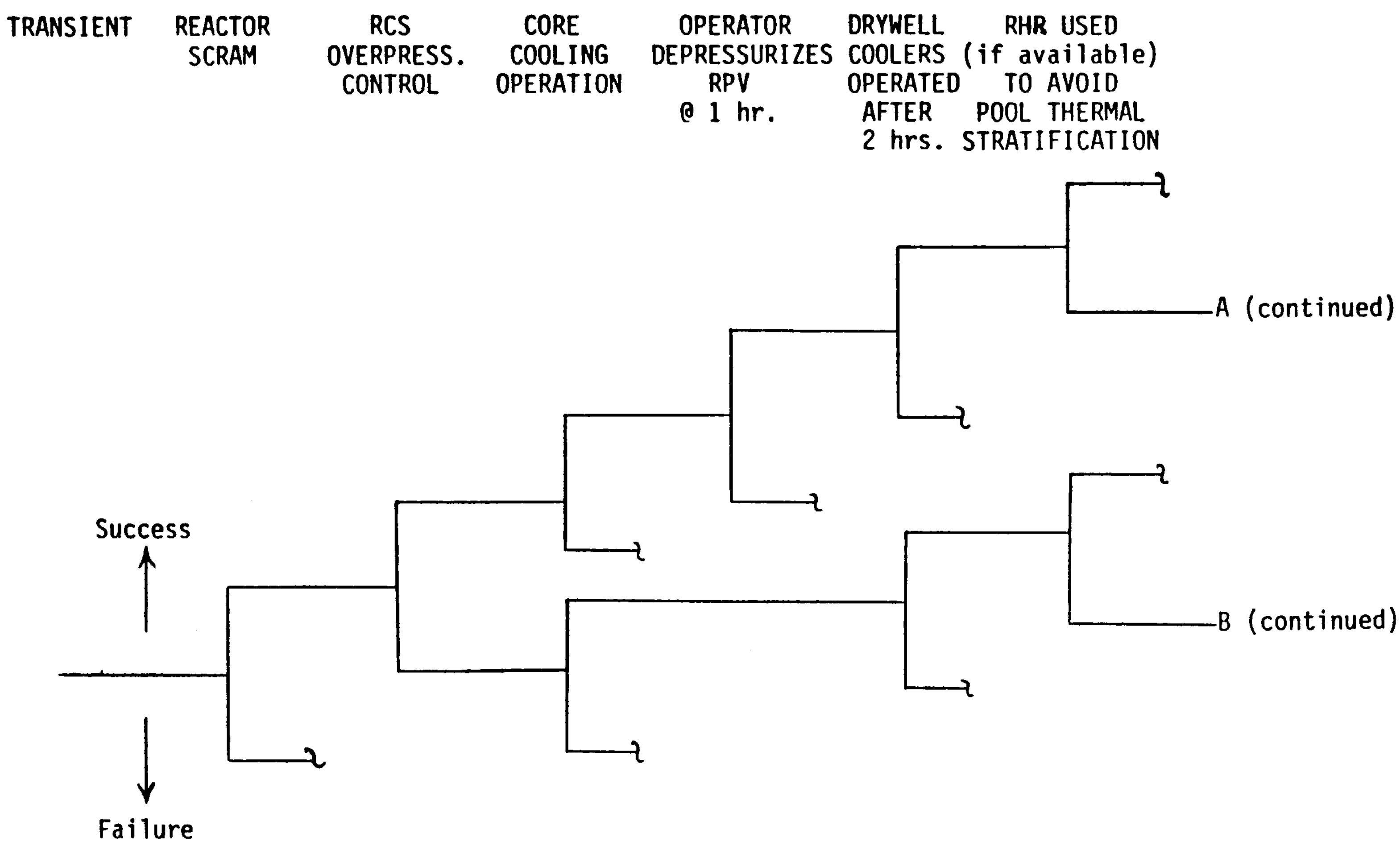


Figure A-3. "Likely" Scenario for TW
(up to potential containment failure)

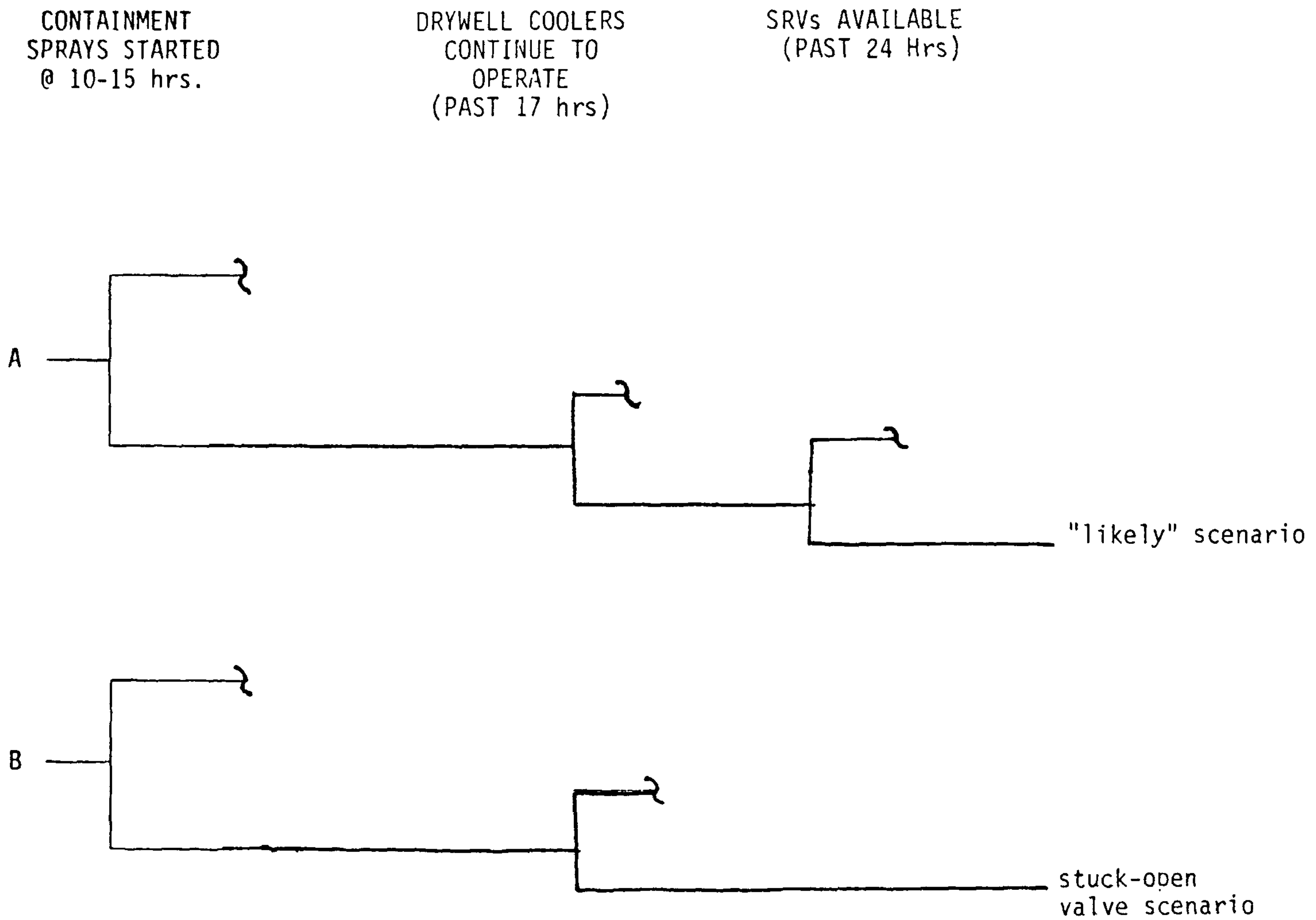


Figure A-3. (continued)

5.0 TC - ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

- Initiated by a variety of transients (most likely other than due to loss of offsite power) with main steam isolation valve (MSIV) closure causing loss of the PCS.

- Functions successful - RCS overpressure protection, Core heat removal

- Functions failed - Reactor subcriticality, Containment heat removal,* Containment overpressure protection,* Radioactivity removal**

- Systems successful - All but the RPS and related scram capabilities. Some systems may be eventually isolated or otherwise made unavailable, such as the drywell coolers, based on possible sequences of events. Although RHR (particularly for suppression pool cooling) is functioning, its adequacy may not be sufficient thus causing failure of the containment heat removal/overpressure functions. This is because RHR can remove typically the equivalent of $\sim$ 5% power while the actual heat to be removed could be higher depending on the degree of RPV level control and whether or not power fluctuations occur.

- Systems failed - RPS and related scram capabilities (e.g. manual scram, deenergizing scram buses, venting air from the scram pilot valve operators, the Alternate Rod Insertion (ARI) system which has been added in some plants, etc.). These systems fail most likely due to electrical hardware faults or there exists mechanical failures associated with control rod insertion.

---

*See discussion about RHR for "Systems Successful"

**Radioactivity removal may or may not be possible by the suppression pool and/or RHR spray following core melt, depending on the location of containment failure (which most likely occurs before core melt) and the survivability of RHR following containment failure.

- SASA program studies and the following information support certain likely scenario paths. Table 4.5 from NUREG/CR-3470 (ATTACHMENT 4) is representative of some of the key events in this sequence. (8)

  - Initial attempts to manually scram are performed per EPG-RC-1. If it is determined that boron injection is required, Contingency #7 should be followed.

  - EPG-RC/Q-4 requires boron injection (using the Standby Liquid Control-SLC-system) if the reactor is not shutdown and the suppression pool temperature exceeds 110^{o}F (will happen in about 2 minutes). ADS initiation is to be prevented. [PEEESAS and Accident Management Program technical staffs have noted that while the initiation criteria for SLC is now clearer, reasons for postponing its use may exist. These include reluctance to borate the core of a BWR and at least in some plants, the possible need to obtain a certain level of management approval before initiation of SLC. Hopes that manual rod insertion or manual scram might successfully occur, also affect this operator decision point. Note that timely SLC operation would result in mitigation of the sequence and hence only its failure, such that the TC sequence results, is of concern here. SLC failure could be due to untimely initiation, hardware faults, or the fact that some questions still exist as to the phenomenological aspects of whether the method of boron addition to the core will indeed assure successful shutdown of the core.]

  - EPG-RC/Q-4.2 calls for continued boron injection until 280 lbs (cold shutdown boron weight) of boron have been injected into the RPV (will take about 20 minutes).

  - EPG-RC/Q-5 provides guidelines for a number of ways to get control rods into the core. For the purpose of studying the TC sequence, these ways are assumed to fail. The final approach is to manually drive in the rods noting that Rod Sequence Control System (RSCS) interlocks may need to be bypassed. Depending on the rods selected and ease of bypassing the RSCS, 20 minutes to 2 1/2 hours may be required to achieve sufficient subcriticality. This action would require a dedicated operator to hold the rod controls in place while driving in the rods. Note that timely manual rod insertion would result in successful mitigation of the sequence.

- If the PCS can not be rapidly restored to control RPV level and pressure per EPGs-RC/L-2 and RC/P-1,2. Contingency #7 calls for lowering of the RPV water level by terminating all injection except the boron systems and the CRD system (to lessen the amount of moderator and increase voiding in the core thereby lowering the power level). Level is to be controlled at the top of the active fuel which is believed to result in sufficiently low power levels although some questions may still exist due to modeling and code uncertainties. NUREG/CR-3470 notes a potential difficulty with water level readings. Two sets of level instruments can be used at Browns Ferry. The first, called Emergency Equipment, is calibrated at normal operating conditions and read down to 13 inches above the active fuel. The second, called Post Accident Flooding, is calibrated for atmospheric pressure and read down to the core midplane. Due to calibration differences, both can read significantly different from each other and from the actual level (depending on operating conditions) as illustrated in Figure A-4.

- Contingency #7, step C7-2 refers to the following cautions: (a) if high suppression pool level or low condensate tank level exists, confirm auto transfer or manually transfer HPCI and RCIC suction to the suppression pool (high level will be reached in $\sim$ 10 minutes) and (b) prevent maximum injection of water from LPCI/LPCS to avoid large power excursions.

- Contingency #7, step C7-3 calls for restoration of RPV level once sufficient boron has been injected or the control rods have been successfully used.

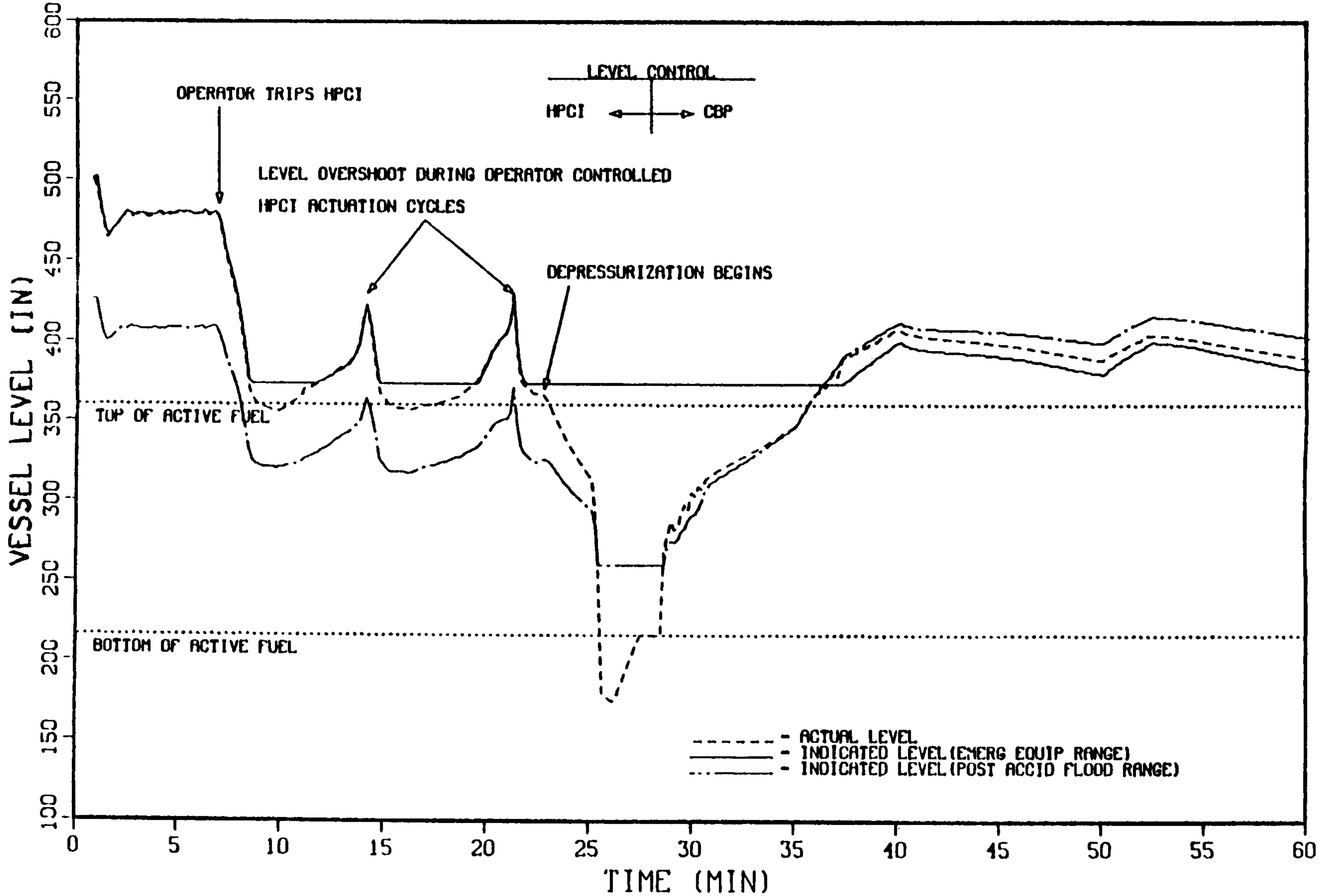


Figure A-4. Example of Level Instrument Discrepancies

- EPG-RC/P-1 calls for control of RPV pressure at $\sim$ 900 psig if any SRV is cycling, which will be the case early in this sequence. EPG-SP/T-4 further calls for controlled depressurization of the RPV (eventually down to 100-200 psig range) to avoid suppression pool heat capacity temperature limits. NUREG/CR-3470 notes potential difficulties with these tasks since no SRV position indication exists adjacent to the SRV switches at Browns Ferry. Without knowing which SRVs are already open due to auto control of RPV pressure, manual opening of SRVs may have no effect (the operator might open an already open SRV or opening of one SRV could be compensated by closure of another SRV not manually opened) until a sufficient number of SRVs are manually opened. Then a rapid depressurization could occur inviting the chances for low pressure system injection thus causing potential power, level, and pressure spikes. Thus, quick shutoff of these low pressure systems might be required.

- EPG-SP/T-2 calls for initiation of suppression pool cooling when the pool temperature exceeds 95^{o}F, which will occur within a few minutes. NUREG/CR-3470 points out potential difficulties with operating RHR in the suppression pool cooling mode. The LPCI mode of the RHR will auto initiate on low-low water level or high drywell pressure and low reactor pressure (< 465 psia). When the operator lowers the water level to the top of the active fuel, LPCI initiation will take place (disrupting suppression pool cooling). If RPV pressure should lower to $\sim$ 350 psia, water injection into the core will take place unless the operator has temporarily shutoff the RHR pumps. The LPCI injection valves are interlocked to full-open for 5 minutes before the suppression pool cooling mode can be restored. Further fluctuations in water level could cause further shifts from the suppression pool cooling mode to the LPCI mode with potential for water injection into the core; causing further power, pressure, and level fluctuations. These fluctuations, if severe enough, could cause a LOCA of the primary system.

Possible bypass capabilities do exist (though not in the procedures) to prevent subsequent shifts to the LPCI mode. However, even then, the LPCI injection valves are still not affected and will remain open for their 5 minute interlock periods. In such cases, the potential exists for RHR flow to be partially diverted from the suppression pool cooling mode to the reactor core through the open injection valves if RPV pressure drops below $\sim$ 350 psia.

- When HPCI suction switches to the suppression pool at a high pool level setpoint in $\sim$ 10 minutes, HPCI could fail due to high pool temperatures in the range of 175^{o}F to > 200^{o}F. Even if RCIC is not also manually switched over per procedures (possibly also causing its failure), RCIC and CRD injection may not be sufficient to maintain RPV level above the active fuel. Temporary core uncovery is expected until low pressure system flow is also initiated. In the meantime, steam cooling is believed sufficient to avoid any significant core damage.

- Auto initiation of the ADS timer (a 2 minute timer which, if not reset, causes auto ADS) could be a constant diversion of operator's attention since it starts on low-low level, which will occur when the RPV level is dropped by the operator.

- Drywell coolers should remain on during the entire sequence until possible failure if the drywell temperature should exceed > 200^{o}F.

- If core cooling is kept under reasonable control but the suppression pool cooling function is not being adequately provided due to temporary or sustained high core power levels, the sequence takes on the characteristics of the TW sequence previously described. In such a scenario, concerns exist for continued drywell cooling capability and manual SRV capability as already discussed for the TW sequence unless venting of the containment (see next item) occurs. If these systems fail, repressurization of the RPV can occur.

- Note that many operators may be needed to concurrently perform all of the above tasks.

- A variation of this sequence includes the addition of one or more stuck-open relief valves. Stuck-open valves would cause some depressurization of the RPV early in the sequence when water level is dropped. HPCI isolation might also occur on low steam pressure. The RPV would also not tend to repressurize to as high a pressure following loss of manual SRV operability.

- Another variation of this sequence includes an ATWS with the MSIVs remaining open and continued feedwater flow. In such a scenario, much of the heat energy generated by the core ($\sim$30% of full power for such a scenario since little level control is called for in the EPGs under these conditions) is directed to the PCS through the open MSIVs. With typical turbine-generator bypass capabilities of $\sim$25%, the remainder of the generated heat energy or about 5%, is directed to the suppression pool via the SRVs. With the RHR having a typical capability of removing 3% to 5% of the core's total heat energy, a rise in containment temperature and pressure is still quite possible. Such a scenario could lead to containment failure, thereby challenging the continued success of core cooling.

- The "likely" TC sequences, on the basis of the above information, appear to be as shown in Figure A-5 up to the point of imminent containment failure that is likely to occur before core damage. Without negative reactivity insertion, containment failure appears likely despite the best attempts to try and control water level and sustain pool cooling, which are, of themselves, difficult to perform. Environmental profiles should be established on the basis of these scenarios.

- Venting of the containment (see EPG-PC/P-7) might prevent catastrophic failure of the containment should it be imminent. Such a catastrophic failure could fail core cooling, thus causing a core melt after the containment failure.

o Key operator actions up to the point of containment failure include:

  - Attempt to manually scram the reactor or otherwise cause induced-scram conditions.

  - Manually control RPV pressure using SRVs to avoid SRV cycling.

  - Control RPV water level contending with the difficulties expressed earlier.

  - Defeat auto ADS.

  - Initiate early and provide sustained pool cooling while contending with the difficulties expressed earlier.

  - Initiate manual rod insertion and SLC early in the sequence.

  - Perform controlled depressurization of RPV if the pool heat capacity limit is reached.

  - Consider whether HPCI/RCIC suction switch to the suppression pool is a correct step depending on pool conditions.

  - Restore RPV level after subcriticality is achieved via rod insertion or sufficient boron injection.

  - Restoration of PCS.

  - Operator vents containment to avoid catastrophic failure, if necessary.

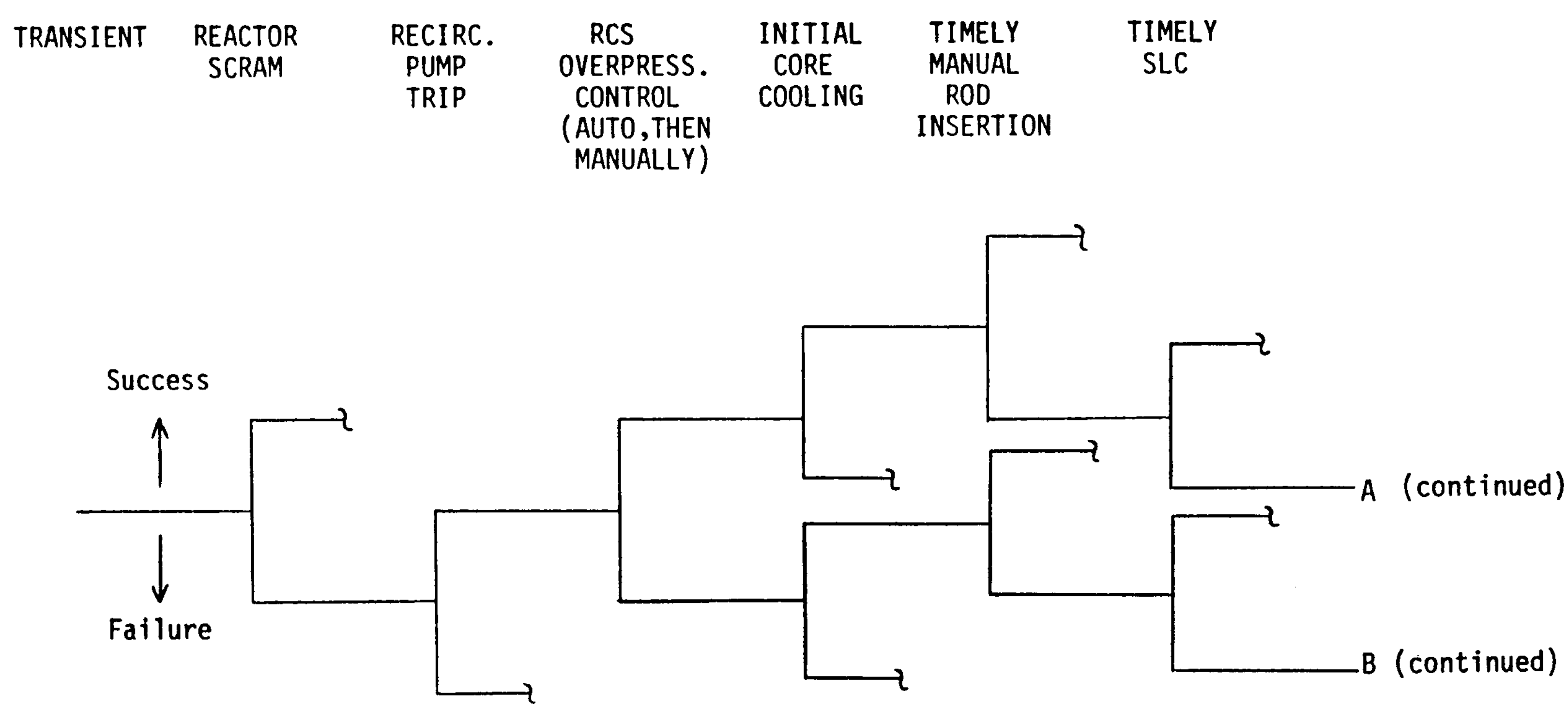


Figure A-5. "Likely" Scenarios for TC
(up to potential containment failure)

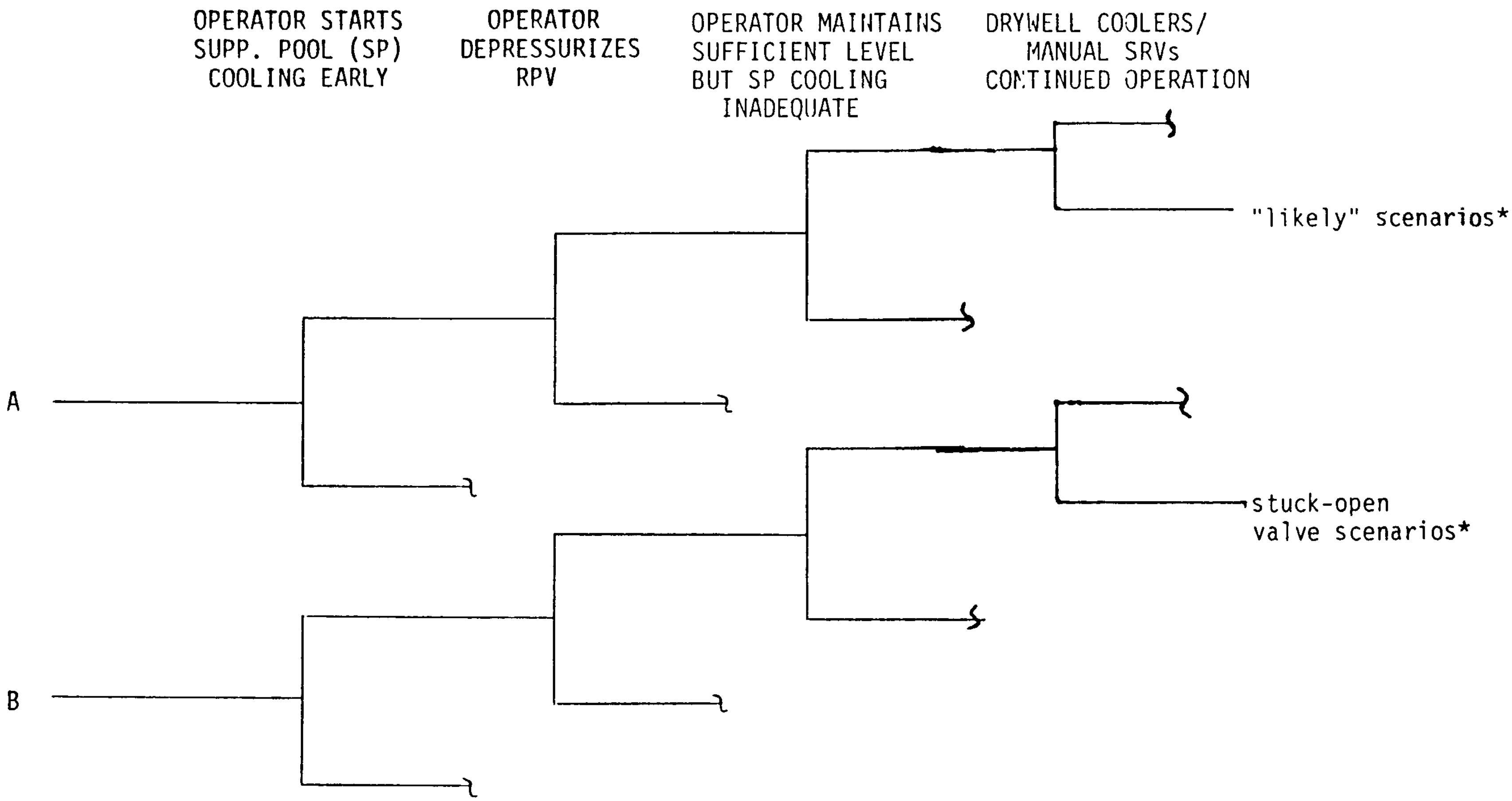


Figure A-5. (continued)

*The "MSIV-close" and "MSIV-open" cases follow vitually the same paths. For the "MSIV-open" case, however, the "Operator Depressurizes RPV" may not be required depending on the depressurization effect of the open MSIVs. While both cases are similar using this pictorial approach, the "MSIV-open" case challenges the containment at a slower rate than the "MSIV-close" case because of the availability of the 25% bypass capability of the PCS in the "MSIV-open" ATWS.

6.0 TQUV - TRANSIENT WITH EARLY LOSS OF CORE COOLING

- Initiated by a variety of transients causing loss of feedwater and failure of both high and low pressure core cooling.

- Functions successful - Reactor subcriticality, RCS overpressure protection

- Functions failed - Core heat removal, Containment heat removal,* Containment overpressure protection,* Radioactivity removal*

- Systems successful - All but HPCI, RCIC, LPCS, LPCI, and possibly CRD, and SLC.

- Systems failed - HPCI, RCIC, LPCS, and LPCI due to hardware/T&M faults or support system faults (power, service water). CRD and SLC (nonboration mode) may be failed due to hardware/T&M faults, or support system faults. Otherwise, these systems are unsuccessful due to operator error to (a) restore and/or increase flow from the CRD system and (b) initiate SLC flow to add to the injection flow. Recovery by using firewater or other systems is also not successful in time to prevent core damage.

- SASA program studies and the following information support certain likely scenario paths resulting in the more "likely" core-damage scenarios shown in Figure A-6. Environmental profiles should be established on the basis of these scenarios.

  - Same key events spelled out for the TB-short term sequence.

*Depending on the particular equipment failures.

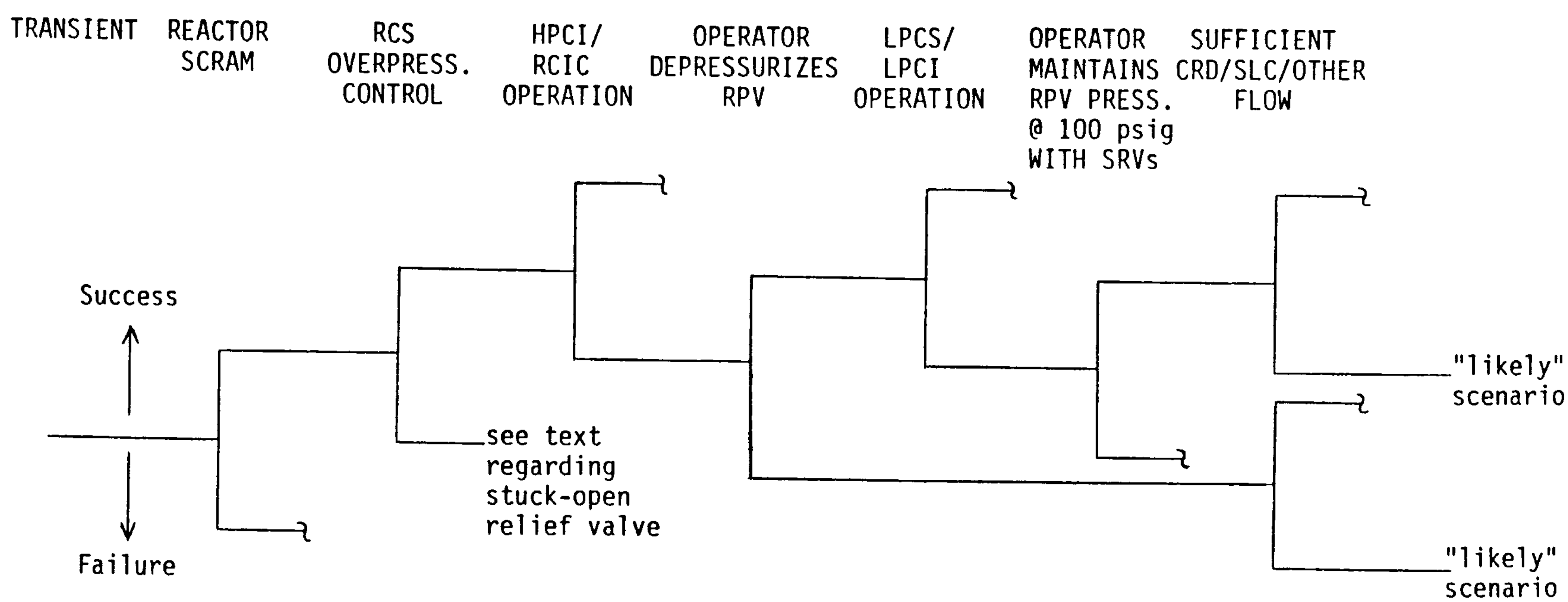


Figure A-6. "Likely" Scenarios for TQUV
(up to reactor vessel failure)

- For either the RPV depressurized or high pressure case, combinations of increased CRD flow or CRD/SLC flow (requiring operator action) are probably required to prevent significant core damage as per Figures A-7 and A-8 (from NUREG/CR-3179). Note that attempts to increase CRD flow could cause trips of the CRD pumps due to low suction pressure. This could add some difficulty to achieving the desired CRD flow.(3)

- SLC initiation in the nonboron mode requires local valve manipulation before initiation.

- Contingency #6 is called for if RPV flooding is required (which it will be for this sequence). These instructions highlight many systems that can be used to restore water level so that chances of operator nonrecovery must be considered to be low unless these systems share a common failure mode(s).

- If RHR is failed, normal containment heat removal and overpressure protection would also be failed. Active spraying for radioactivity removal would also be unavailable.

o Key operator actions up to the point of reactor vessel failure:

  - Operator depressurization of the RPV with ADS/SRVs.

  - Operator maintaining low RPV pressure ($\sim$ 100 psig) with SRVs.

  - Operator initiation of sufficient CRD/SLC/other system injection.

  - Restoration of feedwater/PCS.

o A variation of this sequence includes the addition of a stuck-open relief valve. With probable early depressurization by the operator of the RPV as a likely scenario, this variation is similar to that likely scenario.

From NUREG/CR-3179 (Reference 3)

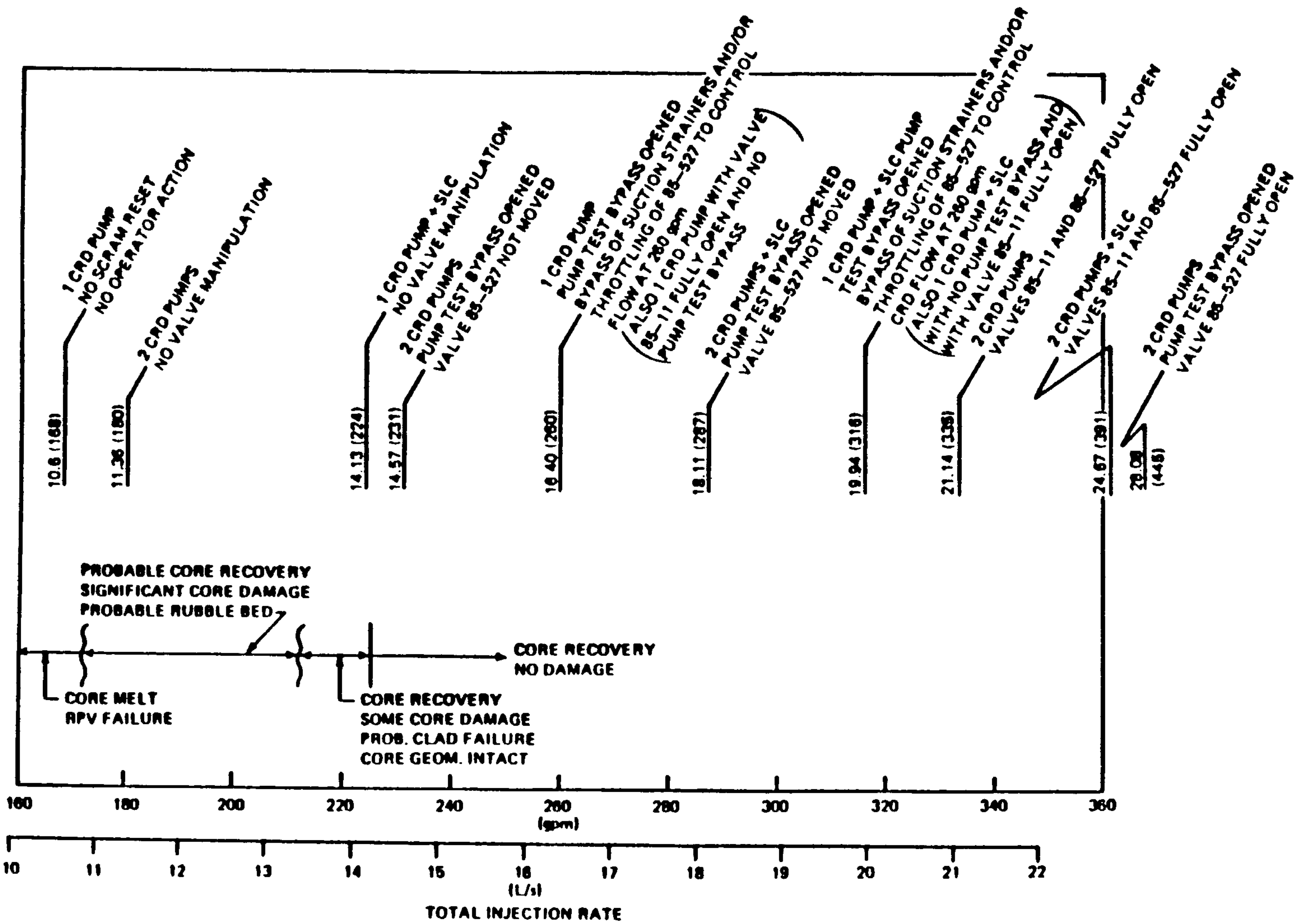


Figure A-7. **BFNP Unit 1 low capacity injection capability after rapid depressurization of the RPV [injection flows evaluated with the RPV at 1.55 MPa (225 psia)].**

From NUREG/CR-3179 (Reference 3)

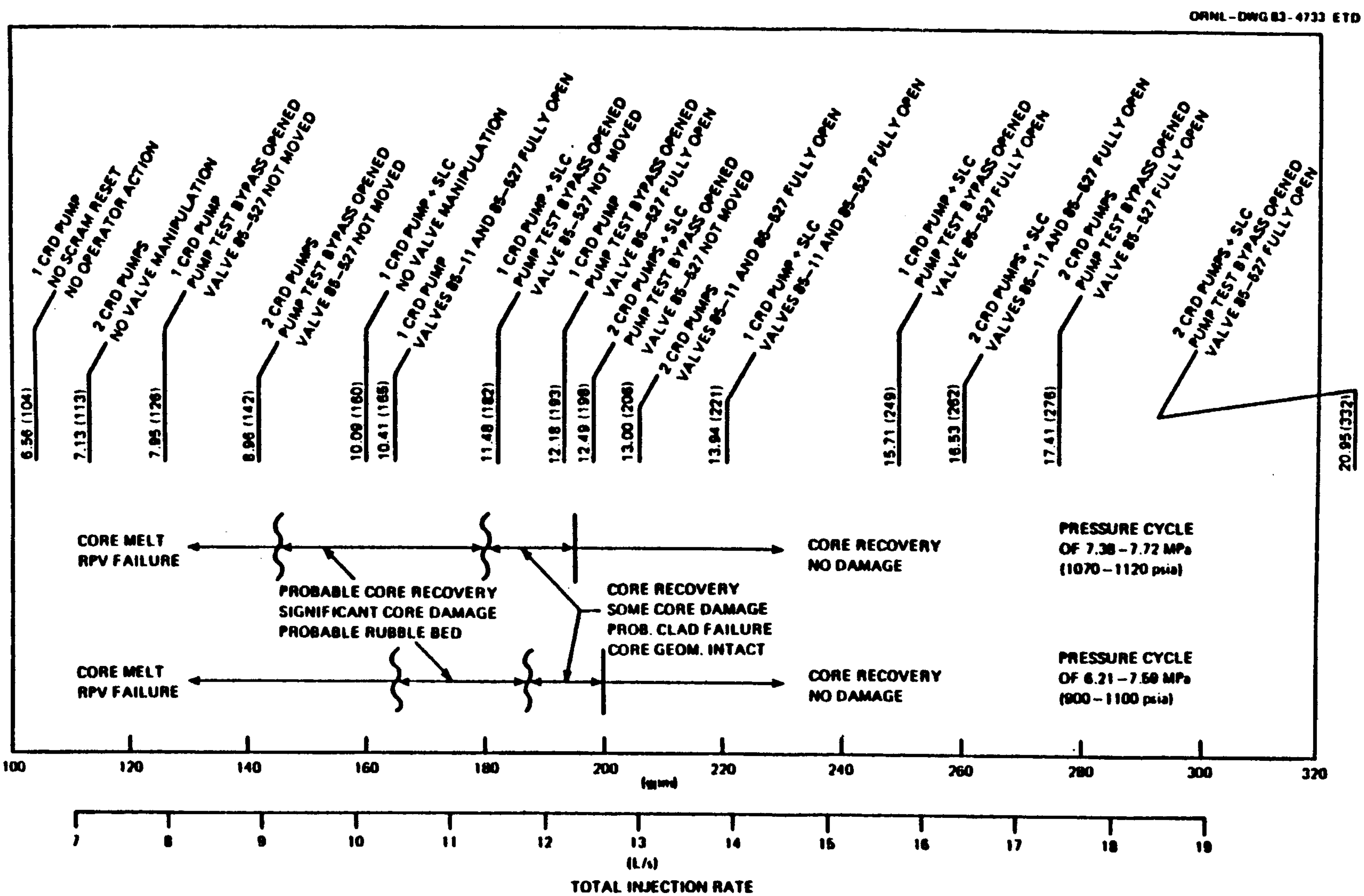


Figure A-8. **BFNP Unit 1 low capacity injection system capability with the reactor at pressure [7.58 MPa (1100 psia)].**

Note: [Past PRAs have also found core damage under high RPV pressure conditions as a dominant scenario. This was due to failure of the operator to depressurize the RPV to allow low pressure injection. With design changes being made to most ADS actuation circuits to allow auto actuation on just low-low level, and due to the EPG requirements that reinforce and clarify conditions requiring operator manual depressurization, the high pressure RPV core-damage scenario appears somewhat less likely. However, due to operator concern at also keeping the core covered for as long as possible and since the decision to depressurize may be very dependent on whether or not a LPCS/LPCI pump at least starts (although it may fail to inject water into the core later because of an injection valve failure, pump cooling failure, etc.), the high pressure RPV case is also considered among the "likely" scenarios by the PEEESAS/Accident Management Program technical staffs.]

7.0 AE - LARGE LOCA WITH EARLY LOSS OF CORE COOLING

- Initiated by a large loss-of-coolant accident (LOCA) such as the double-ended rupture of a recirculation pipe.

- Functions successful - Reactor subcriticality, RCS overpressure protection (by virtue of the LOCA)

- Functions failed - Core heat removal, Containment heat removal,* Containment overpressure protection,* Radioactivity removal*

- Systems successful - All but LPCS, LPCI

- Systems failed - LPCS and LPCI due to hardware/T&M faults or support system faults (power, service water). Recovery actions fail in time to prevent quick core melt.

- Based on past PRA analyses and insights from the BMI-2104 report, the more "likely" core-damage scenario is as shown in Figure A-9.(4) Environmental profiles should be based on this scenario. Note the following:

  - Contingency #6 provides guidelines for attempting to restore RPV level with various systems.

  - EPG-SP/T-2 calls for attempts to provide pool cooling if RHR is available for this mode (unlikely, however).

  - Drywell coolers should operate until the drywell temperature exceeds 200^{o}F.

---

*Depending on particular equipment failures (e.g. RHR) just as discussed for the TQUV sequence.

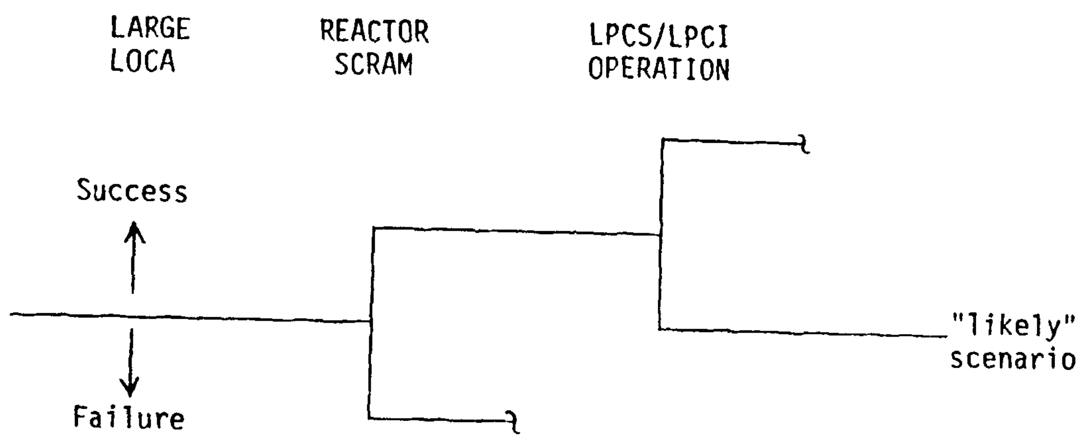


Figure A-9. "Likely" Scenario for AE
(up to significant core melt)

- EPGs-DW/T-3 and PC/P-7 call for drywell spray operation (but most likely RHR is failed) and containment venting.

o Key operator actions up to the point of core melt:

  - Operator attempts to quickly initiate alternate injection systems if available.

8.0 SUMMARY

This appendix summarizes work performed under the Sandia PEEESAS program addressing scenario definition. The accident sequences and resulting "likely" scenarios that will be used to determine the accident environmental profiles up to core melt or containment for this program are highlighted.

It should be recognized that to cope with severe accidents complicated with equipment failures due to the severe accident environment, accident management strategies must focus on contingency plans and operator training. Prior to the TMI-2 incident, accident management strategies were based on dealing with the design basis LOCA. These strategies were procedure-oriented and operator training tended to emphasize the ability to quickly identify and carry out specific procedural steps. After the TMI-2 accident, accident management strategies began changing by using a symptom-oriented philosophy for mitigating an accident. The current Emergency Procedure Guidelines (EPGs) and operator training reflect this new philosophy and cover the possibility of severe accidents.

The severe accident sequence environmental profiles generated in Appendix C reach temperatures and pressures above current equipment qualification levels. This implies that equipment may fail because of the environment. Therefore, it may be helpful if information is included in the EPGs regarding the possible environmentally-induced failure of electrical equipment, within containment, after qualification limits are exceeded.

A specific example of this type of situation involves EPGs which currently recommend transfer of HPCI and RCIC system suction to the suppression pool if "high water level" exists in the pool. Since this procedure may occur when the water in the pool has reached a high temperature and the high temperature may cause the HPCI and RCIC systems to fail, it may be appropriate to delete this instruction, provided the plant can show that the torus can withstand instabilities.

As indicated above, the possibility of environmentally-induced electrical equipment failures could complicate the situation for the operator when he is attempting to mitigate the accident and determine plant status. Therefore, it may be useful for operator training to include the simulation of electrical equipment failure for equipment that may fail due to the environment. This training would enhance operator awareness of the need to use multiple indication devices (and not just rely on one indicator) as well as provide practice in identifying and using alternate mitigating schemes.

9.0 REFERENCES

1 J.W. Minarick, and C.A Kukielka, "Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report," NUREG/CR-2497, ORNL/NSIC-182, June 1982.

2 J.W. Minarick, et al., "Precursors to Potential Severe Core Damage Accidents: 1980-1981 A Status Report," NUREG/CR-3591, ORNL/NSIC-217 (Draft), Februrary 1984.

3 R.M. Harrington and L.J. Ott, "The Effect of Small-Capacity, High -Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One," NUREG/CR-3179, ORNL/TM-8635, September 1983.

4 J.A. Gieseke, et al., "Radionuclide Release Under Specific LWR Accident Conditions," BMI-2104 Volume II, July 1984.

5 A.Kolaczkowski, et al., "Presentation to NRC of ASEP Findings Based on Interim Program Results," May 1985.

6 D.H. Cook, et al., "Loss of DHR Sequences at Browns Ferry Unit One Accident Sequence Analysis," NUREG/CR-2973. ORNL/TM-8532, May 1983.

7 D.H. Cook, et al., "Station Blackout at Browns Ferry Unit One, Accident Sequence Analysis," NUREG/CR-2182, ORNL/NUREG/TM-455/V1 (November 1981), Appendix A.

8 R.M. Harrington (ORNL), S.A. Hodge (ORNL), "ATWS at Browns Ferry Unit One--Accident Sequence Analysis," NUREG/CR-3470, ORNL/TM-8902, July 1984.

REFERENCES (Continued)

9 "Emergency Procedure Guidelines for Browns Ferry Unit One," June 1984.

10 David D. Carlson, et al., "Interim Reliability Evaluation Program Procedures Guide," NUREG/CR-2728, SAND82-1100, January 1983.

11 IDCOR Technical Report, "Task 21.1 Risk Reduction Potential", November 1984.

12 F.E. Haskin, et al., "Containment Event Analysis for Postulated Severe Accidents at the Peach Bottom Nuclear Power Plant",Sandia National Laboratories (Draft Material), March 1985.

From:
NUREG/CR-2182

ATTACHMENT 1

Browns Ferry Nuclear Plant: Complete Station Blackout Sequence of Events

CSB + No HPCI/RCIC

(TUB´)

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.
3.0	Turbine trips off (turbine stop valves fully closed).

Time (sec)	Event
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Seven (7) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 0.51 m (20 in.) from the previous momentary 1.02 m (40-in.) drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 7 S/RVs are completely closed.
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 5.98 m (235.50 in.) above Level 0, or 5.00 m (196.44 in.) above TAF.
22.0	Suppression pool water average temperature rises to 35.13°C (95.24°F) in response to the first S/RV pops.
29.0	All 4 S/RVs are completely closed.
29.7	Two out of 13 S/RVs start to open.

Time (sec)	Event
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 35.3°C (95.54°F).
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 5.50 m (216.00 in.) above Level 0, or 4.50 m (176.94 in.) above TAF.
90.0	Suppression pool water average temperature is approximately 35.4°C (95.72°F).
101.0	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the sequence.
625	Wide range sensed water level reaches low water level set-point (Level 2), i.e., 4.18 m (164.50 in.) above Level 0 at 2/3 core height, or 2.96 m (116.50 in.) above TAF.
625	HPCI and RCIC systems are not turned on because they are assumed to be unavailable.
20 min.	Suppression pool water average temperature reaches 46°C (114°F).
33 min.	Core uncovery time. Steam-water mixture level is at 3.54 m (11.61 ft) above bottom of the core.
40 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The HPCI/RCIC systems are not affected. Drywell and wetwell temperature are 72°C (162°F) and 55°C (130°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	33	4.36×10^3	9.25×10^7	5.26×10^6
Hydrogen	6×10^{-9}	8.62×10^{-7}	2.92×10^{-2}	1.66×10^{-3}

Time (sec)	Event
60 min.	Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	15.6	2.07×10^3	5.06×10^7	2.88×10^6
Hydrogen	2.8×10^{-3}	3.76×10^{-1}	2.15×10^4	1.22×10^3

Time (sec)	Event
70 min.	Core melting starts.
80 min.	Drywell and wetwell temperatures are 75°C (167°F) and 63°C (145°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	5.68	7.51×10^2	2.22×10^7	1.26×10^6
Hydrogen	0.19	2.53×10^1	2.29×10^6	1.30×10^5

Time (sec)	Event
96 min.	Water level in vessel drops below bottom grid elevation.
97 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
99 min.	The corium slumps down to vessel bottom.
101 min.	The debris is starting to melt through the bottom head. Drywell and wetwell temperatures are 97°C (207°F) and 71°C (159°F), respectively. Meanwhile, local pool water temperature at the discharging bay exceeds 149°C (300°F). Steam condensation oscillations could accelerate due to the continuous discharge of superheated noncondensable gases into the suppression pool. Mass and energy addition rates into the wetwell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	18.6	5.46×10^3	5.42×10^7	3.08×10^6
Hydrogen	6.8×10^{-2}	8.93	3.59×10^5	2.04×10^4

Time (sec)	Event
129 min.	Vessel bottom head fails, resulting in a pressure increase of 0.34 MPa (49 psia).
129.03 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 1546°C (2815°F) initially. Internal heat generation in metals and oxides are 1.36×10^7 and 2.50×10^7 watts, respectively.

Time (sec)	Event
165 min.	Drywell and wetwell temperatures are 141°C (286°F) and 74°C (166°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	5.46	722.83	1.59×10^5	9052
Hydrogen	3.3×10^{-2}	4.38	0	0
CO_2	2.58	341.88		
CO	0.69	91.35		

190 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 204°C (400°F) and start to vent through the primary containment.
193 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment.
219 min.	Drywell and wetwell pressures are at 0.10 MPa (14.7 psia). Drywell and wetwell temperatures are 598°C (1109°F) and 78°C (173°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	0.70	92	1.59×10^5	9052
Hydrogen	0.24	32	0	0
CO_2	2.32	307		
CO	5.03	666		

The leak rate through the containment failed areas is ~2.90×10^5 ℓ/s (~6.15×10^5 ft^3/min).

250 min.	Drywell and wetwell temperatures are 675°C (1247°F) and 78°C (173°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	6.84	905	1.59×10^5	9052
Hydrogen	0.25	33	0	0
CO_2	1.53	203		
CO	5.25	695		

Time (sec)	Event
	The leak rate through the containment failed area is ~4.91 × 10^4 ℓ/s (~1.04 × 10^5 ft^3/min).
309 min.	Rate of concrete decomposition is ~4.65 × 10^4 gm/s. Rate of heat added to atmosphere is ~1.20 × 10^4 kW.
367 min.	Drywell and wetwell pressures are at 0.10 MPa (~14.7 psia) and temperatures are 854°C (1570°F) and 77°C (171°F), respectively. The leak rate through the containment failed area is ~3.94 × 10^4 ℓ/s (~8.35 × 10^4 ft^3/min).
733 min.	Drywell and wetwell temperatures are 546°C (~1014°F) and 77°C (170°F), respectively. The leak rate through the containment failed area is ~2.12 × 10^3 ℓ/s (~4.50 × 10^3 ft^3/min).

ATTACHMENT 2

Browns Ferry Nuclear Plant: Complete Station Blackout
Sequence of Events

CSB + Manual RCIC & SRV

($T_V B'$)

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.

Time (sec)	Event
3.0	Turbine trips off (turbine stop valves fully closed).
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Seven (7) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 0.51 m (20 in.) from the previous momentary 1.02 m (40-in.) drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 7 S/RVs are completely closed.
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 5.98 m (235.50 in.) above Level 0, or 5.00 m (196.44 in.) above TAF.
22.0	Suppression pool water average temperature rises to 35.13°C (95.24°F) in response to the first S/RV pops.

Time (sec)	Event
29.0	All 4 S/RVs are completely closed.
29.7	Two out of 13 S/RVs start to open.
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 35.3°C (95.54°F).
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 5.50 m (216.00 in.) above Level 0, or 4.50 m (176.94 in.) above TAF.
90.0	Suppression pool water average temperature is approximately 35.4°C (95.72°F).
101.0	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the subsequent RCIC injections.
625	Wide range sensed water level reaches low water level setpoint (Level 2), i.e., 4.18 m (164.50 in.) above Level 0 at 2/3 core height, or 2.96 m (116.50 in.) above TAF.
625	Operator manually controls RCIC injection to maintain constant vessel water level. The RCIC turbine pump is driven by steam generated by decay heat. System auxiliaries are powered by the 250 V dc system.
655	RCIC flows enter the reactor pressure vessel at 38 ℓ/s (600 gpm) drawing water from the condensate storage tank.
15 min.	Operator manually opens one SRV to depressurize the vessel.
20 min.	Drywell and wetwell temperatures exceed 76°C (169°F) and 50°C (122°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	829.75	1.10×10^5	2.32×10^8	1.32×10^7
Hydrogen	0	0	0	0

Time (sec)	Event
21.14 min.	Core uncovery time.
22.0 min.	Core refloods.
30 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The RCIC system is not isolated.
240 min.	The RCIC pump stops when the batteries run out.
266.3 min.	Wide range sensed water level reaches Level 2 setpoint. Drywell and wetwell temperatures are 99°C (210°F) and 100°C (212°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	19.16	2.53×10^3	5.20×10^7	2.96×10^6
Hydrogen	0	0	0	0

Time (sec)	Event
347 min.	Core uncovers again.
366 min.	Average gas temperature at top of core is 491°C (916°F). Drywell and wetwell temperatures and pressures are 113°C (236°F) and 0.28 MPa (40 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	9.26	1.22×10^3	2.97×10^7	1.69×10^6
Hydrogen	4.09×10^{-5}	5.41×10^{-3}	222.28	12.64

Time (sec)	Event
386 min.	Average gas temperature at top of core is 855°C (1571°F). Drywell and wetwell temperatures and pressures are 115°C (239°F) and 0.29 MPa (41 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	5.05	6.68×10^2	1.81×10^7	1.03×10^6
Hydrogen	1.68×10^{-2}	2.23	1.35×10^5	7.70×10^3

Time (sec)	Event
395.3 min.	Core melting starts.
449.3 min.	Water level in vessel drops below bottom grid elevation.
451.2 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
452 min.	The corium slumps down to vessel bottom.
452.9 min.	Debris starts to melt through the bottom head.
539.3 min.	Vessel bottom head fails, resulting in a pressure increase of 0.0047 MPa (0.68 psia).
539.3 min.	Debris starts to boil water from containment floor.
539.3 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 204°C (400°F) and start to vent through the primary containment at a leak rate of 118 ℓ/s (250 ft^3/min).
539.3 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 1750°C (3182°F) initially. Internal heat generation in metals and oxides are 9.99×10^6 and 1.84×10^7 watts, respectively.
601.05 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment. Mass and energy addition rates into the drywell are:

	Mass Rate		Energy Rate	
	(kg/s	(lb/min)	(w)	(Btu/min)
Steam	4.70	621.51	1.59×10^5	9052
Hydrogen	0.14	18.27	0	0
CO_2	1.29	170.23		
CO	2.88	381.21		

The leak rate through the drywell penetration seals is ~5.33×10^4 ℓ/s (1.13×10^5 ft^3/min).

718.8 min.	Drywell and wetwell pressures are at 0.10 MPa (~14.7 psia) and temperatures are 700°C (1293°F) and 98°C (~209°F), respectively. The leak rate through the containment failed area is ~5.18×10^4 ℓ/s (~1.10×10^5 ft^3/min).

Time (sec)	Event
821.5 min.	Drywell and wetwell temperatures are 737°C (1359°F) and 93°C (199°F), respectively. The leak rate through the containment failed area is ~4.23×10^4 ℓ/s (~8.96×10^4 ft^3/min).
1127.5 min.	Drywell and wetwell temperatures are 468°C (~875°F) and 86°C (~188°F), respectively. The leak rate through the containment failed area is ~4.79×10^4 ℓ/s (~1.02×10^4 ft^3/min).

Table 3.1. Timetable of events for unmitigated loss of DHR with uniform pool heatup

Time (h)	Event
0	Initiating reactor trip followed by MSIV closure and failure of both pool cooling and shutdown cooling modes of the RHR system.
1	High drywell pressure scram at 0.115 MPa (2 psig). Diesel generators and SGTS automatically initiated. Drywell control air compressors isolated. Operators valve station control air into drywell control air header.
1	Pool temperature exceeds 49°C (120°F) — operators begin controlled depressurization of reactor vessel.
2	Core spray initiation signal [reactor vessel pressure <3.21 MPa (465 psia) and drywell pressure >0.115 MPa (2 psig)] causes load shedding if loss of offsite power is still in effect. Operators must use local control stations to restore diesel power to station control air compressors (A and D) and drywell coolers.
2	Suppression pool temperature exceeds the 60°C (140°F) recommended maximum temperature for cooling of RCIC and HPCI lube oil.
4	CRD hydraulic system provides sufficient reactor vessel injection — no RCIC system operation after this time.
8.6	Operators must begin to throttle CRD hydraulic system pump to avoid overfilling the reactor vessel.
13	HPCI and RCIC system steam supply line isolation caused by high [93°C (200°F)] torus room temperature.
14	RCIC turbine high exhaust pressure trip at containment pressure >0.28 MPa (25 psig).
21.5	Drywell design pressure [0.49 MPa (56 psig)] exceeded.
23.5	SRVs become inoperative in remote-manual mode because drywell pressure exceeds 0.55 MPa (65 psig).
35	Drywell fails when internal pressure exceeds 0.91 MPa (117 psig). Suppression pool temperature has increased to 173°C (343°F).

ATTACHMENT 4

Table 4.5. Sequence of events for case without manual rod insertion or SLC injection, but with pool cooling

Time (min)	Event	Comment
0	MSIVs begin to close	Anticipated transient
0.1	No reactor scram	
0.1	Recirculation pumps trip	
1.5	HPCI and RCIC start	Automatic actuation, total injections 5600 gpm (353 l/s)
2	Operator control of vessel pressure begins	To prevent SRV cycling on automatic actuation
7	Operator trips HPCI and RCIC	Per EPG level/power control guideline
8	Core spray and RHR pumps start	At vessel water level <413.5 in. (12.5 m) - reactor vessel pressure too high for injection
8.4	Vessel water level below TAF	Operator restarts HPCI at 1800 gpm (113 l/s)
8.5	Reactor power below 10%	
9	Vessel pressure dropping	Operator shuts all but one SRV
10	Operators initiate suppression pool cooling with all four coolers	"Containment Spray Select" switch actuated
14.8	Vessel water level above TAF	Not back on scale of emergency systems indication
16.8	Power spike	Core thermal power to 35%
16.8	Automatic SRV actuations	
17	Operators decrease HPCI flow	Vessel water level too high
18.7	Operators begin emergency depressurization of reactor vessel	Suppression pool in violation of EPG heat capacity temperature limit
18.7	Operators trip HPCI and RCIC turbines and the core spray, condensate, condensate booster, and RHR pumps	Interrupts suppression pool cooling
19.5	Drywell pressure exceeds 2.45 psig (118 kPa)	
19.6	Core completely uncovered	Subcritical and producing only decay heat
20.1	Vessel pressure below 450 psig (3.21 MPa)	Core spray and LPCI valves open (LPCI valves interlocked open for 5 min)
20.6	Operators resume vessel injection	Using condensate booster pumps, flow controlled by startup bypass valve
27	Operators restart suppression pool cooling	After overriding 2/3 core coverage interlock
27.8	All SRVs shut	Vessel-to-drywell pressure difference <20 psi
31.8	Vessel water level recovered to >TAF	Level not back on scale of emergency systems indication
33.3	Operators discontinue injection flow	Emergency systems indication on scale but increasing too fast

Table 4.5 (continued)

Time (min)	Event	Comment
33.8	SRVs reopen	Vessel-to-drywell pressure difference >50 psi
34.6	Vessel power and pressure spike	Maximum core thermal power = 81%
34.8	Automatic SRV actuations	At 1105 psig (7.72 MPa)
36.5	Vessel pressure below 450 psig (3.1 MPa)	Depressurizing with five open SRVs
40—end	Additional power/pressure spikes	Occurring about every 13 min
120	Suppression pool temperature at 232°F (384 K)	Still increasing
720	Suppression pool temperature at 345°F (447 K)	Drywell overpressure failure imminent

Appendix B

IDENTIFICATION OF ELECTRICAL

EQUIPMENT

TABLE OF CONTENTS

LIST OF TABLES

1.0 OBJECTIVES

This appendix addresses the selection of critical electrical equipment for the selected accident sequences. This portion of the work calls for identification of electrical equipment in boiling water reactors (BWRs), which may be important in preventing or mitigating severe accidents, and which may have to survive environments or service limits beyond current design capabilities while functioning under severe accident conditions.

Identification of this electrical equipment was performed in four steps; first, a cumulative list of equipment needs (see Section 2.2) was formed using a variety of information sources presented in Section 2.1. These information sources each define "important equipment to safety" from a variety of perspectives. Second, this list of equipment needs was reduced to that electrical equipment with components located in the primary containment or reactor vessel of a typical BWR-4, Mark I plant using primarily Browns Ferry-1 design information (see Section 2.3). This latter step focuses on that equipment likely to be in the most severe environments given the accidents being studied, and hence most worthy of examination. Third, the specific components within containment for each category of equipment needs were identified with accompanying manufacturer and model information, where possible. Last, using qualitative arguments, the relative importance of these components was addressed considering the potential functional importance of the components in mitigating selected accident sequences (last two steps are presented in Section 3.0). The potentially more important components makeup the recommended list of items worthy of further examination.

2.0 IMPORTANT SAFETY EQUIPMENT

2.1 INFORMATION SOURCES USED TO DEFINE "IMPORTANT EQUIPMENT TO SAFETY"

A number of information sources were used to establish a cumulative list of equipment needs which are viewed as important to safety and hence might be used to manage an accident. Table B-1 summarizes these information sources and their perspectives regarding power plant safety. The information sources represent a wide variety of safety viewpoints. They also provide current thinking on what equipment is important to safety for preventing/mitigating accidents and providing important plant status information. The information from the references in Table B-1 and general knowledge of equipment found to be important to dominant accident sequences in probabilistic risk assessments (PRA) for BWRs, were used to obtain a list of possible systems and plant status monitoring parameters potentially important to manage any accident.

2.2 EQUIPMENT NEEDS IDENTIFIED AS IMPORTANT TO SAFETY

Based on the varied information sources identified in the previous section, a cumulative list of equipment needs potentially important to accident prevention or mitigation has been developed. This list should be reasonably complete because of the many information sources and perspectives used which represent the combined input of both regulators and industry personnel.

The resulting cumulative list of equipment needs is presented in Table B-2. The systems and plant status monitoring instrumentation making up the list are categorized by the major function with which they are associated. While some equipment serves more than one function, each item is identified with only one function to avoid duplication. This list represents the equipment which could be important in preventing or mitigating an accident for a typical BWR-4, Mark I design.

Table B-1. Information Sources Reviewed for "Important Equipment to Safety"

SOURCE	SAFETY PERSPECTIVE
o BWR Emergency Procedure Guidelines Revision 3	Guidelines for responses to plant upsets in order to prevent or mitigate severe accidents. Identifies important systems and instrumentation.
o Regulatory Guide 1.97 - Revision 3	Regulatory guide indicating important parameters to be monitored during and following an accident.
o Assessment of Generic Instrumentation Systems Used to Meet Provisions of Regulatory Guide 1.97, EGG-EE-6154, February 1983	Expands on current plant system capabilities to meet Regulatory Guide 1.97.
o BWR Status Monitoring During Accident Conditions, NUREG/CR-2100, April 1981.	Identifies important operator actions during certain accident sequences and the information required to take appropriate action.
o Light Water Reactor Engineered Safety Features Status Monitoring Final Report, NUREG/CR-2278, August 1981	Reviews current system monitoring capabilities in comparison with the importance of systems based on PRA.
o Fundamental Safety Parameter Set for Boiling Water Reactors, NSAC/21, December 1980	Identifies those fundamental parameters that provide an overview of plant safety and therefore should be a part of the safety parameter display system.
o Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, NUREG-0654, FEMA-REP-1, Rev. 1, November 1980	Provides examples of conditions defining each emergency action level in reference to formulating emergency responses to potential accident conditions.
o Various IDCOR* Reports	Identifies important systems to BWR plant safety based on PRA and gives examples of equipment important for accident survivability considerations.

*Industry Degraded Core Rulemaking Program.

Table B-1. Information Sources Reviewed for "Important Equipment to Safety" (Concluded)

SOURCE	SAFETY PERSPECTIVE
o Preliminary Interface Specification for NDL Data Acquisition System and NRC Terminal, NUREG/CR-2025, June 1982	Identifies important parameters to be monitored to indicate plant status for the nuclear data link.
o TMI Nuclear Power Plant Accident Hearings Before the Subcommittee on Nuclear Regulation of the Committee on Environment and Public Works, U.S. Senate, 96th Congress, First Session, October 2-3, 1979	Identifies equipment needed to monitor plant conditions during the TMI-2 accident.
o Station Blackout at Browns Ferry Unit One - Accident Sequence Analysis, NUREG/CR-2182, November 1981	Identifies important equipment still operable under station blackout (loss of all AC) conditions.

Table B-2. Equipment Needs Important to Safety in a BWR-4, Mark I Design

FUNCTION - REACTOR SUBCRITICALITY

Neutron flux indication - average, intermediate, and source range monitors (APRM, IRM, SRM) including SRM detector position

Reactor protection system (RPS) scram equipment including trip signal indication

Manual scram and reset equipment

Scram valve controls

Hydraulic control unit (HCU) accumulator charging water header valve

Scram discharge volume vent/drain valves

Scram discharge volume tank level

Control rod scram test switches

Control rod drive (CRD) withdraw line vent valve

Reactor mode switch (turn to shutdown mode)

Control rod position (including magnetic reed switches)

Recirculation pump trip (RPT) equipment and indication of pump discharge pressure, speed, and flow

Standby liquid control (SLC) system operation including indication of flow, pump discharge pressure, and valve positions

SLC boron tank level

Soluble boron concentration (by sampling)

FUNCTION - REACTOR COOLANT SYSTEM (RCS) OVERPRESSURE PROTECTION

RCS pressure indication

Safety/relief valve (SRV) operation and indication of position by valve position, discharge line flow, acoustical monitor, temperature sensor, or valve air pressure

Automatic depressurization system (ADS) operation including the valve position indication by the methods listed above for SRVs and indication of ADS actuation signal

Table B-2. Equipment Needs Important to Safety in a BWR-4, Mark I Design (Continued)

FUNCTION - CORE HEAT REMOVAL

Reactor pressure vessel (RPV) water level

Core temperature indication using in-core thermocouples

High pressure coolant injection (HPCI) system operation including indication of flow, pump discharge pressure, valve positions, turbine backpressure, and steamline flow

Reactor core isolation cooling (RCIC) system operation including indication of flow, pump discharge pressure, valve position, steam line flow, turbine backpressure, and flow controller position

Low pressure core spray (LPCS) system operation and indication of flow, pump discharge pressure, and valve positions

Low pressure coolant injection (LPCI) system operation and indications as above for LPCS

CRD operation including flow indication

Condensate/feedwater operation including indication of flow, pump discharge pressure, feedwater controller position, steam flow to feedwater turbine, and condensate pump current

Reactor water cleanup (RWCU) system operation

SLC operation from demineralized water tank

Residual heat removal (RHR) system operation in the shutdown cooling and steam condensing modes including indication of flow, pump discharge pressure, heat exchanger inlet and outlet temperatures, and valve positions

Emergency core cooling system (ECCS) keep-full systems operation

Residual heat removal service water (RHRSW) operation

Firewater system operation including firemain pressure

Operation of and indications associated with interconnecting unit emergency systems

Main steam isolation valve (MSIV) operation and position indication

Steam line flow and pressure

Condenser operation including indication of pressure, hotwell temperature and level, air ejector flow, circulating water flow, circulating water pump power and discharge header pressure

Table B-2. Equipment Needs Important to Safety in a BWR-4, Mark I Design (Continued)

Core flow (delta-P instrument)

Condensate storage tank (CST) level

Recirculation loop temperatures

Turbine trip signal

Turbine bypass and stop valve positions

Temperature sensors on reactor vessel surface

FUNCTION - CONTAINMENT HEAT REMOVAL

Drywell temperature

Suppression pool temperature

Drywell cooler operation including fans, reactor building closed cooling water (RBCCW) coolers, dampers, and temperature sensors

RHR operation in the suppression pool cooling mode including indication of flow, pump discharge pressure, heat exchanger inlet/outlet temperatures, and valve positions

Secondary containment HVAC operation

HVAC cooler temperatures

Secondary containment temperature

FUNCTION - CONTAINMENT OVERPRESSURE PROTECTION

Drywell pressure

Suppression pool chamber pressure

Vacuum breaker operation and position indication

Drywell spray operation (RHR) including flow and valve positions

Suppression chamber spray operation (RHR) including flow and valve positions

Primary containment venting operation

Secondary containment pressure

Suppression pool level

Table B-2. Equipment Needs Important to Safety in a BWR-4, Mark I Design (Continued)

FUNCTION - RADIOACTIVITY MONITORING AND REMOVAL

Indication of fuel failure such as primary coolant activity or main steam line radiation

Radiation in suppression chamber

Standby gas treatment (SBGT) operation

Drywell purge operation

Drywell radiation

Reactor building radiation

Secondary containment radiation

General area radiations

Radiation (noble gases) and vent flows at release points (including halogen/particulate content with sampling) - e.g. SBGT/drywell purge, secondary containment purge, building release points, HVAC exhausts, mainstack (off-gas) monitor

MSIV leakage control system pressure

Offsite radiation release rate

Liquid tanks/discharge radiation

High radioactivity liquid tank level

MISCELLANEOUS

Primary containment humidity

Secondary containment humidity

RCS high point vent valve operation

Reactor thermal power

Primary/secondary containment isolation including valves, dampers, and signals

Drywell sump level and drain sumps level

Sump pump operation

Floor drain sump levels

Table B-2. Equipment Needs Important to Safety in a BWR-4, Mark I Design (Concluded)

Area water levels

Pump compartment temperatures

Area temperatures

Area coolers operation

Cooling water temperatures and flows to emergency components

Fire suppression operation

Containment and drywell hydrogen/oxygen content

RPV hydrogen

Hydrogen in off-gas

Diesel generator operation

Power supply status including electric, hydraulic, pneumatic (voltages, currents, pressures)

Portable sampling for radiohalogens, etc. throughout plant, environs

Meteorology - wind direction, speed, atmosphere temperatures

2.3 EQUIPMENT WITH ELECTRICAL COMPONENTS WITHIN CONTAINMENT/REACTOR VESSEL

Since the PEEESAS program is concerned with electrical equipment survivability and functionality during and following severe accident conditions, first focus is on that electrical equipment that is located within the primary containment or reactor vessel. It is in these areas where the environmental conditions experienced by the equipment will be generally most severe for the accident sequences being reviewed. With this in mind, the list of equipment needs in Table B-2 has been narrowed down to that equipment with electrical components located in the primary containment or reactor vessel based on review of the Browns Ferry design. The design information used includes References 1 and 2, Piping and Instrumentation Drawings from the Browns Ferry Final Safety Analysis Report, and system schematics from the Browns Ferry PRA listed as Reference 5. Table B-3 contains the resulting list of equipment and includes the purpose of the equipment, when during an accident the equipment may be needed, and the major components believed to makeup the equipment.

Table B-3. Equipment Inside Containment and Potentially Worthy of Review by PEEESAS

EQUIPMENT	PURPOSE	POTENTIAL TIME NEEDED*	COMPONENTS INSIDE CONTAINMENT/Rx VESSEL
Neutron flux (power) measurement using average, intermediate, and start-up range monitors (APRM, IRM, SRM).	Provide indication of power level and reactor shutdown.	Seconds to minutes after scram signal. In TC sequence, may be required until shutdown eventually achieved or significant core damage results.	Ion Chambers, Cabling, Connectors/splices? Terminal blocks?
Control rod position indication (full-in) devices.	Provide indication of "rods in" for reactor shutdown.	Seconds to minutes after scram signal. In TC sequence, may be required until shutdown eventually achieved or significant core damage results.	Magnetic reed switches, Cabling, Connectors/splices? Terminal blocks?
Primary System Safety Relief Valves	Controls primary system pressure.	Throughout sequence up to reactor vessel failure.	Pilot valves, Cabling, Connectors/splices? Terminal blocks? Position indication by: Temperature elements Acoustical devices Pressure transmitters
Core temperature elements	Indication of adequate core cooling.	Throughout sequence up to time of reactor vessel failure.	In-core thermocouples Reactor vessel surface thermocouples Cabling, Connectors/Splices? Terminal blocks?
Drywell/suppression chamber temperature elements	Indication of adequate containment cooling.	Throughout sequence up to and maybe even beyond containment failure.	Temperature elements, Cabling, Connectors/Splices? Terminal blocks?
Drywell pressure indication monitors (in containment?)	Provides indication of containment cooling and pressure control	Throughout sequence up to and maybe even beyond containment failure.	Pressure transmitters, Cabling, Connectors/splices? Terminal blocks?

? Indicates uncertainty in whether or not these components are used.

*For all five sequence classes identified in Task 1. Specific differences for certain sequences are noted.

Table B-3. Equipment Inside Containment and Potentially Worthy of Review by PEEESAS (Continued)

EQUIPMENT	PURPOSE	POTENTIAL TIME NEEDED*	COMPONENTS/INSIDE CONTAINMENT/Rx VESSEL
Drywell/suppression chamber radiation monitors	Monitor radiation levels particularly for consideration of emergency actions.	Throughout sequence and beyond to perhaps months after an accident.	Monitors, Cabling, Connectors/splices? Terminal blocks?
Drywell/suppression chamber hydrogen monitors	Monitor hydrogen levels throughout accident to determine potential for hydrogen burn.	Throughout sequence and beyond until any hydrogen is vented.	Monitors, Cabling, Connectors/splices? Terminal blocks?
Drywell humidity (in containment?) monitors	Monitors humidity level in containment.	Throughout sequence including, perhaps, beyond containment failure.	Monitors, Cabling, Connectors/splices? Terminal blocks?
Drywell cooling system equipment	Provides containment cooling in the drywell.	Throughout sequence up to and maybe even beyond containment failure.	Fan motors, Air solenoid valves for cooling water and damper positioning, Temperature elements, Position switches, Cabling, Connectors/splices? Terminal blocks?

? Indicates uncertainty in whether or not these components are used.

*For all five sequence classes identified in Task 1. Specific differences for certain sequences are noted.

Table B-3. Equipment Inside Containment and Potentially Worthy of Review by PEEESAS (Concluded)

EQUIPMENT	PURPOSE	POTENTIAL TIME NEEDED*	COMPONENTS/INSIDE CONTAINMENT/Rx VESSEL
Drywell/drain sump level and temperature devices	Monitoring for primary system leakage or a LOCA.	Throughout sequence up to reactor vessel failure.	Level elements/switches, Temperature elements, Cabling, Connectors/splices? Terminal blocks?
Sump pumps	Prevent significant water levels in the drywell and to avoid equipment flooding.	Throughout sequence up to reactor vessel failure.	Pump motors, Cabling, Connectors/splices? Terminal blocks?
Recirculation pumps	Pump trip function only; particularly in TC sequence to lower power level.	Important to TC sequence in first seconds of the accident.	Pump motors, Cabling, Connectors/Splices? Terminal Blocks?
Inboard containment isolation valves for HPCI, RCIC, RWCU, RHR, MSIVs etc.	Used to isolate containment pathways out to the environment.	Need to operate at the time that the isolation signal is present (typically within first half-hour of sequence but not necessarily). Also, valves may need to be reopened at any time (even after core damage/containment failure) to recover previously isolated or failed cooling systems.	Valve motors & position switches, Cabling, Connectors/splices? Terminal blocks?

? Indicates uncertainty in whether or not these components are used.

*For all five sequence classes identified in Task 1. Specific differences for certain sequences are noted.

3.0 COMPONENTS RECOMMENDED FOR EXAMINATION

The previous section identified those components within the containment boundary potentially important to safety. For each category of equipment and major component in Table B-3, an attempt has been made to identify representative manufacturers and model numbers for the components used. Browns Ferry equipment qualification information was the major source of this data although two other plant designs were also reviewed to provide additional input. The following subsections summarize this information along with discussions on the relative importance of each set of components in mitigating selected accidents. Based on these qualitative discussions, a few select components are recommended for further examination by the PEEESAS program (since they could be important in accident mitigating strategies). It should be noted that some information is unavailable to the PEEESAS program to complete the data presented in the following subsections. However, for the components recommended for further study, sufficient information exists to progress to other tasks of the PEEESAS program.

3.1 Equipment - Neutron Flux (Power) Measurement Monitors

Component - Ion Detector (RV)*:

Browns Ferry Design - ?
Other Designs - Limerick (3)
General Atomic Model #?
- Cooper (4)
General Electric Model #?

- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Blocks (DW):

see "general information" in Section 3.12

Importance to Sequences -

The power range devices would provide indication of reactor shutdown in all the sequences of interest. The monitors would be used in only the first seconds to minutes for sequences TB, TW, TQUV, and AE. In the TC sequence, this measurement could assist the operator in maintaining low power level; however, other indications (e.g., water level) are likely to be more important. Backups such as control rod "in" indicators are also available. Because of the short time the measurement is used in four of the five sequences, and the questionable importance in the TC sequence, these components are not considered high on the list for further evaluation.

*Indicates location of component of interest. (RV) = reactor vessel, (DW) = drywell, (SP) = suppression pool.

3.2 Equipment - Control Rod Position Indication (full-in) Devices

Component - Magnetic Reed Switches (RV):

- Designs - ?

- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Blocks (DW):

see "general information" in Section 3.12

Importance to Sequences -

Same discussion as for the neutron flux measurement in Section 3.1.

3.3 Equipment - Primary System Safety Relief Valves (SRVs)

Component - Pilot Valves for SRVs (DW):

Browns Ferry Design (1&2) - Target Rock Model 1/2 SMS-A-01-1
(Not sure whether this is the pilot valve or SRV itself)

- Position indication by temperature elements (DW): see general information in Section 3.12

- Position indication by acoustical device (DW): Browns Ferry Design (1&2) - Endevco Model 2273A

- Position indication by pressure transmitter (DW):

  Other Designs - Cooper (4)
  Pressure Controls, Inc.

- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Blocks (DW):

  see "general information" in Section 3.12

Importance to Sequences -

Controlling primary system pressure by using the SRVs and accompanying components could be important in four of the five accident sequences of interest (not AE since reactor vessel remains at low pressure). In the TB sequence (with HPCI/RCIC failure), the ability to use the SRVs to achieve low pressure in the reactor vessel should AC power be restored could be important since all remaining high quantity injection systems require low pressure operation. Following containment failure in the TW and TC sequences, continued ability to use the SRVs could mean the difference between continuing to cool the core or core melt if only low pressure injection systems remain operable. In the TQUV sequence, the ability to achieve or maintain low pressure in the vessel can increase the amount of injection flow supplied by such low flow systems as the Control Rod Drive (CRD) system. Depending on the flow, the sequence may be mitigated. Should a low pressure injection system be recovered before core melt in the TQUV sequence, SRV operation could be an important prerequisite for low pressure injection. Based on the above discussion, these components are of potentially high importance pending comparison of the accident environmental profiles to the qualification profiles. Once vessel breach occurs in any sequence, these components are no longer important.

3.4 Equipment - Core Temperature Elements

Component - In-core Thermocouples (RV) and Reactor Vessel Surface Thermocouples (DW):

Designs - ?

- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Blocks (DW):

see "general information" in Section 3.12

Importance to Sequences -

These devices can provide an indication of adequate core cooling throughout any of the five sequences of interest up to the point of reactor vessel failure. This indication could provide additional input to the operator particularly if other indications are confusing. Since other indications (e.g., water level) are just as likely to be used, these components are qualitatively judged as moderately important to PEEESAS future consideration.

3.5 Equipment - Drywell/Suppression Chamber Temperature Elements

Component - Temperature Element (DW, SP):

Browns Ferry Design (1&2) - Weed SP601-1A-A-3-C-275-SN4-2 (Drywell)

- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Blocks (DW):

see "general information" in Section 3.12

Importance to Sequences

Monitoring of the containment temperature is a direct indication of adequate containment cooling and could be of some value depending on the accident sequence. This indication is of limited value in those sequences where containment failure occurs very quickly after qualification temperatures are reached, since little time exists for the operator to take any actions based on the indicated temperature. These sequences include the TC and AE sequences. In those sequences where containment failure occurs some time after core damage (TB, TQUV) or the approach to containment failure is very slow (TW), the survivability of this indication may be of some value to the operator in knowing when to take certain actions to prevent containment failure (e.g., venting). This equipment is therefore judged to be moderately important to PEEESAS for the sequences indicated.

3.6 Equipment - Drywell Pressure Monitors

Component - Pressure Transmitters (DW):

- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Blocks (DW):

see "general information" in Section 3.12

Importance to Sequences

Same discussion as for drywell/suppression chamber temperature in Section 3.5.

*Have not been able to verify that any component associated with this measurement is actually inside containment.

3.7 Equipment - Drywell/Suppression Chamber Monitors for Radiation, Hydrogen, and Humidity.

Component - Monitors (DW):

Browns Ferry Design (2)
Radiation - General Electric Model?
Hydrogen - GE Space Model, 47E226428G2
Limerick (3)
Radiation - General Atomic Model?

Not certain if humidity monitors are inside containment.

- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Blocks (DW):

see "general information" in Section 3.12

Importance to Sequences -

These monitors provide indications to the operator of the status of the environment inside containment regarding radiation, hydrogen, and humidity levels. The radiation could be important to all the accident sequences in assessing plant damage and for deciding on sheltering or evacuating the general public. Other monitors (fixed and portable), however, could also be used to make such decisions. The hydrogen monitors could indicate to the operator when hydrogen levels are approaching flammable limits and thus be potentially of some importance in all the sequences. Since equipment is typically qualified for all levels of humidity, such a monitor, if it is inside containment, appears to be of little value as an effective indication for accident mitigation. Therefore, the hydrogen and radiation monitor equipment are judged as being moderately important for PEEESAS consideration while the humidity equipment is of little concern.

3.8 Equipment - Drywell Cooling System Equipment

Component - Fan Motors (DW):

Designs - ?

- Damper Motors (DW):
  Browns Ferry Design (2) - Honeywell M445A

- Air Solenoid Valves (DW):
- Temperature Elements (DW):
- Position Switches (DW):
- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Blocks (DW):

see "general information" in Section 3.12

Importance to Sequences -

While the drywell cooling system operability could potentially add some delay time to containment failure for any of the five accident sequences of interest, this time is not considered significant under the large heat loads of an uncovered core. Therefore this equipment is considered of low importance for PEEESAS to pursue for accident mitigating purposes since it appears that it would have limited value under degraded core conditions.

3.9 Equipment - Drywell/Drains Sump Pumps, Sump Level and Temperature Devices

Component - Drain Sump Temperature (DW):
Browns Ferry Design (2) - Thermo Elect B7582-1

- Drywell Sump Level Switch and Transmitter (DW):
Browns Ferry Design (2) - GEMS XM-36425

- Sump Pumps Design (DW): ?
- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Blocks (DW):

see "general information" in Section 3.12

Importance to Sequences -

While the sump levels and temperatures could be used as additional indications of primary system leakage or failure, other parametric indicators can be used and thus these indications are not particularly important to any of the accident sequences of interest. Sump pump operation could be used to prevent equipment flooding, but no important equipment in the containment is believed to be located at near-submergence levels. Therefore the sump pumps and corresponding indications are considered of low value for future PEEESAS consideration.

3.10 Equipment - Recirculation Pumps

Component - Pump Motors (DW):

Design - ?

- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Blocks (DW):

see "general information" in Section 3.12

Importance to Sequences

Tripping of the recirculation pumps is particularly important in only the TC sequence in order to lower the core power level in the first seconds of the accident. While this category has equipment in the containment, it is the tripping of these pumps (not their operation) that is crucial. Therefore, these components are considered of low importance to the PEEESAS program.

3.11 Equipment - Inboard Containment Isolation Valves

Component - Valve Motors and Position Switches (DW):
(see Table 4)

- Cabling (DW):
- Connectors/Splices (DW):
- Terminal Boxes (DW):

see "general information" in Section 3.12

Importance to Sequences -

There are a variety of systems which have inboard valves that could be important to mitigating an accident. Table 4 summarizes the systems with valves inside containment along with comments regarding the importance of these valves for future PEEESAS consideration. In all cases, the initial closure of the valves (if required) to isolate containment should happen early in all the sequences. This is of lesser importance than the ability to reopen later to restore or otherwise start a cooling or heat rejection system after environmental conditions have degraded.

Table B-4. Inboard Containment Isolation Valves of Potential Interest to the PEEESAS Program

SYSTEM	VALVE ELECTRICAL COMPONENTS*	COMMENTS
Main Steam	o MSIV Manifold Assy (Solenoid Valve) Automatic Valve Corp C-5497 o MSIV Position Switch NAMCO EA740-50100 o Main Steam Drain Valve Actuator Limitorque SMB-000 (Bypass line)	The valves could be highly important to restore or maintain this heat rejection path (along with feedwater) once (a) AC is restored for TB or (b) up to the time of vessel breach for TQUV, or (c) up to the time of containment failure for TW and TC. Of little value to AE. Position switches not as important as valves themselves. Drain valve is usually opened to equalize pressure around MSIVs.
Service Air for Drywell Control Air	Solenoid Valve AAA502	As this system may support SRV/drywell cooler operation to maintain air to these systems, this valve's operability is as important as the SRV (high importance) or drywell cooling (low importance) systems.
Cooling Water	o Emergency Equipment Cooling Water (EECW) Limitorque SMB-000 o Reactor Building Closed Cooling Water (RBCCW) System Discharge Header Valve Limitorque SMB-00	Depending on the loads these systems serve, sustained opening or re-opening of these valves could be important in any of the accident sequences. However, review of available FSAR P&IDs does not show such valves inside containment.
Reactor Water Cleanup (RWCU)	o Isolation Valve Limitorque SMB-0	RWCU can act as another injection source therefore possibly requiring re-opening of this valve if the RWCU is isolated. With limited injection flow and when considering CRD, SLC and other systems, the importance of this valve's survivability is considered low.

*All entries are for Browns Ferry from References (1) and (2).

Table B-4. Inboard Containment Isolation Valves of Potential Interest to the PEEESAS Program (Continued)

SYSTEM	VALVE ELECTRICAL COMPONENTS*	COMMENTS
High Pressure Coolant Injection (HPCI) Reactor Core Isolation Cooling (RCIC) Systems	Inboard Isolation Valve RCIC - Limitorque SMB-00 HPCI - Limitorque SMB-2	These valves could be important in TC and TW sequences up to vessel breach. Valve opening must be sustained or if the system isolates, re-opening of the isolation valve could be important to restore core coolant injection. In TQUV, could be an important factor in restoring coolant injection before vessel breach. For TB, after AC restored, these valves may at first isolate due to high pump room temperature, etc. Re-opening of the valves under degraded conditions, before vessel breach, could be important. Systems have little influence on AE sequence. Overall, the importance of these valves is high.
Residual Heat Removal (RHR)	o Inboard Shutdown Cooling Valve Limitorque SMB-2	Consideration for this valve would be most important in the TW and possibly the TC sequences where containment heat removal is failed. Opening of this normally closed valve to restore a RHR heat removal path could be important as a means to prevent containment failure in these sequences. The importance of this valve is therefore considered high.
Containment Inerting/ Dilution Systems	o Inboard Torus Return Flow Solenoid Valves Valcor 526D o Suppression Chamber N_2 Solenoid Valves Target Rock 73FF-005	Unsure of how these valves may be important in mitigating actions unless they can be used as part of containment venting procedures. Reopening of these valves could then be important particularly in TW, TC sequences. Note that Standby Gas Treatment System pathways exist for containment venting which appear to have no in-containment valves. Therefore, the importance of these valves is considered low.

*All entries are for Browns Ferry from References (1) and (2).

Table B-4. Inboard Containment Isolation Valves of Potential Interest to the PEEESAS Program (Concluded)

SYSTEM	VALVE ELECTRICAL COMPONENTS*	COMMENTS
Miscellaneous	o Torus H_2 Analyzer Solenoid Valve Valcor 526D o Drywell H_2/O_2 Analyzer Valves Valcor 526D o Water Quality Sampling Valve and Position Switch ASCO WPHTX8300B68F Microswitch OPD-AR, OPD-AR30	As these are valves used for sampling and indication type functions, their relative importance to PEEESAS is considered low.

*All entries are for Browns Ferry from References (1) and (2).

3.12 Equipment - General Information

Miscellaneous information has been identified on such components as cabling, temperature elements, terminal blocks, etc. without specific applications identified. This information is listed here for potential future use by PEEESAS.

Browns Ferry Information (1&2):

Terminal blocks in drywell
GE Models CR-15182, EB-5

Cabling - Anaconda, Silicone Rubber Insulation

Limerick Information (3):

Cabling - XLPR insulation w/neoprene jacket
XLP2 insulation w/neoprene jacket
XLPE insulation w/neoprene jacket

Cooper Information (4):

Valve position switches - NAMCO D2400X

Solenoid valves - AVCO Models C-5450, C-5577, C-5140-8H, C-5140-4H
ASCO Model NP8320-A-193

Temperature element - American Std. (Copper Constantan or Iron Constantan)

Terminal block - GE Model EB5 - CR151A6, Buchanon Model 514

Cabling - Boston Insulated Wire LSS-1942B, 993-H-002
Kerrite
Cerro
Raychem 10483 Coax

On the basis of the information presented in Sections 3.1 - 3.12, it appears that a few components located within containment could be important to mitigate or assess plant conditions for some of the selected accident sequences (Task 1). Survivability during a severe accident is most important for these components so that they can be relied upon during the accident. These components and the accident sequences for which their continued functionality is most important are presented in Table B-5 which follows Section 3.12. In later tasks, the PEEESAS program will examine the qualification and selected accident sequence environmental profiles for these components to see if the accident profiles appear significantly worse than the qualification requirements. For the components in Table B-5 which could mitigate accident sequences, the ability to use the component during the severe accident would (1) provide restoration of a core heat removal or coolant injection system or (2) allow for low vessel pressure operation so that low pressure coolant injection systems could be used.

Table B-5. Components Recommended for Further Examination by the PEEESAS Program

COMPONENT	SEQUENCE*	TIME PERIOD*	USEFULNESS
Inboard MSIV Solenoid Valves	TW, TC TQUV, TB (after AC restored)	- Containment failure - Vessel breach	Reopening of MSIVs will restore a heat rejection path to avoid containment failure in TW or TC and possibly a core melt in TQUV or TB (if feedwater is also supplied).
HPCI, RCIC Inboard Isolation Valves	TW, TC, TQUV, TB (after AC restored)	Vessel breach	Reopening or sustained opening of the valves provides core cooling and hence prevents core melt.
Pilot Valves and Service Air Solenoid Valves for SRVs	TW, TC, TQUV, TB (after AC restored)	Vessel breach	Sustained functionality allows for low reactor vessel pressure operation to sustain or restore low pressure injection to the core, thus preventing core melt.
RHR Inboard Shutdown Cooling Valve	TW, TC	Containment failure	Opening of this valve to provide a RHR cooling path could be important in preventing containment overpressurization and failure.
In-Core Thermocouples and Reactor Vessel Surface Thermocouples	TW, TC, TQUV, TB, AE	Vessel breach	Provide status of core cooling adequacy.
Drywell Temperature Element (RTD)	TB, TQUV, TW	Containment failure	Provides status of D.W. cooler operation and an indication of margin to containment failure.
Drywell Pressure Monitor	TB, TQUV, TW	Containment failure	Provides indication of containment venting effectiveness and margin to containment failure.
Drywell H_2 and Radiation Monitor	TB, TW, TC, TQUV, AE	Beyond containment failure	Provides estimate of core condition and release of fission products.
Cabling, Connectors/Splices, Terminal Blocks for Above Components	See Above	See Above	See Above

*Listed are those sequences for which survivability of the component could be most important for mitigating the accident or providing plant status information up to the time period indicated.

4.0 OTHER EQUIPMENT CONSIDERATIONS

Section 1.0 indicated that while the focus of the PEEESAS program is on electrical components located within containment, equipment outside containment could also be worthy of examination. Numerous components in the reactor building makeup the core cooling and containment cooling systems. These include such devices as valves, pumps, sensors, actuation and control hardware, etc. If containment should fail, the continued operability of these systems could prevent core melt or at least lessen the consequences of the accident. Before containment failure, operability or restoration of these systems could occur under stressful environmental conditions depending on the state of the core.

One system is particularly important to note even though all its electrical components are outside the primary containment barrier. This is the Standby Gas Treatment System (SBGTS). The operation of this system during a severe accident could lessen the consequences of the accident by filtering some radioactive fission products before any release occurs to the environment. In addition, its operation as part of the containment venting process could prevent containment failure as might occur in the TW and TC sequences. Operation of this system might have to proceed under degraded core conditions including air flow within the system which might be higher in temperature and radiation levels than for which the system is qualified. Therefore, it is recommended that if equipment outside containment should ever come under review, the survivability of the SBGTS should be considered.

5.0 SUMMARY

This task report presents the results to date of Task 3 of the Sandia PEEESAS program. Based on the information thus far, electrical components worthy of further examination in the PEEESAS program include valve motors, pilot valves for safety/relief valve operation, solenoid valves, cabling, connectors, splices, and terminal blocks (see Table 5). These electrical component may be most important in preventing, mitigating, or assessing plant conditions for the selected severe accidents in a BWR-4, Mark I plant design. These components are located within the primary containment where environmental conditions are generally more severe. It is this equipment, whose survivability and functionality during severe accidents could be important and yet may be questionable, that is most worthy of review.

6.0 REFERENCES

1. Browns Ferry Nuclear Power Station, Units 1, 2 and 3, Safety Evaluation of Environmental Qualification of Safety-Related Electrical Equipment, USNRC, January 1983.

2. Browns Ferry Nuclear Plant Units 1-3, Response to NRC IE Bulletin 79-01B - Environmental Qualification of Class 1E Equipment, Tennessee Valley Authority, 1980.

3. Limerick Generating Station, Unit 1, Electrical Equipment Environmental Qualification Report, Philadelphia Electric Co., October 1983.

4. Cooper Nuclear Station, Technical and Safety Evaluation Reports of Environmental Qualification of Safety-Related Electrical Equipment, USNRC, December 1980 and June 1981.

5. Interim Reliability Evaluation Program: Analysis of Browns Ferry Unit 1 Nuclear Plant, EG&G Idaho, Inc., July 1982.

APPENDIX C

GENERATION OF ENVIRONMENTAL PROFILES FOR SELECTED ACCIDENT SEQUENCES

TABLE OF CONTENTS

TABLE OF CONTENTS (CONTINUED)

TABLE OF CONTENTS (CONCLUDED)

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES (CONTINUED)

LIST OF FIGURES (CONTINUED)

LIST OF FIGURES (CONTINUED)

LIST OF FIGURES (CONTINUED)

LIST OF FIGURES (CONTINUED)

LIST OF FIGURES (CONCLUDED)

1.0 INTRODUCTION

1.1 Areas Addressed By This Appendix

This appendix focuses on defining the environmental profiles in the containment for the five selected severe accidents and comparing these profiles to typical equipment qualification profiles. Areas are identified where the environmental profile exceeds the qualification profile.

1.2 Organization

This appendix is organized into 7 sections

- Section 1.0 includes the specific areas addressed in the appendix and the organization of the document.

- Section 2.0 contains environmental profiles. It presents the methodology and results of the effort to define pressure and temperature vs. time plots for each of the five severe accidents being examined.

- Section 3.0 presents typical qualification profiles. These profiles are based on IEEE 323-1974 Appendix A.

- Section 4.0 is the results section. It presents the comparison of the environmental profiles from Section 2.0 and the qualification profiles from Section 3.0.

- Section 5.0 reviews environments, other than the pressure and temperature environments considered in Section 2.0, occurring during a severe accident.

- Section 6.0 contains a brief report summary.

- Section 7.0 contains the reference list.

2.0 ENVIRONMENTAL PROFILES

2.1 Introduction

This section of the appendix develops the accident environmental profiles. Each of the 14 "likely scenarios" and their profiles are discussed. Table C-1 summarizes the 14 likely scenarios for the five major accident sequences examined (see Ref. 1). Computer models described in Sections 2.2 and 2.3 are used to construct the profiles. It is important to note that most of the constructed profiles represent a composite of several data sources and therefore must be considered general in nature. However, the good agreement between current LTAS computer model results and past Meltdown Accident Response CHaracteristics (MARCH) code results lends confidence to the accuracy of the general trends and approximate levels for the developed profiles.

2.2 Long Term Accident Sequence code (LTAS)

The major source of data concerning pressure and temperature environments up to the point of core damage was the LTAS code (see Ref. 4). This code was originally developed to study the station blackout sequence at Browns Ferry Unit 1, a boiling water reactor (BWR). Since the code was written specifically for the plant chosen by the PEEESAS program, it was ideally suited to model thermohydraulic environmental behavior up to the point of core damage. The code was expanded by Oak Ridge National Laboratory personnel from its original version to investigate other accident sequences. The version used for this study modeled all five accident sequences. The major advantage of the LTAS code is its ability to simulate a wide variety of possible operator actions. This allows very tight control over scenario definition in terms of equipment operating and operator control of the equipment. It is also relatively fast and inexpensive to run compared to other codes. The code demonstrates good agreement with other models and test data lending confidence to its ability to accurately predict plant response up to the point of core damage. Originally the PEEESAS program had intended to construct the

TABLE C-1 - LIKELY ACCIDENT SCENARIOS FOR EXAMINATION BY THE PEEESAS PROGRAM

ACCIDENT CATEGORY	SEQUENCE DESCRIPTION	DISTINGUISHING FEATURE WITH SEQUENCE	SCENARIO NUMBER
TB	TOTAL BLACKOUT WITH HPCI/RCIC FAILURE.	OPERATOR DEPRESSURIZES RPV	1
		NO OPERATOR ACTION	2
		STUCK OPEN RELIEF VALVE	3
	TOTAL BLACKOUT WITH HPCI/RCIC AVAILABLE UNTIL BATTERY FAILURE	OPERATOR DEPRESSURIZES RPV	4
		STUCK OPEN RELIEF VALVE	5
TW	TRANSIENT WITH NO RHR AVAILABLE FOR SUPPRESSION POOL COOLING. DRYWELL COOLER FAILURE AT 17 HOURS. SRVs FAIL AFTER 24 HOURS.	OPERATOR DEPRESSURIZES RPV	6
		STUCK OPEN RELIEF VALVE	7
TC	TRANSIENT WITHOUT SCRAM. MANUAL ROD INSERTION AND LIQUID BORATION ARE FAILED LEVEL CONTROL, DRYWELL COOLER OPERATOR, AND MANUAL SRV CONTROL EVENTUALLY FAIL.	OPERATOR PERFORMS SUPPRESSION POOL COOLING OPERATOR DEPRESSURIZES RPV MANUAL LEVEL CONTROL (MSIVs CLOSED)	8
		AS ABOVE FOR A MSIV OPEN EVENT	8A
		OPERATOR PERFORMS SUPPRESSION POOL COOLING STUCK OPEN RELIEF VALVE/OPERATOR DEPRESS. RPV MANUAL LEVEL CONTROL (MSIVs CLOSED)	9
TQUV	TRANSIENT WITHOUT HPCI/RCIC/LPCS/LPCI AVAILABLE. CRD FLOW IS AVAILABLE BUT NOT SUFFICIENT TO PREVENT CORE DAMAGE.	OPERATOR DEPRESSURIZES RPV	10
		STUCK OPEN RELIEF VALVE	11
		OPERATOR LEAVES RPV AT OPER. PRESSURE	12
AE	LARGE LOCA WITH NO HPCI/RCIC/LPCS/LPCI AVAILABLE	NO OPERATOR ACTION	13

environmental profiles based only on past studies. The addition of the LTAS data represents a significant improvement in the ability to accurately depict accident environmental profiles.

2.3 Other Data Sources

As mentioned previously, the LTAS code does have the limitation of not being able to accurately model thermohydraulic response of the core or containment past the point of core damage. Therefore, other sources of information were used to supplement the LTAS results once core damage was predicted. Primarily, the Meltdown Accident Response CHaracteristics (MARCH) code results were used. Although somewhat cumbersome to run, the MARCH code is designed to simulate plant response throughout the core damage period. Some of the past studies consulted used MARCH results exclusively, while others used MARCH to provide inputs to other specialized codes like MERGE, CORSOR, VANESA, TRAP-MELT, and SPARC. The MARCH code and these other codes used in past studies form the other data sources used in construction of the accident profiles (Refs. 3,5,6,8,9,& 10).

2.4 Methodology

This section discusses the methodology used to construct the environmental profiles. As previously mentioned, LTAS data is used up to the point of core damage. For each scenario a LTAS input deck was constructed specifying the appropriate operator actions, plant conditions, and accident parameters. The code was then compiled and run. These results were graphed to form the environmental profile up to the point of core damage. The LTAS results were then compared to MARCH runs for a similar sequence. If the LTAS and MARCH results agreed, it was assumed the MARCH data was constructed from similar plant parameters and operator actions. Therefore, the LTAS code performs two important functions: (1) it serves as an independent data source for the sequence being examined and (2) it helps to verify other code data was generated under similar initial conditions. After core damage, results from the selected MARCH runs were used to complete the environmental profile.

This process was completed for all 14 scenarios. In cases where the environmental profiles for two or more scenarios were found to be very similar, a composite profile was constructed. The data was then graphed on the same scale as the typical qualification profiles to allow overlay comparisons between the environmental profiles and the qualification profiles.

2.5 Results

2.5.1 TB (short term)

The first accident sequence consists of a complete station blackout without any injection available. The first variation on this sequence, scenario 1, involved the operator depressurizing the reactor vessel 40 minutes into the accident. The LTAS results for the two methods of vessel water level indications and the actual water level are shown in Figure C-1. The Top of the Active Fuel (TAF) at Browns Ferry is at 360 inches. Note that Figure C-1 indicates the TAF is actually uncovered at about 1600 seconds (27 min.) into the accident. This is in good agreement with the 33 minutes predicted by MARCH data for this sequence. Therefore LTAS data was used up to 1600 seconds and the MARCH data was used after that to complete the profile. Figures C-2 and C-3 show the LTAS results for drywell and suppression pool temperatures and drywell pressure. Table C-2 information (Ref. 9, pg.110 and 145) contains the MARCH sequence of events used to complete the profiles after the 1600 second point and a drywell pressure versus time plot for this sequence. All of this data is combined to form the final profiles of drywell temperature, suppression pool temperature, and drywell pressure shown as Figures C-4, C-5, and C-6 respectively. Each of the profiles presented in Figures C-4, C-5, and C-6 is discussed in some detail in the following paragraphs. These profiles represent a generic example of environmental progressions during a severe accident where vessel breach precedes containment failure.

Figure C-4 shows the expected rapid rise in drywell temperature due to loss of the drywell coolers at the accident initiation and the heating up of the

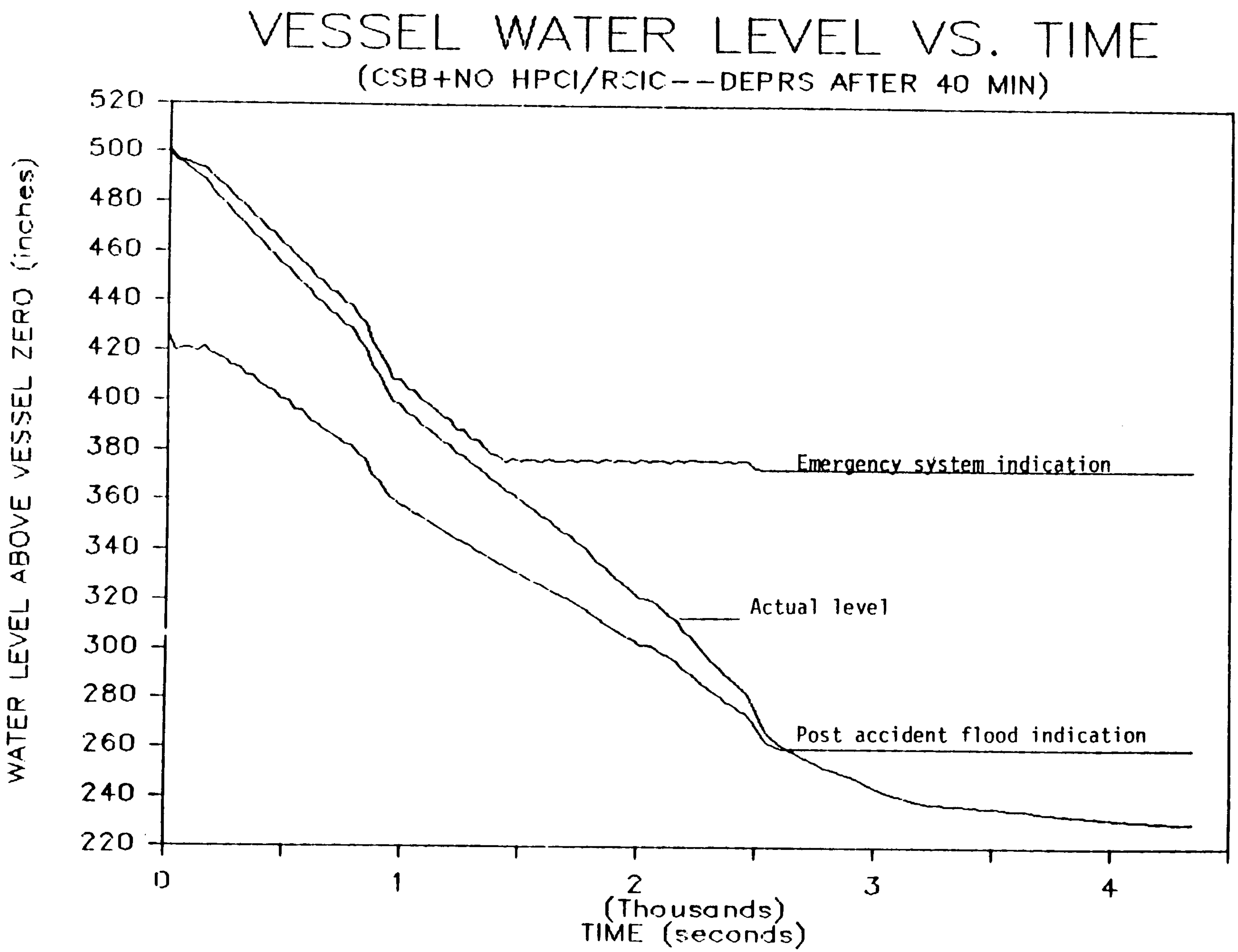


FIGURE C-1 - LTAS PREDICTED WATER LEVELS FOR THE FIRST SCENARIO

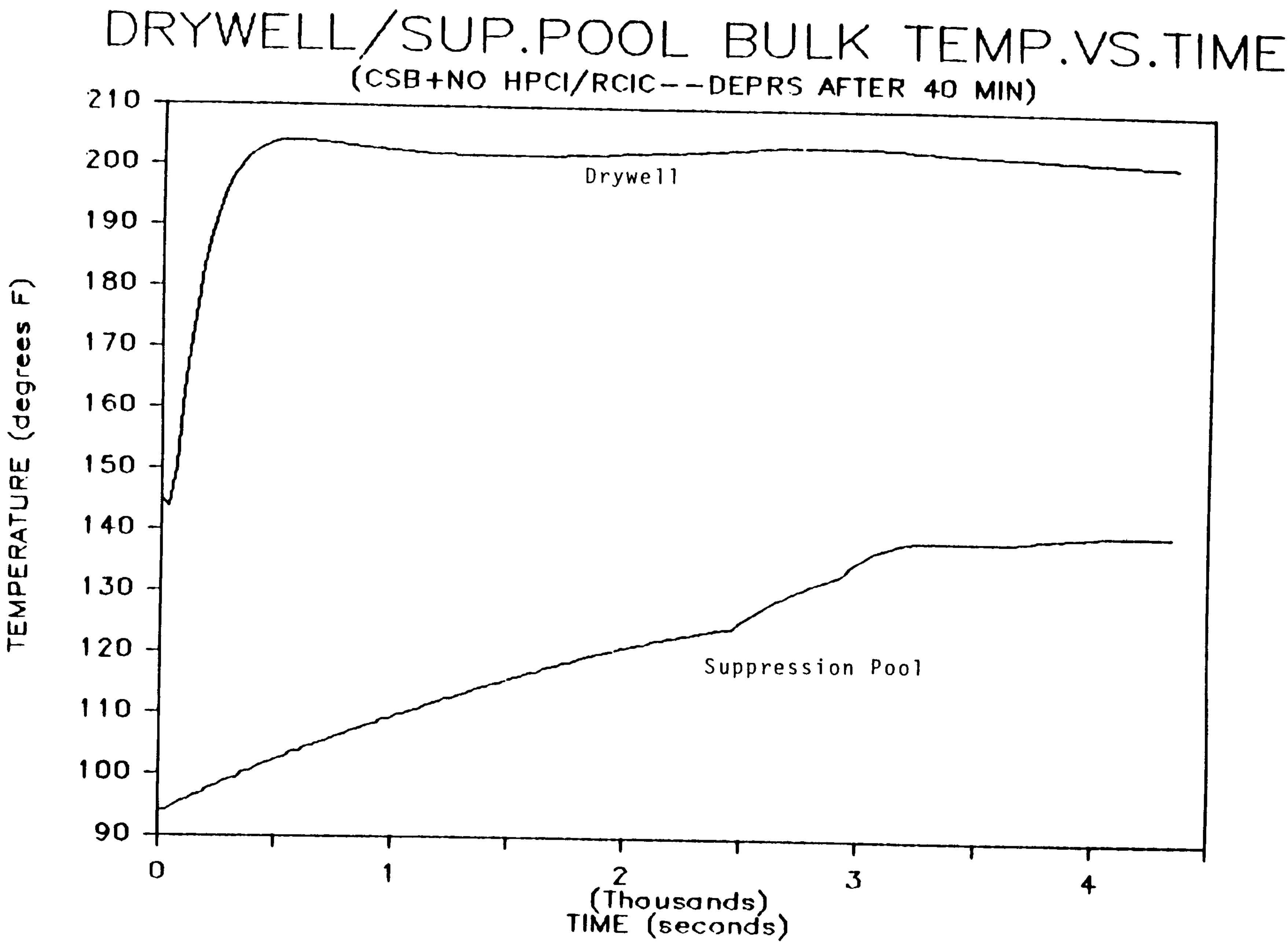


FIGURE C-2 - LTAS PREDICTED DRYWELL AND WETWELL TEMPERATURES FOR THE FIRST SCENARIO

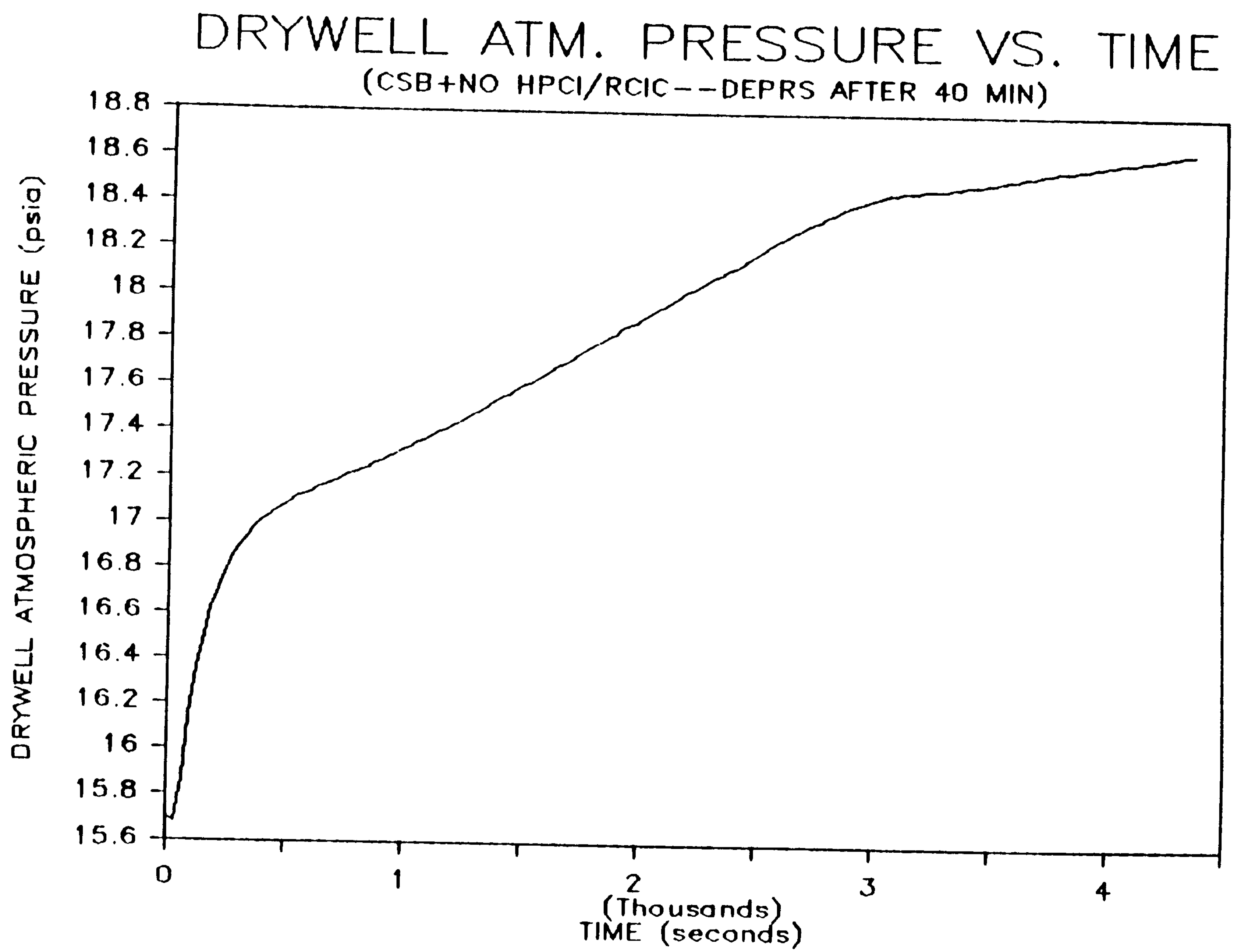


FIGURE C-3 - LTAS PREDICTED CONTAINMENT PRESSURE FOR THE FIRST SCENARIO

Table C-2

Browns Ferry Nuclear Plant: Complete Station Blackout
Sequence of Events

CSB + No HPCI/RCIC

(TUB′)

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.
3.0	Turbine trips off (turbine stop valves fully closed).

Time (sec)	Event
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Seven (7) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 0.51 m (20 in.) from the previous momentary 1.02 m (40-in.) drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 7 S/RVs are completely closed.
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 5.98 m (235.50 in.) above Level 0, or 5.00 m (196.44 in.) above TAF.
22.0	Suppression pool water average temperature rises to 35.13°C (95.24°F) in response to the first S/RV pops.
29.0	All 4 S/RVs are completely closed.
29.7	Two out of 13 S/RVs start to open.

Time (sec)	Event
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 35.3°C (95.54°F).
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 5.50 m (216.00 in.) above Level 0, or 4.50 m (176.94 in.) above TAF.
90.0	Suppression pool water average temperature is approximately 35.4°C (95.72°F).
101.0	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the sequence.
625	Wide range sensed water level reaches low water level set-point (Level 2), i.e., 4.18 m (164.50 in.) above Level 0 at 2/3 core height, or 2.96 m (116.50 in.) above TAF.
625	HPCI and RCIC systems are not turned on because they are assumed to be unavailable.
20 min.	Suppression pool water average temperature reaches 46°C (114°F).
33 min.	Core uncovery time. Steam-water mixture level is at 3.54 m (11.61 ft) above bottom of the core.
40 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The HPCI/RCIC systems are not affected. Drywell and wetwell temperature are 72°C (162°F) and 55°C (130°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	33	4.36×10^{3}	9.25×10^{7}	5.26×10^{6}
Hydrogen	6×10^{-9}	8.62×10^{-7}	2.92×10^{-2}	1.66×10^{-3}

60 min. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	15.6	2.07×10^{3}	5.06×10^{7}	2.88×10^{6}
Hydrogen	2.8×10^{-3}	3.76×10^{-1}	2.15×10^{4}	1.22×10^{3}

Time (sec)	Event
70 min.	Core melting starts.
80 min.	Drywell and wetwell temperatures are 75°C (167°F) and 63°C (145°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	5.68	7.51×10^2	2.22×10^7	1.26×10^6
Hydrogen	0.19	2.53×10^1	2.29×10^6	1.30×10^5

Time (sec)	Event
96 min.	Water level in vessel drops below bottom grid elevation.
97 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
99 min.	The corium slumps down to vessel bottom.
101 min.	The debris is starting to melt through the bottom head. Drywell and wetwell temperatures are 97°C (207°F) and 71°C (159°F), respectively. Meanwhile, local pool water temperature at the discharging bay exceeds 149°C (300°F). Steam condensation oscillations could accelerate due to the continuous discharge of superheated noncondensable gases into the suppression pool. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	18.6	5.46×10^3	5.42×10^7	3.08×10^6
Hydrogen	6.8×10^{-2}	8.93	3.59×10^5	2.04×10^4

Time (sec)	Event
129 min.	Vessel bottom head fails, resulting in a pressure increase of 0.34 MPa (49 psia).
129.03 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 1546°C (2815°F) initially. Internal heat generation in metals and oxides are 1.36×10^7 and 2.50×10^7 watts, respectively.

Time (sec)	Event
165 min.	Drywell and wetwell temperatures are 141°C (286°F) and 74°C (166°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	5.46	722.83	1.59×10^5	9052
Hydrogen	3.3×10^{-2}	4.38	0	0
CO_2	2.58	341.88		
CO	0.69	91.35		

Time (sec)	Event
190 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 204°C (400°F) and start to vent through the primary containment.
193 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment.
219 min.	Drywell and wetwell pressures are at 0.10 MPa (14.7 psia). Drywell and wetwell temperatures are 598°C (1109°F) and 78°C (173°F), respectively. Mass and energy addition rates into the drywell are:

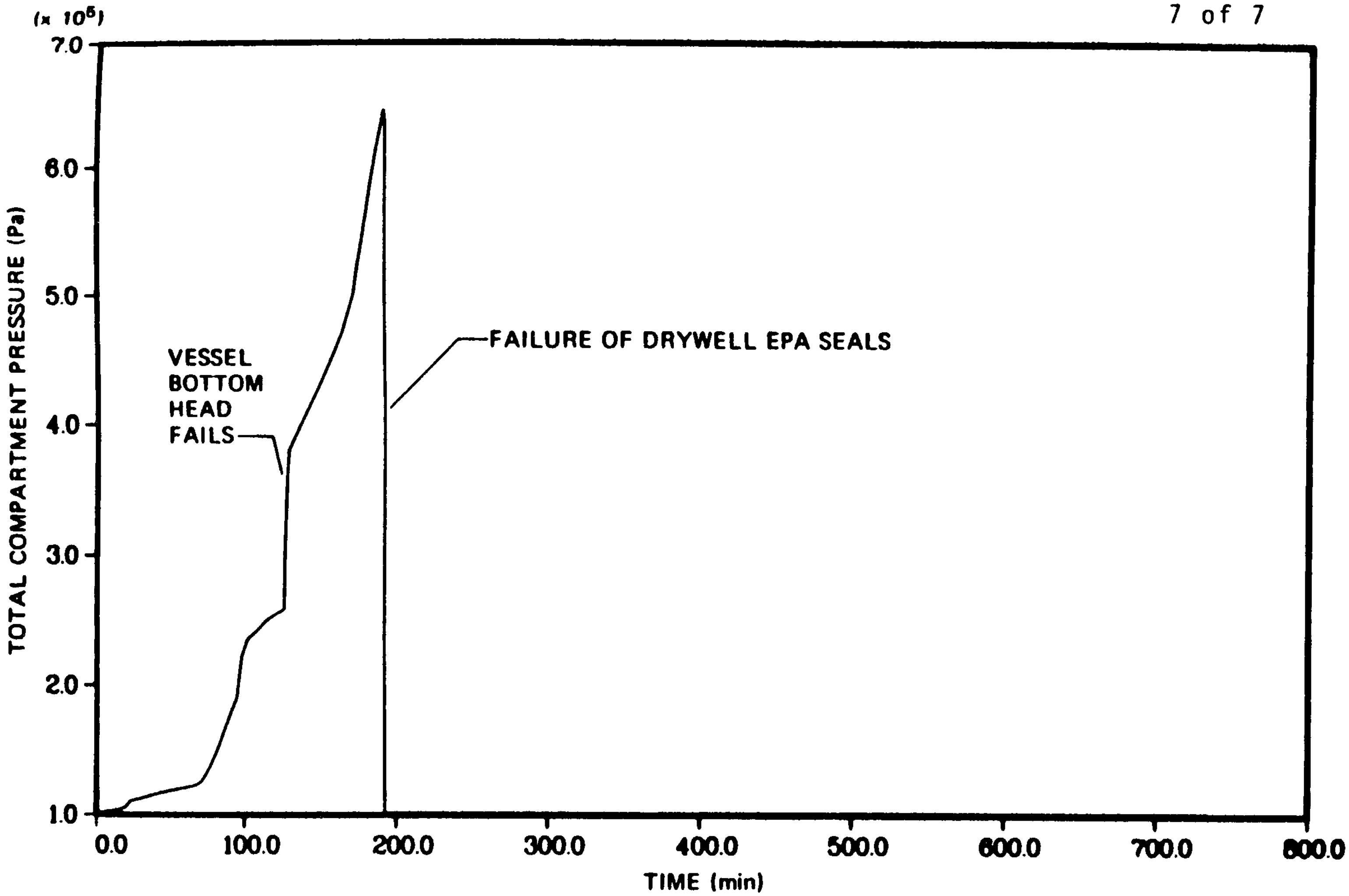
	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	0.70	92	1.59×10^5	9052
Hydrogen	0.24	32	0	0
CO_2	2.32	307		
CO	5.03	666		

The leak rate through the containment failed areas is $\sim 2.90 \times 10^5$ ℓ/s ($\sim 6.15 \times 10^5$ ft^3/min).

Time (sec)	Event
250 min.	Drywell and wetwell temperatures are 675°C (1247°F) and 78°C (173°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	6.84	905	1.59×10^5	9052
Hydrogen	0.25	33	0	0
CO_2	1.53	203		
CO	5.25	695		

Time (sec)	Event
	The leak rate through the containment failed area is ~4.91 × 10^4 ℓ/s (~1.04 × 10^5 ft^3/min).
309 min.	Rate of concrete decomposition is ~4.65 × 10^4 gm/s. Rate of heat added to atmosphere is ~1.20 × 10^4 kW.
367 min.	Drywell and wetwell pressures are at 0.10 MPa (~14.7 psia) and temperatures are 854°C (1570°F) and 77°C (171°F), respectively. The leak rate through the containment failed area is ~3.94 × 10^4 ℓ/s (~8.35 × 10^4 ft^3/min).
733 min.	Drywell and wetwell temperatures are 546°C (~1014°F) and 77°C (170°F), respectively. The leak rate through the containment failed area is ~2.12 × 10^3 ℓ/s (~4.50 × 10^3 ft^3/min).



Drywell total pressure (TUB´).

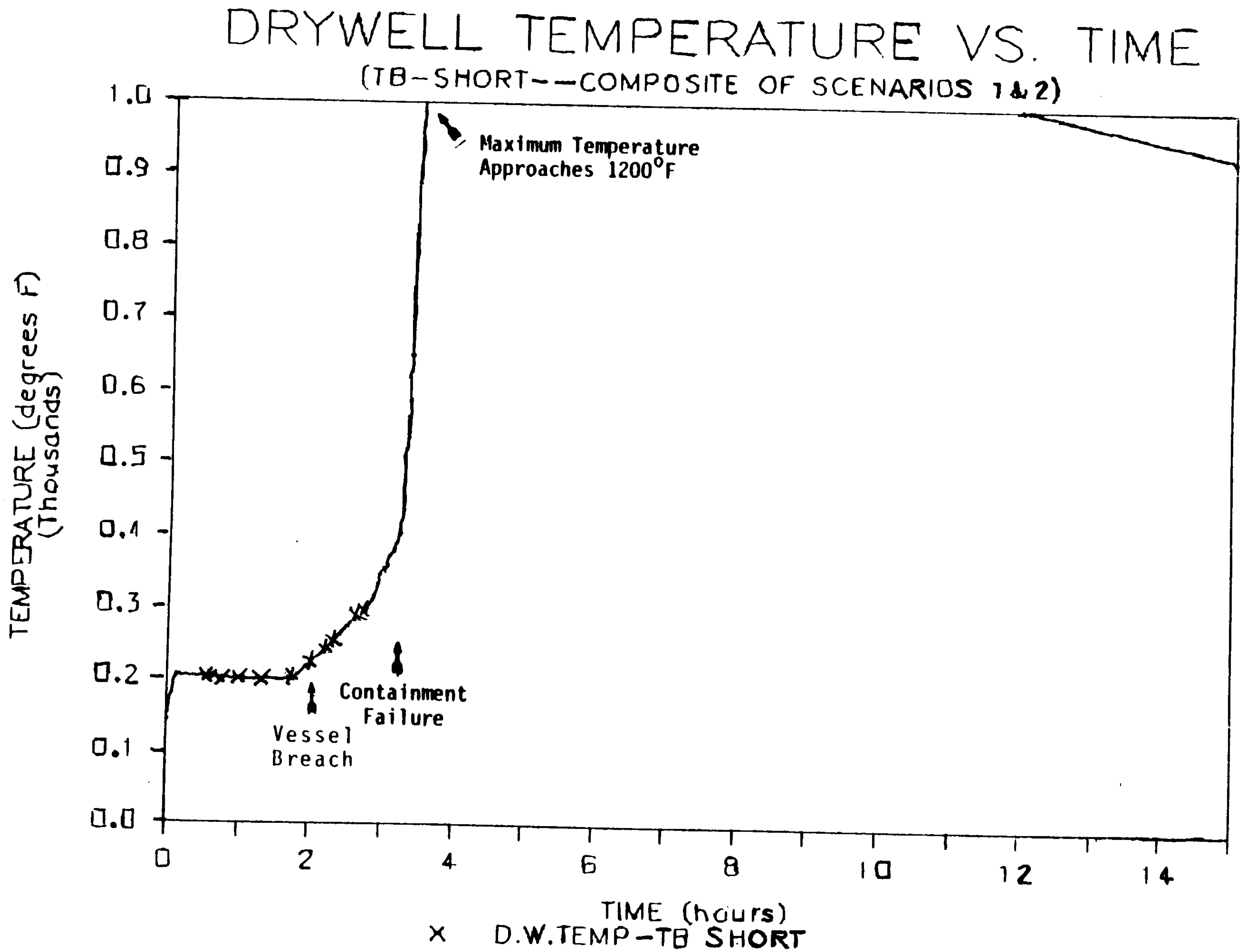


FIGURE C-4 - DRYWELL TEMPERATURE PROFILE FOR THE FIRST AND SECOND SCENARIOS

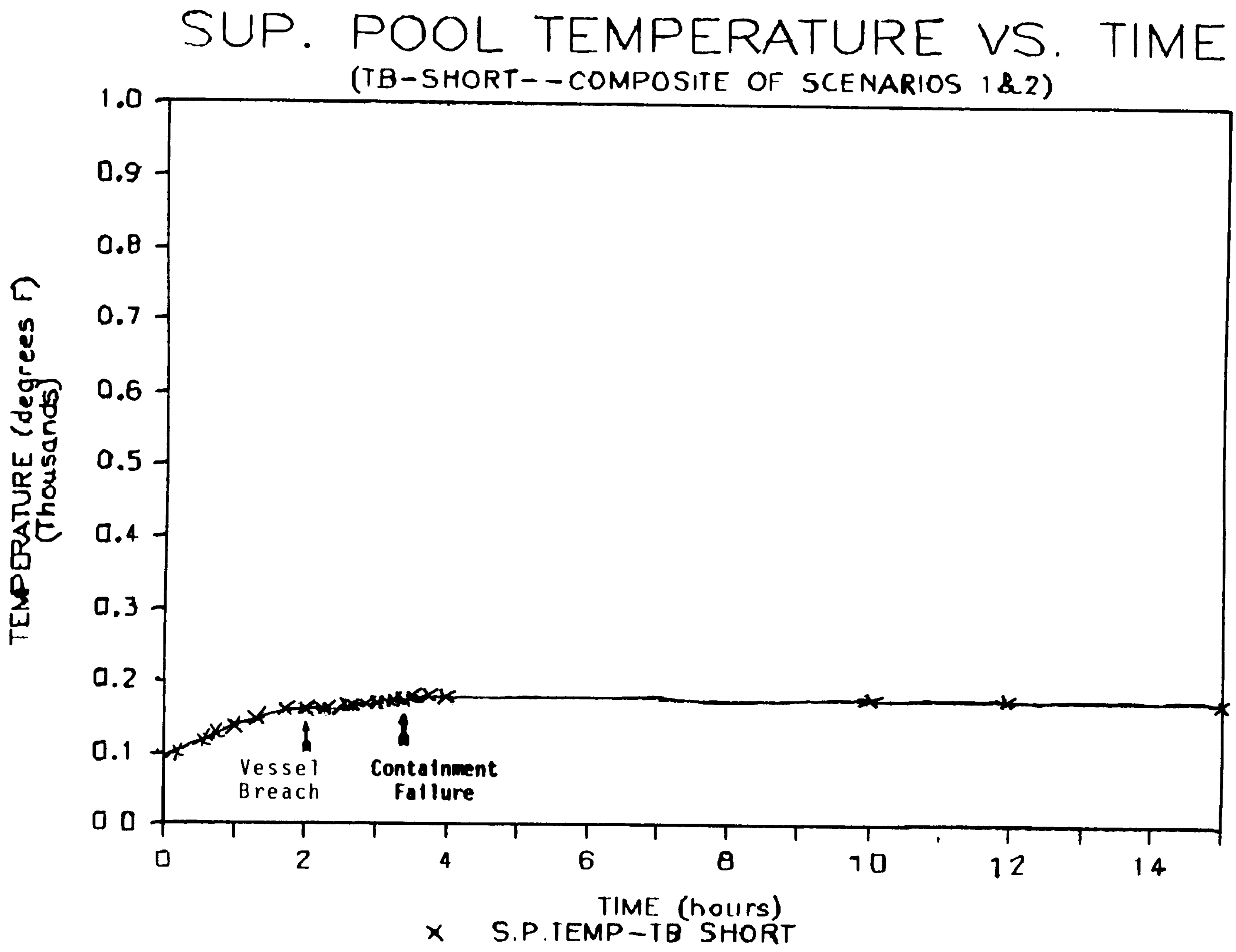


FIGURE C-5 - WETWELL TEMPERATURE PROFILE FOR THE FIRST AND SECOND SCENARIOS

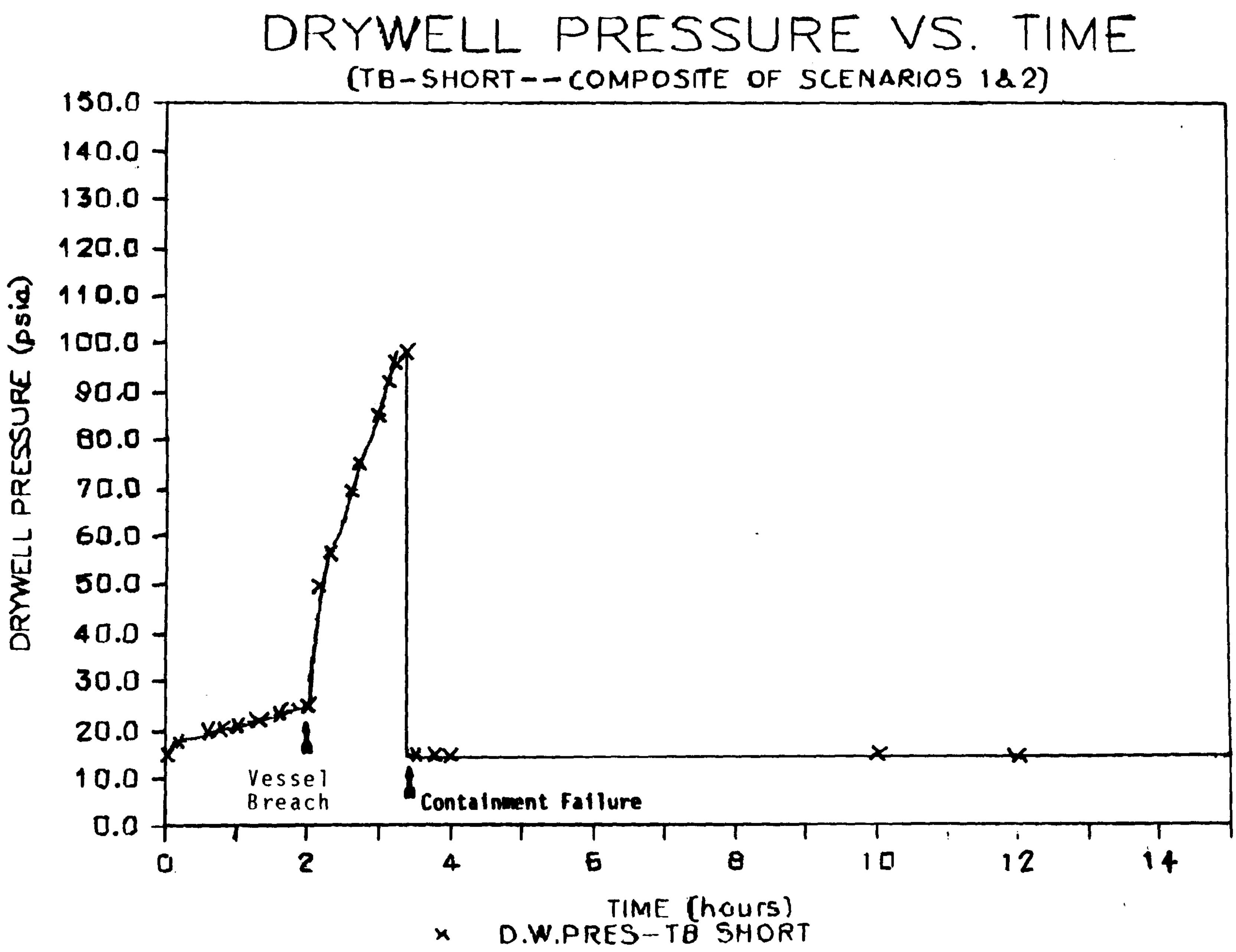


FIGURE C-6 - CONTAINMENT PRESSURE FOR THE FIRST AND SECOND SCENARIOS primary system due to immediate loss of core coolant injection. During this portion of the accident, decay heat energy is removed to the wetwell (suppression pool) by intermittent safety relief valve operation during the boil-off process. Therefore, drywell temperature quickly approaches a new equilibrium value which just compensates for the lost heat removal capacity of the drywell coolers. In the meantime, boil-off has continued leading to core uncovery, the beginning of core melt, and reactor vessel breach. Once vessel breach occurs, decay heat energy has direct access to the drywell atmosphere. This energy release causes an increase in the temperature and pressure of the containment atmosphere (see Figure C-6). This trend continues until the containment fails at which point almost all the energy previously stored within the closed containment system is now released. The heat energy previously applied to elevate the pressure within the containment,now contributes to a rapid temperature spike after which the system slowly begins to approach atmospheric equilibrium. In this particular case, containment failure occurs due to electrical penetration failure under high temperature conditions of 500°F. If the penetrations did not fail, the pressure would increase to the point of containment failure (estimated range from 117 psia to 132 psia depending on the initiating conditions and the accident scenario examined). Based on the size of the containment breach and the resulting leak rate, the depressurization could be rapid or slow. Regardless, the general trend of the profiles and the absolute values attained would not change significantly.

The profile of the suppression pool temperature is shown in Figure C-5. Note that while water still exists in the reactor vessel, the pool serves as a heat sink for the transfer of decay heat energy to the wetwell through the safety relief valves (SRVs). The rate of temperature rise in the suppression pool decreases from the start of the accident to the time of vessel breach as the fixed water inventory depletes itself through the SRVs. Once the core is uncovered and vessel breach occurs, the only form of pool temperature rise is from radiative heating by the drywell atmosphere. Thus the slope of the temperature curve decreases. This trend continues until the time of containment failure when the suppression pool temperature will join the drywell temperature in slowly approaching atmospheric equilibrium.

The pressure profile is shown in Figure C-6. The general shape of this curve is indicative of accidents where reactor vessel breach precedes containment failure. The initial quick rise in pressure is primarily due to failure of the drywell coolers. This parallels the drywell temperature profile shown in Figure C-4. Once the new equilibrium is reached due to the loss of drywell coolers, a slow rise in pressure due to radiative heating of the containment atmosphere by the reactor vessel is observed. This continues until vessel breach at which time the corium has direct contact with the containment atmosphere causing a rapid rise in containment pressure to the point of failure at approximately 110 psia.

The second variation on the short term TB sequence (scenario 2) involved a case of no operator action so the reactor vessel remains at pressure. Results from this run were so similar to the depressurized case that the environmental profiles constructed for scenario 1 were also used to represent this scenario. Hence, Figures C-4, C-5, and C-6 are labeled as composites for scenarios 1 and 2. Figure C-7 is a plot of drywell and suppression pool temperature overlayed for these two scenarios demonstrating the similarities of these cases. The fact that most of the boil-off occurred before the operator depressurized in the first scenario accounts for the similarities in the environmental response.

The last scenario for a short term station blackout involved a stuck open relief valve at 600 seconds into the accident (scenario 3). Note that this scenario is representative of what would have occurred if the operator had depressurized earlier in the first sequence. The shapes of the environmental profile curves are generally the same as seen in the first scenario. Figure C-8 shows the LTAS results for water level behavior over the course of the accident. The LTAS results indicate that the core uncovers about 900 seconds (15 min.) into the accident. Again, allowing for differences in the time the two codes have the relief valve sticking open, this figure is in good agreement with results of the MARCH sequence shown in Table C-3. The

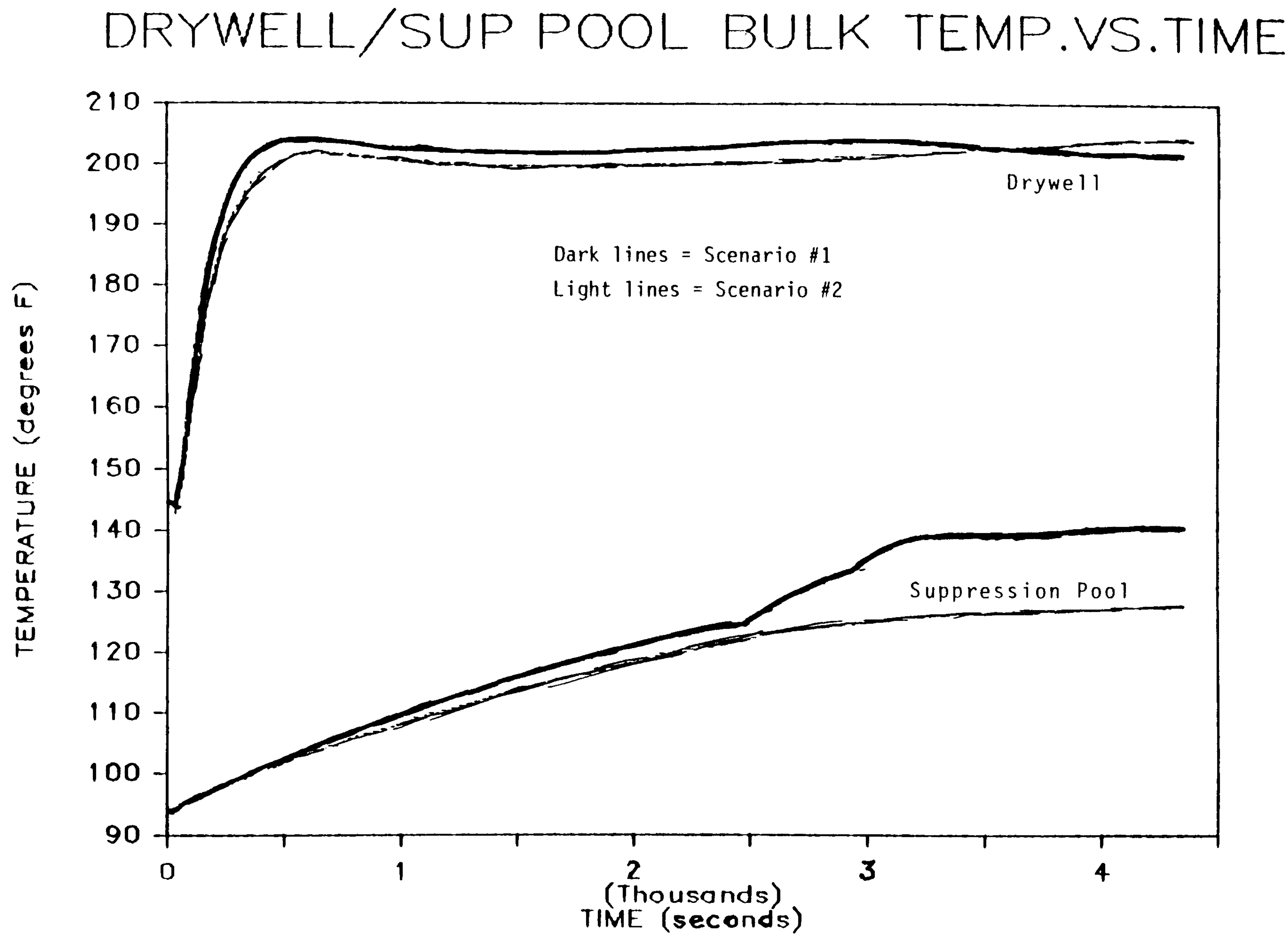


FIGURE C-7 - LTAS PREDICTED DRYWELL AND WETWELL TEMPERATURES COMPARED FOR THE FIRST AND SECOND SCENARIOS

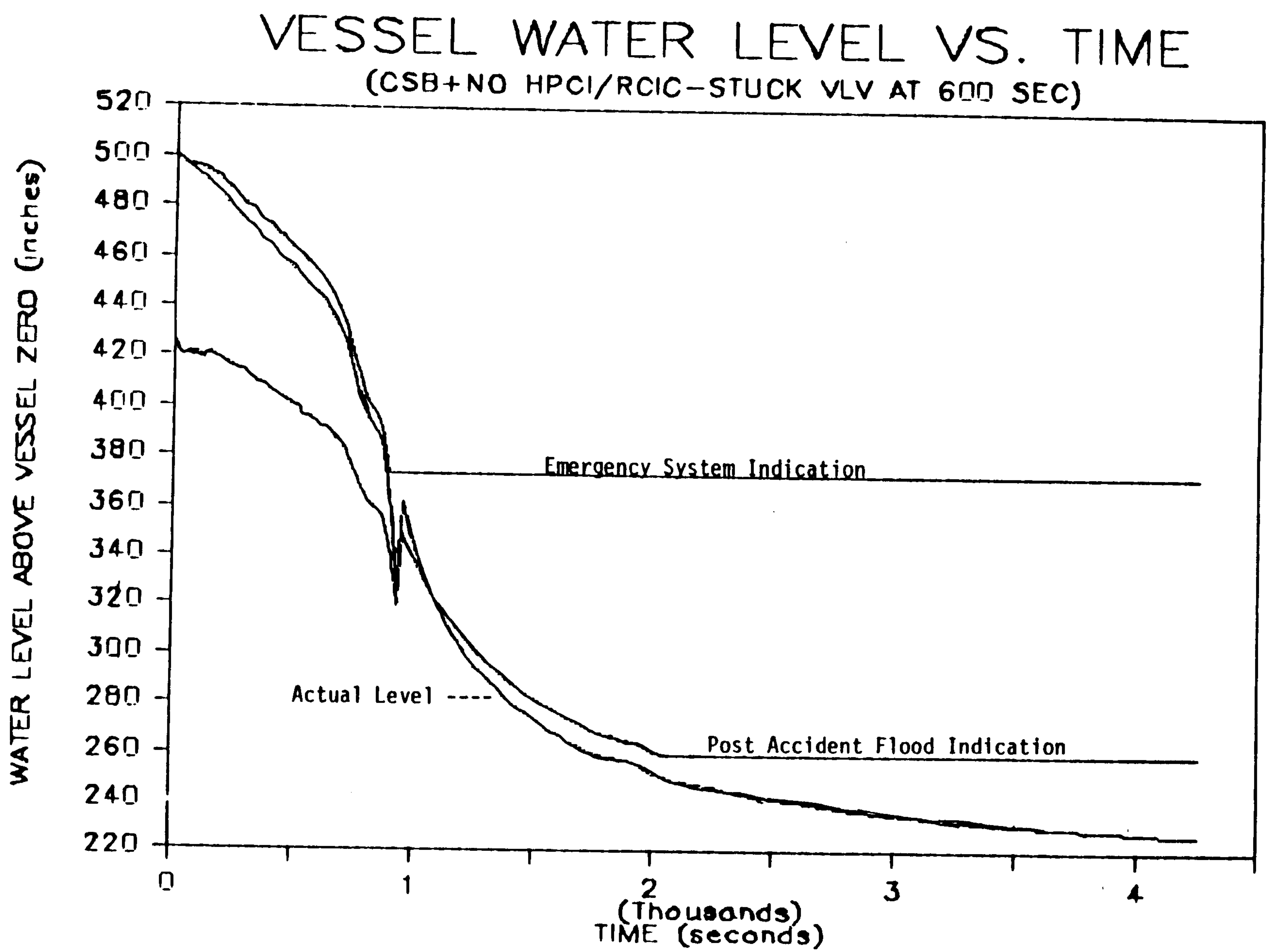


FIGURE C-8 - LTAS PREDICTED WATER LEVEL FOR THE THIRD SCENARIO

Table C-3

Browns Ferry Nuclear Plant: Complete Station Blackout Sequence of Events

CSB + No HPCI/RCIC & SORV (Small Break LOCA)

(TUPB′)

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.
3.0	Turbine trips off (turbine stop valves fully closed).

Time (sec)	Event
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Seven (7) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 0.51 m (20 in.) from the previous momentary 1.02 m (40-in.) drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 6 S/RVs are completely closed. One S/RV is stuck open (SORV); this has the same effect as a small break LOCA of equivalent break area of 0.015 m^2 (0.1583 ft^2).
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 5.98 m (235.50 in.) above Level 0, or 5.00 m (196.44 in.) above TAF.
22.0	Suppression pool water average temperature rises to 35.13°C (95.24°F) in response to the first S/RV pops.
29.0	All 4 S/RVs are completely closed.
29.7	Two out of 13 S/RVs start to open.

Time (sec)	Event
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 35.3°C (95.54°F).
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 5.50 m (216.00 in.) above Level 0, or 4.50 m (176.94 in.) above TAF.
90.0	Suppression pool water average temperature is approximately 35.4°C (95.72°F).
101.0	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the sequence.
625	Wide range sensed water level reaches low water level set-point (Level 2), i.e., 4.18 m (164.50 in.) above Level 0 at 2/3 core height, or 2.96 m (116.50 in.) above TAF.
625	HPCI and RCIC systems are not turned on because they are assumed to be unavailable.
17.2 min.	Core uncovery time. Steam-water mixture level is at 3.62 m (11.88 ft) above bottom of the core.
20 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The HPCI/RCIC systems are not affected. Drywell and wetwell temperature are 73°C (163°F) and 55°C (130°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	60	7.94×10^{3}	1.69×10^{8}	9.61×10^{6}
Hydrogen	8.53×10^{-13}	1.13×10^{-10}	3.19×10^{-6}	1.81×10^{-7}

Time (sec)	Event
40 min.	Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	10.30	1.36×10^{3}	3.41×10^{7}	1.94×10^{6}
Hydrogen	2.38×10^{-4}	3.15×10^{-2}	1.61×10^{3}	91.56

Time (sec)	Event
56.6 min.	Core melting starts.
60 min.	Drywell and wetwell temperatures are 75°C (167°F) and 63°C (145°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	2.73	3.61×10^2	1.23×10^7	6.99×10^5
Hydrogen	0.49	6.48×10^1	7.05×10^6	4.01×10^5

Time (sec)	Event
78 min.	Water level in vessel drops below bottom grid elevation.
79 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
81 min.	The corium slumps down to vessel bottom.
81.5 min.	The debris is starting to melt through the bottom head. Drywell and wetwell temperatures are 82°C (180°F) and 71°C (159°F), respectively. Meanwhile, local pool water temperature at the discharging bay exceeds 149°C (300°F). Steam condensation oscillations could accelerate due to the continuous discharge of superheated noncondensable gases into the suppression pool.
101 min.	Mass and energy addition rates into the wetwell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	1.25	165.35	3.47×10^6	1.97×10^5
Hydrogen	4.45×10^{-4}	0.06	1.03×10^3	58.58

Time (sec)	Event
142.5 min.	Vessel bottom head fails, resulting in a pressure increase of 0.34 MPa (49 psia).
152.5 min.	Debris starts to boil water from containment floor.
162.5 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 2013°C (3655°F) initially. Internal heat generation in metals and oxides are 2.43×10^7 and 1.26×10^7 watts, respectively.

Time (sec)	Event
162.5 min.	Drywell and wetwell temperatures are 128°C (262°F) and 74°C (166°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	0.057	7.54	1.59×10^5	9052
Hydrogen	0	0	0	0

Time (sec)	Event
167.8 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 204°C (400°F) and start to vent through the primary containment.
175.2 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment.
185.3 min.	Drywell and wetwell pressures are at 0.10 MPa (14.7 psia). Drywell and wetwell temperatures are 314°C (598°F) and 78°C (173°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	1.65	218.26	1.59×10^5	9052
Hydrogen	0.025	3.31	0	0
CO_2	5.79	765.88		
CO	0.526	69.58		

The leak rate through the containment failed areas is ~3.00×10^4 ℓ/s (~6.36×10^4 ft^3/min).

Time (sec)	Event
206 min.	Drywell and wetwell temperatures are 610°C (1130°F) and 78°C (173°F), respectively. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
CO_2	1.83	242		
Hydrogen	0.20	26	0	0
Steam	1.36	180	1.59×10^5	9052
CO	4.15	549		

Time (sec)	Event
	The leak rate through the containment failed area is ~2.94 × 10^4 ℓ/s (~6.24 × 10^4 ft^3/min).
222.5 min.	Rate of concrete decomposition is ~4.46 × 10^4 gm/s. Rate of heat added to atmosphere is ~3.71 × 10^4 kW.
254.5 min.	Drywell and wetwell pressures are at 0.10 MPa (~14.7 psia) and temperatures are 746°C (1375°F) and 77°C (171°F), respectively. The leak rate through the containment failed area is ~5.54 × 10^4 ℓ/s (~1.17 × 10^5 ft^3/min).
501 min.	Drywell and wetwell temperatures are 815°C (~1500°F) and 77°C (170°F), respectively. The leak rate through the containment failed area is ~2.34 × 10^4 ℓ/s (~4.96 × 10^4 ft^3/min).

LTAS results were used up to the 900 second point and the MARCH sequence results were used to construct the remainder of the environmental profiles. Figures C-9 and C-10 show drywell and suppression pool temperatures and drywell pressure as predicted by the LTAS code. Table C-3 (Ref. 9, pg.116) contains the MARCH data used to complete the environmental profiles shown in Figures C-11, C-12, and C-13. In the case of plotting the drywell pressure pulse shown in Figure C-13, no curves or data could be found in the available documentation of MARCH results. Instead, the pressure pulse shape was approximated by the shape of the pressure pulse in Figure C-6 but using the vessel breach and containment failure times given in Table C-3.

2.5.2 TB (long term)

A long term blackout with core injection available until battery failure at 4 hours was also examined. This sequence is characterized in much the same manner as the short term sequence with all key events being delayed by about 7 hours due to injection initially being available. Two variations were investigated with the LTAS code. Scenario 4 involved all expected operator actions including depressurizing the vessel about 15 minutes into the accident. Scenario 5 examined the effect of a stuck open relief valve 250 seconds into the accident. Even though the reactor vessel returns to pressure after battery failure (due to loss of power to the SRVs) in scenario 4, pressure was not found to have a significant effect in this longer sequence. Therefore, a composite profile of both scenarios was used to describe the long term blackout sequence.

The LTAS results predicting water level behavior as a function of time are shown in Figure C-14. Top of the active fuel is uncovered about 20,000 seconds (333 min.) into the accident. This is in good agreement with the MARCH result shown in Table C-4 (Ref. 1, pg.45) of 347 minutes. Therefore, LTAS results are used to construct the scenario up to 333 minutes and the March results shown in Table C-4 are used to complete the profiles. Figures C-15 and C-16 show the LTAS results for drywell and wetwell temperature and drywell pressure used up to the 333 minute point. Figures C-17, C-18, and

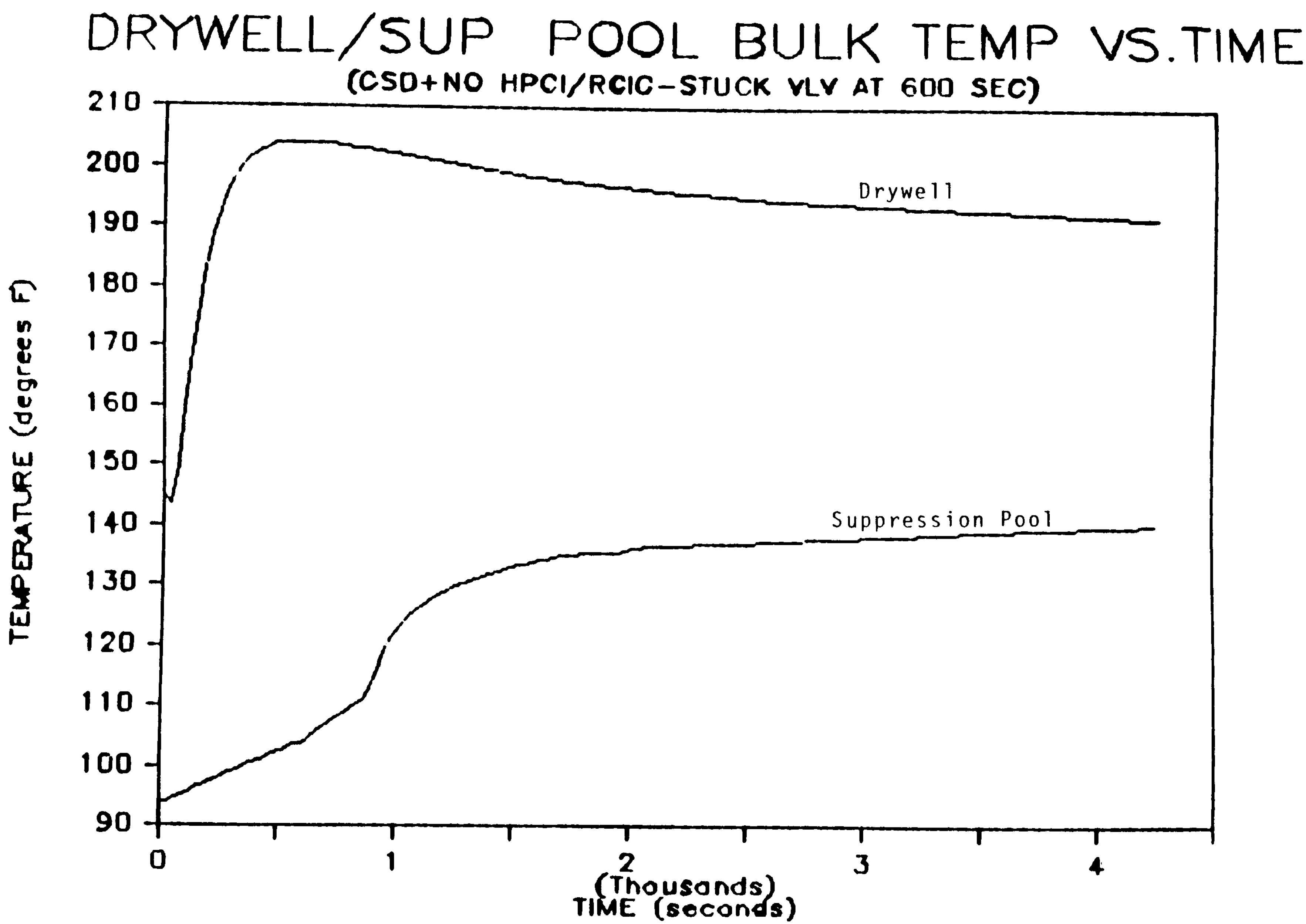


FIGURE C-9 - LTAS PREDICTED DRYWELL AND WETWELL TEMPERATURES FOR THE THIRD SCENARIO

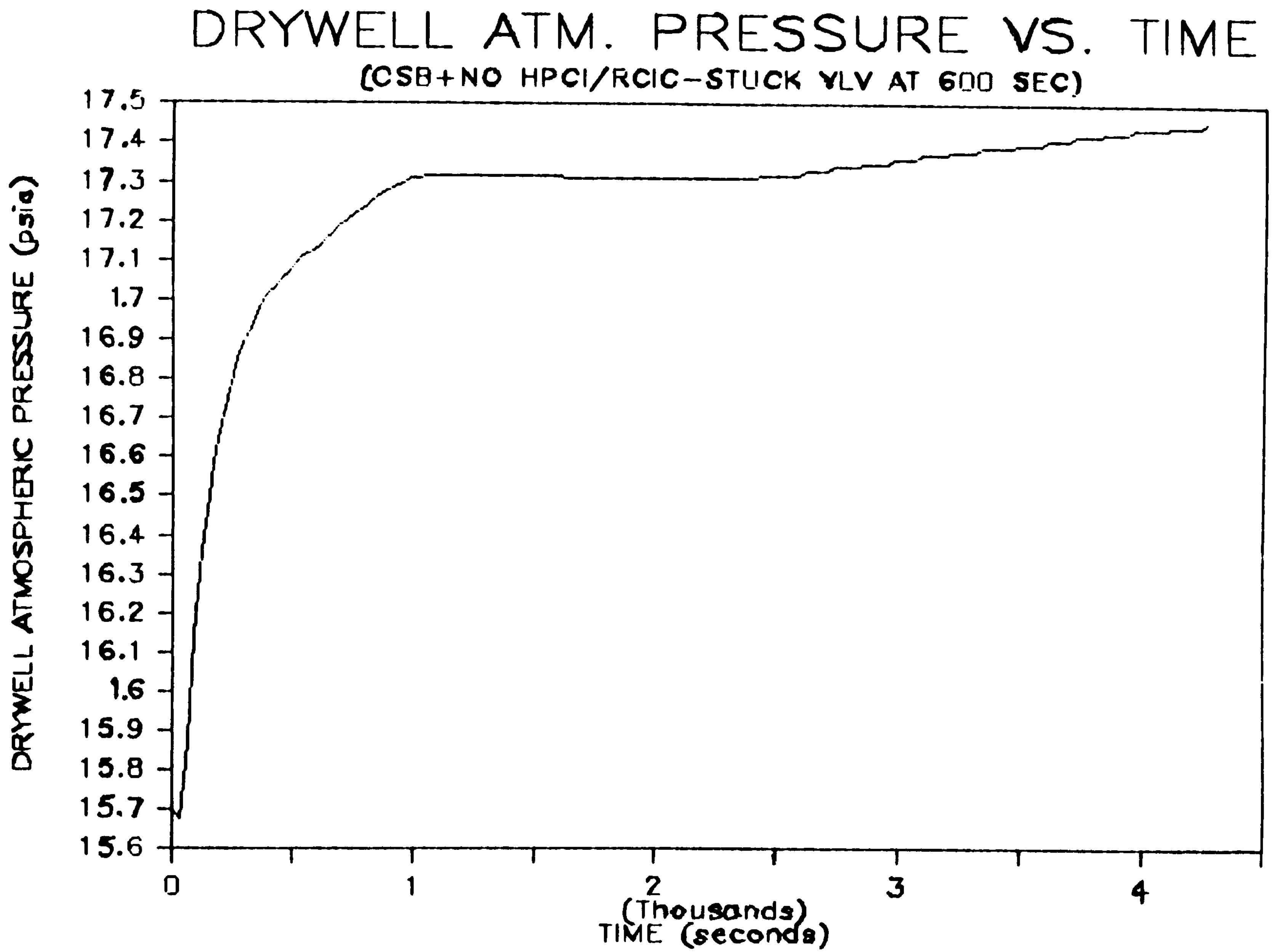


FIGURE C-10 - LTAS PREDICTED CONTAINMENT PRESSURE FOR THE THIRD SCENARIO

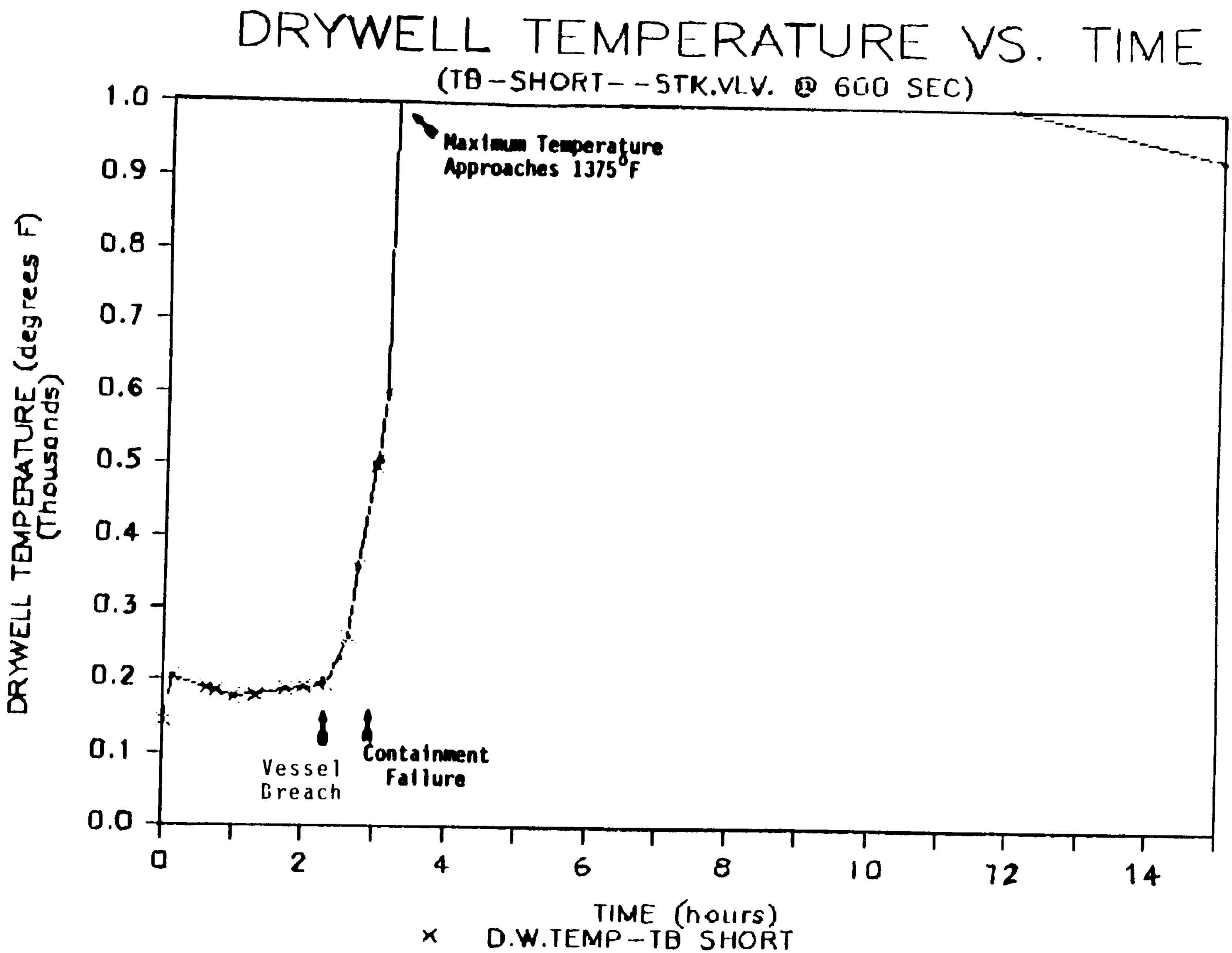


FIGURE C-11- DRYWELL TEMPERATURE PROFILE FOR THE THIRD SCENARIO

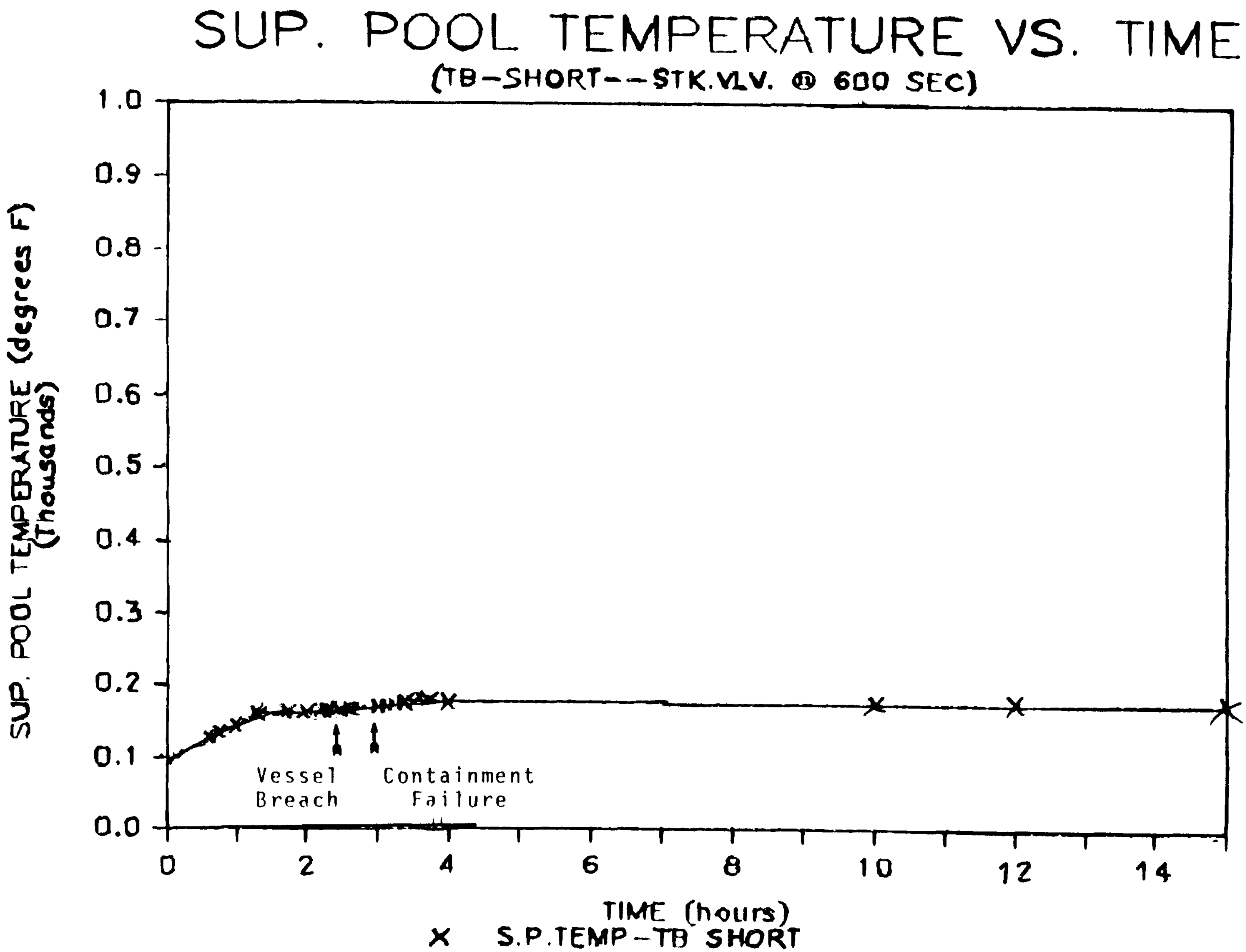


FIGURE C-12 - WETWELL TEMPERATURE PROFILE FOR THE THIRD SCENARIO

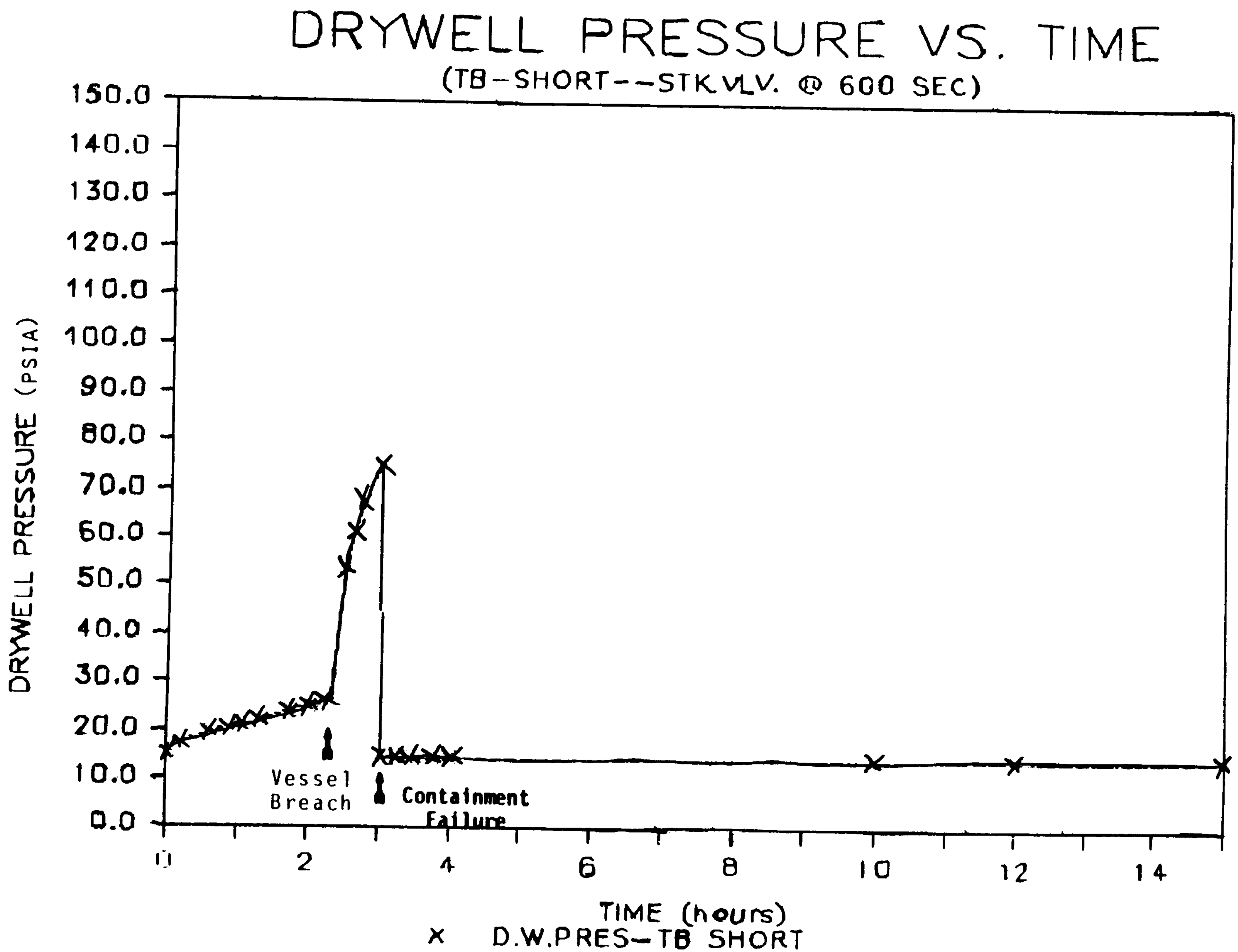


FIGURE C-13 - CONTAINMENT PRESSURE PROFILE FOR THE THIRD SCENARIO

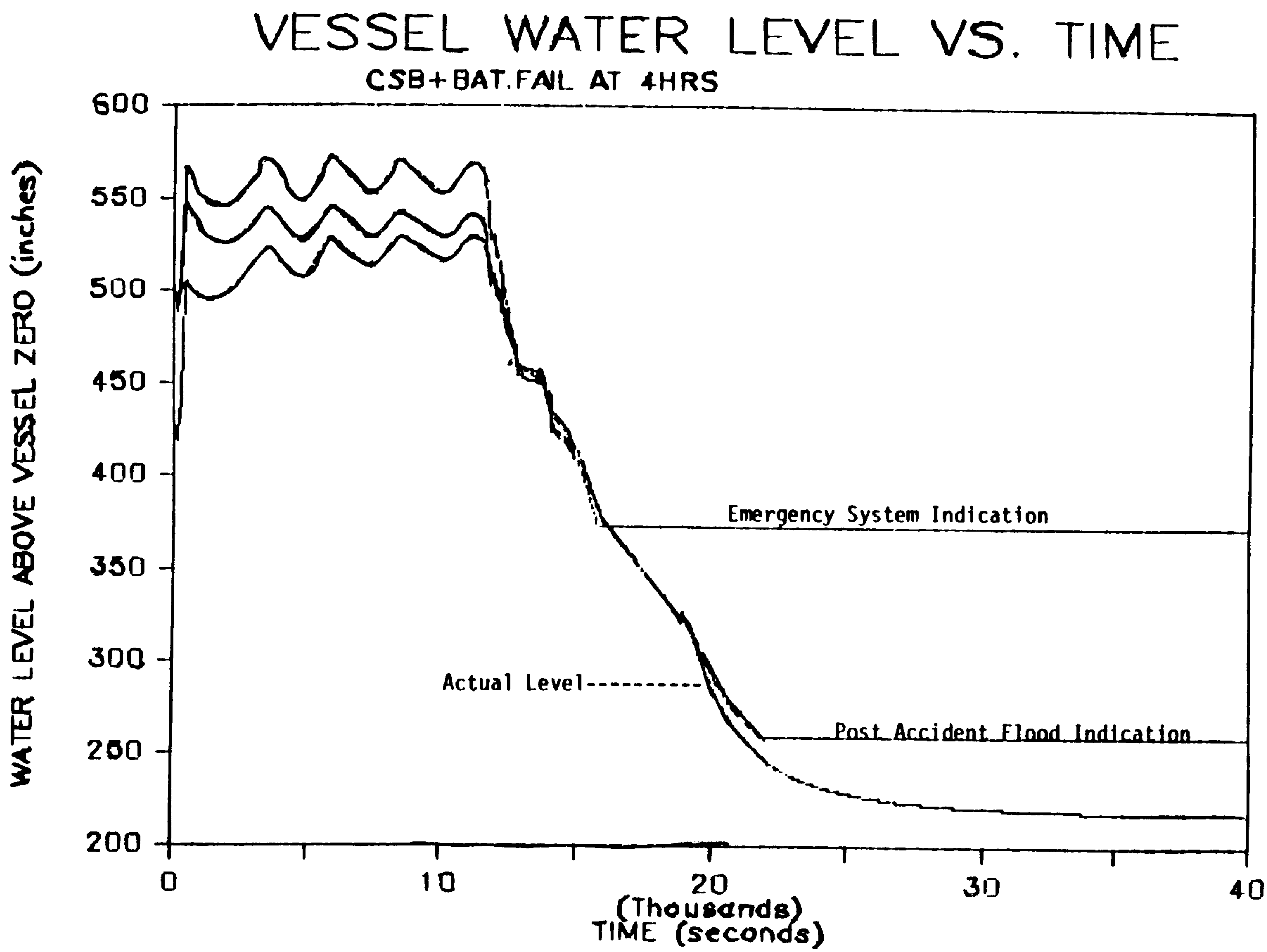


FIGURE C-14 - LTAS PREDICTED WATER LEVEL FOR THE FIFTH SCENARIO

From:
NUREG/CR-2182

Table C-4

Browns Ferry Nuclear Plant: Complete Station Blackout
Sequence of Events

CSB + Manual RCIC & SRV

(T_VB')

Time (sec)	Event
0.0	Loss of all AC power and diesel generators. The plant is initially operating at 100% power. Initial drywell temperature = 66°C (150°F) Initial wetwell temperature = 35°C (95°F)
0.2	Full load rejection (i.e., fast closure of turbine control valves) occurs.
0.2	Recirculation pumps and condenser circulatory water pumps trip off. Loss of condenser vacuum occurs. Core flow is provided by natural circulation.
0.2	Reactor pressure increases suddenly due to load rejection.
0.3	Scram pilot valve solenoids are deenergized due to load rejection. Control rod motion begins.
0.5	Turbine bypass valves start to open due to load rejection.
1.0	Neutron flux starts to decrease after an initial increase to over 100% rated power level.
1.0	Reactor power starts to decrease slowly after an initial rise.
2.0	Control rods are 40% inserted from fully withdrawn position.
2.0	Main steamline isolation valves (MSIVs) start to close (relay-type reactor trip system), resulting in a rapid steam-line pressure rise.
2.0	Turbine bypass valves are tripped to close.
3.0	Control rods are 75% inserted from fully withdrawn position.

Time (sec)	Event
3.0	Turbine trips off (turbine stop valves fully closed).
3.5	Power generation due to delayed neutrons and fission product decay drops to 10% of initial rated power generation.
4.0	Feedwater turbines trip off.
5.0	MSIVs are fully closed, resulting in a momentary 0.69 MPa (100 psi) pressure increase and 1.02 m (40-in.) drop of water-steam mixture level due to collapsing of voids.
5.0	All control rods are fully inserted.
5.0	Reactor vessel pressure exceeds the lowest setpoint at 7.52 MPa (1090 psi) of safety/relief valves (S/RVs).
5.0	Seven (7) out of thirteen (13) S/RVs start to open in response to pressure rise above the setpoint.
5.2	Water-steam mixture level recovers 0.51 m (20 in.) from the previous momentary 1.02 m (40-in.) drop.
5.5	S/RV steam blowdowns into the pressure suppression pool through the T-quenchers begin.
7.5	Feedwater flow drops below 20%.
9.0	Feedwater flow decreases to zero.
10.0	Power generation due to fission product decay drops to approximately 7.2% of rated power generation.
15.0	All 7 S/RVs are completely closed.
15.7	Four out of 13 S/RVs start to open.
17.0	Neutron flux drops below 1% of initial full power level.
21.0	Narrow range (NR) sensed water level reaches low alarm (Level 4), i.e., 5.98 m (235.50 in.) above Level 0, or 5.00 m (196.44 in.) above TAF.
22.0	Suppression pool water average temperature rises to 35.13°C (95.24°F) in response to the first S/RV pops.

Time (sec)	Event
29.0	All 4 S/RVs are completely closed.
29.7	Two out of 13 S/RVs start to open.
47.0	All 2 S/RVs are completely closed.
47.7	One out of 13 S/RVs starts to open.
56.0	Suppression pool water average temperature is approximately 35.3°C (95.54°F).
56.0	NR sensed water level reaches low level alarm (Level 3), i.e., 5.50 m (216.00 in.) above Level 0, or 4.50 m (176.94 in.) above TAF.
90.0	Suppression pool water average temperature is approximately 35.4°C (95.72°F).
101.0	The S/RV is completely closed. The same S/RV continues to cycle on and off on setpoints throughout the subsequent RCIC injections.
625	Wide range sensed water level reaches low water level setpoint (Level 2), i.e., 4.18 m (164.50 in.) above Level 0 at 2/3 core height, or 2.96 m (116.50 in.) above TAF.
625	Operator manually controls RCIC injection to maintain constant vessel water level. The RCIC turbine pump is driven by steam generated by decay heat. System auxiliaries are powered by the 250 V dc system.
655	RCIC flows enter the reactor pressure vessel at 38 ℓ/s (600 gpm) drawing water from the condensate storage tank.
15 min.	Operator manually opens one SRV to depressurize the vessel.
20 min.	Drywell and wetwell temperatures exceed 76°C (169°F) and 50°C (122°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	829.75	1.10×10^5	2.32×10^8	1.32×10^7
Hydrogen	0	0	0	0

Time (sec)	Event
21.14 min.	Core uncovery time.
22.0 min.	Core refloods.
30 min.	Auto-isolation signal initiates as increase of drywell pressure exceeds 13.8 KPa (2.0 psi). The RCIC system is not isolated.
240 min.	The RCIC pump stops when the batteries run out.
266.3 min.	Wide range sensed water level reaches Level 2 setpoint. Drywell and wetwell temperatures are 99°C (210°F) and 100°C (212°F), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	19.16	2.53×10^3	5.20×10^7	2.96×10^6
Hydrogen	0	0	0	0

Time (sec)	Event
347 min.	Core uncovers again.
366 min.	Average gas temperature at top of core is 491°C (916°F). Drywell and wetwell temperatures and pressures are 113°C (236°F) and 0.28 MPa (40 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	9.26	1.22×10^3	2.97×10^7	1.69×10^6
Hydrogen	4.09×10^{-5}	5.41×10^{-3}	222.28	12.64

Time (sec)	Event
386 min.	Average gas temperature at top of core is 855°C (1571°F). Drywell and wetwell temperatures and pressures are 115°C (239°F) and 0.29 MPa (41 psia), respectively. Mass and energy addition rates into the wetwell are:

	Mass Rate		Energy Rate	
	(kg/s)	(lb/min)	(w)	(Btu/min)
Steam	5.05	6.68×10^2	1.81×10^7	1.03×10^6
Hydrogen	1.68×10^{-2}	2.23	1.35×10^5	7.70×10^3

Time (sec)	Event
395.3 min.	Core melting starts.
449.3 min.	Water level in vessel drops below bottom grid elevation.
451.2 min.	Bottom grid fails and temperature of structures in bottom head is above water temperature.
452 min.	The corium slumps down to vessel bottom.
452.9 min.	Debris starts to melt through the bottom head.
539.3 min.	Vessel bottom head fails, resulting in a pressure increase of 0.0047 MPa (0.68 psia).
539.3 min.	Debris starts to boil water from containment floor.
539.3 min.	Drywell electric penetration assembly seals have failed as the containment temperature exceeds 204°C (400°F) and start to vent through the primary containment at a leak rate of 118 ℓ/s (250 ft^3/min).
539.3 min.	Debris starts to melt the concrete floor of the containment building. Temperature of debris is 1750°C (3182°F) initially. Internal heat generation in metals and oxides are 9.99 $\times 10^6$ and 1.84 $\times 10^7$ watts, respectively.
601.05 min.	Containment failed as the containment temperature exceeds 260°C (500°F) and all electric penetration modules are blown out of the containment. Mass and energy addition rates into the drywell are:

	Mass Rate (kg/s)	Mass Rate (lb/min)	Energy Rate (w)	Energy Rate (Btu/min)
Steam	4.70	621.51	1.59×10^5	9052
Hydrogen	0.14	18.27	0	0
CO_2	1.29	170.23		
CO	2.88	381.21		

The leak rate through the drywell penetration seals is ~5.33×10^4 ℓ/s (1.13×10^5 ft^3/min).

Time (sec)	Event
718.8 min.	Drywell and wetwell pressures are at 0.10 MPa (~14.7 psia) and temperatures are 700°C (1293°F) and 98°C (~209°F), respectively. The leak rate through the containment failed area is ~5.18×10^4 ℓ/s (~1.10×10^5 ft^3/min).

Time (sec)	Event
821.5 min.	Drywell and wetwell temperatures are 737°C (1359°F) and 93°C (199°F), respectively. The leak rate through the containment failed area is ~4.23×10^4 ℓ/s (~8.96×10^4 ft^3/min).
1127.5 min.	Drywell and wetwell temperatures are 468°C (~875°F) and 86°C (~188°F), respectively. The leak rate through the containment failed area is ~4.79×10^4 ℓ/s (~1.02×10^4 ft^3/min).

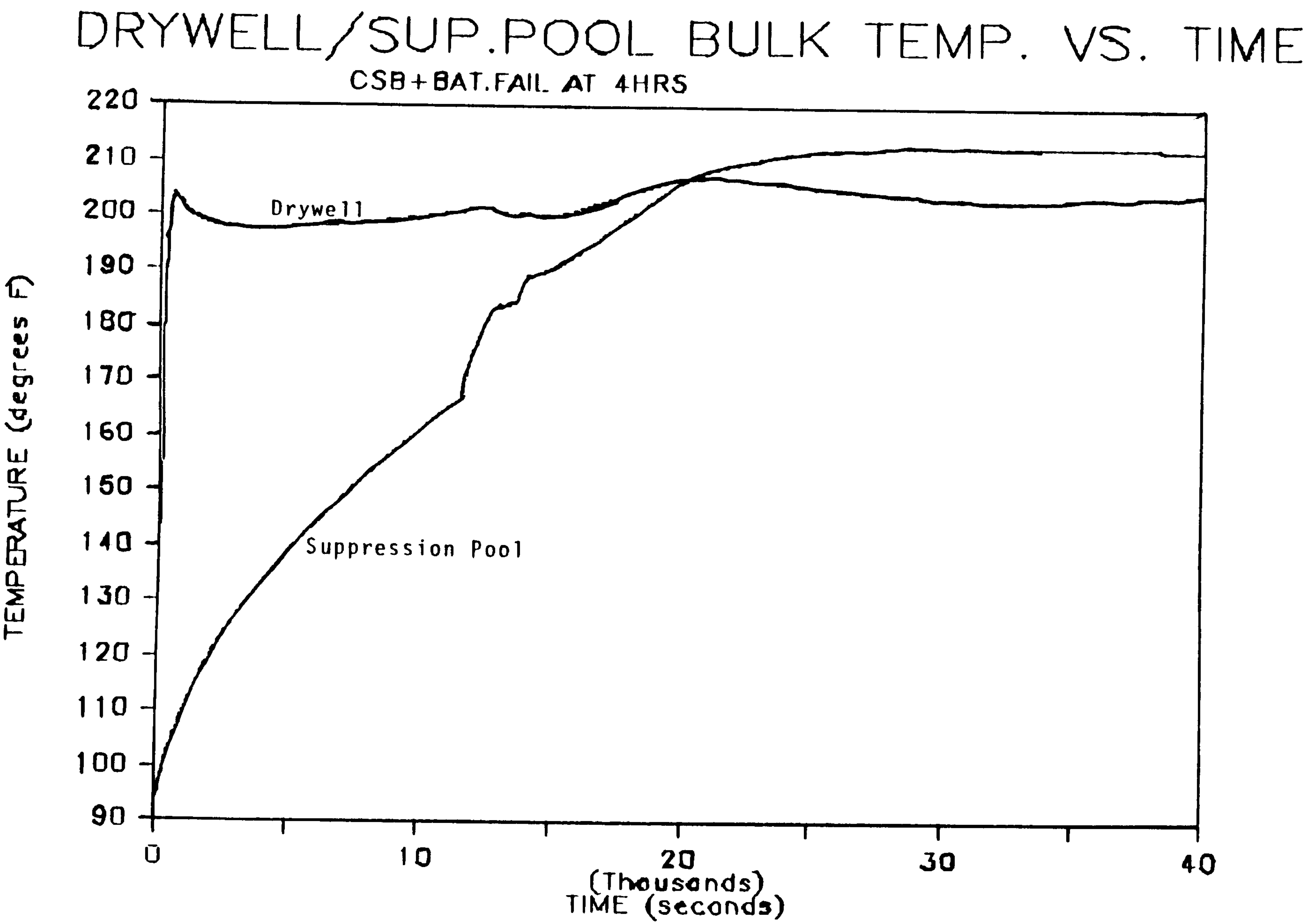


FIGURE C-15 - LTAS PREDICTED DRYWELL AND WETWELL TEMPERATURES FOR THE FIFTH SCENARIO

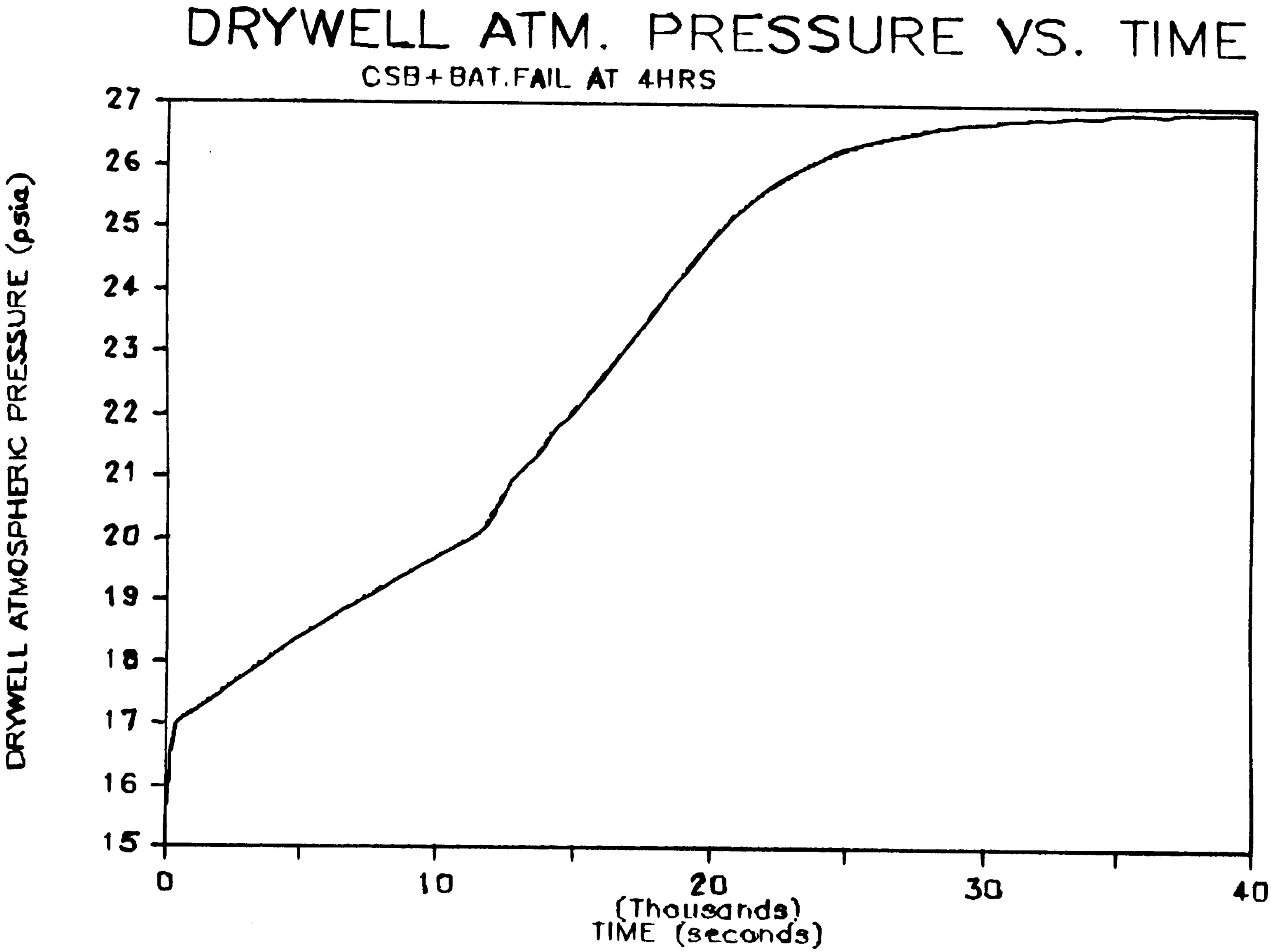


FIGURE C-16 - LTAS PREDICTED CONTAINMENT PRESSURE FOR THE FIFTH SCENARIO

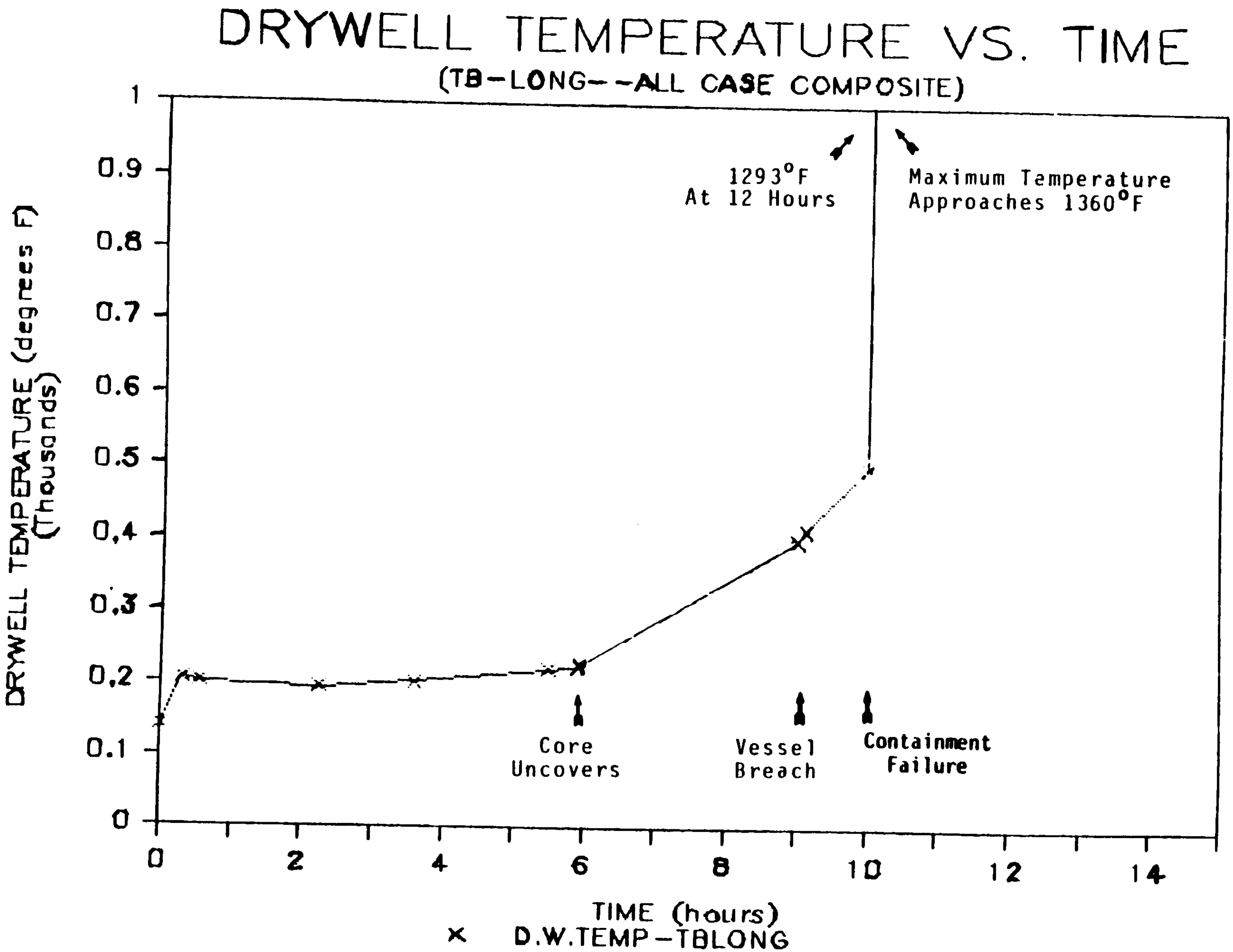


FIGURE C-17 - DRYWELL TEMPERATURE PROFILE FOR THE FOURTH AND FIFTH SCENARIO

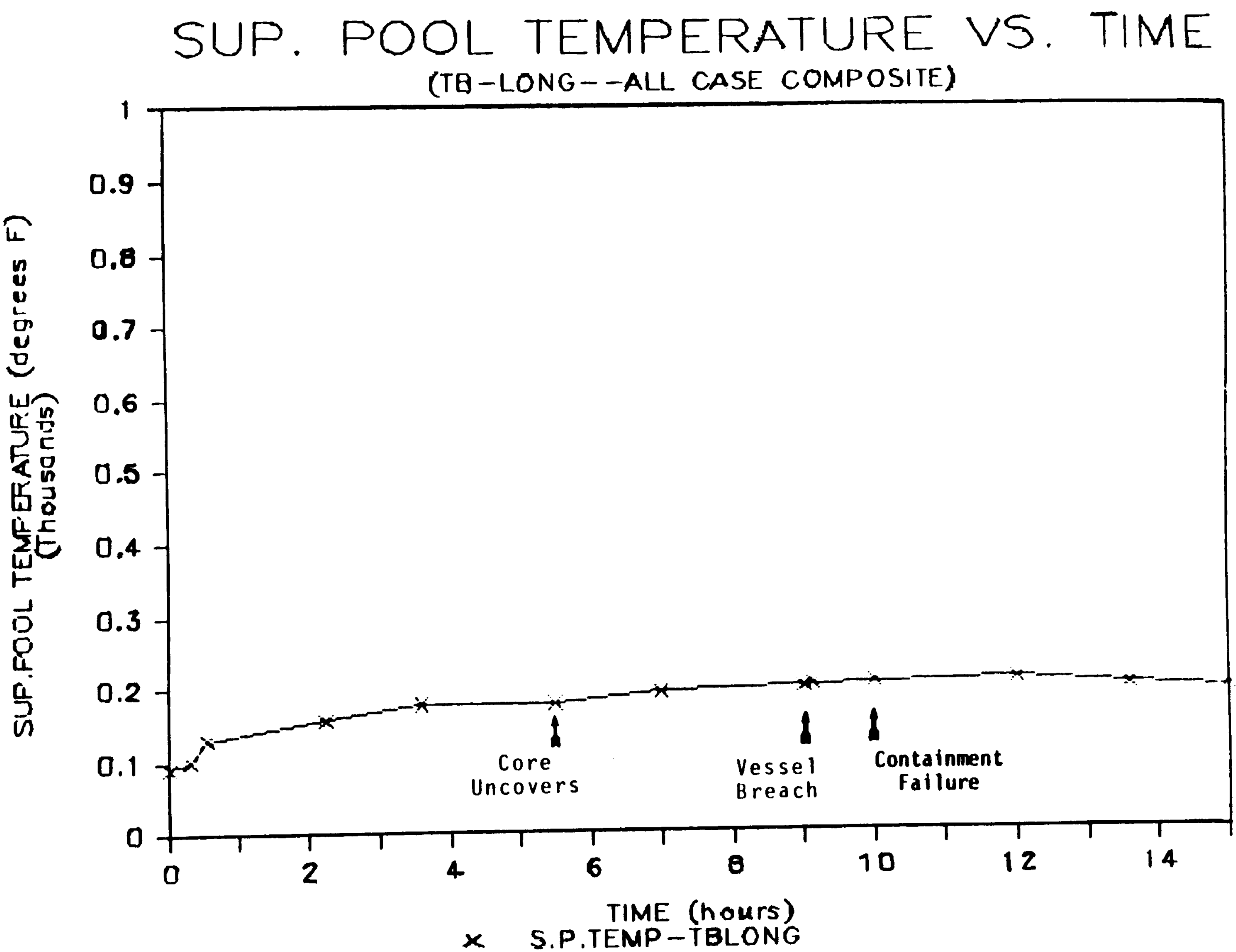


FIGURE C-18 - WETWELL TEMPERATURE PROFILE FOR THE FOURTH AND FIFTH SCENARIO

C-19 are the resulting environmental profiles for drywell temperature, suppression pool temperature, and drywell pressure respectively. The explanation in Section 2.5.1 relating the general shape of the profile curves to specific events in the sequence also applies here. Note that in Figure C-19 no specific values for the pressure pulse were given in the available documentation, but it is probable that this pulse is similar to the pressure pulse displayed in Figure C-13 for the short term sequence. One difference worth mentioning is that with injection available for the first four hours, more decay heat energy is removed to the wetwell. This is evidenced by the fact that wetwell temperature reaches a higher peak value in the long term sequence than in the previous short term sequence. The general shape and behavior of the curve, however, is the same as the short term sequence described in Section 2.5.1.

2.5.3 TW

The TW sequence, a transient without means to cool the suppression pool, was also examined. Two separate scenarios were examined for this sequence. Scenario 6 considered the effects of operator action to depressurize the vessel about one hour into the accident. Scenario 7 examined the consequences of a stuck open relief valve. A single set of environmental profiles was constructed for the two scenarios as they produced very similar results. In this sequence, containment failure precedes core damage. This means that the LTAS code should be able to generate good data for most of the sequence because the core will be covered for a longer period of time. Figures C-20 and C-21 show the LTAS data used to form the first 15 hours of the environmental profiles found in Figures C-22-24. MARCH generated data found in Table C-5 (Ref. 6, pgs 19,25,29,30,96, and 98) was used to verify the LTAS curves and complete the profiles. A brief discussion explaining the behavior of the environmental profiles for this sequence follows.

Figure C-22 shows the drywell temperature as a function of time for the TW sequence. Note that following reactor scram drywell temperature slowly decreases as would be expected since the reactor is no longer generating

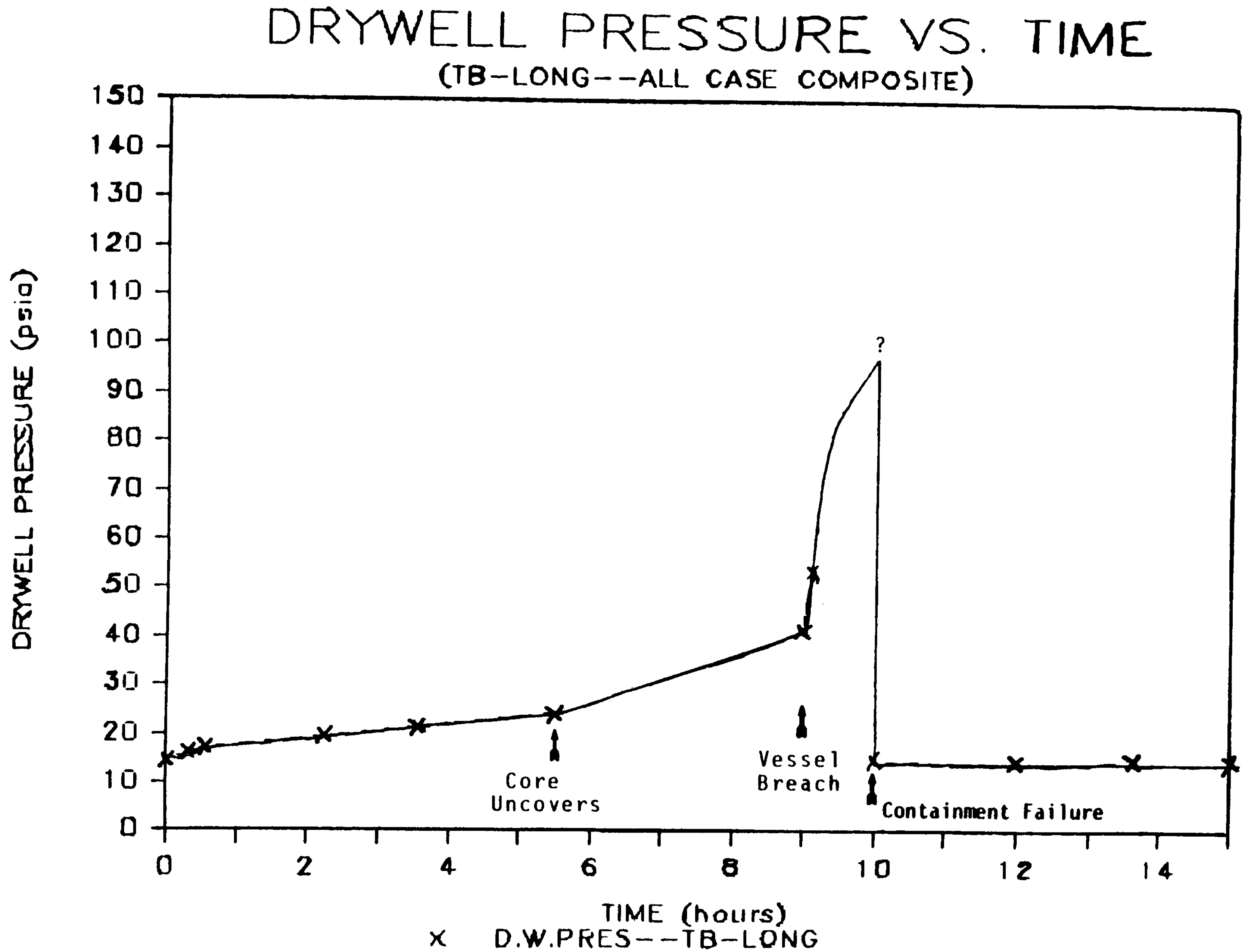


FIGURE C-19 - CONTAINMENT PRESSURE PROFILE FOR THE FOURTH AND FIFTH SCENARIO

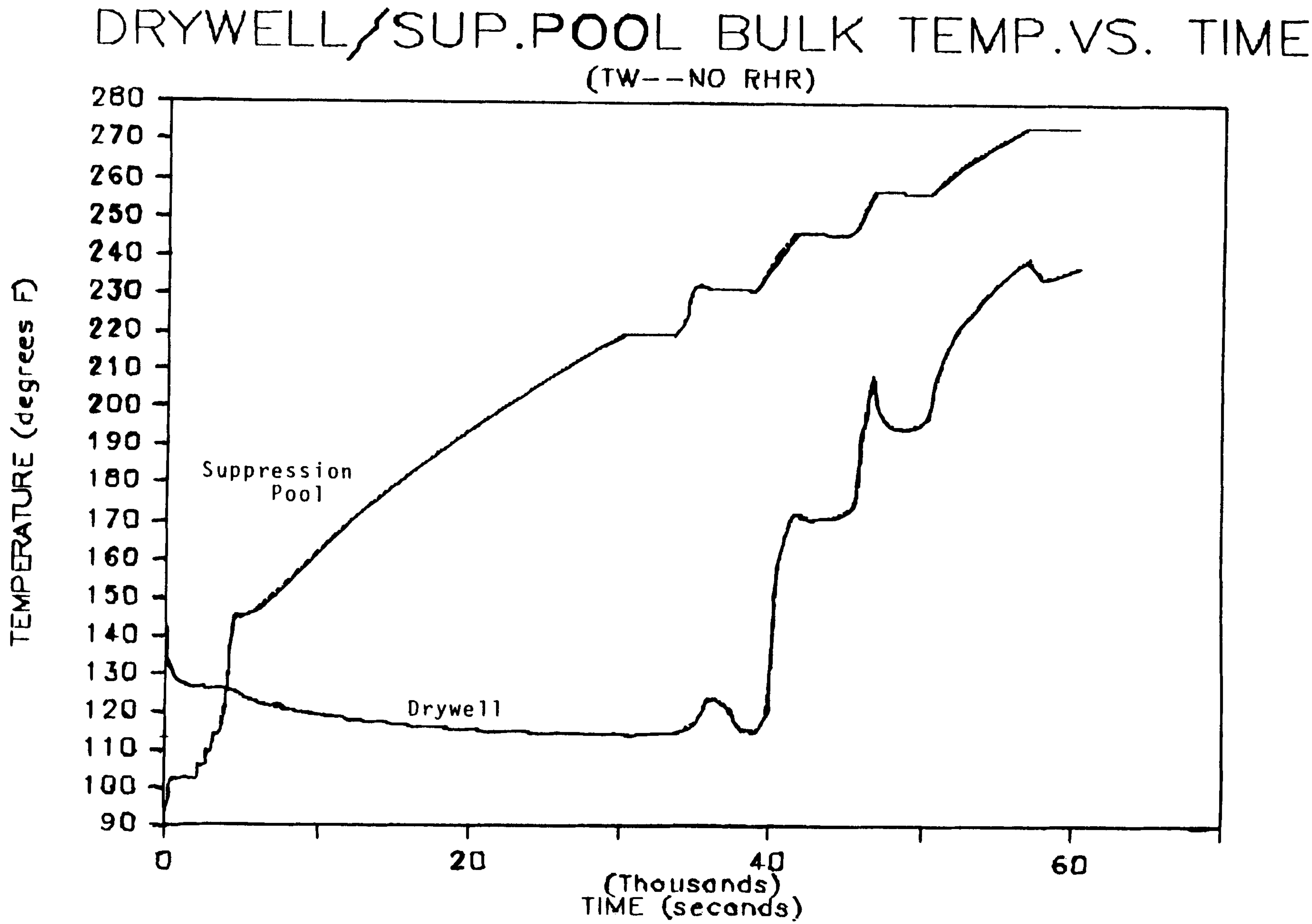


FIGURE C-20 - LTAS PREDICTED DRYWELL AND WETWELL TEMPERATURES FOR THE SIXTH SCENARIO

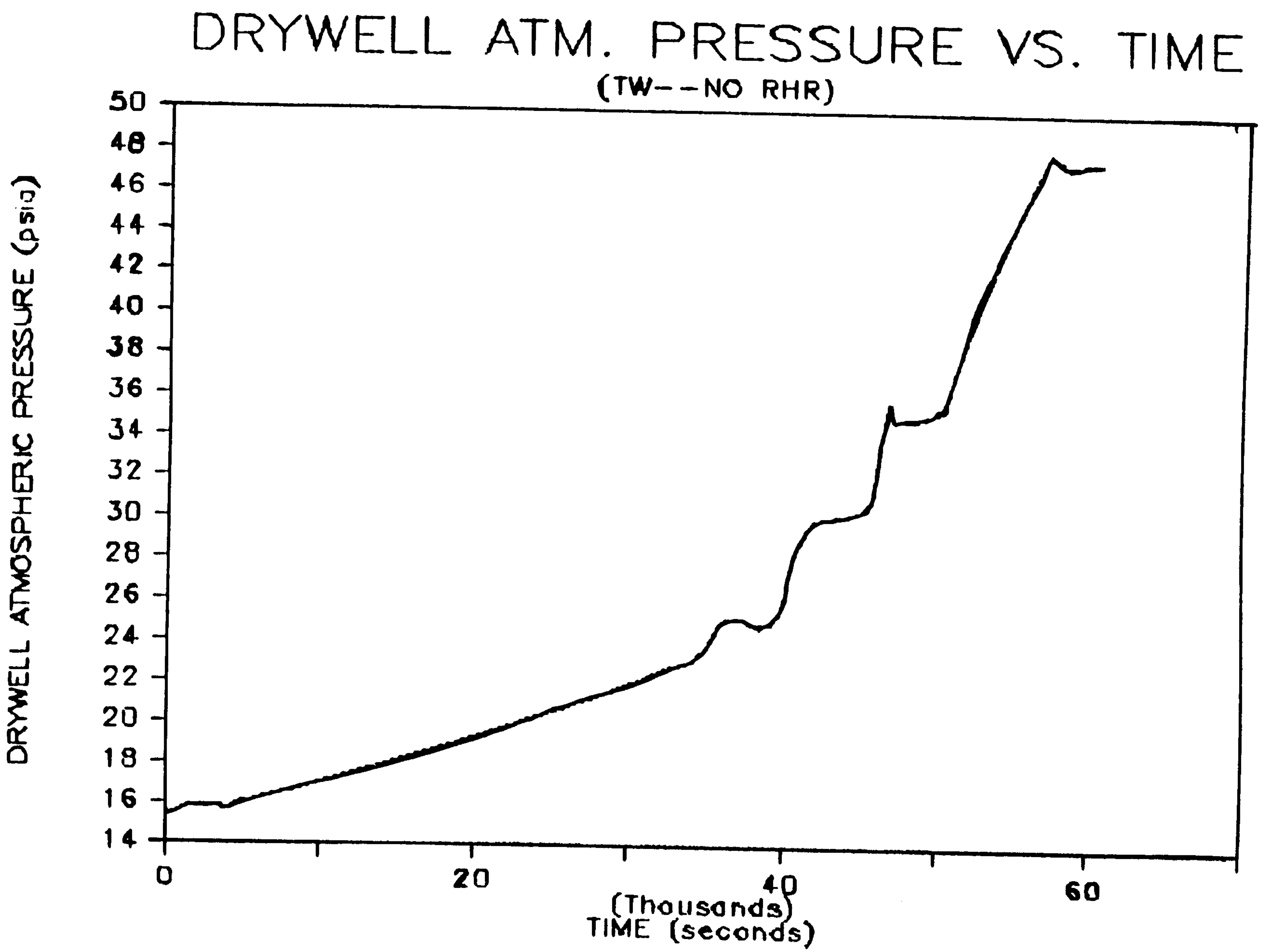


FIGURE C-21- LTAS PREDICTED CONTAINMENT PRESSURE FOR THE SIXTH SCENARIO

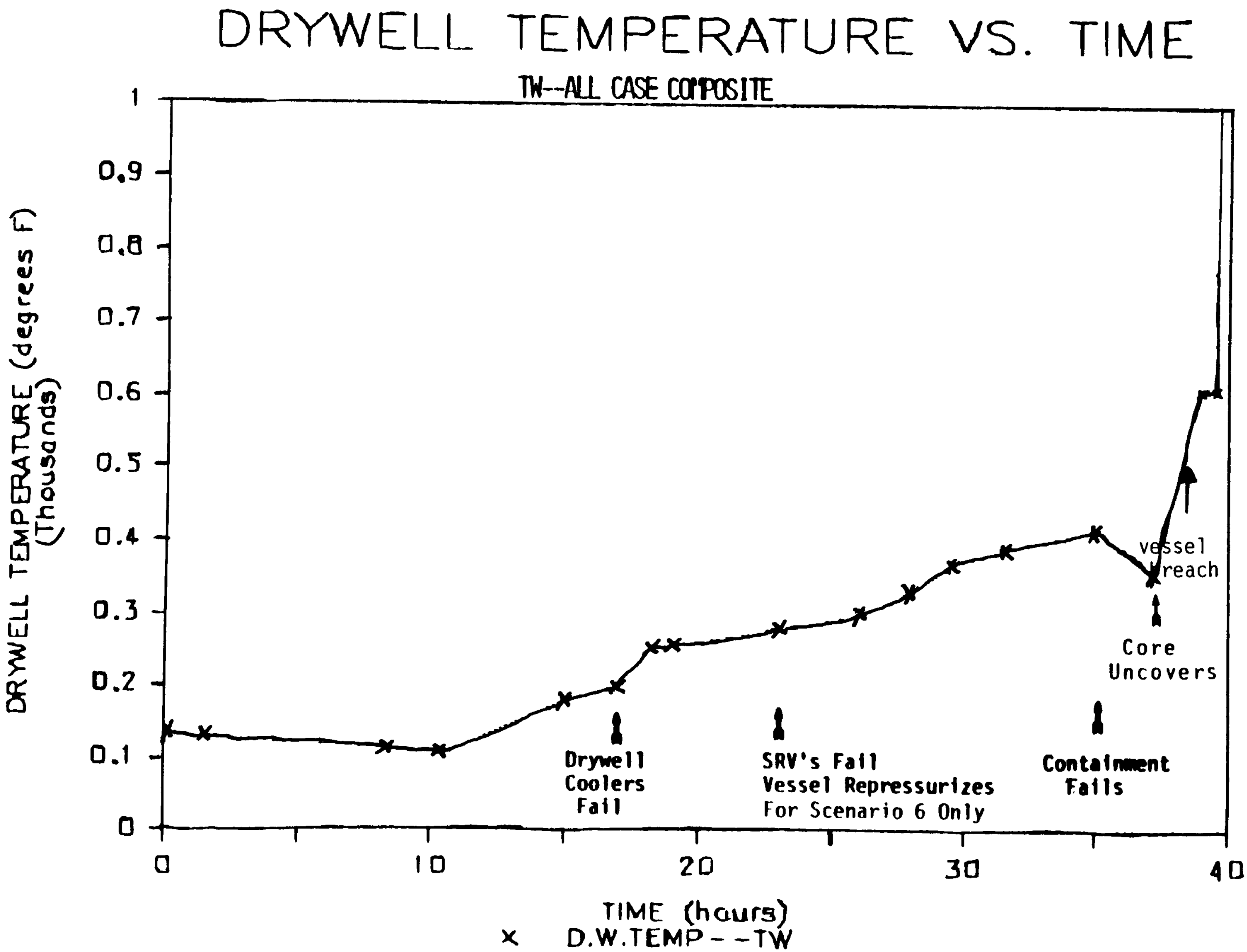


FIGURE C-22 - DRYWELL TEMPERATURE PROFILE FOR THE SIXTH AND SEVENTH SCENARIO

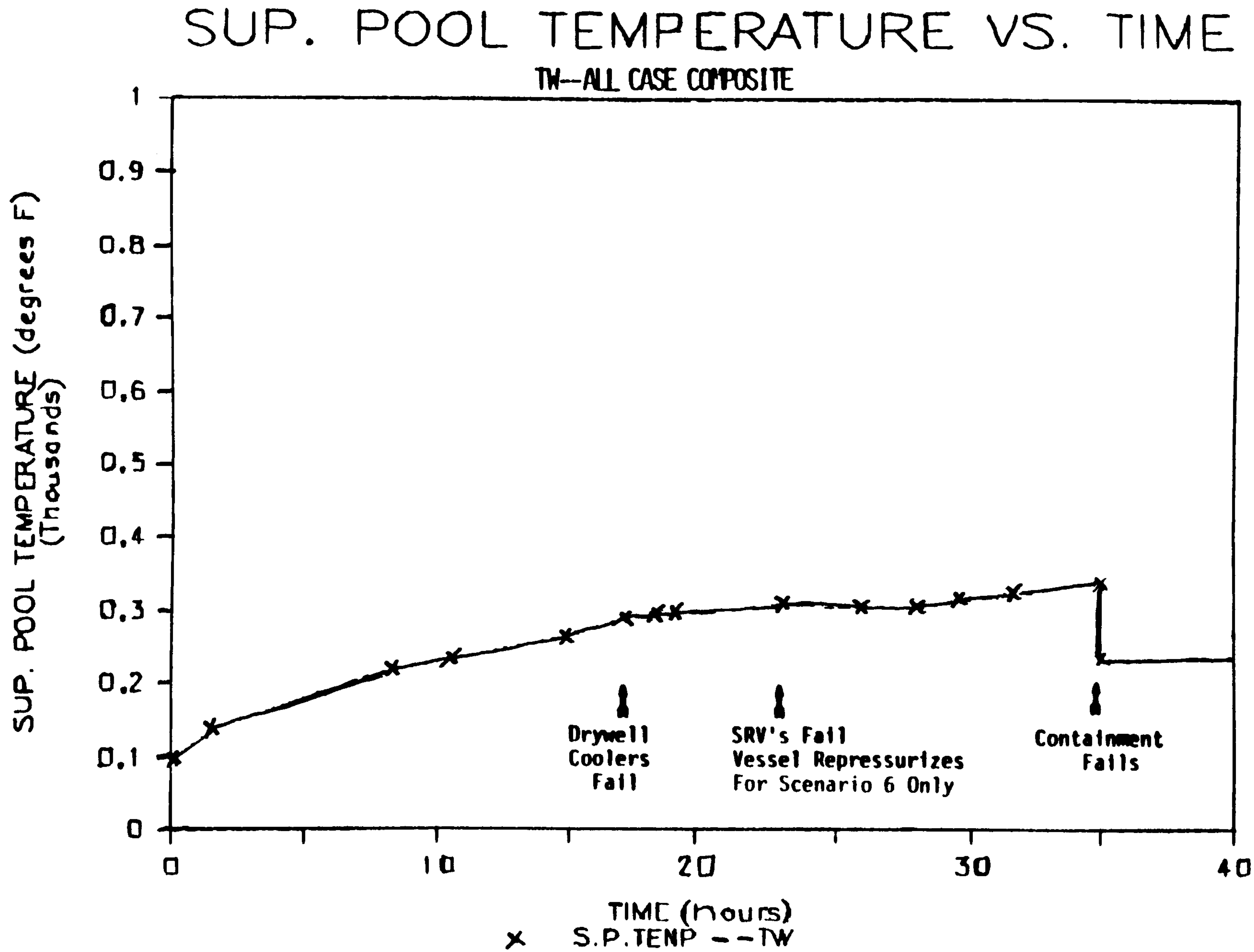


FIGURE C-23 - WETWELL TEMPERATURE PROFILE FOR THE SIXTH AND SEVENTH SCENARIO

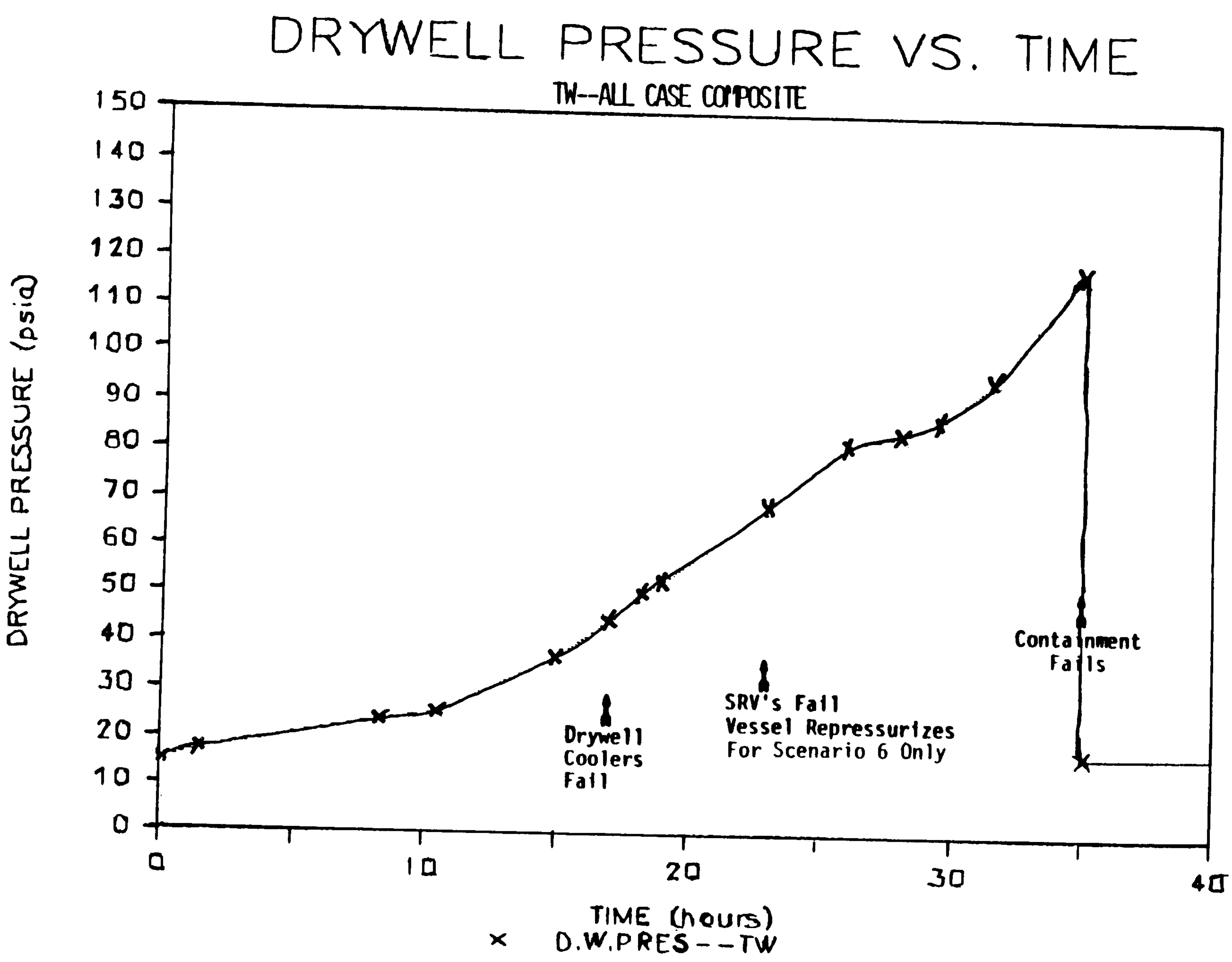
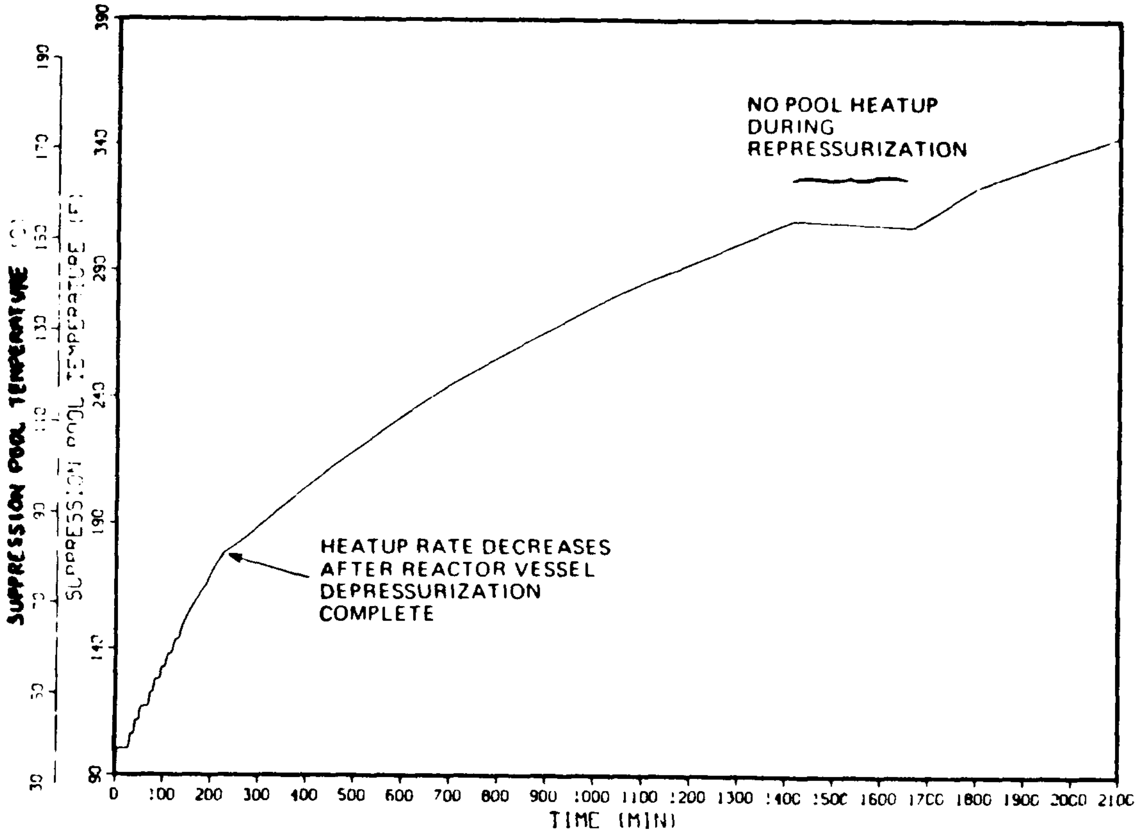


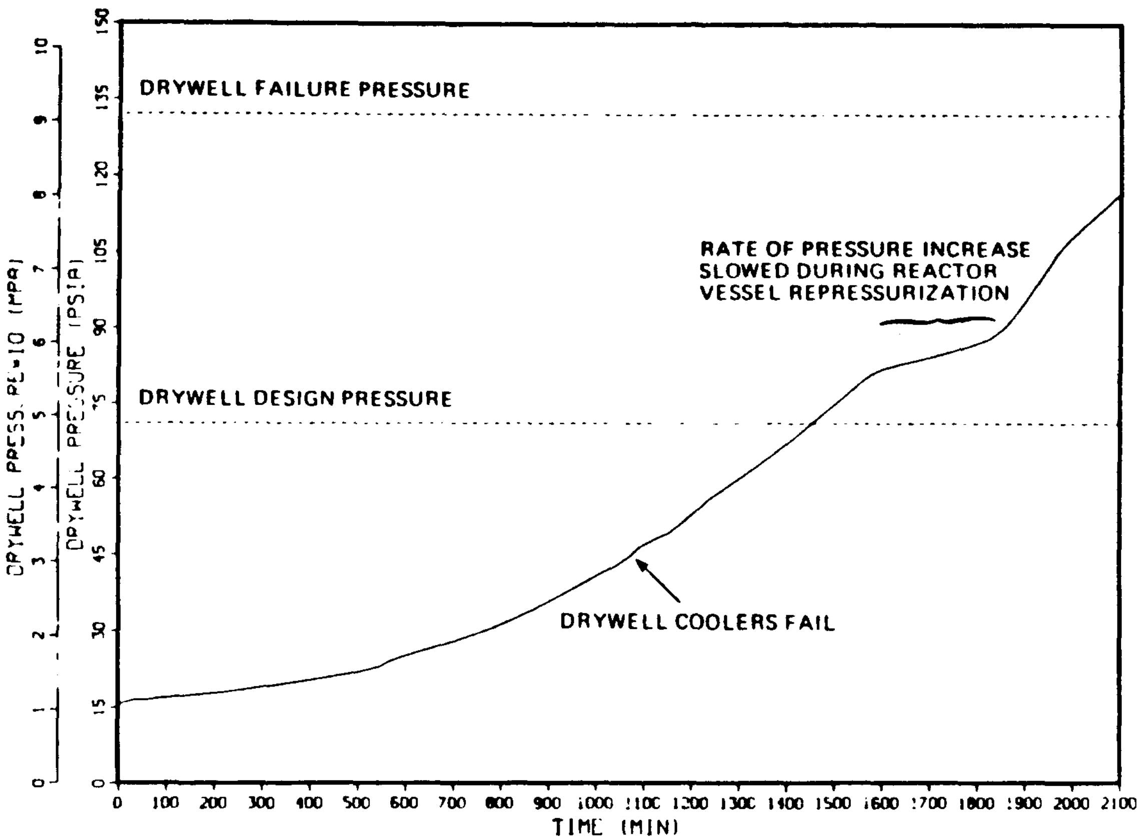
FIGURE C-24 - CONTAINMENT PRESSURE PROFILE FOR THE SIXTH AND SEVENTH SCENARIO

Table C-5 Timetable/Plots of events for
unmitigated loss of DHR with uniform pool heatup

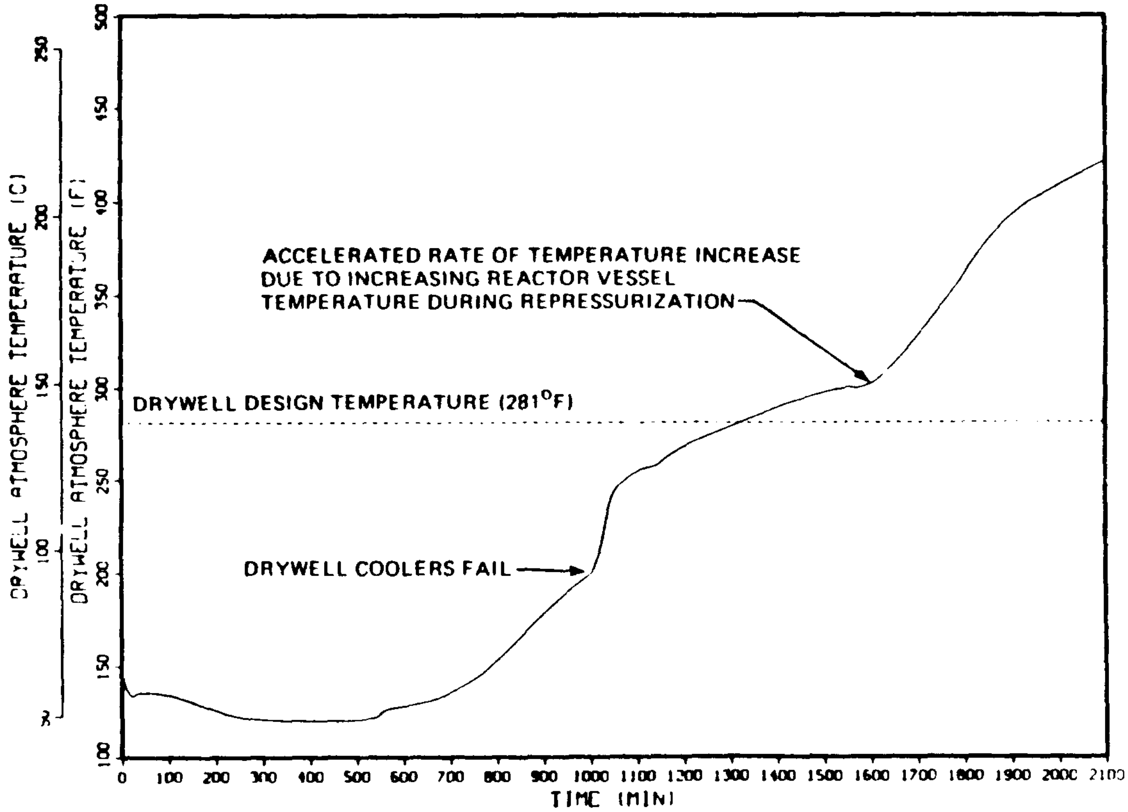
Time (h)	Event
0	Initiating reactor trip followed by MSIV closure and failure of both pool cooling and shutdown cooling modes of the RHR system.
1	High drywell pressure scram at 0.115 MPa (2 psig). Diesel generators and SGTS automatically initiated. Drywell control air compressors isolated. Operators valve station control air into drywell control air header.
1	Pool temperature exceeds 49°C (120°F) — operators begin controlled depressurization of reactor vessel.
2	Core spray initiation signal [reactor vessel pressure <3.21 MPa (465 psia) and drywell pressure >0.115 MPa (2 psig)] causes load shedding if loss of offsite power is still in effect. Operators must use local control stations to restore diesel power to station control air compressors (A and D) and drywell coolers.
2	Suppression pool temperature exceeds the 60°C (140°F) recommended maximum temperature for cooling of RCIC and HPCI lube oil.
4	CRD hydraulic system provides sufficient reactor vessel injection — no RCIC system operation after this time.
8.6	Operators must begin to throttle CRD hydraulic system pump to avoid overfilling the reactor vessel.
13	HPCI and RCIC system steam supply line isolation caused by high [93°C (200°F)] torus room temperature.
14	RCIC turbine high exhaust pressure trip at containment pressure >0.28 MPa (25 psig).
21.5	Drywell design pressure [0.49 MPa (56 psig)] exceeded.
23.5	SRVs become inoperative in remote-manual mode because drywell pressure exceeds 0.55 MPa (65 psig).
35	Drywell fails when internal pressure exceeds 0.91 MPa (117 psig). Suppression pool temperature has increased to 173°C (343°F).



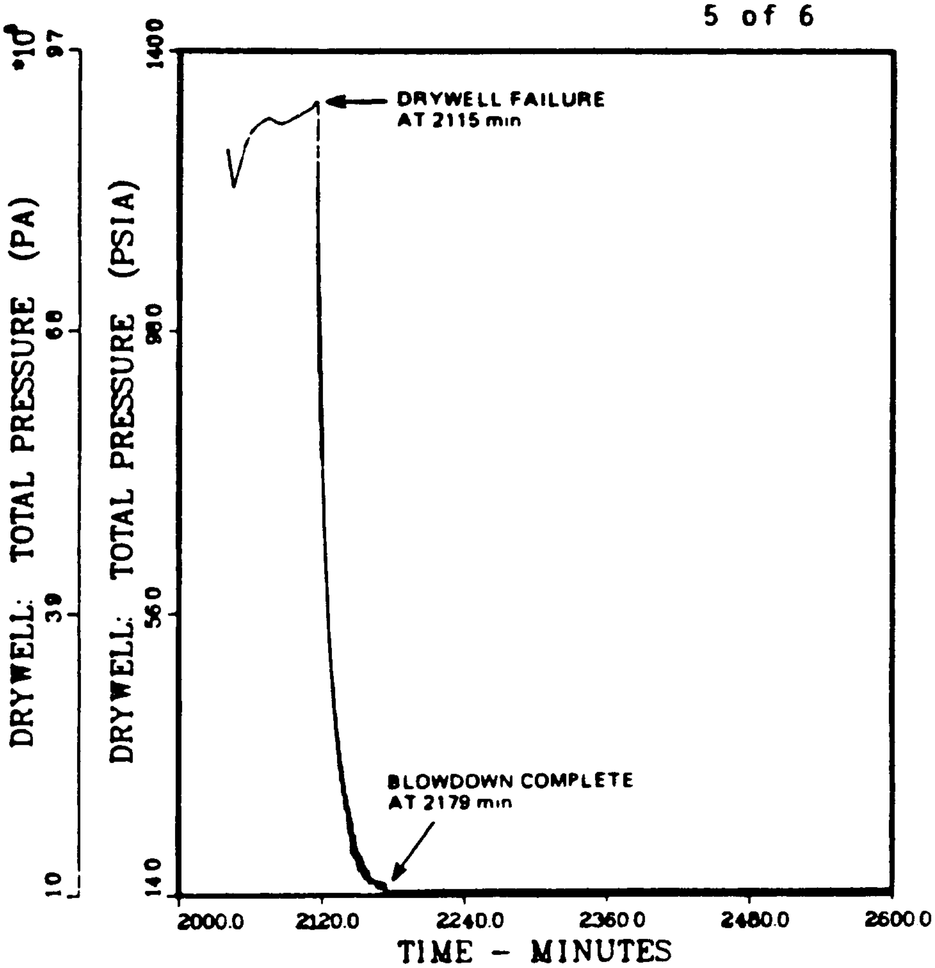
Unmitigated Loss of DHR - suppression pool temperature.



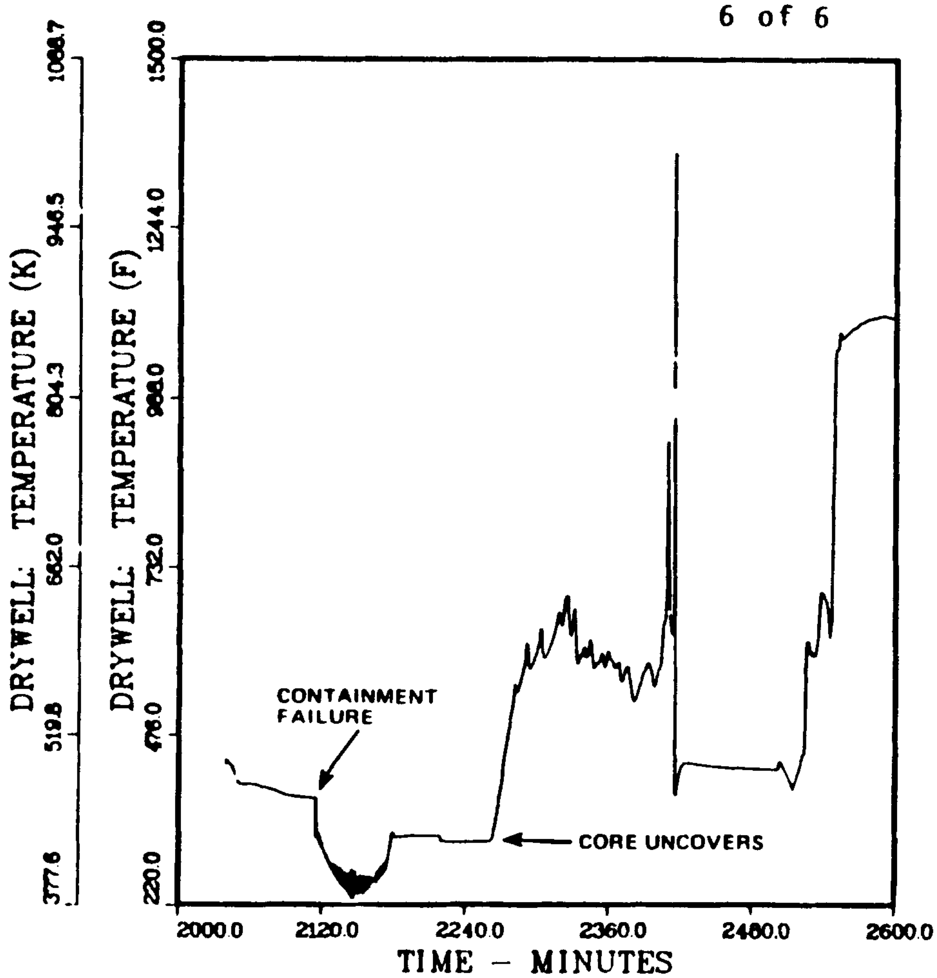
Unmitigated Loss of DHR - drywell pressure (drywell coolers operated until failure).



Unmitigated Loss of DHR - drywell temperature (drywell coolers operated until failure).



Drywell pressure.



Drywell temperature.

heat, drywell coolers are still operating, and decay heat loads are being sent to the suppression pool through the SRVs. This continues until about the 10 hour point when temperature turns from a negative to a positive slope. Close examination of the suppression pool temperature profile shown in Figure C-23 and the drywell pressure profile shown in Figure C-24 produces an explanation for this change in slope. At the 10 hour point the suppression pool has actually reached the boiling point for the indicated drywell pressure. Thus, the steam energy built up in the wetwell due to decay heat loads is now transferred to the drywell. This results in a positive sloped temperature and pressure profile within the drywell. This trend continues in all three profiles until the 17 hour point when the drywell coolers are assumed to fail due to high temperature (200°F). This causes the slope of the drywell temperature profile to increase until a new equilibrium is established accounting for the loss of the drywell coolers. The drywell cooler failure has little effect on the drywell pressure and wetwell temperature profiles. The next point of interest occurs at about 23 hours when operator control of the SRVs fails due to insufficient pressure differential. In scenario 6, this causes the vessel to repressurize. During this time, the wetwell gets a short reprieve from the decay heat load previously being dumped through the SRVs. This is not the case in scenario 7 where the stuck valve remains open to continue steam removal from the vessel to the wetwell. However, this produces only a minor difference in the profiles and the two cases end up at the same temperature and pressure. There is a small reduction in the slope of the wetwell temperature profile seen in Figure C-23 following the SRV failure due to the lack of SRV flow until the reactor vessel is fully repressurized (scenario 6). This process of decay heat removal to the wetwell and drywell atmospheres continues until containment pressure finally reaches the failure level. Note that this sequence is a very slow, gradual process needing over 35 hours to complete. Following containment failure, for purposes of this study, it is assumed that all coolant injection is lost. Note that in fact, core cooling capability may survive containment failure thereby preventing core melt. Temperatures initially decrease as energy is released to the environment, but rapidly increase in the drywell as the core uncovers. Reactor vessel breach is estimated at about 39 hours.

2.5.4 TC

The TC (with MSIV closure) sequence is characterized by the control rods failing to insert following a transient event which causes the power conversion system to fail or otherwise isolate. Reactor power levels at a point dependent on the coolant makeup rate. Heat from the fission process is dumped to the suppression pool through the safety relief valves. With the reactor remaining at power the suppression pool is stressed beyond the capacity of the heat removal system. This leads to a temperature rise in the wetwell causing containment pressure to increase to the point of failure. Two variations of this sequence were investigated to construct the TC environmental profiles.

In both scenarios, the operator manually controls water level initially and starts suppression pool cooling. In scenario 8 the operator depressurizes the vessel 1000 seconds into the accident, while in scenario 9, a safety relief valve sticks open 250 seconds into the accident. The results for both these scenarios were very similar so a composite of the scenarios was constructed to form the environmental profiles for the TC-MSIV closure sequence.

Figure C-25 displays LTAS predicted water level behavior with time for the operator depressurization scenario. Note that the water level drops to below the top of the active fuel within the first 300 seconds. However, the core is subsequently re-covered as the operator attempts to keep the water level at the top of the active fuel. Subsequent highs and lows in water level are attained. Figures C-26 and C-27 show the LTAS results for drywell and wetwell temperatures and drywell pressure which agree quite well with MARCH data (Ref. 10, pgs 6-21,6-22, and 6-45 to 6-50). Figures C-28 and C-29 display the MARCH data used to complete the profiles beyond the 4000 second (about 1 hour) cut-off for the LTAS data. The resulting profiles are shown in Figures C-30, C-31, and C-32. A brief discussion of the resulting environmental profiles follows.

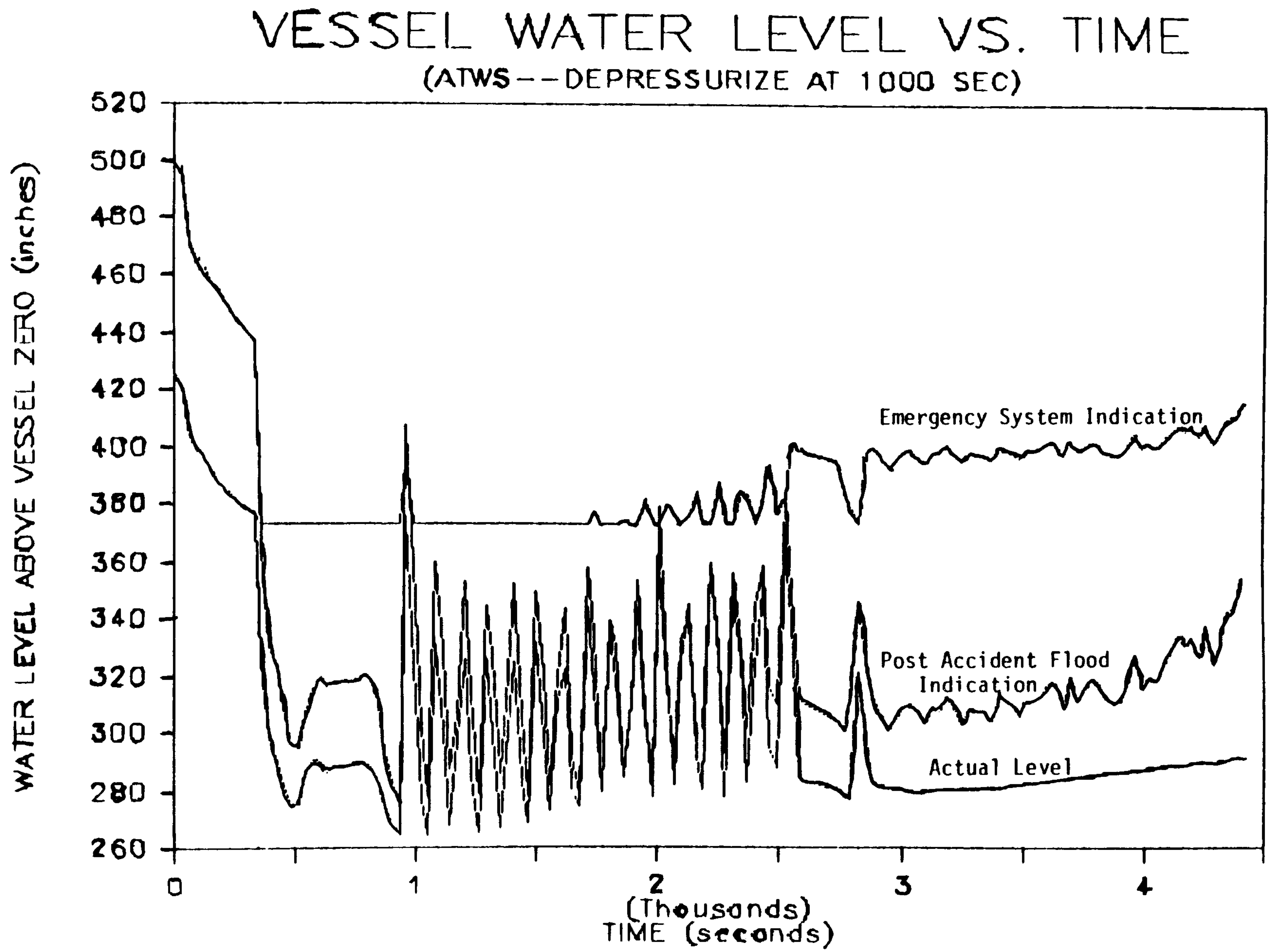


FIGURE C-25 - LTAS PREDICTED WATER LEVEL FOR THE EIGHTH SCENARIO

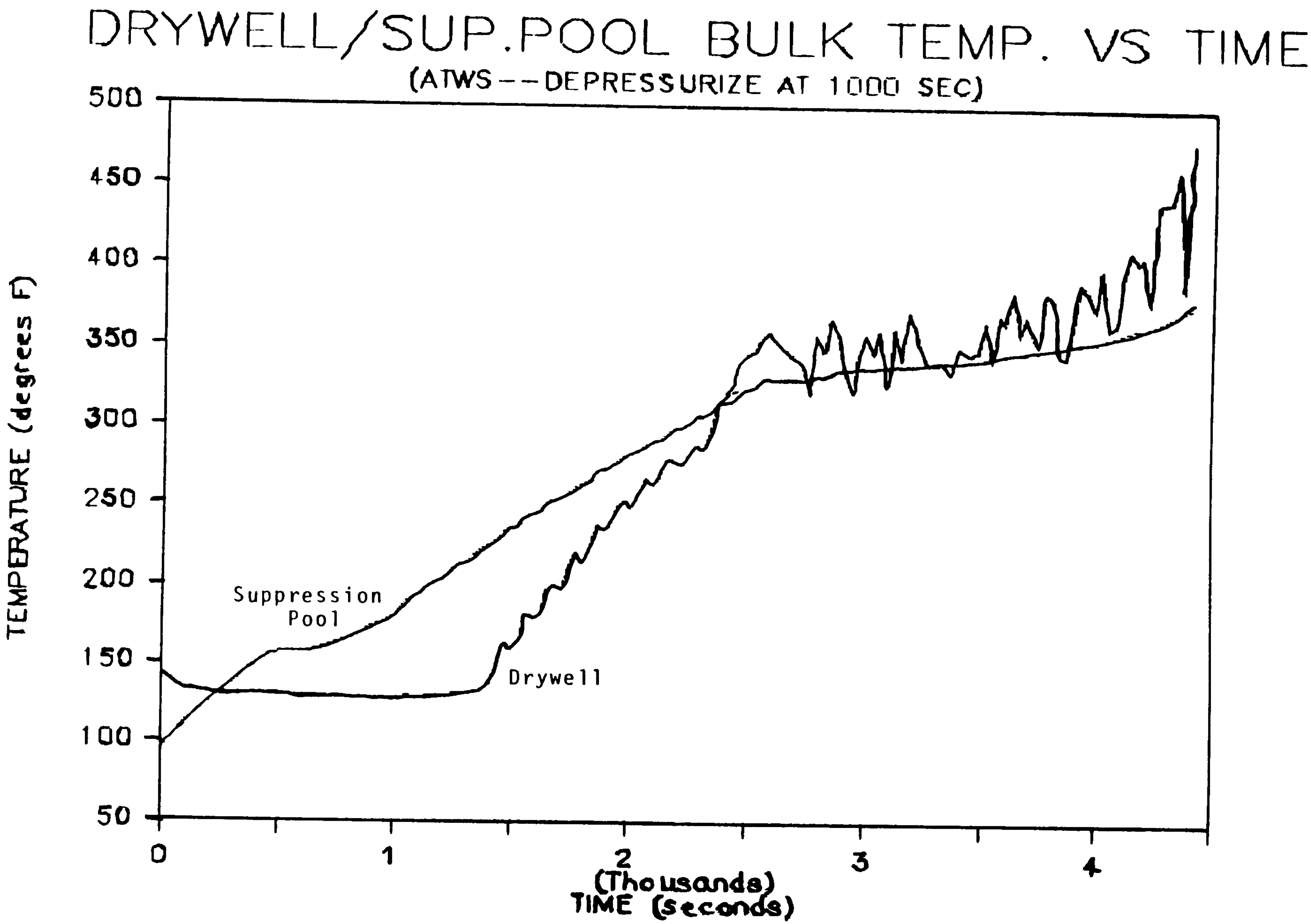


FIGURE C-26 - LTAS PREDICTED DRYWELL AND WETWELL TEMPERATURES FOR THE EIGHTH SCENARIO

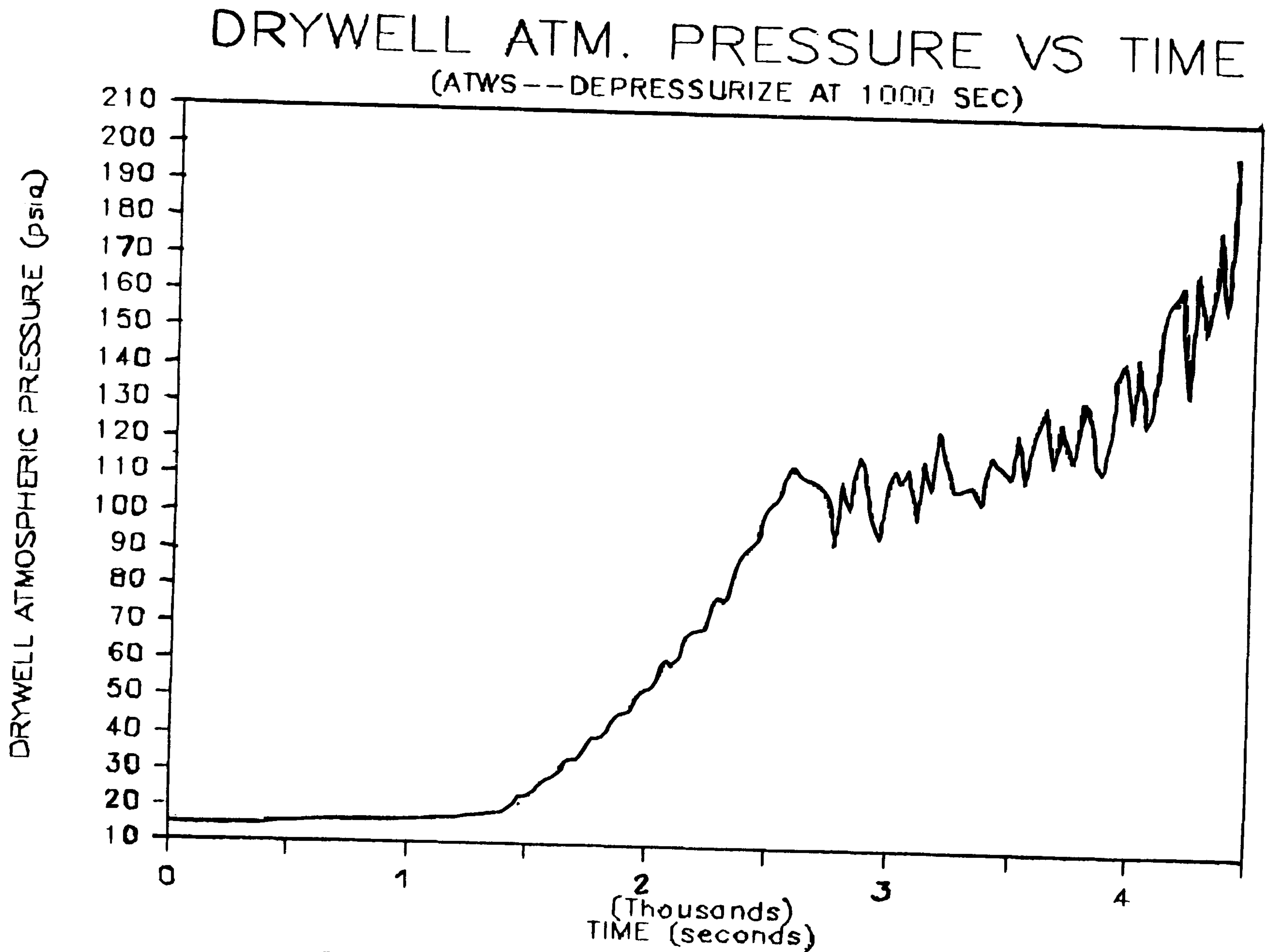


FIGURE C-27 - LTAS PREDICTED CONTAINMENT PRESSURE FOR THE EIGHTH SCENARIO

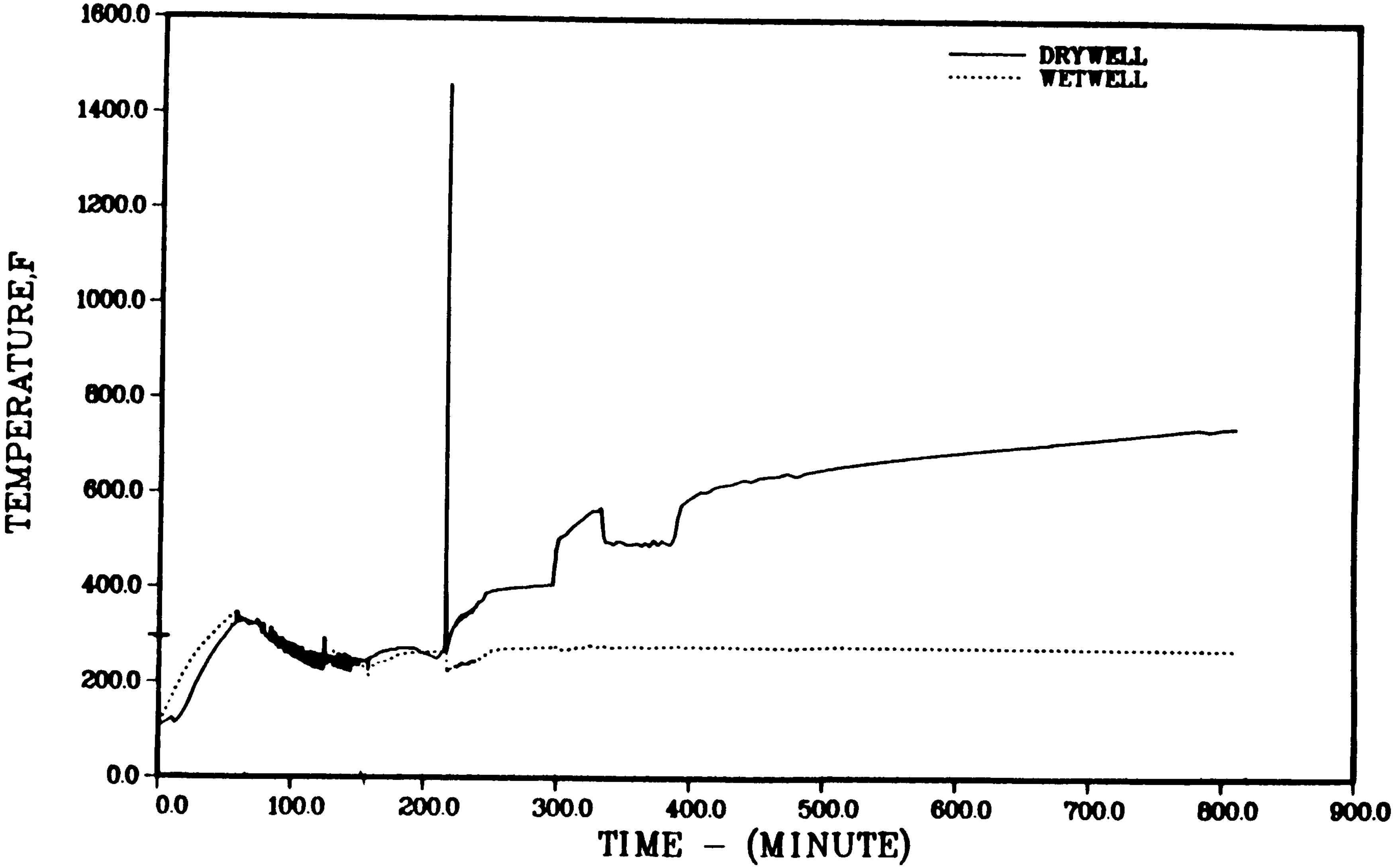


FIGURE C-28 GAS TEMPERATURES IN CONTAINMENT VOLUMES - SEQUENCE TC

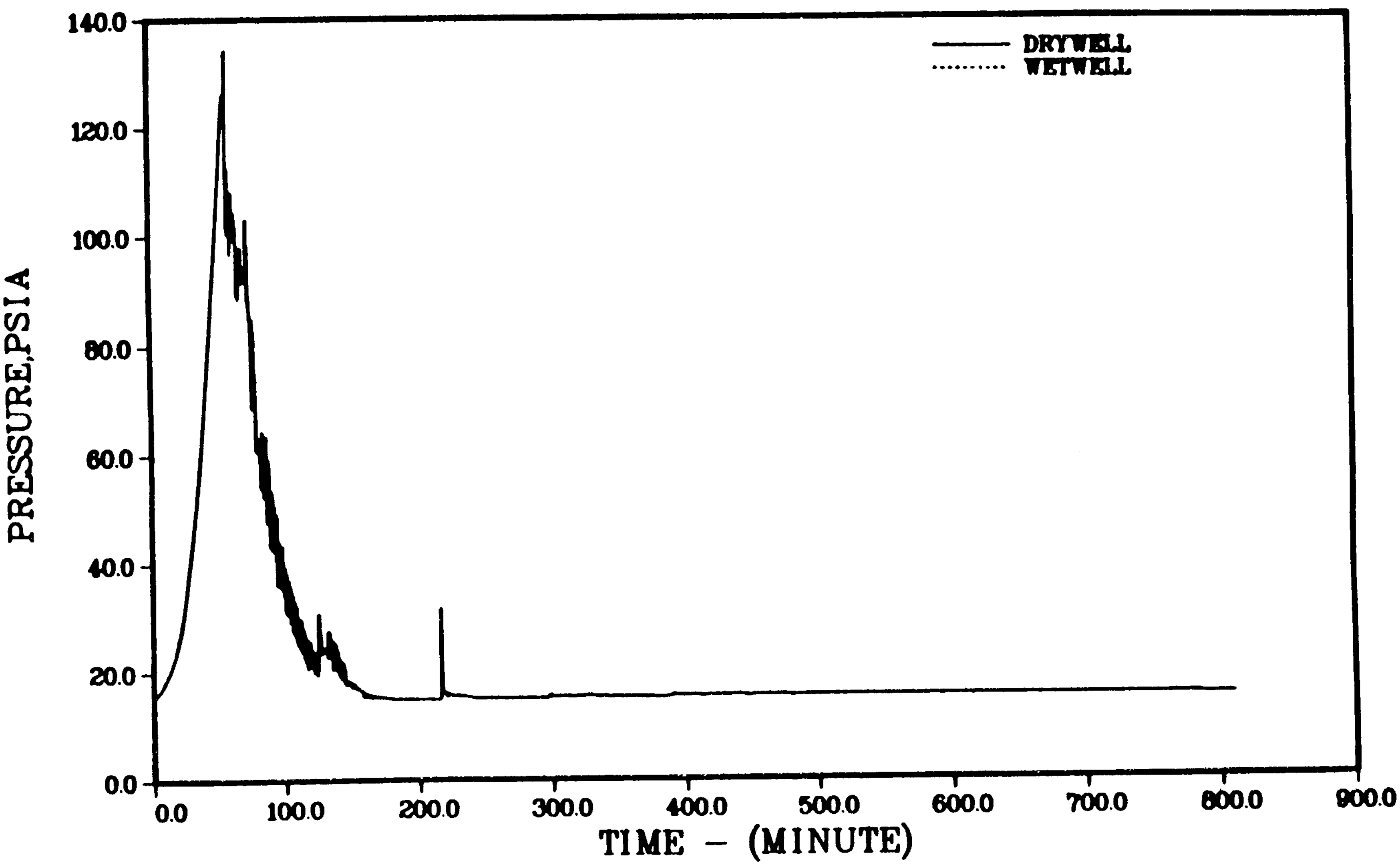


FIGURE C-29 PRESSURES IN CONTAINMENT VOLUMES - SEQUENCE TC

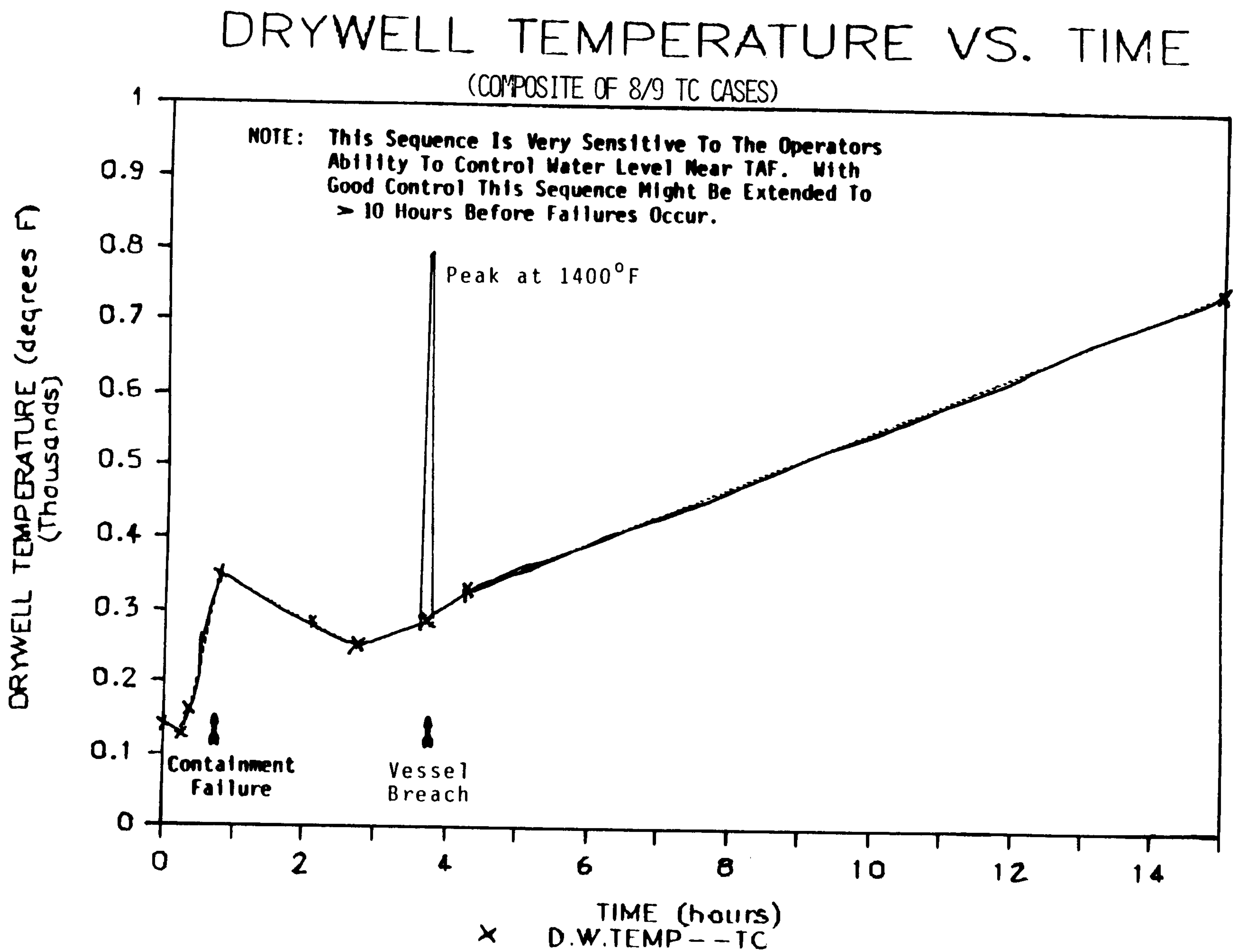


FIGURE C-30 - DRYWELL TEMPERATURE PROFILE FOR THE EIGHTH AND NINTH SCENARIO

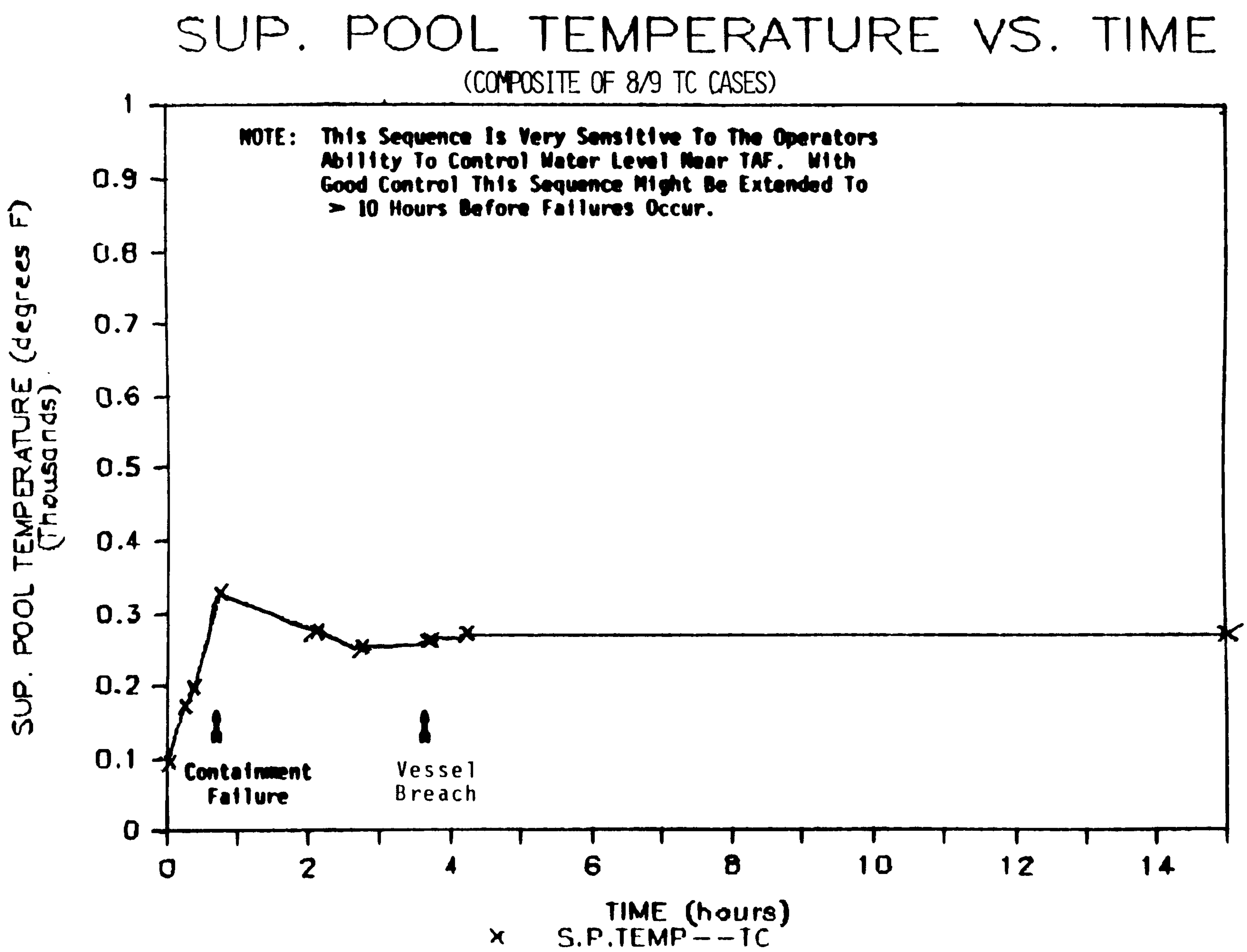


FIGURE C-31 - WETWELL TEMPERATURE PROFILE FOR THE EIGHTH AND NINTH SCENARIO

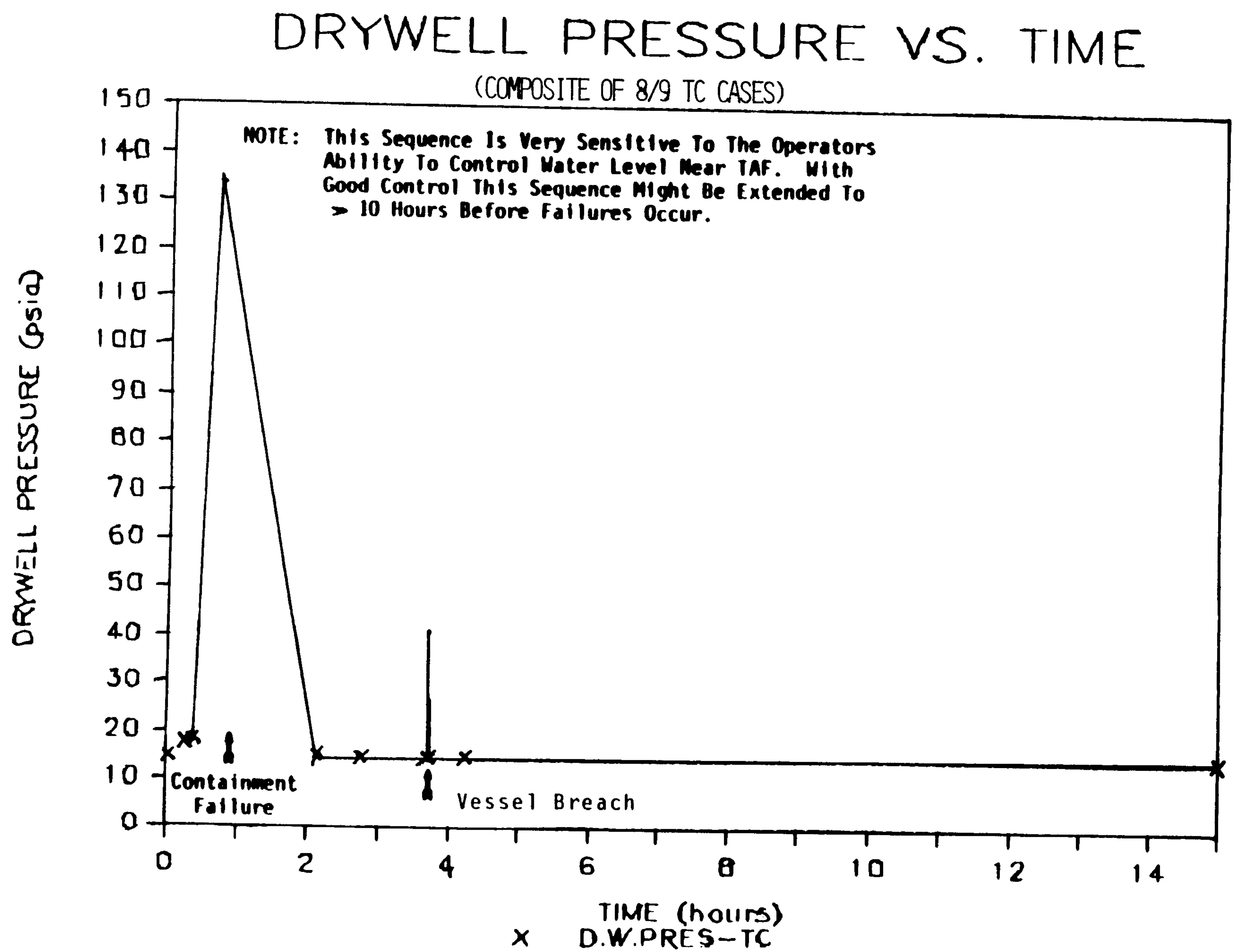


FIGURE C-32 - CONTAINMENT PRESSURE PROFILE FOR THE EIGHTH AND NINTH SCENARIO

Drywell temperature behavior for this sequence is best understood by examining Figures C-26 and C-30 closely. Note that drywell temperature remains constant for the first 1400 seconds of the accident. During this time the reactor is dumping the majority of its energy to the wetwell and the drywell coolers are operating. At 1400 seconds into the accident the suppression pool temperature has reached the boiling point for the containment pressure. Once the wetwell starts to boil, its energy is released to the drywell. Thus the drywell temperature and pressure start to rapidly climb at this point. This continues to the point of containment failure about one hour into the accident. At this point vessel injection is assumed to fail and the reactor is shut down due to the lack of a moderator. The core uncovers leading to vessel failure with a rapid drywell temperature response, and then a slow rise in compartment temperature due to radiative heating from the debris bed.

The wetwell profile shown in Figure C-31 shows how the suppression pool temperature initially leads the drywell temperature but then matches it once saturation conditions are reached. After the point of vessel failure, decay heat energy is predominantly deposited in the drywell so the wetwell temperature profile levels to a constant value.

The pressure profile shown in Figure C-32 shows how containment pressure starts to rapidly rise once the wetwell heat removal capacity is exceeded and saturation conditions are reached. This rapid rise continues to the point of containment failure at about 132 psia. The pressure profile then drops to atmospheric pressure over the next hour since the containment leak rate is assumed relatively small. At the point of vessel failure such a large heat load occurs so quickly that the compartment can again pressurize because the containment opening to the atmosphere is small. Once this heat load is absorbed, the containment pressure drops to and maintains atmospheric pressure.

A third scenario (8A) models a TC sequence with the MSIVs remaining open (no isolation of the PCS). Current estimates indicate that the PCS is capable of dissipating about 20% - 25% power with the main turbine off line. Power varies between 15% and 35% in this sequence depending on the success of water level control. Since the degree of level control is not expected to be of primary concern to the operators, it is assumed that average power is approximately 30%. Thus the sequence was modeled as a TW sequence with 5% power being sent to the suppression pool via the SRVs. The sequence was modeled in this way because the LTAS code doesn't have provisions to model the MSIVs open for any transient sequence. By forcing the calculated value of the decay heat load for the TW sequence to be a constant 5%, a good approximation of the TC/MSIV open sequence can be obtained. The entire sequence was modeled using the LTAS code. No MARCH data was used. Figures C-32A,B,C, and D show vessel water level, drywell temperature, suppression pool temperature, and containment pressure respectively. Observing these figures shows that the sequence behavior is almost identical to the previous TC sequences with the MSIVs shut. The only difference is having the MSIVs open allows some of the energy to be directed to the PCS and thus it takes a longer time for the suppression pool to reach saturation conditions resulting in the rapid containment pressure rise as previously described. The MSIV open case takes approximately 4 hours to reach the point of containment failure as opposed to 1.0 hours. The remainder of the temperature and pressure profiles (after containment failure) were estimated using the TC (MSIV-closed) profiles described earlier. Core melt/vessel breach is estimated at approximately 6.7 hours into the sequence.

2.5.5 TQUV

The TQUV sequence is characterized by a transient-induced scram followed by a loss of all high and low pressure injection except for limited Control Rod Drive (CRD) system flow. The computer data for this sequence assumed one CRD pump was in operation. Three separate scenarios for this sequence were examined to determine the environmental profile sets for this accident. The first involved the operator depressurizing the reactor vessel, the second was a case of a stuck open relief valve, and the third involved no operator action with the vessel remaining at pressure. These are LTAS scenarios

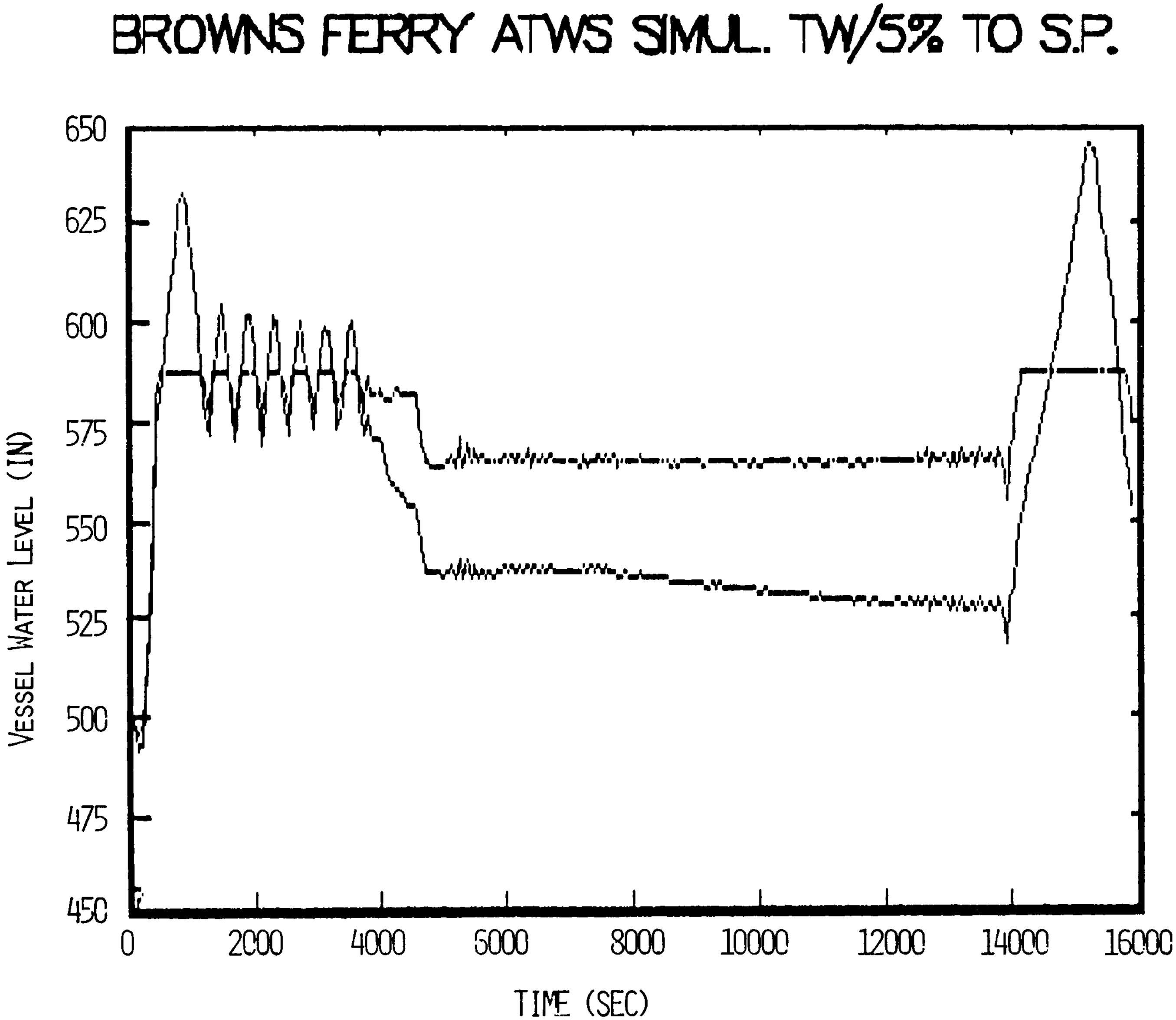


FIGURE C-32A - LTAS PREDICTED WATER LEVEL FOR SCENARIO 8A

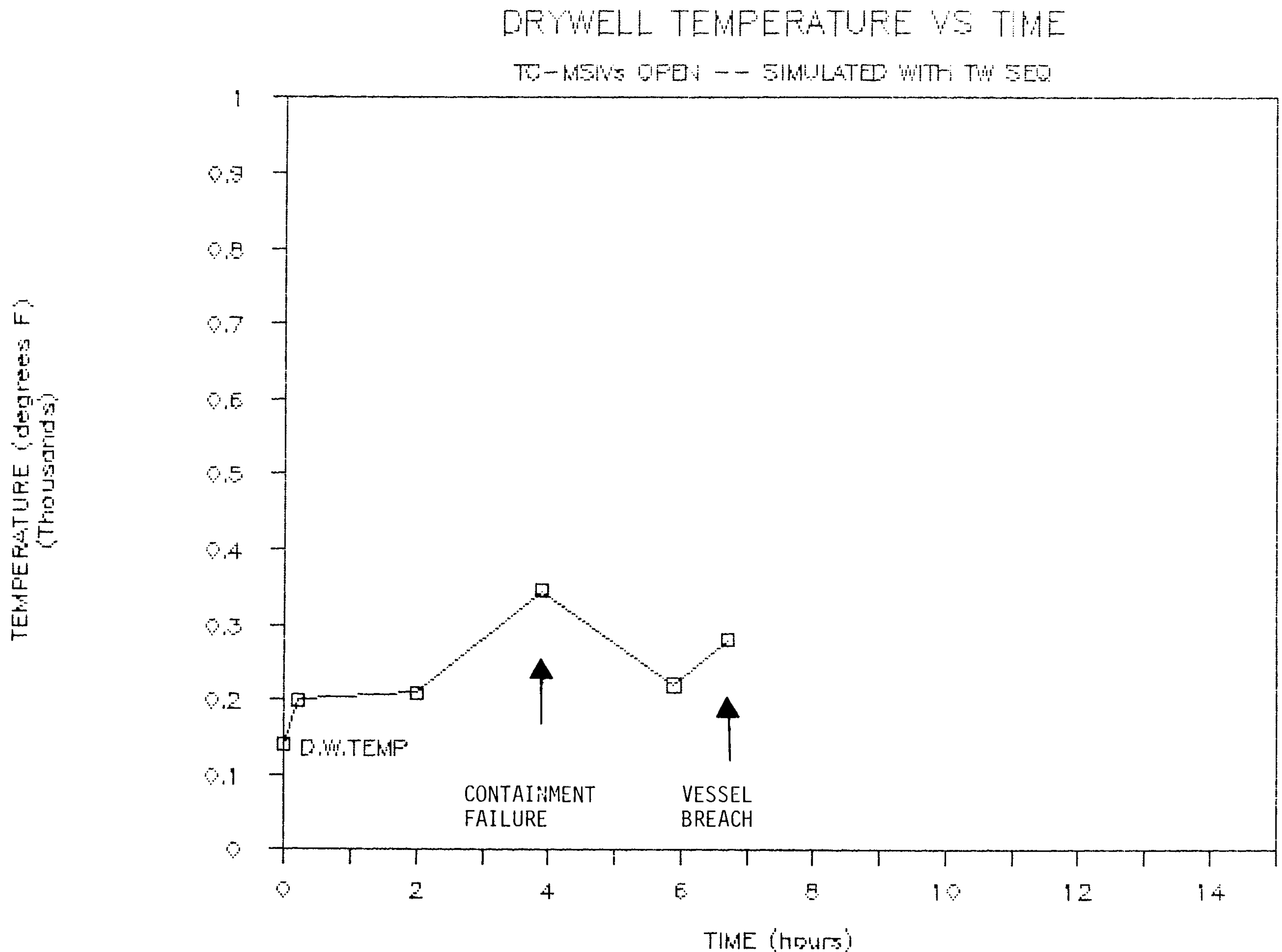


FIGURE C-32B - LTAS PREDICTED DRYWELL TEMPERATURE FOR SCENARIO 8A

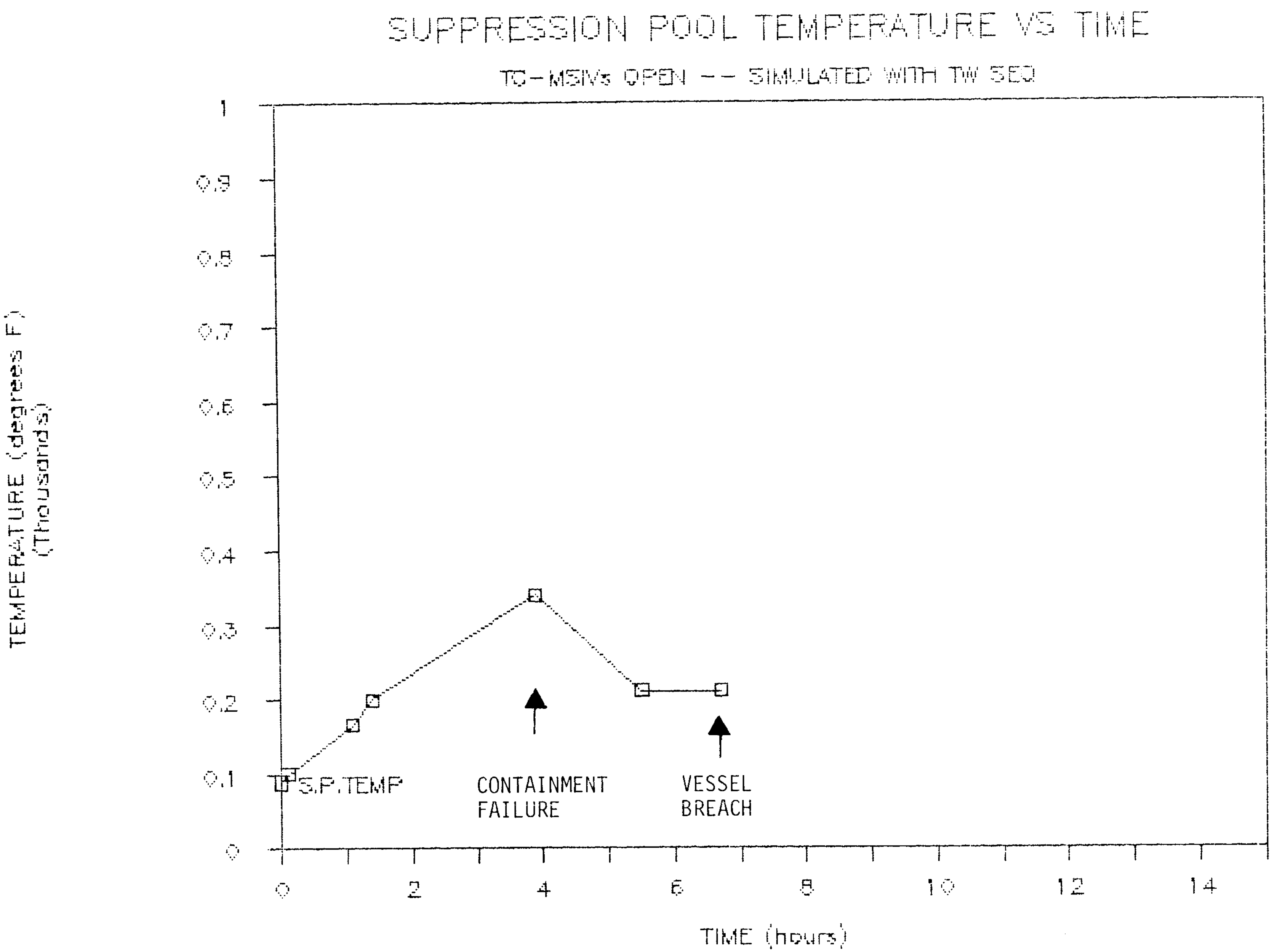


FIGURE C-32C - LTAS PREDICTED SUPPRESSION POOL TEMPERATURE FOR SCENARIO 8A

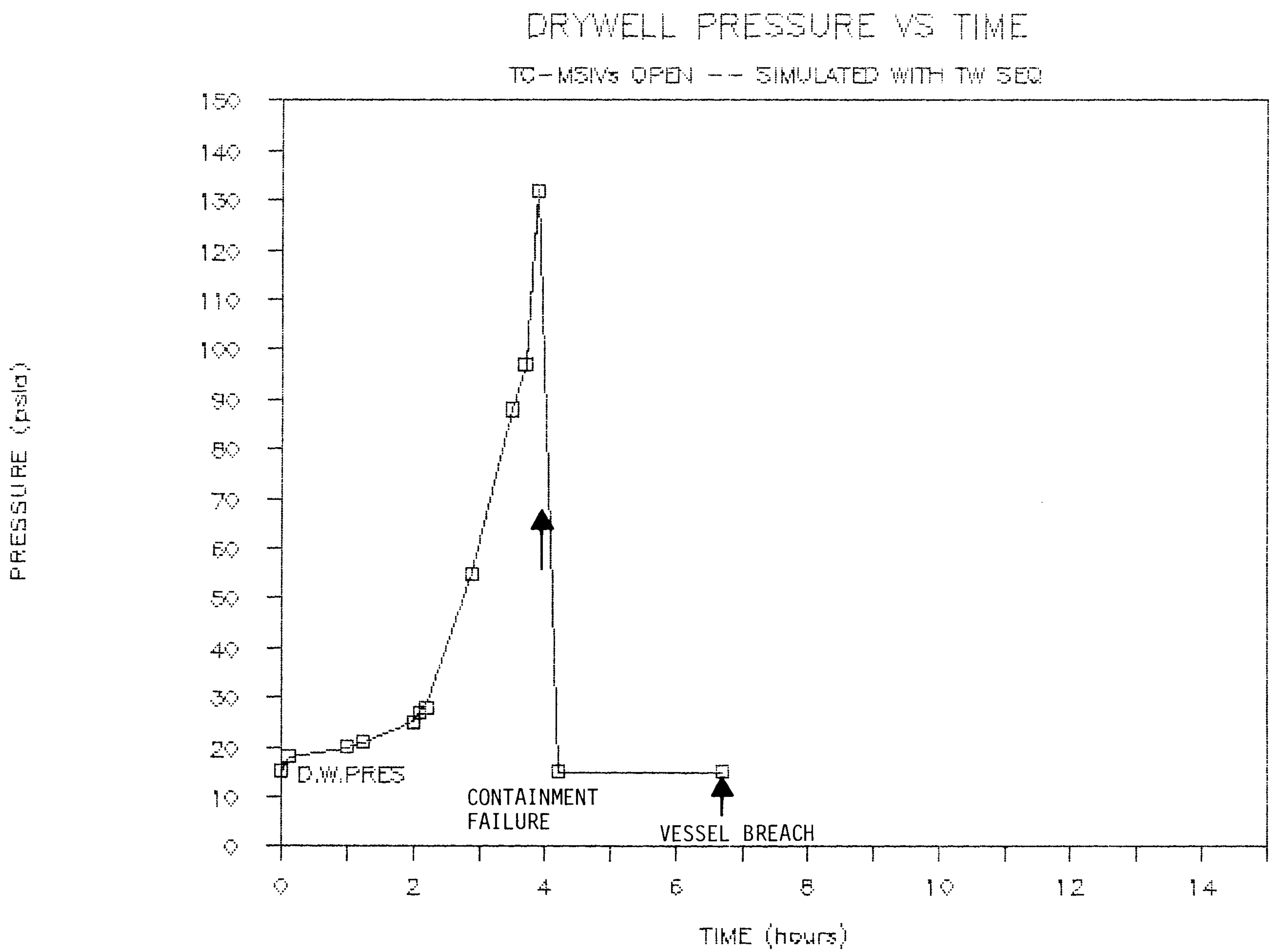


FIGURE C-32D - LTAS PREDICTED CONTAINMENT (DRYWELL) PRESSURE FOR SCENARIO 8A numbers 10, 11, and 12 respectively. It was found that scenarios 10 and 11 were similar enough to be grouped together into a single set of environmental profiles, while the no operator action case had enough unique points to warrant a separate profile construction.

The first set of profiles considered is the one constructed for scenario 12, the no operator action or pressurized case. In this scenario the CRD pump must work against a higher head and this results in a lower flow output. Figure C-33 shows the LTAS results for water level behavior in this scenario. The core uncovers about 2000 seconds into the accident (about 33 min.). Therefore, LTAS data was used up to 2000 seconds. Figures C-34 and C-35 show the LTAS results for drywell and wetwell temperature and drywell pressure used to construct the profiles up to the 2000 second point. The remainder of the profiles were developed using the MARCH timing data represented by Case 1 of Table C-6 (Ref. 3, pg. 34) and using the general profile shapes of a similar boil-off sequence (TB) with additional guidance provided by old MARCH run plots (Ref. 5, pg. 31). As was the case for the TB sequences, the temperatures and drywell pressure for TQUV were assumed to change insignificantly until the point of vessel breach since both are similar boil off calculations. At that point, temperature and pressure profiles were approximated by those of the TB sequence knowing that containment failure occurs at 430 minutes into the accident (from Table C-6) at a drywell temperature of 400-500 degrees F (when the electrical penetrations fail). The suppression pool temperature was assumed to not change significantly as was the case for TB. The drywell pressure pulse was approximated using the general shape of the pressure pulse for TB but for the time periods presented in Case 1 of Table C-6 for vessel breach and containment failure. The shape and general trends for the curves presented in Reference 5, page 31, added further validity to the approximations. Figures C-36, C-37, and C-38 display the resulting completed profiles. In terms of the profile behavior, the discussion in Section 2.5.1 applies here as the sequence is similar to a station blackout with no injection.

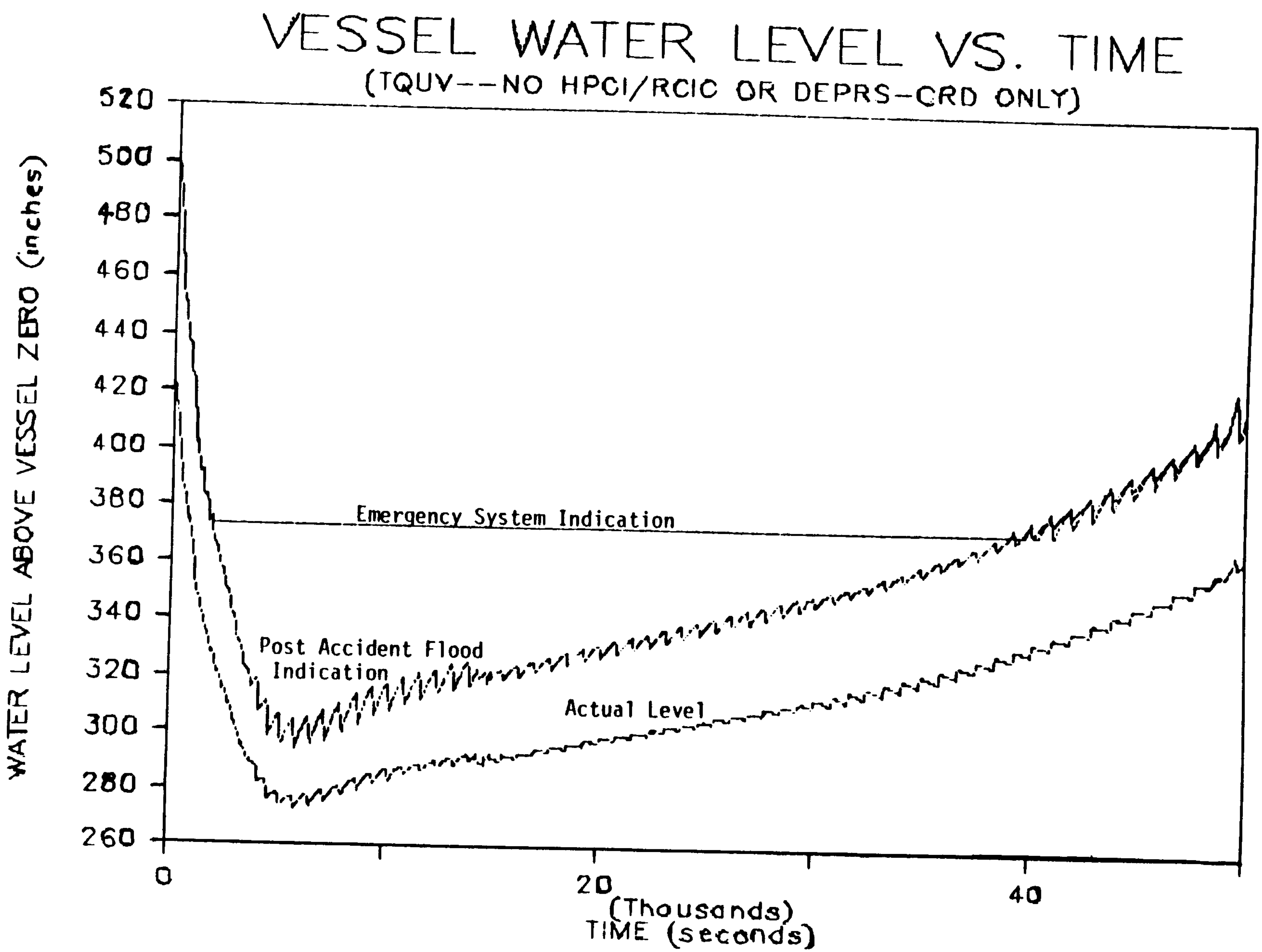


FIGURE C-33 - LTAS PREDICTED WATER LEVEL FOR THE TWELFTH SCENARIO

DRYWELL/SUP.POOL BULK TEMP. VS. TIME

(TQUV--NO HPCI/RCIC OR DEPRS-CRD ONLY)

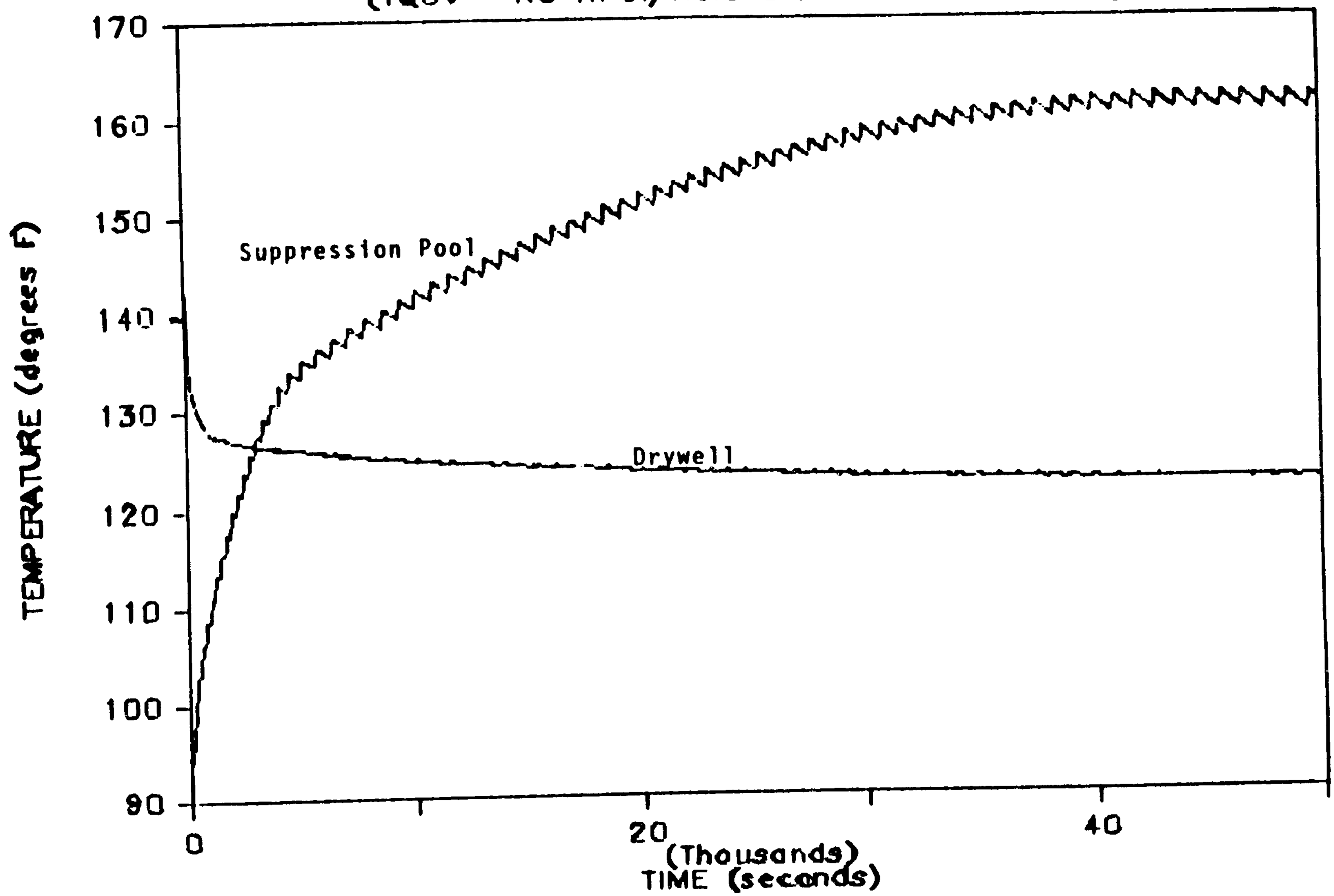


FIGURE C-34 - LTAS PREDICTED DRYWELL AND WETWELL TEMPERATURES FOR THE TWELFTH SCENARIO

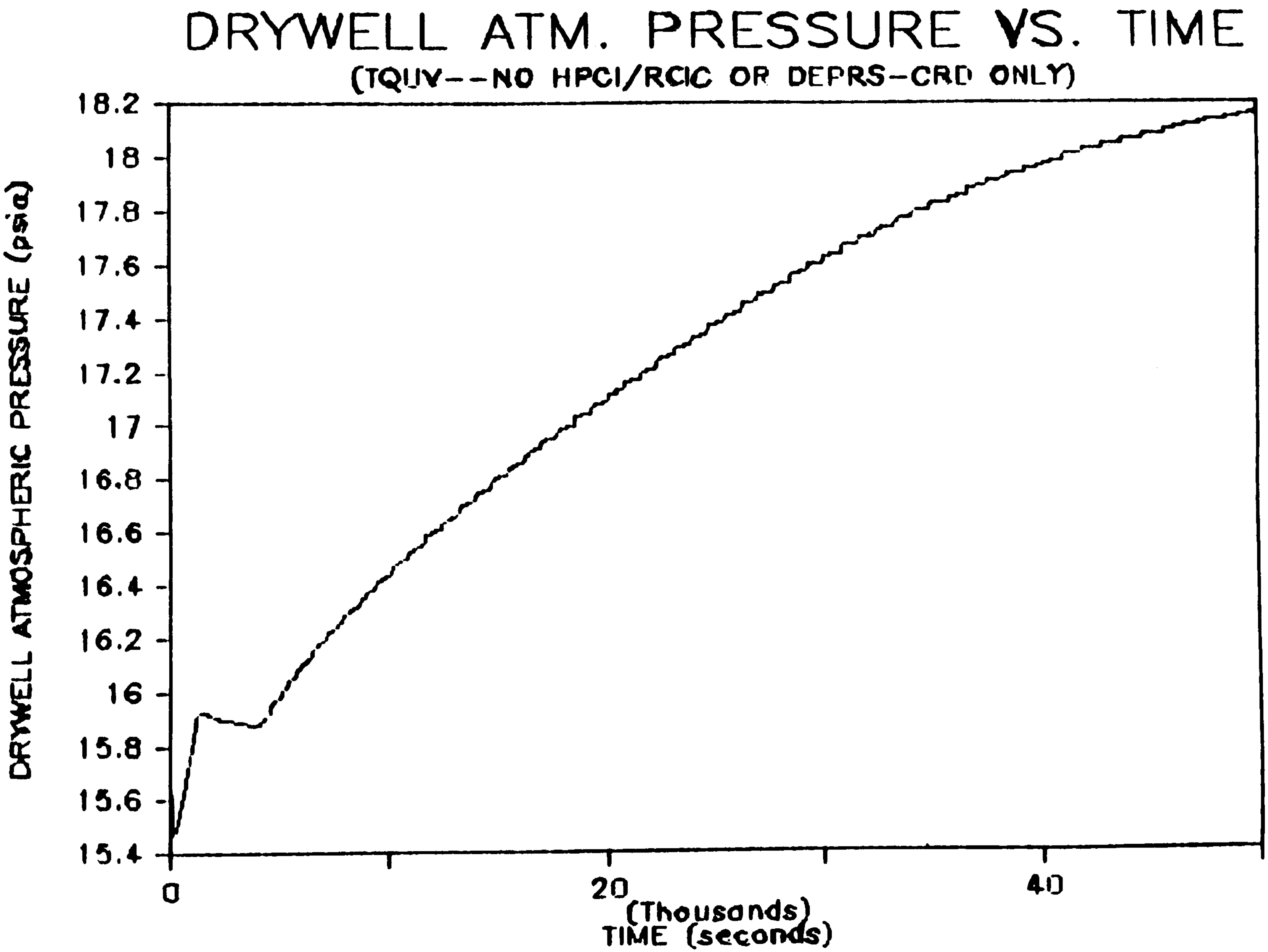


FIGURE C-35 - LTAS PREDICTED CONTAINMENT PRESSURE FOR THE TWELFTH SCENARIO

Table C-6 - Comparison of accident event timing for the at-pressure cases (with no operator action).

	Case 1[b]	Case 2[c]
Start of fuel melting	105	96
Core slump	276	145
RPV head failure[d]	280	211
Containment failure[e]	430	267

[a]Minutes after reactor scram.

[b]Pressure control between 7.38 and 7.72 MPa (1055 and 1105 psig) with average CRD injection of 6.68 L/s (~106 gpm).

[c]Pressure control between 7.38 and 7.72 MPa (1055 and 1105 psig) with no CRD injection.

[d]Case 1 head failure caused by vessel overpressurization after core slump rather than corium attack on head, as in Case 2.

[e]Due to overtemperature failure of dry-well electrical penetration assemblies.

Table C-7 - Comparison of accident event timing for the SORV cases with no operator action.

	Case 1[b]	Case 2[c]
Start of fuel melting	80	73
Core slump	206	111
RPV head failure[d]	434	253
Containment failure[e]	472	286

[a]Minutes after reactor scram.

[b]CRD injection increases from 104 gpm at 1100 psia to ~180 gpm after 60 min.

[c]No injection

[d]In Case 1, the corium debris is quenched when the core slumps; thus the corium must re-heat before attacking the bottom head.

[e]Due to overtemperature failure of the drywell electrical penetration assemblies.

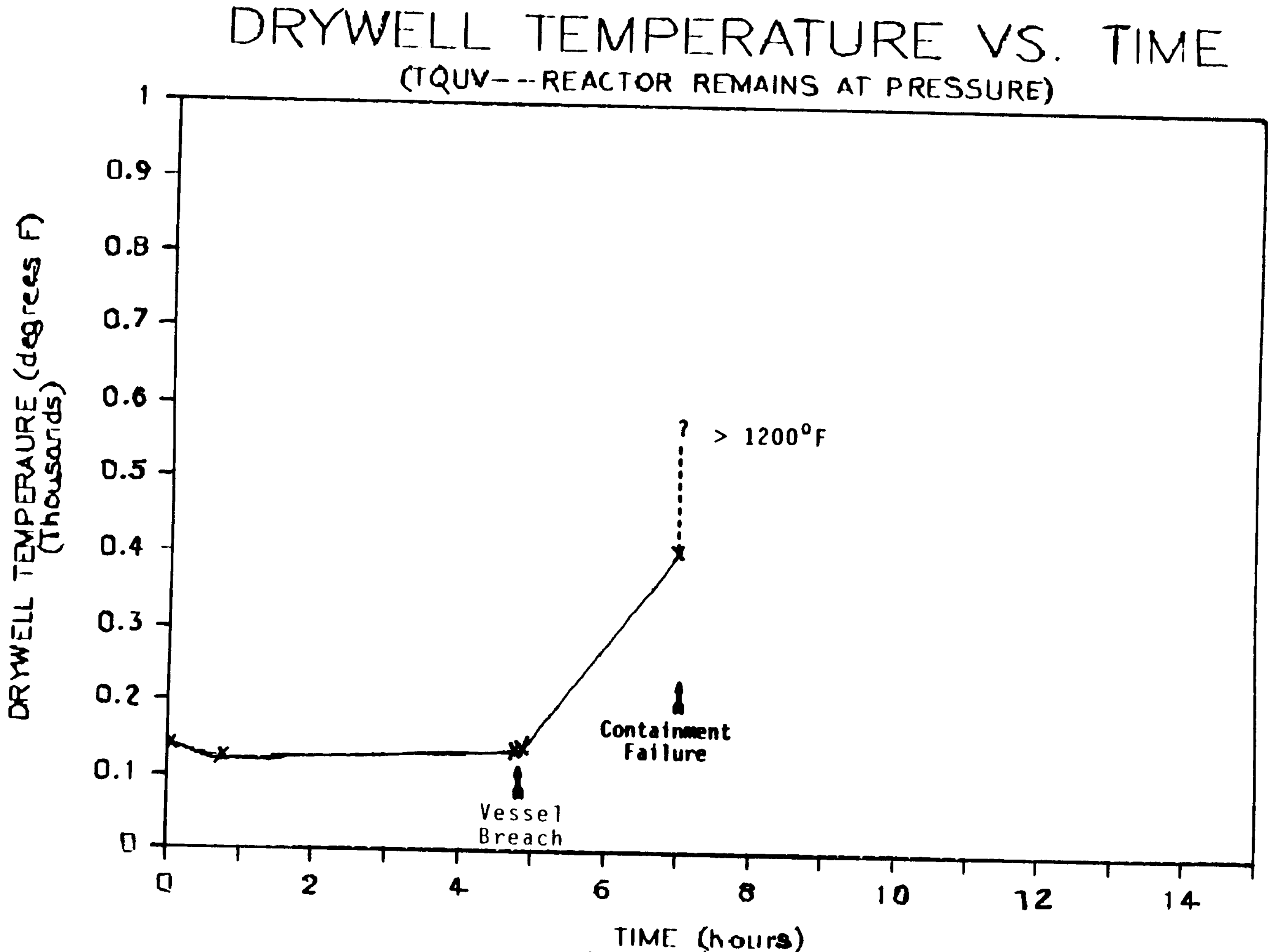


FIGURE C-36 - DRYWELL TEMPERATURE PROFILE FOR THE TWELFTH SCENARIO

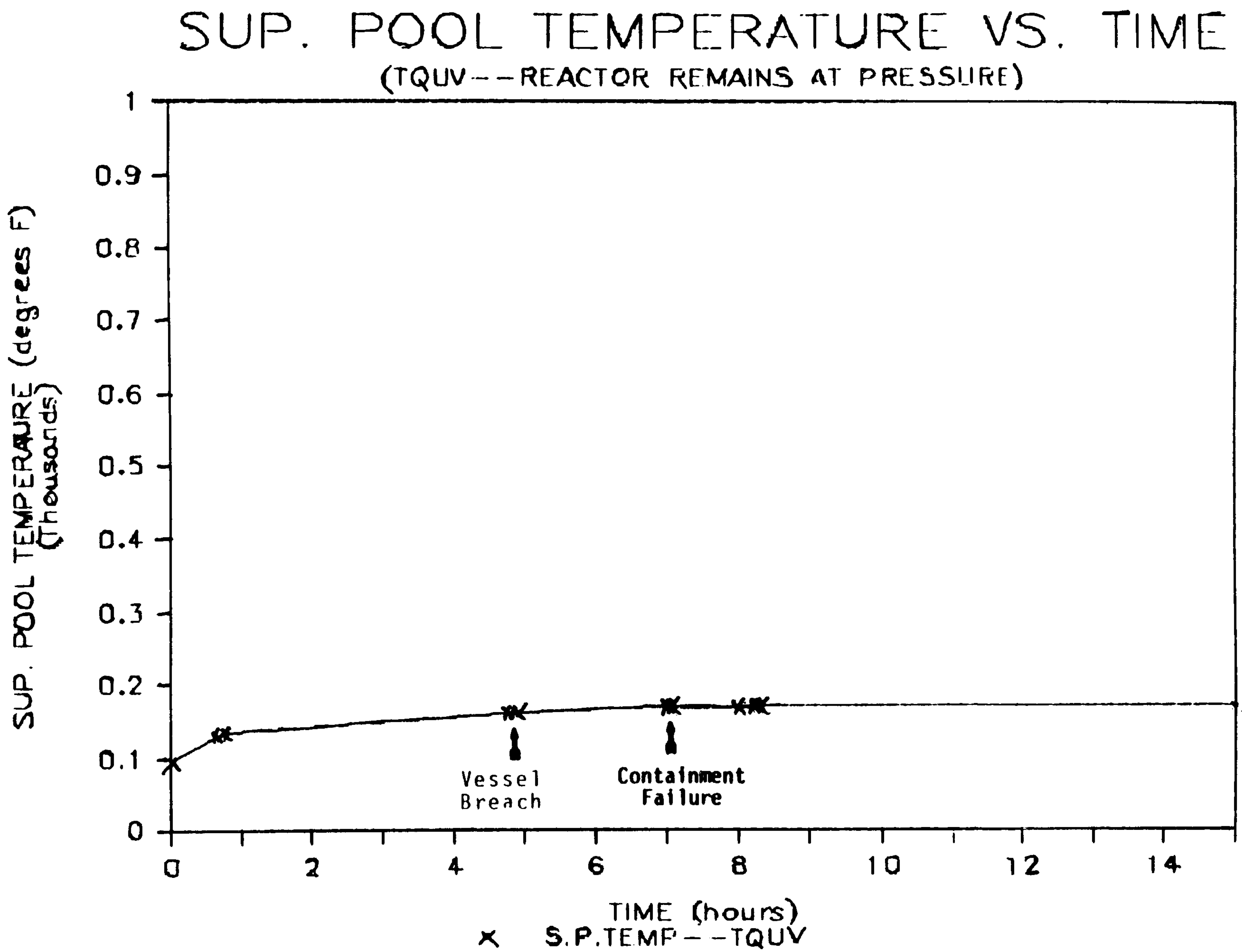


FIGURE C-37 - WETWELL TEMPERATURE PROFILE FOR THE TWELFTH SCENARIO

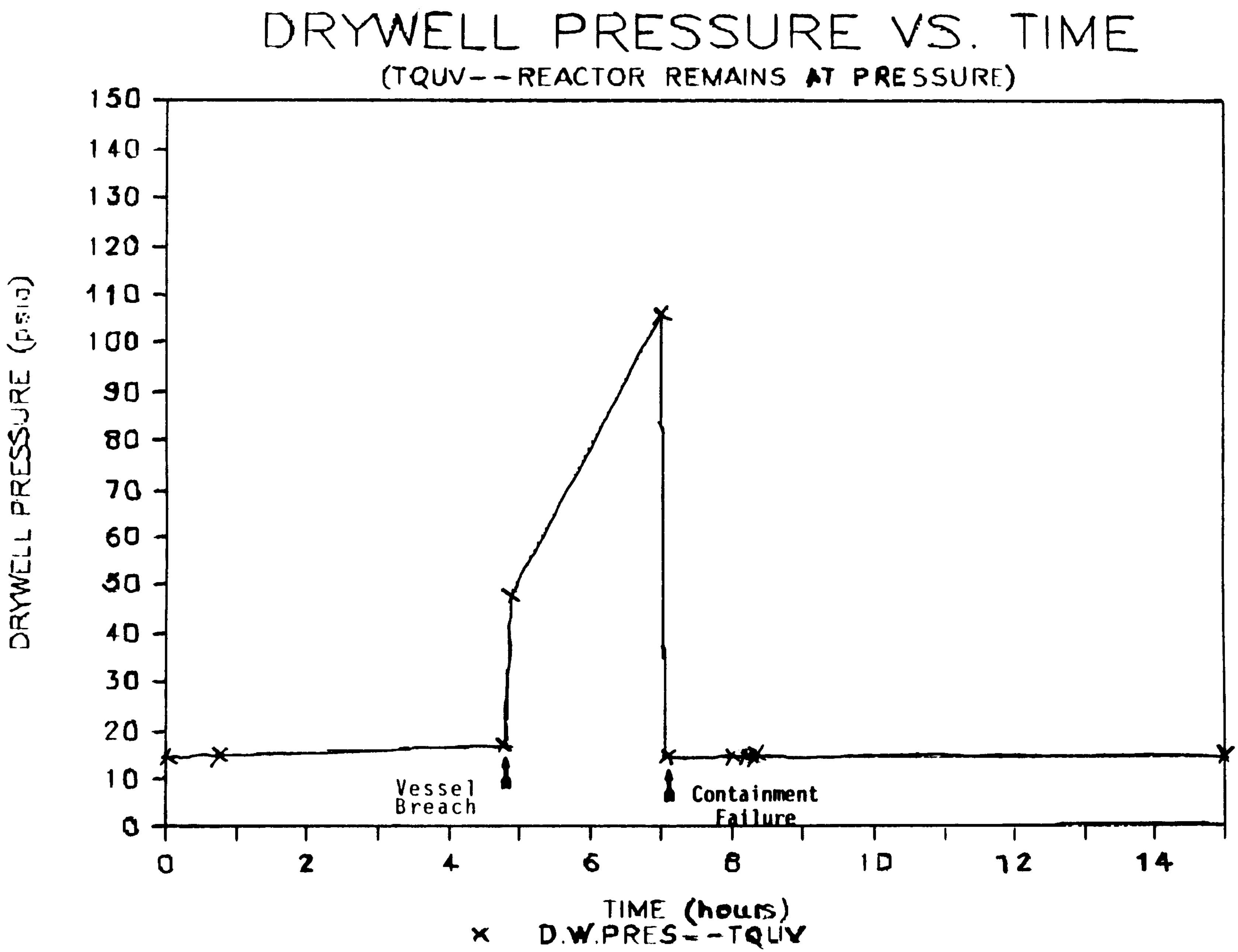


FIGURE C-38 - CONTAINMENT PRESSURE PROFILE FOR THE TWELFTH SCENARIO

The other set of profiles constructed was for those scenarios involving the reactor being depressurized (scenarios 10 & 11). Figure C-39 shows the LTAS predicted water level behavior for this case. Again the core uncovers about 2000 seconds into the accident. This time however, because the vessel is depressurized, the CRD pump is able to operate at a higher flow rate. This has the effect of postponing core damage and containment failure. Figures C-40 and C-41 display the LTAS data for drywell and wetwell temperature and drywell pressure used to construct the environmental profiles for the first 2000 seconds. MARCH data found in Table C-7 for Case 1 (Ref. 3, pg. 34) was used to complete the profiles. Figures C-42, C-43, and C-44 show the resulting completed profiles. Again the reader is referred to the text in Section 2.5.1 for an explanation as to the general behavior of these curves.

2.5.6 AE

The AE sequence is characterized by a large pipe break in the primary system such as in a recirculation line. The reactor blows down to the drywell which relieves to the wetwell through vertical vents. Injection systems are assumed to be inoperable. The blowdown results in subsequent core uncovery and melt. The suppression pool remains subcooled throughout the event. Only one scenario was investigated for this sequence because without injection, there are not many operator actions which can mitigate the severity of the accident. Therefore, a case with no operator action was run to construct the environmental profiles for this sequence.

Figure C-45 shows the LTAS predicted water level behavior for this sequence using the largest break size which can be currently analyzed by the code. The LTAS results indicate the core is uncovered 700 seconds (11 min.) into the accident. This is not in agreement with the MARCH data because the MARCH run was completed on the basis of a larger break size. Therefore, only MARCH data (Ref 10, pgs. 6-9,6-10,6-45,6-47, and 6-49) was used to complete this sequence in order to remain consistent throughout the profile construction. Figures C-46 and C-47 show the MARCH plots used to construct the profiles for drywell and wetwell temperature and drywell pressure. The

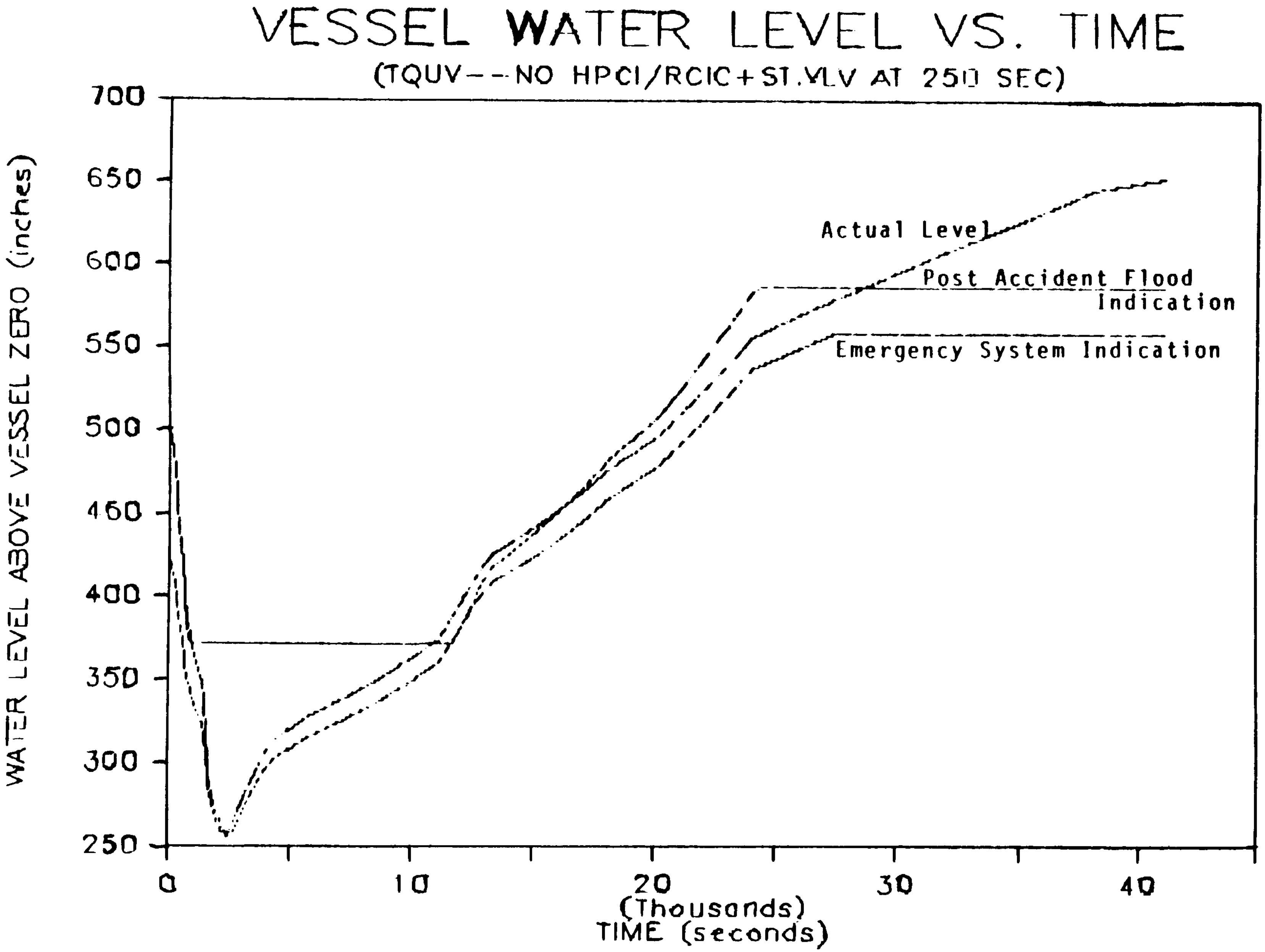


FIGURE C-39 - LTAS PREDICTED WATER LEVEL FOR THE TENTH AND ELEVENTH SCENARIO

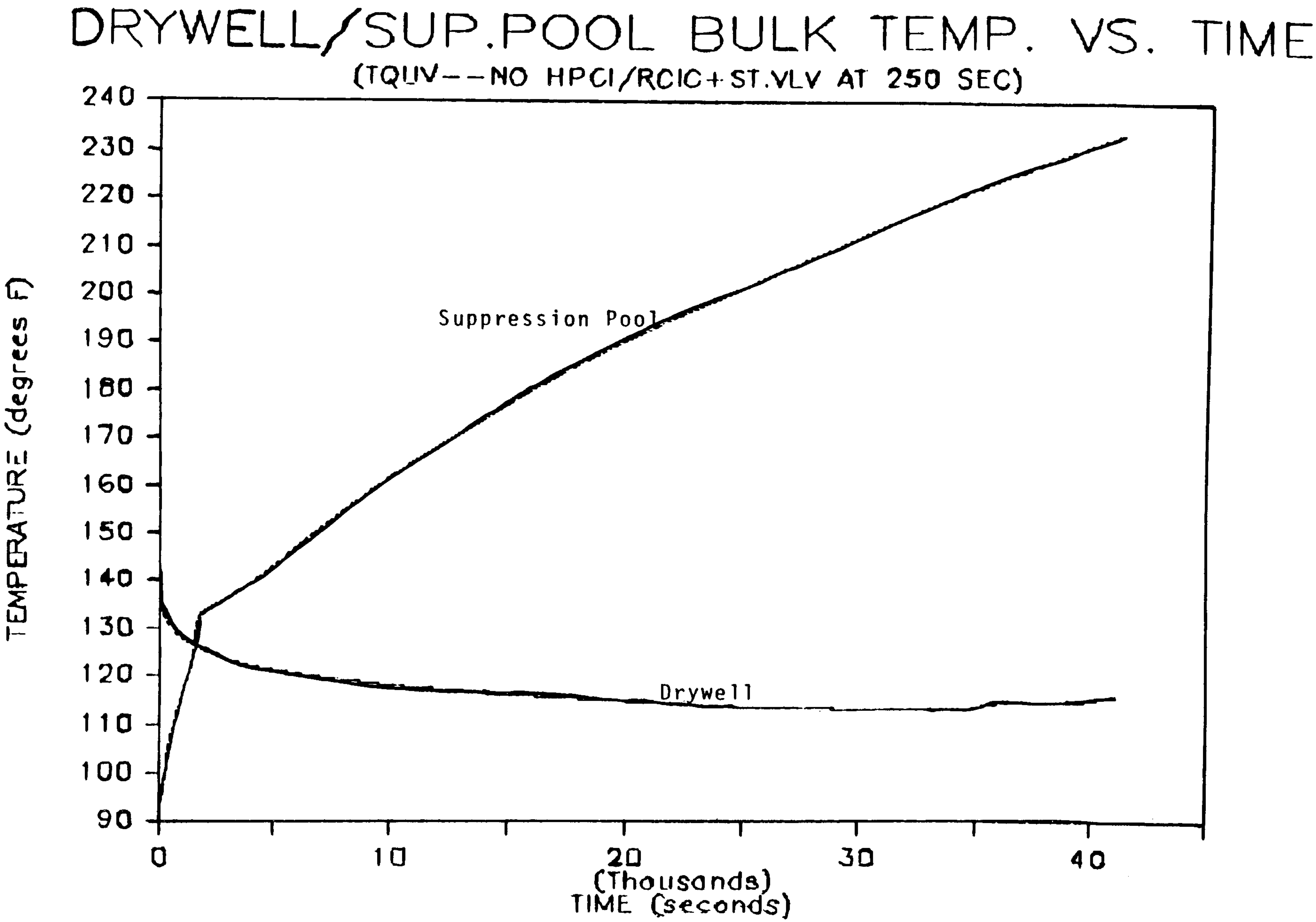


FIGURE C-40 - LTAS PREDICTED DRYWELL & WETWELL TEMPERATURES FOR THE TENTH AND ELEVENTH SCENARIO

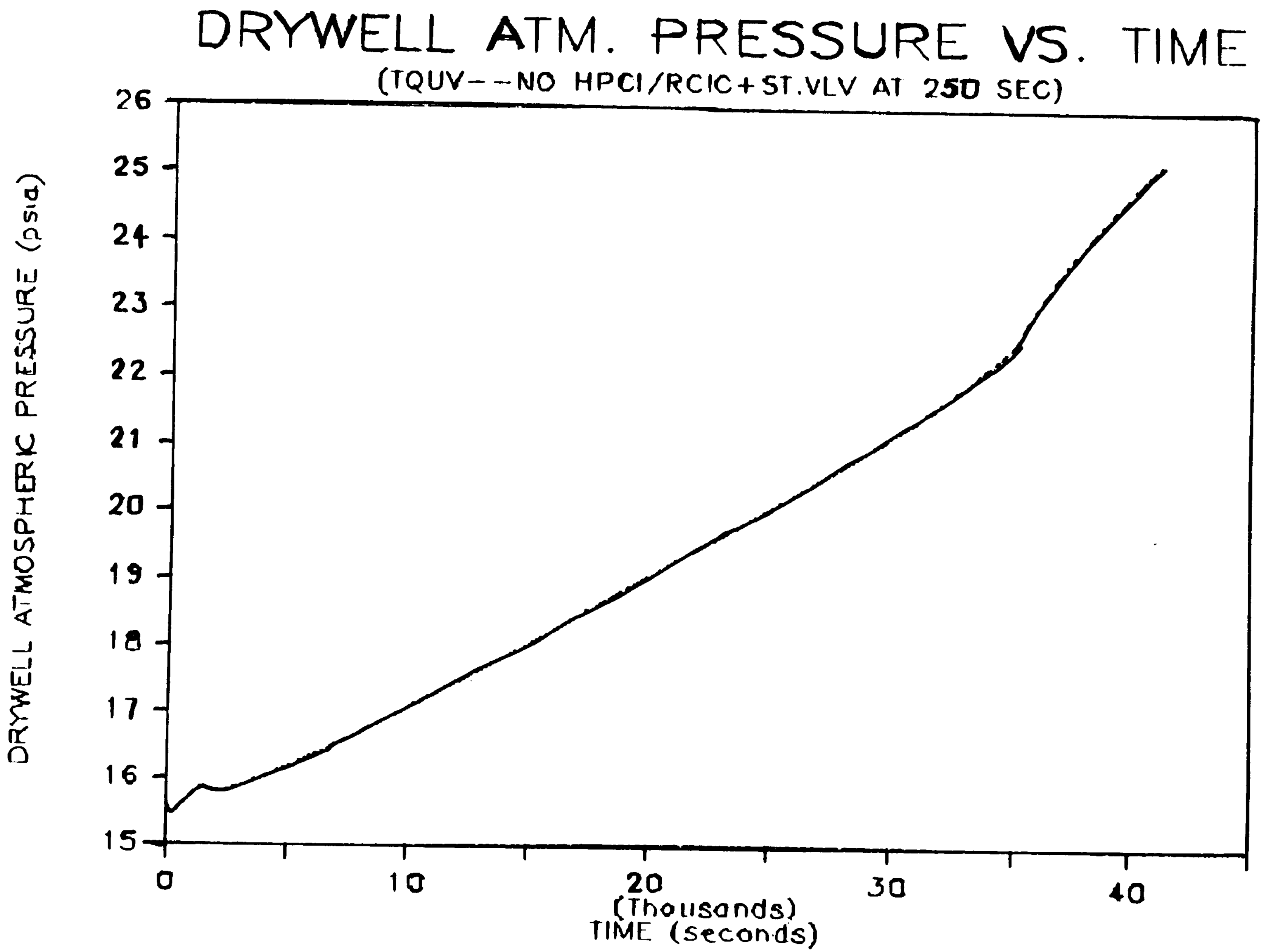


FIGURE C-41 - LTAS PREDICTED CONTAINMENT PRESSURE FOR THE TENTH & ELEVENTH SCENARIO

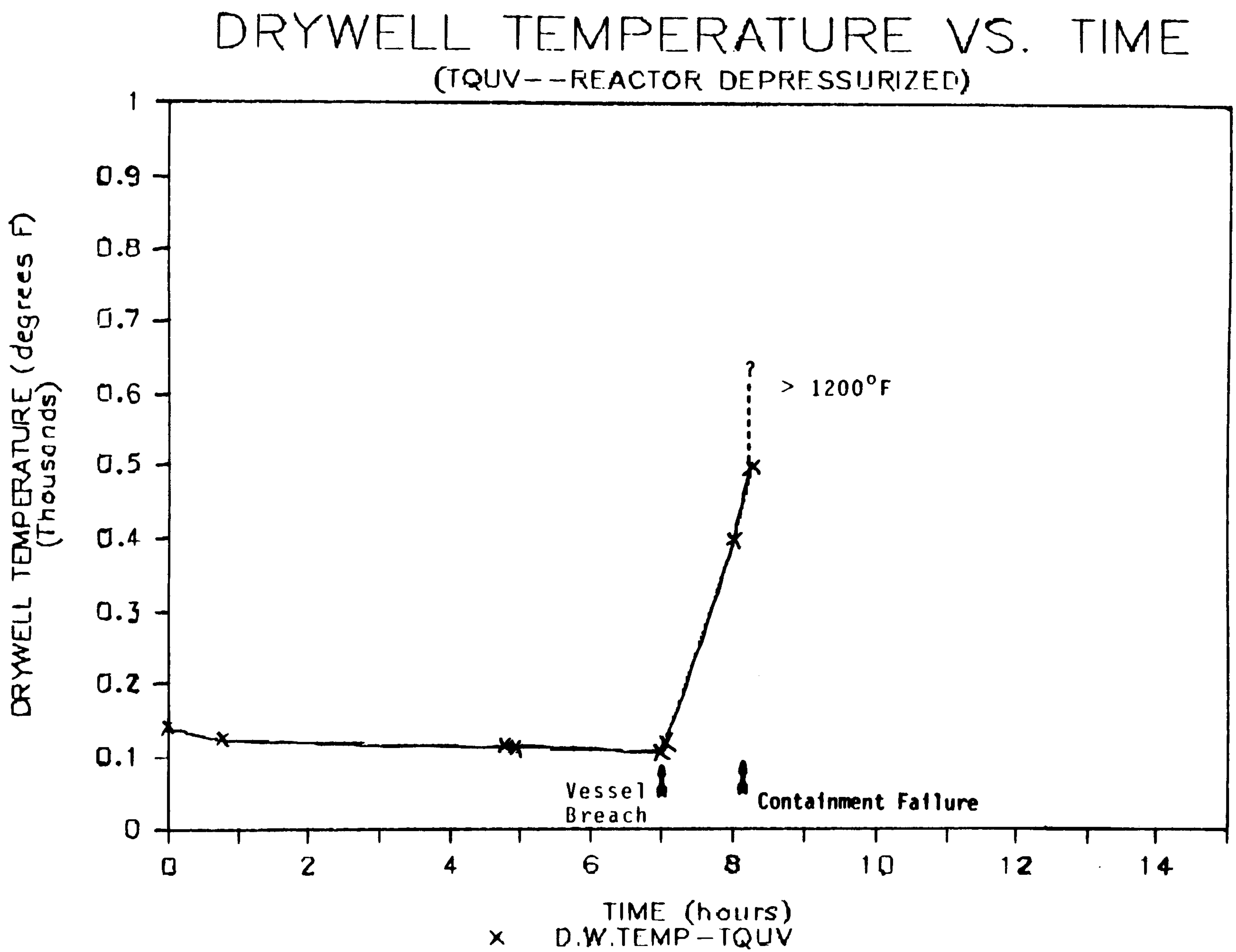


FIGURE C-42 - DRYWELL TEMPERATURE PROFILE FOR THE TENTH & ELEVENTH SCENARIO

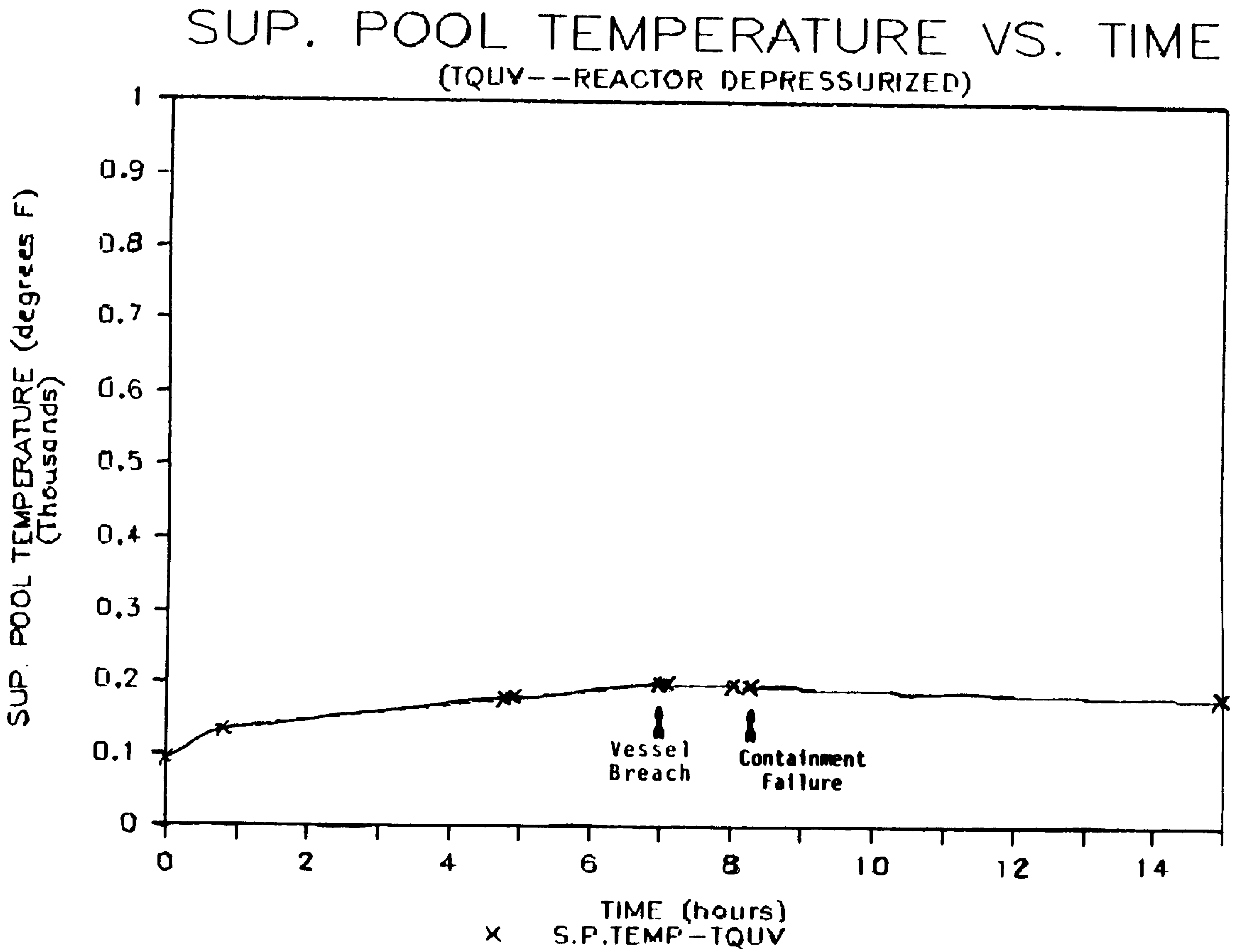


FIGURE C-43 - WETWELL TEMPERATURE PROFILE FOR THE TENTH & ELEVENTH SCENARIO

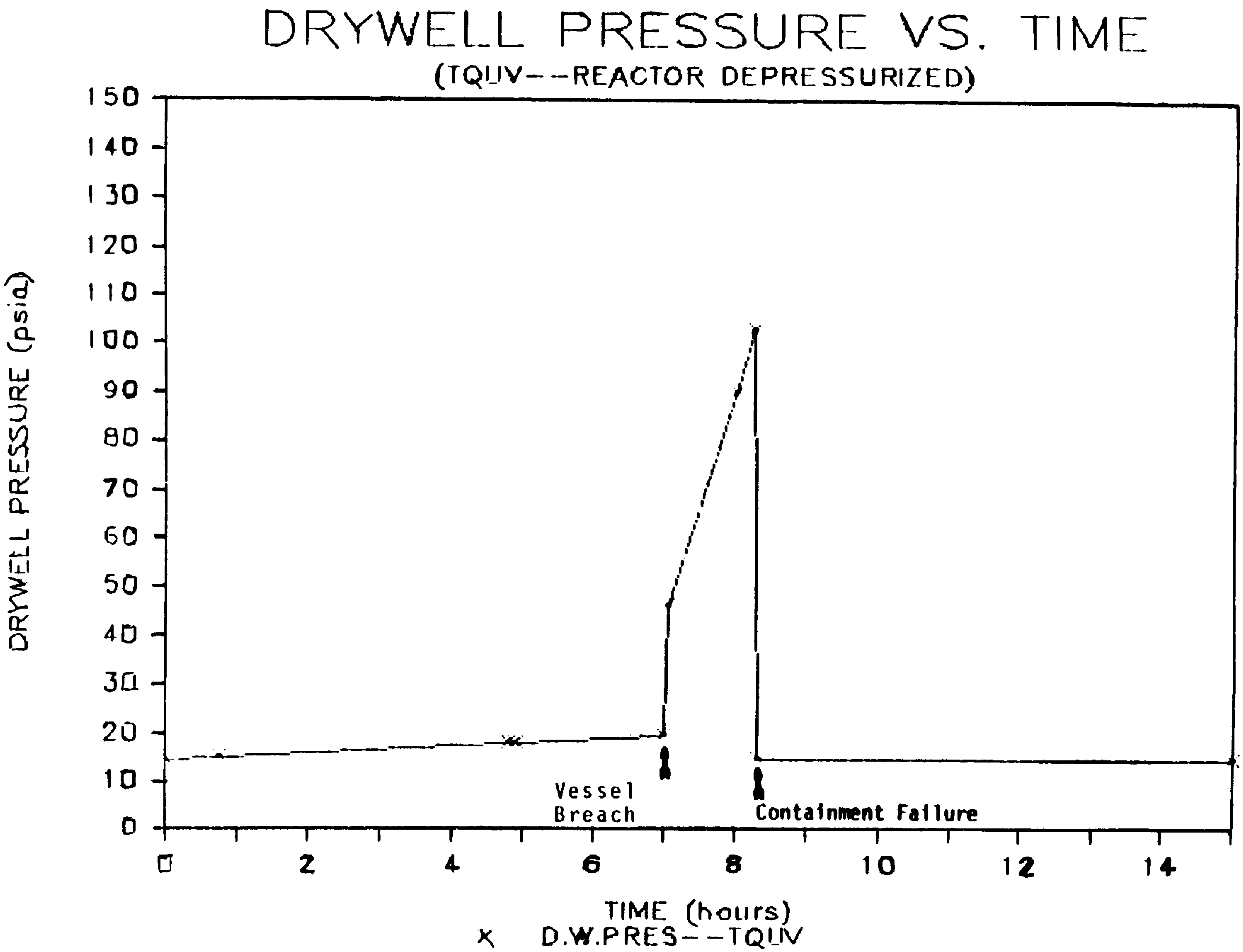


FIGURE C-44 - CONTAINMENT PRESSURE PROFILE FOR THE TENTH & ELEVENTH SCENARIO

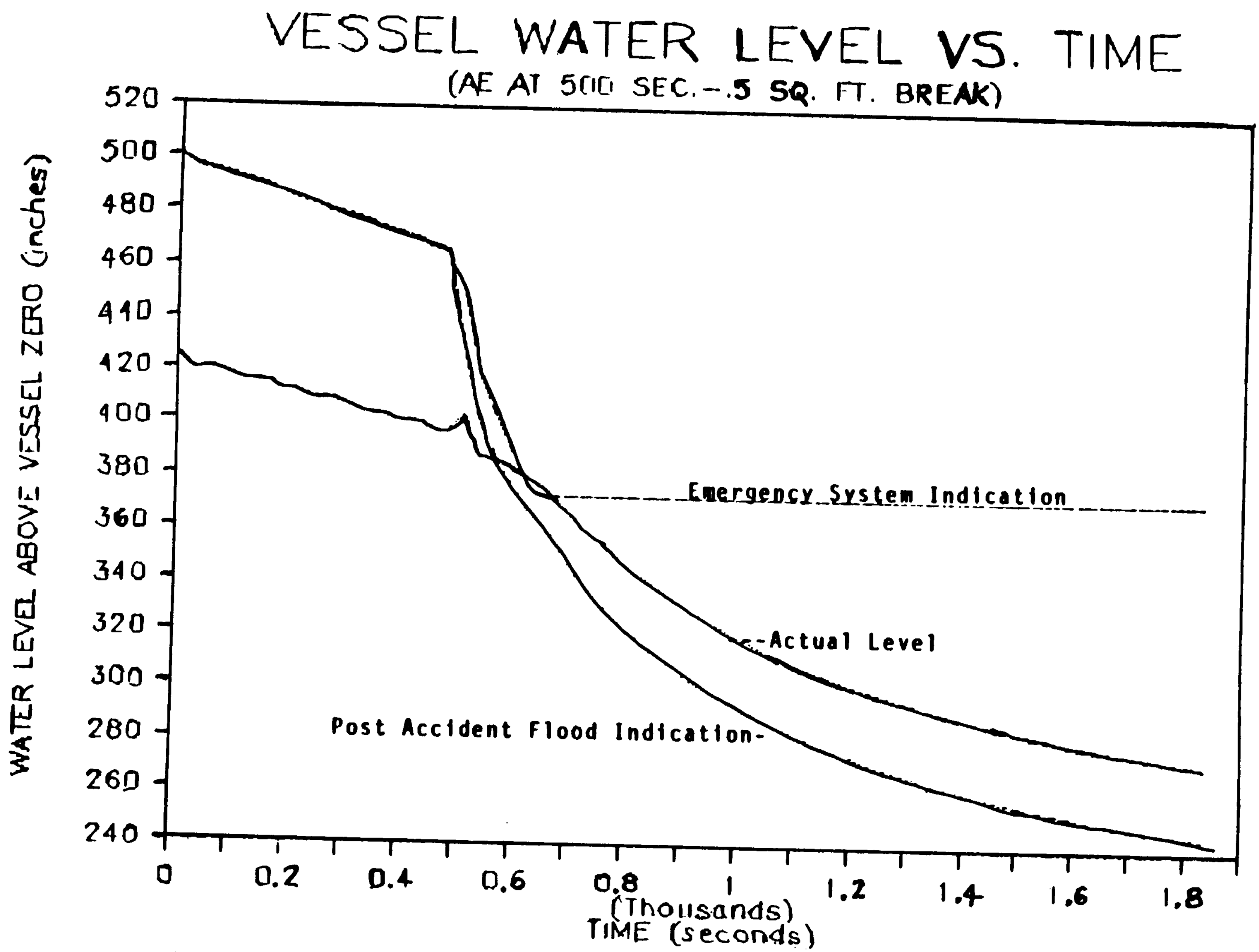


FIGURE C-45 - LTAS PREDICTED WATER LEVEL FOR THE THIRTEENTH SCENARIO

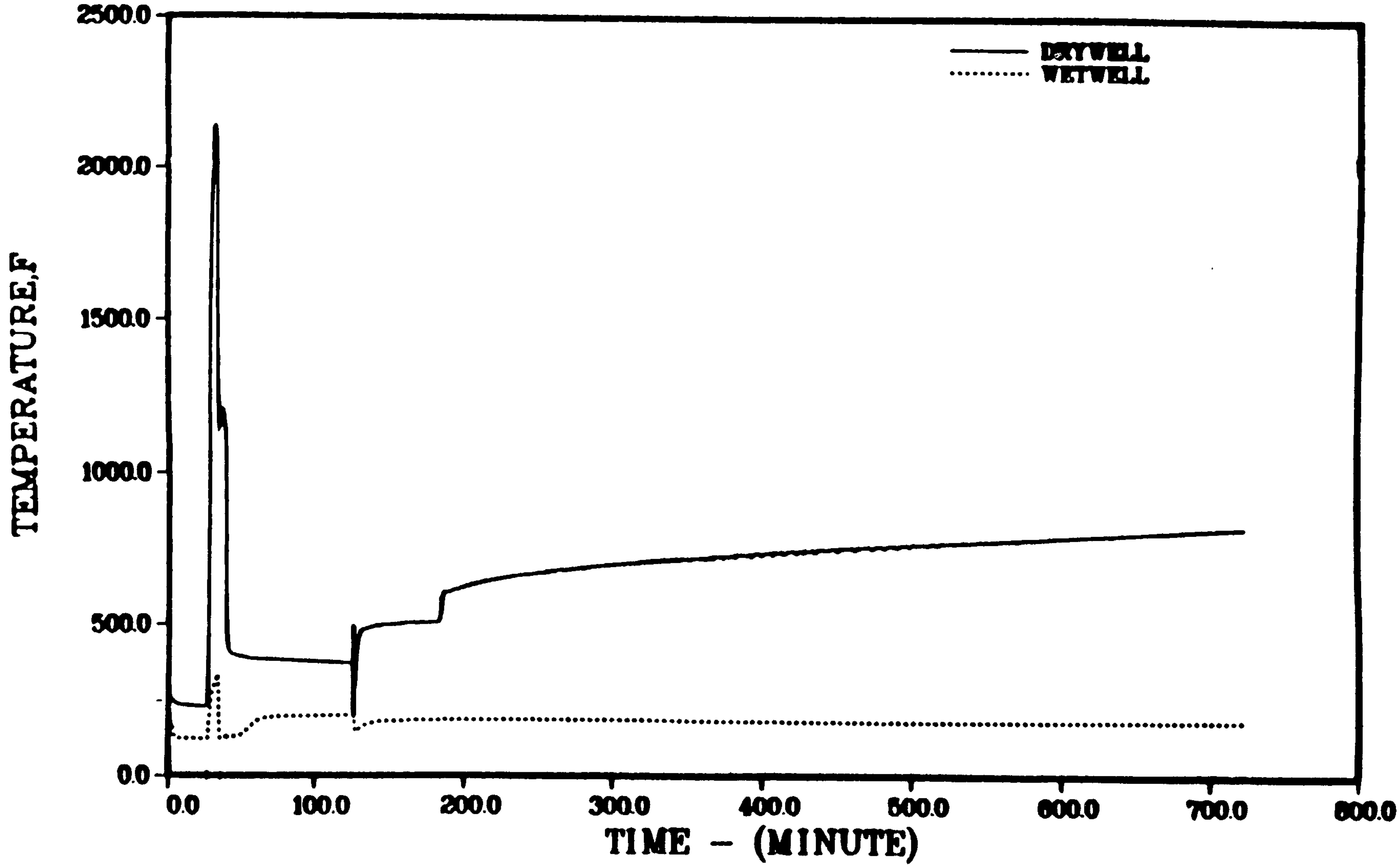


FIGURE C-46 - TEMPERATURES IN CONTAINMENT VOLUMES - SEQUENCE AE

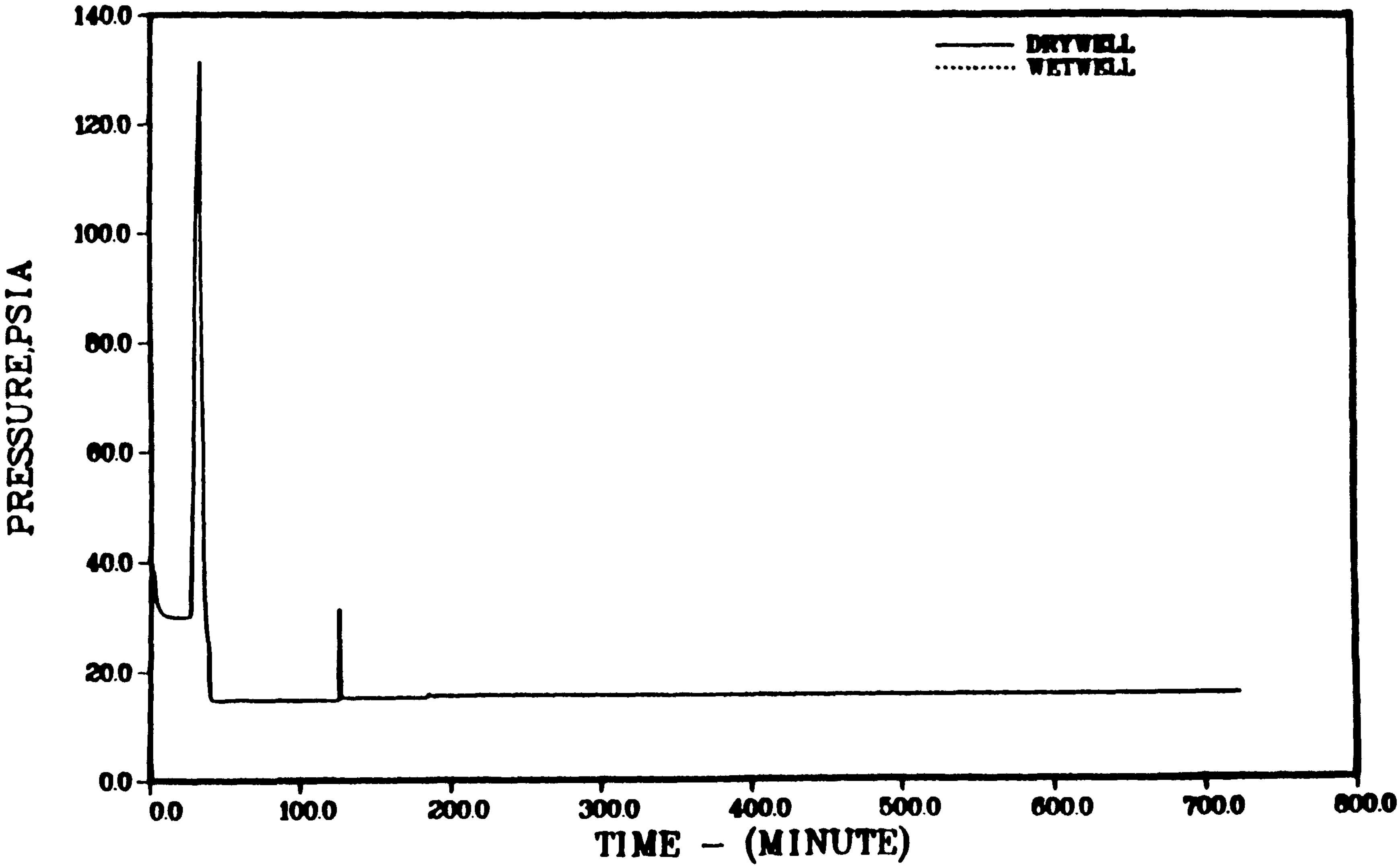


FIGURE C-47 - PRESSURES IN CONTAINMENT VOLUMES - SEQUENCE AE completed profiles are shown in Figures C-48, C-49, and C-50. A brief discussion of the profiles follows.

Drywell temperature and pressure are shown in Figures C-48 and C-50. Initially these parameters rise quickly in response to direct exposure of the drywell atmosphere to the superheated steam/water mixture from the reactor vessel. This continues until all the coolant has left the vessel resulting in the core becoming uncovered. The drywell temperature and pressure then level until decay heat causes core slump about 26 minutes into the accident. At this point the containment pressure and temperature experience a tremendous rise due to the production of hydrogen and the transport of the noncondensible gases into the wetwell due to the zircaloy reaction ($Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$) as the fuel rods melt. Containment failure occurs almost immediately with peak containment temperature predicted to be in excess of 2000^oF and pressure exceeding 130 psia. When containment fails, pressure falls to atmospheric and temperature starts to decrease. This continues until vessel breach when another smaller spike occurs. Decay heat from the molten core is sufficient to cause a slow temperature rise of the open drywell atmosphere which continues past the 15 hour point when the analysis ends.

Suppression pool temperature behavior is shown in Figure C-49. The wetwell also experiences an initial temperature rise due to the transfer of heated reactor coolant to its volume. Because of the large wetwell volume, the addition of this coolant does not have a major impact and the temperature rise remains within manageable levels. Once core slump occurs, the sudden release of noncondensible gases to the wetwell causes a momentary temperature spike, but this heat load is also quickly absorbed by the wetwell volume. The wetwell volume begins a phase of very slow heating due to continued release of heat energy from the reactor vessel. Once vessel failure occurs, wetwell temperature levels for the rest of the sequence.

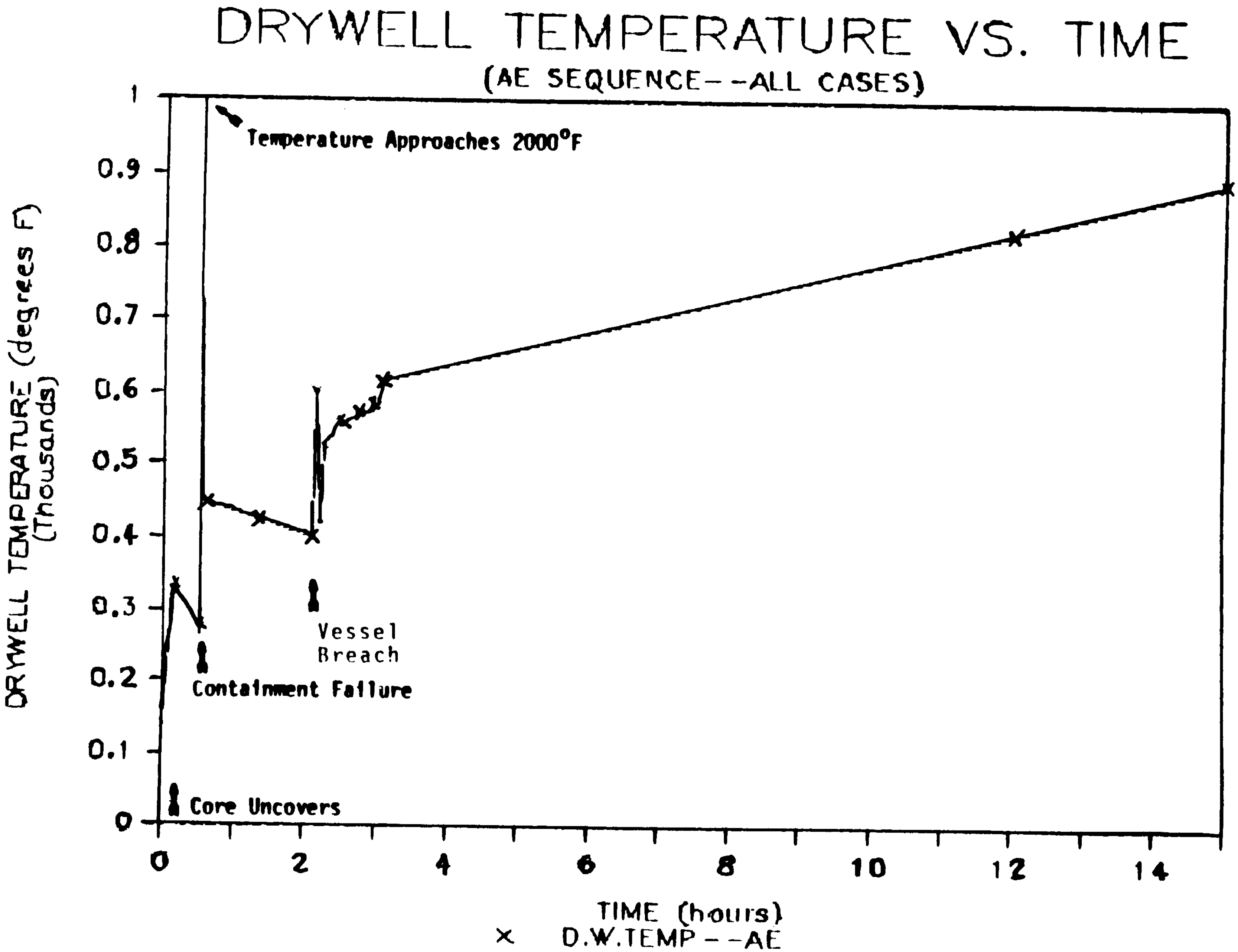


FIGURE C-48 - DRYWELL TEMPERATURE PROFILE FOR THE THIRTEENTH SCENARIO

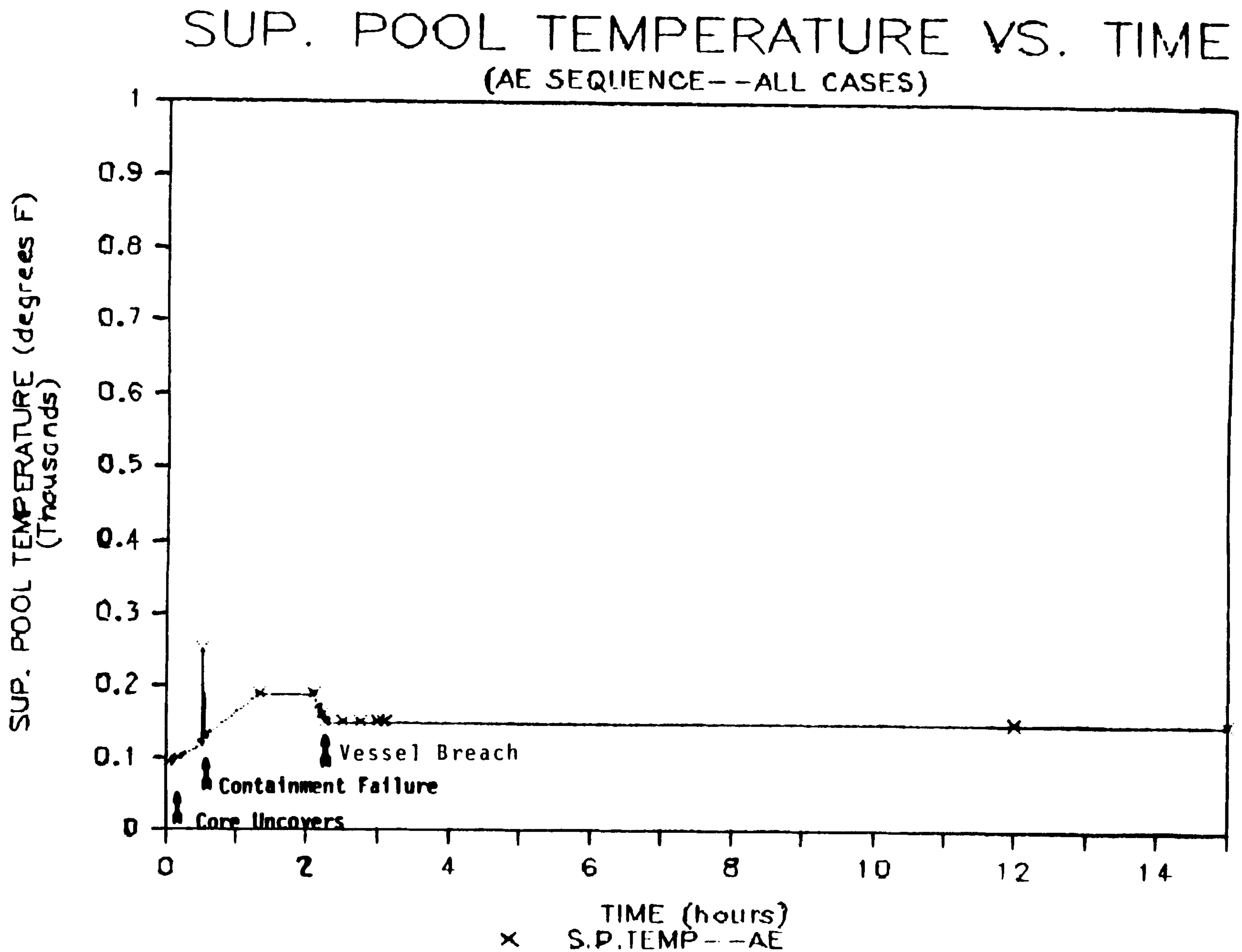


FIGURE C-49 - WETWELL TEMPERATURE PROFILE FOR THE THIRTEENTH SCENARIO

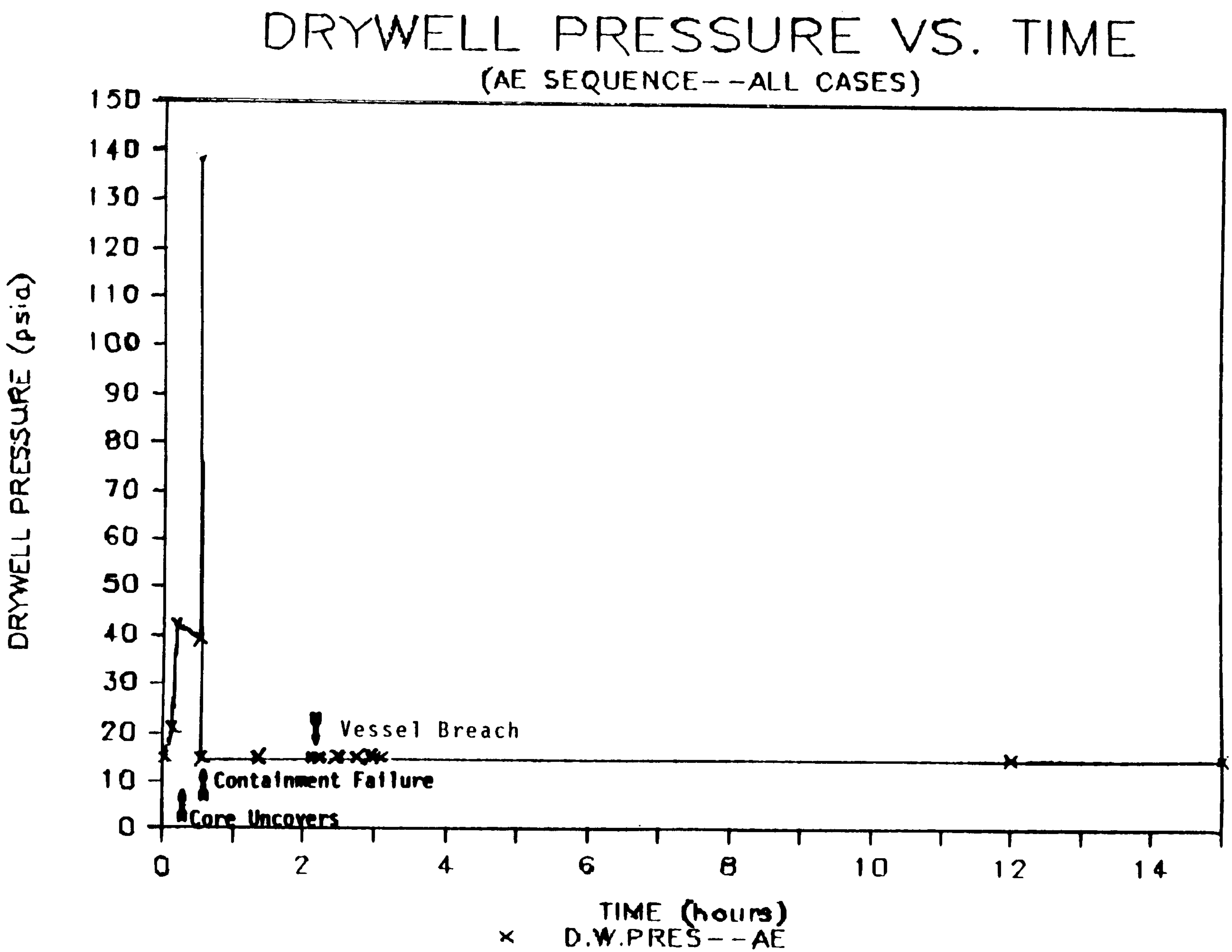


FIGURE C-50 - CONTAINMENT PRESSURE PROFILE FOR THE THIRTEENTH SCENARIO

2.6 Summary

This section has presented the construction methodology for the environmental profiles generated for examination by the PEEESAS program. Sources of information used in their construction were reviewed and results from past programs were compared to present data in an effort to validate the profiles. Table C-8 presents a synopsis of the profile generation results. As can be seen from the table, a total of 9 sets of profiles were constructed to describe the 14 scenarios used to represent the most probable path of the 5 accident sequences chosen for this study.

TABLE C-8 PROFILE GENERATION SUMMARY

CATEGORY	DESIGNATED PROFILE NUMBER	LTAS SCENARIO(S) USED	OTHER DATA SOURCES USED	PROFILE DESCRIPTION
TB	1	1 & 2 (0-1600 SEC)	REF 9 Pg 110, 145 (1600 SEC ON)	Total Blackout With No Injection. Late Or No Operator Action To Control Vessel Pressure.
	2	3 (0-900 SEC)	REF 9 Pg 116 (900 SEC ON)	Total Blackout With No Injection. Early Action To Lower Vessel Pressure.
	3	4 & 5 (0-333 MIN)	REF 9 Pg 100 (333 MIN ON)	Total Blackout With Injection Until Battery Failure. Vessel Depressurized.
TW	4	6 & 7 (0-15 HRS.)	REF 11 Pg 19, 25, 29 30, 96, 98 (15 HRS. ON)	Transient With No Wetwell Cooling. Vessel Depressurized.
TC	5	8 & 9 (0-4000 SEC)	REF 10 Pg 6-21 6-22 & 6-45--6-50 (4000 SEC ON)	Transient Without Scram. Vessel Depressurized.
	5A	8A (0-5 hours)	Results from TW Scenario	Transient without scram. MSIVs open.
TQUV	6	12 (0-33 MIN)	REF 3 Pg 34 REF 5 Pg 31 (33 MIN ON)	Transient With Loss Of High And Low Pressure Injection. CRD Flow Only. Vessel Remains At Pressure.
	7	10 & 11 (0-33 MIN)	REF 3 Pg 34 (33 MIN ON)	Transient With Loss Of High And Low Pressure Injection. CRD Flow Only. Vessel Depressurized.
AE	8	- - -	REF 10 Pgs 6-9, 6-10 6-45, 6-47, 6-49 (Entire Sequence)	Large LOCA. No Injection. No Operator Action.

3.0 QUALIFICATION PROFILES

3.1 Introduction

This section of the appendix presents the qualification profiles used for comparison purposes in this phase of the PEEESAS program. Although all test profiles for a given plant must envelope plant specific calculations, it is assumed that all equipment of interest has been qualified to the levels presented here. The PEEESAS program used a typical qualification profile based on IEEE 323-1974. In cases where equipment is not qualified to the typical qualification profile, the results of this study may have to be adjusted.

3.2 Profile Discussion

IEEE standard 323-1974 addresses the qualification of Class 1E equipment for nuclear power plants. The standard states that testing is the preferred qualification method. Equipment must be tested to the environmental profile based on the postulated design basis event (large LOCA). In addition, the test profile must add margin to the environmental profile to account for variations in manufacturing and uncertainty in defining satisfactory performance. To assure performance, the test profile includes margin; additional peak transient, increasing the temperature by 15^{o}F, increasing the pressure by 10% (gauge), and increasing the time (that equipment must operate following the design basis event) by 10%.

Although the actual test profile must be based on plant-specific calculations, a representative test profile is presented in IEEE Std 323-1974, Appendix A. Temperatures and pressures, as a function of time, were calculated for a typical LOCA in a PWR and a BWR. The larger value of temperature or pressure, at time, was used to develop the typical test profile. These resulting temperature and pressure profiles are shown in Figures C-51 and C-52. IEEE 323-1974, Appendix A also gives an accident dose of 150 Mrad and a demineralized water spray rate, for a BWR, of 0.15 (gal/min)/sq.ft.

IEEE 323-1974   APPENDIX A

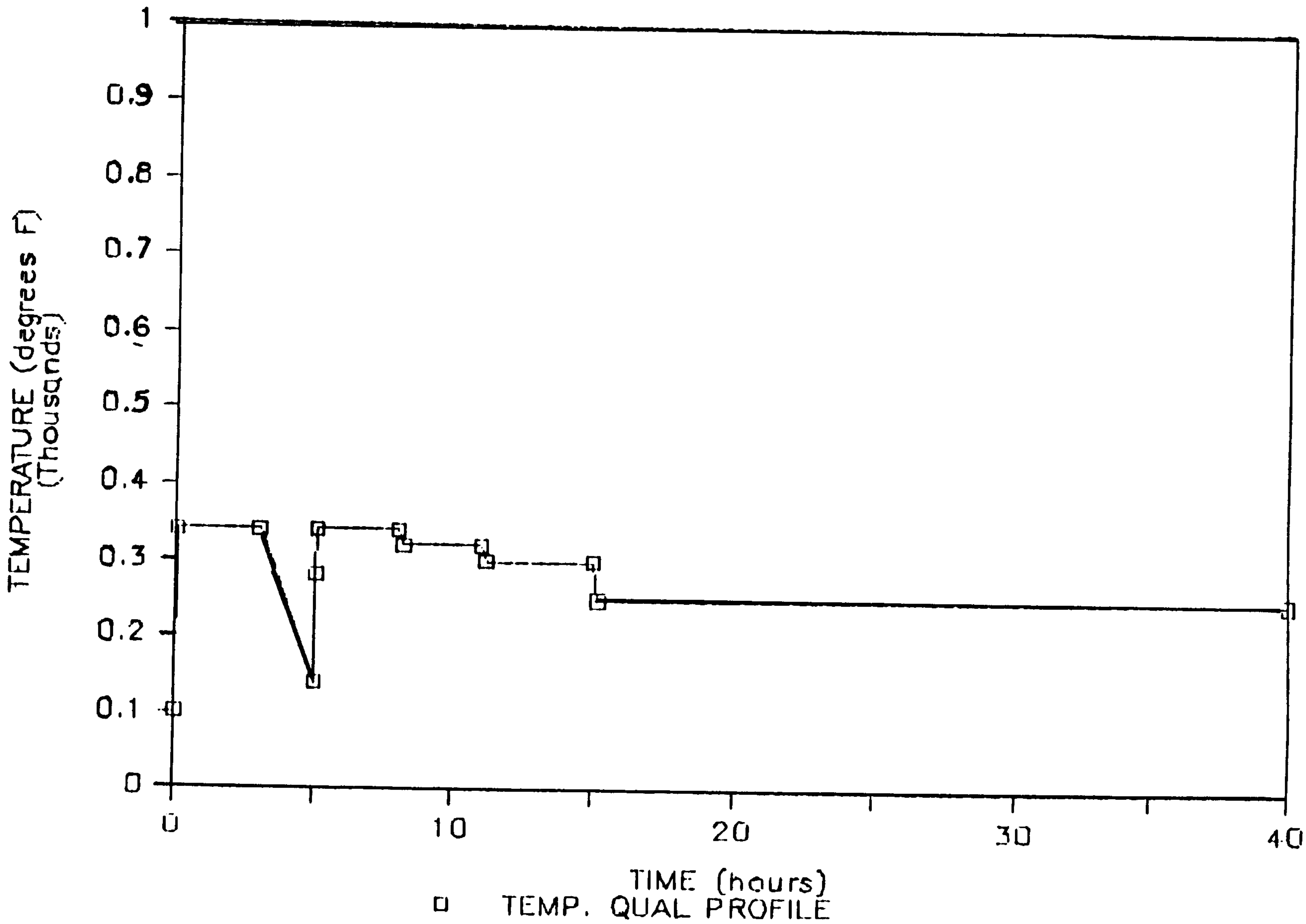


FIGURE C-51 - TYPICAL QUALIFICATION PROFILE FOR TEMPERATURE

IEEE 323-1974 APPENDIX A

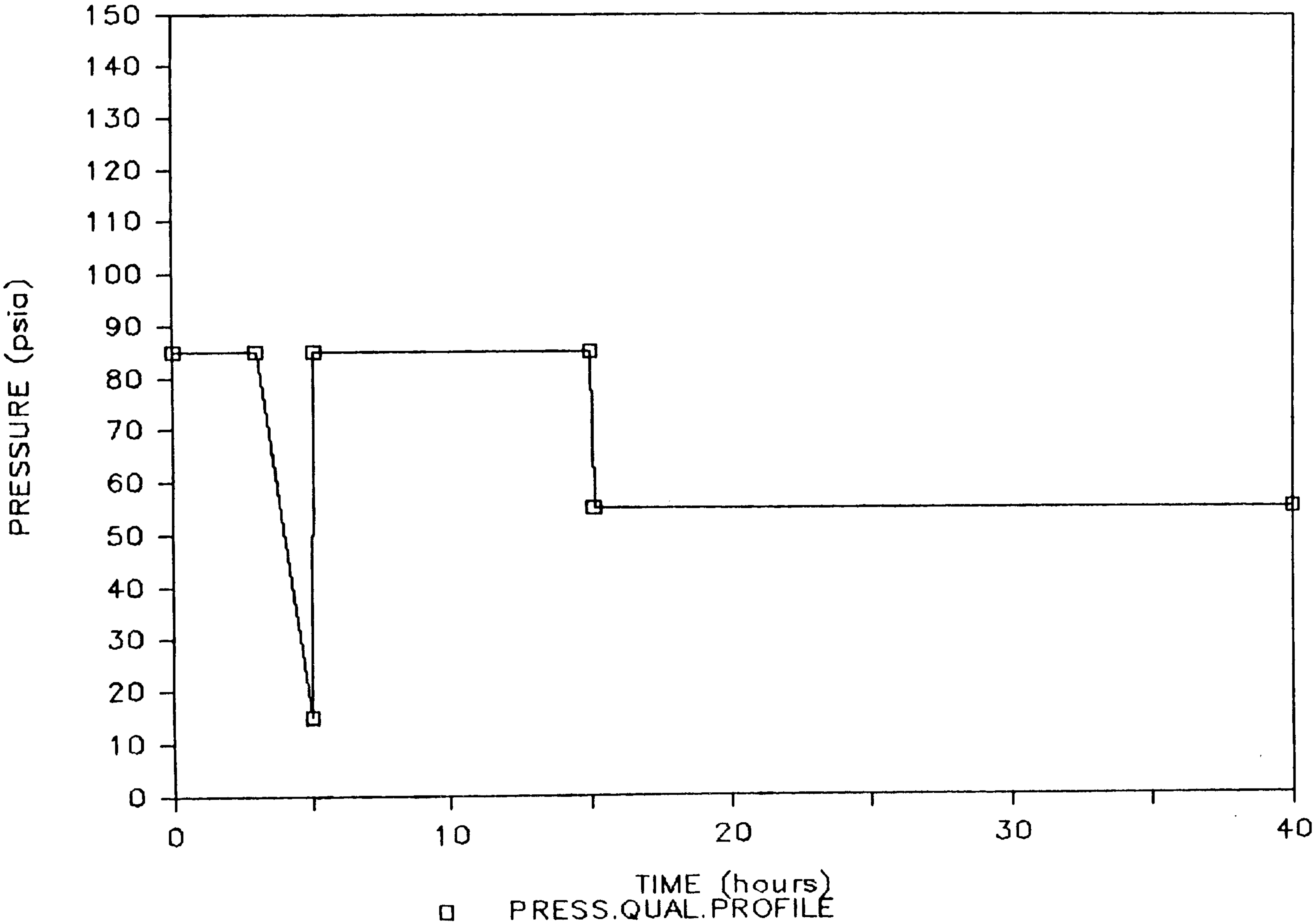


FIGURE C-52 - TYPICAL QUALIFICATION PROFILE FOR PRESSURE

3.3 Summary

This section has presented the reasons for using the IEEE Std 323-1974 qualification criteria for comparison of accident profiles and qualification profiles. It was assumed for the purposes of this study that all the equipment identified for further test and analysis has been qualified to these criteria. Specific issues, including the limitations in using this profile, margin, temperature and pressure response, total radiation dose, and demineralized water spray rate were discussed.

4.0 RESULTS

4.1 Introduction

This section of the appendix compares the nine environmental profiles sets (section 2.0) to the qualification profiles described in the last section. Points where the environmental profiles exceed the qualification profiles are some areas for closer examination and possible testing. The appropriate qualification profile was overlayed with each environmental profile. This overlay allows easy identification of those areas in excess of the qualification limit. Data from each of these overlay comparisons was then tabulated in summary form permitting quick review of the results.

4.2 Profile Comparisons

4.2.1 TB (short term)

The short term blackout sequence had two sets of profiles. The first set was for the case with late or no operator action. Figures C-53, C-54 and C-55 present the comparisons for this case. Note that drywell temperature exceeds the maximum qualification temperature 3 hours into the accident. Between vessel breach and containment failure, the drywell temperature spends approximately 15 minutes above the maximum temperature of the qualification profile. During this time temperature approaches 500°F. Figure C-54 indicates that suppression pool temperature never exceeds the maximum qualification temperature and thus is not an area for concern. Figure C-55 shows that drywell pressure exceeds the maximum qualification pressure at about 3.0 hours into the accident. Between vessel breach and containment failure the drywell atmosphere pressure spends about 10 minutes above the qualification profile level and reaches a peak level of approximately 100 psia. This is approximately 1.25 times the qualification level of 85 psia.

The second set of profiles constructed for the short term blackout sequence involved the case where the vessel is depressurized. Figures C-56, C-57,

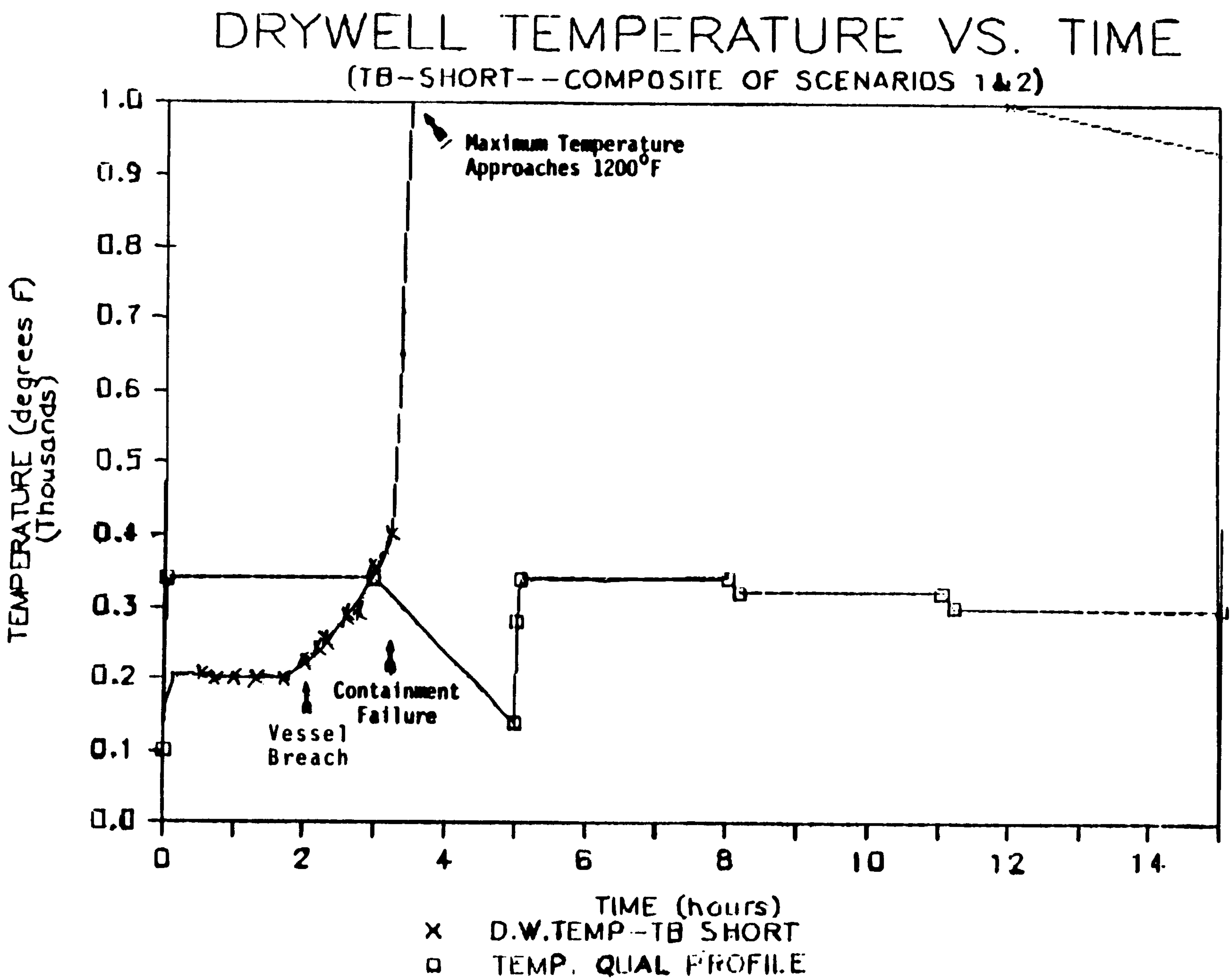


FIGURE C-53- ENVIRONMENTAL PROFILE NO. 1 COMPARISON TO QUALIFICATION LEVELS FOR DRYWELL TEMPERATURE

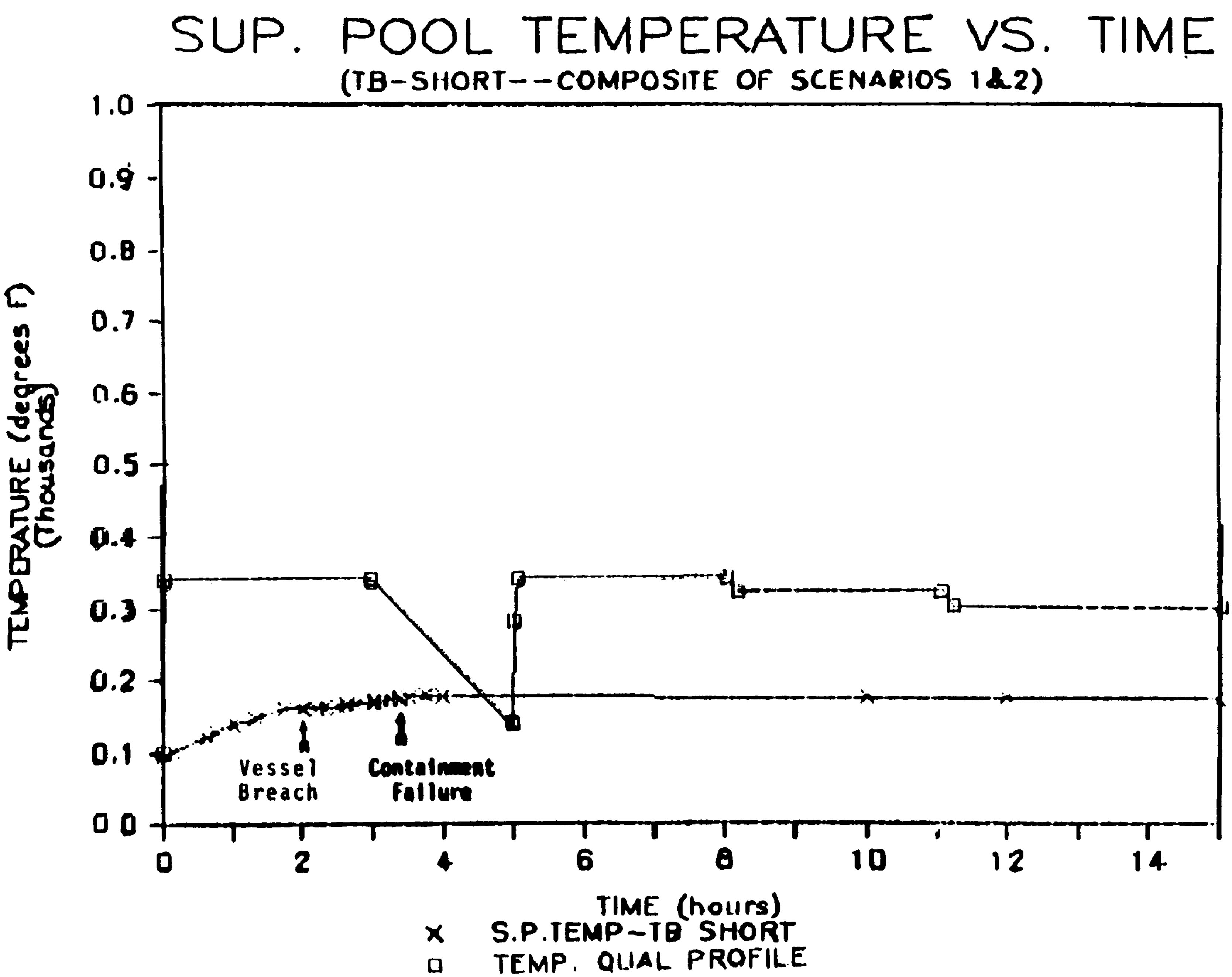


FIGURE C-54 - ENVIRONMENTAL PROFILE NO.1 COMPARISON TO QUALIFICATION LEVELS FOR WETWELL TEMPERATURE

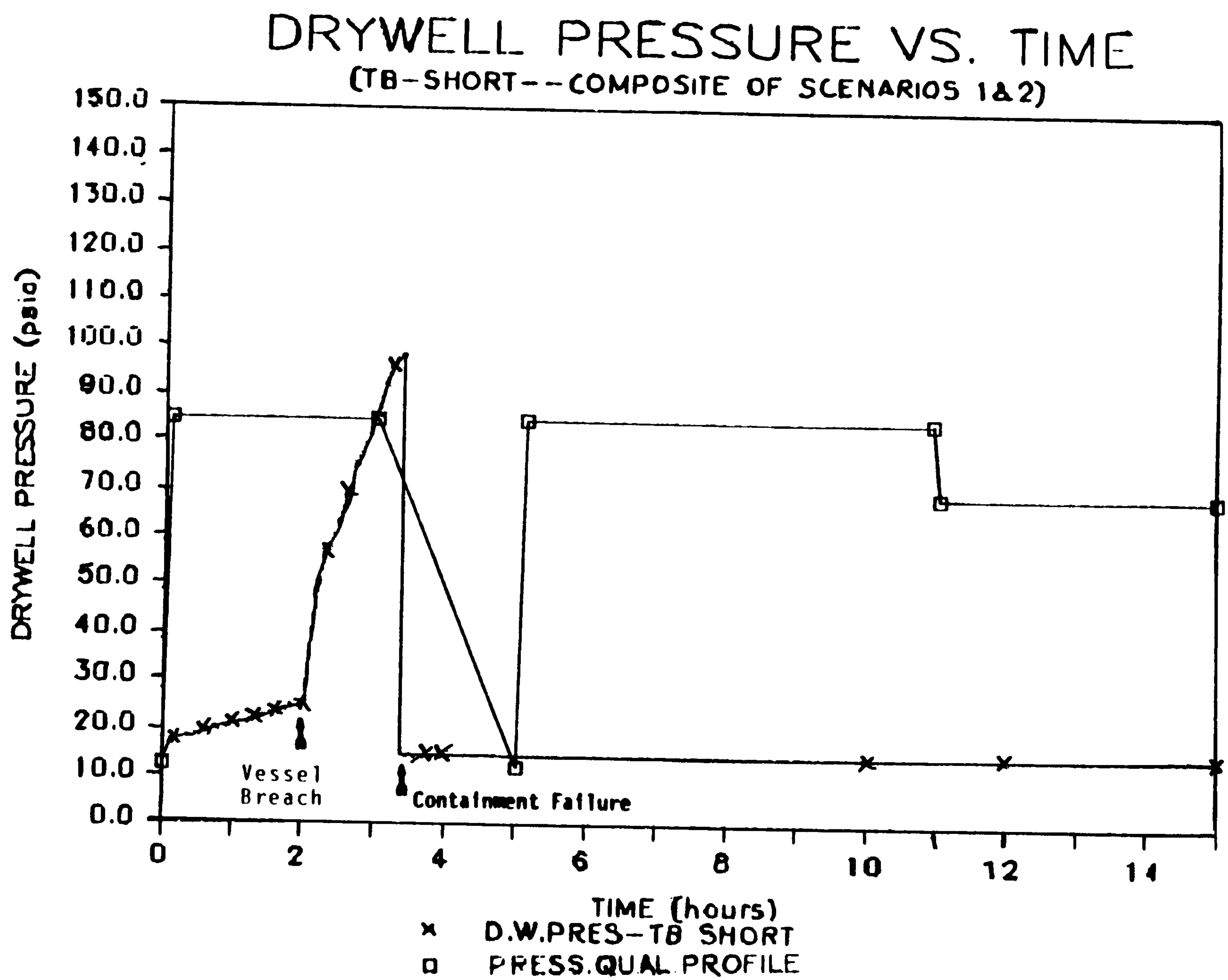


FIGURE C-55 - ENVIRONMENTAL PROFILE NO. 1 COMPARISON TO QUALIFICATION LEVELS FOR CONTAINMENT PRESSURE

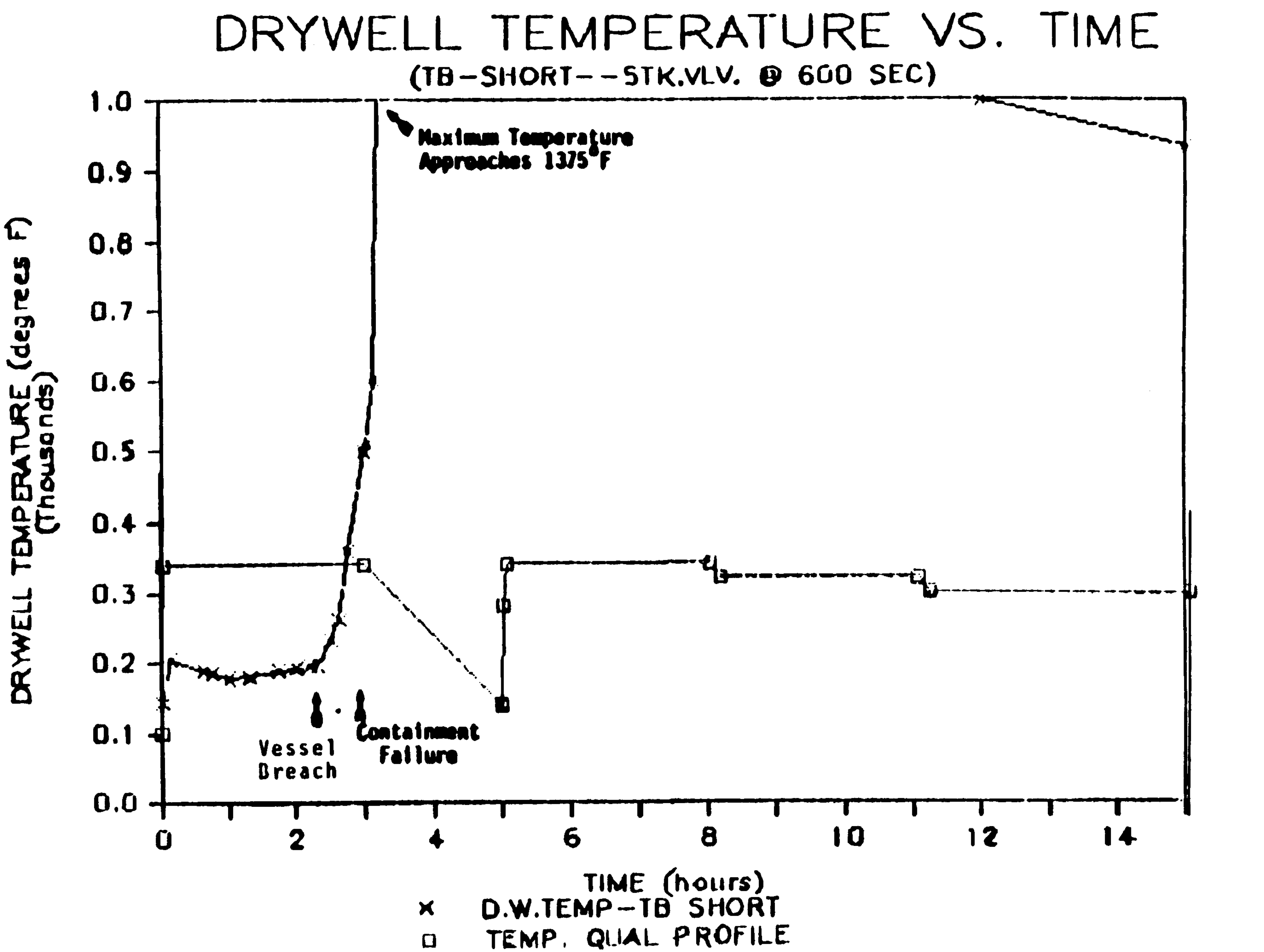


FIGURE C-56- ENVIRONMENTAL PROFILE NO. 2 COMPARISON TO QUALIFICATION LEVELS FOR DRYWELL TEMPERATURE

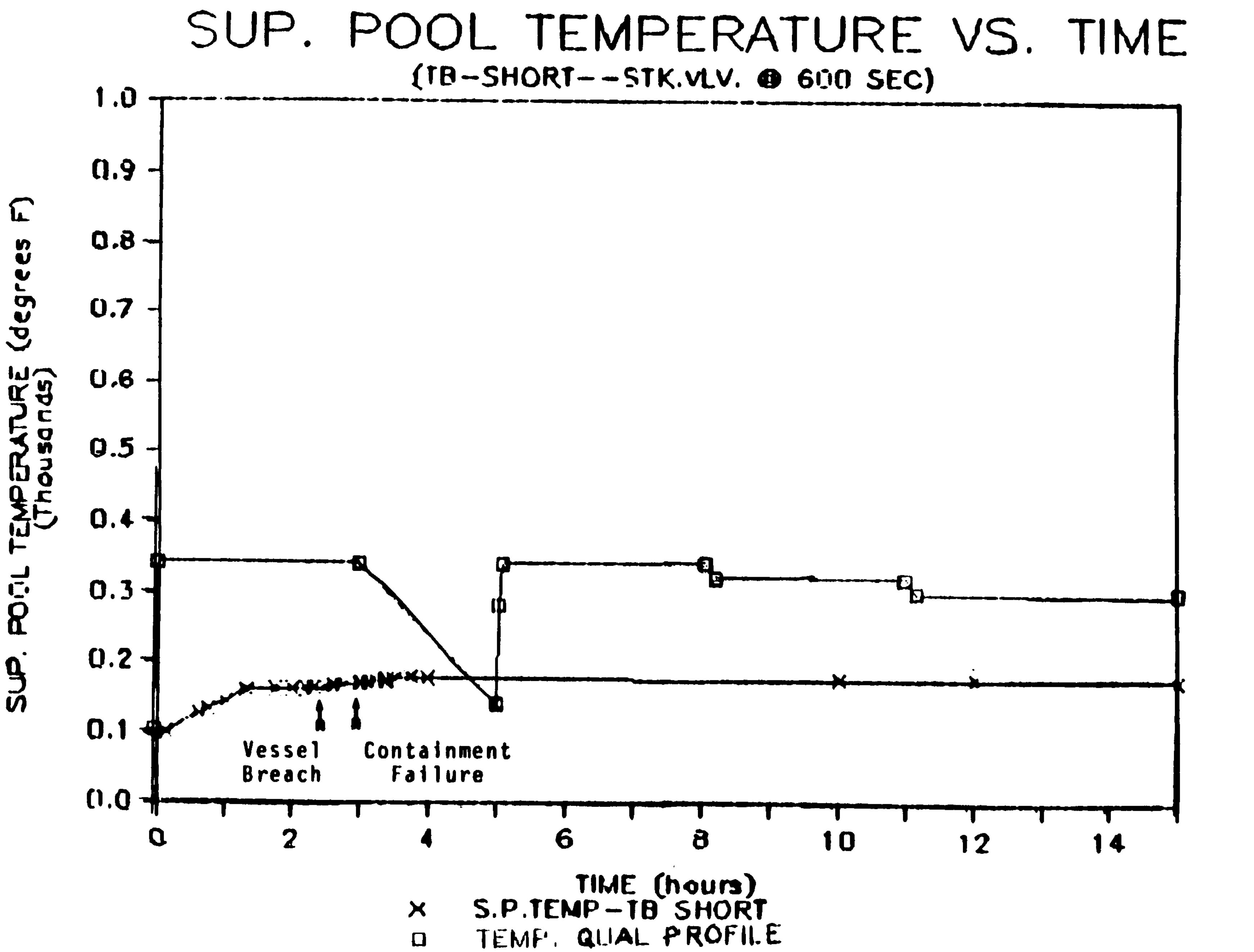


Figure C-57 - Environmental Profile No. 2 Comparison To Qualification Levels For Wetwell Temperature

and C-58 present the comparisons for this case. As shown in Figure C-56, drywell temperature exceeds the maximum qualification temperature about 2.7 hours into the accident. From time of vessel breach to the time of containment failure at 500 degrees, drywell temperature spends about one-quarter hour above the maximum qualification temperature. Again it can be seen from Figure C-57, that wetwell temperature never even approaches the maximum qualification temperature. Figure C-58 shows that containment pressure never exceeds maximum qualification pressure. This indicates that the temperature environment is the driving force in causing containment failure about 3 hours into the accident (due to electrical penetration failure at about 500°F).

4.2.2 TB (long term)

Profile set number 3 represents environmental conditions for the long term TB sequence. Figures C-59, C-60, and C-61 show the environmental profiles overlayed with the qualification profiles. Figure C-59 indicates that drywell temperature exceeds the maximum qualification temperature 8.0 hours into the accident. This precedes both vessel breach and containment failure. The drywell temperature remains above qualification limits over the next 2.0 hours until containment failure. Suppression pool temperature, as with the short term sequence, never approaches the maximum temperature value as shown in Figure C-60. In Figure C-61, containment pressure is shown to approach about 90-100 psia. This is an estimate based on behavior seen in the short term sequence. No actual value was found for the peak pressure value, but it is assumed that containment failure was caused by temperature as was seen in the short term blackout sequence. This means that containment pressure would peak below 130 psia (which is the estimated containment failure point due to pressure). Using a 90-100 psia peak estimate, it is seen that pressure exceeds the maximum qualification pressure about 9.5 hours into the accident and remains in excess of this level until containment failure at 10 hours into the accident. A peak level of 90-100 psia is about 1.2 times the maximum qualification pressure of 85 psia.

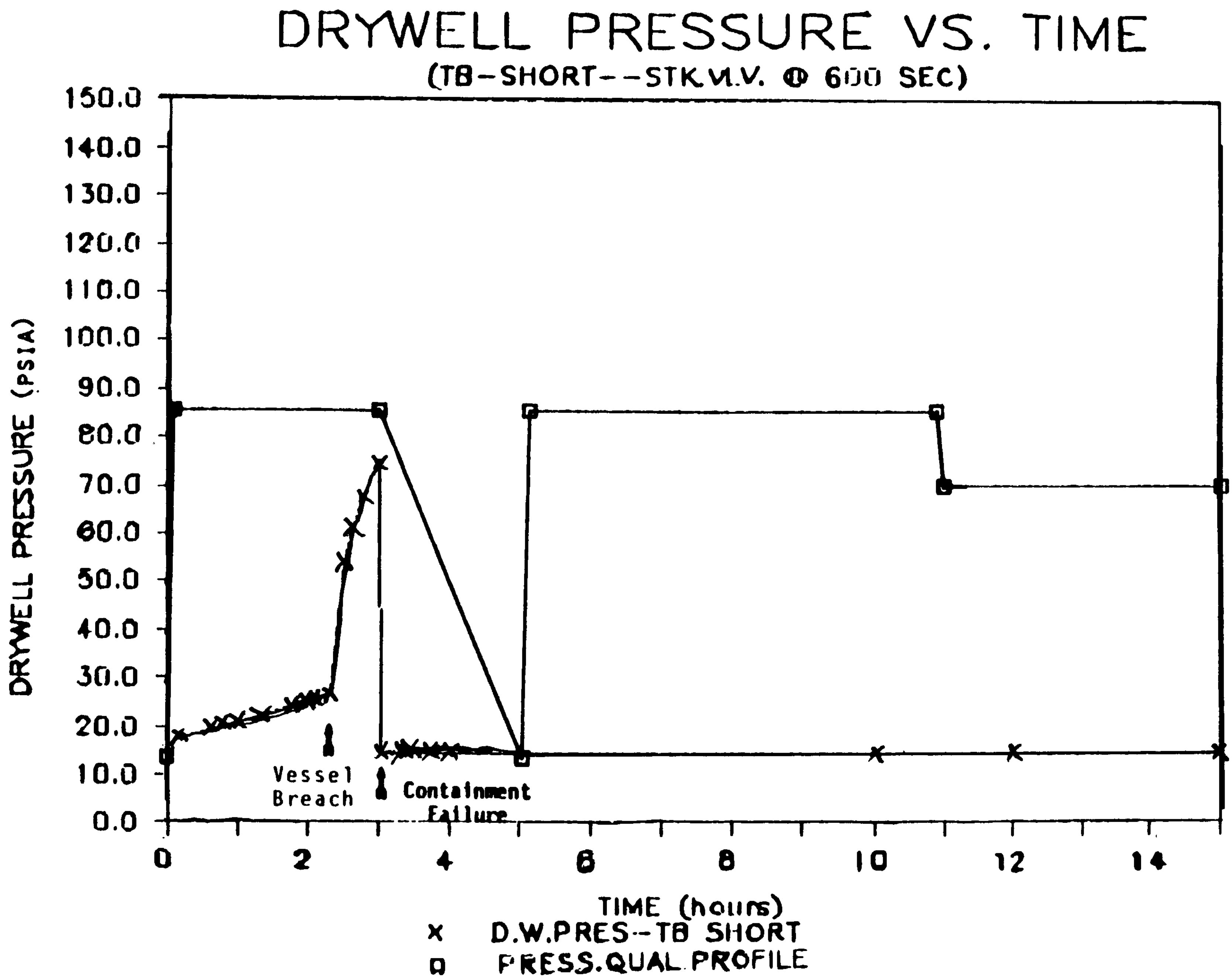


FIGURE C-58 - ENVIRONMENTAL PROFILE NO. 2 COMPARISON TO QUALIFICATION LEVELS FOR CONTAINMENT PRESSURE

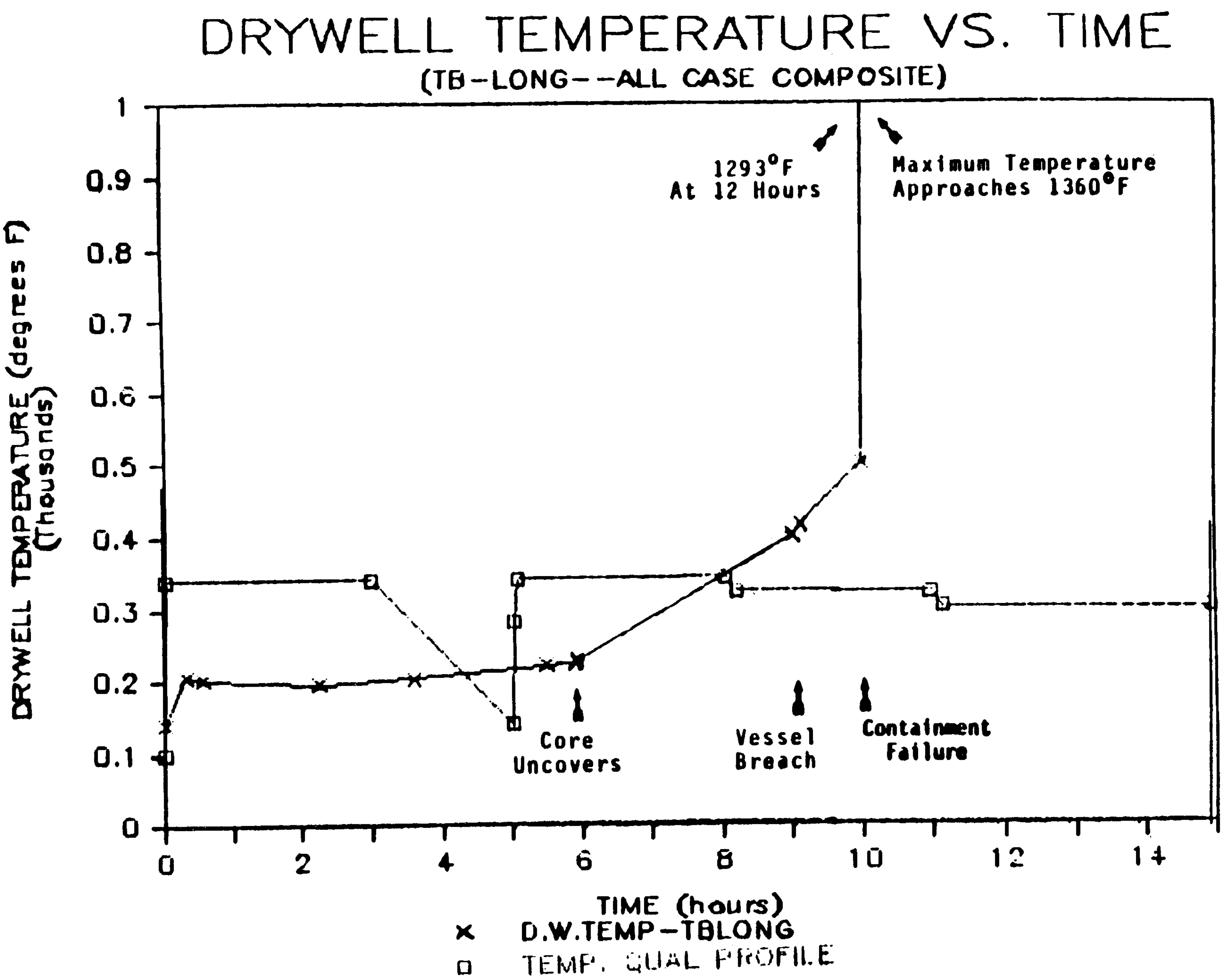


Figure C-59 - Environmental Profile No. 3 Comparison To Qualification Levels For Drywell Temperature

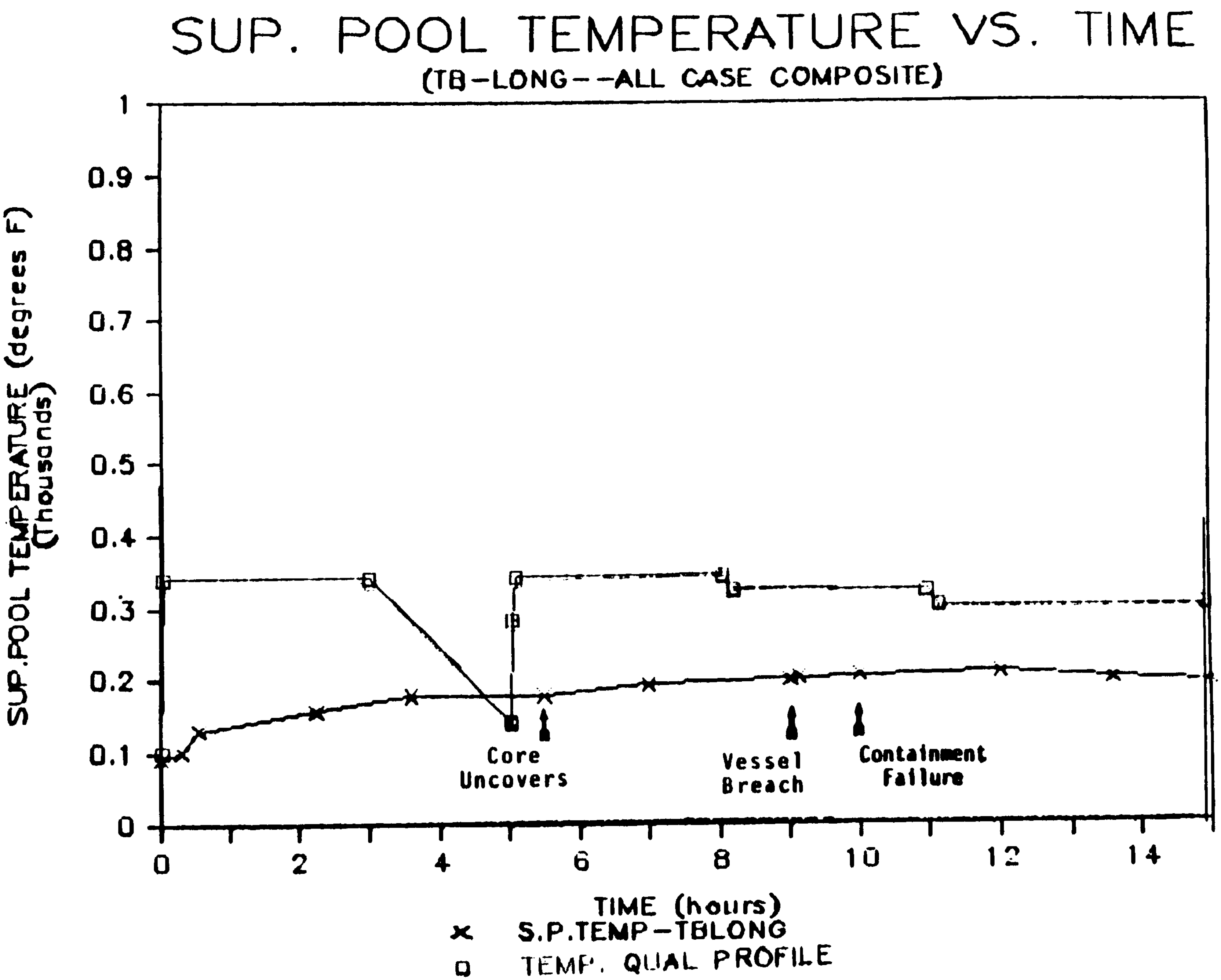


FIGURE C-60 - ENVIRONMENTAL PROFILE No. 3 COMPARISON TO QUALIFICATION LEVELS FOR WETWELL TEMPERATURE

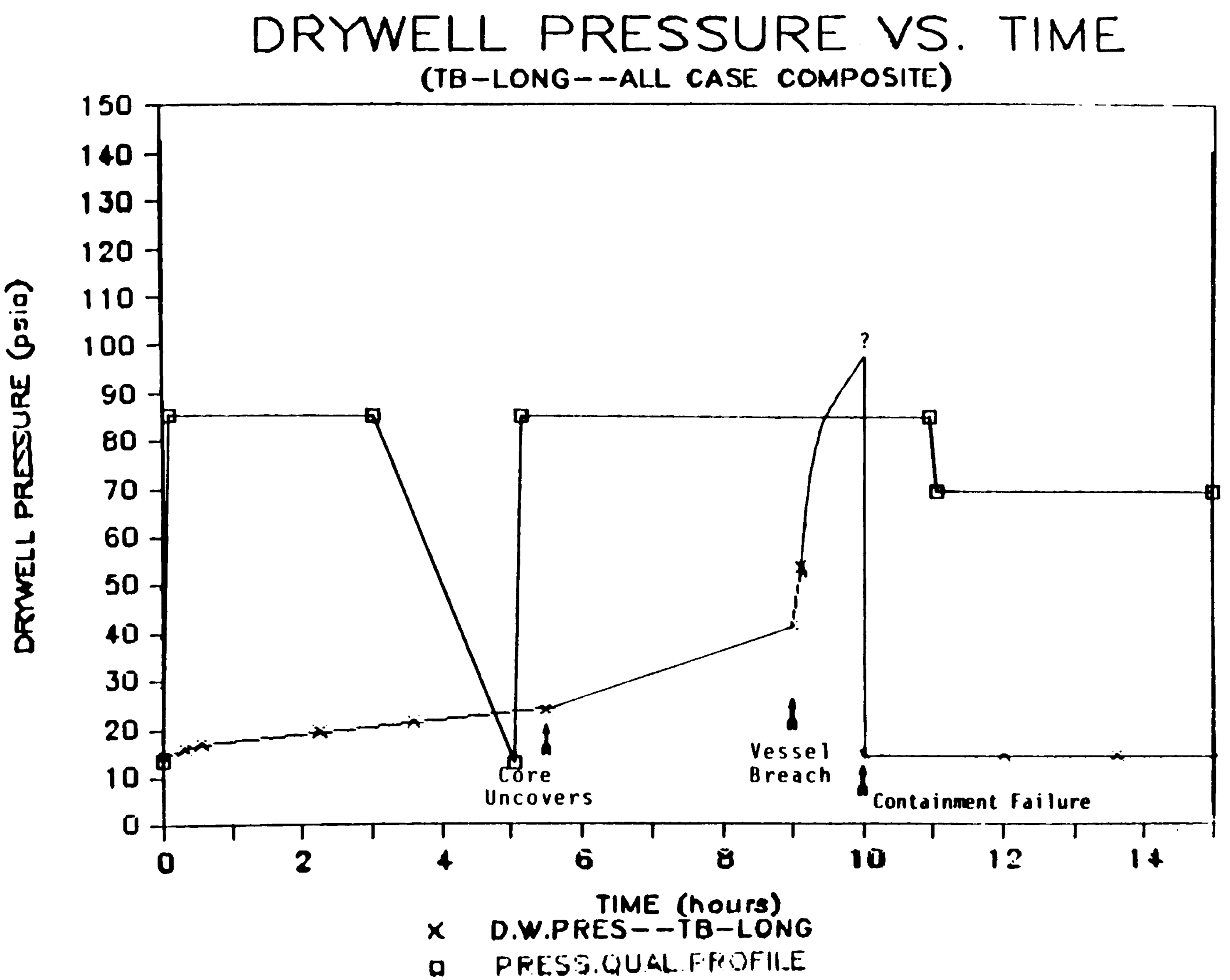


FIGURE C-61 - ENVIRONMENTAL PROFILE NO. 3 COMPARISON TO QUALIFICATION LEVELS FOR CONTAINMENT PRESSURE

4.2.3 TW

The TW sequence was represented with environmental profile set number 4. The results of overlaying the environmental profile curves with the qualification profiles are shown in Figures C-62, C-63, and C-64. Figure C-62 shows the behavior of the drywell temperature environment in relation to the qualification profile. Note that drywell temperature doesn't exceed the qualification limit until about 28 hours into the accident. From this point temperature continues to climb to about 425°F when containment failure occurs due to high pressure at 35 hours. Past the 35 hour point, drywell temperature rises steeply up to the time of core slump and vessel breach at about 36 hours and 39 hours respectively. Figure C-63 shows that suppression pool temperature exceeds the maximum qualification temperature at about 34 hours into the accident and remains in a slow climb for the remainder of the accident up to the point of containment failure. Figure C-64 shows how containment pressure behaves throughout the accident. Pressure exceeds the maximum qualification pressure about 28 hours into the sequence and continues an almost linear climb to 120-130 psia where containment fails 35 hours into the accident. After containment failure pressure drops to atmospheric and remains there for the rest of the sequence.

4.2.4 TC

The TC sequence was represented with profile set number 5 and 5A. Figures C-65, C-66, and C-67 show the results of comparing the set number 5 environmental profiles with the qualification profiles. In Figure C-65, it can be seen that drywell temperature approaches and just exceeds the maximum qualification temperature briefly at the point of containment failure (occurring about 1 hour into the accident). The temperature then dips down below the qualification profile and remains there until the point of vessel breach which occurs 3.8 hours into the accident. These results indicate that the drywell temperature does not significantly exceed the qualification level until after both containment failure and vessel breach. Figure C-66 shows that wetwell temperature approaches but never exceeds the maximum

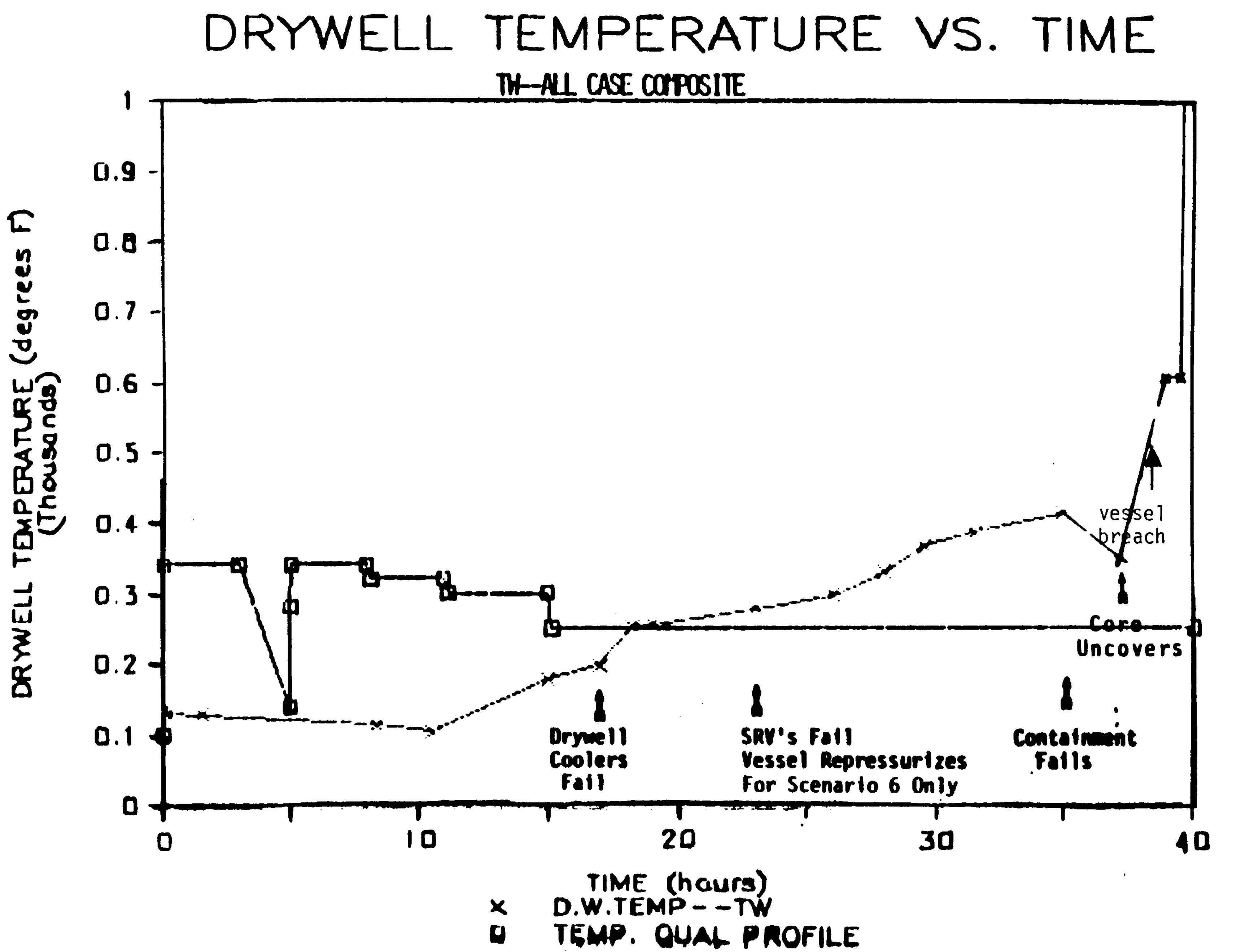


FIGURE C-62 - ENVIRONMENTAL PROFILE NO. 4 COMPARISON TO QUALIFICATION LEVELS FOR DRYWELL TEMPERATURE

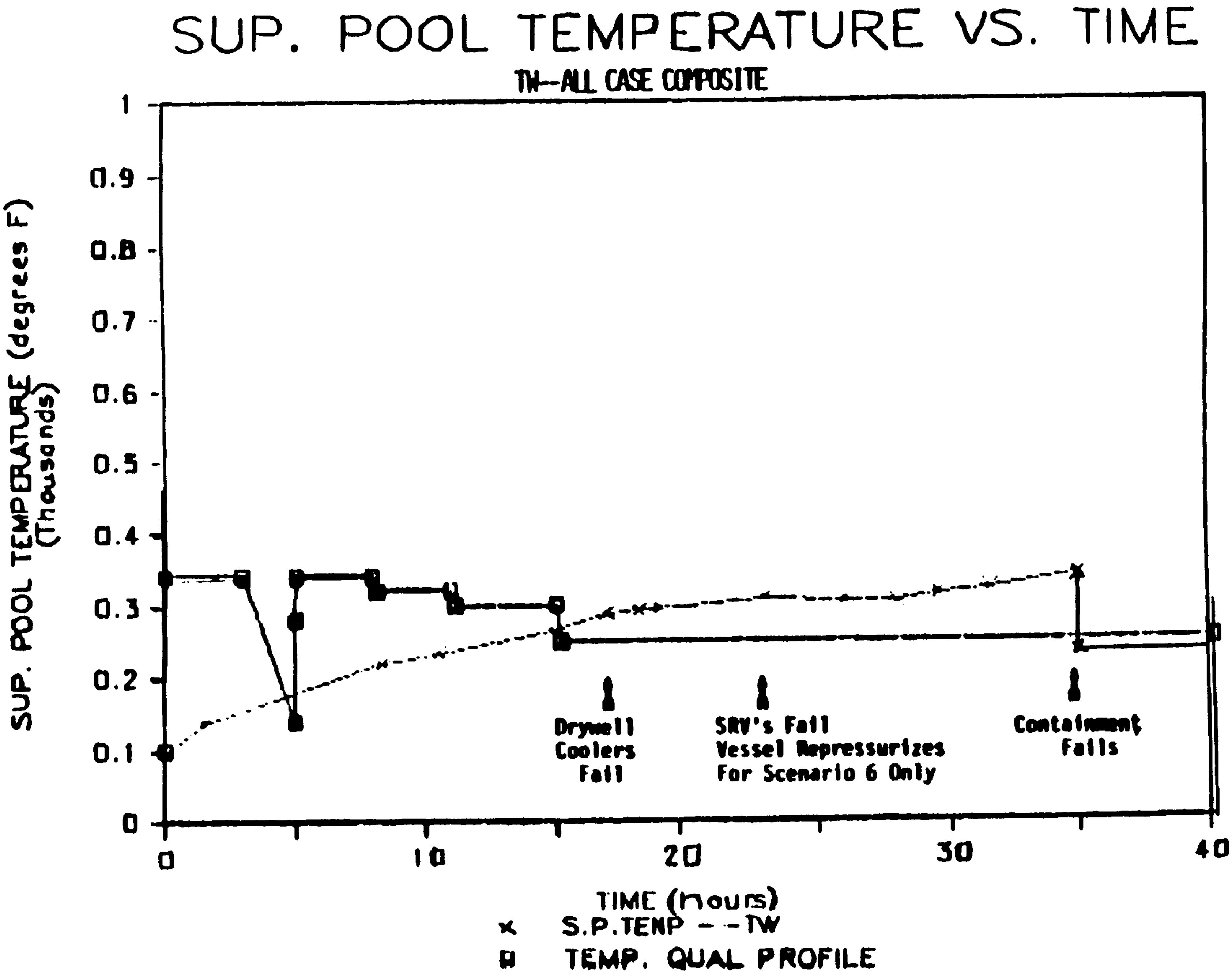


FIGURE C-63- ENVIRONMENTAL PROFILE NO. 4 COMPARISON TO QUALIFICATION LEVELS FOR WETWELL TEMPERATURE

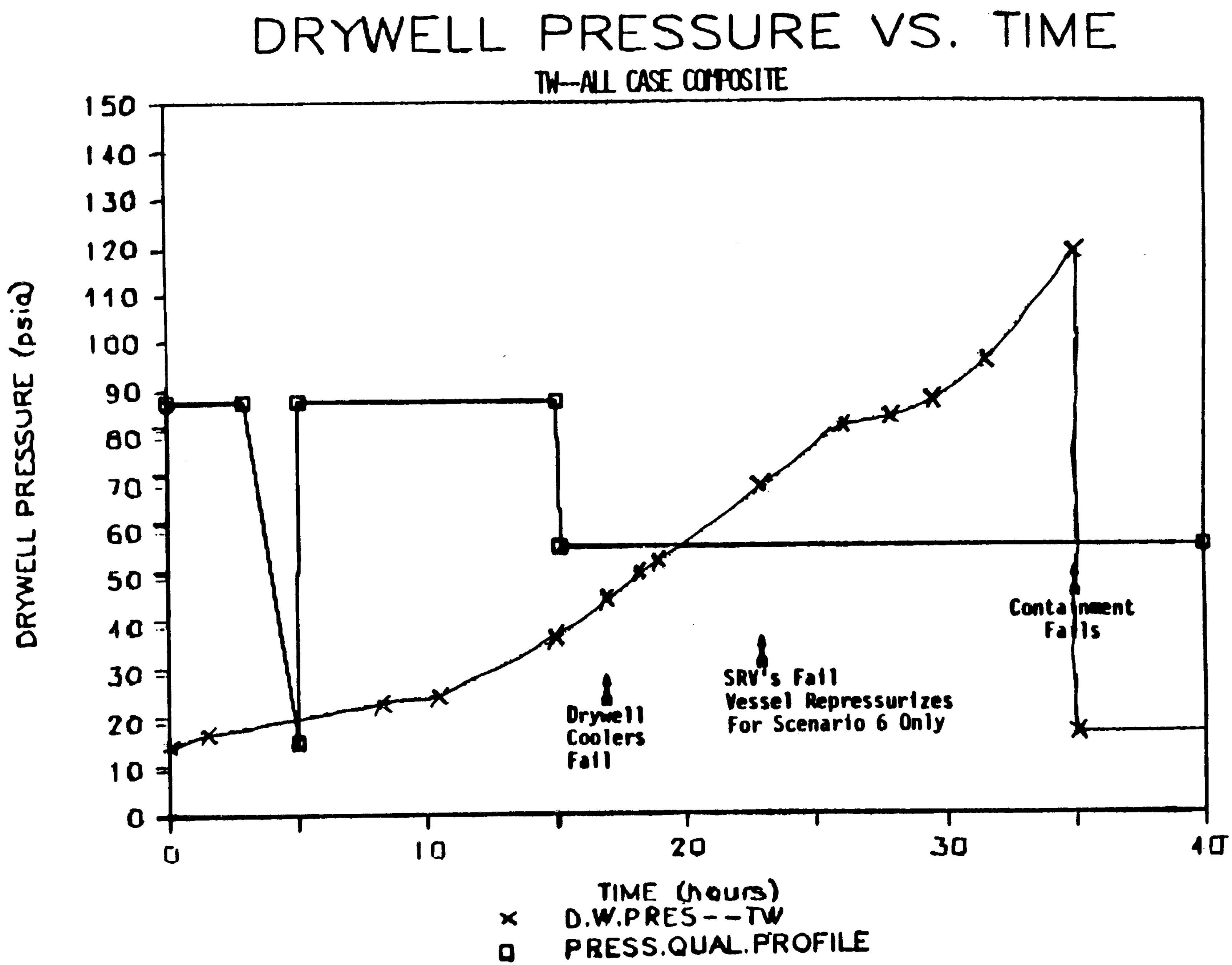


FIGURE C-64 - ENVIRONMENTAL PROFILE NO. 4 COMPARISON TO QUALIFICATION LEVELS FOR CONTAINMENT PRESSURE

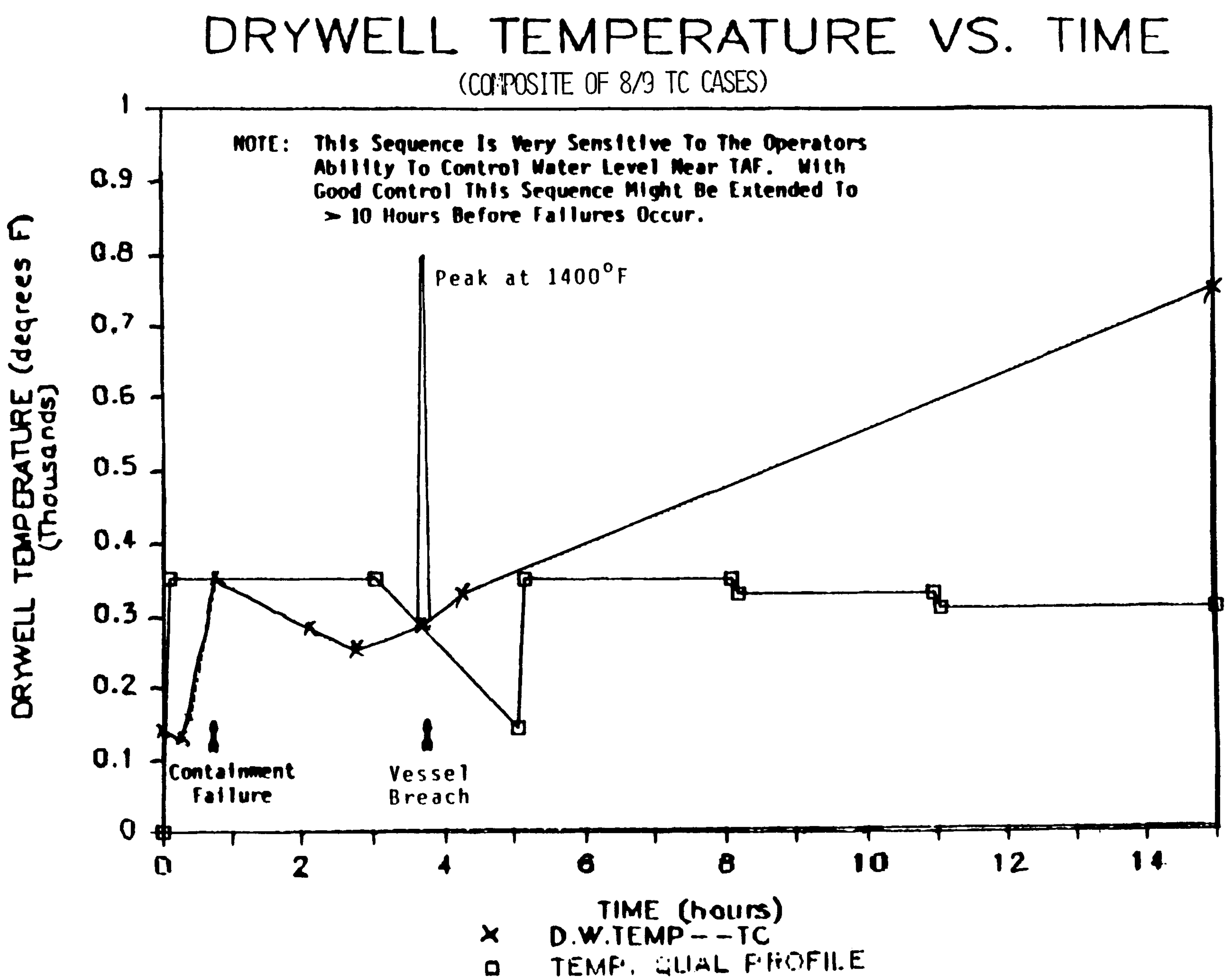


FIGURE C-65 - ENVIRONMENTAL PROFILE NO. 5 COMPARISON TO QUALIFICATION LEVELS FOR DRYWELL TEMPERATURE

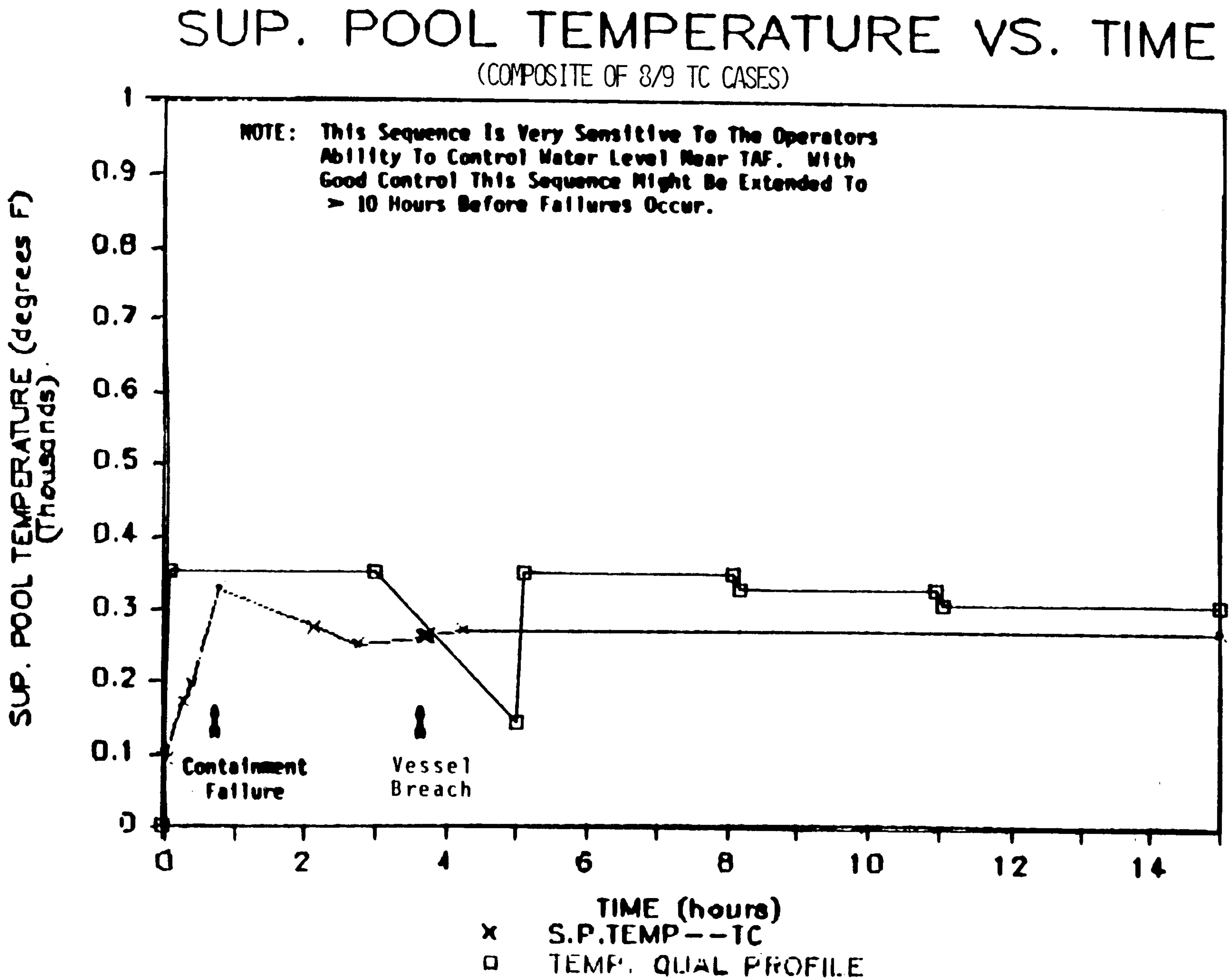


FIGURE C-66 - ENVIRONMENTAL PROFILE NO. 5 COMPARISON TO QUALIFICATION LEVELS FOR WETWELL TEMPERATURE

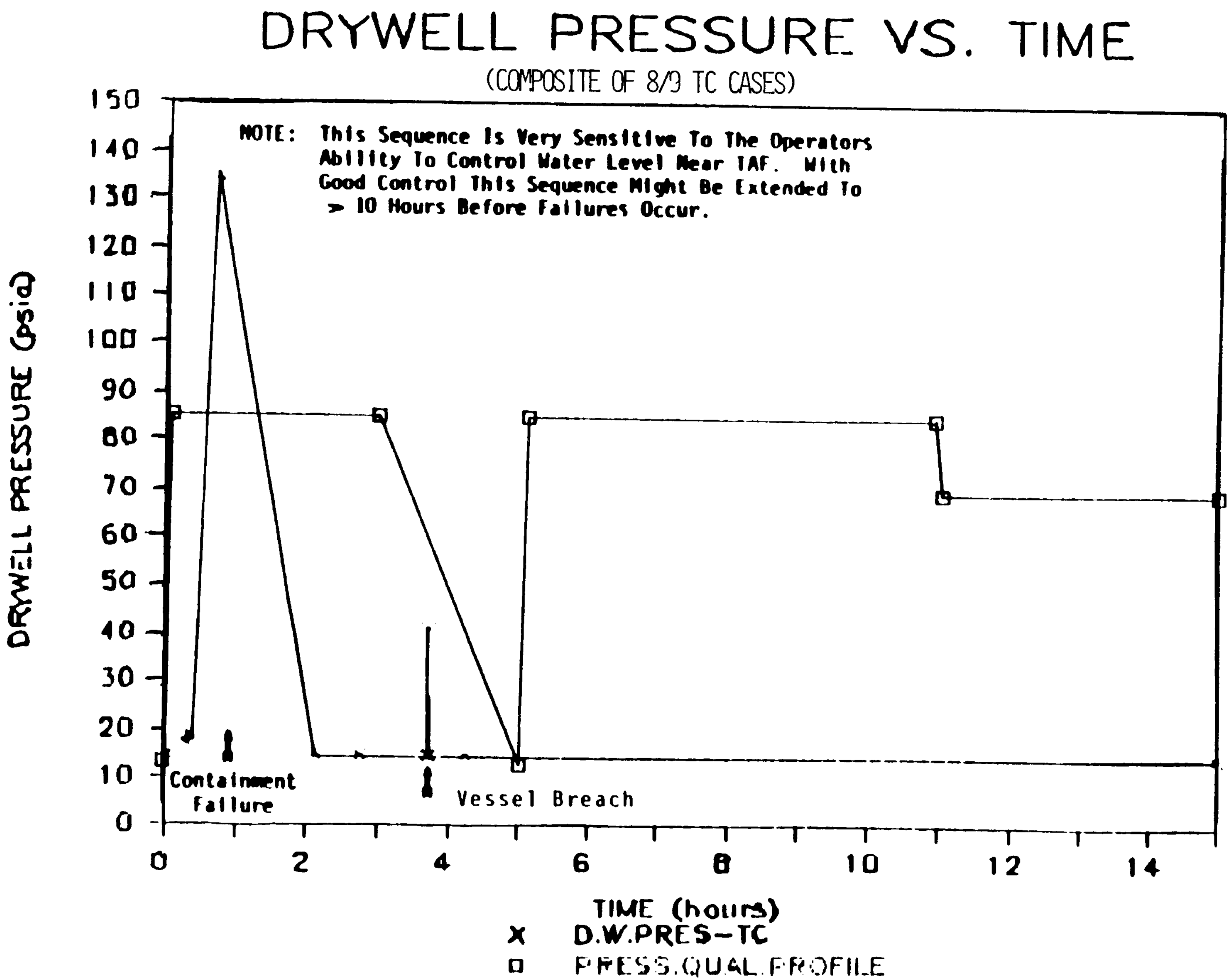


FIGURE C-67- ENVIRONMENTAL PROFILE NO. 5 COMPARISON TO QUALIFICATION LEVELS FOR CONTAINMENT PRESSURE qualification temperature at any time during the accident. Figure C-67 demonstrates that containment failure is definitely due to an overpressure condition. Pressure exceeds the maximum qualification pressure about 30 minutes into the sequence rising to the containment failure level of approximately 130 psia at about 1 hour into the accident. Note that the MARCH data used to generate this profile shows a rather gradual return of compartment pressure to atmospheric level after containment failure. The LTAS code and other MARCH versions show a more sudden drop in compartment pressure. Actual pressure behavior would be largely dependent on the method and size of containment failure. The gradual drop in pressure is displayed here to be consistent with the data the curve is based on. In actuality, a more sudden drop in pressure may occur. The figure indicates that the pressure environment spends about .75 hours in excess of qualification levels reaching a peak amplitude of 132 psia which is about 1.6 times greater than the qualification level.

Figures C-65A, C-66A, and C-67A compare the set number 5A profiles with the qualification profiles. It can be seen from Figure C-65A, that maximum qualification temperature is not exceeded by the time of containment failure (3.9 hours). Assuming that the MSIV open profile follows the MSIV closed profile from this point on, it can be seen that drywell temperature will never exceed maximum qualification level until after vessel breach (estimated at ~6.7 hours). Since it was assumed that suppression pool temperature follows the TC MSIV closed curve from containment failure on, the suppression pool temperature may briefly exceed maximum qualification levels after containment failure but before vessel breach. Containment pressure exceeds maximum qualification level 3.4 hours into the accident with containment failure occurring at about 3.9 hours. Figure C-67A indicates that containment pressure spends ~.5 hours in excess of maximum qualification pressure reaching a peak amplitude of ~132 psia (1.6 times greater than maximum qualification level).

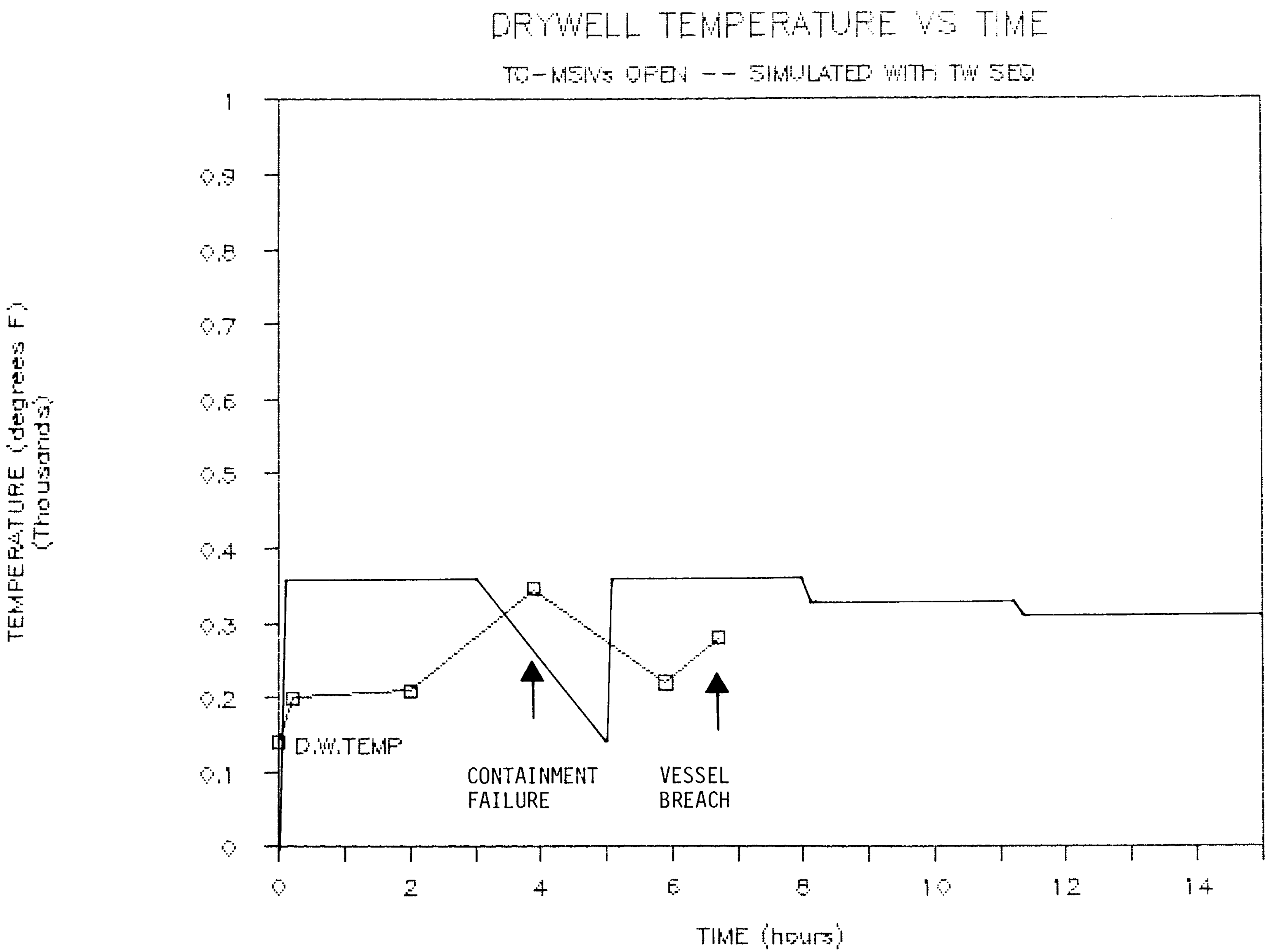


FIGURE C-65A - ENVIRONMENTAL PROFILE NO. 5A COMPARISON TO QUALIFICATION LEVELS FOR DRYWELL TEMPERATURE

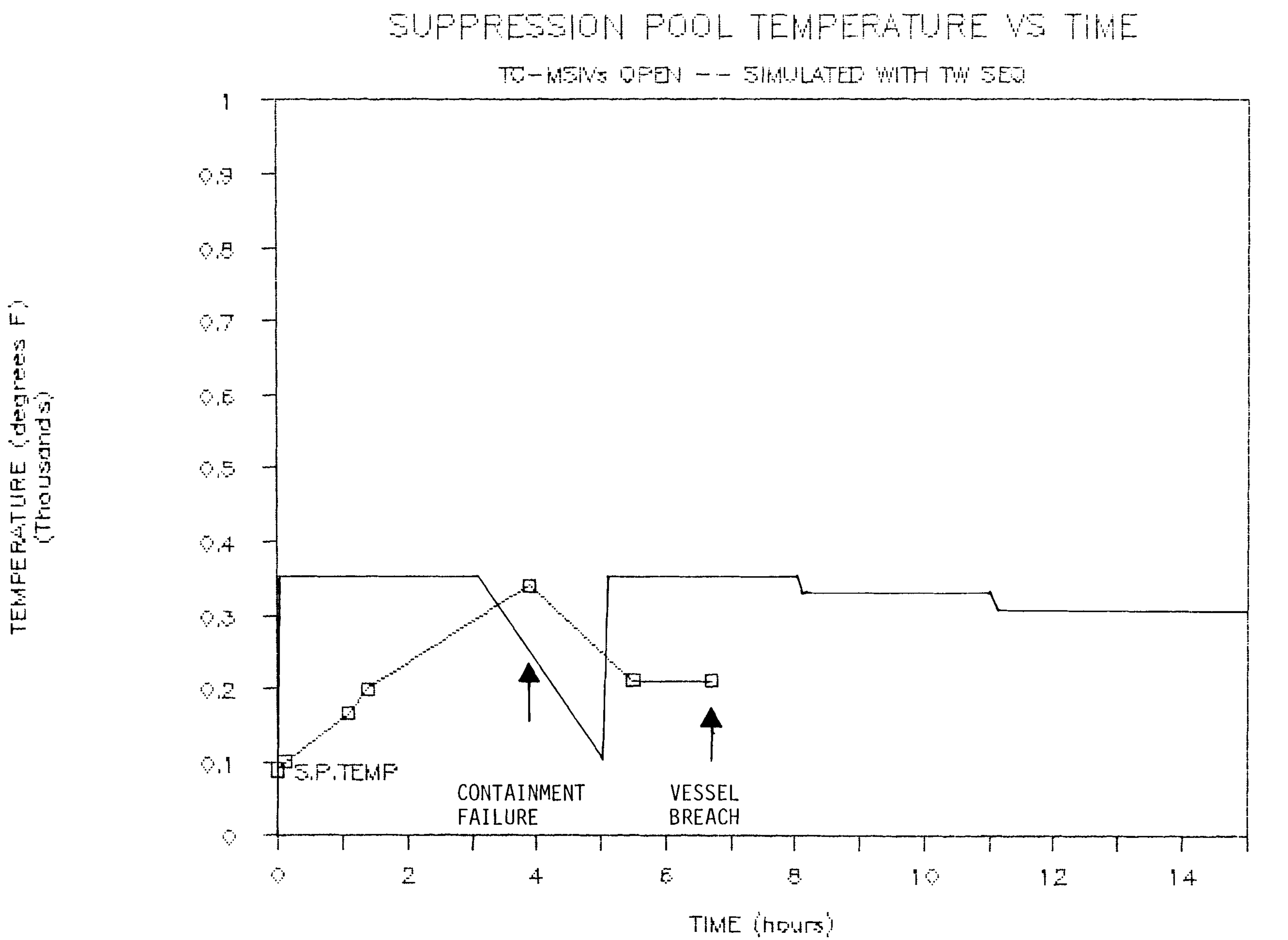


FIGURE C-66A - ENVIRONMENTAL PROFILE NO. 5A COMPARISON TO QUALIFICATION LEVELS FOR WETWELL TEMPERATURE

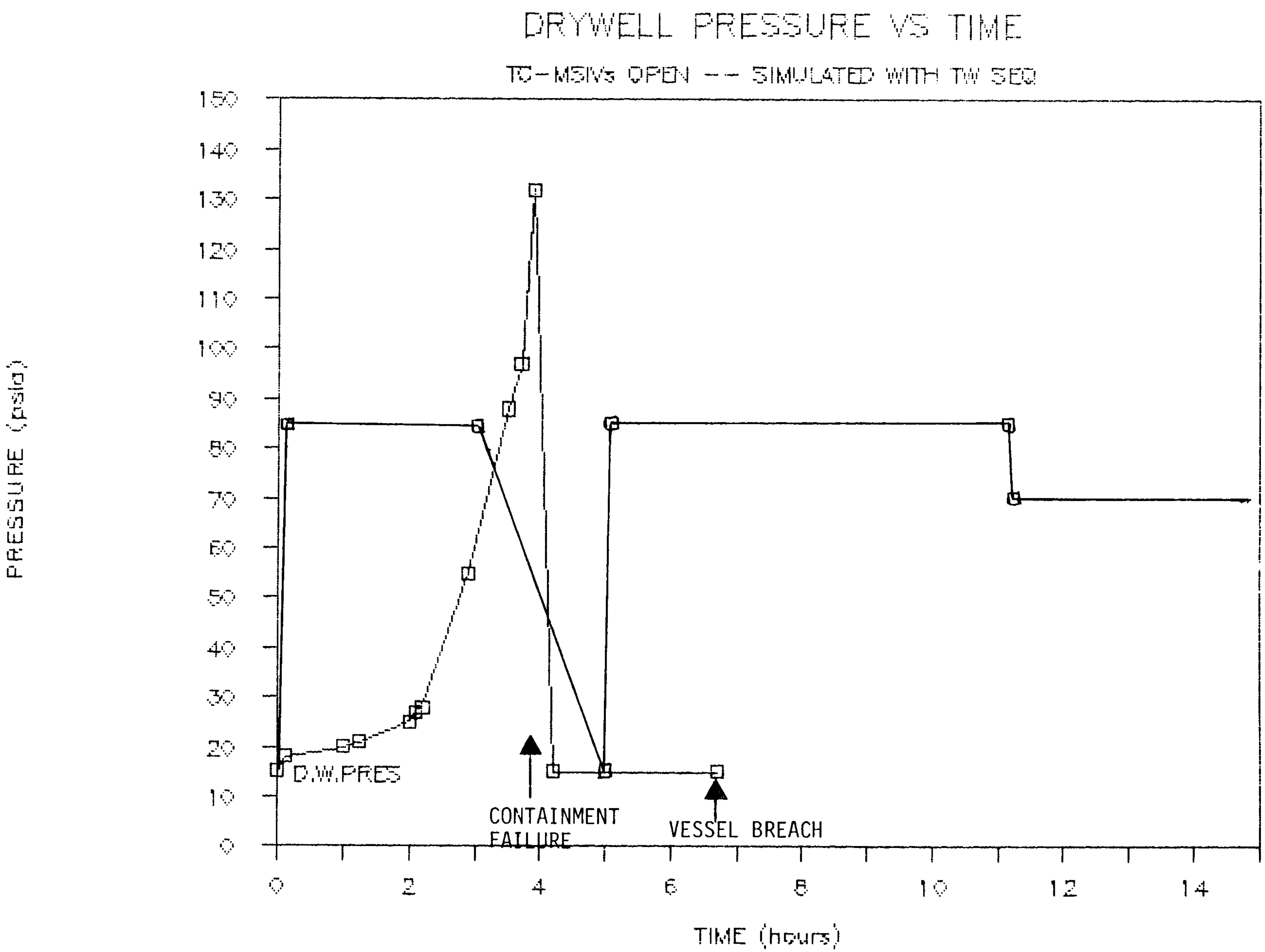


FIGURE C-67A - ENVIRONMENTAL PROFILE NO. 5A COMPARISON TO QUALIFICATION LEVELS FOR CONTAINMENT (DRYWELL) PRESSURE

4.2.5 TQUV

The TQUV sequence is represented by profile sets 6 and 7. Profile set 6 shows the environment response to the cases where the reactor vessel remains at pressure throughout the sequence. Figures C-68, C-69, and C-70 show the environmental comparisons to qualification profiles for the sixth set of environmental profiles. Figure C-68 shows that drywell temperature remains very constant until the point of vessel failure about 5 hours into the accident. From this point drywell temperature rises rapidly exceeding the maximum qualification temperature about 6.5 hours into the accident. The containment spends approximately 1/2 hour in excess of the maximum qualification temperature before containment failure at about 500°F. Figure C-69 indicates that suppression pool temperature never exceeds the qualification profile. Containment pressure exceeds qualification levels approximately 6.2 hours into the accident as shown in Figure C-70. Maximum pressure obtained approaches 110 psia. The containment spends about .8 hours in excess of the maximum qualification pressure before containment failure occurs and pressure returns to atmospheric.

Profile set number 7 described the TQUV sequence when the reactor vessel was depressurized. The behavior of this sequence is identical to the previous TQUV profile except that all major events occur later in time. Figures C-71, C-72, and C-73 show environmental behavior in respect to qualification profiles. Figure C-71 shows drywell temperature exceeding maximum qualification temperature 7.8 hours into the accident and spending about 25 minutes above the qualification profile before containment failure at about 500°F. As before, wetwell temperature never exceeds qualification levels as seen in Figure C-72. Figure C-73 shows containment pressure exceeding qualification levels 7.9 hours into the sequence and remaining in excess of the qualification limit until containment failure at 8.2 hours. Again, maximum pressure is about 110 psia which occurs at the point of containment failure.

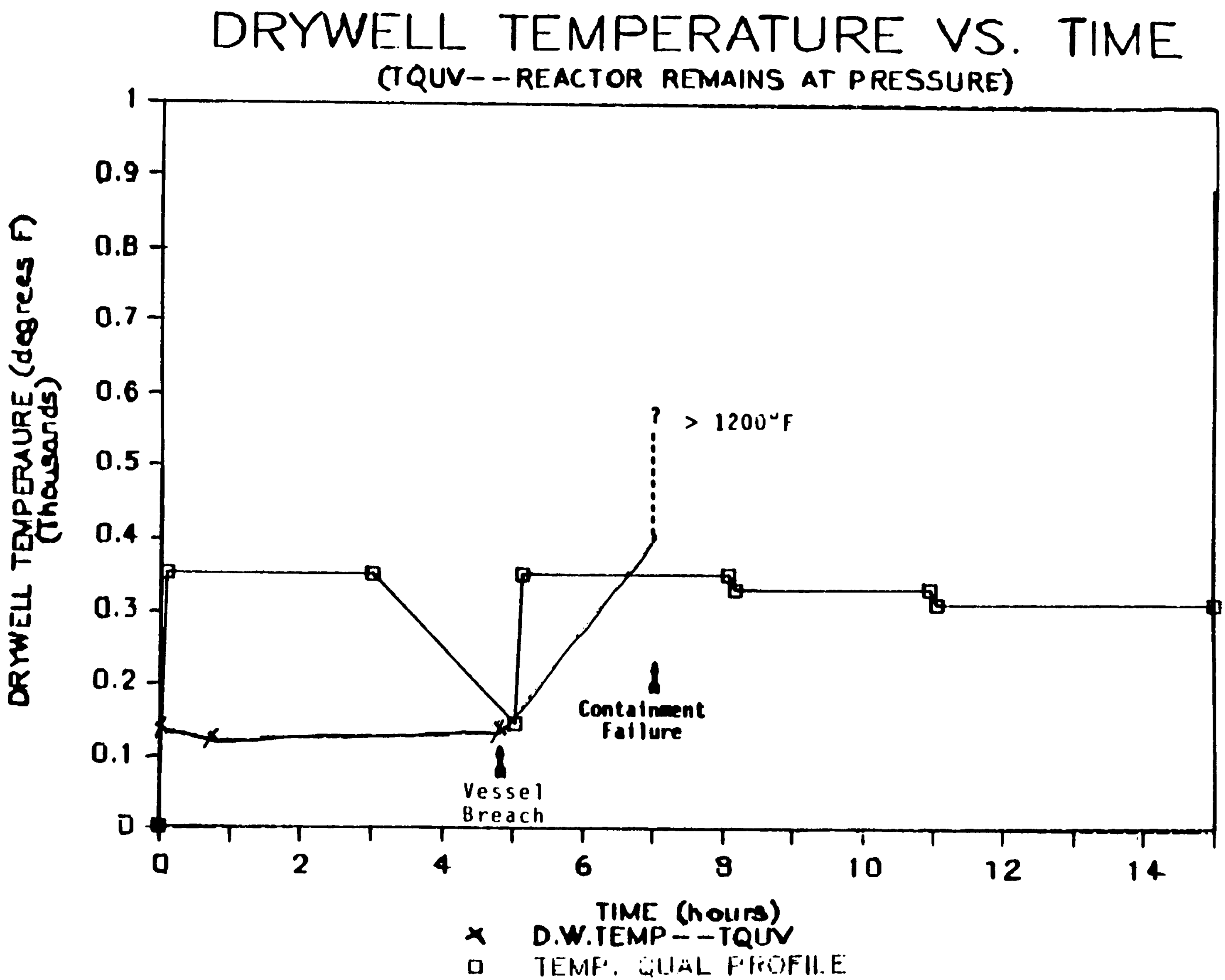


FIGURE C-68 - ENVIRONMENTAL PROFILE NO. 6 COMPARISON TO QUALIFICATION LEVELS FOR DRYWELL TEMPERATURE

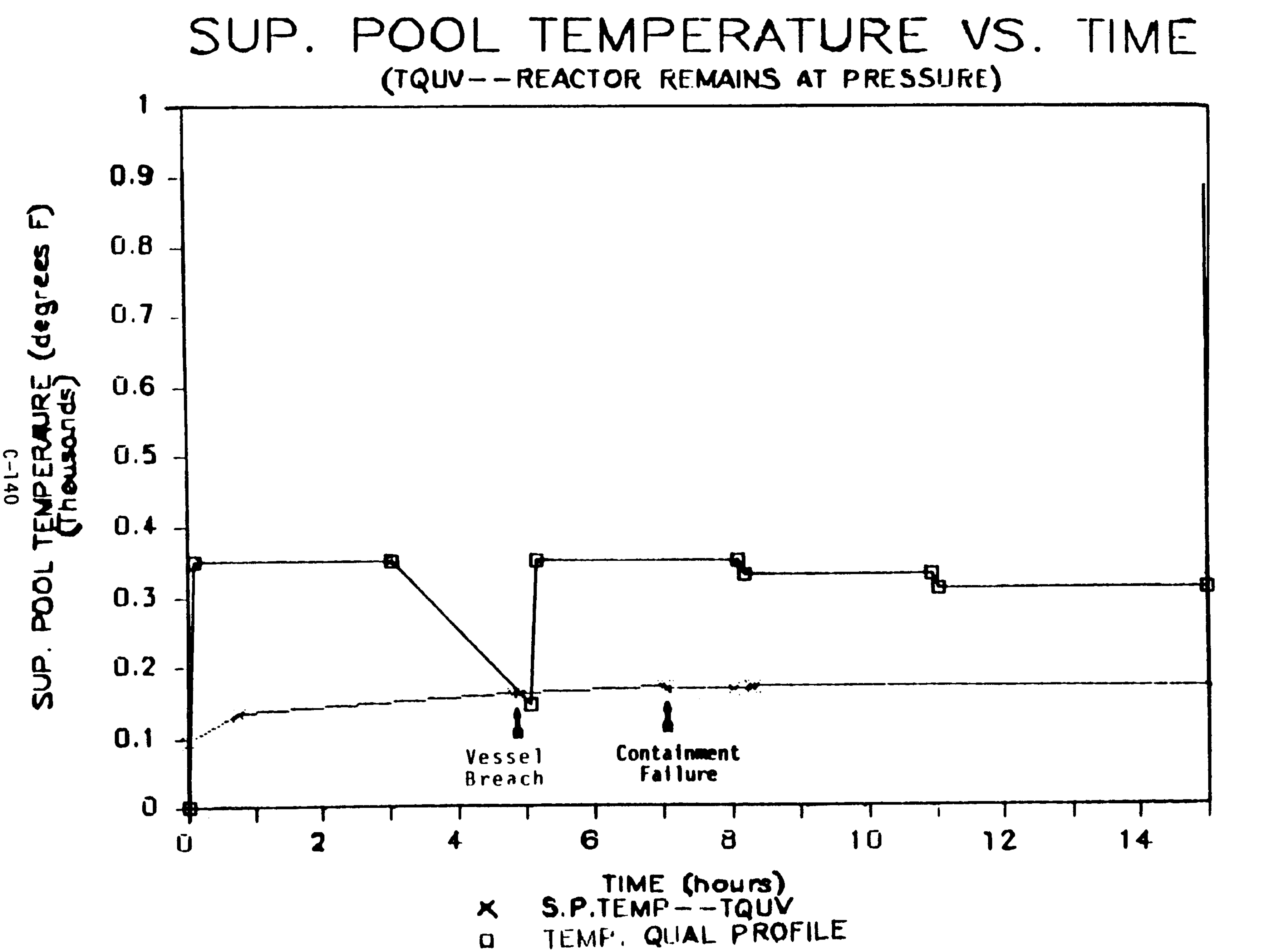


Figure C-69 - Environmental Profile No. 5 Comparison To Qualification Levels For Wetwell Temperature

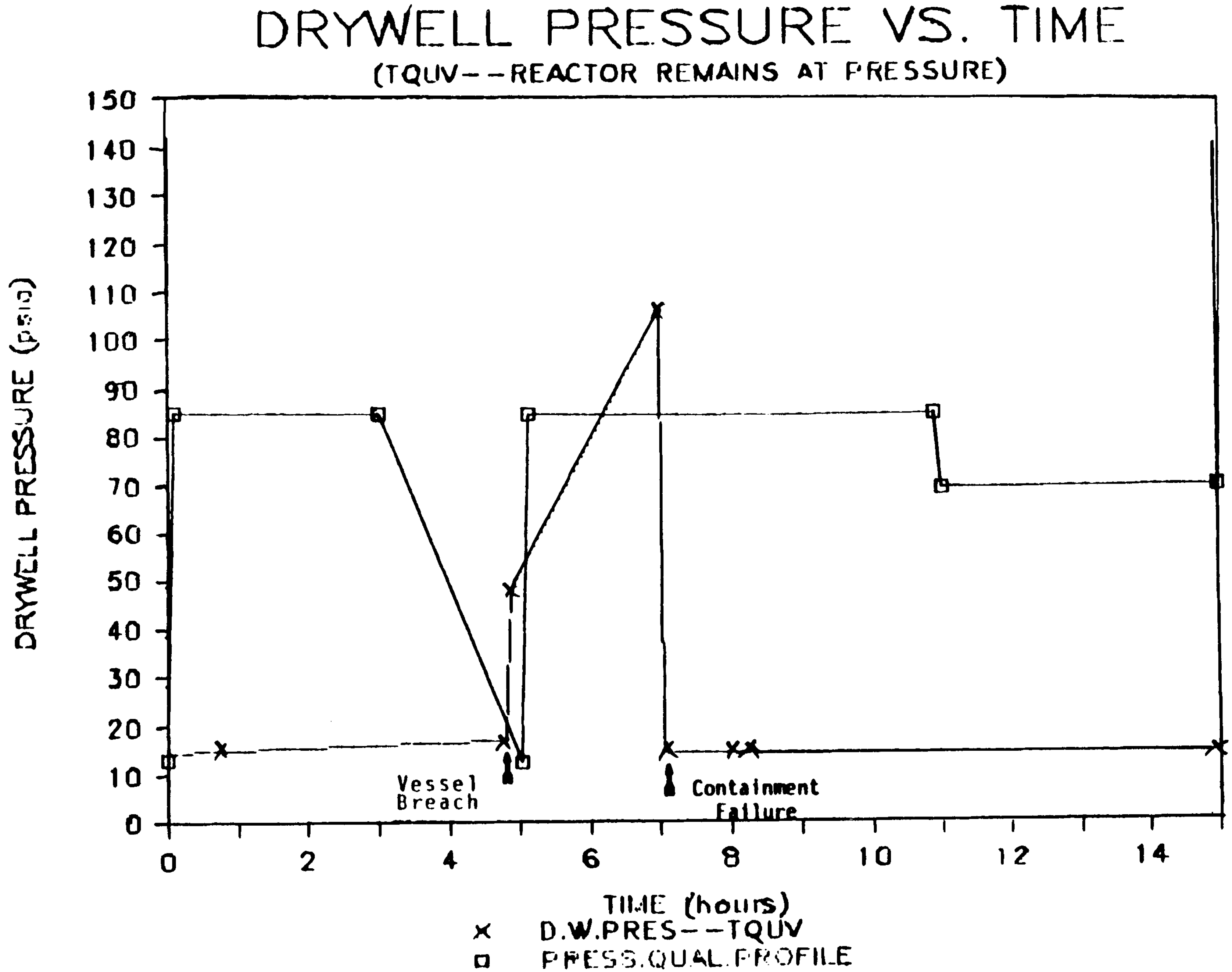


FIGURE C-70 - ENVIRONMENTAL PROFILE NO. 6 COMPARISON TO QUALIFICATION LEVELS FOR CONTAINMENT PRESSURE

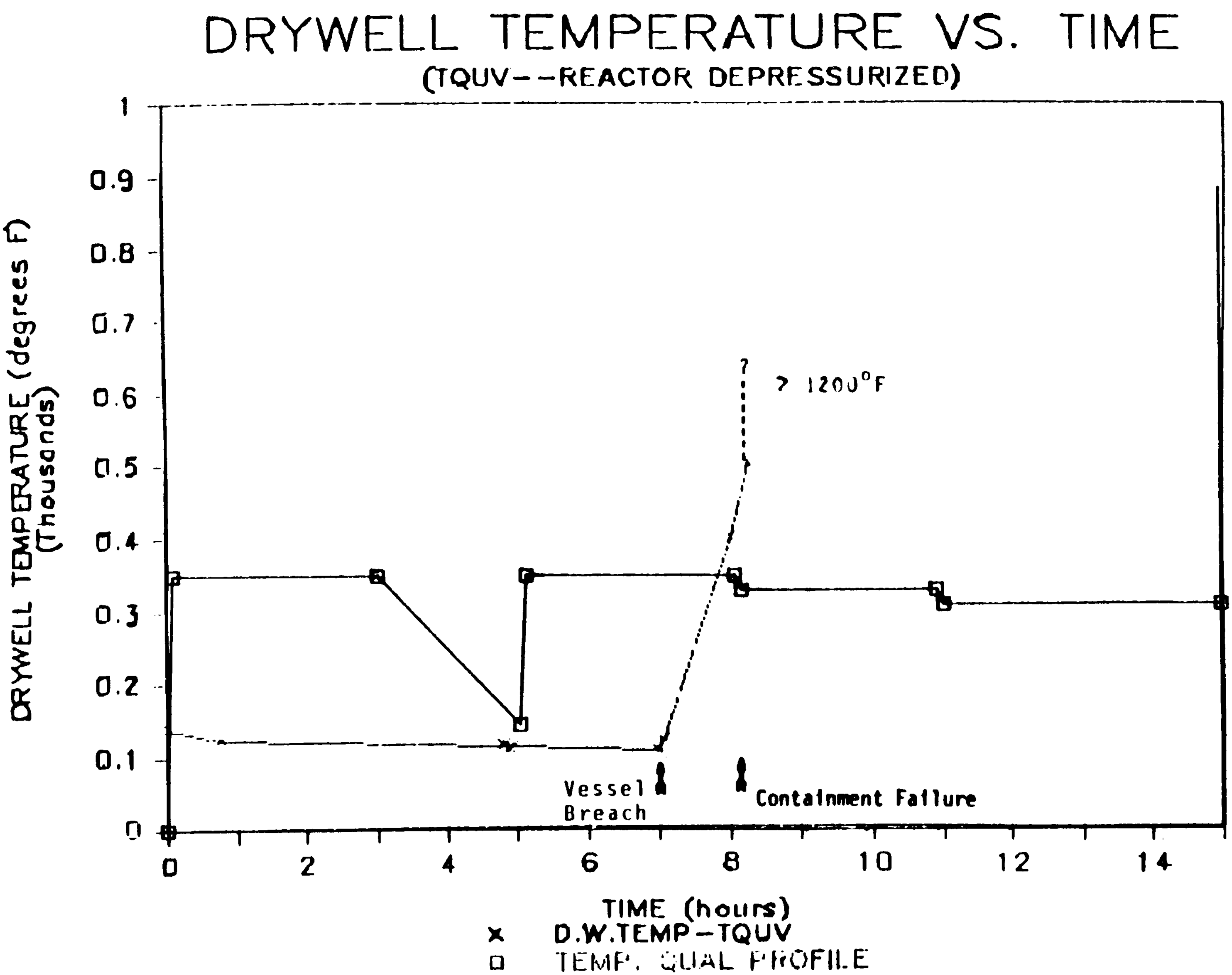


FIGURE C-71 - ENVIRONMENTAL PROFILE NO. 7 COMPARISON TO QUALIFICATION LEVELS FOR DRYWELL TEMPERATURE

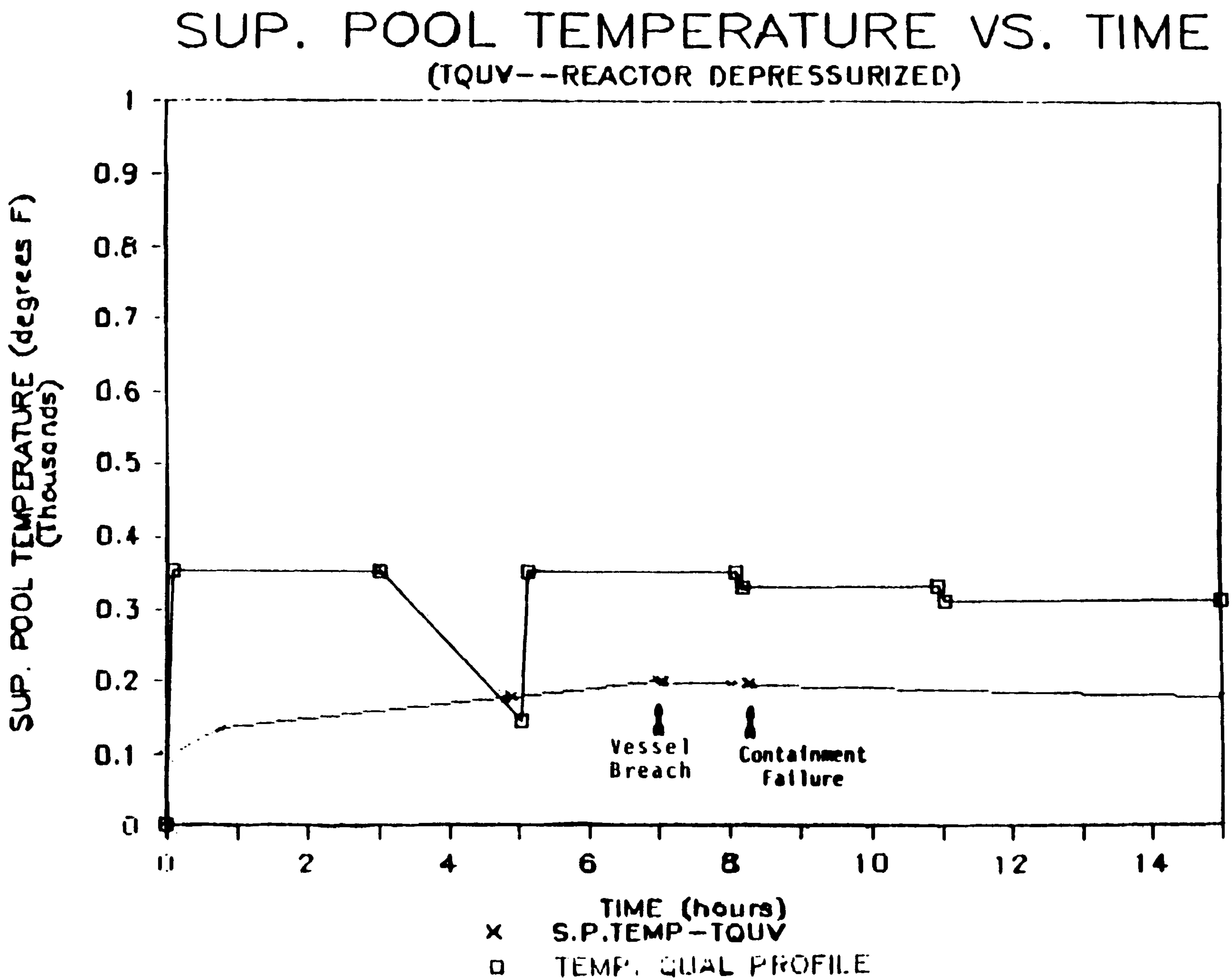


FIGURE C-72 - ENVIRONMENTAL PROFILE NO. 7 COMPARISON TO QUALIFICATION LEVELS FOR WETWELL TEMPERATURE

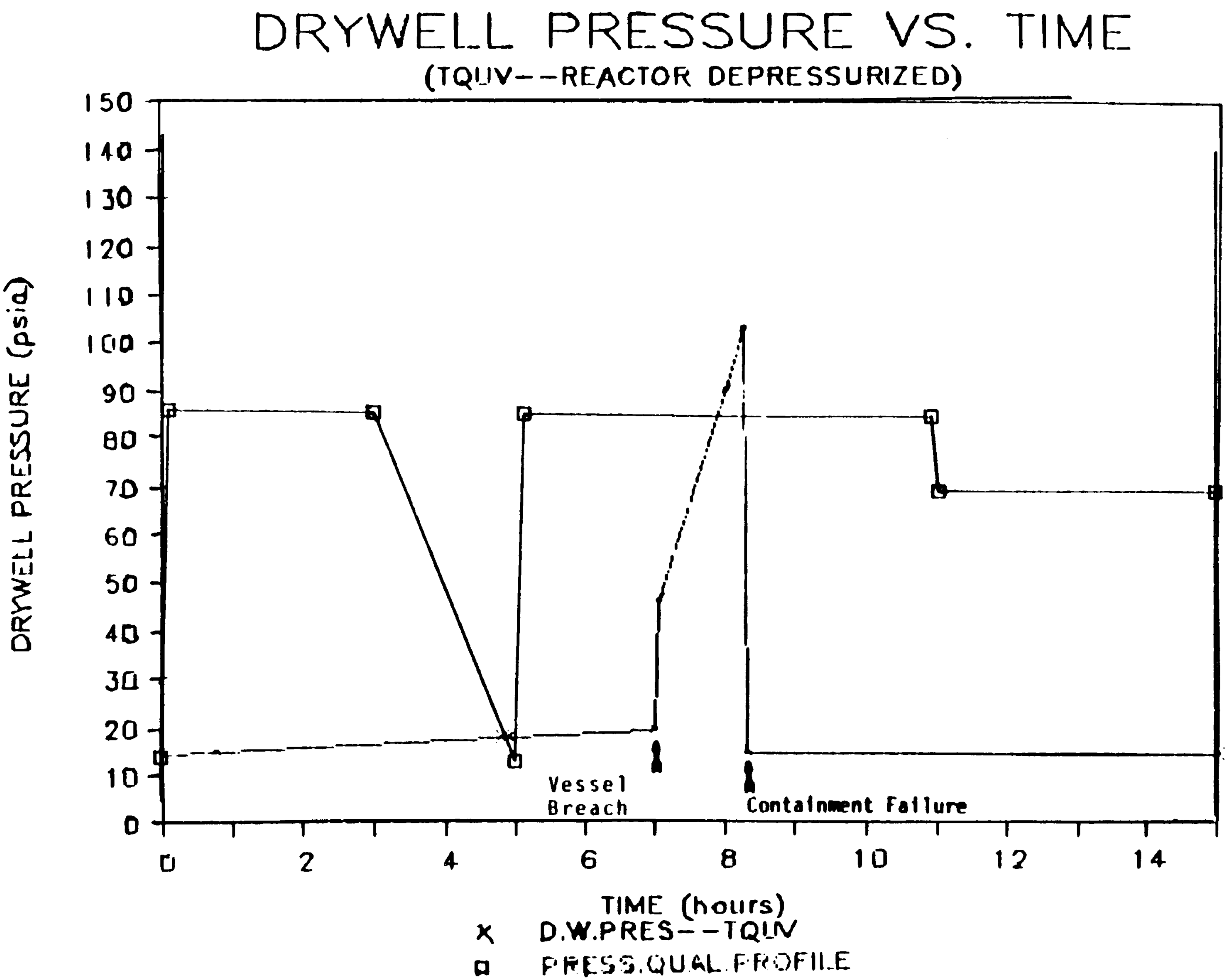


Figure C-73 - Environmental Profile No. 7 Comparison To Qualification Levels For Containment Pressure

4.2.6 AE

Environmental profile set number 8 represents the AE sequence. Figures C-74, C-75, and C-76 show the results of comparing the AE environmental profiles with the qualification profiles. Figure C-74 shows that drywell temperature exceeds maximum qualification temperature about 30 minutes into the accident attaining a peak temperature of over 2000°F. This pulse width is on the order of 10 minutes. The temperature remains well above qualification levels for the entire 100 minutes between containment failure and vessel breach. As seen in previous sequences, suppression pool temperature never exceeds the maximum qualification temperature as shown in Figure C-75. Figure C-76 shows the pressure spike associated with containment failure which exceeds the maximum qualification pressure by a large margin. This spike lasts about 5 minutes above the 85 psia qualification limit. The pressure reaches the containment failure point which is about 1.6 times the qualification limit.

4.3 Summary

This section has presented the comparison of the nine environmental profile sets to typical qualification profiles. Areas where the environmental profiles exceeded the qualification profiles were noted. Maximum temperatures and pressures in excess of the qualification limits as well as total time above qualification levels were also noted. Table C-9 summarizes the useful data from the environmental and qualification profile comparisons.

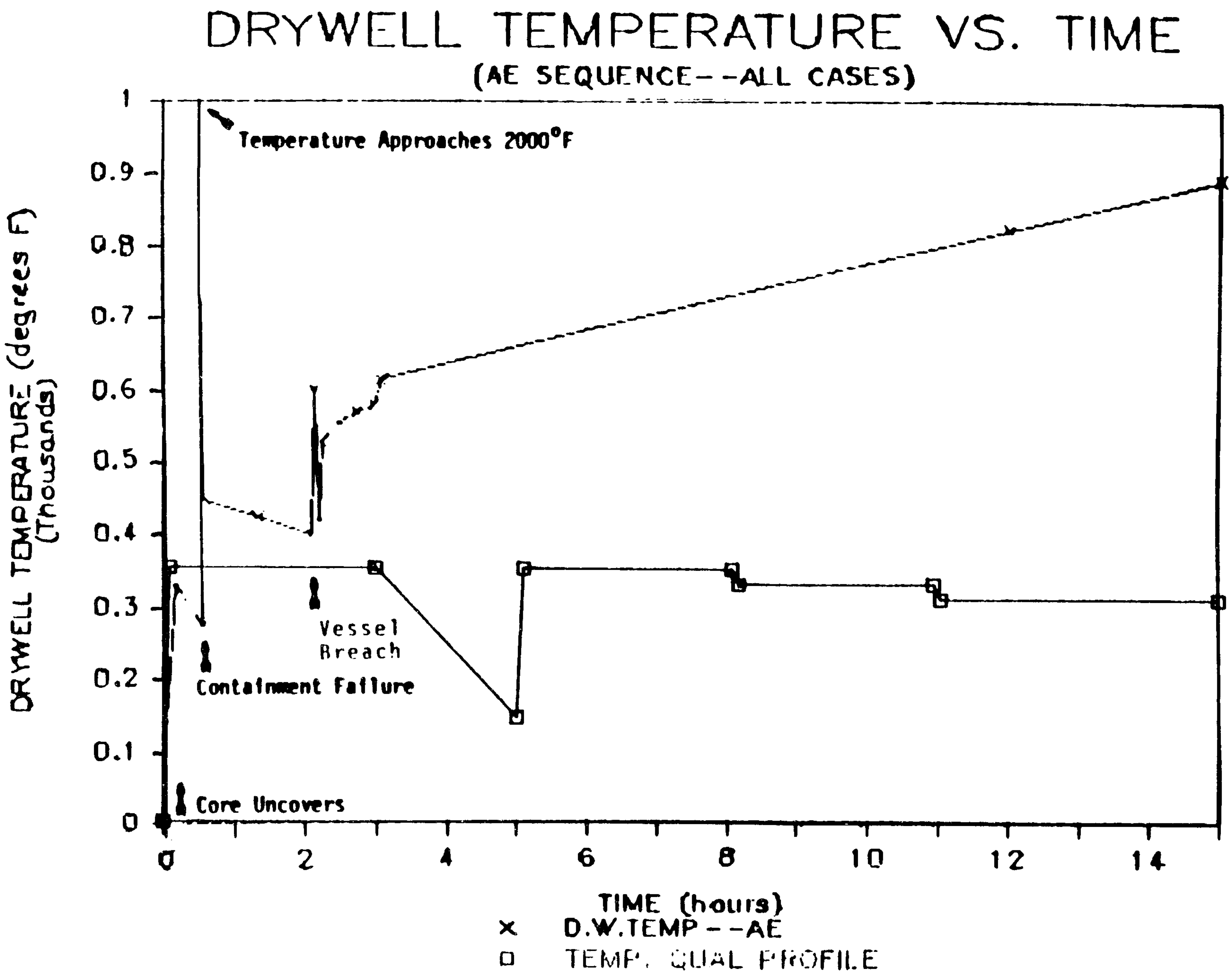


FIGURE C-74 - ENVIRONMENTAL PROFILE NO. 8 COMPARISON TO QUALIFICATION LEVELS FOR DRYWELL TEMPERATURE

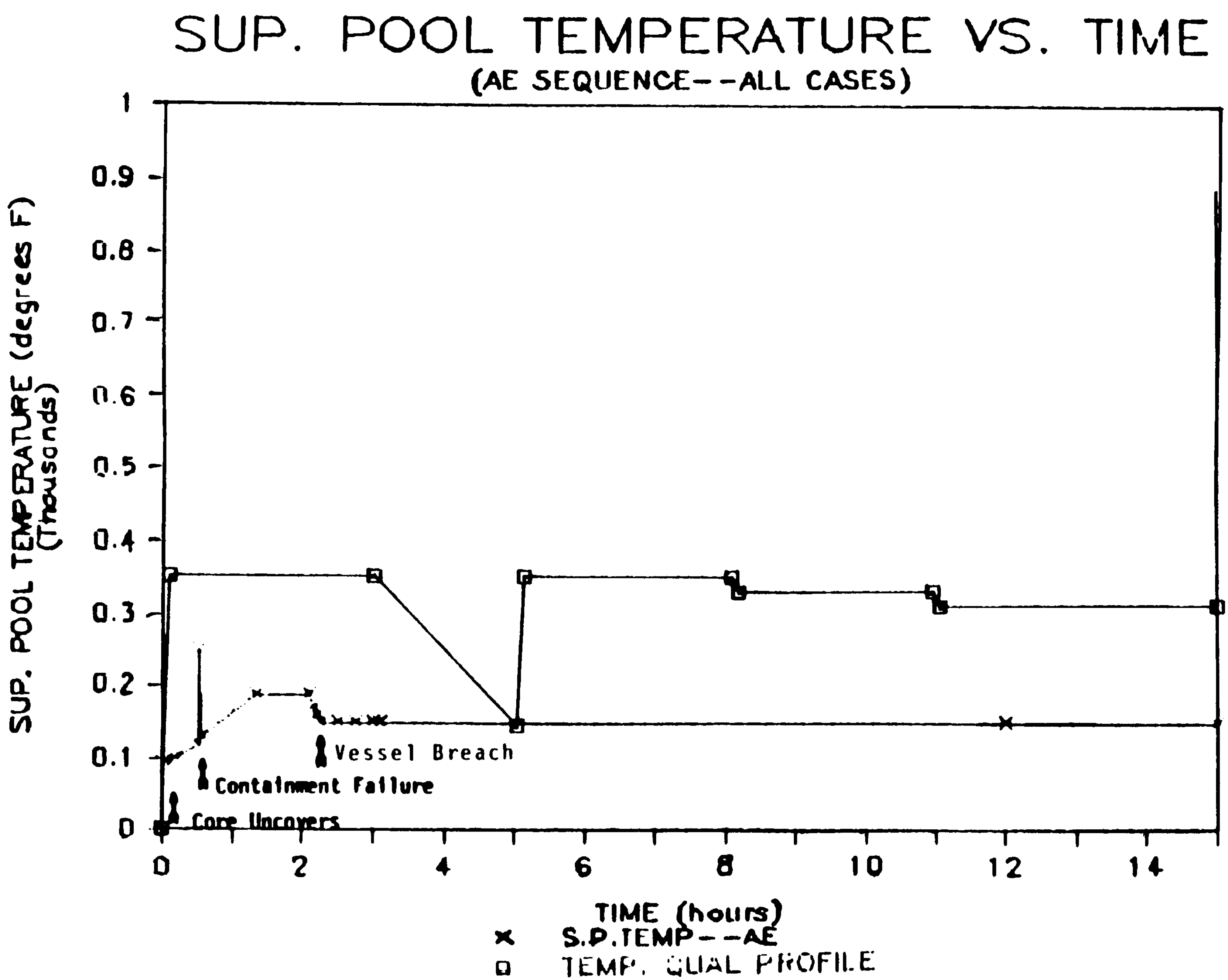


FIGURE C-75 - ENVIRONMENTAL PROFILE NO. 3 COMPARISON TO QUALIFICATION LEVELS FOR WETWELL TEMPERATURE

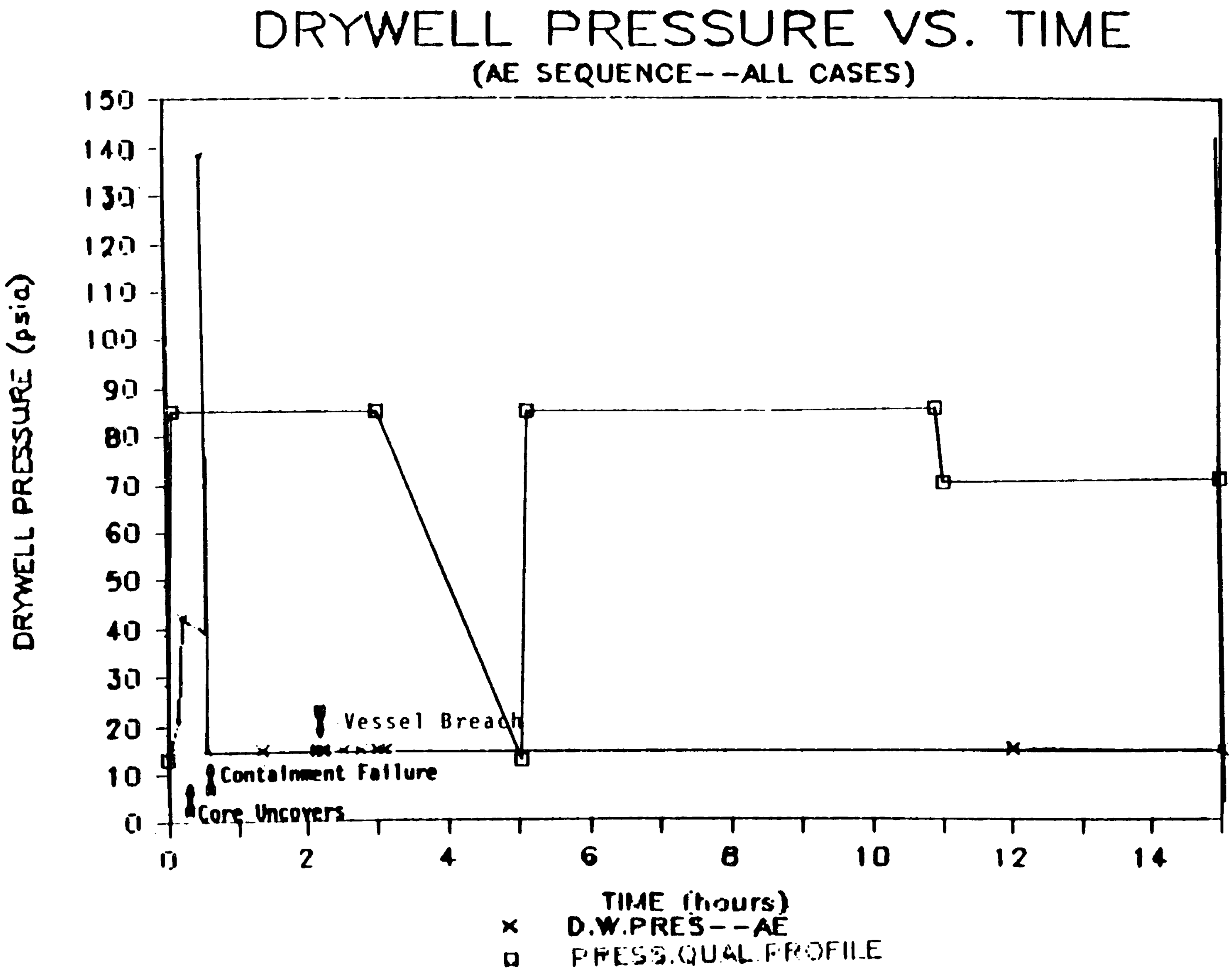


FIGURE C-76 - ENVIRONMENTAL PROFILE NO. 8 COMPARISON TO QUALIFICATION LEVELS FOR CONTAINMENT PRESSURE

TABLE C-9 - ENVIRONMENTAL PROFILE SUMMARY

PROFILE NO.	ACCIDENT	TIME OF VESSEL BREACH (1)	TIME OF CONT. FAIL (1)	ENVIR. VARIABLE	TIME ABOVE QUAL LVL (1)	PEAK AMP./% QUAL LVL (2, 3)	TIME OF (1) PEAK AMP.	TIME FIRST (1) EXCEEDS QUAL. LVL
1	TB-Short No/late Oper. Action	2.1	3.2	D.W. Temp.	.20	500° / 142%	3.2	3.0
				S.P. Temp.	0	180° / 51%	3.2	---
				D.W. Press.	.2	100 / 118%	3.2	3.0
2	TB-Short Vessel Depress. Early	2.3	2.9	D.W. Temp.	.2	500° / 142%	2.9	2.7
				S.P. Temp.	0	170° / 49%	2.9	---
				D.W. Press.	0	75 / 88%	2.9	---
3	TB-Long	9.0	10.0	D.W. Temp.	2.0	500° / 142%	10.0	8.0
				S.P. Temp.	0	212° / 60%	10.0	---
				D.W. Press.	.5	100 / 118%	10.0	9.5
4	TW	39	35	D.W. Temp.	11	500° / 142%	39	28
				S.P. Temp.	8	350° / 100%	35	34
				D.W. Press.	7	125 / 147%	35	28
5(5A)*	TC	3.8(6.7)*	0.9(3.9)*	D.W. Temp.	<.1(0)*	360°/103% (345°/98%)*	.9(3.9)*	.9(-)*
				S.P. Temp.	0(0)*	340°/97% (340°/97%)*	.9(3.9)*	---(-)*
				D.W. Press.	.75(.5)*	132/155% (132°/155%)*	.9(3.9)*	.6(3.4)*
6	TQUV Vessel At Pressure	4.9	7.0	D.W. Temp.	.5	500° / 142%	7.0	6.5
				S.P. Temp.	0	170° / 49%	7.0	---
				D.W. Press.	.8	110 / 129%	7.0	6.2
7	TQUV-- Vessel Depress.	7.0	8.2	D.W. Temp.	.4	500° / 142%	8.2	7.8
				S.P. Temp.	0	200° / 57%	7.0	---
				D.W. Press.	.3	110 / 129%	8.2	7.9
8	AE	2.1	.66	D.W. Temp.	1.44	2000° / 571%	.7	.6
				S.P. Temp.	0	270° / 77%	.66	---
				D.W. Press.	.1	138 / 162%	.7	.6

Notes:

(1) All Times in Hours
(2) All Temperatures in °F/All Pressures in psia
(3) Percentage Based on Exceeding Qualification Profile Maximum Value

*Information in parenthesis is for MSIV open case.

5.0 OTHER ENVIRONMENTAL CONSIDERATIONS

The previous sections have focused on the pressures and temperatures for the selected severe accident sequences. However, other environmental considerations must be addressed to complete the investigation into the survivability of the components of interest under severe accident conditions. These considerations include humidity, flooding, water spray, radiation, and vibration. Each is addressed briefly in the following sections.

5.1 Humidity Considerations

Based on the fact that all electrical equipment within the Browns Ferry containment has been qualified to 100% humidity conditions, humidity has been adequately addressed by design basis conditions.

5.2 Flooding

The components of interest are all believed to be located well above any possible submergence levels based on plant qualification reports and equipment location information. It is therefore assumed that flooding represents little or no hazard to the components of interest. However information regarding the basis for the flooding calculations, for Browns Ferry Unit 1, could change this conclusion and cause the submergence issue to have to be reevaluated.

5.3 Water Spray

Equipment could be exposed to water spray. This would be the case when, for instance, drywell sprays (RHR) operate to cool and prevent overpressurization of the containment. Since direct water spray may represent a worse condition than 100% humidity, it is suggested that any tests performed as part of the PEEESAS program consider spraying the components.

5.4 Radiation and Aerosols

Current qualification criteria require consideration of radiation dose levels derived from assuming that 100% of the noble gases, 50% of the halogens, and 1% of other fission products, including aerosols, are released to the containment environment. This leads to a maximum dose of 150 Mrads. Current information (Ref 11) suggests that this dose is appropriate for severe accident sequences. However, there are uncertainties when considering aerosol and other fission product dispersal patterns, such as direct plateout on the equipment or preferential radiation shine. Aerosal generation from concrete attack was not considered in this study due to the wide variablity of concrete types.

5.5 Vibration

Particularly for those sequences where containment failure occurs before vessel breach (TW, TC, AE), the resulting blowdown forces could cause vibration of equipment. Those components needed after the blowdown and until vessel failure may need to be examined under blowdown forces. Further review is necessary to determine if any resulting vibration could be worse than current seismic testing requirements.

6.0 SUMMARY

The appendix began by explaining the LTAS code and other data sources used to construct environmental profiles for the 5 accident sequences to be examined by this program. The methodology used to construct the profiles was then explained in Section 2.4. The rest of Section 2 explained the actual construction of the 14 scenario profile sets used to describe the 5 accident sequences. Because of similarities in some of the profiles, only 9 actual profile sets were needed to describe the environments encountered in the 5 accident sequences.

With the environmental profiles calculated and explained, the next area of the appendix described current qualification standards which electrical equipment must meet. Section 3 of the appendix showed typical temperature and pressure qualification profiles.

Section 4 of the appendix went through a careful comparison of the environmental profiles constructed in Section 2 with the typical qualification profiles presented in Section 3. Details such as time the environmental profile first exceeds the maximum level of the qualification profile, peak amplitude, and total time above the maximum qualification level were noted.

The last section of the appendix addressed environments other than temperature and pressure. These environments included humidity, flooding, water spray, radiation/aerosols, and vibration.

7.0 REFERENCES

1. A.M. Kolaczkowski (SAIC), "INFORMATION REGARDING LIKELY SCENARIOS FOR CERTAIN ACCIDENT SEQUENCES", SAIC, May 21, 1985.

2. A.M. Kolaczkowski (SAIC), "IDENTIFICATION OF ELECTRICAL EQUIPMENT TO BE EXAMINED BY THE PEEESAS PROGRAM", SAIC, June 4, 1985.

3. R.M. Harrington (ORNL), L.J. Ott (ORNL), "THE EFFECT OF SMALL-CAPACITY, HIGH-PRESSURE INJECTION SYSTEMS OF TQUV SEQUENCES AT BROWNS FERRY UNIT ONE", NUREG/CR-3179, ORNL/TM-8635, September 1983.

4. R.M. Harrington (ORNL), L.C. Fuller (ORNL), "BWR-LTAS: A BOILING WATER REACTOR LONG-TERM ACCIDENT SIMULATION CODE", NUREG/CR-3764, ORNL-TM-9163, February 1985.

5. D.D. Yue (ORNL) T.E. Cole (ORNL), "BWR 4/MARK I ACCIDENT SEQUENCES ASSESSMENT", NUREG/CR-2825, ORNL/TM-8148, November 1982.

6. D.H. Cook (ORNL), R.M. Harrington (ORNL), S.R. Greene (ORNL), S.A. Hodge (ORNL), "LOSS OF DHR SEQUENCES AT BROWNS FERRY UNIT ONE-ACCIDENT SEQUENCE ANALYSIS", NUREG/CR-2973, ORNL/TM-8532, May 1983.

7. R.R. Schultz (NRC), S.R. Wagoner (NRC), "THE STATION BLACKOUT TRANSIENT AT THE BROWNS FERRY UNIT ONE PLANT--A SEVERE ACCIDENT SEQUENCE ANALYSIS", EGG-NTAP-6002, September 1982.

8. R.M. Harrington (ORNL), S.A. Hodge (ORNL), "ATWS AT BROWNS FERRY UNIT ONE-ACCIDENT SEQUENCE ANALYSIS", NUREG/CR-3470, ORNL/TM-8902, July 1984.

9. D.H. Cook (ORNL), S.R. Greene (ORNL), R.M. Harrington (ORNL), S.A. Hodge (ORNL), D.D. Yue (ORNL), "STATION BLACKOUT AT BROWNS FERRY UNIT ONE--ACCIDENT SEQUENCE ANALYSIS", NUREG/CR-2182, ORNL/NUREG/TM-455/VI, November 1981.

10. J.A. Gieseke (BCL), P. Cybulskis (BCL), R.S. Denning (BCL), K.W. Lee (BCL), M.R. Kuhlman (BCL), H. Chen (BCL), "RADIONUCLIDE RELEASE UNDER SPECIFIC LWR ACCIDENT CONDITIONS", BMI-2104 Volume II, July 1984.

11. Communication with W. Farmer (NRC/RES), Programmatic review meeting for the Performance Evaluation of Electrical Equipment During Severe Accident States program, Rockville, MD, July 23, 1985.

APPENDIX D

DATA ANALYSIS AND FINAL TEST RECOMMENDATIONS

TABLE OF CONTENTS

TABLE OF CONTENTS (CONTINUED)

TABLE OF CONTENTS (CONCLUDED)

LIST OF TABLES

LIST OF TABLES (CONTINUED)

LIST OF TABLES (CONCLUDED)

LIST OF FIGURES

LIST OF FIGURES (CONCLUDED)

1.0 INTRODUCTION

This appendix includes the analysis of the data presented in Appendices A, B, and C to arrive at a best estimate of which equipment and environments should be chosen for testing.

1.1 Review of Appendices A, B, and C

Appendix A reviewed current PRA estimates to select the accident sequences for a BWR/Mark I (Based on the amount of data available and large number of past studies performed, Browns Ferry-1 was chosen to represent a BWR/Mark I). The accidents were chosen based on being among the most probable and the highest risk contributors, but not necessarily the most severe. The accident sequences selected included TB, TW, TC, TQUV, and AE. Within each accident sequence, the more likely scenarios (accident progressions) were determined. This resulted in 14 scenarios representing the 5 accident sequences.

Appendix B identified critical electrical equipment within the containment. This equipment would be subjected to the harshest environments generated in a severe accident. Only that equipment which could help mitigate or provide important plant status for the selected accident sequences were carried forward for examination. The selected equipment included MSIVs, HPCI/RCIC isolation valves, an RHR shutdown valve, SRV air solenoid and pilot assemblies, thermocouples, RTDs, pressure indications, and radiation and hydrogen monitoring devices.

Appendix C dealt with defining the environments that would be generated in each of 14 scenarios representing the 5 selected accident sequences. Meltdown Accident Response Characteristics (MARCH) code results from past studies were coupled with new Long Term Accident Sequence (LTAS) code results to form a composite environmental profile describing pressure and temperature behavior verses time for each of the accident scenarios. The resulting environmental profiles were then compared to typical qualification profiles to indicate areas where equipment might experience environmental levels in excess of typical qualification levels. Besides the pressure and temperature profiles created by the above, additional environmental parameters were examined: radiation, humidity, flooding, spray, and mechanical vibration. Although these additional environments were deemed insufficient to cause failure by themselves, synergistic effects are investigated in Section 2.3 of this appendix.

1.2 Goals For This Appendix

The primary focus of this appendix is to develop an analysis methodology to sort and rank the data presented in the first three appendices. An overview of the four step analysis methodology is presented as Figure D-1. The first step in the methodology is a three phase screening process used to examine the survivability of containment equipment based on environment, possible failure modes, and equipment functional importance for a given accident sequence. This three phase screening process is described in detail in Section 2.0 of this Appendix.

After the screening process is complete, the results from each phase of the screening process are qualitatively ranked to allow comparison of a given piece of equipment's relative standing in terms of environment, failure mode, and function. A piece of equipment which scores high in all categories becomes a potential candidate for further analysis and possible test pending the PRA analysis review. This second step in the analysis methodology is implemented in Section 3.0 of this appendix.

The third step of the analysis methodology examines possible effects, if any, that this survivability data may imply for current PRA estimates of sequence probabilities. This step in the analysis methodology is developed in Section 4.0 of the appendix.

The fourth and last step in the analysis methodology is the recommendations based on the results of the first three steps. The recommendations for equipment test are presented in Section 5.0 of this appendix.

ANALYSIS METHODOLOGY

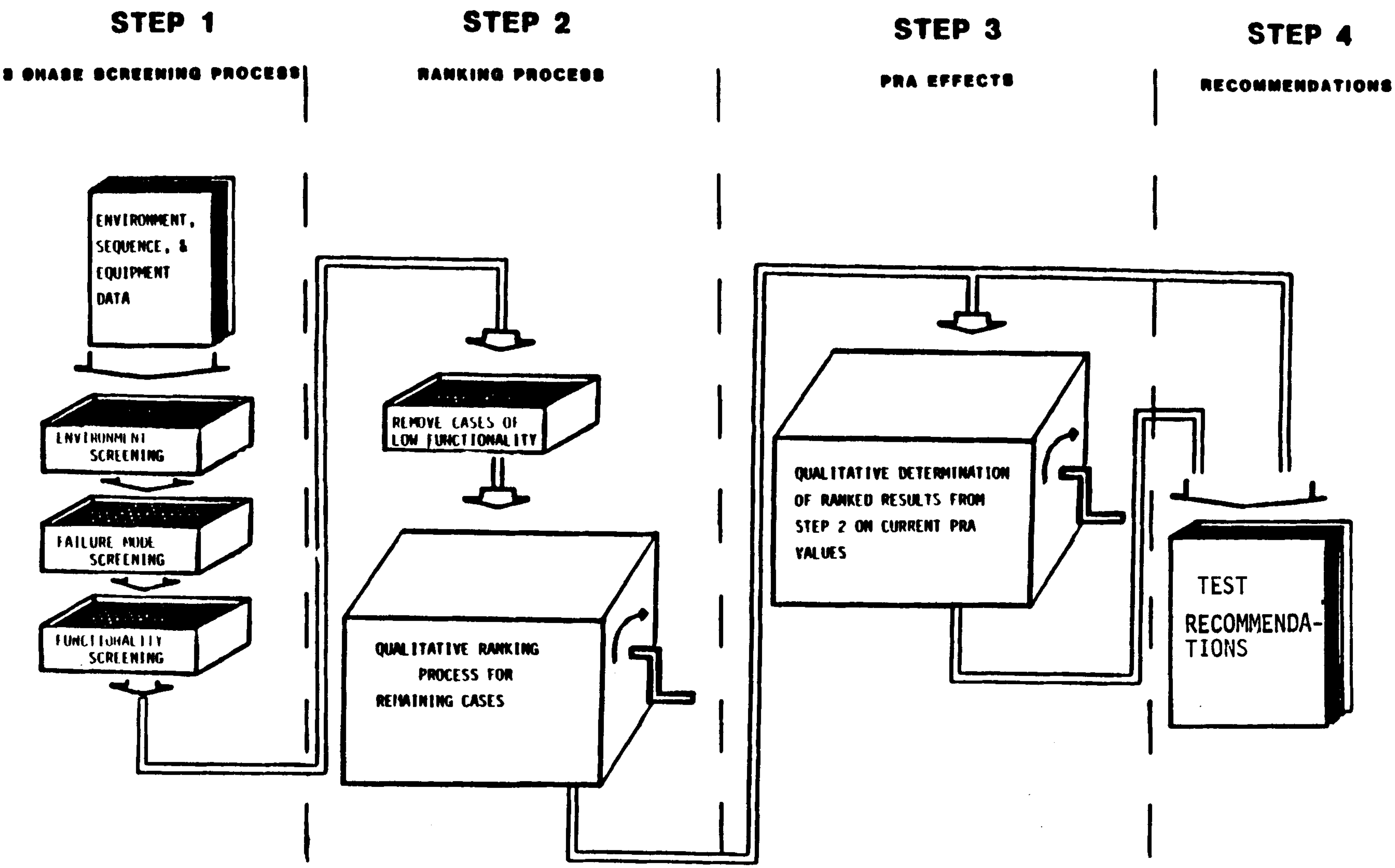


FIGURE D-1 - ANALYSIS METHODOLOGY OVERVIEW

1.3 Appendix Organization

This appendix is organized into seven sections.

- Section 1.0 is the introduction. It reviews Appendix A, B, and C, defines the analysis methodology, and presents the organization of the report.

- Section 2.0 defines and implements the three phase screening process which is step one in the analysis methodology.

- Section 3.0 presents the qualitative ranking of the three phase screening results. This is step two in the analysis methodology.

- Section 4.0 addresses the implications PEEESAS results may have on current PRA sequence probabilities. Where appropriate, qualitative estimates of changes in probabilities are made. This is the third step in the analysis methodology.

- Section 5.0 lists recommendations for equipment test based on the results of the first three steps in the analysis methodology. This is the fourth and final step in the analysis methodology process.

- Section 6.0 is a report summary reiterating key results from implementation of the analysis methodology.

- Section 7.0 is a list of references.

2.0 3 PHASE SCREENING PROCESS

2.1 Introduction

This section implements step 1 of the analysis methodology by screening current PEEESAS data for environment, failure modes, and functional considerations. Figure D-2 presents an overview of the process illustrating each of these three major phases. The results from this process serve as input to the ranking function (step 2) in the analysis methodology.

2.2 Phase 1 - Environmental/Qualification Profile Screening

2.2.1 Introduction

When the accident environmental profile exceeds the typical qualification profile for a given environment, a determination must be made concerning equipment survivability at the elevated environmental level. The screening tasks in this section are designed to identify the equipment and profiles where equipment survivability is questionable. The screening tasks consider the accident profile only up to the latter of vessel breach or containment failure. Although some actions may be taken after both vessel breach and containment failure, these actions would have less impact than actions taken before vessel breach and/or containment failure. The major focus is to investigate what equipment can be used before both of these failures occur. Figure D-3 illustrates the six screens used in phase 1 to sort the equipment and profiles. Note each of the screens is numbered for easy reference. The screening tasks are examined in four discrete steps described in subsequent paragraphs. The results of the screening process are then placed in a summary matrix to facilitate later ranking of the data.

2.2.2 Step One --- Profile Screens (Screens 1 & 2)

Appendix C (ref. 21) compared the projected environmental profiles to typical qualification profiles. Table D-1 summarizes the results for easy reference. The results, in Table D-1, are the input to Screen # 1 shown in Figure D-4. Screen 1 eliminates the accident scenarios where the environmental profile

3 PHASE SCREENING PROCESS

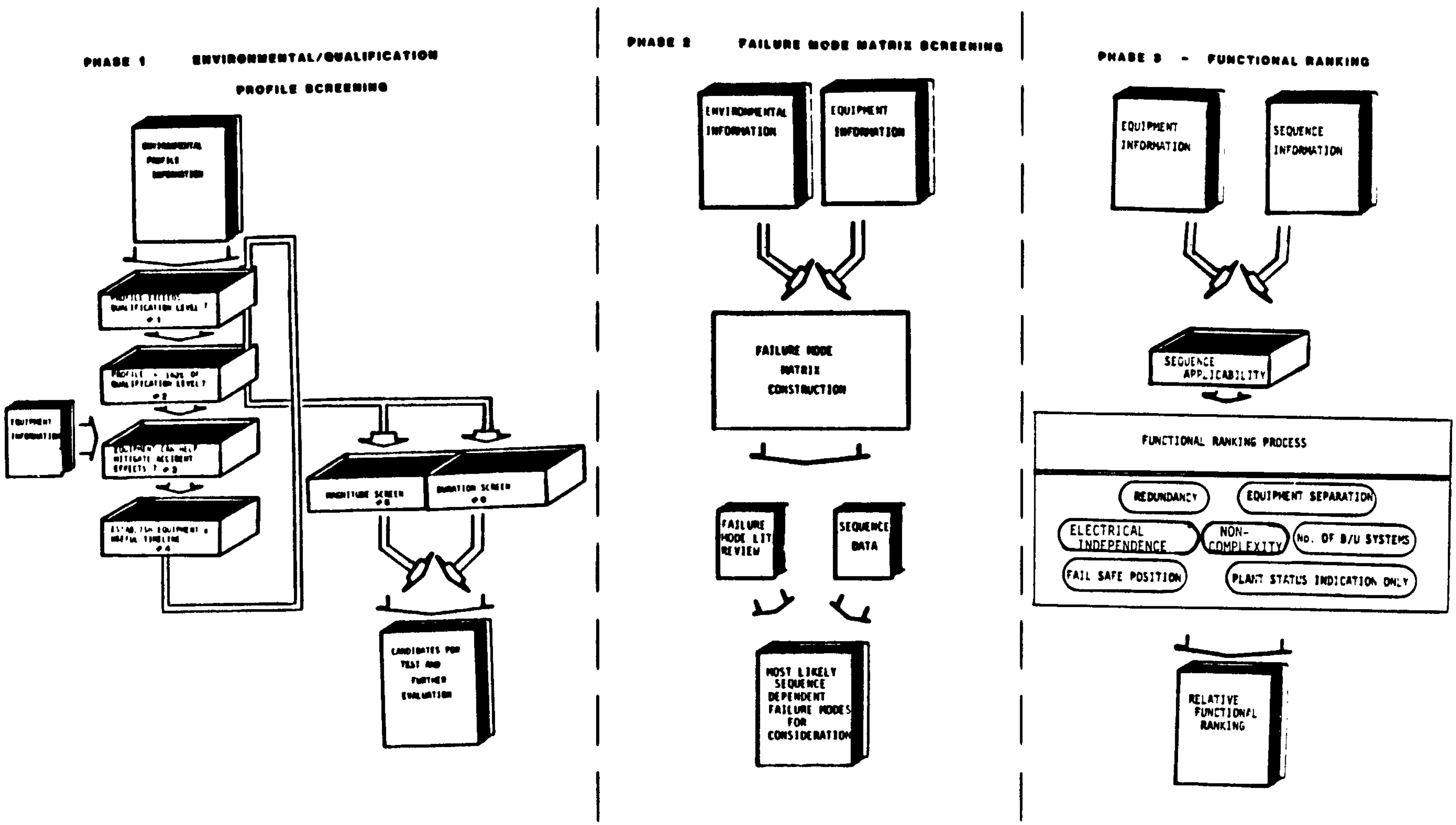


FIGURE D-2 - THREE PHASE SCREENING PROCESS OVERVIEW

PHASE 1 ENVIRONMENTAL/QUALIFICATION PROFILE SCREENING

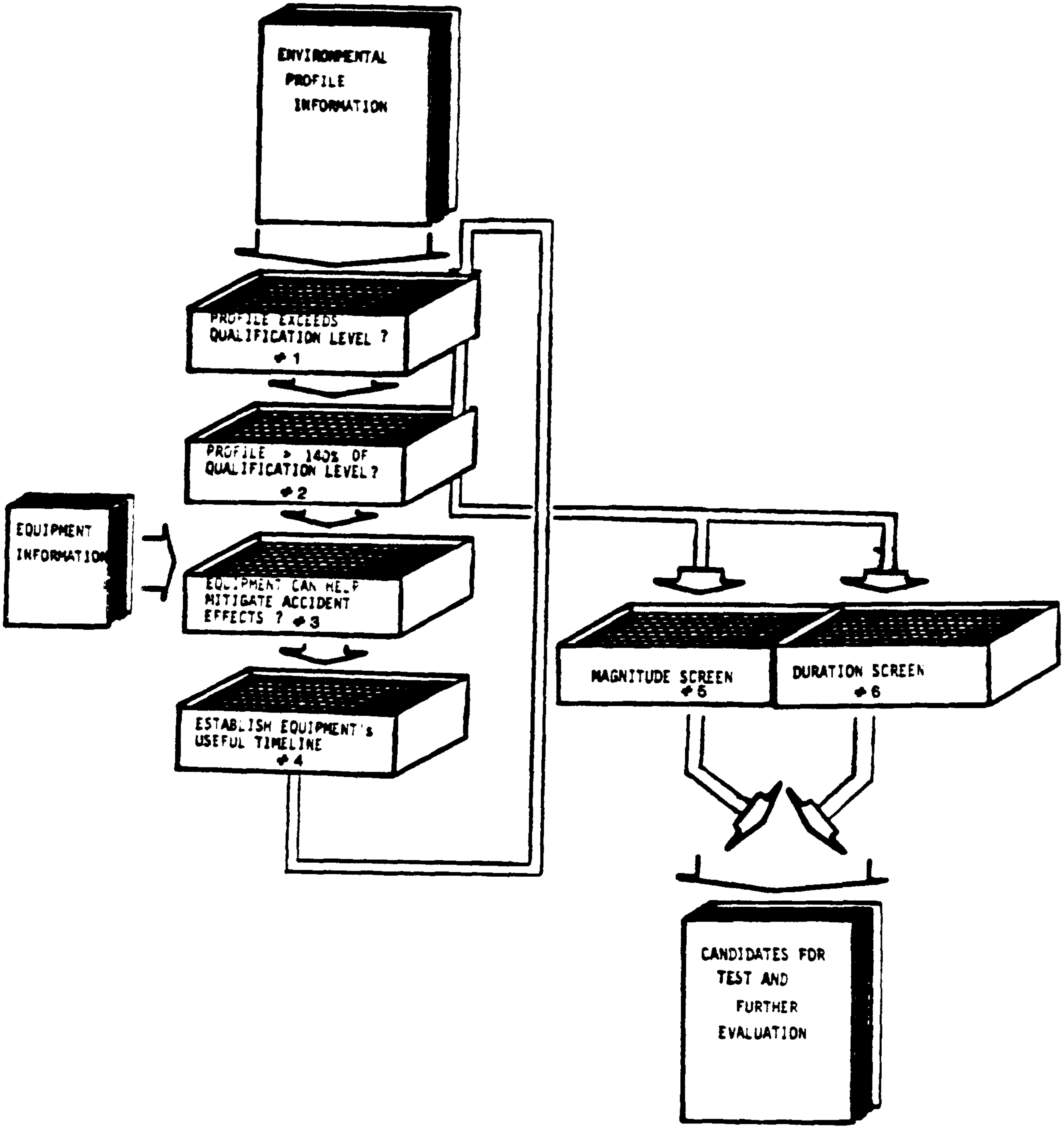


FIGURE D-3 - PHASE 1 PROFILE SCREENING FUNCTION

TABLE D-1 - RESULTS SUMMARY

PROFILE NO.	ACCIDENT	TIME OF VESSEL BREACH (1)	TIME OF CONT. FAIL (1)	ENVIR. VARIABLE	TIME ABOVE QUAL LVL (1)	PEAK AMP./%> QUAL LVL (2, 3)	TIME OF (1) PEAK AMP.	TIME FIRST (1) EXCEEDS QUAL. LVL
1	TB-Short No/late Oper. Action	2.1	3.2	D.W. Temp.	.20	500° / 142%	3.2	3.0
				S.P. Temp.	0	180° / 51%	3.2	---
				D.W. Press.	.2	100 / 118%	3.2	3.0
2	TB-Short Vessel Depress. Early	2.3	2.9	D.W. Temp.	.2	500° / 142%	2.9	2.7
				S.P. Temp.	0	170° / 49%	2.9	---
				D.W. Press.	0	75 / 88%	2.9	---
3	TB-Long	9.0	10.0	D.W. Temp.	2.0	500° / 142%	10.0	8.0
				S.P. Temp.	0	212° / 60%	10.0	---
				D.W. Press.	.5	100 / 118%	10.0	9.5
4	TW	39	35	D.W. Temp.	11	500° / 142%	39	28
				S.P. Temp.	8	350° / 100%	35	34
				D.W. Press.	7	125 / 147%	35	28
5	TC	3.8(6.7)*	0.9(3.9)*	D.W. Temp.	<.1 (-)*	360° / 103%(345°/98%)*	.9(3.9)*	.9(-)*
				S.P. Temp.	0 (-)*	340° / 97% (340°/97%)*	.9(3.9)*	---(-)*
				D.W. Press.	.75 (.5)*	132 / 155% (132°/155%)*	.9(3.9)*	.6(3.4)*
6	TQUV Vessel At Pressure	4.9	7.0	D.W. Temp.	.5	500° / 142%	7.0	6.5
				S.P. Temp.	0	170° / 49%	7.0	---
				D.W. Press.	.8	110 / 129%	7.0	6.2
7	TQUV-- Vessel Depress.	7.0	8.2	D.W. Temp.	.4	500° / 142%	8.2	7.8
				S.P. Temp.	0	200° / 57%	7.0	---
				D.W. Press.	.3	110 / 129%	8.2	7.9
8	AE	2.1	.66	D.W. Temp.	1.44	2000° / 571%	.7	.6
				S.P. Temp.	0	270° / 77%	.66	---
				D.W. Press.	.1	138 / 162%	.7	.6

Notes:

(1) All Times in Hours

(2) All Temperatures in °F/All Pressures in psia

(3) Percentage Based on Exceeding Qualification Profile Maximum Value

*Information in parenthesis is for MSIV open case.

PHASE 1 ENVIRONMENTAL/QUALIFICATION

PROFILE SCREENING

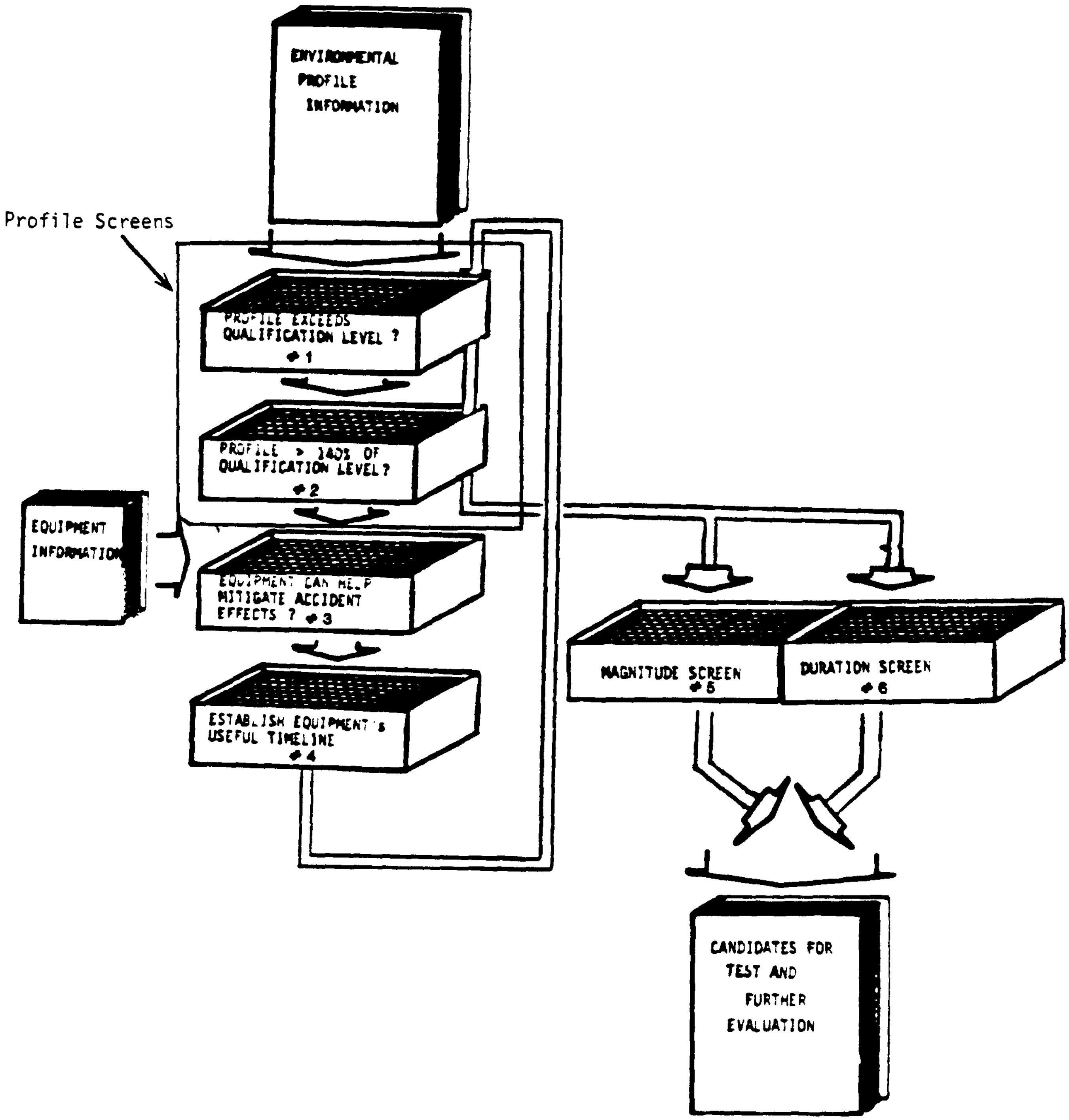


FIGURE D-4 - STEP ONE - PROFILE SCREENING does not exceed the typical qualification profile since the equipment should have been qualified to perform at the typical qualification profile levels. Table D-2 shows those profiles remaining after application of screen 1.

In order to differentiate between those profiles which are slightly above the qualification limits and those which definitely exceed the qualification limit, (Screen #2), the profiles which were 40% above the qualification profile were retained. Note that 500 degrees (where temperature is estimated to fail the containment) is 42% greater than the maximum qualification limit of 350 degrees. This means that every case where the environmental profile shows containment failure due to temperature will not be eliminated by screen #2. Table D-3 identifies those profiles which remain after applying the second screen. Application of these two screens reduces the 27 original profiles to 11 total profiles for consideration with the equipment screens.

2.2.3 Step Two --- Equipment Screens (Screens 3 & 4)

Appendix B focused on identifying equipment which could be important to accident mitigation. Table D-4 lists the equipment recommended for further analysis. The screens in this section are designed to couple equipment of interest with the sequences they influence. Figure D-5 highlights the two equipment screens described in this section. Screen 3 determines if a given piece of equipment is of possible use in an accident sequence. Reference 20 provides a detailed account of how each piece of equipment might help mitigate a given accident sequence or provide important plant status information during the accident. Not all of the equipment shown in Table D-4 can help reduce the effect of a given accident sequence. For example, a Safety Relief Valve (SRV) is not required during an AE sequence since the vessel is depressurized anyway. Therefore, determining the survivability of an SRV actuation assembly for the environments encountered during an AE sequence is not needed. On the other hand, determining the survivability of an SRV actuation assembly for the environments predicted in a TQUV sequence (when low pressure is required to maximize CRD flow or restore low pressure cooling) would be useful. Table D-5 presents the results of applying Screen 3 to the equipment of interest.

TABLE D-2 - SCREEN 1 SUMMARY RESULTS

PROFILE NO.	ACCIDENT	TIME OF VESSEL BREACH (1)	TIME OF CONT. FAIL (1)	ENVIR. VARIABLE	TIME ABOVE QUAL LVL (1)	PEAK AMP./%> QUAL LVL (2, 3)	TIME OF (1) PEAK AMP.	TIME FIRST (1) EXCEEDS QUAL. LVL
1	TB-Short No/late Oper. Action	2.1	3.2	D.W. Temp.	.20	500^{o} / 142%	3.2	3.0
				S.P. Temp.	*1			
				D.W. Press.	.2	100 / 118%	3.2	3.0
2	TB-Short Vessel Depress. Early	2.3	2.9	D.W. Temp.	.2	500^{o} / 142%	2.9	2.7
				S.P. Temp.	*1			
				D.W. Press.	*1			
3	TB-Long	9.0	10.0	D.W. Temp.	2.0	500^{o} / 142%	10.0	8.0
				S.P. Temp.	*1			
				D.W. Press.	.5	100 / 118%	10.0	9.5
4	TW	39	35	D.W. Temp.	11	500^{o} / 142%	39	28
				S.P. Temp.	8	350^{o} / 100%	35	34
				D.W. Press.	7	125 / 147%	35	28
5	TC	3.8(6.7)*	0.9(3.9)*	D.W. Temp.	<.1 (*1)*	360^{o} / 103%( )	.9( )	.9( )
				S.P. Temp.	*1 (*1)*			
				D.W. Press.	.75(.5)*	132 / 155%(132/155%)*	.9(3.9)*	.6(3.4)*
6	TQUV Vessel At Pressure	4.9	7.0	D.W. Temp.	.5	500^{o} / 142%	7.0	6.5
				S.P. Temp.	*1			
				D.W. Press.	.8	110 / 129%	7.0	6.2
7	TQUV-- Vessel Depress.	7.0	8.2	D.W. Temp.	.4	500^{o} / 142%	8.2	7.8
				S.P. Temp.	*1			
				D.W. Press.	.3	110 / 129%	8.2	7.9
8	AE	2.1	.66	D.W. Temp.	1.44	2000^{o} / 571%	.7	.6
				S.P. Temp.	*1			
				D.W. Press.	.1	138 / 162%	.7	.6

Notes:

(1) All Times in Hours
(2) All Temperatures in ^{o}F/All Pressures in psia
(3) Percentage Based on Exceeding Qualification Profile Maximum Value

*1 - Indicates Those Profiles Removed by Screen #1.

* - Information in parenthesis is for MSIV open case.

TABLE D-3 - SCREEN 2 SUMMARY RESULTS

PROFILE NO.	ACCIDENT	TIME OF VESSEL BREACH (1)	TIME OF CONT. FAIL (1)	ENVIR. VARIABLE	TIME ABOVE QUAL LVL (1)	PEAK AMP./%> QUAL LVL (2, 3)	TIME OF (1) PEAK AMP.	TIME FIRST (1) EXCEEDS QUAL. LVL
1	TB-Short No/late Oper. Action	2.1	3.2	D.W. Temp.	.20	500^{o} / 142%	3.2	3.0
				S.P. Temp.	*1			
				D.W. Press.	*2			
2	TB-Short Vessel Depress. Early	2.3	2.9	D.W. Temp.	.2	500^{o} / 142%	2.9	2.7
				S.P. Temp.	*1			
				D.W. Press.	*1			
3	TB-Long	9.0	10.0	D.W. Temp.	2.0	500^{o} / 142%	10.0	8.0
				S.P. Temp.	*1			
				D.W. Press.	*2			
4	TW	39	35	D.W. Temp.	11	500^{o} / 142%	39	28
				S.P. Temp.	*2			
				D.W. Press.	7	125 / 147%	35	28
5	TC	3.8(6.7)*	0.9(3.9)*	D.W. Temp.	*2 (*1)*			
				S.P. Temp.	*1 (*1)*			
				D.W. Press.	.75(.5)*	132/155%(132/155%)*	.9(3.9)*	.6(.6)*
6	TQUV Vessel At Pressure	4.9	7.0	D.W. Temp.	.5	500^{o} / 142%	7.0	6.5
				S.P. Temp.	*1			
				D.W. Press.	*2			
7	TQUV-- Vessel Depress.	7.0	8.2	D.W. Temp.	.4	500^{o} / 142%	8.2	7.8
				S.P. Temp.	*1			
				D.W. Press.	*2			
8	AE	2.1	.66	D.W. Temp.	1.44	2000^{o} / 571%	.7	.6
				S.P. Temp.	*1			
				D.W. Press.	.1	138 / 162%	.7	.6

Notes:

(1) All Times in Hours
(2) All Temperatures in ^{o}F/All Pressures in psia
(3) Percentage Based on Exceeding Qualification Profile Maximum Value

*1 Indicates Those Profiles Removed by Screen #1
*2 Indicates Those Profiles Removed by Screen #2
* Indicates Values for TC5A (MSIV Open) Case

TABLE D-4 - EQUIPMENT RECOMMENDED FOR FURTHER EXAMINATION BY THE PEEESAS PROGRAM

COMPONENT	SEQUENCE*	TIME PERIOD*	USEFULNESS
Inboard MSIV Solenoid Valves	TW, TC TQUV, TB (after AC restored)	- Containment failure - Vessel breach	Reopening of MSIVs will restore a heat rejection path to avoid containment failure in TW or TC and possibly a core melt in TQUV or TB (if feedwater is also supplied).
HPCI, RCIC Inboard Isolation Valves	TW, TC, TQUV, TB (after AC restored)	Vessel breach	Reopening or sustained opening of the valves provides core cooling and hence prevents core melt.
Pilot Valves and Service Air Solenoid Valves for SRVs	TW, TC, TQUV, TB (after AC restored)	Vessel breach	Sustained functionality allows for low reactor vessel pressure operation to sustain or restore low pressure injection to the core, thus preventing core melt.
RHR Inboard Shutdown Cooling Valve	TW, TC	Containment failure	Opening of this valve to provide a RHR cooling path could be important in preventing containment overpressurization and failure.
In-Core Thermocouples and Reactor Vessel Surface Thermocouples	TW, TC, TQUV, TB, AE	Vessel breach	Provide status of core cooling adequacy.
Drywell Temperature Element (RTD)	TB, TQUV, TW	Containment failure	Provides status of D.W. cooler operation and an indication of margin to containment failure.
Drywell Pressure Monitor	TB, TQUV, TW	Containment failure	Provides indication of containment venting effectiveness and margin to containment failure.
Drywell N_2 and Radiation Monitor	TB, TW, TC, TQUV, AE	Beyond containment failure	Provides estimate of core condition and release of fission products.
Cabling, Connectors/Splices, Terminal Blocks for Above Components	See Above	See Above	See Above

*Listed are those sequences for which survivability of the component could be most important for mitigating the accident or providing plant status information up to the time period indicated.

PHASE 1 ENVIRONMENTAL/QUALIFICATION PROFILE SCREENING

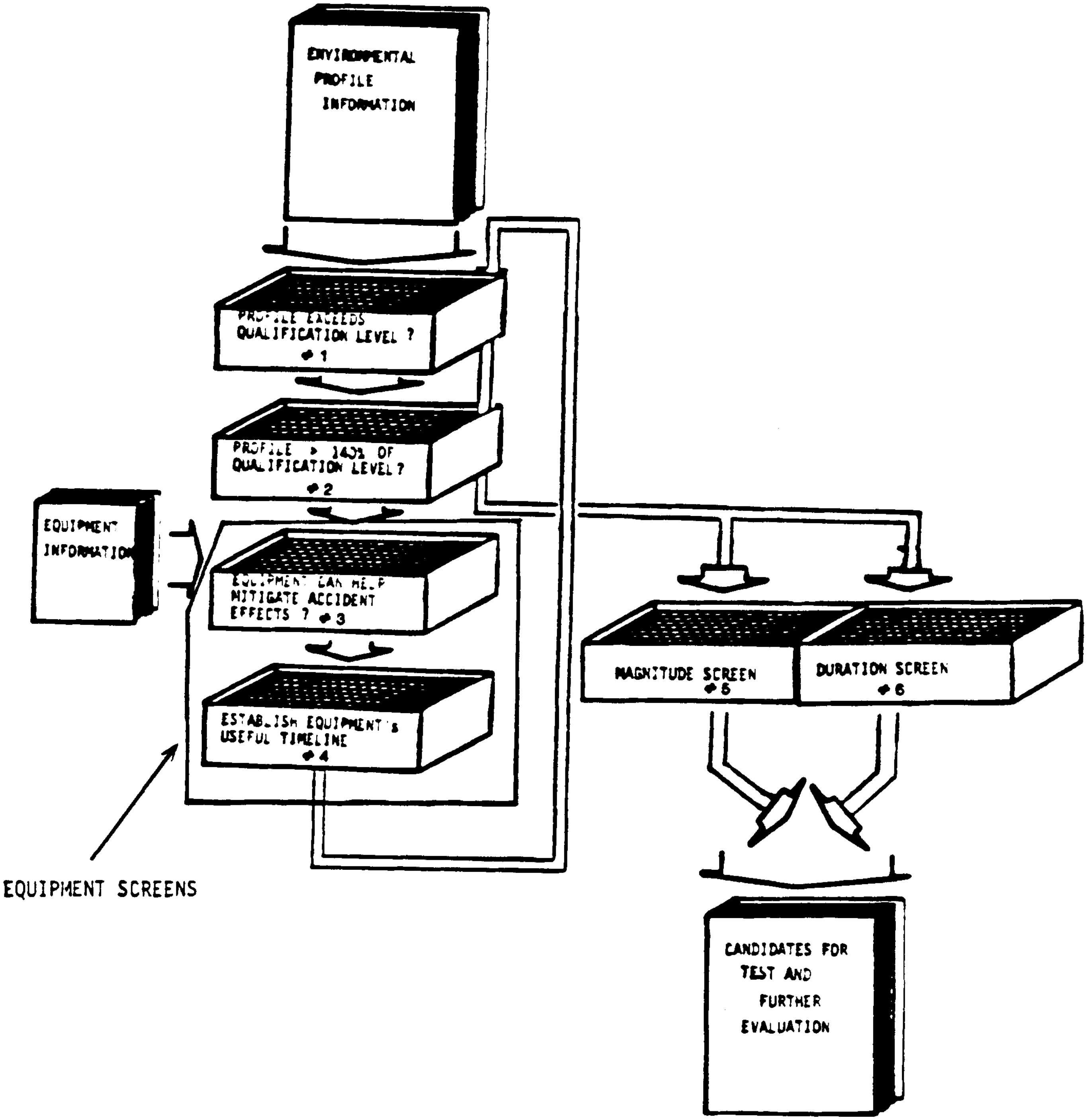


FIGURE D-5 - STEP TWO - EQUIPMENT SCREENING

TABLE D-5 - SCREEN 3 SUMMARY RESULTS

PROFILE	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMOCOUPLE	DRYWELL TEMP. RTD	DRYWELL PRESSURE	H_2/RADIATION MONITOR
TB1 - D.W. TEMP.	YES	YES*	NO	YES	YES	YES	YES	YES
TB2 - D.W. TEMP.	YES	YES*	NO	YES	YES	YES	YES	YES
TB3 - D.W. TEMP.	YES	YES	NO	YES	YES	YES	YES	YES
TW4 - D.W. TEMP.	YES	YES	YES	YES	YES	YES	YES	YES
TW4 - D.W. PRESS.	YES	YES	YES	YES	YES	YES	YES	YES
TC5 - D.W. PRESS.	YES(YES)**	YES(YES)**	YES(YES)**	YES(YES)**	YES(YES)**	NO(NO)**	NO(NO)**	YES(YES)**
TQUV6 - D.W. TEMP.	YES	YES*	NO	YES	YES	YES	YES	YES
TQUV7 - D.W. TEMP.	YES	YES*	NO	YES	YES	YES	YES	YES
AE8 - D.W. TEMP.	NO	NO	NO	NO	YES	NO	NO	YES
AE8 - D.W. PRESS.	NO	NO	NO	NO	YES	NO	NO	YES

* If system failures can be overcome before Vessel Breach
** Indicates TC5A (MSIV Open) Results

Screen 4 is a timeline consideration. It determines when any potential corrective action for a given sequence would no longer be of any use. For example, the ability to operate an SRV past the point of vessel breach is useless since the vessel depressurizes when the breach occurs. Therefore, a piece of equipment should only be tested to the highest level expected within its "useful" timeline. Table D-6 shows the results of this screen as applied to all equipment and profiles of interest.

2.2.4 Step Three --- Profile Screens Repeated (Screens 1A & 2A)

The next step in the screening process takes the coupled results from the first 4 screens shown in Figure D-3 and refilters them through screens 1 and 2. This step is illustrated in Figure D-6. The reapplication of these screens is beneficial because now that a piece of equipment's "useful" timeline has been determined, some equipment may never see an environmental level in excess of typical qualification levels until after its usefulness has expired. An example may help to explain this process. Suppose it is desired to know if the MSIV solenoid assembly would survive the drywell temperature environment predicted for the short term blackout sequence described in profile set number 1 (short term blackout with no operator action). Figure D-7, taken from Appendix C, shows the results of comparing the projected drywell temperature for this sequence against a typical temperature qualification profile. Note that this profile was not eliminated by screen number 1 the first time through since the sequence does exceed maximum qualification temperature levels. But now this profile can be coupled with a specific piece of equipment and be reexamined. Screen 3 said that the ability to reopen the MSIVs could help mitigate the effects of this accident (Ref. 20). Screen 4 said that the ability to open these valves could help only up to the time of vessel breach. With these two facts in mind screen 1 is reapplied. Now it can be seen that before the time of vessel breach (end of "useful" time) maximum predicted drywell temperature in Figure D-5 is only 220^{o}F. This is well below the maximum typical qualification level of 350^{o}F and thus the MSIV solenoid is removed from further consideration for the drywell temperature environment in the short term TB sequence. Table D-7 shows the results of reapplication of screen number 1 under full equipment and profile considerations.

TABLE D-6 - SCREEN 4 SUMMARY RESULTS

PROFILE	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMOCOUPLE	DRYWELL TEMP. RTD	DRYWELL PRESSURE	H_2/RADIATION MONITOR
TB1 - D.W. TEMP.	V.B.	V.B.	*3	V.B.	V.B.	CONT.FAIL.	CONT.FAIL.	CONT.FAIL.
TB2 - D.W. TEMP.	V.B.	V.B.	*3	V.B.	V.B.	CONT.FAIL.	CONT.FAIL.	CONT.FAIL.
TB3 - D.W. TEMP.	V.B.	V.B.	*3	V.B.	V.B.	CONT.FAIL.	CONT.FAIL.	CONT.FAIL.
TW4 - D.W. TEMP.	CONT.FAIL.	V.B.	CONT.FAIL.	V.B.	V.B.	CONT.FAIL.	CONT.FAIL.	V.B.
TW4 - D.W. PRESS.	CONT.FAIL.	V.B.	CONT.FAIL.	V.B.	V.B.	CONT.FAIL.	CONT.FAIL.	V.B.
TC5 - D.W. PRESS.	CONT.FAIL.(C.F.)**	V.B.(V.B.)**	CONT-FAIL.(C.F.)**	V.B.(V.B.)**	V.B.(V.B.)**	*3(*3)**	*3(*3)**	V.B.(V.B.)**
TQUV6 - D.W. TEMP.	V.B.	V.B.	*3	V.B.	V.B.	CONT.FAIL.	CONT.FAIL.	CONT.FAIL.
TQUV7 - D.W. TEMP.	V.B.	V.B.	*3	V.B.	V.B.	CONT.FAIL.	CONT.FAIL.	CONT.FAIL.
AE8 - D.W. TEMP.	*3	*3	*3	*3	V.B.	*3	*3	V.B.
AE8 - D.W. PRESS.	*3	*3	*3	*3	V.B.	*3	*3	V.B.

Key: V.B. = Vessel Breach Cont.Fail. = Containment Failure

*3 Indicates Equipment Coupled to a Specific Profile that was Removed by Screen #3
** Indicates TC5A (MSIV Open) Results

PHASE 1 ENVIRONMENTAL/QUALIFICATION PROFILE SCREENING

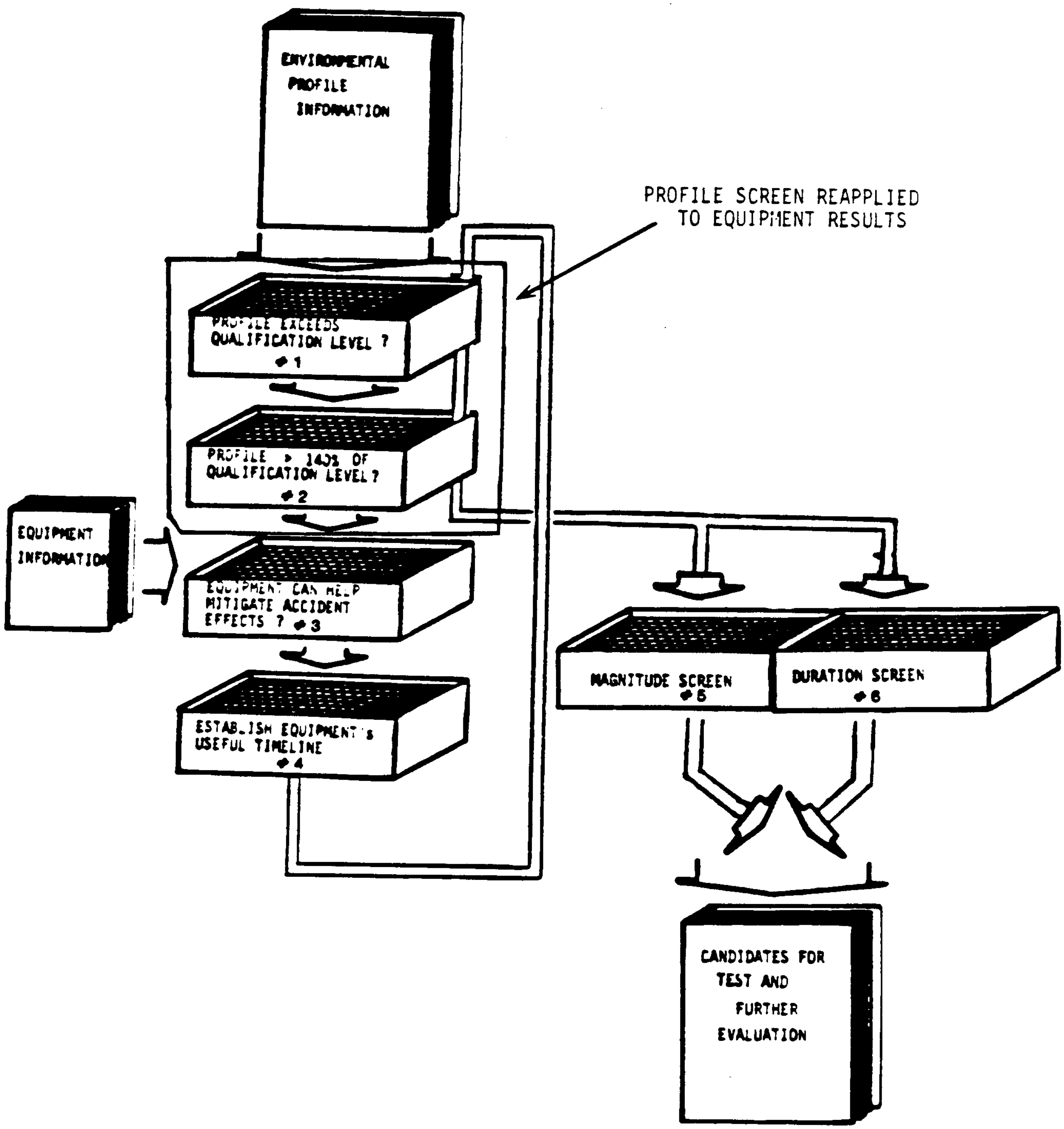


FIGURE D-6 - STEP THREE - REAPPLICATION OF PROFILE SCREENS TO EQUIPMENT RESULTS

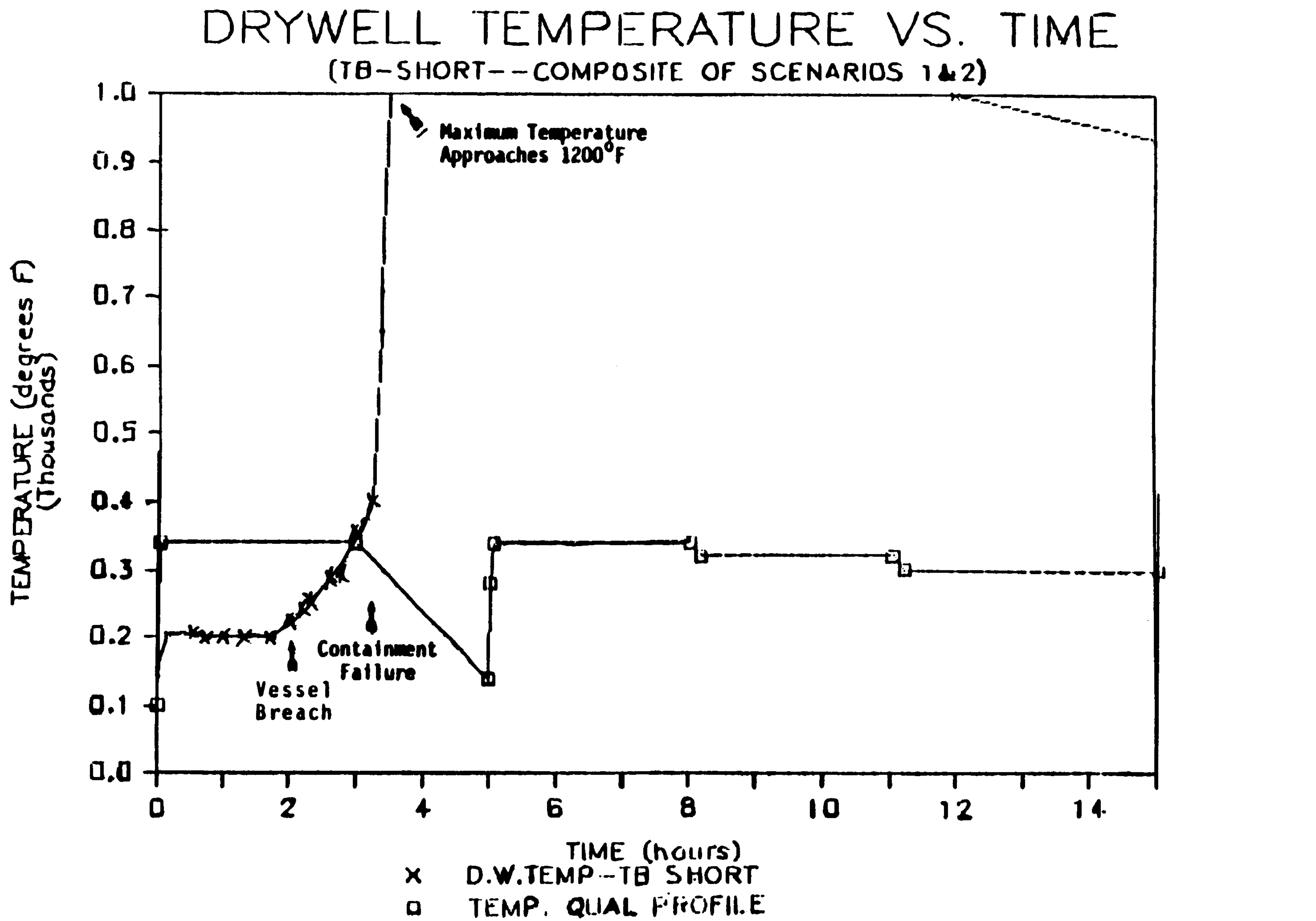


FIGURE D-7 - DRYWELL TEMPERATURE COMPARISON TO QUALIFICATION LEVELS FOR SHORT TERM TB SEQUENCE (SCENARIO 2)

TABLE D-7 - SCREEN 1A SUMMARY RESULTS

PROFILE	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMOCOUPLE	DRYWELL TEMP. RTD	DRYWELL PRESSURE	H_2/RADIATION MONITOR
TB1 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	142% .2hrs / 500°	142% .2hrs / 500°	142% .2hrs / 500°
TB2 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	142% .2hrs / 500°	142% .2hrs / 500°	142% .2hrs / 500°
TB3 - D.W. TEMP.	120% 1.5hrs / 420°	120% 1.5hrs / 420°	*3	120% 1.5hrs / 420°	120% 1.5hrs / 420°	142% 2.0hrs / 500°	142% 2.0hrs / 500°	142% 2.0hrs / 500°
TW4 - D.W. TEMP.	120% 7hrs / 420°	142% 11hrs / 500°	120% 7hrs / 420°	142% 11hrs / 500°	142% 11hrs / 500°	120% 7hrs / 420°	120% 7hrs / 420°	142% 11hrs / 500°
TW4 - D.W. PRESS.	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#
TC5 - D.W. PRESS.	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]	*3 (*3)[7]	*3 (*3)[7]	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]
TQUV6 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	142% .4hrs / 500°	142% .4hrs / 500°	142% .4hrs/ 500°
TQUV7 - D.W.	*1A	*1A	*3	*1A	*1A	142% .4hrs / 500°	142% .4hrs / 500°	142% .4hrs / 500°
AE8 - D.W. TEMP.	*3	*3	*3	*3	571% 1.4hrs / 2000°	*3	*3	571% 1.4hrs / 2000°
AE8 - D.W. PRESS.	*3	*3	*3	*3	162% .1hr / 138#	*3	*3	162% .1hr / 138#

Notes:

(1) All Percentages Indicate Projected environmental Level Relative to the Maximum Qualification Level
(2) All Times Indicate How Long the Projected Environment is in Excess of Maximum Qualification Level
(3) Listed Temperatures (°F) and Pressures (PSIA) Indicate Maximum Projected Environmental Level
(4) *B Indicates Equipment Coupled to a Specific Profile that was Removed by Screen #3
(5) *1A Indicates Equipment Coupled to a Specific Profile that was Removed by Reapplication of Screen #1
(6) *3 Indicates Equipment Coupled to a Specific Profile that was Removed by Screen #3.
(7) Indicates TC5A (MSIV Open) Results

The next screen is a reapplication of screen number 2 accounting for equipment considerations. It identifies that equipment which will see environmental levels of at least 40% above the typical qualification levels during the equipments' "useful" period. Table D-8 shows the results of this screen. The result of the profile and equipment screening process gives 37 possible test cases.

2.2.5 Step Four --- Magnitude and Duration Screens (Screens 5 & 6)

Table D-8 results indicate a large number of profiles to consider and a further reduction in the number would be beneficial. The number of test cases in Table D-8 can be reduced without excluding any profiles which should be considered. For each piece of equipment, the profile with the maximum environmental level and the profile with the maximum time above qualification levels should be retained. All other profiles should fall within these limits. Therefore, only those profiles which predict a maximum amplitude or duration of a particular environment, for each piece of equipment, need be carried forward as possible test and analysis candidates. This reduction is achieved by using screens 5 and 6 as shown in Figure D-8. Screen 5 looks for those profiles which produce the maximum drywell temperature or pressure for each piece of critical equipment. The results of this screen are presented in Table D-9. Screen 6 looks for those profiles with the longest duration above the qualification maximums for a given piece of equipment and environment. Table D-10 shows the results of this screen. Whether the magnitude or duration is more severe depends on the profile and the equipment. Therefore both are carried forward. Thus Table D-9 and D-10 results are combined to form Table D-11 which lists those profiles judged to be most severe.

2.2.6 Summary Matrix

The six screens employed to this point have been used to identify the most severe profiles. The results of this screening process are now placed in a summary matrix for each of the 8 major equipment assemblies in order to understand relative sequence and environment comparisons. Tables D-12 through D-19 show the summary matrix for each piece of equipment. The matrix is constructed by looking at the results of the environmental/qualification

TABLE D-8 - SCREEN 2A SUMMARY RESULTS

PROFILE	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMOCOUPLE	DRYWELL TEMP. RTD	DRYWELL PRESSURE	H_2/RADIATION MONITOR
TB1 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	142% .2hrs / 500°	142% .2hrs / 500°	142% .2hrs / 500°
TB2 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	142% .2hrs / 500°	142% .2hrs / 500°	142% .2hrs / 500°
TB3 - D.W. TEMP.	*2A	*2A	*3	*2A	*2A	142% 2.0hrs / 500°	142% 2.0hrs / 500°	142% 2.0hrs / 500°
TW4 - D.W. TEMP.	*2A	142% 11hrs / 500°	*2A	142% 11hrs / 500°	142% 11hrs / 500°	*2A	*2A	142% 11hrs / 500°
TW4 - D.W. PRESS.	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#
TC5 - D.W. PRESS.	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]	*3 (*3)[7]	*3 (*3)[7]	155%(155%)[7] .75hrs / 132#[7] (.5hrs / 132#)[7]
TQUV6 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	142% .4hrs / 500°	142% .4hrs / 500°	142% .4hrs/ 500°
TQUV7 - D.W.	*1A	*1A	*3	*1A	*1A	142% .4hrs / 500°	142% .4hrs / 500°	142% .4hrs / 500°
AE8 - D.W. TEMP.	*3	*3	*3	*3	571% 1.4hrs / 2000°	*3	*3	571% 1.4hrs / 2000°
AE8 - D.W. PRESS.	*3	*3	*3	*3	162% .1hr / 138#	*3	*3	162% .1hr / 138#

Notes:

(1) All Percentages Indicate Projected Environmental Level Relative to Maximum Qualification Level.
(2) All Times Indicate How Long the Projected Environment is in Excess of Maximum Qualification Level.
(3) Listed Temperatures (°F) and Pressures (PSIA) Indicate Maximum Projected Environmental Level.
(4) *3 Indicates Equipment Coupled to a Specific Profile that was Removed by Screen #3.
(5) *1A Indicates Equipment Coupled to a Specific Profile that was Removed by Screen #1.
(6) *2A Indicates Equipment Coupled to a Specific Profile that was Removed by Reapplication of Screen #2.
(7) Indicates TC5A (MSIV Open) Results

PHASE 1 ENVIRONMENTAL/QUALIFICATION

PROFILE SCREENING

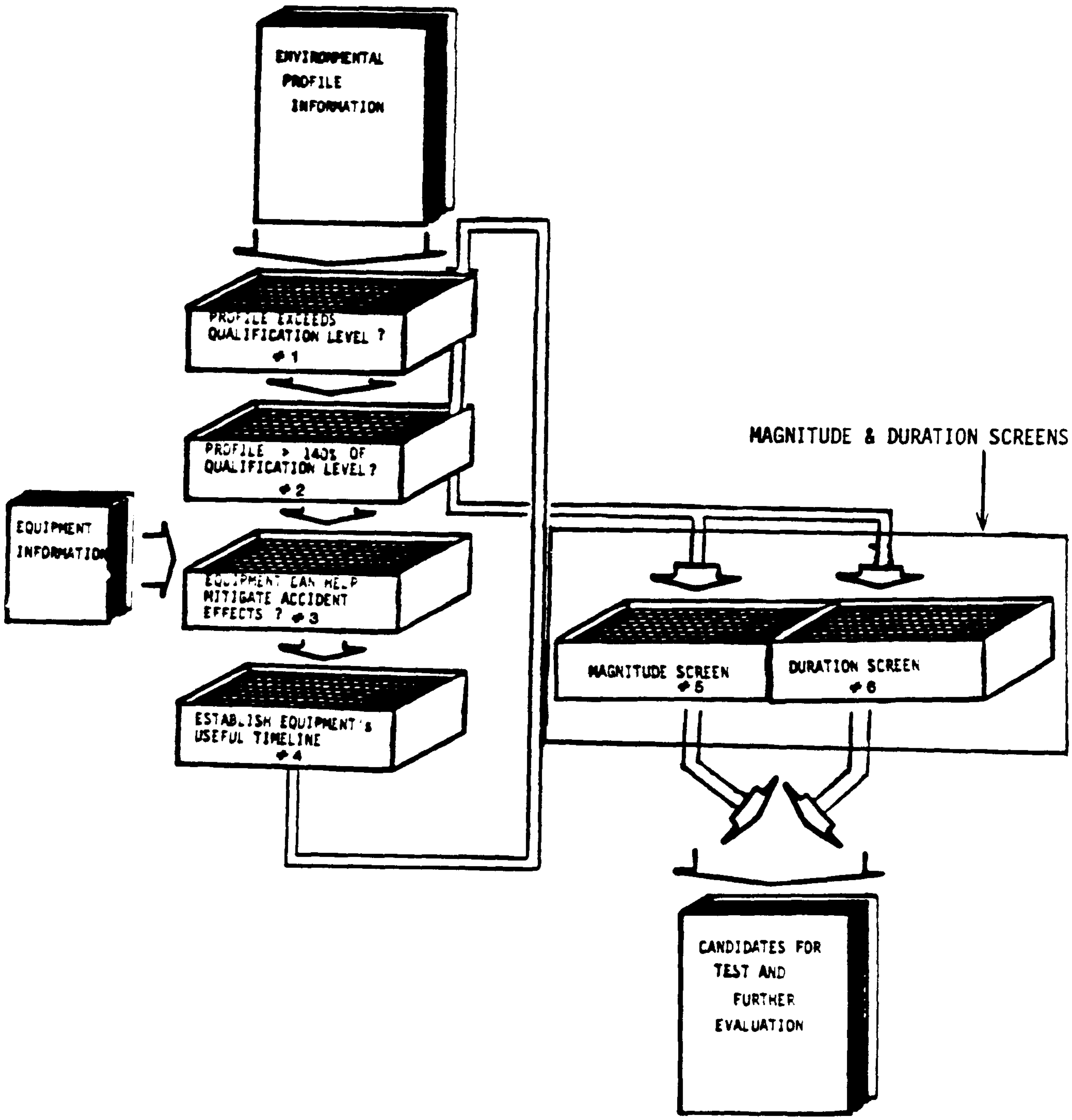


FIGURE D-8 - STEP FOUR - PROFILE MAGNITUDE & DURATION SCREENS

TABLE D-9 - SCREEN 5 SUMMARY RESULTS

PROFILE	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMOCOUPLE	DRYWELL TEMP. RTD	DRYWELL PRESSURE	H_2/RADIATION MONITOR
TB1 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*5	*5	*5
TB2 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*5	*5	*5
TB3 - D.W. TEMP.	*2A	*2A	*3	*2A	*2A	142% 2.0hrs / 500°	142% 2.0hrs / 500°	*5
TW4 - D.W. TEMP.	*2A	142% 11hrs / 500°	*2A	142% 11hrs / 500°	*5	*2A	*2A	*5
TW4 - D.W. PRESS.	*5	*5	*5	*5	*5	147% 7hrs / 125#	147% 7hrs / 125#	*5
TC5 - D.W. PRESS.	155%(155%)[8] .75hrs / 132# (.5hrs / 132#)[8]	155%(155%)[8] .75hrs / 132# (.5hrs / 132#)[8]	155%(155%)[8] .75hrs / 132# (.5hrs / 132#)[8]	155%(155%)[8] .75hrs / 132# (.5hrs / 132#)[8]	*5 (*5)[8]	*3 (*3)[8]	*3 (*3)[8]	*5 (*5)[8]
TQUV6 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*5	*5	*5
TQUV7 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*5	*5	*5
AE8 - D.W. TEMP.	*3	*3	*3	*3	571% 1.4hrs / 2000°	*3	*3	571% 1.4hrs / 2000°
AE8 - D.W. PRESS.	*3	*3	*3	*3	162% .1hr / 138#	*3	*3	162% .1hr / 138#

Notes:

(1) All Percentages Indicate Projected Environmental Level Relative to the Maximum Qualification Level
(2) All Times Indicate How Long the Projected Environment is in Excess of Maximum Qualification Level
(3) Listed Temperatures (°F) and Pressures (PSIA) Indicate Maximum Projected Environmental Level
(4) *3 Indicates Equipment Coupled to a Specific Profile that was Removed by Screen #3
(5) *1A Indicates Equipment Coupled to a Specific Profile that was Removed by Reapplication of Screen #1
(6) *2A Indicates Equipment Coupled to a Speiciffi Profile that was Removed By Reapplication of Screen 2
(7) *5 Indicates Equipment Coupled to a Specific Profile that was Removed by Screen 5
(8) Indicates TC5A (MSIV Open) Results

TABLE D-10 - SCREEN 6 SUMMARY RESULTS

PROFILE	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMOCOUPLE	DRYWELL TEMP. RTD	DRYWELL PRESSURE	H_2/RADIATION MONITOR
TB1 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*6	*6	*6
TB2 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*6	*6	*6
TB3 - D.W. TEMP.	*2A	*2A	*3	*2A	*2A	*6	*6	*6
TW4 - D.W. TEMP.	*2A	142% 11hrs / 500°	*2A	142% 11hrs / 500°	142% 11hrs / 500°	*2A	*2A	142% 11hrs / 500°
TW4 - D.W. PRESS.	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#
TC5 - D.W. PRESS.	*6 (*6)[8]	*6 (*6)[8]	*6 (*6)[8]	*6 (*6)[8]	*6 (*6)[8]	*3 (*6)[8]	*3 (*6)[8]	*6 (*6)[8]
TQUV6 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*6	*6	*6
TQUV7 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*6	*6	*6
AE8 - D.W. TEMP.	*3	*3	*3	*3	*6	*3	*3	*6
AE8 - D.W. PRESS.	*3	*3	*3	*3	*6	*3	*3	*6

Notes:

(1) All Percentages Indicate Projected Environmental Level Relative to the Maximum Qualification Level
(2) All Times Indicate How Long the Projected Environment is in Excess of Maximum Qualification Level
(3) Listed Temperatures (°F) and Pressures (PSIA) Indicate Maximum Projected Environmental Level
(4) *3 Indicates Equipment Coupled to a Specific Profile that was Removed by Screen #3
(5) *1A Indicates Equipment Coupled to a Specific Profile that was Removed by Reapplication of Screen #1
(6) *2A Indicates Equipment Coupled to a Specific Profile that was Removed by Reapplication of Screen #2
(7) *6 Indicates Equipment Coupled to a Specific Profile that was Removed by Screen #6
(8) Indicates TC5A (MSIV Open) Results

TABLE D-11 - COMBINED PROFILE/EQUIPMENT SCREENING RESULTS

PROFILE	MSIV	HPCI/RCIC	RHR	SRV PILOT VALVE	INCORE THERMOCOUPLE	DRYWELL TEMP. RTD	DRYWELL PRESSURE	H_2/RADIATION MONITOR
TB1 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*5&6	*5&6	*5&6
TB2 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*5&6	*5&6	*5&6
TB3 - D.W. TEMP.	*2A	*2A	*3	*2A	*2A	142% 2.0hrs / 500°	142% 2.0hrs / 500°	*5&6
TW4 - D.W. TEMP.	*2A	142% 11hrs / 500°	*2A	142% 11hrs / 500°	142% 11hrs / 500°	*2A	*2A	142% 11hrs / 500°
TW4 - D.W. PRESS.	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#	147% 7hrs / 125#
TC5 - D.W. PRESS.	155% .75hrs / 132# ( )[8]	155% .75hrs / 132# ( )[8]	155% .75hrs / 132# ( )[8]	155% .75hrs/ 132# ( )[8]	*5&6 (*5&6)[8]	*3 (*3)[8]	*3 (*3)[8]	*5&6 (*5&6)[8]
TQUV6 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*5&6	*5&6	*5&6
TQUV7 - D.W. TEMP.	*1A	*1A	*3	*1A	*1A	*5&6	*5&6	*5&6
AE8 - D.W. TEMP.	*3	*3	*3	*3	571% 1.4hrs / 2000°	*3	*3	571% 1.4hrs / 2000°
AE8 - D.W. PRESS.	*3	*3	*3	*3	162% .1hr / 138#	*3	*3	162% .1hr / 138#

Notes:

(1) All Percentages Indicate Projecte Environmental Level Relative to the Maximum Qualification Level
(2) All Times Indicate How Long the Projected Environment is in Excess of Maximum Qualification Level
(3) Listed Temperatures (°F) and Pressures (PSIA) Indicate Maximum Projected Environmental Level
(4) *3 Indicates Equipment Coupled to a Specific Profile that was Removed by Screen #3
(5) *1A Indicates Equipment Coupled to a Specific Profile that was Removed by Reapplication of Screen #1
(6) *2A Indicates Equipment Coupled to a Specific Profile that was Removed by Reapplication of Screen #2
(7) *5&6 Indicates Equipmetn Coupled to a Specific Profile that was Removed by Both Screens 5 and 6
(8) Since the TC press profile for the MSIV open case is the same magnitude and shorter duration, the profile will be represented in further analysis by the TC MSIV closed profile.

TABLE D-12 - MSIV ENVIRONMENTAL SUMMARY MATRIX

MAJOR EQUIPMENT CATEGORY MSIV

SEQUENCE / ENVIRONMENT	TB	TW	TC	TQUV	AE
TEMPERATURE	MEDIUM	MEDIUM	MEDIUM	LOW	---
PRESSURE	LOW	HIGH	HIGH	LOW	---
HUMIDITY	LOW	LOW	LOW	LOW	---
SPRAY/SUBMERGENCE	LOW	LOW	LOW	LOW	---
RADIATION	LOW	LOW	LOW	LOW	---
VIBRATION	LOW	LOW	LOW	LOW	---
PRESSURE/HUMIDITY	LOW	HIGH/LOW	HIGH/LOW	LOW	---
TEMPERATURE/HUMIDITY	MEDIUM/LOW	MEDIUM/LOW	MEDIUM/LOW	LOW	---
TEMPERATURE/RADIATION	MEDIUM/LOW	MEDIUM/LOW	MEDIUM/LOW	LOW	---
TIME OF APPLICABILITY	V.B.	CONT.FAIL.	CONT.FAIL.	V.B.	---

TABLE D-13 - HPCI/RCIC ENVIRONMENTAL SUMMARY MATRIX

MAJOR EQUIPMENT CATEGORY HPCI/RCIC

SEQUENCE / ENVIRONMENT	TB	TW	TC	TQUV	AE
TEMPERATURE	MEDIUM	HIGH	MEDIUM	LOW	---
PRESSURE	LOW	HIGH	HIGH	LOW	---
HUMIDITY	LOW	LOW	LOW	LOW	---
SPRAY/SUBMERGENCE	LOW	LOW	LOW	LOW	---
RADIATION	LOW	LOW	LOW	LOW	---
VIBRATION	LOW	MEDIUM	MEDIUM	LOW	---
PRESSURE/HUMIDITY	LOW	HIGH/LOW	HIGH/LOW	LOW	---
TEMPERATURE/HUMIDITY	MEDIUM/LOW	HIGH/LOW	MEDIUM/LOW	LOW	---
TEMPERATURE/RADIATION	MEDIUM/LOW	HIGH/LOW	MEDIUM/LOW	LOW	---
TIME OF APPLICABILITY	V.B.	V.B.	V.B.	V.B.	---

TABLE D-14 - RHR ENVIRONMENTAL SUMMARY MATRIX

MAJOR EQUIPMENT CATEGORY RHR

SEQUENCE / ENVIRONMENT	TB	TW	TC	TQUV	AE
TEMPERATURE	---	MEDIUM	MEDIUM	---	---
PRESSURE	---	HIGH	HIGH	---	---
HUMIDITY	---	LOW	LOW	---	---
SPRAY/SUBMERGENCE	---	LOW	LOW	---	---
RADIATION	---	LOW	LOW	---	---
VIBRATION	---	LOW	LOW	---	---
PRESSURE/HUMIDITY	---	HIGH/LOW	HIGH/LOW	---	---
TEMPERATURE/HUMIDITY	---	MEDIUM/LOW	MEDIUM/LOW	---	---
TEMPERATURE/RADIATION	---	MEDIUM/LOW	MEDIUM/LOW	---	---
TIME OF APPLICABILITY	---	CONT.FAIL.	CONT.FAIL.	---	---

TABLE D-15 - SRV ENVIRONMENTAL SUMMARY MATRIX

MAJOR EQUIPMENT CATEGORY SRV

SEQUENCE / ENVIRONMENT	TB	TW	TC	TQUV	AE
TEMPERATURE	MEDIUM	HIGH	MEDIUM	LOW	---
PRESSURE	LOW	HIGH	HIGH	LOW	---
HUMIDITY	LOW	LOW	LOW	LOW	---
SPRAY/SUBMERGENCE	LOW	LOW	LOW	LOW	---
RADIATION	LOW	LOW	LOW	LOW	---
VIBRATION	LOW	MEDIUM	MEDIUM	LOW	---
PRESSURE/HUMIDITY	LOW	HIGH/LOW	HIGH/LOW	LOW	---
TEMPERATURE/HUMIDITY	MEDIUM/LOW	HIGH/LOW	MEDIUM/LOW	LOW	---
TEMPERATURE/RADIATION	MEDIUM/LOW	HIGH/LOW	MEDIUM/LOW	LOW	---
TIME OF APPLICABILITY	V.B.	V.B.	V.B.	V.B.	---

TABLE D-16 - THERMOCOUPLE ENVIRONMENTAL SUMMARY MATRIX

MAJOR EQUIPMENT CATEGORY THERMOCOUPLE

SEQUENCE / ENVIRONMENT	TB	TW	TC	TQUV	AE
TEMPERATURE	MEDIUM	HIGH	MEDIUM	LOW	HIGH
PRESSURE	LOW	HIGH	HIGH	LOW	HIGH
HUMIDITY	LOW	LOW	LOW	LOW	MEDIUM
SPRAY/SUBMERGENCE	LOW	LOW	LOW	LOW	MEDIUM
RADIATION	LOW	LOW	LOW	LOW	MEDIUM
VIBRATION	LOW	MEDIUM	MEDIUM	LOW	MEDIUM
PRESSURE/HUMIDITY	LOW	HIGH/LOW	HIGH/LOW	LOW	HIGH/MEDIUM
TEMPERATURE/HUMIDITY	MEDIUM/LOW	HIGH/LOW	MEDIUM/LOW	LOW	HIGH/MEDIUM
TEMPERATURE/RADIATION	MEDIUM/LOW	HIGH/LOW	MEDIUM/LOW	LOW	HIGH/MEDIUM
TIME OF APPLICABILITY	V.B.	V.B.	V.B.	V.B.	V.B.

TABLE D-17 - RTD ENVIRONMENTAL SUMMARY MATRIX

MAJOR EQUIPMENT CATEGORY RTD

SEQUENCE / ENVIRONMENT	TB	TW	TC	TQUV	AE
TEMPERATURE	HIGH	MEDIUM	---	HIGH	---
PRESSURE	MEDIUM	HIGH	---	MEDIUM	---
HUMIDITY	MEDIUM	LOW	---	MEDIUM	---
SPRAY/SUBMERGENCE	MEDIUM	LOW	---	MEDIUM	---
RADIATION	MEDIUM	LOW	---	MEDIUM	---
VIBRATION	LOW	LOW	---	LOW	---
PRESSURE/HUMIDITY	MEDIUM	HIGH/LOW	---	MEDIUM	---
TEMPERATURE/HUMIDITY	HIGH/MEDIUM	MEDIUM/LOW	---	HIGH/MEDIUM	---
TEMPERATURE/RADIATION	HIGH/MEDIUM	MEDIUM/LOW	---	HIGH/MEDIUM	---
TIME OF APPLICABILITY	CONT.FAIL.	CONT.FAIL.	---	CONT.FAIL.	---

TABLE D-18 - DRYWELL PRESS. MONITOR ENVIRONMENTAL SUMMARY MATRIX

MAJOR EQUIPMENT CATEGORY <u>PRESSURE INST.</u>

SEQUENCE / ENVIRONMENT	TB	TW	TC	TQUV	AE
TEMPERATURE	HIGH	MEDIUM	---	HIGH	---
PRESSURE	MEDIUM	HIGH	---	MEDIUM	---
HUMIDITY	MEDIUM	LOW	---	MEDIUM	---
SPRAY/SUBMERGENCE	MEDIUM	LOW	---	MEDIUM	---
RADIATION	MEDIUM	LOW	---	MEDIUM	---
VIBRATION	LOW	LOW	---	LOW	---
PRESSURE/HUMIDITY	MEDIUM	HIGH/LOW	---	MEDIUM	---
TEMPERATURE/HUMIDITY	HIGH/MEDIUM	MEDIUM/LOW	---	HIGH/MEDIUM	---
TEMPERATURE/RADIATION	HIGH/MEDIUM	MEDIUM/LOW	---	HIGH/MEDIUM	---
TIME OF APPLICABILITY	CONT.FAIL.	CONT.FAIL.	---	CONT.FAIL.	---

TABLE D-19 - H_2/RAD. MONT. ENVIRONMENTAL SUMMARY MATRIX

MAJOR EQUIPMENT CATEGORY RAD/H_2 MON

SEQUENCE / ENVIRONMENT	TB	TW	TC	TQUV	AE
TEMPERATURE	HIGH	HIGH	MEDIUM	HIGH	HIGH
PRESSURE	MEDIUM	HIGH	HIGH	MEDIUM	HIGH
HUMIDITY	MEDIUM	LOW	LOW	MEDIUM	MEDIUM
SPRAY/SUBMERGENCE	MEDIUM	LOW	LOW	MEDIUM	MEDIUM
RADIATION	MEDIUM	LOW	LOW	MEDIUM	MEDIUM
VIBRATION	LOW	MEDIUM	MEDIUM	LOW	MEDIUM
PRESSURE/HUMIDITY	MEDIUM	HIGH/LOW	HIGH/LOW	MEDIUM	HIGH/MEDIUM
TEMPERATURE/HUMIDITY	HIGH/MEDIUM	HIGH/LOW	MEDIUM/LOW	HIGH/MEDIUM	HIGH/MEDIUM
TEMPERATURE/RADIATION	HIGH/MEDIUM	HIGH/LOW	MEDIUM/LOW	HIGH/MEDIUM	HIGH/MEDIUM
TIME OF APPLICABILITY	CONT.FAIL.	V.B.	V.B.	CONT.FAIL.	V.B.

profile comparison as presented in Appendix C, and the screening functions as summarized in Table D-11. If a piece of equipment, coupled with its profile comparison, sees above 140% of the qualification level for a given environment (during its useful time) it is rated as "high" for that environment. Between 100% and 140% gets a "medium" rating and less than 100% gets a "low."

For those environments (humidity, spray, submergence, radiation, and vibration) found not to exceed 100% of qualification conditions for a sequence, the highest ranking possible is a "medium." A "medium" or 100% of qualification level environment was assumed to exist for the humidity, spray/submergence, and radiation environments if vessel breach is projected to occur within the equipment's "useful" timeline. Prior to vessel breach a low ranking is assumed for these environments (based on the assumption that the vessel and pool will approximate a closed system until breach occurs) except for the AE sequence where a medium ranking is assumed from the start of the sequence. If containment failure occurs during the equipment's "useful" timeline, then the vibration environment is assigned a "medium" value based on possible blowdown effects. Otherwise a "low" ranking is assumed.

Three environmental combinations deemed likely to occur were investigated for synergistic effects (pressure/humidity, temperature/humidity, and temperature-/radiation) and treated separate from the above single environment descriptions. These combinations were chosen based on being among the most likely to occur from past test data. The combination environments are ranked with a combined additive value of their constituent parts. For example, a "medium" humidity environment and a "high" pressure environment form a ranking of "medium-high" for the pressure/humidity combination.

With the ranking process defined, an arbitrary point system can now be implemented to make the qualitative comparisons desired. Assigning a value of 5 for a "high", 3 for a "medium" and 1 for a "low" allows comparison data to be derived for each piece of equipment. The synergistic environments have the constituent single environment values added for the overall score. A series of vertical additions (lower half of Table D-20) allows a qualitative determination of the projected worst case sequence for a given piece of equipment. A series of horizontal additions (upper half of Table D-20)

TABLE D-20 - SUMMARY MATRIX RANKING RESULTS

EQUIPMENT / ENVIRONMENT	MSIV	HPCI/RCIC	RHR	SRV	THERM	RTD	PRESS. MONT.	H_2/RAD MONT.
TEMP.	10	12	6	12	17	13	13	23
PRESS.	12	12	10	12	17	11	11	21
HUMIDITY	4	4	2	4	7	7	7	11
SPRAY	4	4	2	4	7	7	7	11
RAD	4	4	2	4	7	7	7	11
VIBRATION	4	8	2	8	11	3	3	11
PRESS/HUM	16	16	12	16	24	18	18	32
TEMP/HUM	14	16	8	16	24	22	22	34
TEMP/RAD	14	16	8	16	24	22	22	34

EQUIPMENT / SEQUENCE	MSIV	HPCI/RCIC	RHR	SRV	THERM	RTD	PRESS. MONT.	H_2/RAD MONT.
TB	18	18	--	18	18	40	40	40
TW	26	34	26	34	34	26	26	34
TC	26	28	26	28	28	--	--	28
TQUV	12	12	--	12	12	37	37	40
AE	--	--	--	--	46	--	--	46

provides an indication of which environment tends to be the most severe for a given piece of equipment accounting for all of the five major accident sequences. Table D-20 presents the results of the summary matricies ranking process.

2.2.7 Summary

Section 2.2 has presented the results of the first phase of the 3 phase process shown in Figure D-2. The results were based on the use of six logical screens designed to filter the equipment and environment information presented in Appendices B and C. The screens filtered less severe cases from further consideration. The screen results are summarized in a ranking matrix format shown as Tables D-12 through D-19. Table D-20 shows the relative ranking of these results in tabular form. This data will be combined with failure mode (phase 2) and functional information (phase 3) as input into to the next step of the analysis methodology.

2.3 Phase 2 - Failure Mode Matrix Screening

2.3.1 Introduction

Section 2.3 deals with construction and screening of a failure mode matrix for the components and profiles of interest. Figure D-2 shows that this is the second phase of the 3 phase screening process. This phase will examine possible failure modes for the equipment identified in Appendix B (Ref. 20) and then remove those failure modes deemed impossible or unlikely. Figure D-9 presents a functional flow of this process. The results from this section will be combined with the results from the rest of the 3 phase process to be used in ranking equipment for testing.

2.3.2 Failure Mode Matrix Construction

This section addresses the construction of a failure mode matrix for each of the eight equipment assemblies. The form used for the matrix construction is shown as Table D-21. Note that for each major equipment assembly, all the environments (and important combinations of environments) encountered are

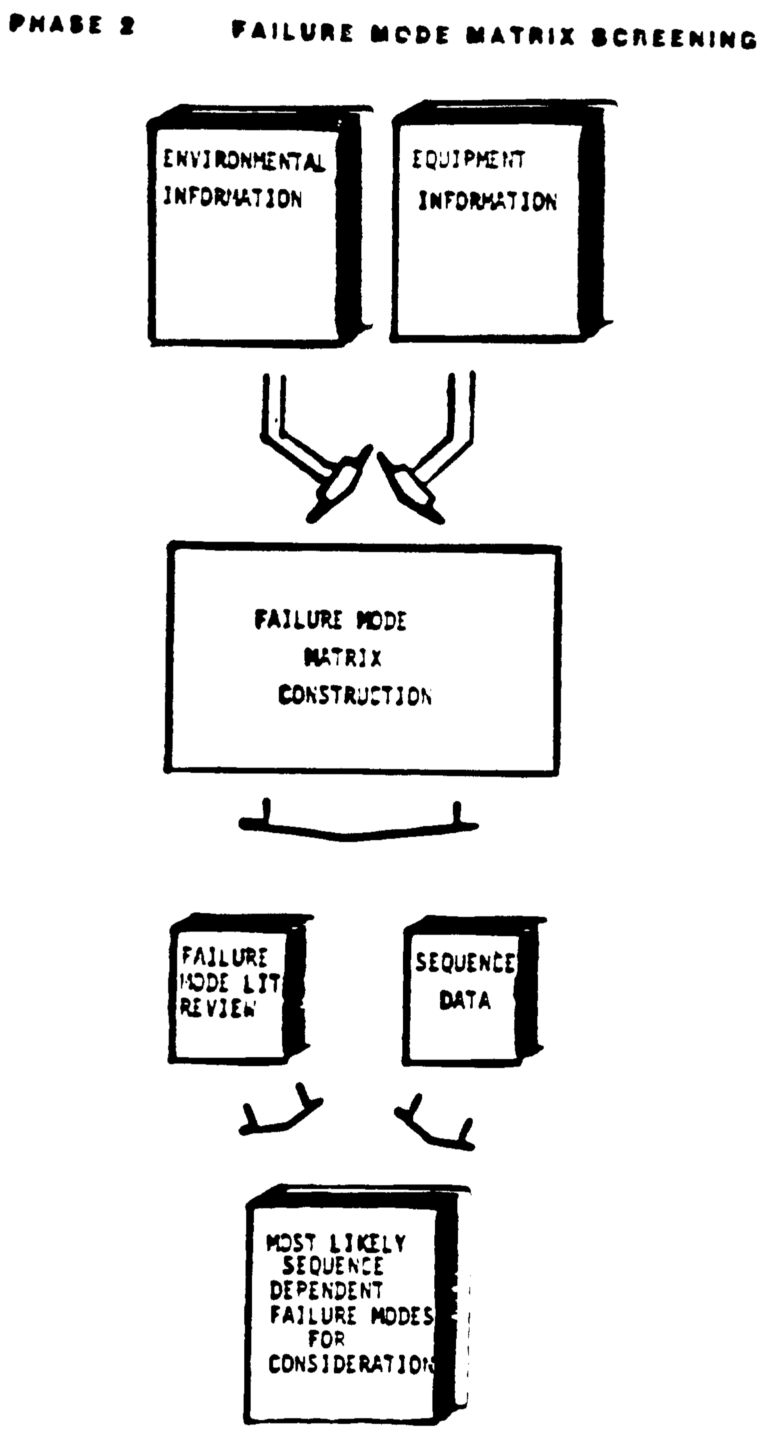


FIGURE D-9 - FAILURE MODE MATRIX METHODOLOGY FLOW

TABLE D-21 - FAILURE MODE MATRIX FORM

SUBASSEMBLIES ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE						
PRESSURE						
HUMIDITY						
SPRAY/SUBMERGENCE						
RADIATION						
VIBRATION						
COMBINATION 1						
COMBINATION 2						
COMBINATION 3						

COMB. ENVIR. #1 = PRESSURE/HUMIDITY   COMB. ENVIR. #2 - TEMP/HUMIDITY   COMB. ENVIR. #3 = RADIATION/TEMPERATURE

listed down the vertical column while each of the subassemblies comprising the major equipment category is listed horizontally across the top. The spray and submergence environments are combined as they both produce similar failure modes. Each slot in the matrix is filled in for how the listed environment may effect the given subassembly. The information to postulate these failure modes is derived from Licensee Event Reports (LER), manufacturer data, a literature review, and engineering judgement. Because some major equipment assemblies have the same generic subassembly breakdown, they are combined onto a single table. Tables D-22 through D-27 present the completed matrices for the postulated failure modes. The data on these tables is the input to the failure mode screening functions described in the subsequent sections.

2.3.3 Literature Review Screening Function

The literature review screening function is intended to reduce the list of postulated failure modes to those deemed most likely by past experience. Literature surveys covering plant LERs, operating and maintenance records, I&E Information Notices, Qualification Testing Evaluation Program Reports, and vendor reports form the basis for this screen. Three Mile Island (TMI) accident analysis reports also contribute valuable information. The screening function itself is quite simple. If a particular failure mode is mentioned as being significant in the literature review, it is carried forward for further analysis. If a failure mode is indicated to be unlikely then it is removed from further consideration. For those failure modes where no or indeterminate information is found, a case by case determination is made based on engineering judgement as to whether to include or eliminate the failure mode. Where no clear determination is possible, several failure modes may be included to preserve a conservative estimate with this screen.

In implementing this screening function, a "one" is assigned those failure modes considered possible based on the literature review. For those failure modes deemed unlikely a "zero" value is entered in the matrix. The score for a given piece of equipment is then normalized by dividing the number of possible failure modes carried forward by the total number of failure modes projected. The normalization process is necessary because some pieces of equipment have more subassemblies and thus more failure modes possible than

TABLE D-22 - MSIV/SRV VALVE FAILURE MODE MATRIX

SUBASSEMBLIES / ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY	WINDING INSULATION MELT/HOT SPOT MECHANICAL BINDING	---	MICROSWITCH OVER-HEATING & FAILURE	---
PRESSURE	FAULT PROPAGATION	GASKET SEAL MOVEMENT/ MISALIGNMENT	SEAL MISALIGNMENT /FAILURE	---	MISALIGNMENT	---
HUMIDITY	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRATION ELECTRICAL SHORT	ELECTRICAL SHORT THROUGH EXISTING FAULT	---	ELECTRICAL FAULT	---
SPRAY/SUBMERGENCE	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRATION ELECTRICAL SHORT	ELECTRICAL SHORT THROUGH EXISTING FAULT	---	ELECTRICAL FAULT	---
RADIATION	EMBRITTLEMENT/ SHRINKAGE LEADING TO BREAKS & CRACKS	SEAL EMBRITTLEMENT	SEAL/WINDING INSULATION EMBRITTLEMENT (POSSIBLE SHORT) DETERIORATES	---	ELECTRONICS DEGRADATION	---
VIBRATION	RAPID FAULT PROPAGATION MECHANICAL SEVERANCE	SEAL FAILURE CABLE TO CONNECTOR MECHANICAL SEVERANCE	MECH. BINDING LOSS OF WINDING CONTINUITY	---	LOSS OF ELECTRICAL CONTINUITY - MECH. MISALIGNMENT	---
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	---	ELECTRICAL SHORT MISALIGNMENT	---
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS MECH. BINDING	---	MICROSWITCH MISALIGNMENT	---
COMBINATION 3	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION	JOINT EXP/SEAL EMBRITTLEMENT LEADING TO LOSS	WINDING EMBRITTLEMENT /HOT SPOTS	---	CONTACT MISALIGNMENT /ELECTRONICS DEGRADATION	---

COMB. ENVIR. #1 = PRESSURE/HUMIDITY COMB. ENVIR. #2 - TEMP/HUMIDITY COMB. ENVIR. #3 = RADIATION/TEMPERATURE

TABLE D-23 - HPCI/RCIC/RHR VALVE FAILURE MODE MATRIX

SUBASSEMBLIES / ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY	---	MOTOR WINDING MECH./HOT SPOTS MECH. BINDING ELECTRICAL SHORT	MICROSWITCH OVER-HEATING & FAILURE	---
PRESSURE	FAULT PROPAGATION	GASKET SEAL MOVEMENT/ MISALIGNMENT	---	PACKING FAILURE MOTOR BEARING MISALIGNMENT/ HEATING & FAILURE	MISALIGNMENT	---
HUMIDITY	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRA-TION ELECTRICAL SHORT	---	ELECTRICAL SHORT THROUGH EXPOSED FAULTS	ELECTRICAL FAULT	---
SPRAY/SUBMERGENCE	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRA-TION ELECTRICAL SHORT	---	ELECTRICAL SHORT THROUGH EXPOSED FAULTS	ELECTRICAL FAULT	---
RADIATION	EMBRITTLEMENT/ SHRINKAGE LEADING TO BREAKS & CRACKS	SEAL EMBRITTLE-MENT	---	WINDING INSULATION DEGRADATION (SHORT) VALVE SEAT & DISK DECOMPOSITION	ELECTRONICS DEGRADATION	---
VIBRATION	RAPID FAULT PROPAGATION MECHANICAL SEVERANCE	SEAL FAILURE CABLE TO CONNECTOR MECHANICAL SEVERANCE	---	MECH. BINDING ELECTRICAL DISCONNECTS	LOSS OF ELECTRICAL CONTINUITY - MECH. MISALIGNMENT	---
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	---	ELECTRICAL SHORT DUE TO MOISTURE INTRUSTION OF MOTOR WINDINGS	ELECTRICAL SHORT MISALIGNMENT	---
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	---	ELECTRICAL SHORT DUE TO MOISTURE INTRUSTION OF MOTOR WINDINGS MECH. BINDING	MICROSWITCH ELECTRONICS FAIL-URE DUE TO OVER-HEATING & SHORT CIRCUIT	---
COMBINATION 3	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION	JOINT EXP/SEAL EMBRITTLEMENT LEADING TO LOSS	---	WINDING EMBRITTLE-MENTS/HOT SPOTS	CONTACT MISALIGNMENT /ELECTRONICS DEGRADATION	---

COMB. ENVIR. #1 = PRESSURE/HUMIDITY   COMB. ENVIR. #2 - TEMP/HUMIDITY   COMB. ENVIR. #3 = RADIATION/TEMPERATURE

TABLE D-24 - THERMOCOUPLE FAILURE MODE MATRIX

SUBASSEMBLIES / ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY	---	---	---	900° VIRTUAL JUNCTION PHENOMENON
PRESSURE	FAULT PROPAGATION	GASKET SEAL MOVEMENT/ MISALIGNMENT	---	---	---	JUNCTION DEFORMATION
HUMIDITY	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRATION ELECTRICAL SHORT	---	---	---	INACCURATE INDICATION
SPRAY/SUBMERGENCE	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRATION ELECTRICAL SHORT	---	---	---	INACCURATE INDICATION
RADIATION	EMBRITTLEMENT/ SHRINKAGE LEADING TO BREAKS & CRACKS	SEAL EMBRITTLEMENT	---	---	---	INACCURATE INDICATION
VIBRATION	RAPID FAULT PROPAGATION MECHANICAL SEVERANCE	SEAL FAILURE CABLE TO CONNECTOR MECHANICAL SEVERANCE	---	---	---	JUNCTION SEPARATION
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	---	---	---	JUNCTION DEFORMATION AND INACCURATE INDICATION
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	---	---	---	VIRTUAL JUNCTION PHENOMENON
COMBINATION 3	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION	JOINT EXP/SEAL EMBRITTLEMENT LEADING TO LOSS	---	---	---	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION

COMB. ENVIR. #1 = PRESSURE/HUMIDITY   COMB. ENVIR. #2 - TEMP/HUMIDITY   COMB. ENVIR. #3 = RADIATION/TEMPERATURE

TABLE D-25 - RTD FAILURE MODE MATRIX

SUBASSEMBLIES / ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY	---	---	---	SEAL DEFORMATION
PRESSURE	FAULT PROPAGATION	GASKET SEAL MOVEMENT/ MISALIGNMENT	---	---	---	SEAL DEFORMATION
HUMIDITY	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRA- TION ELECTRICAL SHORT	---	---	---	HEAD SEAL GASKET PENETRATION LEADING TO ELECTRICAL SHORT
SPRAY/SUBMERGENCE	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRA- TION ELECTRICAL SHORT	---	---	---	HEAD SEAL GASKET PENETRATION LEADING TO ELECTRICAL SHORT
RADIATION	EMBRITTLEMENT/ SHRINKAGE LEADING TO BREAKS & CRACKS	SEAL EMBRITTLE- MENT	---	---	---	HEAD SEAL GASKET DEFORMATION
VIBRATION	RAPID FAULT PROPAGATION MECHANICAL SEVERANCE	SEAL FAILURE CABLE TO CONNECTOR MECHANICAL SEVERANCE	---	---	---	MECHANICAL SEAL STRESS- LOSS OF CONTINUITY
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	---	---	---	ELECTRICAL SHORT DUE TO SEAL PENETRATION
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	---	---	---	ELECTRICAL SHORT DUE TO SEAL PENETRATION
COMBINATION 3	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION	JOINT EXT/SEAL EMBRITTLEMENT LEADING TO LOSS	---	---	---	SEAL EMBRITTLEMENT

COMB. ENVIR. #1 = PRESSURE/HUMIDITY COMB. ENVIR. #2 - TEMP/HUMIDITY COMB. ENVIR. #3 = RADIATION/TEMPERATURE

TABLE D-26 - DRYWELL PRESSURE MONITOR FAILURE MODE MATRIX

SUBASSEMBLIES / ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY	---	---	---	CALIBRATION POTENTIONMETER ERROR
PRESSURE	FAULT PROPAGATION	GASKET SEAL MOVEMENT/ MISALIGNMENT	---	---	---	SEAL EROSION
HUMIDITY	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRA- TION ELECTRICAL SHORT	---	---	---	ELECTRONICS DAMAGE
SPRAY/SUBMERGENCE	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRA- TION ELECTRICAL SHORT	---	---	---	ELECTRONICS DAMAGE
RADIATION	EMBRITTLEMENT/ SHRINKAGE LEADING TO BREAKS & CRACKS	SEAL EMBRITTLE- MENT	---	---	---	ONLY IF INTERNAL SEAL LEAK-POSSIBLE EMBRITTLEMENT & ELEC. FAILURE
VIBRATION	RAPID FAULT PROPAGATION MECHANICAL SEVERANCE	SEAL FAILURE CABLE TO CONNECTOR MECHANICAL SEVERANCE	---	---	---	SEAL DAMAGE
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	---	---	---	ELECTRONICS SHORT
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	---	---	---	ELECTRONICS SHORT
COMBINATION 3	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION	JOINT EXP/SEAL EMBRITTLEMENT LEADING TO LOSS	---	---	---	SEAL EMBRITTLEMENT

COMB. ENVIR. #1 = PRESSURE/HUMIDITY COMB. ENVIR. #2 - TEMP/HUMIDITY COMB. ENVIR. #3 = RADIATION/TEMPERATURE

TABLE D-27 - H_2/RAD. MONITOR FAILURE MODE MATRIX

SUBASSEMBLIES / ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY	---	---	---	ELECTRONICS FAILURE
PRESSURE	FAULT PROPAGATION	GASKET SEAL MOVEMENT/ MISALIGNMENT	---	---	---	GM TUBE STRESS-LOSS OF QUENCH GAS
HUMIDITY	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRA-TION ELECTRICAL SHORT	---	---	---	ELECTRONICS PACKAGE SHORT
SPRAY/SUBMERGENCE	ELECTRICAL SHORT THROUGH EXISTING FAULT	MOISTURE PENETRA-TION ELECTRICAL SHORT	---	---	---	ELECTRONICS PACKAGE SHORT
RADIATION	EMBRITTLEMENT/ SHRINKAGE LEADING TO BREAKS & CRACKS	SEAL EMBRITTLE-MENT	---	---	---	MOS-TRANSISTOR FAIL-URE IN ELECTRONICS PACKAGE
VIBRATION	RAPID FAULT PROPAGATION MECHANICAL SEVERANCE	SEAL FAILURE CABLE TO CONNECTOR MECHANICAL SEVERANCE	---	---	---	RUPTURE OF GM TUBE-LOSS OF INERT GAS
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS	---	---	---	ELECTRONICS SHORT DUE TO MOISTURE PENETRATION
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE.	---	---	---	ELECTRONICS SHORT DUE TO MOISTURE PENETRATION
COMBINATION 3	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION	JOINT EXP/SEAL EMBRITTLEMENT LEADING TO LOSS	---	---	--	ELECTRONICS DEGRAD-/ATION SEMICONDUCTOR JUNCTION FAILURE

COMB. ENVIR. #1 = PRESSURE/HUMIDITY   COMB. ENVIR. #2 - TEMP/HUMIDITY   COMB. ENVIR. #3 = RADIATION/TEMPERATURE

others. Therefore, the normalization produces a usable number for comparison of various pieces of equipment. The information presented in Section 5.0 of Appendix C suggests that the humidity, spray, radiation, and vibration environments (each by themselves), at the levels projected for the 5 accident sequences of interest, will be insufficient to induce failure. Therefore, a zero is entered in the screened failure mode matrix for these environments. An investigation of the failure data provided no basis for further distinction between failure modes and thus the rest of the projected failure modes were assigned a value of one. Tables D-28 through D-33 present the screened failure mode matricies for each piece of equipment, and Table D-34 shows a relative comparison of failure mode ranking between all equipment of interest. As can be seen from the Table D-34 results, all equipment scored similar results in the failure mode ranking and thus failure mode analysis cannot be used as a distinguishing factor. Although specific conclusions are difficult to deduce from the failure mode analysis, some conclusions can be made at a more general level. The literature review provided overwhelming evidence to suggest that the combination of moisture with temperature or pressure will be the dominate environmental stress in almost all cases of equipment failure. Thus, the failure modes listed with combination 1 and combination 2 environments on the failure mode matricies should be of highest consideration. In addition, recent testing has indicated aged cable failure in the temperature/radiation environment. For this reason the temperature/radiation environment was chosen as the third combination environment to investigate for synergistic effects. This combination should also be of high concern when developing testing strategies.

2.3.4 Summary

This section has presented the construction of the failure mode matrix based on a literature survey screening. The screening results showed all equipment scoring the same and thus no distinction is made on the basis of possible failure modes. The survey did indicate that moisture intrusion was the dominate failure mechanism in the majority of the cases. This "likely" failure mode data is combined with the environmental data from the last section and the functional data presented in the next section as input to the second step in the analysis methodology.

TABLE D-28 - MSIV/SRV VALVES LITERATURE REVIEW SCREENING RESULTS

SUBASSEMBLIES / ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION 1	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY 1	WINDING INSULATION MELT/HOT SPOT MECHANICAL BINDING 1	---	MICROSWITCH OVER-HEATING & FAILURE 1	---
PRESSURE	FAULT PROPAGATION 1	GASKET SEAL MOVEMENT/ MISALIGNMENT 1	SEAL MISALIGNMENT /FAILURE 1	---	MISALIGNMENT 1	---
HUMIDITY	0	0	0	---	0	---
SPRAY/SUBMERGENCE	0	0	0	---	0	---
RADIATION	0	0	0	---	0	---
VIBRATION	0	0	0	---	0	---
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	---	ELECTRICAL SHORT MISALIGNMENT 1	---
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS MECH. BINDING 1	---	ELECTRICAL SHORT MISALIGNMENT 1	---
COMBINATION 3	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION 1	JOINT EXP/SEAL EMBRITTLEMENT LEADING TO LOSS 1	WINDING EMBRITTLEMENT /HOT SPOTS 1	---	CONTACT MISALIGNMENT /ELECTRONICS DEGRADATION 1	---

COMB. ENVIR. #1 = PRESSURE/HUMIDITY COMB. ENVIR. #2 - TEMP/HUMIDITY COMB. ENVIR. #3 = RADIATION/TEMPERATURE

TABLE D-29 - HPCI/RCIC/RHR VALVE LITERATURE REVIEW SCREENING RESULTS

SUBASSEMBLIES / ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION 1	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY 1	---	MOTOR WINDING MECH./HOT SPOTS MECH. BINDING- ELECTRICAL SHORT 1	MICROSWITCH OVER- HEATING & FAILURE 1	---
PRESSURE	FAULT PROPAGATION 1	GASKET SEAL MOVEMENT/ MISALIGNMENT 1	---	PACKING FAILURE MOTOR BEARING MISALIGNMENT/ HEATING & FAILURE 1	MISALIGNMENT 1	---
HUMIDITY	0	0	---	0	0	---
SPRAY/SUBMERGENCE	0	0	---	0	0	---
RADIATION	0	0	---	0	0	---
VIBRATION	0	0	---	0	0	---
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	---	ELECTRICAL SHORT DUE TO MOISTURE INTRUSTION OF MOTOR WINDINGS 1	ELECTRICAL SHORT MISALIGNMENT 1	---
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	---	ELECTRICAL SHORT DUE TO MOISTURE INTRUSTION OF MOTOR WINDINGS MECH. BINDINGS 1	ELECTRICAL SHORT MISALIGNMENT 1	---
COMBINATION 3	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION 1	JOINT EXP/SEAL EMBRITTLEMENT LEADING TO LOSS 1	---	WINDING EMBRITTLE- MENTS/HOT SPOTS 1	CONTACT MISALIGNMENT /ELECTRONICS DEGRADATION 1	---

COMB. ENVIR. #1 = PRESSURE/HUMIDITY   COMB. ENVIR. #2 - TEMP/HUMIDITY   COMB. ENVIR. #3 = RADIATION/TEMPERATURE

TABLE D-30 - THERMOCOUPLE LITERATURE REVIEW SCREENING RESULTS

SUBASSEMBLIES / ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION 1	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY 1	---	---	---	900° VIRTUAL JUNCTION PHENOMENON 1
PRESSURE	FAULT PROPAGATION 1	GASKET SEAL MOVEMENT/ MISALIGNMENT 1	---	---	---	JUNCTION DEFORMATION 1
HUMIDITY	0	0	---	---	---	0
SPRAY/SUBMERGENCE	0	0	---	---	---	0
RADIATION	0	0	---	---	---	0
VIBRATION	0	0	---	---	---	0
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	---	---	---	JUNCTION DEFORMATION AND INACCURATE INDICATION 1
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	---	---	---	VIRTUAL JUNCTION PHENOMENON 1
COMBINATION 3	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION 1	JOINT EXP/SEAL EMBRITTLEMENT LEADING TO LOSS 1	---	---	---	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION 1

COMB. ENVIR. #1 = PRESSURE/HUMIDITY COMB. ENVIR. #2 - TEMP/HUMIDITY COMB. ENVIR. #3 = RADIATION/TEMPERATURE

TABLE D-31 - RTD LITERATURE REVIEW SCREENING RESULTS

SUBASSEMBLIES ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION 1	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY 1	---	---	---	SEAL DEFORMATION 1
PRESSURE	FAULT PROPAGATION 1	GASKET SEAL MOVEMENT/ MISALIGNMENT 1	---	---	---	SEAL DEFORMATION 1
HUMIDITY	0	0	---	---	---	0
SPRAY/SUBMERGENCE	0	0	---	---	---	0
RADIATION	0	0	---	---	---	0
VIBRATION	0	0	---	---	---	0
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	---	---	---	ELECTRICAL SHORT DUE TO SEAL PENETRATION 1
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	---	---	---	ELECTRICAL SHORT DUE TO SEAL PENETRATION 1
COMBINATION 3	EMBRITTLEMENT/MELT LEADING TO BREAKS & CRACKS IN INSULATION 1	JOINT EXT/SEAL EMBRITTLEMENT LEADING TO LOSS 1	---	---	---	SEAL EMBRITTLEMENT 1

COMB. ENVIR. #1 = PRESSURE/HUMIDITY COMB. ENVIR. #2 - TEMP/HUMIDITY COMB. ENVIR. #3 = VIBRATION/PRESSURE

TABLE D-32 - DRYWELL PRESSURE MONITOR LITERATURE REVIEW SCREENING RESULTS

SUBASSEMBLIES ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION 1	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY 1	---	---	---	CALIBRATION POTENTIONMETER ERROR 1
PRESSURE	FAULT PROPAGATION 1	GASKET SEAL MOVEMENT/ MISALIGNMENT 1	---	---	---	SEAL EROSION 1
HUMIDITY	0	0	---	---	---	0
SPRAY/SUBMERGENCE	0	0	---	---	---	0
RADIATION	0	0	---	---	---	0
VIBRATION	0	0	---	---	---	0
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	---	---	---	ELECTRONICS SHORT 1
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	---	---	---	ELECTRONICS SHORT 1
COMBINATION 3	EMBRITTLEMENT LEADING TO BREAKS & CRACKS IN INSULATION 1	JOINT EXP/SEAL EMBRITTLEMENT LEADING TO LOSS 1	---	---	---	SEAL EMBRITTLEMENT 1

COMB. ENVIR. #1 = PRESSURE/HUMIDITY COMB. ENVIR. #2 - TEMP/HUMIDITY COMB. ENVIR. #3 = VIBRATION/PRESSURE

TABLE D-33 - H_2/RAD. MONITOR LITERATURE REVIEW SCREENING RESULTS

SUBASSEMBLIES ENVIRONMENTS	CABLING	JUNCTION BOXES/ CONNECTORS	SOLENOID ASSEMBLY	MOTOR OPERATOR	POSITION INDICATION SWITCH	OTHER
TEMPERATURE	INSULATION MELT/ FAULT PROPAGATION 1	JOINT EXPANSION LEADING TO LOSS OF ELECTRICAL CONTINUITY 1	---	---	---	ELECTRONICS FAILURE 1
PRESSURE	FAULT PROPAGATION 1	GASKET SEAL MOVEMENT/ MISALIGNMENT 1	---	---	---	GM TUBE STRESS-LOSS OF QUENCH GAS 1
HUMIDITY	0	0	---	---	---	0
SPRAY/SUBMERGENCE	0	0	---	---	---	0
RADIATION	0	0	---	---	---	0
VIBRATION	0	0	---	---	---	0
COMBINATION 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	ELECTRICAL SHORT DUE TO MOISTURE PENETRATION THROUGH SEALS 1	---	---	---	ELECTRONICS SHORT DUE TO MOISTURE PENETRATION 1
COMBINATION 2	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	CARBONIZING JACKET MATERIAL LEADING TO SHORT DUE TO MOISTURE PENE. 1	---	---	---	ELECTRONICS SHORT DUE TO MOISTURE PENETRATION 1
COMBINATION 3	EMBRITTLEMENT LEADING TO BREAKS & CRACKS IN INSULATION 1	JOINT EXP/SEAL EMBRITTLEMENT LEADING TO LOSS 1	---	---	---	ELECTRONICS DEGRAD- ATION/SEMI CONDUCTOR JUNCTION FAILURE 1

COMB. ENVIR. #1 = PRESSURE/HUMIDITY   COMB. ENVIR. #2 - TEMP/HUMIDITY   COMB. ENVIR. #3 = VIBRATION/PRESSURE

TABLE D-34 - FAILURE MODE RANKING COMPARISON

EQUIPMENT / RANKING	MSIV/SRV	HPCI/RCIC/RHR	THERMOCOUPLE	RTD	PRESSURE XMETER	H_2/RAD. MONITOR
	$\frac{20}{36}$ = 55%	$\frac{20}{36}$ = 55%	$\frac{15}{27}$ = 55%	$\frac{15}{27}$ = 55%	$\frac{15}{27}$ = 55%	$\frac{15}{27}$ = 55%

2.4 Functional Screening

2.4.1 Introduction

Section 2.4 considers the functional importance of a given equipment assembly. This is the third phase of the 3 phase screening process identified in Figure D-2 and completes the first step in the analysis methodology. Figure D-10 illustrates the functional screening process. The first step in the process indicates the sequences for which the equipment may be useful. Once sequences in which the equipment has no function are removed, the actual ranking process was found to be sequence independent in most cases. Thus, a common ranking suffices for a given piece of equipment in all sequences where applicability was established.

The ranking process assigns points for each of seven factors used to establish functional importance. Each of these seven factors (redundancy, number of back-up systems, equipment separation, electrical independence, degree of noncomplexity, fail safe position, and plant status indication only) have the property of de-emphasizing functional importance and thus the higher the number of points obtained, the less functionally important a particular equipment assembly. The information for scoring each of the seven functional criteria comes from limited Browns Ferry schematic drawings, Appendix B, and engineering judgement based on other "typical" BWR designs.

In those rare cases where a piece of equipment shows some sequence dependence in one of the seven ranking categories, (ie, the MSIVs are much more important in the TC sequence because they have no back-up system with sufficient capacity to perform their heat removal function) the more conservative (the ranking showing the larger functional importance) scoring is applied for the common ranking. This section defines each of the seven functional categories and presents each of the eight equipment assembly rankings in these categories.

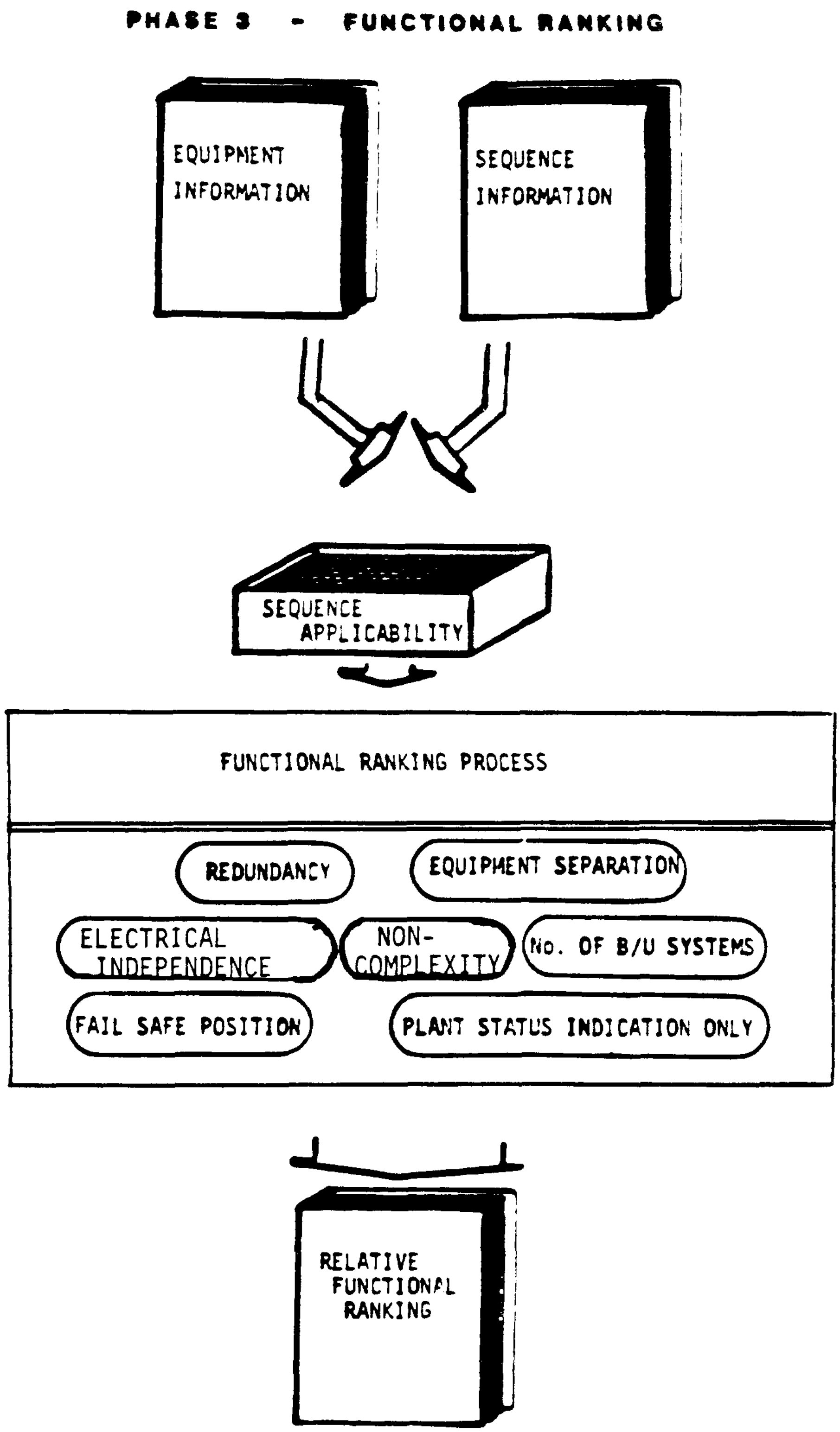


FIGURE D-10 - FUNCTIONAL SCREENING FLOW

2.4.2 Functional Criteria Definition

Functional criteria are used as a measure of equipment importance. Seven criteria were chosen as a basis to qualitatively rank the functional importance of the equipment. The seven criteria include the degree of redundancy for the equipment of interest, whether or not other (back-up) equipment can perform the function, non-complexity of the equipment, electrical independence of the equipment, equipment's failure position, whether or not the equipment can actively provide preventive/mitigative action (not just plant status), and physical separation of redundant equipment. Each of these functional criteria are defined in subsequent paragraphs and summarized in Table D-35.

Redundancy deals with having more than one component to perform the equipment's mitigation function. The RTD indication system is an example of equipment which exhibits redundancy. Typically, plants have two or more drywell temperature (RTD) devices. The failure of any one RTD does not prevent indication of plant status. Thus, having multiple devices will score points in this category and implies the equipment is somewhat less functionally important than equipment which doesn't exhibit this aspect.

The back-up systems category identifies different equipment assemblies which are able to perform the same function. If a set of equipment has one or more other systems able to perform its function, points are scored for this category and it is judged to be less functionally important. As with all the functional criteria, grey areas exist in judging the number of points assigned since the use of a back-up system may not apply for all circumstances. For example the RHR system can serve as a back-up system to the MSIVs as a source of heat removal. However the capacity of the RHR system may or may not indicate it is a viable back-up system to the MSIV system depending on the circumstances involved.

System noncomplexity is a factor directly related to survivability and thus functional capability. The fewer moving parts a system has, the more points the equipment receives since it has a better chance of surviving environmental stresses. Therefore the less functionally important it is judged to be. The noncomplexity criteria looks at both mechanical and electrical functions to form an equipment score.

TABLE D-35 - FUNCTIONAL FACTOR DEFINITIONS

REDUNDANCY -	The state of having multiple components for the equipment of interest (e.g., four MSIV valves and independent cables).
BACK-UP (B/U) SYSTEMS -	Refers to having a totally different system able to perform the desired mitigation function.
NONCOMPLEXITY -	A measure of a system's potential for malfunction based on the number and functions of the components involved.
ELECTRICAL INDEPENDENCE -	Refers to an equipment's devices being electrically independent such that failure of a single bus does not negate the intended mitigation function.
FAIL-SAFE POSITION APPROPRIATE -	A measure of whether or not the equipment can perform the required mitigative action in the deenergized or failed state.
PLANT STATUS INDICATIONS ONLY -	Differentiates between passive systems which only provide status and those systems with the potential to actively influence sequence outcome.
SEPARATION -	The state of having redundant devices with sufficient physical distance between them such that a localized environmental peak won't affect multiple devices.

Electrical independence refers to having the failure of a single power supply or bus not negate the equipment performance. Equipment which exhibits this property scores points for this category and becomes somewhat less functionally important.

Having an appropriate fail safe position is a measure of the potential of the equipment to perform the desired function when it is in the de-energized or failed state. Having the ability to fail "safe" scores points in this category and reduces functional importance.

Plant status indication differentiates between that equipment which is a passive contributor to accident mitigation and that which can actively influence sequence outcome. That equipment which provides status only scores points for this criteria and is considered less functionally important then equipment which has the potential to terminate a sequence.

Lastly, component separation refers to the physical distance between redundant devices in equipment systems. Since humidity-caused moisture intrusion enhanced by pressure and temperature effects is considered the dominant environmentallyinduced failure mode, separation differences were judged to be relatively unimportant assuming that these environment parameters may not vary significantly throughout the containment volume in a severe accident situation. Thus a "zero" is entered for all equipment in this category indicating that insufficient distance between redundant devices exists to enhance the probability of system function in adverse environments.

All these definitions are summarized in tabular form in Table D-35. At this point it is important to note that equipment becomes much more functionally important if it is the primary means of mitigating the sequence of interest. It is also important to realize that all the equipment has already been judged to be functionally important in Appendix B. The ranking system is simply used to judge relative importance. The next section describes and implements the ranking system.

2.4.3 Ranking Process

The first step of the ranking process illustrated in Figure D-10 is determining if a particular piece of equipment would be used for a given accident sequence. This allows the functional ranking to proceed for each piece of equipment with the results of the process applying only to those sequences where potential for sequence mitigation or plant status exists.

Implementing the ranking process involves developing a methodology to apply the seven factors. The process used is a simple one. If an equipment assembly exhibits a given criteria it scores points for that factor. Employing this idea implies the higher the equipment's total score for the seven functional categories, the less functionally important it is since all the factors describe conditions which imply lower functional importance. The actual ranking is done for the most part on a pass/fail basis. If the equipment exhibits the factor, it scores the points assigned to that category and if it doesn't it scores nothing. Each functional category is assigned a single point value. In some cases, it is not known with certainty the "typical" configuration for the equipment in most plants. A range of point values are then assigned to depict this uncertainty.

To further illustrate the methodology, two specific equipment examples follow. One example is for the MSIV equipment assembly, and the other is for thermocouple indication of core temperature.

Table D-36 shows the completed ranking results for all eight equipment assemblies. The results for the MSIV system are shown in column 1 of the table. Note that the MSIV system scores one point in the redundancy category. The MSIV system contains four separate valves, solenoids, and perhaps cable runs to make up four separate sets of equipment. Exhibiting this property indicates a lessened functional importance.

The primary function of the MSIV system is to allow primary heat removal through the Power Conversion System (PCS). There are alternate means of heat removal in the plant. The primary one in most sequences is the RHR system. This however is one of the cases where sequence dependence can influence the

TABLE D-36 - FUNCTIONAL RANKING RESULTS

EQUIPMENT MONT. / FUNCTIONAL CATEGORY	MSIV	HPCI/RCIC	RHR	SRV	THERMO	RTD	PRESS.MONT.	H_2/RAD
DEGREE OF REDUNDANCY	1	1(0 for TC Sequence)	1	1	1	1	0	1
BACK-UP SYSTEMS	1(0 for TC Sequence)	1	1	0	1	1	1	1
NON-COMPLEXITY	0	0	0	0	1	1	1	1
ELECTRIC INDEPENDENCE	0 or 1	1	1	0 or 1	0 or 1	0 or 1	1	0 or 1
FAIL SAFE	0	1	0	0	1	0	0	0
PLANT STATUS	0	0	0	0	1	1	1	1
PHYSICAL SEPARATION	0	0	0	0	0	0	0	0
TOTAL SCORE	1 - 3	3 - 4	3	1 - 2	5 - 6	4 - 5	4	4 - 5

scoring. The PCS has a much larger capacity for heat removal than the RHR system. In a TC sequence, where the reactor is still at power, the RHR system can't handle the demands placed on it and therefore can't be considered a realistic back-up system to the PCS. Even so, venting the containment may be a possible alternative although its success is considered unlikely. Considering these differences, a 0-1 score is given to this category for MSIVs.

The MSIV system is complex both mechanically and electrically. There are active electronic components in the solenoid control circuitry and mechanically the valve has many moving parts and critical seals. The MSIV assembly therefore scores a 0 for noncomplexity.

Consulting the electrical schematics for BWR designs indicates all four MSIV solenoids are often powered from the same bus. Therefore a fault on this bus has the potential to eliminate the ability to operate all the MSIVs. Thus electrical isolation does not exist and a 0 is entered for this category. To account for plants which may load their MSIV solenoids on separate buses, a 1 score is also shown.

The deenergized or failed state for the MSIV solenoid is shut. Since the MSIVs need to be open to perform their heat removal function, no points can be assigned for a favorable failure position.

The MSIVs obviously do more than provide plant status information. They provide a means of heat removal which actively mitigates many accident sequences. A 0 score is entered for this category.

Adding the points from all seven categories for the MSIV system shows a total score of from 1 to 3 points. This total score may be compared to the other seven devices to indicate a relative standing in terms of functional importance.

As an example of the scoring process for an indication device, column 5 of Table D-36 shows the scoring for the in-core thermocouples. In the first category the thermocouples scored a point for degree of redundancy. There are many individual thermocouples in various locations throughout the core. This implies a degree of redundancy exists since any one thermocouple can provide a general indication of core status.

Thermocouples scored a point in the back-up systems category because there are alternate, independent means of determining core temperatures. For example, steam table determination of saturation temperature for indicated vessel pressure provides a means of determining core temperature behavior without thermocouples.

The thermocouple equipment assembly scored 1 point in the noncomplexity category. Thermocouples display neither mechanical or electrical complexity being esentially nothing more than a cable run of two dissimilar metals.

It was not determined if the indication system associated with the thermocouples is electrically independent. Lack of equipment specific information on the thermocouples leads to assigning a 0 or a 1 to this category.

Significant experimental evidence exists to show that thermocouples exhibit a "virtual junction phenomenon." This means that even if the thermocouple junction in the core is melted or lost, that an accurate temperature indication is still provided at the point where the melted dissimilar materials are fused. This implies that thermocouples "fail safe" and provide valuable information even after a junction failure. Thus a point is scored in this category.

The thermocouple scores a point in the plant status category because it provides no mitigative action by itself, only plant status.

As stated previously, for this evaluation, no cases of sufficient physical separation were found and a score of 0 was placed in this category.

The other devices were ranked using the same methodology as just illustrated for the MSIV and thermocouple systems. The results from this ranking process are displayed in Table D-36.

2.4.4 Conclusions

This section has investigated the relative functional importance of the critical equipment assemblies within containment. To accomplish this, seven functional factors were defined and each equipment assembly was graded against each of these functional factors. The results of this ranking process indicate that on a relative basis, equipment which can actively perform a prevention/mitigation function in severe accidents is more important than equipment which provides plant status information only. Among the "active" equipment categories, MSIVs and SRVs rank higher relative to the HPCI/RCIC or RHR categories. This conclusion can be additionally supported by the following observations. In the MSIV case, restoring the PCS for heat removal (which includes the re-opening of one or more MSIVs) directly mitigates most accident sequences and actually places the plant in a preferred configuration (in accordance with EPGs). The SRVs are the only means for maintaining pressure control of the primary system when the reactor is isolated from the PCS. The HPCI and RCIC systems and the RHR shutdown cooling path, on the otherhand, are redundant within themselves or have other low pressure systems or RHR cooling paths which back-up these equipment categories.

In considering these conclusions, it is important to remember that these rankings are relative to one another but that all eight equipment assemblies are deemed functionally important to severe accidents.

2.5 Summary

Section 2.0 has presented the development and implementation of the 3 phase screening process. The 3 phase screening process was the first step in the analysis methodology. The 3 phases examined included environmental screening, failure mode screening, and functional screening. The results from each phase serve as input to the next step in the analysis methodology which will rank equipment for further analysis based on environmental, failure mode, and functional data.

3.0 RANKING OF THE 3 PHASE RESULTS

3.1 Introduction

Section 3.0 is designed to examine the individual results from the 3 phase ranking process. This is the second step in the analysis methodology which was shown as Figure D-1. The ranking process is composed of 3 basic steps: (1) development and explanation of the system used to rank the 3 phase results, (2) implementing the system, and (3) looking at the results. These results will serve as input to the PRA screening function which is the third step in the analysis methodology.

3.2 Ranking Methodology

Failure mode data is not used in the ranking methodology because no distinctions were found on the basis of failure modes. The failure mode data is used to suggest specific failure mechanisms to be aware of in developing test plans.

The ranking methodology must incoporate both functional and environmental results along with probable failure modes to project the best candidates for test or further analysis. Figure D-11 illustrates the comparative process used to determine these candidates. The first step involves taking the environmental results from Section 2.2 and the functional results from Section 2.4 and grouping them into high, moderate, and low categories. Assignment of the high, moderate, or low ranking is determined by examining the range of environmental or functional results and dividing it roughly into thirds. Thus, approximately one third of the thirty sequence/equipment environmental rankings shown in Table D-20 will be judged to fall into the high risk category, and one third of the eight functional rankings shown in Table D-36 will also fall into this category.

The next major step in the ranking process is to combine these results in order to easily determine an equipment assembly's relative overall standing. At this point it would be nice to be able to test any piece of equipment which showed a combined functional/environmental ranking greater than low/low.

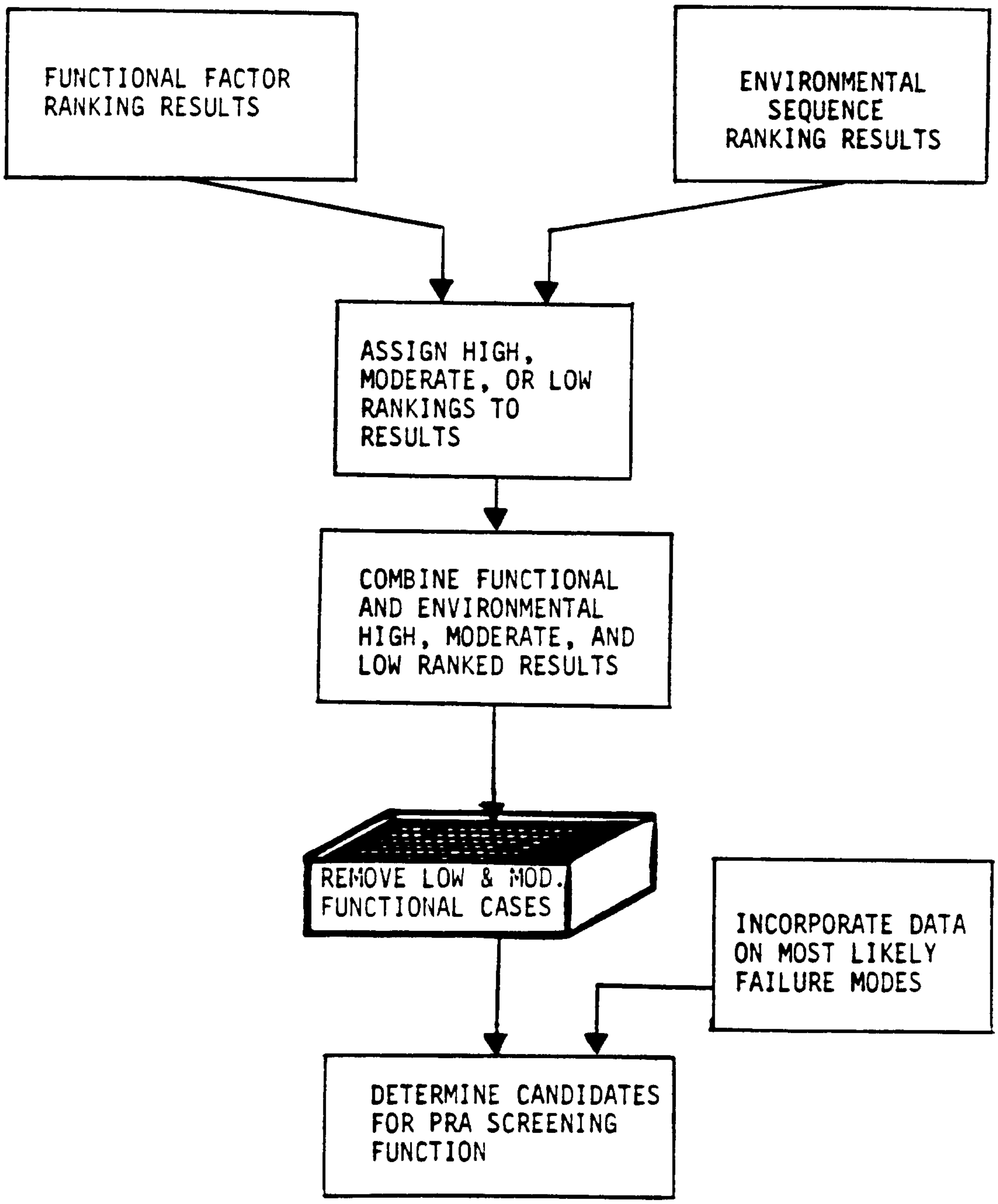


FIGURE D-11 - RANKING METHODOLOGY FLOW

Because of limited resources this is not possible so it is recomended that only those equipment assemblies which show a high functional ranking coupled with a high or moderate environmental ranking be examined further. This sufficiently narrows the list of possible test candidates and focuses attention on the most significant functional and environmental conditions observed.

3.3 Methodology Implementation

The first step in implementing the ranking methodology was to divide the eight equipment assemblies into high, moderate, or low functional and environmental importance for the applicable sequences. To accomplish this the Table D-36 functional ranking results were divided as follows: any equipment assembly which could score less than 3 points was judged to be of "high" functional importance, any equipment assembly with a mean score of 4 points was judged to be of "moderate" functional importance, and any equipment assembly with a mean score greater than 4 points was judged to be of "low" functional importance. Table D-37 summarizes these results. The table shows that MSIVs and SRVs were the two equipment assemblies which ranked "high" in functional importance.

The lower half of Table D-20, Summary Matrix Ranking Results, showed the environmental stress on each of the eight major equipment assemblies for the five sequences of interest. These results were then divided into the high, moderate and low categories. Any equipment assembly which received 34 or more points for a given sequence was rated in the "high" environmental risk category. The 19 - 33 point range earned a "moderate" ranking, and 18 points or less was judged to be a "low" environmental stress. Any equipment assembly which fell into the "high" or "moderate" environmental risk category has at least one environment which is projected to exceed typical qualification levels. The results of the ranking are also summarized in Table D-37.

The next step in the ranking methodology was to examine the results of Table D-37 to screen those cases of moderate or low functional importance and low environmental importance from further consideration. This is done to reduce the list of test candidates to a managable level and to emphasize that it is more important to test equipment which is functionally important. After implementing this screen, the remaining cases become those recommended for PRA analysis.

TABLE D-37 - FUNCTIONAL/ENVIRONMENTAL RANKING COMPARISONS

EQUIPMENT	MSIV	HPCI/RCIC	RHR	SRV	THERM	RTD	PRESS. MONT.	H_2/RAD MONT.
FUNCTIONAL IMPORTANCE	(H)	(M)	(M)	(H)	(L)	(L)	(M)	(L)
SEQUENCES:								
TB	L	L	-	L	L	H	H	H
TW	M	H	M	H	H	M	M	H
TC	M	M	M	M	M	-	-	M
TQUV	L	L	-	L	L	H	H	H
AE	-	-	-	-	H	-	-	H

H - HIGH M - MODERATE L - LOW

3.4 Ranking Methodology Results

Implementation of the ranking methodology resulted in four cases recommended for PRA analysis. These cases include the MSIV and SRV equipment assemblies stressed for the environmental conditions projected for the TC and TW sequences. Of the four cases, the SRV equipment assembly was the only one to score high in both the functional and environmental categories. The other three cases scored high in the functional category and moderate in the environmental category. In actuality there is probably little difference between the four cases and all four represent viable test candidates pending PRA analysis results. Section 2.3 identified that the most prevalent failure modes to be aware of for these components is electrical short circuiting due to moisture penetration through seals; solenoid winding localized melting due to hot spots; and mechanical binding of valve components due to heating and pressure induced misalignment. In addition, cable insulation melt or carbonizing allowing moisture penetration and limit switch temperature/pressure related failures are failure modes of concern.

3.5 Summary

Section 3.0 has presented the implementation of the methodology used to rank the overall 3 phase results. To accomplish this, functional and environmental results from the 3 phase process were assigned to high, moderate, or low risk categories. The results were then combined and screened to remove moderate or low functional cases and low environmental cases. The four cases of interest were found to be the MSIV and the SRV equipment assembly response to the TW and TC sequences. These four cases were judged to be equivalent in terms of the test potential with no one case presenting itself as the "best" test candidate. These 4 cases will be the input to the PRA screening function (step 3 of the analysis methodology).

4.0 PROBABILISTIC RISK ASSESSMENT PERSPECTIVES

4.1 Introduction

This section is designed to examine current PRA estimates of sequence probabilities and determine what changes, if any, might be introduced in these probabilities by environmentally-induced equipment failure. This perspective is important for two major reasons. First, current PRA estimates do not directly consider environmentally-induced equipment failure probabilities. The probability of equipment failure is based on estimates of operator action, maintainance schedules, and past performance data. One of the major goals of the PEEESAS program is to determine the environmental impacts on equipment and how they should be incorporated into PRA techniques. The second reason to use a PRA perspective is that it serves as a convenient method (other considerations being equal) to choose a first test candidate. With these observations in mind, we proceed to examine the PRA estimates for the four selected cases of interest. This will be done from a relative point of view concentrating on the degree of change in estimated probabilities due to possible environmental equipment failures rather than absolute numbers. This avoids disagreement over absolute numbers for current sequence probabilities.

4.2 Qualitative Arguments for Considering the MSIVs and SRVs as Test Candidates

In order to understand why the MSIVs and SRVs may be of particular importance in the two sequences of interest, it is important to understand certain elements that make up the scenarios and expected operator responses. Transients can be typically grouped into three major categories as was done in WASH-1400 [Ref.23]. These include loss of offsite power, an initial loss of the power conversion system (PCS), and transients in which the PCS is initially available but is subsequently lost as a result of pertubations to the PCS following reactor trip conditions. Based on EPRI data [Ref.24] and other NRC reports such as the one on station blackout [Ref.25], these three transient categories contribute typically about 1%, 10%, and nearly 90% respectively to the overall frequency of transients at nuclear power plants.

History of offsite power events has shown that the probability of recovering offsite power, and hence the ability to restore the use of the PCS or other AC-driven systems as a primary heat removal path, is greater than 50% in about one-half hour and exceeds 90% by approximately four to five hours after the initial power loss. Events involving loss of the PCS are similar in that there is an estimated 90% chance of restoring the PCS by approximately four to five hours after its initial loss [Ref.26]. Particularly in cases where the PCS has been lost due to pertubations in the system (that is, hardware faults have not occurred), recovery of the PCS is considerably more likely.

Because of these generally high chances of recovering from the initial transient conditions, particularly in the case of 90% of all transients in which PCS recovery is quite likely, the Emergency Procedure Guidelines often stress the use of the PCS as a preferred source of heat removal for the plant. Steps regarding water level control (RC/L-2), vessel pressure control (RC/P-2), using the main condenser as the preferred heat sink (RC/P-1), and use of the main condenser for emergency depressurization (C2-1) are examples [Ref.27]. This is not surprising when one considers that operators are extremely familiar with operation of the PCS since it is the normal heat rejection path for the plant.

Considering these points, one can begin to see the possible importance that the ability of opening the MSIVs might have in the TC and TW sequences. With PCS recovery likely, especially for the extreme long time periods associated with the recovery potential for the TW scenario, the ability of the MSIVs to be reopened following exposure to significant environmental conditions within the containment could spell the difference between successful recovery of the PCS and failure. The added facts that the Residual Heat Removal system is of little use in a TC scenario (power is too high) and is likely failed and hence its recovery is uncertain in the TW scenario, add further arguments for the importance of the MSIVs in these two sequences.

Given the PCS can not be restored, operability of the primary system SRVs becomes more important particularly if conditions arise (as they will in these two sequences) where the high pressure emergency core cooling systems eventually fail due to environmental conditions within containment (for example, HPCI/RCIC will fail due to high temperature suppression pool water which ultimately becomes the suction source for these systems in both sequences). The only way that low pressure systems can then be used to continue the core heat removal process is through use of the SRVs so that low pressure can be maintained in the primary system.

It is on the basis of these qualitative considerations that both the MSIV solenoids and the SRV solenoids appear as potentially reasonable candidates for testing in the PEEESAS program. The following subsections address in a more quantitative manner, the potential importance of these devices in the TC and TW scenarios.

4.3 MSIVs in the TC Sequence

The first case considered is operation of MSIVs during a TC sequence. Using current Accident Sequence Evaluation Program (ASEP) estimates as a baseline, there is roughly a 50/50 chance of MSIV closure in this sequence [ref.28]. Current PRA estimates give no credit for reopening the MSIVs in the case where the MSIVs close at or shortly following sequence initiation. Reasons for this include the time necessary to equalize pressure around and open the valves, the short time to restore other portions of the PCS if they were lost, and operator occupation with other aspects of accident management. Therefore, the current non-recovery probability associated with opening the MSIVs is = 1.0. This implies that any environmentally induced failure of the MSIVs (due to the rapidly forming severe conditions experienced within containment) will not produce any change in the sequence probability as currently estimated. However, if future PRA estimates give some credit for opening MSIVs in this sequence, environmentally induced valve failure will become a factor to consider.

When the MSIVs initially stay open in this sequence there are different factors to consider. The sequence becomes much longer affording more opportunity for recovery as long as the PCS remains in operation. Without the main turbine on line, the PCS in most plants is typically capable of absorbing about 25% reactor power. Current estimates for power equilibrium level is about 30% in this sequence which means the suppression pool will be absorbing

5% reactor power. The 5% power absorbed in the suppression pool will eventually raise environmental conditions to critical levels. If the MSIVs should subsequently shut because of a previously unconsidered environmentally induced failure of the MSIVs, the MSIV open contribution to the sequence probability is eliminated, and the MSIV closure contribution doubles (100% MSIV closure). This results in the same net core damage probability, but the MSIV-closed cases have a higher chance of containment failure due to the smaller amount of time available for recovery actions. This is a more severe condition from a risk standpoint, and so the increased MSIV closure probability introduced by an environmentally induced failure of the valves could significantly influence the risk associated with the total TC scenario.

4.4 MSIVs in the TW Sequence

The TW sequence is characterized by failure to restore suppression pool cooling (RHR) or reopen MSIVs in order to use the PCS. Current PRA estimates assume that use of the PCS can mitigate the sequence right up to containment failure. Figure D-12 shows that as the sequence progresses, the probability of failing to restore the PCS is currently estimated to decrease exponentially with time [Ref.26]. In Figure D-12 it is assumed the MSIVs are able to operate throughout the sequence and opening them is only a matter of clearing the trip signal, rectifying any other PCS problems, and equalizing the pressure around the valves. In fact, the actual availability of these valves may be very different. Figure D-13 is taken from Appendix C and shows how projected drywell temperature for the TW sequence compares to typical qualification profiles. Note from Figure D-13 that projected drywell temperature exceeds the qualification profile 18 hours into the accident and exceeds maximum qualification level about 27 hours into the accident. If an environmentally induced failure of the MSIV assembly occurs when qualification levels are exceeded, then there may be only 18-27 hours to recover the PCS instead of the 35 hours presently used in PRA analysis. Looking back at Figure D-12, it can be seen that the nonrecovery probability at 18 hours is about .02 while the nonrecovery probability at 35 hours is only .004. This implies that the sequence probability could change by as much as a factor of 5 if environmentally induced failure of the MSIVs were to occur after about 18 hours into the sequence.

NONRECOVERY PROBABILITY VS. TIME

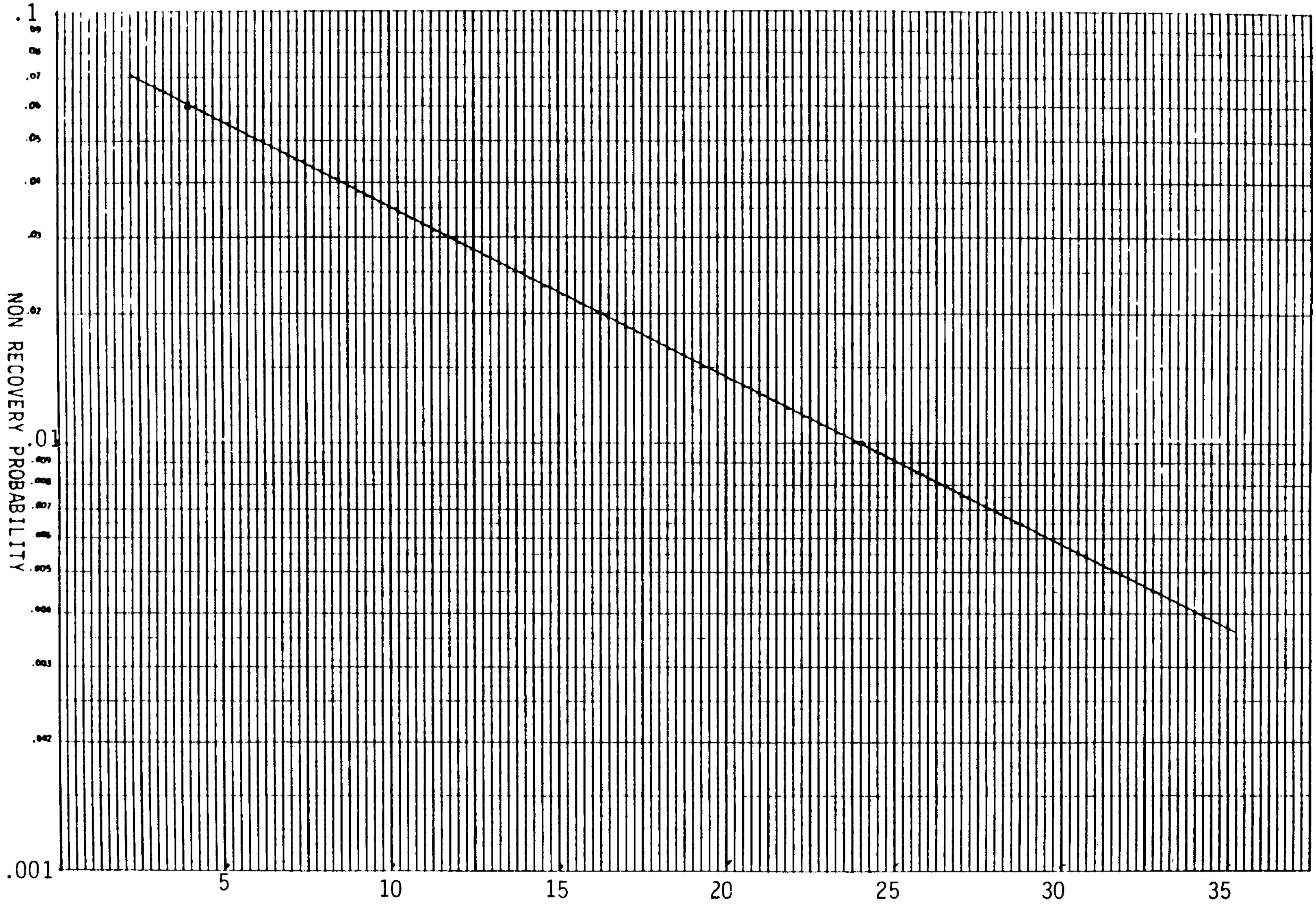


FIGURE D-12 - MSIV NONRECOVERY PROBABILTY FOR THE TW SEQUENCE

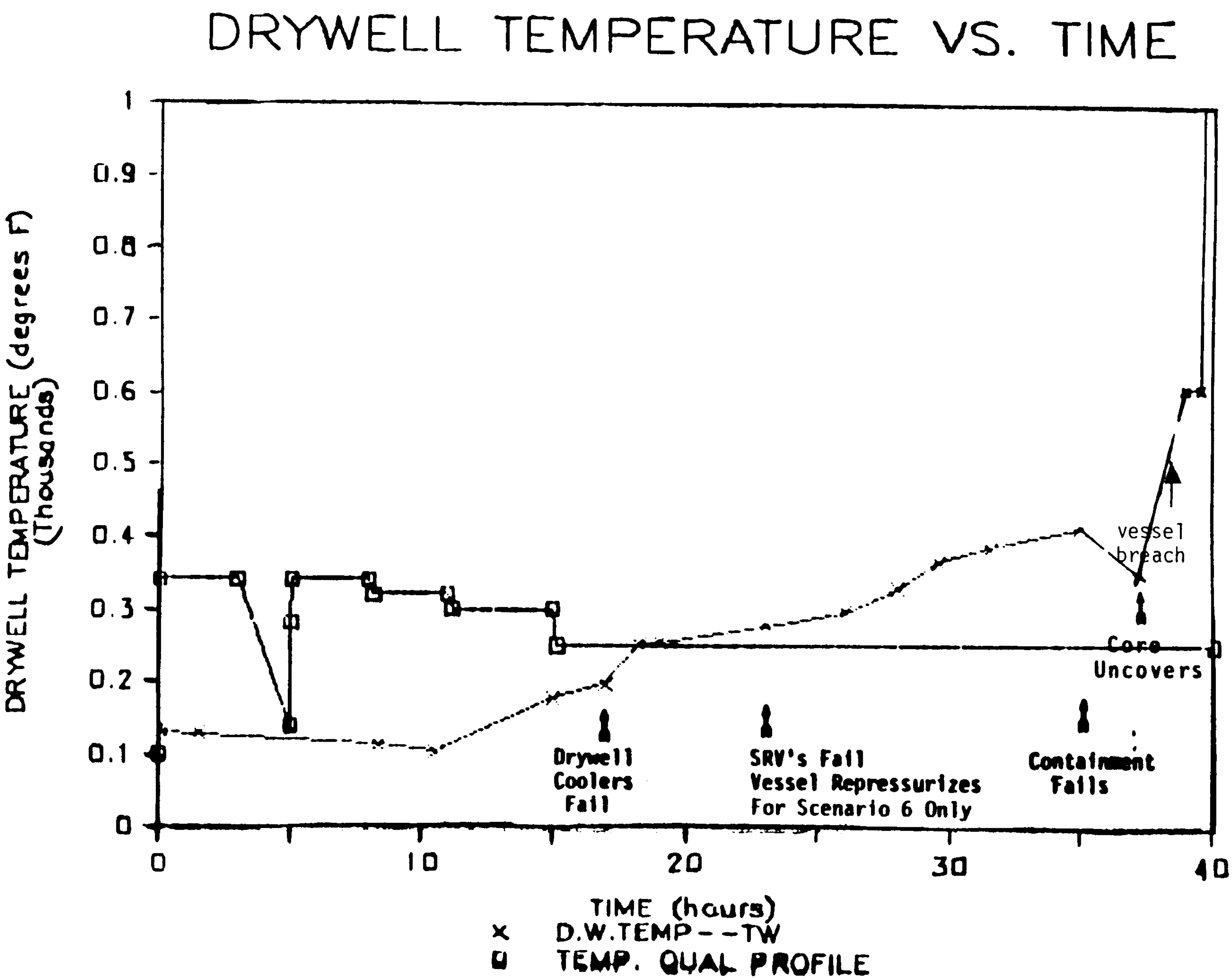


FIGURE D-13 - PROJECTED DRYWELL TEMPERATURE COMPARISON TO QUALIFICATION LEVELS FOR THE TW SEQUENCE

4.5 SRV Air Solenoid Valves in the TC Sequence

For the TC sequence, as explained in section 4.3, PRAs distinguish between MSIV open and shut cases. SRV operation is important in the MSIV shut case so the PRA analysis centers on that scenario. Current PRA projections indicate that initially the HPCI/RCIC systems will operate to maintain water level in the core. This can be expected to continue until the suppression pool temperature reaches about 200°F. Past this point, pump failure can occur due to high lube oil temperatures or low suction head causing possible cavitation in the pump impeller leading to system failure. At this time the operator must be able to operate SRVs in order to obtain the low pressures necessary to use alternate injection systems. Current PRA estimates indicate the probability of failure to depressurize of the order of 0.1 [Ref.28]. Should depressurization failure occur, core melt is likely. Again, PEEESAS results indicate these numbers bear investigation. Figures D-14, D-15, and D-16 show the projected environmental profiles for drywell temperature, suppression pool temperature, and drywell pressure in the TC sequence. These figures indicate that just as suppression pool temperature is approaching the failure level for the high pressure injection systems, the entire containment volume is experiencing elevated environmental levels approaching and then exceeding qualification limits. This suggests a possible SRV environmentally induced failure resulting in the inability to depressurize and engage low pressure injection systems. If environmentally induced failures were considered to change the failure to depressurize probability to 1.0, then the core melt probability would become 1.0 instead of 0.1. This results in a potential factor of 10 increase in the core melt probability.

4.6 SRV Air Solenoid Valves in the TW Sequence

The need for low pressure systems prior to containment failure in this sequence is relatively low. Even if HPCI/RCIC fail due to suppression pool temperature (which is likely to occur) the CRD system is still available and should be adequate to maintain vessel level. Once the containment has been vented or if containment failure occurs, the probability of continued operation of the CRD system is currently estimated at approximately .9 (failure probability = .1) [ref.28]. Should CRD failure occur, then the SRVs would be

DRYWELL TEMPERATURE VS. TIME

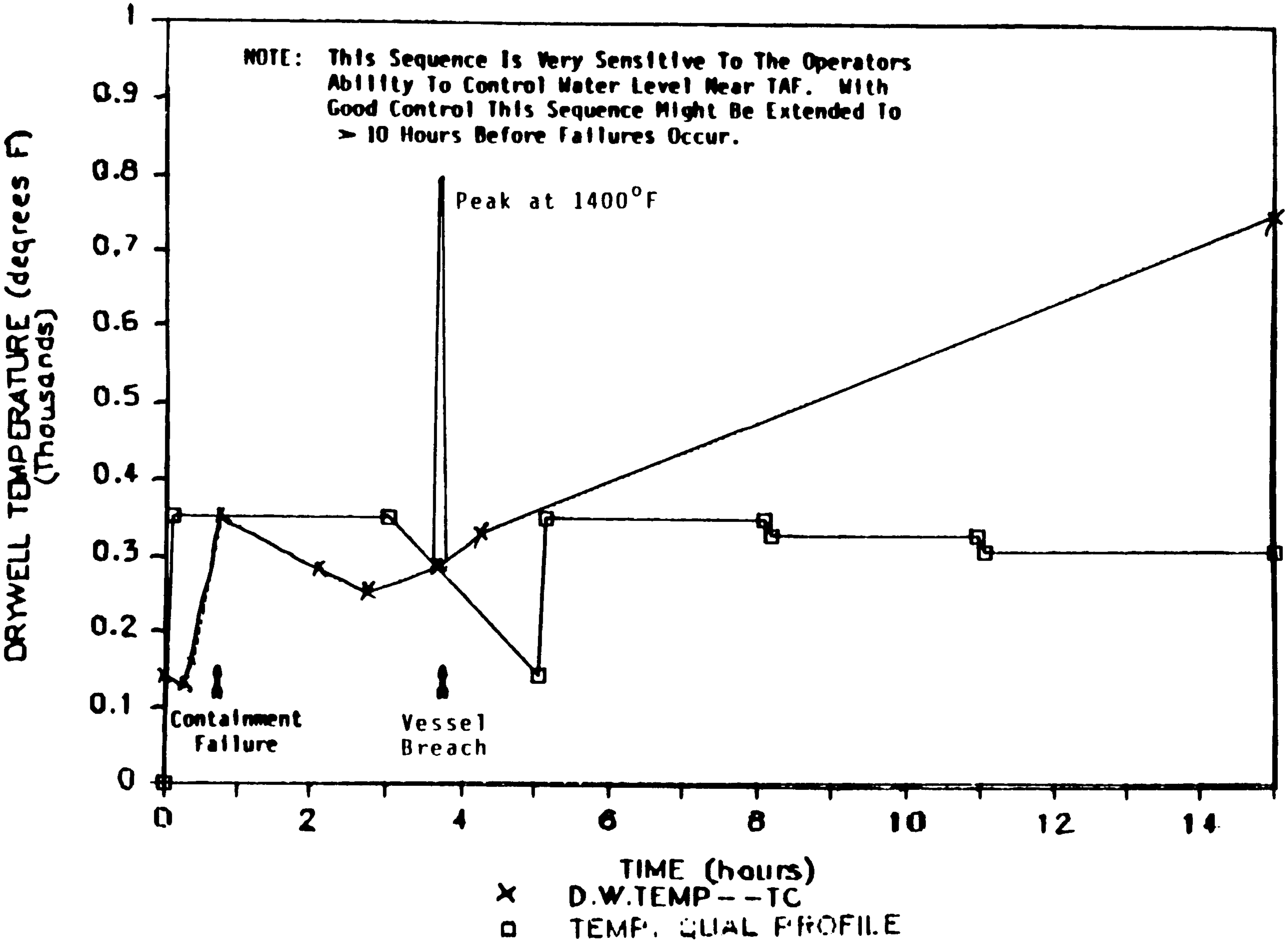


FIGURE D-14 - PROJECTED DRYWELL TEMPERATURE COMPARISON TO QUALIFICATION LEVELS FOR THE TC SEQUENCE

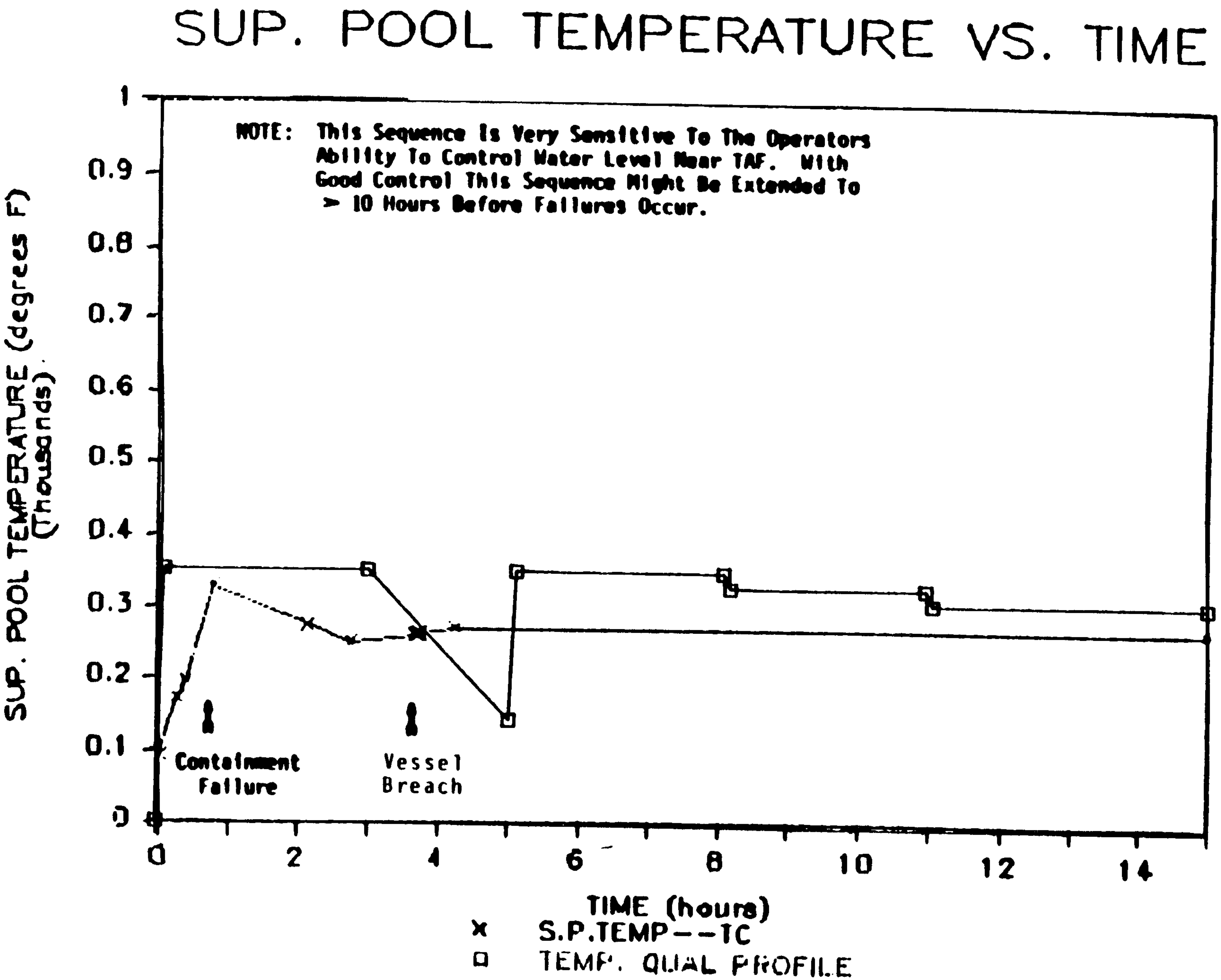


FIGURE D-15 - PROJECTED SUPPRESSION POOL TEMPERATURE COMPARISON TO QUALIFICATION LEVELS FOR THE TC SEQUENCE

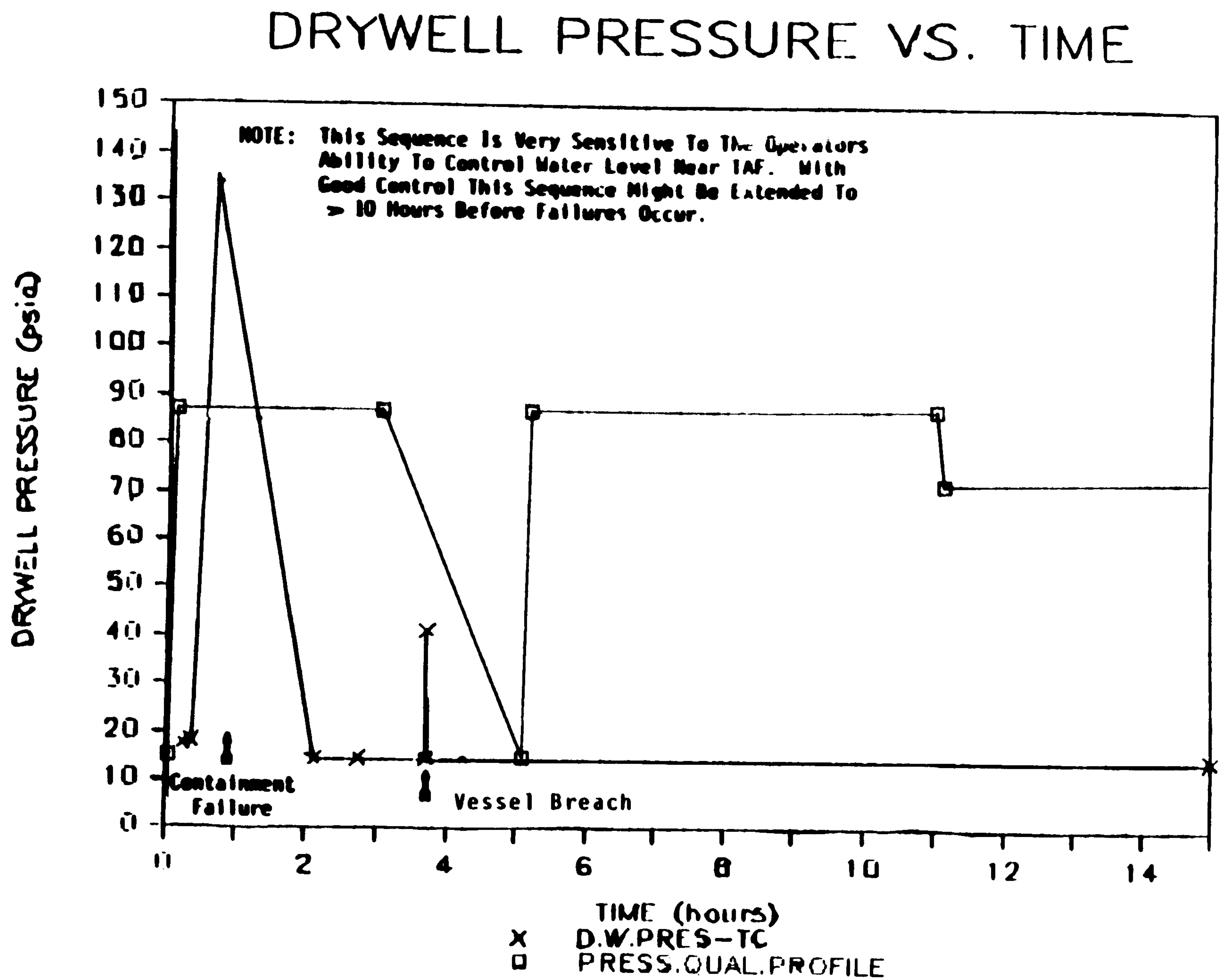


FIGURE D-16 - PROJECTED DRYWELL PRESSURE COMPARISON TO QUALIFICATION LEVELS FOR THE TC SEQUENCE needed in order to depressurize the reactor vessel such that low pressure cooling systems could be used to maintain proper coolant level. As in the TC sequence, containment environmental levels will exceed qualification limits in this sequence raising the question of environmentally induced failure of the SRV air solenoid valves. Current PRA estimates of depressurization failure are less than or equal to .1 given the containment has failed [Ref.28]. If it is found that because of the environments encountered the actual probability of SRV failure is greater than .1 for this sequence, then current sequence probabilities could change by a factor of up to ten. However, since the probability of the TW sequence coupled with a CRD failure are relatively low, the effect of environmental considerations on SRV performance is relatively small.

4.7 Summary

This section has investigated the potential effects of environmentally induced component failure on current PRA estimates. It was shown that both MSIVs and SRVs will experience environments in excess of qualification levels for the sequences of interest and thus are potential candidates for environmentally induced failure. This in turn suggests a possible revision in the current PRA estimates of failure for these components. MSIVs appear to have the greater influence in the TW sequence with the potential to change the sequence probabilities by a factor of five. SRVs are most important in the TC sequence with environmental considerations also having the potential of changing current sequence probabilities by up to an order of magnitude. In addition, MSIV operability can have a significant effect on the risk associated with TC sequences.

The results from this section also lend insight to what environmentally induced failures could mean to overall core melt probability. Based on past IDCOR and ASEP work, the total core melt probability per reactor year for an average BWR-4 design is on the order of 1E-5. TC sequences generally account for about 50% of this total probability, TW sequences 10%, and TB sequences for approximately 40%. The results of this section show that environmental considerations could change TC sequence probabilities by a factor of 10. If the TC sequence accounts for 50% of the total core melt probability, then a factor of 10 increase in the sequence probability implies a factor of 5 increase in overall core melt probability. Likewise, if the TW sequence accounts for 10% of the total core melt probability and the results of this section show a possible factor of 5 increase in this sequence, then this implies a possible factor of 1.5 increase in total core melt probability.

These conclusions indicate that both MSIVs and SRVs are viable test candidates from a PRA perspective. The fact that incorporating environmental failure data for these components has a potential to change current PRA estimates adds additional support to choosing them for the testing phase of this program.

5.0 RECOMMENDATIONS

The recommendations presented represent the best estimate use of limited resources available for testing. The final recommended test candidates were identified in section 4.0 to be the MSIV for the TW or TC sequences and the SRV for the TC accident sequence.

6.0 APPENDIX SUMMARY

This appendix investigated the survivability of electrical equipment in a BWR Mk I containment during severe accident conditions. The goal of this appendix was to analyze data presented in the first three appendices to arrive at a best estimate of which equipment and environments should be chosen for testing. To accomplish this goal, a methodology was developed to systematically examine the factors necessary to make viable test recommendations. Each step of the methodology is discussed in subsequent paragraphs.

Step one of the methodology implemented a three phase screening process to examine critical equipment for projected environments, functional importance, and potential failure modes. Each of the five accident sequences was examined to find out where projected environmental levels would exceed typical qualification levels. These areas were screened against the equipments "useful" period for the sequence and any overlap of these areas was noted as a potential concern. The next phase of the screening process determined which failure modes would dominate in the projected environmental conditions. This was accomplished by a literature review of previous tests and analysis of containment equipment. The final phase of the process determined functional importance of equipment systems based on critical factors including redundancy, noncomplexity, number of back-up systems, fail safe position of the equipment, providing plant status indication only, physical separation of redundant components, and electrical isolation.

The second step of the methodology ranked the three phase screening results. Environmental severity and functional importance were summed together to arrive at a final ranking based on these screening criteria. Failure mode data was added to this ranking to indicate the most likely failure patterns to look for in the environments of concern. Recommended test candidates were chosen based on these results.

The third step of the analysis methodology looked at current PRA estimates to determine how the equipment of interest might affect sequence probabilities. The results from this section formed the final criterion to judge the "best" test candidates. The results showed the MSIV for the TW or TC accident sequence and the SRV for the TC accident sequence to be the recommended test specimens and environments.

7.0 REFERENCES

1. Garrison, William A. (EG&G), "ACCIDENT QUALIFICATION TEST SUMMARY OF QUALIFICATION TESTING PERFORMED BY WYLE LABORATORIES," EG&G, July 1980.

2. KMC, Inc., "EQUIPMENT QUALIFICATION TEST SUMMARY OF SOME OF THE EQUIPMENT TESTED AT FRC," KMC, Inc., July 1980.

3. FRC, "QUALIFICATION TESTS OF ELECTRICAL CABLES UNDER SIMULATED REACTOR CONTAINMENT SERVICE CONDITIONS INCLUDING LOSS-OF-COOLANT ACCIDENT WHILE ELECTRICALLY ENERGIZED," Final Report F-C3798, March 1974.

4. Component Testing Division of Isoomedix, Inc. "RADIATION EXPOSURE SIMULATED LOCA/DBE," Test Report No. IFR-V877-01, August 1977.

5. Witcher, Leroy E. (FRC) and W. H. Steigelmann (FRC), "QUALIFICATION TESTS OF ELECTRICAL CABLES UNDER SIMULATED POST-ACCIDENT REACTOR CONTAINMENT SERVICE CONDITIONS," Final Report F-C2525, October 1969.

6. FRC, "LONG TERM TESTING OF ELECTRICAL CABLES UNDER SIMULTANEOUS EXPOSURE TO GAMMA RADIATION, STEAM AND CHEMICAL SPRAY," Final Report F-C3341, January 1973.

7. FRC, "TESTS OF ELECTRICAL CABLES SUBJECTED TO THERMAL AGING, GAMMA RADIATION AND A LOSS-OF-COOLANT ACCIDENT SIMULATION," Final Report F-C4350-3, July 1976.

8. Carfagno, S.P. (FRC), L.E. Witcher (FRC), and G.R. Boardman (FRC), "TESTS OF ELECTRICAL CABLES UNDER SIMULATED REACTOR CONTAINMENT SERVICE CONDITIONS," Final Report F-C3033, April 1971.

9. The Rockbetos Company, "QUALIFICATION OF FIREWALL III CLASS 1E ELECTRIC CABLES," Copy 12, February 1977.

REFERENCES (CONTINUED)

10. Target Rock Corporation, "ENVIRONMENTAL TEST REPORT ON TRC MODEL 72V TYPE, 1" "Y" PATTERN SOLENOID MOTOR VALVE PER PROVISIONS OF TR TEST PROCEDURE #1383," Report No. 1500, Summer 1974.

11. Pardue, E.F., "VALVE MONITORING SYSTEM SEISMIC AND ENVIRONMENTAL TEST PROGRAM," BWNP-20210-1 (1-78), December 1980.

12. Browns Ferry Nuclear Power Plant, "QUALIFICATION TEST REPORT ON EIGHT CABLE SPLICE ASSEMBLIES TO BE USED AT BROWNS FERRY NUCLEAR POWER PLANT UNIT 3 VOLUME 1 OF 1," WYLE 43854-3, March 1978.

13. Sugerman, Alfred C. (NUTECH Engineers), "AGING-SEISMIC CORRELATION STUDY ON CLASS 1 E EQUIPMENT," NUREG/CR-3808 SAND84-7135, September 1984.

14. Hess, Thomas (LCTL), "QUALIFICATION TYPE TEST REPORT LIMITORQUE DC VALVE ACTUATORS FOR NUCLEAR POWER STATION SERVICE CONDITIONS, PER IEEE 382, 323 and 344," September to November 1975.

15. US Nuclear Regulatory Commission Office of Inspection & Enforcement, "ENVIRONMENTAL TESTS OF ELECTRICAL TERMINAL BLOCKS," SSINS: 6835. March 1982.

16. Meininger, Richard D., "THREE MILE ISLAND TECHNICAL INFORMATION AND EXAMINATION PROGRAM INSTRUMENTATION AND ELECTRICAL SUMMARY REPORT," GEND 050, July 1985.

17. Sandia Laboratories Staff, "STATUS REPORT ON EQUIPMENT QUALIFICATION ISSUES RESEARCH AND RESOLUTION," NUREG/CR-4301 SAND85-1309 RV, June 1985.

18. Jacobus, Mark J., "REVIEW OF SOLENOID AND MOTOR OPERATED VALVE USAGE AND FAILURE DATA IN THE NUCLEAR POWER INDUSTRY," RS/6445/84-06, October 1984.

REFERENCES (CONCLUDED)

19. Kolaczkowski, A.M.(SAIC), Bennett, P.R.(SNL), and Medford, G.T.(SAIC), "INFORMATION REGARDING LIKELY SCENARIOS FOR CERTAIN ACCIDENT SEQUENCES," Science Applications International Corporation, May 1985.

20. Kolaczkowski, A.M.(SAIC), Bennett, P.R.(SNL), and Medford, G.T.(SAIC), "IDENTIFICATION OF ELECTRICAL EQUIPMENT TO BE EXAMINED BY THE PEEESAS PROGRAM," Science Applications International Corporation, June 1985.

21. Medford, G.T.(SAIC), Kolaczkowski, A.M.(SAIC), and Bennett, P.R.(SNL), "GENERATION OF ENVIRONMENTAL PROFILES FOR SELECTED ACCIDENT SEQUENCES," Science Applications International Corporation, August 1985.

22. U.S. Nuclear Regulatory Commission/Federal Emergency Management Agency, "CRITERIA FOR PREPARATION AND EVALUATION OF RADIOLOGICAL EMERGENCY RESPONSE PLANS AND PREPAREDNESS IN SUPPORT OF NUCLEAR POWER PLANTS," November 1980.

23. U.S. Nuclear Regulatory Commission, "REACTOR SAFETY STUDY", NUREG-75/014, WASH-1400, October 1975.

24. Science Applications International Corporation for EPRI, "ATWS: A REAPPRAISAL-PART 3; FREQUENCY OF ANTICIPATED TRANSIENTS," EPRI-NP2230, January 1982.

25. U.S. Nuclear Regulatory Commission, "EVALUATION OF STATION BLACKOUT ACCIDENTS AT NUCLEAR POWER PLANTS," NUREG-1032, May 1985.

26. Accident Sequence Evaluation Program (ASEP) Generic Data Base (not published), Sandia National Laboratories.

27. Emergency Procedure Guidelines for Browns Ferry, June 1984.

28. Ongoing ASEP Analysis of Peach Bottom-2 (a BWR-4, Mark I) for NUREG-1150 based on discussions with Alan Kolaczkowski, Fall 1985.

Appendix E: Test Plan for MSIV Pneumatic Control Manifold Assembly

TEST PLAN FOR MSIV PNEUMATIC CONTROL MANIFOLD ASSEMBLY
Revision 1, 3/5/86

1.0 INTRODUCTION

1.1 Test Objective

The purpose of this test is to evaluate the performance of electrical equipment during a severe accident. The selection of test equipment and severe accident environments was completed as a part of the Performance Evaluation of Electrical Equipment during Severe Accident States program (PEEESAS). These topics are elaborated on in Section 2.0.

1.2 Background of the PEEESAS Program

The Reactor Safety Study and subsequent probabilistic risk assessments have predicted that severe accidents dominate the risk. The Office of Nuclear Reactor Research established a Severe Accident Research Plan to provide an experimental and analytical basis for more accurate assessments of severe accident risks in nuclear power plants. An important part of the severe accident effort is to reduce the many substantial uncertainties in severe accident analysis. One significant source of uncertainty is the lack of data on component performance during a severe accident.

Severe accidents are defined as those which lead to either vessel breach or containment failure and which include the potential for core melt and/or release of radioactivity. (The resulting environment may or may not be more severe than the design basis LOCA environment.)

The objective of the PEEESAS program is to develop information for electrical component performance under severe accident conditions. These components include those which would be used to mitigate an accident or provide information on the status of the plant. In order to evaluate the performance of electrical components, a test program was planned. The test plan is described below and the test will be conducted in FY 86.

1.3 Scope of the PEEESAS Program

During FY85, a method was established to determine which components are important during severe accidents. Browns Ferry Unit 1, a BWR/MARK 1, was used to demonstrate the method. Accident sequences, operator actions likely to occur during those sequences, environmental profiles, and a recommended list of components for testing were determined for this plant.

The component in the test plan is one of the recommended components. One component will be tested in each of two different severe accident environments.

2.0 TEST SPECIMENS AND ENVIRONMENTAL PROFILES

2.1 Basis for Selection of Test Specimens and Environmental Profiles

The test specimen was selected by the following procedure. From work done earlier this year on this program (Appendix B or Ref.. 1), nine components were identified as being important to mitigating accident sequences or providing plant status.

The nine components were further analyzed to determine the best candidates for testing by using a three step process: screening, ranking, and the effect on probabilistic risk assessments (Appendix D or Ref. 2). The screening process identified (1) when the equipment was needed during the accident and whether the severe accident profile was above a typical qualification profile prior to or during that time, (2) failure modes associated with the component, and (3) the functional importance of the component.

The results of the screening process were evaluated by a ranking process. Equipment with high functional importance and high/medium environmental conditions (environmental conditions above the current qualification levels) were retained as possible test candidates. The remaining test candidates were the MSIV (main steam isolation valve) and SRV (safety relief valve) equipment assemblies. Both test candidates were required to operate during two accident sequences: TC (a transient with failure of the reactor protection system) and TW (a transient with failure of the decay heat removal system).

These components were then evaluated for their importance to probabilistic risk assessments. That is, will environmentally-induced failures affect current PRA accident sequence probabilities or the risk associated with these sequences? MSIV environmentally-induced failures may increase the TW accident sequence probability by a factor of five and may increase the risk associated with the TC sequence (for the case where the MSIVs have been assumed to stay open). SRVs are most important in the TC sequence (MSIV closed case) where environmentally-induced failures have the potential to increase the current sequence probabilities by a factor of ten.

An environmentally-induced failure of the SRVs may affect the probability of only one accident sequence; however, an environmentally-induced failure of the MSIV may affect the probability or risk of two accident sequences. Therefore, the MSIVs will be the first testing choice.

Several pieces of equipment are associated with the MSIV equipment assembly: pneumatic control manifold assembly, position switch, main steam drain valve actuator, and globe valve. As discussed in Appendix B or Reference 1, the position switch is less important than the manifold assembly since the manifold assembly is required to operate the MSIV globe valve. The drain valve actuator may be needed to equalize the pressure across the MSIV, but since the MSIV is a globe valve, the globe valve may open even if the pressure across the valve is not equalized. In addition, the heat rejection path associated with the main steam drain valve actuator is smaller than the heat rejection path associated with the manifold assembly. Furthermore the manifold assembly, a complex electrical component, is more susceptible to failure than the globe valve.

Therefore, the MSIV pneumatic control manifold assembly was chosen to be the FY-86 test candidate. Both the TC and TW accident sequence profiles will be used.

2.2 Sample Selected

The MSIV pneumatic control manifold assembly actuates the disc of the main steam line check valve. The MSIV pneumatic control manifold assembly, used at Browns Ferry Unit 1, was manufactured by Automatic Valve Corporation and has three solenoids (250Vdc, 120Vac, and 120Vac). This particular manifold assembly is only used at Browns Ferry and one plant in Japan. However, the same manifold using a 125Vdc solenoid instead of a 250Vdc solenoid is in place in many licensed BWR plants. Therefore, the manifold assembly with the 125Vdc solenoid will be used for this test. Based on discussions with the manufacturer, this component will be purchased for about $18,000 and delivered within three to four months.

The maximum dimensions of the test specimen are as follows: height = 12.5 inches, width = 12.5 inches, and length = 23.2 inches. The approximate weight of the assembly is 100 pounds.

The test specimen has three solenoids (125 Vdc, 120 Vac, and 120 Vac) to operate three valves (4-way, 3-way, and 2-way). The solenoids have Class H insulation and the valves have Viton seals. The valves are lubricated with Parker Super-O-Lube.

The assembly operates in the following fashion: either the 120 Vac or 125 Vdc main control solenoids activate the 4-way valve; if either main control solenoid fails, the 4-way valve may be operated by the other main control solenoid; and if the 4-way valve fails to cause the MSIV to close, the 2-way valve may be used to close the MSIV. The remaining 120 Vac exercise control solenoid operates the 3-way valve. The 3-way valve is normally used to slowly close the MSIV, during normal plant operation, to determine if the MSIV will shut. Although the 3-way valve and exercise control solenoid can only slowly close the MSIV, they may be used if all other values and solenoids fail.

2.3 Environmental Profiles Selected

The environmental profiles for the severe accident portion of the test are shown in Figures E-1 through E-4. As described in Reference 3 or Appendix C, the profiles are based on the LTAS and MARCH computer codes. Figures E-1 and E-2 show the temperature and pressure profiles versus time for the TC sequence until containment failure at approximately 4.5 hours. The temperature and pressure profiles for the TW sequence are shown in Figures E-3 through E-4. For this sequence, containment failure occurs at 35 hours into the accident. For Figures E-1 - E-4, a typical qualification profile is overlayed to show the sections of the severe accident profiles that are above the qualification profile.

For the TC and TW sequences, the MSIV will only be opened prior to containment failure. Since core melt occurs after containment failure, the test specimens need not be exposed to severe accident radiation levels. In addition, the containment spray system is not operated in the TC and TW sequences; therefore, the test specimens will not be exposed to demineralized spray.

3.0 TEST STRATEGY

The MSIV pneumatic control manifold assemblies will be exposed to a simultaneous aging exposure. Then, each assembly will be exposed to an accident profile. In Test #1, the TC accident sequence profile will be used. For this profile, the MSIV is required to remain open until containment failure. However, in Test #2, the valve will start in the closed position and will be reopened periodically during the TW accident sequence profile (until containment failure). After containment failure, the valves (Test #1 and #2) will be exposed to a fragility test with step increases in pressure and temperature. At the completion of the test, the valves must close in both Tests #1 and #2.

Both MSIV pneumatic control manifold assemblies will be subjected to baseline tests at designated points throughout the test. See Section 5.0.

4.0 ACCEPTANCE CRITERIA

The acceptance criteria is based on the operational performance of the MSIV pneumatic control manifold assembly. The assembly must perform its required safety function throughout the accident exposure. Note: the safety function (valves open or close when demanded) varies as a function of time.

4.1 Test #1

For Test #1, the valve must be maintained in the open position throughout the accident exposure. At the conclusion of the test, the valves must reclose and remain in the closed position.

4.2 Test #2

For Test #2, the valve will be closed and must be able to be opened upon demand during the accident exposure. At the conclusion of the test, the valves must reclose and remain in the closed position.

5.0 BASELINE TESTS AND HEAT RISE MEASUREMENTS

5.1 Baseline tests

Baseline tests, representative of typical solenoid valve qualification tests (Ref.. 4, 5, 6, and 7), will be conducted prior to aging, after aging, and after the accident exposure. These tests will only be used to gain further insight into the performance of the valves. The following baseline tests will be conducted.

1. Operational tests at minimum and maximum rated voltages (rated voltage $\pm$ 10 percent) and minimum and maximum pressures (150 $\pm$ 10 psig). This pressure, 150 $\pm$ 10 psig, is the upper limit provided by the instrument air system.

   When the applied inlet pressure is between the minimum and maximum operating pressure differentials, the valve must shift to the energized position (open) upon application of a voltage within the voltage limits. The valve must shift to the de-energized position (closed) upon removal of the applied voltage.

2. External leakage of valve bodies

While energized, apply a bubble solution to all joints and pressurize the valve to a safe working pressure.

3. Seat leakage

In the de-energized state, check the valve for seat leakage at the minimum and maximum operating pressures.

4. Insulation resistance measurements

Measure the insulation resistance after 1 minute at 500 V.

5. High potential withstand test

Measure the leakage current, after 1 minute, at twice the rated voltage plus 1000 Vac. This test will only be done at the conclusion of the test.

5.2 Heat Rise Measurements

The heat rise due to the solenoid assembly being energized (heat rise from the coil) will be measured. When the solenoids are energized at the service temperature and the temperature has stabilized for two hours, the heat rise temperatures will be recorded. After these measurements are made, the thermal aging calculations will be reviewed to ensure that the heat rise temperature of 27-30°C was appropriate (Ref.. 9). Section 8.1 explains the use of the heat rise measurements.

6.0 MONITORING DURING THE TEST

The test specimen and the test environment will be monitored throughout the test.

6.1 Monitoring of the Test Specimen

In order to monitor the operation of the manifold assembly, the cycle rate, the on/off time, the cycle count, the pressure at the valve inlet and exhaust cylinder ports, and the supply voltage will be monitored. These parameters are representative of typical solenoid valve qualification tests (Ref. 4, 5, and 7).

For the heat rise measurements, thermocouples will be placed at the following locations: (1) top and bottom seats of the plunger of the 125 Vdc solenoid, (2) inside the NEMA 4 box as close to the neoprene gasket as possible, (3) inside the solenoid assembly housing of the 125 Vdc solenoid as close as possible to the coil, and (4) on the silicone lead wire of the 125 Vdc solenoid as close as possible to the solenoid coil.

6.2 Monitoring of the Test Environment

The test environments will be monitored using thermocouples positioned throughout the steam chamber and near the test specimen. (Two differential thermocouples will be connected directly to the test specimen.) The pressure in the test environment will also be monitored.

Automated measurements of temperature and pressure during the accident exposure will be made at five minute intervals. Continuous chart recording will provide backup capability.

7.0 MOUNTING AND CONNECTIONS

As shown in Figure E-5, the test specimen will be mounted at a forty-five degree angle with the solenoids upside down during the accident exposure. This is the usual installed configuration at nuclear power plants.

Electrical lead wires and piping connections for energizing, pressurizing, and monitoring shall be attached according to manufacturing specifications and plant installation requirements. The wires and piping connections will be brought out of the chamber and connected to monitoring equipment for recording.

8.0 TEST DESCRIPTION

8.1 Aging Simulation

The aging simulation will consist of simultaneous radiation and thermal aging. The solenoids will be energized throughout aging; however, no cycling of the valve will be done during the aging simulation.

The manufacturer recommends replacing some of the organic materials of the valve after 15 months. Therefore, the valve will be aged to an equivalent of 15 months.

The radiation exposure will be 1.25 Mrads at the lowest dose rate available at the HIACA facility (approximately .09 Mrad/hr.). The radiation damage threshold is approximately 1 Mrad for several of the materials in the manifold assembly and Viton is susceptible to dose rate effects.

The maximum service temperature was given as 85°C. Self-heating of the coil will raise the service temperature to

112°C. Since the solenoids are isolated from the rest of the valve, only the solenoids need to be analyzed using the higher service temperature. In order to avoid over-aging any part of the valve, part of the aging of the solenoids will done separately. Then the solenoids will be reinstalled into the valve to complete aging. The two-step aging process was used in Refs. 5, 6, and 7 in order to avoid over-aging the elastomeric materials. The thermal aging calculations are found in Attachment A.

First, the solenoids will be thermally aged at 130°C (160°C including self-heating) for 4 days. An activation energy of 1.0 eV was used. Then the solenoids will be replaced and the entire assembly will be exposed to 130°C (160°C for the solenoids because of self-heating) for 12.2 days and 1.25 Mrads. For this portion of the aging calculation, an activation energy of 1.0 eV was used.

8.2 Accident Simulation

As required during the accident simulation, the solenoids will be energized (rated voltage ± 10 percent) and the valves will be pressurized at the inlet ports with instrument air. The applied inlet pressure will be 150 ± 10 psig.

The test valves will be energized at the rated voltage for a minimum of four hours, at 85°C, to produce thermal saturation of test valve coils and to simulate the maximum typical temperature prior to an accident (Ref. 7).

8.2.1 Test #1

The valve must be energized until Test #1 has been completed. The temperature and pressure profiles for Test #1 are shown in Figures E-6 and E-7. These profiles correspond to the TC accident sequence, until containment failure at 4.5 hours. If the valve is still operable at containment failure (remains open), the pressure and temperature will be increased in steps to determine the fragility level of the manifold assembly.

The following procedure will be used during the fragility portion of the profile. The environmental temperature will be increased in 25°F increments and held at that temperature until the valve has stabilized at the environmental temperature for ten minutes. The temperature will continue to be increased in 25°F increments until the valve fails to remain open. The minimum differential pressure to open the manifold assembly is 28 psig (Ref. 9). Therefore, the chamber pressure will be increased in 5 psig increments until the pressure reaches a maximum pressure of 132 psig.

At the conclusion of the test, the valve will be cycled (if necessary) to the closed position. The valve must close, at that time, to perform its required safety function.

8.2.1 Test #2

The profiles for Test #2 are shown in Figures E-8 and E-9. This profile corresponds to the TW accident sequence until containment failure at 35 hours. The valve will be cycled every 2 hours during the profile until containment failure. The valve will be open long enough to allow for observation and then the valve will be de-energized. If the valve can still be cycled from the closed position to the open position at containment failure, the pressure and temperature will be increased to determine the fragility level of the manifold assembly.

The following procedure will be followed during the fragility portion of the profile. The valve will be cycled at each fragility plateau. The valve will be open long enough to allow for observation and then the valve will be de-energized. The temperature will be increased in 25°F increments and held at that temperature for 10 minutes after the valve has stabilized at the environmental temperature. The temperature will continue to increase in 25°F increments until the valve fails to open. The pressure will be increased in 5 psig increments until the pressure reaches 132 psig. Since the minimum differential pressure to open the manifold assembly is 28 psig (Ref. 9), the maximum chamber pressure may be 132 psig.

At the conclusion of the test, the valve will be cycled (if necessary) to the closed position. The safety function of the valve is to close at that time.

9.0 TEST FACILITIES

9.1 HIACA: Radiation and Thermal Aging Facility

The accident simulation test will take place in Sandia's High Intensity Adjustable Cobalt Array (HIACA). A complete description of the HIACA is contained in Reference 8. A brief description follows.

The HIACA consists of thirty-two 24-inch long Cobalt-60 source pencils. The pencils are arranged in a circle, giving a cylindrical test volume that supplies a relatively uniform dose to test specimens. The array is adjustable: the pencils can be moved to accommodate larger test specimens at lower dose rates or smaller specimens at higher dose rates. Dosimetry runs will be made prior to the experiment.

9.2 Steam System: Accident Simulation

Capabilities of the steam system are given in Table 1. The system is designed to accommodate severe accident testing.

TABLE 1: General Capabilities of the Steam System

Design (system and vessels)	
Pressure	200 psig
Temperature	750°F
Accumulators	16 $ft.^3$
Total volume	16
Water volume	8
Saturated steam ramps--	
12,000 lb. peak (16 $ft.^3$ chamber with ID=20.5in)	1 chamber
0-70 psig (314°F sat.)	10 sec.
0-110 psig (343°F sat.)	22 sec.
Superheat steam ramps	
ramp (pressure to 70 psig)	to 750°F
continuous	$\leq$ 10 minutes
	300 lb./hr.

9.3 Test Chamber

The test chamber to be used for this test will depend on availability. If possible, simultaneous radiation and thermal aging and the accident simulation will take place in the same chamber so that handling of the test specimen will be minimized.

9.4 Instrumentation

Appropriate instrumentation will be employed to monitor the test environment and the test specimen response. The test facility is equipped with a variety of data acquisition systems capable of monitoring and recording the necessary pressures, temperatures, ramp times, and other pertinent data. The instrumentation channels for monitoring the test specimen response will be on a separate data acquisition system, which can be interfaced to a computer to facilitate analysis.

10.0 QUALITY ASSURANCE

Quality assurance will be handled in accordance with QAP 6447-2, Revision A. (QAP 6447-2, Revision A is found in Attachment B.) Some specific areas addressed by the QAP are described below.

10.1 Test Specimens

The purchase order will specify that a Certificate of Compliance/Conformance accompanies the test specimens to ensure that the test specimens are Class 1E qualified in accordance with appropriate requirements. The Certificate must indicate the nuclear standards to which the equipment has been qualified, including 10 CFR 50 Appendix B, NUREG-0588, IEEE-382, IEEE-344, and IEEE 323-1974. The manufacturing lot and data codes are also required to appear on the certificate.

10.2 Documentation Control

The documents will provide an auditable trail of information describing test specimens, any preparation of the test specimens, test configuration, test environments, procedures, test results, analysis methodology, and data analysis. The following information will be stored in the appropriate files: data sheets, log books, drawings, calibration data, plots, photographs, datalogger paper tapes, hard copy printouts of computer data files, datalogger and computer channel configuration documents, test plan, and final report.

The final report will be subject to peer review for technical accuracy and management review and approval prior to issuance.

10.3 Photographs

Prior to testing, color photographs will be taken of the test setup, instrumentation, test chamber, and test specimen. These will include both general overview shots as well as close-ups of pertinent details. After the test, similar photographs will be taken of the test specimen. Photographs of the test specimen will be taken during set-up of the test, during testing if warranted, after the test, and during disassembly of the test.

10.4 Nonconformance/Unusual Occurrences

In the event of a nonconformance, the nonconformance will be identified in the lab notebook. This identification will include the date and time of the occurrence, nature of the deviation, expected or planned occurrence or procedure, magnitude of the deviation from the planned procedures, effect of the deviation on the test and test results, disposition of the test items, and any corrective action required.

10.5 Equipment Calibration

The principal investigator will assure that test and measurement equipment used during the test will have valid calibration stickers and that calibration will not expire during the planned course of the experiment (calibration against standards which are traceable to the National Bureau of Standards). If the time required for testing is modified due to unanticipated occurrences and the calibration of the equipment expires during the test program, the principal investigator may elect to continue using the equipment. In such a case, the equipment will be calibrated after completion of the test. The principal investigator will analyze the results of the post-test calibration to ascertain the effect on test results.

11.0 REFERENCES

1. PEEESAS Task 3 Letter Report to R. Feit, Identification of Electrical Equipment to be Examined by the PEEESAS Program, July 8, 1985.

2. PEEESAS Tasks 5, 6, and 7 Letter Report to R. Feit, Data Analysis and Final Test Recommendations, December 12, 1985.

3. PEEESAS Task 4 Letter Report to R. Feit, Generation of Environmental Profiles for Selected Accident Sequences, August 1, 1985.

4. IEEE Standard for Qualification of Safety-Related Valve Actuators, IEEE 382-1980.

5. Draft of A Comparison of Three Test Programs for ASCO Solenoid Valves, FRC Report No. P-C5260-3013, Franklin Research Center, November 25, 1981.

6. Equipment Qualification Research: Test Program and Failure Analysis of Class 1E Solenoid Valves, NUREG/CR-3424, Franklin Research Center, November 1983.

7. Qualification Specification for Automatic Switch Co. (ASCO) Catalog NS-2 Solenoid Valves, Specification #AQS-21682, Rev. 0, Automatic Switch Company, September 20, 1984.

8. William H. Buckalew and Frank V. Thome, Radiation Capabilities of the Sandia High Intensity Adjustable Cobalt Array, NUREG/CR-2582, SAND81-2655, Sandia National Laboratories, Albuquerque, NM, March 1982.

9. R. Bell and T. Akos, "Lessons Learned in the Environmental Qualification of Class 1E Equipment at Tennessee Valley Authority: Qualification Testing of a Main Steam Isolation Valve Control Manifold Assembly by Wyle Laboratories," TVA/PUB--85/10, DE85 900869, Tennessee Valley Authority, Knoxville, Tennessee, presented at the IEEE 1984 Nuclear Science Symposium, Orlando, Florida, November 1, 1984.

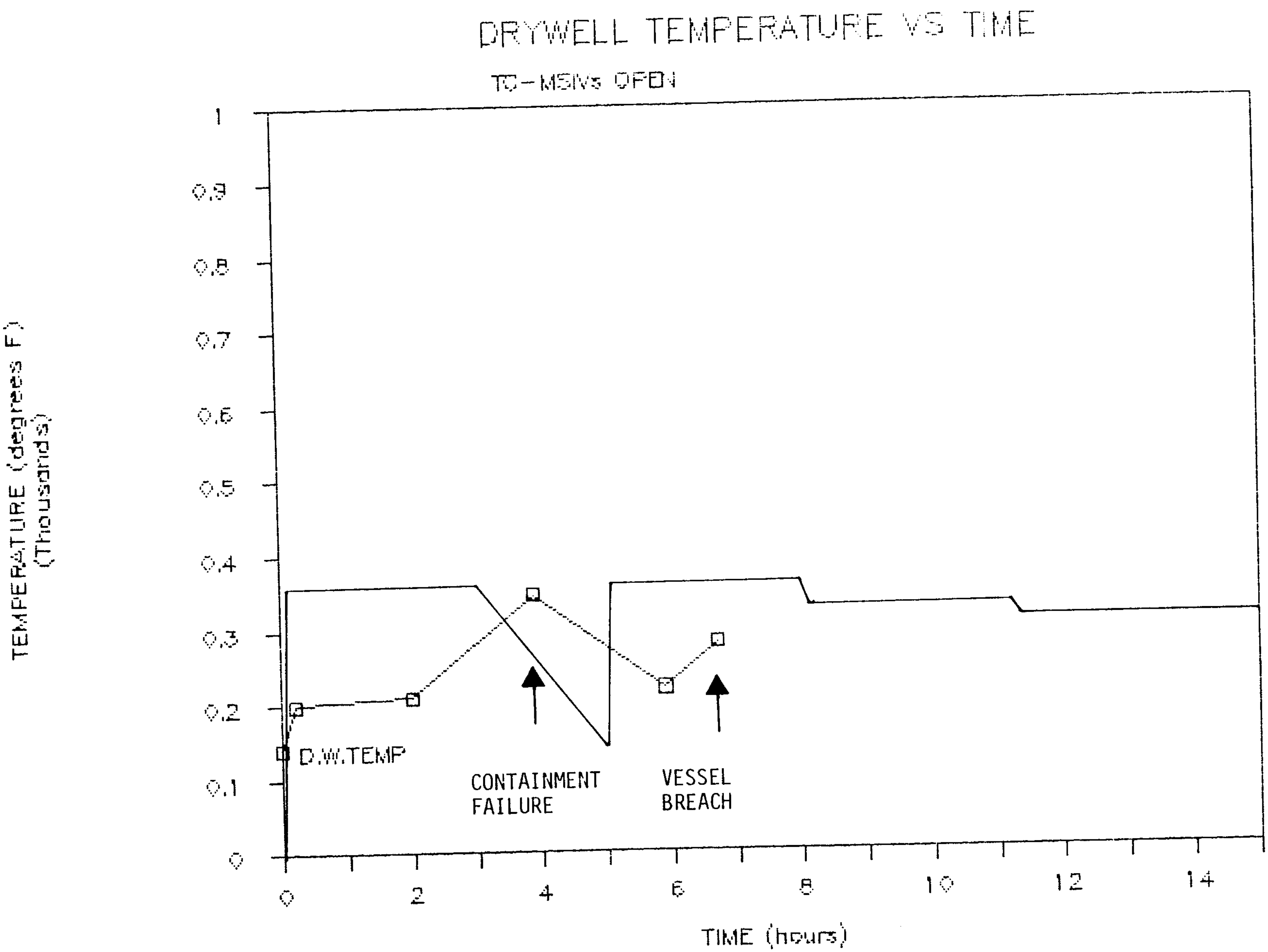


FIGURE E-1 - ENVIRONMENTAL PROFILE COMPARISON TO QUALIFICATION LEVELS FOR DRYWELL TEMPERATURE

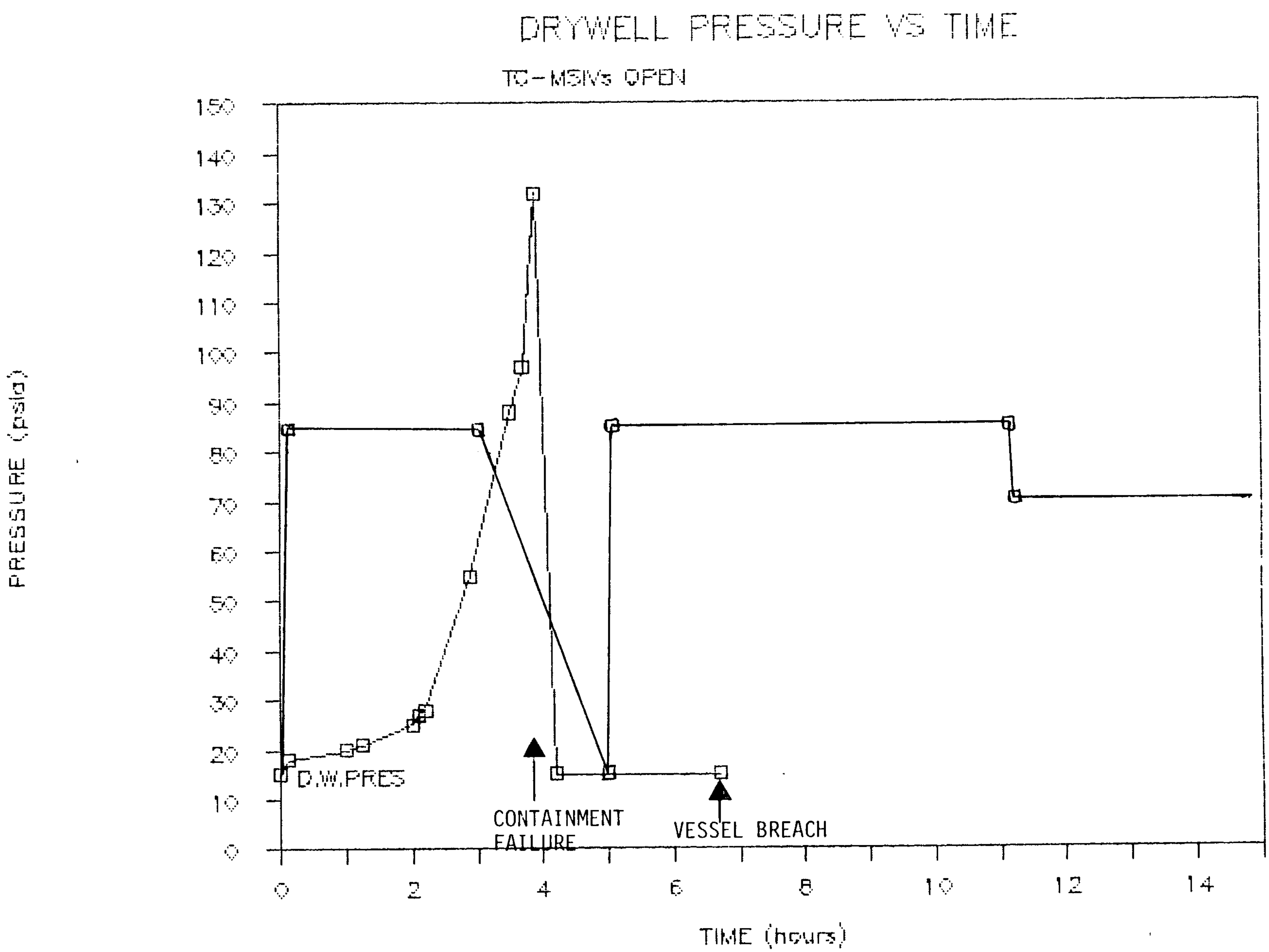


FIGURE E-2 - ENVIRONMENTAL PROFILE COMPARISON TO QUALIFICATION LEVELS FOR CONTAINMENT (DRYWELL) PRESSURE

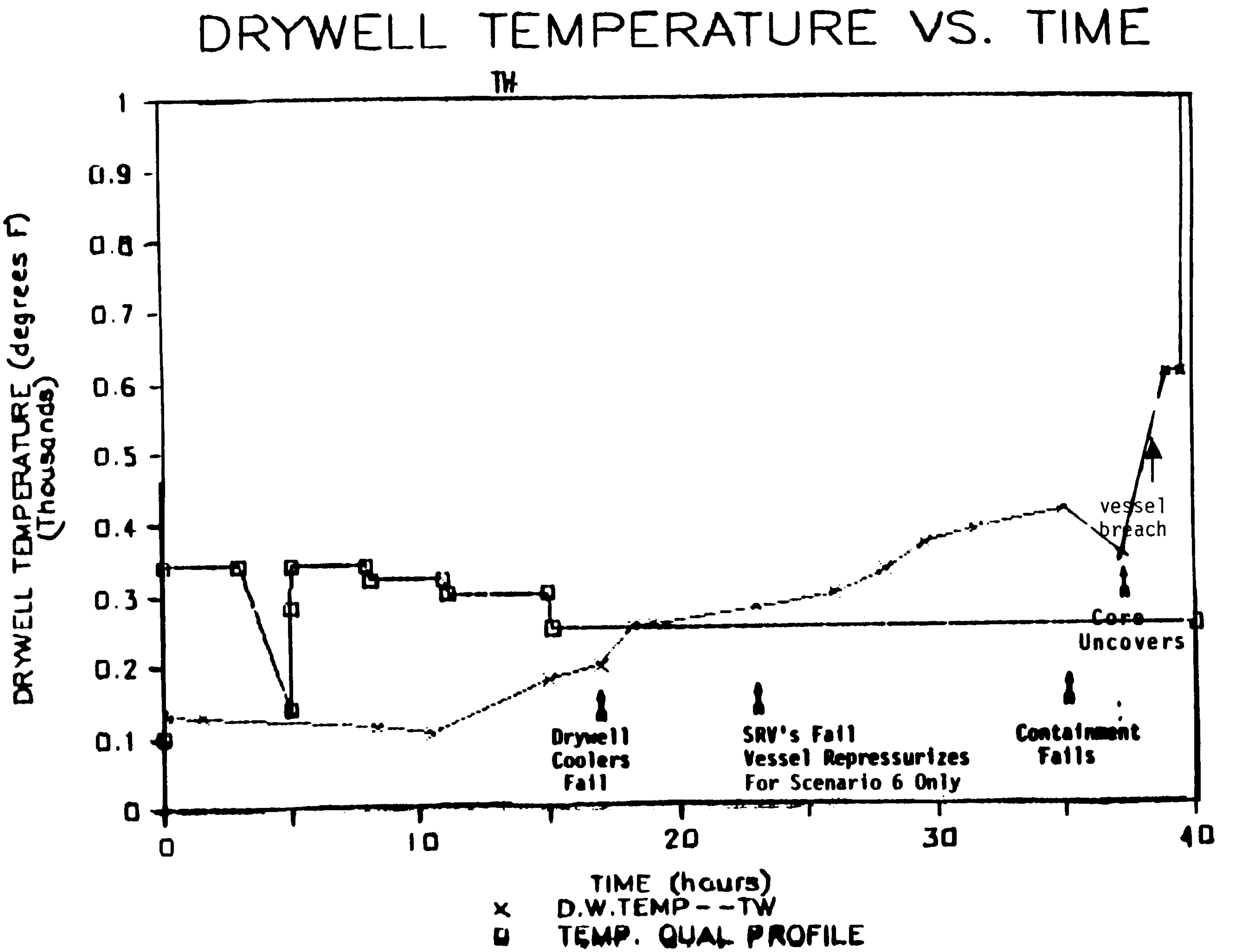


FIGURE E-3 ENVIRONMENTAL PROFILE COMPARISON TO QUALIFICATION LEVELS FOR DRYWELL TEMPERATURE

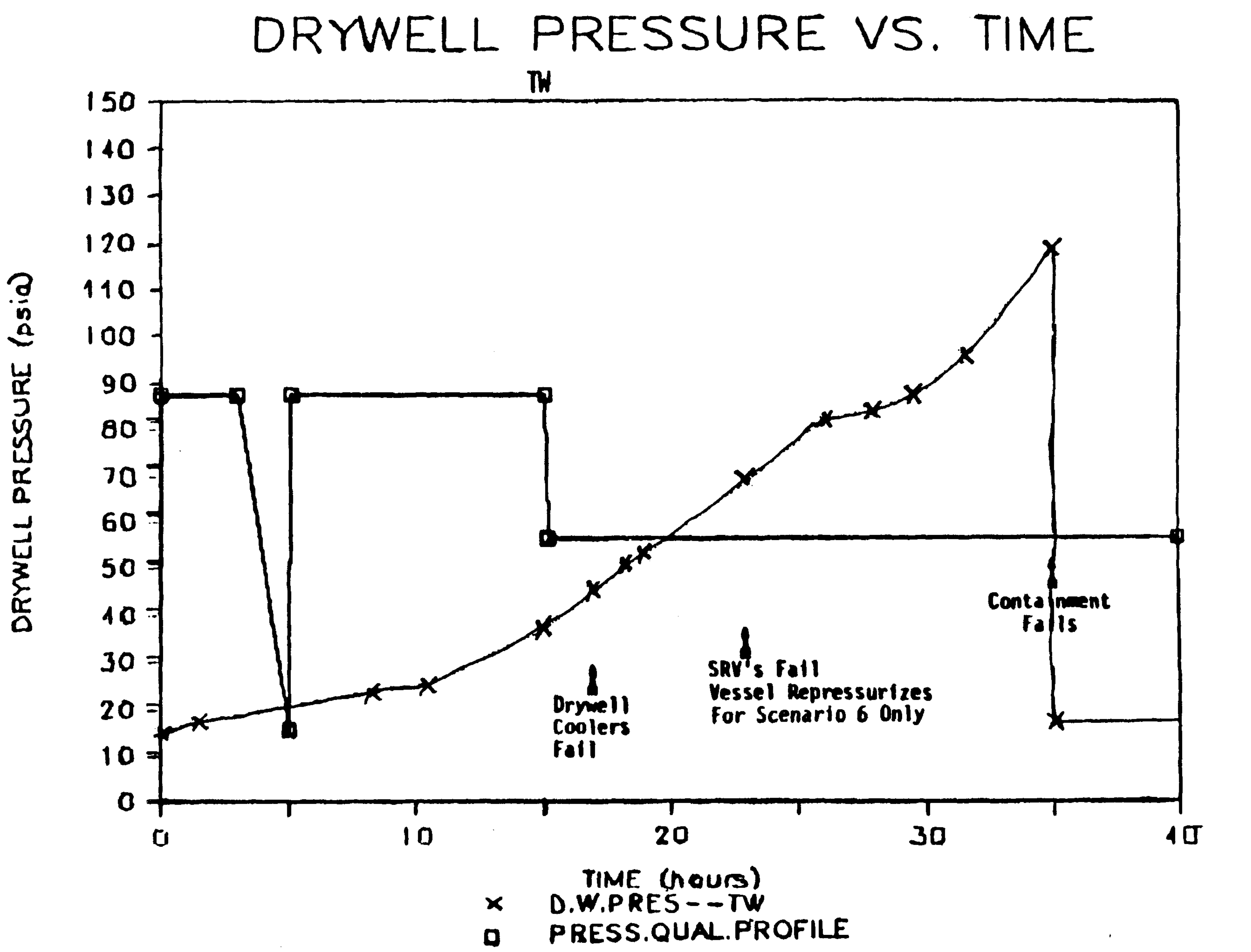


FIGURE E-4- ENVIRONMENTAL PROFILE COMPARISON TO QUALIFICATION LEVELS FOR CONTAINMENT PRESSURE

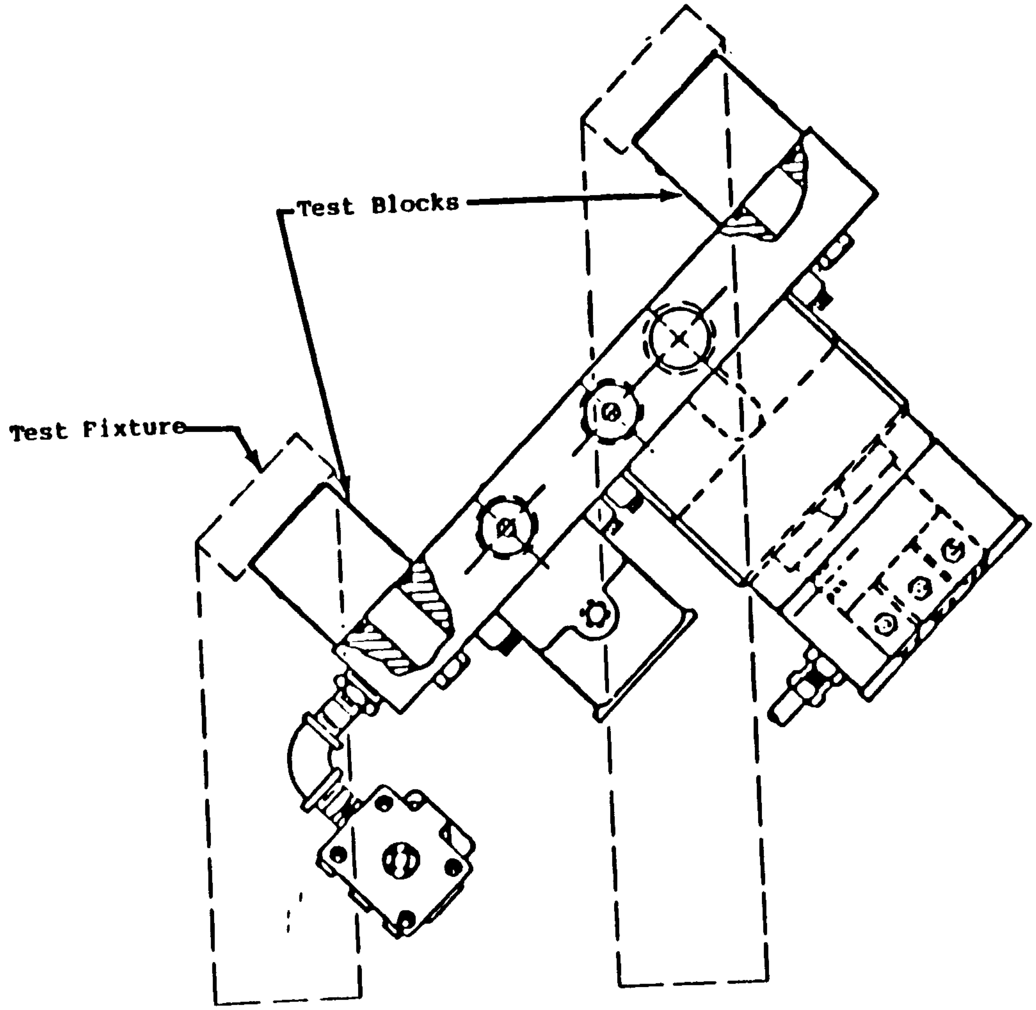


FIGURE E-5 TYPICAL MOUNTING CONFIGURATION

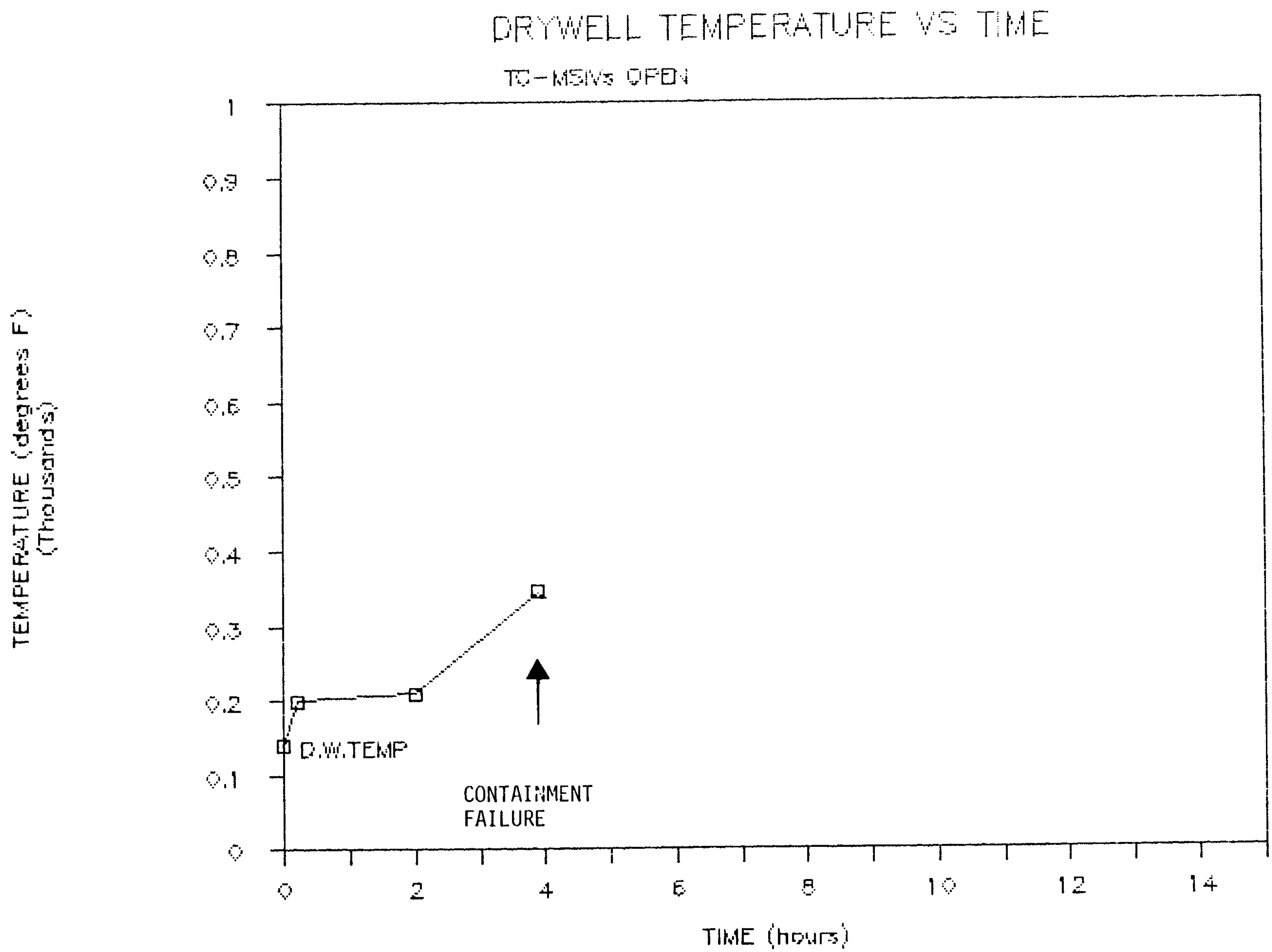


FIGURE E-6 DRYWELL TEMPERATURE

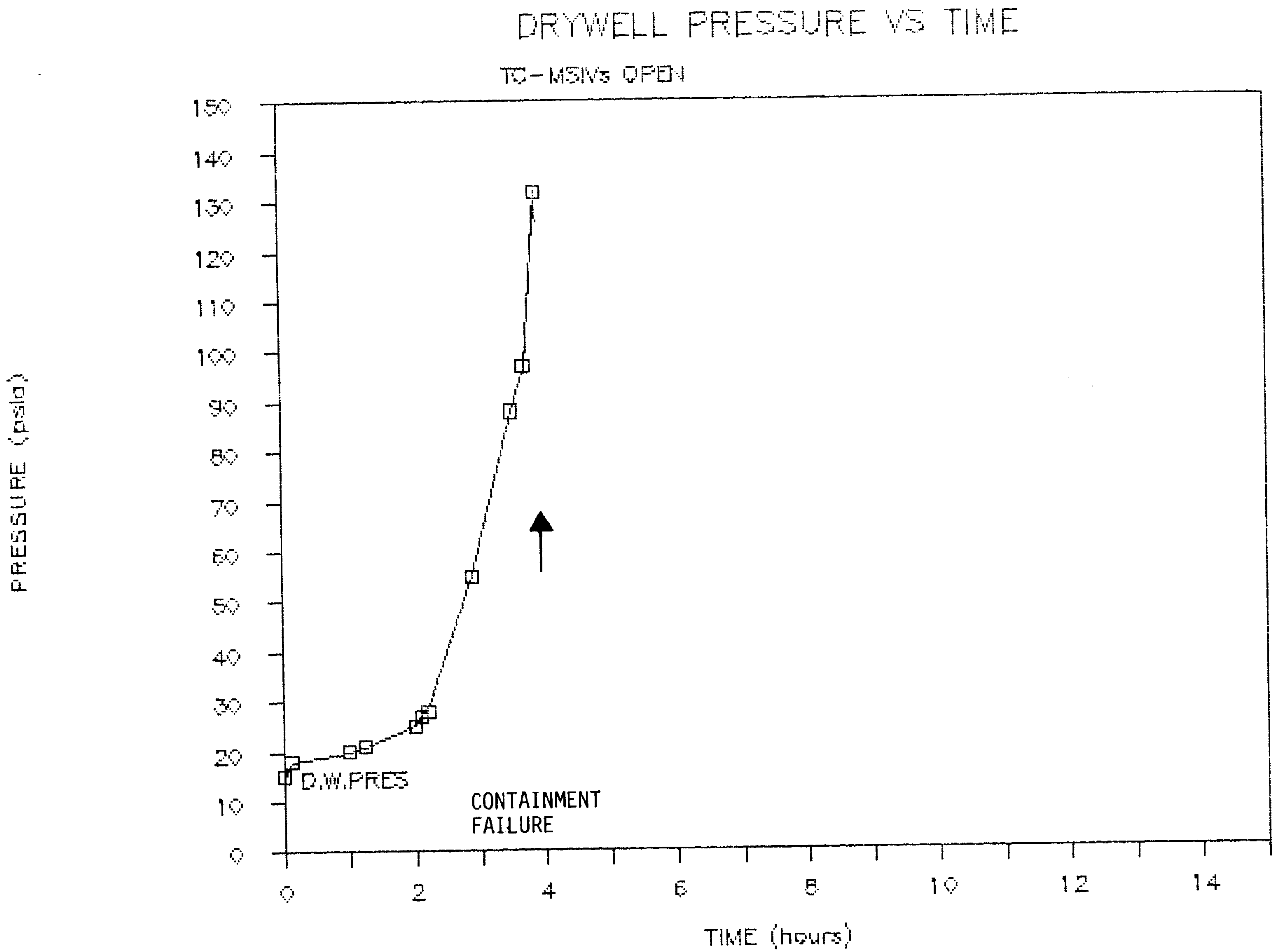


FIGURE E-7 - CONTAINMENT (DRYWELL) PRESSURE

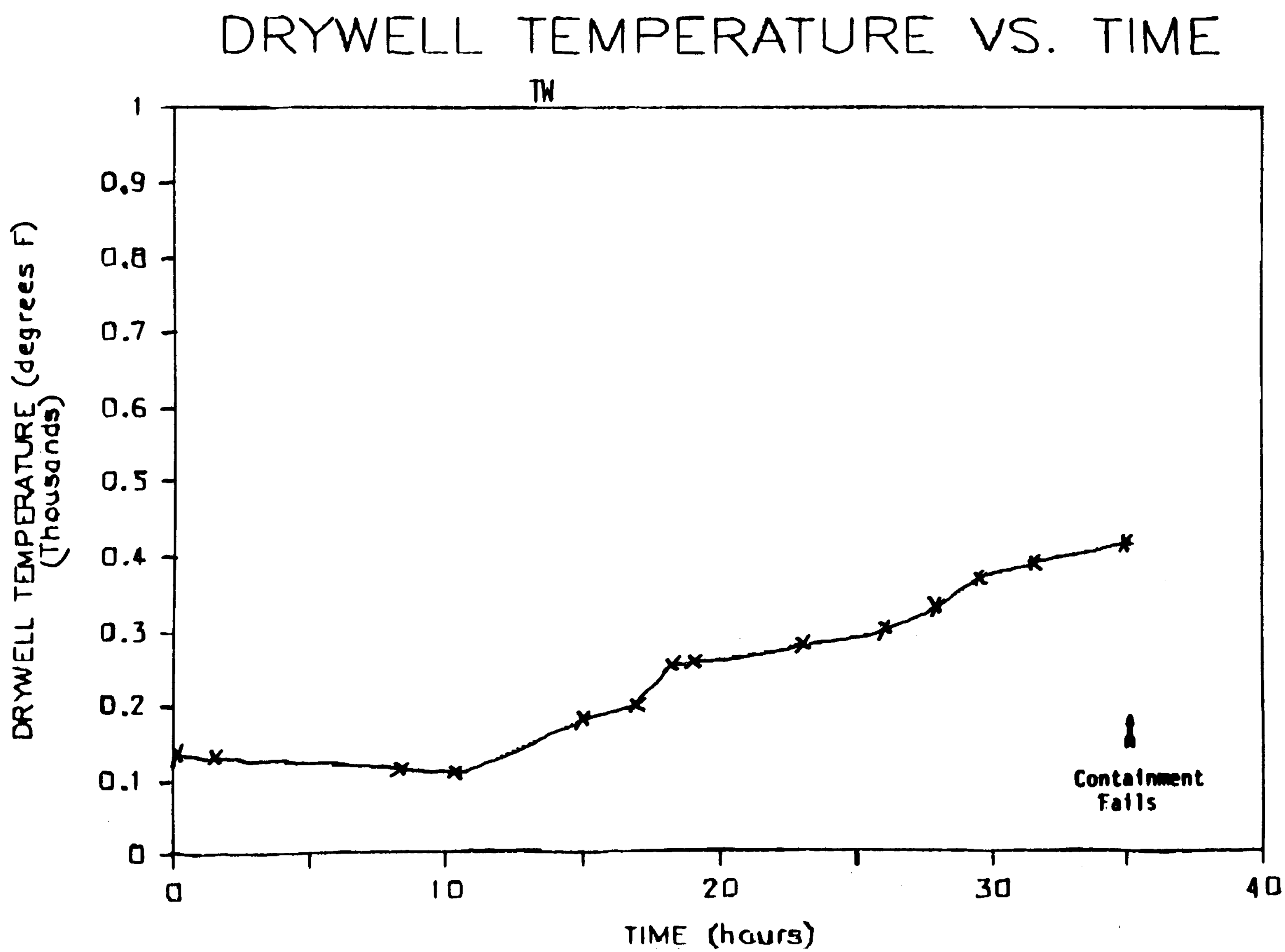


FIGURE E-8 - DRYWELL TEMPERATURE PROFILE

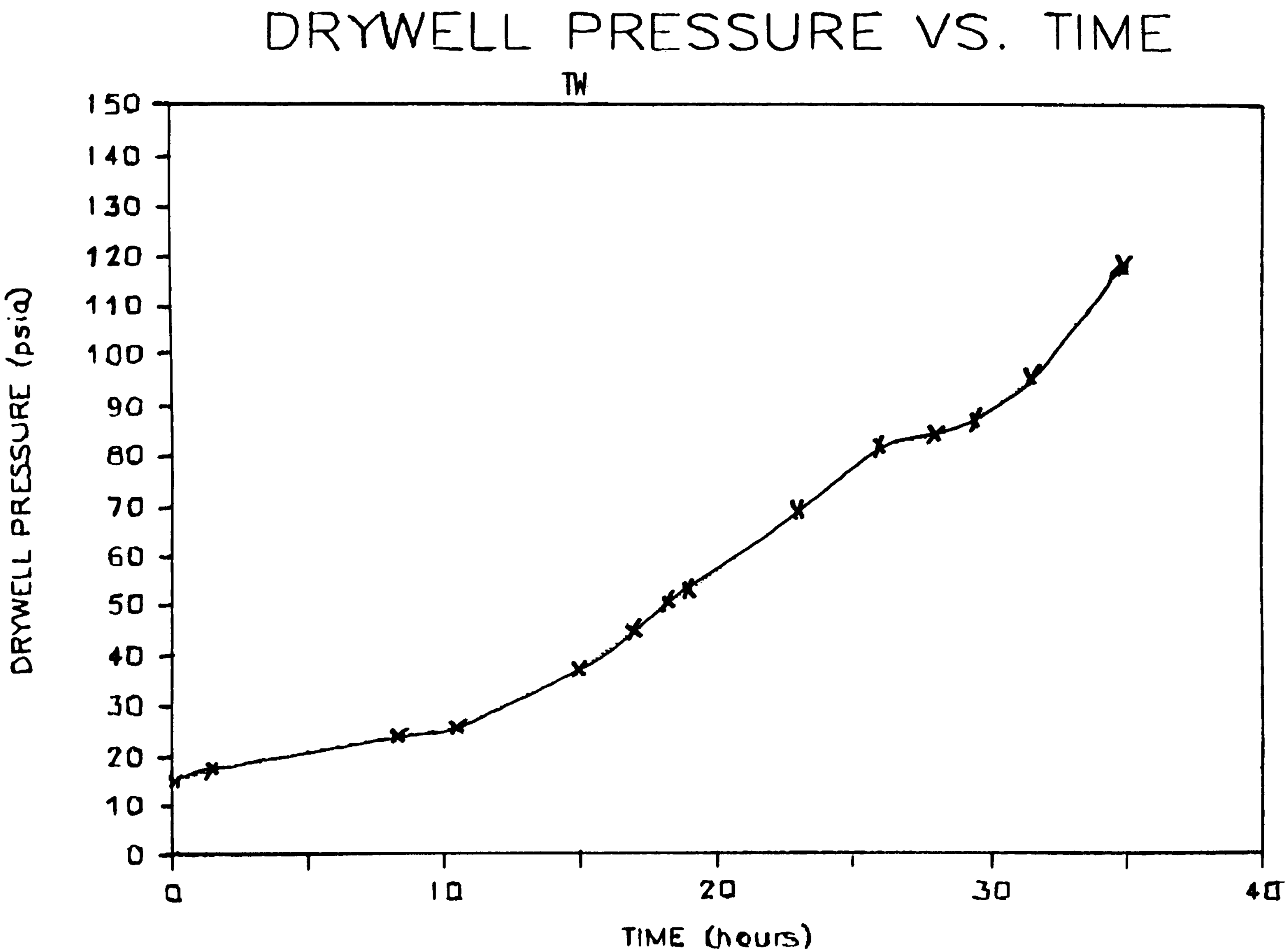


FIGURE E-9 - CONTAINMENT PRESSURE PROFILE

ATTACHMENT A

THERMAL AGING CALCULATIONS

1. Given Conditions:

a. Service Time

t_s = 15 months (manufacturer's suggested replacement period)

b. Service Temperature

For all components but the solenoid:

T_s = 185 F = 85 C = 358 K

For the solenoid:

T_s = 185 F + ΔT (ΔT due to self-heating)

T_s = 85 C + 27 C = 112 C = 385 K

c. Activation Energies

Viton: E_a = 1.11 eV

Assume for solenoid: E_a = 1.0 eV

2. Accelerated Aging:

a. Calculation for all components but solenoids (manifold assembly energized)

T_s = 358 K

t_s = 15 months

E_a = 1.0 eV

$k_B = 1.38 \text{ E-23 } \frac{J}{K}$

1eV = 1.602 E-19 J

T_a = 130 C = 403 K (130 C used in a previous test of AVCO manifold assembly)

t_a = ?

$$\frac{t_a}{t_s} = \exp\left[\frac{-E_a}{k_B}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right]$$

$$t_a = (15\text{ months})\exp\left[-\frac{1.0\text{ eV}}{1.38\text{ E-}23\ \frac{J}{K}}\left(\frac{1.602\text{ E-}19J}{1\text{ eV}}\right)\left(\frac{1}{358K}-\frac{1}{403K}\right)\right]$$

t_a = .40 months = 12.2 days

b. Calculation for solenoids

1. When assembled in manifold assembly and energized

T_a = 130 C + 30 C

= 160 C = 433 K

t_a = 12.2 days

E_a = 1.0 eV

k_B = 1.38 E-23 $\frac{J}{K}$

1 eV = 1.602 E-19 J

T_s = 385 K

t_s = ?

$$\frac{t_a}{t_s} = \exp\left[-\frac{E_a}{k_B}\left(\frac{1}{T_s}-\frac{1}{T_a}\right)\right]$$

$$t_s = \frac{12.2\text{ d}}{\exp\left[-\frac{1.0\text{ eV}}{1.38\text{ E-}23\ \frac{J}{K}}\left(\frac{1.602\text{ E-}19\text{ J}}{1\text{ eV}}\right)\left(\frac{1}{385\text{ K}}-\frac{1}{433\text{ K}}\right)\right]}$$

t_s = 345.1 d = 11.3 months

2. Therefore, the solenoid must be aged (and energized) for an additional t_s = 15 - 11.3 months = 3.7 months

T_s = 385 K

t_s = 3.7 months

E_a = 1.0 eV

$k_B = 1.38\ E{-}23\ \frac{J}{K}$

1eV = 1.602 E-19 J

T_a = 130 C + 30 C = 160 C = 433 K

t_a = ?

$$\frac{t_a}{t_s} = \exp\left[-\frac{E_a}{k_B}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)\right]$$

$$t_a = (3.7\ \text{months})\ \exp\left[-\frac{1.0\ \text{eV}}{1.38\ E{-}23\ \frac{J}{K}}\left(\frac{1.602\ E{-}19\ J}{1\ \text{eV}}\right)\left(\frac{1}{385\ K} - \frac{1}{433\ K}\right)\right]$$

t_a = .13 months = 4.0 days

QUALITY ASSURANCE PROCEDURE

Performance Evaluation of Electrical Equipment During Severe Accident States (PEEESAS) Test Program

Page	1	2	3	4	5	6	7	8	9	10
Revision	A	A	A	A	A	A	A	A	A	A

Approved by: D. A. Brosseau — 12/3/85
Q. A. Coordinator — Date

D. L. Berry — 12/3/85
Program Manager — Date

PERFORMANCE EVALUATION OF ELECTRICAL EQUIPMENT DURING SEVERE ACCIDENT STATES TEST PROGRAM (PEEESAS)

1.0 POLICY

Sandia National Laboratories (SNL) has as a primary objective the solution of engineering and scientific problems of interest to the public and to sponsoring organizations such as the NRC. It is the policy of SNL to take appropriate steps through selective application of appropriate quality assurance program controls to ensure that work is done to quality standards commensurate with the activities performed.

1.1 Purpose The purpose of this Quality Assurance Procedure (QAP) is to summarize the quality assurance requirements that have been identified for implementation on a project-specific basis to the tasks identified herein. This QAP and its' set of requirements shall provide the basis for documentation of all quality-related tasks of the PEEESAS Test Program.

1.2 Scope This QAP defines the total applicable set of QA requirements based upon the 18 QA criteria of 10CFR50, Appendix B, and provisions of the 6410/6440 Quality Assurance Program Plan. Further, the extent of application of each QA criteria is governed by the Quality Level assigned to the PEEESAS Test Program.

1.3 Applicability This QAP applies to the task or tasks defined herein to the extent required. All requirements shall be adhered to by project personnel directly involved with or who interface with the specific SNL project task. No other quality assurance plans, except as noted in this procedure, apply to the particular program covered by this QAP. Changes or revisions of project QA requirements due to project scope or task definition changes shall be accomplished by revisions to this QAP.

2.0 GENERAL

2.1 Introduction This QAP is subordinate to a number of key quality assurance documents, including:

a. The 6410/6440 Quality Assurance Program Plan (QAPP)
b. The Organization 6000 QAPP
c. 10CFR50, Appendix B
d. ANSI/ASME NQA-1

This project-specific QAP is consistent with the governing QA requirements, to the extent appropriate, for the specific project tasks and associated Quality Levels as further defined herein.

2.2 Organization This QAP falls within the existing organization structure and provisions outlined in its' parent QAPP documents. No specific organizational changes are required to implement the project tasks or quality assurance provisions. Existing organizational responsibilities and authorities are unaffected and shall prevail for all project work and review/approval requirements.

2.3 Quality Assurance Program The administration, documentation, implementation, monitoring and control of the applicable SNL Quality Assurance Programs shall not be negatively affected or diminished by this QAP. All QAPP guidelines shall apply to the extent defined in this QAP as appropriate to the particular project tasks. Reference is made within this QAP to Energy Programs Instructions (EPIs) that apply to tasks specific to this project. EPIs shall be implemented as appropriate to the nature and scope of project tasks. All questions regarding specific implementation requirements in EPI's shall be referred to the QA Coordinator for resolution.

3.0 GENERAL PROJECT DESCRIPTION

This QAP specifically applies to testing associated with the PEEESAS Test Program (Division 6447). The purpose of the testing is to evaluate the performance of electrical equipment, during a severe accident, which would be used to mitigate an accident or provide information on the status of the plant. The results of this program will provide data on environmentally-induced equipment failures.

For FY-86 testing, the test specimens will be exposed to simultaneous radiation and thermal aging and a severe accident test profile. Accident test profiles are based on accident sequences for a BWR/MARK I. Throughout the accident profiles, the operational performance of the test specimen will be monitored to ensure that the test specimen performs its required safety function.

Overall task Quality Level is III or Minor as defined in EPI II-3. The following section defines the applicable quality assurance criteria and tasks that are required during the execution of this experimental program.

4.0 REQUIREMENTS

It shall be the responsibility of the principal investigator (PI) to plan and conduct all project tasks within the quality assurance requirements listed in this section of this QAP. Determination of requirements is based upon the assigned Quality Level (III, Minor) and the overall program requirements of referenced governing documents, including the 6000 QAPP and 6410/6440 QAPP. The requirements are listed in the order provided in 10CFR50, Appendix B. Attachment I to this QAP provides an itemized checklist of QA requirements for all major project tasks.

4.1 Design Control Required tasks associated with design control include the following:

a. Identify project objectives in appropriate statements of work and program plans.

b. Develop and document detailed test/experiment plans and procedures, including appropriate acceptance criteria.

c. Obtain one technical peer review for test plans/procedures and document using appropriate forms in accordance with EPI III-2 or by use of memoranda to file. See QA Coordinator regarding EPI implementation requirements.

d. Obtain SAND document reviews and approvals, as appropriate, and document on established forms.

e. Control document changes and drawing revisions in accordance with established procedures.

f. Experimental results and log book data shall be maintained in an auditable, retrievable manner.

Route all test plans and peer review records to the QA Coordinator for review and filing.

4.2 Procurement Procurement activities and associated quality assurance provisions are only marginally applicable to this project due to the nature and limited scope of the overall experimental program. Procurement QA documentation shall be established in accordance with EPI IV-2 (see QA Coordinator for details on current EPI) as follows:

a. Include technical and QA requirements in the PRs for the test specimens. QA requirements shall include, as appropriate:

(1) Right of access for inspection and/or audit.

(2) Required supplier submittals of certifications, evidence of quality.

(3) QA Plan, if required, which meets 10CFR50 requirements.

(4) Identification of inspection requirements and hold points.

(5) Other special instructions and acceptance requirements.

b. Complete the form "Quality Requirements for Purchase Requisition" (EPI IV-2) in consultation with the QA Coordinator, and obtain QA Coordinator or alternate designee review of the PR package to verify the adequacy of QA requirements, prior to management approval.

c. Process Change Requisitions in accordance with established procedures to include the original review and approval requirements.

d. Ensure adequate translation of technical requirements in the PO and contract documents.

e. Close out contract documentation in accordance with established procedures, Section 9, EPI IV-2 (see QA Coordinator).

4.3 Instructions, Procedures and Drawings Instructions and procedures, including appropriate drawings or sketches, shall be developed, referenced and utilized to accomplish the project tasks associated with the PEEESAS Program. Procedures shall be prepared, reviewed and approved in accordance with EPI XI-1 and EPI III-2, with inclusion of quantitative and/or qualitative acceptance criteria to assure measures of successfully meeting requirements. See QA Coordinator for EPI implementation requirements.

4.4 Document Control The following requirements apply:

a. Specify and control preparation, review, approval, distribution and changes made to project documentation in accordance with established departmental procedures.

b. Ensure latest issue of all documents is at all appropriate work locations.

c. Project files shall be controlled and indexed to ensure access to current issues of project documents.

4.5 Control of Purchased Items and Services Required tasks associated with this QA category include the following:

a. Specify or request source inspections of suppliers and appropriate hold points, if deemed appropriate, in accordance with EPI VII-1. (See QA Coordinator for implementation details.) Participation is recommended to ensure technical requirements are adhered to.

b. Coordinate and specify acceptance or rejection (deviation) of materials with the receiving and purchasing organizations and the QA Coordinator.

c. Specify and verify receipt of required supplier certifications with test components.

d. Determine nonconformances or deviations, coordinate or specify corrective action, and conduct follow-up activities in consultation with the QA Coordinator.

4.6 Identification and Control of Items Quality assurance requirements associated with this category on the PEEESAS Test Program are limited to ensuring adequate marking, identification, and traceability of the test specimens throughout the experimental program. Identification shall be appropriately cross-referenced to procedures and lab notebook entries and data.

4.7 Control of Special Processes There are no applicable requirements associated with special processes in the PEEESAS Test Program. Operation of test facilities is conducted through well-established SNL procedures using qualified personnel. Operator and test personnel qualification and certification records are documented and maintained in accordance with established SNL procedures.

4.8 Inspection The need for and control of any required inspection or testing activities and personnel qualifications shall be defined in appropriate specifications, test plans, or procedures.

4.9 Test Control All experiments and tests shall be conducted in accordance with appropriate test plans and procedures prepared, distributed and revised in accordance with EPI XI-1 (for details, see QA Coordinator) and established departmental standards and procedures. SNL facility operation and desired test conditions shall be called out and controlled by test procedures. Test log books shall be maintained and controlled. All test plans and procedures shall include project or task Quality Level assignment and any applicable hold points.

4.10 Control of Measuring and Test Equipment QA requirements are as follows:

a. Ensure all equipment, gages and standards are either entered into the SNL calibration system or are calibrated and controlled by project personnel via appropriate procedures (EPI XII-4; see QA Coordinator).
b. Verify all calibration stickers are current and will remain so for the expected duration of the test.

4.11 Handling, Storage and Shipping Quality assurance requirements associated with handling, storage and shipping are limited to "common sense" care and handling of test specimens to protect them from loss, damage, exposure to elements or sunlight, or other degradation. Handling requirements shall be outlined, as appropriate, in test plans and procedures. Supplier handling, packaging, and shipping requirements shall be specified to meet the requirements typical of shipments for nuclear power plant use.

4.12 Inspection, Test and Operating Status QA requirements are as follows:

a. Identify the test specimens and indicate test status by appropriate stickers, marking, or notations in accordance with written procedures or instructions.

b. Provide controls and instructions authorizing application and removal of status indicators and for segregation and control of nonconforming items.

c. Provide necessary status indicators, as appropriate, to inform other SNL personnel of the operating status of test and experimental facilities.

4.13 Control of Nonconforming Items Required QA activities are as follows:

a. Report nonconformances via entries in lab or field notebooks.

b. Determine and document all dispositions, including material segregation and/or rejection.

c. Determine and implement corrective actions (paragraph 4.14).

d. Implement means to prevent recurrences of nonconformances.

4.14 Corrective Action QA requirements are as follows:

a. Following identification of nonconformances and disposition, determine the cause(s) and appropriate corrective actions, document in lab notebooks, and implement.

b. Monitor the effectiveness of any corrective actions.

4.15 Quality Assurance Records QA requirements are as follows:

a. Maintain and control records by ensuring all records are identifiable and retrievable.

b. Determine and document records retention requirements. Minimum retention shall be for the life of the project, including follow-on work.

c. Following project completion and official transfer of all records to sponsor files, archive all records in accordance with SNL procedures.

4.16 Audits All project personnel shall assist, participate, cooperate and follow up on all audit or surveillance activities in areas affected and in accordance with the outline procedure provided in EPI XVIII-1. Regarding specific implementation procedures and principal investigator responsibilities, consult with the QA Coordinator for details.

Project Activity	QA Requirements	QAP Ref.	EPI Ref.
Test Plan	Preparation/Peer Review	4.1b,c	XI-1,III-2
	Revision Review	4.1e	XI-1
	Include: Quality Level, Acceptance Criteria, Applicable Hold Points	4.1b 4.9	XI-1
	Specify Test Specimen Handling and Applicable Inspection Requirements	4.6,4.8, 4.11	XI-1
	Distribution to QA Coordinator	4.1	XI-1
Purchase Requisitions:	Include Technical Specifications and Requirements	4.2a	IV-2
Major	Require Supplier Certifications	4.2a,4.5c	IV-2
	Handling, Storage, Shipping Requirements	4.2a,4.11	-
	Specify Right of Access to Supplier Facilities (if applicable)	4.2a	IV-2
	Require Contractor Inspections	4.8	-
	Source Inspections, Hold Points (if applicable)	4.2a,4.5a	IV-2,VII-1
	"QA Requirements for Purchase Requisition" Form and QA Coordinator Review	4.2b	IV-2
	PR to PO Translation Verification	4.2d	IV-2
	Receiving Inspection (Inspection Code X)	4.2a,4.5b	IV-2
	Non-conformance/Deviation/Corrective Action Documentation	4.5d	IV-2
Minor	None	-	-
Test Fixture Design	Informal Peer Review Suggested	-	-
Test Procedures Baseline Tests	Preparation/Review/Approval	4.1b,4.3	XI-1,III-2
	Modifications	-	V-1
	Test Status Designation	4.9	-
	Peer Review, if Appropriate	4.1c,4.3	III-2
	Contractor Test Control	4.9	
Data Acquisition:	Auditable and Retrievable	4.1f	-
Log Books Photographs	Identification and Control of Test Specimen	4.6	-
Data Sheets Computer Tapes,	Test Status Designation (if applicable)	4.12	-
Files, Etc.	Non-conformance/Corrective Action Documentation During Test	4.13,4.14	-

Project Activity	QA Requirements	QAP Ref.	EPI Ref.
Instrument Calibration	Verify Calibration Documentation	4.10a	XII-4
	Calibration Stickers Current During Test	4.10b	
	Ensure Contractor Calibration	4.10c	XII-3
Data Analysis	None	–	–
Quick Look Reports	None	–	–
SAND Reports	Preparation (Format and Content)	4.1d	–
	Peer Reviews	4.1d	III-2
	Management Approval	4.1d	III-2

DISTRIBUTION:

U.S. Government Printing Office
Receiving Branch (Attn: NRC Stock)
8610 Cherry Lane
Laurel, MD 20707
375 copies for RV

Ansaldo Impianti
Centro Sperimentale del Boschetto
Corso F.M. Perrone, 118
16161 Genova
ITALY
Attn: C. Bozzolo

Ansaldo Impianti
Via Gabriele D'Annunzio, 113
16121 Genova
ITALY
Attn: S. Grifoni

ASEA-ATOM
Department KRD
Box 53
S-721 04
Vasteras
SWEDEN
Attn: A. Kjellberg

ASEA-ATOM
Department TQD
Box 53
S-721 04
Vasteras
SWEDEN
Attn: T. Granberg

ASEA KABEL AB
P.O. Box 42 108
S-126 12
Stockholm
SWEDEN
Attn: B. Dellby

Atomic Energy of Canada, Ltd.
Chalk River Nuclear Laboratories
Chalk River, Ontario K0J 1J0
CANADA
Attn: G. F. Lynch

Atomic Energy of Canada, Ltd.
1600 Dorchester Boulevard West
Montreal, Quebec H3H 1P9
CANADA
Attn: S. Nish

Atomic Energy Research Establishment
Building 47, Division M.D.D.
Harwell, Oxfordshire
OX11 ORA
ENGLAND
Attn: S. G. Burnay

Bhabha Atomic Research Centre
Health Physics Division
BARC
Bombay-85
INDIA
Attn: S. K. Mehta

British Nuclear Fuels Ltd.
Springfields Works
Salwick, Preston
Lancs
ENGLAND
Attn: W. G. Cunliff, Bldg 334

Brown Boveri Reaktor GMBH
Postfach 5143
D-6800 Mannheim 1
WEST GERMANY
Attn: R. Schemmel

Bundesanstalt fur Materialprufung
Unter den Eichen 87
D-1000 Berlin 45
WEST GERMANY
Attn: K. Wundrich

CEA/CEN-FAR (2)
Departement de Surete Nucleaire
Service d'Analysex Fonctionnelle
B.P. 6
92260 Fontenay-aux-Roses
FRANCE
Attn: M. Le Meur
J. Henry

CERN
Laboratorie 1
CH-1211 Geneve 23
SWITZERLAND
Attn: H. Schonbacher

Canada Wire and Cable Limited
Power & Control Products Division
22 Commercial Road
Toronto, Ontario
CANADA M4G 1Z4
Attn: Z. S. Paniri

Centro Elettrotecnico
  Sperimentale Italiano
Research and Development
Via Rubattino 54
20134 Milan
ITALY
Attn: Carlo Masetti

Commissariat a l'Energie Atomique (3)
ORIS/LABRA
BP No 21
91190 Gif-Sur-Yvette
FRANCE
Attn: G. Gaussens
       J. Chenion
       F. Carlin

Commissariat a l'Energie Atomique
CEN Cadarche DRE/STRE
PB No 1
13115 Saint Paul Lez Durance
FRANCE
Attn: J. Campan

Conductores Monterrey, S.A.
P.O. Box 2039
Monterrey, N.L.
MEXICO
Attn: P. G. Murga

Electricite de France (2)
(S.E.P.T.E.N.)
12, 14 Ave. Dubrieroz
69628 Villeurbarnie
Paris, FRANCE
Attn: H. Herouard
       M. Hermant

Electricite de France (2)
Direction des Etudes et Recherches
1, Avenue du General de Gaulle
92141 CLAMART CEDEX
FRANCE
Attn: J. Roubault
       L. Deschamps

Electricite de France (3)
Direction des Etudes et Recherches
Les Renardieres
Boite Postale no 1
77250 MORET SUR LORING
FRANCE
Attn: Ph. Roussarie
       V. Deglon
       J. Ribot

Energia Nucleare e delle
  Energie Alternative
CKE Casaccia
1-0060 Rome
ITALY
Attn: A. Cabrini

EURATOM
Commission of European Communities
C.E.C. J.R.C.
21020 Ispra (Varese)
ITALY
Attn: G. Mancini

FRAMATOME (2)
Tour Fiat - Cedex 16
92084 Paris La Defense
FRANCE
Attn: G. Chauvin
       E. Raimondo

Furukawa Electric Co., Ltd.
Hiratsuka Wire Works
1-9 Higashi Yawata - 5 Chome
Hiratsuka, Kanagawa Pref
JAPAN 254
Attn: E. Oda

Gesellschaft fur
  Reaktorsicherheit (GRS) mbH
Glockengasse 2
D-5000 Koln 1
WEST GERMANY
Attn: Library

Health & Safety Executive
Thames House North
Milbank
London SW1P 4QJ
ENGLAND
Attn: W. W. Ascroft-Hutton

ITT Cannon Electric Canada
Four Cannon Court
Whitby, Ontario L1N 5V8
CANADA
Attn: B. D. Vallillee

Imatran Voima Oy (2)
Electrotechn. Department
P.O. Box 138
SF-00101 Helsinki 10
FINLAND
Attn: B. Regnell
K. Koskinen

Institute of Radiation Protection
Department of Reactor Safety
P.O. Box 268
00101 Helsinki 10
FINLAND
Attn: L. Reiman

Instituto de Desarrollo y Diseno
Ingar - Santa Fe
Avellaneda 3657
C.C. 34B
3000 Santa Fe
REPUBLICA ARGENTINA
Attn: N. Labath

ISMES S.p.A. (2)
Viale G. Cesare, 29
24100 Bergamo, ITALY
Attn: A. Castoldi
M. Salvetti

Japan Atomic Energy (3)
Research Institute
Takasaki Radiation Chemistry
Research Establishment
Watanuki-machi
Takasaki, Gunma-ken
JAPAN
Attn: N. Tamura
K. Yoshida
T. Seguchi

Japan Atomic Energy
Research Institute
Tokai-Mura
Naka-Gun
Ibaraki-Ken
319-11, JAPAN
Attn: Y. Koizumi

Japan Atomic Energy
Research Institute
Osaka Laboratory for
Radiation Chemistry
25-1 Mii-Minami machi,
Neyagawa-shi
Osaka 572
JAPAN
Attn: Y. Nakase

Kalle Niederlassung der Hoeschst AG
Postfach 3540
6200 Wiesbaden 1,
WEST GERMANY
Biebrich
Attn: Dr. H. Wilski

Kraftwerk Union AG
Department R361
Hammerbacherstrasse 12 + 14
D-8524 Erlangen
WEST GERMANY
Attn: I. Terry

Kraftwerk Union AG
Section R541
Postfach: 1240
D-8757 Karlstein
WEST GERMANY
Attn: W. Siegler

Kraftwerk Union AG
Hammerbacherstrasse 12 + 14
Postfach: 3220
D-8524 Erlangen
WEST GERMANY
Attn: W. Morell

Motor Columbus
Parkstrasse 27
CH-5401
Baden
SWITZERLAND
Attn: H. Fuchs

National Nuclear Corporation (2)
Cambridge Road, Whetstone
Leicester LE8 3LH
ENGLAND
Attn: A. D. Hayward
J. V. Tindale

NOK AG Baden
Beznau Nuclear Power Plant
CH-5312 Doettingen
SWITZERLAND
Attn: O. Tatti

Norsk Kabelfabrik
3000 Drammen
NORWAY
Attn: C. T. Jacobsen

Nuclear Power Engineering
Test Center
6-2, Toranomon, 3-Chome
Minato-ku, #2 Akiyana Bldg.
Tokyo 105
JAPAN
Attn: K. Takumi

Ontario Hydro (2)
700 University Ave
Toronto, Ontario M5G 1X6
CANADA
Attn: R. Wong
B. Kukreti

Oy Stromberg Ab
Helsinki Works
Box 118
Fi-00101 Helsinki 10
FINLAND
Attn: P. Paloniemi

Radiation Center of
Osaka Prefecture
Radiation Application-
Physics Division
Shinke-Cho, Sakai
Osaka, 593, JAPAN
Attn: S. Okamoto

Rappini
ENEA-PEC
Via Arcoveggio 56/23
Bologna
ITALY
Attn: Ing. Ruggero

Rheinisch-Westfallscher
Technischer Uberwachunge-Vereln e.V.
Postfach 10 32 61
D-4300 Essen 1
WEST GERMANY
Attn: R. Sartori

Sydkraft
Southern Sweden Power Supply
21701 Malmo
SWEDEN
Attn: O. Grondalen

Technical University Munich
Institut for Radiochemie
D-8046 Garching
WEST GERMANY
Attn: Dr. H. Heusinger

UKAEA
Materials Development Division
Building 47
AERE Harwell
OXON OX11 ORA
ENGLAND
Attn: D. C. Phillips

United Kingdom
Atomic Energy Authority
Safety & Reliability Directorate
Wigshaw Lane
Culcheth
Warrington WA3 4NE
ENGLAND
Attn: M. A. H. G. Alderson

Waseda University
Department of Electrical Engineering
3-4-1 Ohkubo, Shinjuku-ku
Tokyo 160
JAPAN
Attn: Y. Ohki

A. M. Kolaczkowski (2)
Science Applications
International Corporation
505 Marquette Avenue, NW
Albuquerque, NM 87102

G. T. Medford (2)
Science Applications
International Corporation
505 Marquette Avenue, NW
Albuquerque, NM 87102

J. B. Komendera
Automatic Valve Corporation
P.O. Box 435
41144 Vincenti Court
Novi, MI 48050-0435

J. Edson
EG&G Idaho, Inc.
1955 Fremont Avenue
Bldg. WCB-W2
Idaho Falls, ID 83415

Professor David Okrent
5532 Boelter Hall
Univeristy of California at
Los Angeles
School of Applied Science & Engineering
Los Angeles, CA 90024

1200	J. P. VanDevender
1800	R. L. Schwoebel
1810	R. G. Kepler
1811	R. L. Clough
1812	J. M. Zeigler
1812	K. T. Gillen
1813	J. G. Curro
2126	J. E. Gover
2126	O. M. Stuetzer
2514	L. L. Bonzon
5114	W. D. Ulrich
5114	P. R. Bennett (5)
6200	V. L. Dugan
6300	R. W. Lynch
6400	A. W. Snyder
6410	J. W. Hickman
6411	D. M. Kunsman
6412	A. C. Payne
6420	J. V. Walker
6430	N. R. Ortiz
6433	F. V. Thome
6433	F. J. Wyant
6440	D. A. Dahlgren
6442	W. A. von Riesemann
6444	L. D. Buxton
6446	W. H. Buckalew
6446	L. D. Bustard
6446	J. W. Grossman
6446	M. J. Jacobus
6446	J. D. Keck
6447	D. L. Berry
6447	V. J. Dandini
6447	D. B. King
6447	V. F. Nicolette
6447	S. P. Nowlen
6447	B. L. Spletzer
6447	W. T. Wheelis
6449	K. D. Bergeron
6450	T. R. Schmidt
8024	P. W. Dean
3141	S. A. Landenberger (5)
3151	W. L. Garner

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION — **BIBLIOGRAPHIC DATA SHEET** — SEE INSTRUCTIONS ON THE REVERSE	1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any) NUREG/CR-4537 SAND86-0451
2. TITLE AND SUBTITLE SUMMARY REPORT: ELECTRICAL EQUIPMENT PERFORMANCE UNDER SEVERE ACCIDENT CONDITIONS (BWR/MARK I PLANT ANALYSIS)	3. LEAVE BLANK 4. DATE REPORT COMPLETED MONTH / YEAR
5. AUTHOR(S) P. R. Bennett, A. M. Kolaczkowski, G. T. Medford	6. DATE REPORT ISSUED MONTH: July YEAR: 1986
7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Sandia National Laboratories Albuquerque, NM 87185	8. PROJECT/TASK/WORK UNIT NUMBER 9. FIN OR GRANT NUMBER
10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Division of Engineering Technology Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, DC 20555	11a. TYPE OF REPORT Technical b. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

The purpose of the Performance Evaluation of Electrical Equipment during Severe Accident States program is to determine the performance of electrical equipment, important to safety, under severe accident conditions. In FY 85, a method was devised to identify important electrical equipment and the severe accident environments in which the equipment was likely to fail. This method was used to evaluate the equipment and severe accident environments for Browns Ferry Unit 1, a BWR/Mark I. In addition, a test plan was written to experimentally determine the performance of one selected component to two severe accident environments.

Specifically, equipment was identified that was important to safety for a BWR--equipment which would mitigate severe accident sequences or provide plant status. For this list of equipment, only that equipment located in the primary containment or reactor vessel of Browns Ferry Unit 1 was analyzed further. For the five selected BWR severe accident sequences (TB, TC, TW, TQUV, and AE), environmental conditions within containment reached temperatures and pressures exceeding the current equipment qualification testing requirements prior to or during the time the equipment was needed. The results of this analysis suggest the need for testing the performance of the pneumatic control manifold assembly (part of the main steam isolation valve equipment assembly) during the TC and TW accident sequences.

Insights from the analysis portion of this study were used to recommend changes in the areas of accident management, emergency planning, probabilistic risk assessments, probability and risk reduction, and current equipment qualification requirements. However, confirmation of the analytical results should be demonstrated by testing the equipment before the recommendations are implemented.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS b. IDENTIFIERS/OPEN-ENDED TERMS	15. AVAILABILITY STATEMENT: Unlimited 16. SECURITY CLASSIFICATION (This page) Unclassified (This report) Unclassified 17. NUMBER OF PAGES: 60 18. PRICE

☆ U.S. GOVERNMENT PRINTING OFFICE: 1986—773-049/41016