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Instrumentation Availability During Severe Accidents for a Boiling Water Reactor with a Mark I Containment

Prepared by
W. C. Arcieri, D. J. Hanson

Idaho National Engineering Laboratory
EG&G Idaho, Inc.

Prepared for
U.S. Nuclear Regulatory Commission

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Prepared by
W. C. Arcieri, D. J. Hanson

Idaho National Engineering Laboratory
Managed by the U.S. Department of Energy

EG&G Idaho, Inc.
Idaho Falls, ID 83415

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ABSTRACT

In support of the U.S. Nuclear Regulatory Commission Accident Management Research Program, the availability of instruments to supply accident management information during a broad range of severe accidents is evaluated for a Boiling Water Reactor with a Mark I containment. Results from this evaluation include: (a) the identification of plant conditions that would impact instrument performance and information needs during severe accidents, (b) the definition of envelopes of parameters that would be important in assessing the performance of plant instrumentation for a broad range of severe accident sequences, and (c) assessment of the availability of plant instrumentation during severe accidents. A similar evaluation for a pressurized water reactor with a large, dry containment design is presented in NUREG/CR-5691.

FIN No. L1425—Accident Management Information Needs

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EXECUTIVE SUMMARY

The Nuclear Regulatory Commission (NRC) identifies accident management as an essential element of the Integration Plan for the closure of severe accident issues.[1] Accident management ensures that planned actions and preparatory measures are developed to enhance the ability of nuclear power plant personnel to effectively manage severe accidents. An area that affects this ability is the availability of timely and accurate information that will assist in determining the status of the plant, selecting preventative or mitigative actions, and monitoring the effectiveness of these actions. The plant instrumentation is relied on to supply this information.

Because instrumentation is an important element to accident management, the NRC needs a strong technical basis to understand the capabilities and shortcomings of instrument systems under severe accident conditions that are representative of instruments used in existing plants. The data provided by the series of studies on information needs and instrument capabilities for both the Pressurized Water Reactor (PWR) and the Boiling Water Reactor (BWR) plants (which this study is a part) will enable NRC staff to evaluate the validity of licensees claims concerning the ability of plant staff to detect the onset of a severe accident, diagnose and evaluate severe accident status, select appropriate corrective actions, and monitor the effectiveness of these actions.

The capability of representative plant instruments to supply the information needed to manage a broad range of severe accidents is conducted for a BWR with a Mark I containment in this study. The objectives of this study are to: (a) identify plant conditions that would influence the availability and performance of the instrumentation and the information needs during severe accidents, (b) define envelopes of parameters that would be important in assessing the availability of plant instrumentation for a broad range of severe accident sequences, and (c) assess the availability of plant instrumentation during severe accidents.

The approach used to meet these objectives includes the following steps:

1. Identify a set of possible severe accident sequences that represent the spectrum of accident types that have a principal impact on the risk for a BWR with a Mark I containment.

2. Define the expected conditions within the reactor coolant system, containment (drywell and torus), and reactor building for the identified severe accident sequences. Define bounding envelopes for these conditions.

3. Assess instrument availability during the severe accident sequences, based on the location of the instrument components and conditions that would influence instrument performance.

4. Provide an accident management information assessment that discusses information needs and the instruments that are available to meet these needs. Identify potential limitations on the information available for assessing the status of plant safety functions.

The set of severe accident sequences that has the potential to influence risk for a BWR with a Mark I containment is based on NUREG–1150[2] results. These results represent the most recent evaluation of the types of accidents that will dominate core damage frequency and risk to the public. The set of sequences (plant damage states) identified were the following: station blackout (SBO), large and small break loss–of–coolant accidents, anticipated transients without scram (ATWS), and all transients other than SBO and ATWS. Results from existing studies (BMI–2104[3] and NUREG/CR–4624[4]) were used to define thermal hydraulic data within the reactor system and containment for accident sequences representative of the NUREG–1150 results.

Assessment of instrument availability is primarily based on the environmental qualification limits, instrument range, and the source of backup power for each instrument. Instrument information for this evaluation is based on the implementation of Regulatory Guide 1.97[5] requirements at the Peach Bottom Atomic Power Station,[6] the technical evaluation report for equipment qualification at Peach Bottom,[7] and information on pressure and temperature qualification conditions from the Peach Bottom Final Safety Analysis Report (FSAR).[8]

The assessment of instrument availability assumes that instrument performance will be degraded if the pressure, temperature, or radiation conditions in the vicinity of the instrument exceeds the specified qualification limits, or if the parameter being measured is outside the instrument range. This definition includes the possibility of instrument failure. Degraded instrument performance denotes that the indicated magnitude or trend of the measured parameter is in error. This error may cause the operator to take inappropriate action, cause premature termination of the operation of an automatic safety system, or start the operation of an automatic safety system when it is not required. An example would be termination of the operation of the high pressure coolant injection system (HPCI) due to an false indication of high vessel water level.

Pressure and temperature conditions have the greatest impact on instrument availability, particularly in the early stages of the accident. Degraded instrument performance due to severe pressure and temperature conditions can occur prior to core damage for accidents involving an ATWS with a standby liquid control system failure. For these types of accidents, severe conditions can occur in the containment (drywell and torus) and in the reactor building upon containment failure. Containment venting could cause severe reactor building conditions if vents other than the hardened vent system are used during an ATWS with Standby Liquid Control System (SLCS) failure, or if a hardened vent system is not installed. Typical hardened vent systems are being designed for decay heat levels and may not have sufficient capacity for an ATWS. For long–term accident management situations (days or weeks), radiation exposure could affect instrument performance. The effects of radiation on instrument components located in the reactor building are considered to be particularly significant, because the hardware was qualified for radiation levels resulting from a design basis accident where the primary containment stays intact.

Results from the evaluation of instrument availability based on the Peach Bottom design and the thermal hydraulic conditions for a broad range of severe accidents are presented in this report. These results are summarized as follows:

- The detectors used by the neutron monitoring system are available prior to core damage. After the onset of core damage, temperature in the vicinity of the detectors will exceed the qualification temperature and the instrument performance would degrade. The performance of the system may degrade during an ATWS, if the pressure and temperature in the drywell exceeds qualification limits, because components of this system are located in the drywell.
- Performance of instruments in the primary containment (drywell and torus) could degrade prior to the onset of core damage if the containment pressure exceeds the qualification pressure during an ATWS with SLCS failure or during sequences involving failure of the containment heat removal systems.
- Performance of instruments in the primary containment (drywell and torus) could degrade if the reactor vessel fails.
- Performance of instrument systems with components in the reactor building could degrade prior to the onset of core damage for ATWS sequences with SLCS failure due to containment failure or failure of nonhardened ducts after containment venting. Degradation of these systems would affect the capability to monitor and control conditions in the reactor coolant system and containment.
- Performance of instrument systems with components in the reactor building are

available prior to containment failure for non–ATWS sequences where core melt occurs. If containment failure occurs after core melt, then severe conditions in the reactor building could cause degraded instrument performance.

Because of differences in the electrical power system configuration at different plants, it is not possible to generically evaluate instrument availability for a station blackout. It is noted that many plants provide battery backup for all Regulatory Guide 1.97, Category 1 instrumentation although it is not a requirement. If battery backup is provided, then most of the information required to monitor the status of the reactor coolant system and containment will be available until the batteries deplete or accident conditions challenge instrument availability. In addition, systems used to obtain and analyze samples of reactor coolant, containment atmosphere, and suppression pool water may not be available in the event of a station blackout. Information needs that require sampling information may not be met as a result.

Information needs were reviewed for each of the safety functions defined in the safety objective trees developed as a result of an NRC sponsored information needs evaluation presented in NUREG/CR–5702.[9] This review shows that the ability to meet safety functions associated with maintaining pressure and temperature control for the reactor and containment will be impeded during an accident, particularly if severe conditions develop in the reactor building.

The results from this instrument availability evaluation are intended to provide scoping information that can be used to understand the general characteristics of instrument availability for a wide range of plant conditions during severe accidents. These results are conservative in that less availability is predicted in this study than would be predicted by a more detailed, plant–specific study for the following reasons:

1. Specified instrument qualification conditions were used rather than actual qualification conditions. The actual conditions may exceed the specified qualification conditions because most instruments are tested to more severe environments than those specified by the licensee. This difference could increase instrument availability for some accident sequences.

2. More detailed analysis of the environmental conditions at the location of instrument components in the containment and reactor building would tend to increase availability. Location of instrument components varies widely from plant to plant and specific locations may be protected from severe accident conditions.

3. Degraded performance of instruments is likely influenced by the length of time and the magnitude of the difference between the environmental and qualification conditions. If the environmental conditions exceed the specified qualification conditions by small amounts or for short periods of time, the instruments would likely remain available.

Plant–specific evaluations of instrument availability would be necessary to eliminate conservatism from the results. The evaluations would need to include an assessment of the relationship between the instrument uncertainties and the timing and degree to which the qualification conditions are exceeded, based on a detailed study of basic instrument capabilities and failure modes. These plant–specific evaluations are beyond the scope of this study.

The results of the study should provide an understanding of conditions for which instrumentation system response may become unreliable, and could adversely affect the ability of licensees to effectively diagnose and manage severe accidents. When coupled with the results of a proposed study to evaluate the actual response characteristics of selected representative systems when operated beyond their qualification or design limits, the NRC staff should have a strong technical basis to evaluate the accident management claims of licensees, as well as evidence provided by them to justify the adequacy of their instrument systems for implementing appropriate accident management procedures and guidance.

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ACRONYMS

ac	alternating current
ADS	automatic depressurization system
ATWS	anticipated transient without scram
BWR	Boiling Water Reactor
CRD	control rod driveline
dc	direct current
ECCS	Emergency Core Cooling System
HPCI	high-pressure coolant injection
IRM	intermediate range monitor
LOCA	loss–of–coolant accident
LPCI	low pressure coolant injection
MSIV	main steam isolation valves
NRC	Nuclear Regulation Commission
ORNL	Oak Ridge National Laboratory
PCS	Power Conversion System
PDS	plant damage state
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RH	relative humidity
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	reactor pressure vessel
SBO	station blackout
SGTS	Standby Gas Treatment System
SLCS	Standby Liquid Control System
SRM	source range monitor
SRV	safety relief valve

Instrumentation Availability During Severe Accidents for a Boiling Water Reactor with a Mark I Containment

1. INTRODUCTION

The Nuclear Regulatory Commission (NRC) identifies accident management as an essential element of the Integration Plan for the closure of severe accident issues.[1] Accident management ensures that planned actions and preparatory measures are developed to enhance the capability of nuclear power plant personnel to effectively manage severe accidents. Successful accident management is strongly influenced by instrument availability.

A methodology to assess information needs and instrument availability was developed in NUREG/CR-5702 to identify (a) the information needed to determine the status of a BWR for a broad range of severe accidents, including recovery actions, (b) the existing plant measurements that could be directly or indirectly used to supply these information needs, and (c) the conditions in which information from the measurement systems could mislead plant personnel. A four step approach was developed in NUREG/CR-5702 for identifying nuclear power plant information needs during severe accidents and for determining the extent to which these needs will be met by instrumentation currently in use at the plants. The steps are (1) to develop safety objective trees, (2) to determine information needs and sources of information, (3) to identify available instruments, and (4) to identify potentially misleading information.

Results presented in this report show the impact of possible plant environmental conditions during a range of severe accidents on the availability of instruments needed to meet the severe accident information needs discussed in NUREG/CR-5702. This evaluation is performed for a BWR with a Mark I containment. Section 2 describes the approach and the data used to evaluate instrument availability, and defines an envelope of plant conditions. Section 3 discusses important accident sequences for evaluating information needs and instrument availability. Section 4 presents an evaluation of instrument availability. Section 5 presents an evaluation of information needs based on available instruments. Section 6 discusses an envelope of severe accident plant conditions and event timing. The summary and conclusions are presented in Section 7 and the references are listed in Section 8. Appendices A through D discuss Peach Bottom plant damage states, results from Peach Bottom thermal hydraulic analyses, accident management information assessment, and long-term effects of radiation on instrument availability.

2. APPROACH USED TO EVALUATE INSTRUMENT AVAILABILITY AND DEFINE THERMAL HYDRAULIC ENVELOPE

The approach used to evaluate the availability of instrumentation during severe accidents is described in this section. The sources of information on potential accident sequences and severe accident thermal hydraulic behavior are also described.

2.1 Approach

The approach used to evaluate instrument availability for various severe accident conditions are summarized in the following steps:

Step 1–Identification of Types of Severe Accidents

The types of severe accident sequences that potentially influence risk are identified for a spectrum of severe accidents using the probabilistic risk assessment results presented in NUREG–1150[2] for Unit 2 of the Peach Bottom Atomic Power Station. The Peach Bottom station has two General Electric boiling water reactors (BWR–4). Both have a rated thermal power output of 3293 MW_{th} and are housed in a Mark I containment. NUREG–1150 results are used because they represent the most recent evaluation of all credible types of accidents that will dominate core damage frequency, and risk to the public. Although the results are specific to Peach Bottom–2, the sequence categories identified in this document are sufficiently broad that they would apply to any BWR with a Mark I containment. A brief overview of NUREG–1150 methodology is presented in Section 2.2.

Step 2–Determination of Severe Accident Conditions

The conditions within the reactor coolant system (RCS), containment, and reactor building are determined from a review of the results of severe accident analyses available for BWR plants with Mark I containments. The results from the BMI–2104[3] and NUREG/CR–4624[4] analyses performed for the Peach Bottom plant were used to determine the thermal hydraulic conditions for a range of important BWR accident sequences. A brief overview of these analyses are presented in Section 2.3. These analyses are used because most of the important events expected during a severe accident from core melt through lower head failure and beyond are found in these reports. This includes the possible effects on the primary containment and the reactor building. These analyses provide a baseline for gaining insight into challenges to instrument availability. The assignment of the sequences analyzed by Battelle–Columbus to the NUREG–1150 results is discussed in Section 3. Appendix B provides a discussion of the results of the Battelle analyses and how they were categorized in NUREG–1150.

Step 3–Evaluation of Instrument Availability

Instrument availability (Step 3) is evaluated based on accident conditions, principally pressure and temperature in the vicinity of the instrument, relative to the range and qualification conditions established for the instrument. The source of backup power for each instrument is also considered for a station blackout event. The evaluation focuses on the impact of pressure and temperature conditions because they appear to strongly influence instrument availability, particularly in the early stages of the accident. Relative humidity, steam condensation, and radiation are also factored into the evaluation for instruments in the reactor building.

Instrument information for this evaluation is based on the implementation of Regulatory Guide 1.97[5] requirements for the Peach Bottom station.[6] The instrument qualification temperature and pressure conditions used for this evaluation are based on the results of the Technical Evaluation Report for equipment qualification.[7] Information on pressure and temperature qualification conditions from the Peach Bottom FSAR is also utilized.[8]

The assessment of instrument availability assumes that instrument performance will be degraded if the pressure, temperature, or radiation conditions in the vicinity of the instrument exceeds the specified qualification limits, or if the parameter being measured is outside the instrument range. This definition includes the possibility of instrument failure. Degraded instrument performance denotes that the indicated magnitude or trend of the measured parameter is in error. This error may cause the operator to take inappropriate action, cause premature termination of the operation of an automatic safety system, or start the operation of an automatic safety system when it is not required. An example would be termination of the operation of the high-pressure coolant injection system (HPCI) due to an false indication of high vessel water level.

Step 4–Assessment of Accident Management Information Needs

An assessment of accident management information needs considering instrument availability is performed. This accident management information assessment utilizes the safety objective trees, and the information needs tables developed in NUREG/CR–5702. The safety objective trees define the relationship among the safety objectives and safety functions, possible challenges to them, mechanisms causing the challenges, and strategies to prevent or mitigate the consequences of the mechanisms causing the challenges.

There are three safety objective trees used in this report for BWR plants with a Mark I containment design. Prevent Core Dispersal from Vessel, Maintain Containment Integrity, and Prevent Fission Product Release from Containment. These trees are shown in Figures 1, 2, and 3. Tables of information needs developed from the safety objective trees are presented in Appendix A of NUREG/CR–5702. These tables provide a tabulation of the information needs, and available or potential instruments for meeting a given information need.

Instrument availability is reviewed to determine the degree to which the information needs can be fulfilled. Potential limitations in terms of range and qualification conditions of the existing instrumentation that would otherwise be capable of satisfying information needs are reviewed considering the range of conditions expected during a severe accident. The accident management information assessment is discussed in Section 5 and Appendix C.

2.2 NUREG–1150 Overview

The objective of NUREG–1150 is to provide an assessment of the severe accident risks for five plants of different designs. The plants considered in the NUREG–1150 analysis were Zion (Unit 1), Surry (Unit 1), Peach Bottom (Unit 2), Grand Gulf (Unit 1), and Sequoyah (Unit 1). The Peach Bottom analysis is the basis for the plant damage state (PDS) and accident progression bin assignment discussed in Section 3 of the report.

The general approach used in NUREG–1150 is based on the systematic elicitation of expert opinion on plant system analysis that determines core damage frequency and severe accident phenomena as described by Hora and Iman.[10] Experts from various nuclear industry organizations were selected and organized into panels convened to study particular aspects of severe accidents. These experts were trained in the methods used for systematic elicitation of expert opinion. Issues were then presented to the expert panels to establish consistency and common understanding of the issues addressed. A period of time was allowed so that the assigned issues could be studied by the experts, thus allowing the development of preliminary subjective probability assessments. Meetings were held during this time to allow for the exchange of information on various issues among the experts. Afterwards, elicitation sessions were held to obtain and document the opinion of each expert. This elicitation included the justification of each opinion and a probability distribution for parameters pertinent to a given issue. The probability distributions from each expert were assembled for use in NUREG–1150.

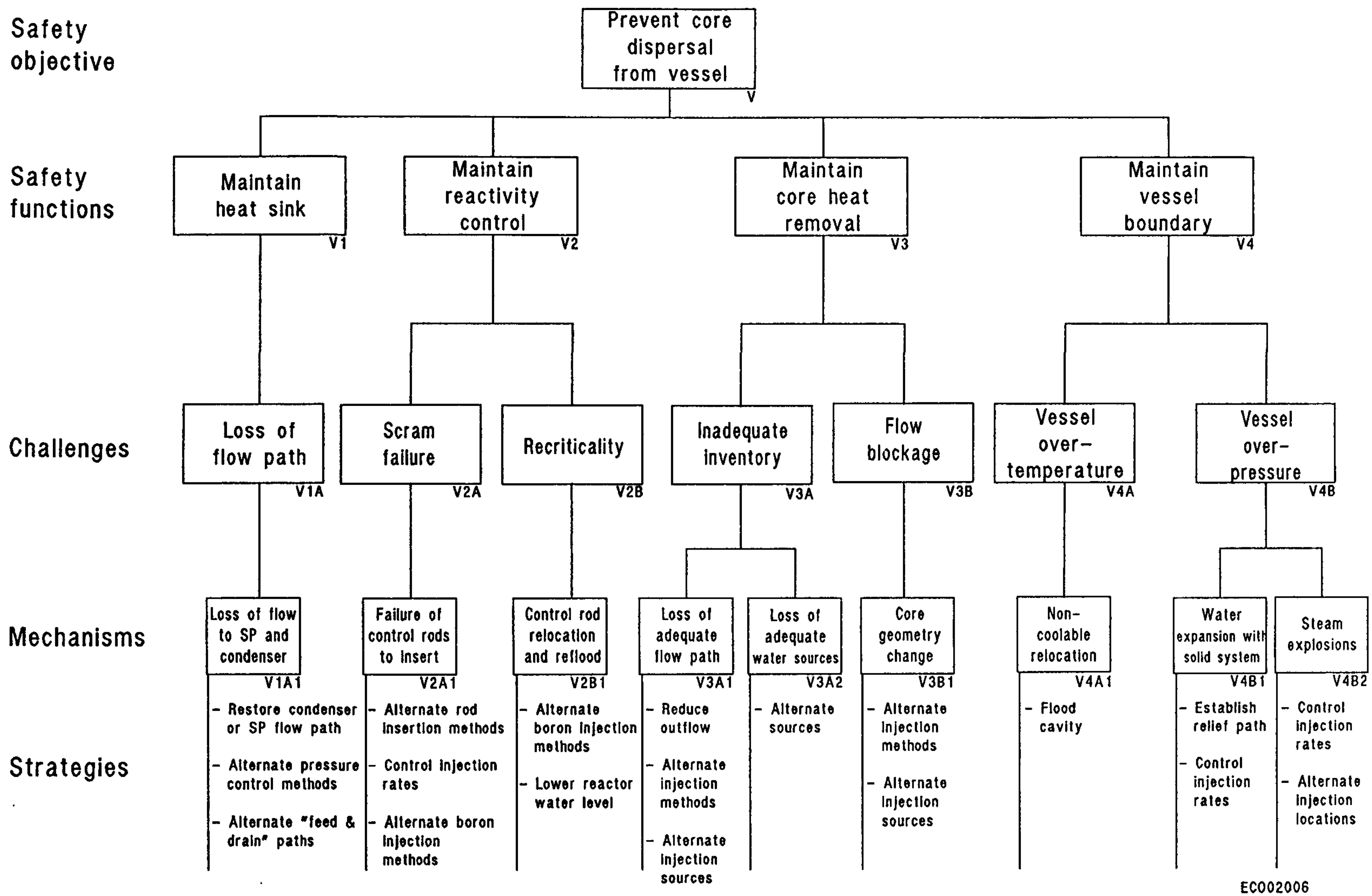

Figure 1. Safety objective tree: prevent dispersal from vessel (from NUREG/CR–5702).

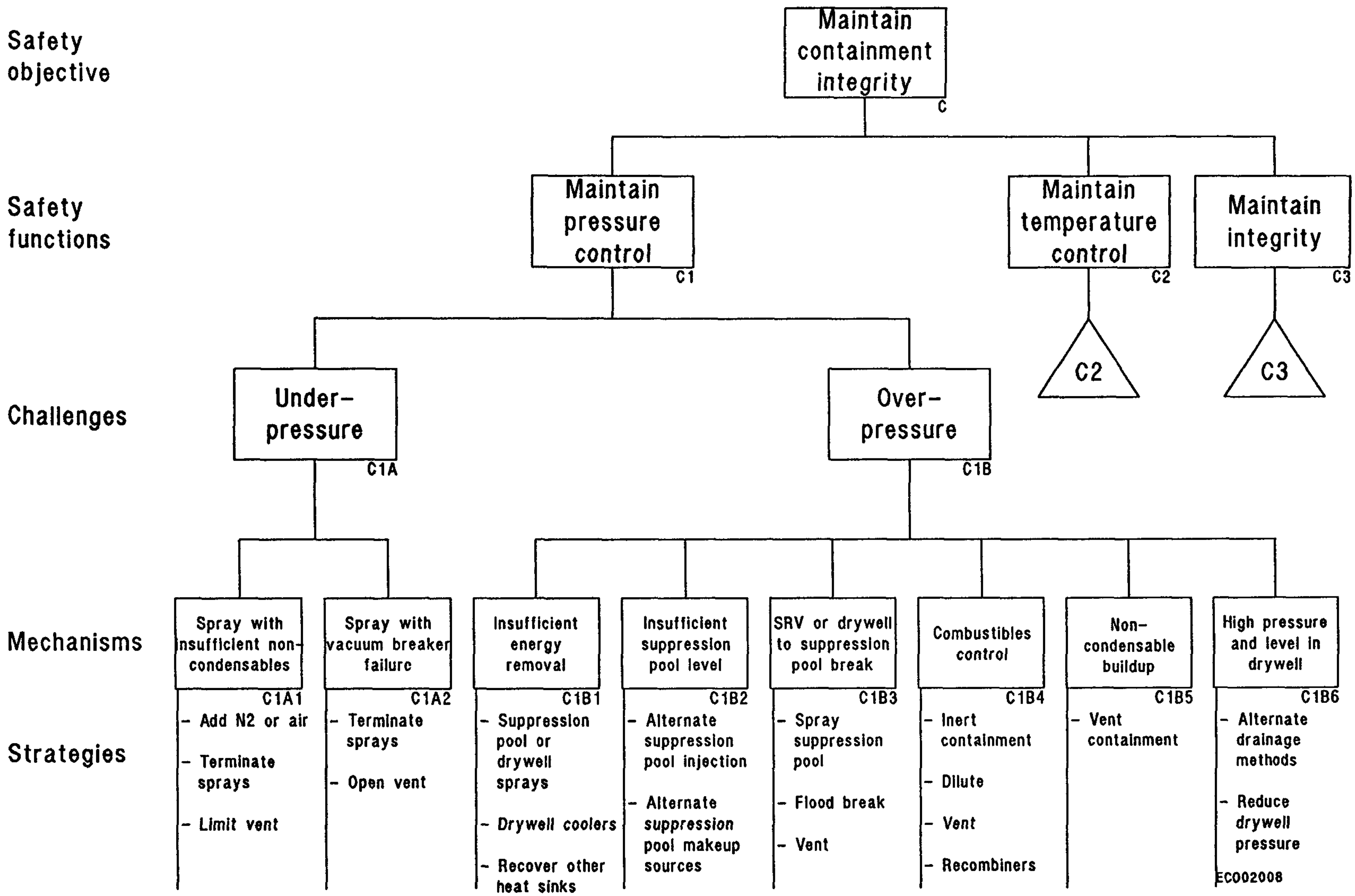

Figure 2. Safety objective tree: prevent contaminent failure (from NUREG/CR-5702).

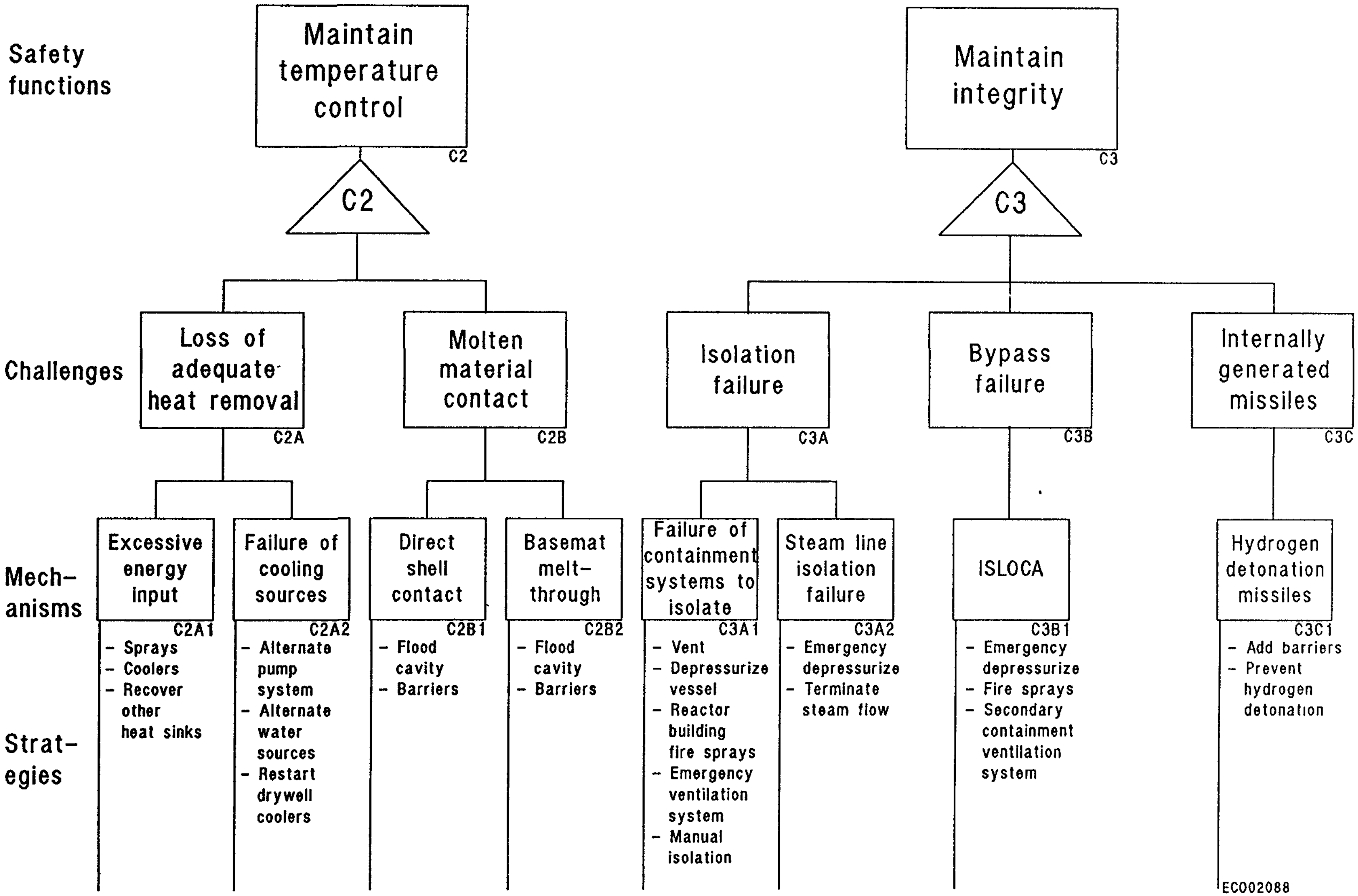

Figure 2. (continued).

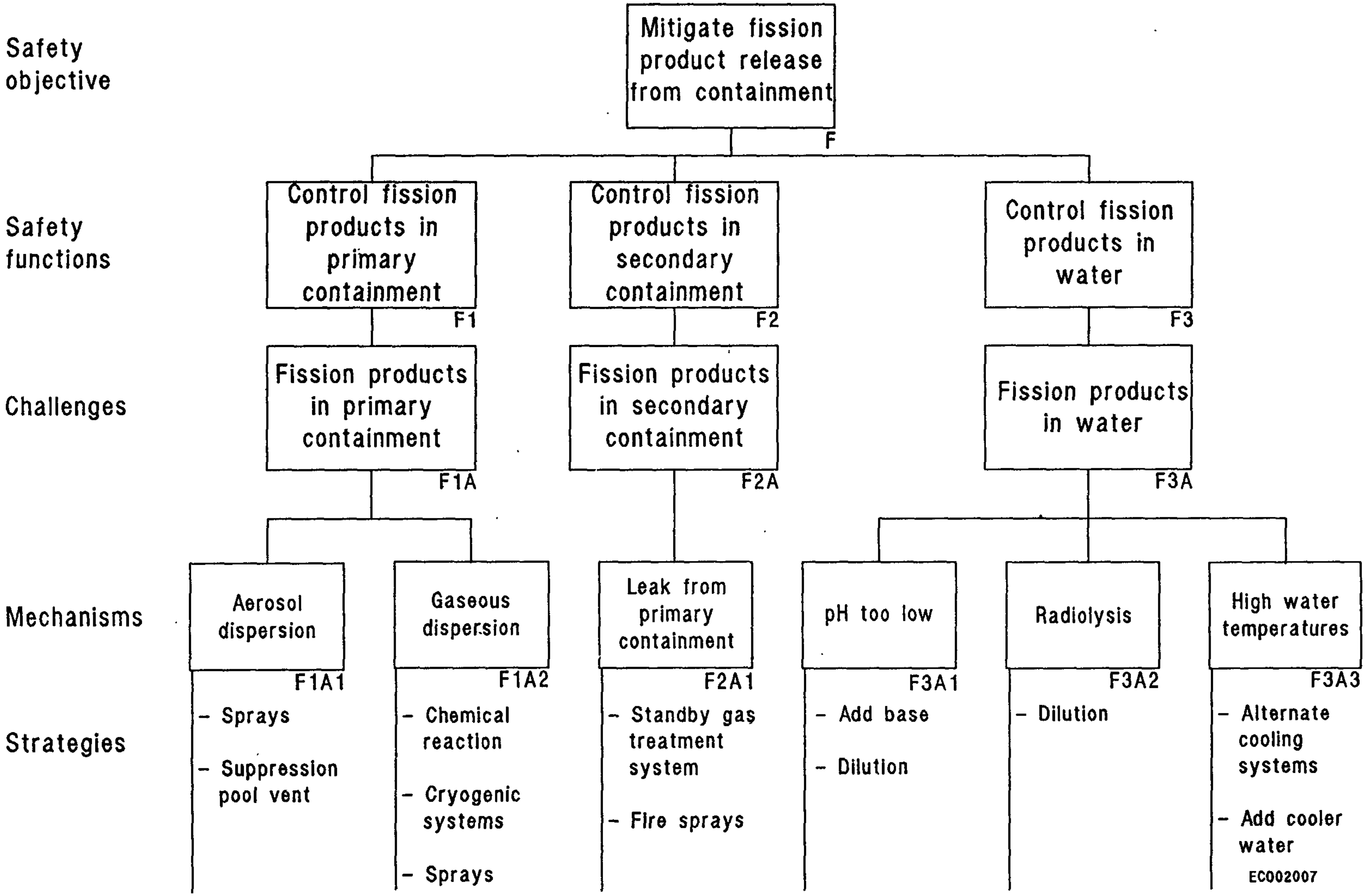

Figure 3. Safety objective tree: mitigate fission product release from contaminent (from NUREG/CR–5702).

Included in NUREG–1150, is an accident frequency analysis and an accident progression analysis. The accident frequency analysis identifies the combination of events that can lead to core damage and estimates their frequency of occurrence. This analysis results in a set of PDSs with corresponding probability levels that are subsequently used in the accident progression analysis. The accident progression analysis utilizes the results of the accident frequency analysis to investigate the physical processes affecting the reactor core after the initiating event. The results of this analysis are presented as a set of accident progression bins defining the possible outcomes for a severe accident.

2.3 Overview of BMI–2104 and NUREG/CR–4624 Analyses

The purpose of the BMI–2104 and NUREG/CR–4624 analyses was to estimate the source term magnitude for various severe accident sequences that are important to risk. The results published in BMI–2104 and NUREG/CR–4624 are based on computations performed with the MARCH2 and MERGE programs, which are currently part of the Source Term Code Package. An overview of the Source Term Code Package is presented in NUREG–0956.[11] Descriptive information on MARCH2 and MERGE is presented in Section 5 of BMI–2104, Volume VI. The MARCH2 program is used to determine the plant thermal hydraulic conditions during a severe accident and incorporates models for primary system and containment response, fuel meltdown and slump, and lower head failure, among others. The MERGE program is developed to determine detailed flow and temperature information in the upper plenum, piping, and other primary system components. This information is not available from MARCH2 and is needed to determine fission product retention in the primary system.

The BMI–2104 and NUREG/CR–4624 results are useful in performing this instrument availability evaluation since data for most of the important events expected during a severe accident from core melt through lower head failure and beyond is presented. The data is a good baseline for gaining insight into the challenges of instrument availability. There is uncertainty in the BMI–2104 and NUREG/CR–4624 results that are discussed in Section 6 of this report. However, the results are considered to be adequate for evaluating instrument availability.

3. IMPORTANT ACCIDENT SEQUENCES FOR EVALUATING INFORMATION NEEDS AND INSTRUMENT AVAILABILITY

This section presents the results from Steps 1 and 2 of the methodology discussed in Section 2.1. Identification of important accident sequences for use in the evaluation of instrument availability is based on plant damage states and accident progression bins used in the NUREG–1150 analysis for Unit 2 of the Peach Bottom Atomic Power Station. Although these PDSs and accident progression bins are specific for Peach Bottom–2, results from other probabilistic risk assessments show them to be typical for other BWRs with Mark I containments. As discussed in Section 2, thermal hydraulic conditions are determined for the PDSs and accident progression bins based on BMI–2104 and NUREG/CR-4624 results.

3.1 Plant Damage States and Accident Progression Bins

In the NUREG–1150 analysis, accident sequences for the Peach Bottom plant are grouped into four summary PDSs. They are

- Station blackout
- Large and small break loss of coolant accidents
- Anticipated transients without scram (ATWS)
- All other transients except station blackout and ATWS.

Each PDS is defined by a group of accident sequences that has similar characteristics with respect to accident progression and containment engineered safety feature operability.

Several accident progression bins are developed to cover the range of potential outcomes for each PDS. A set of ten accident progression bins are identified in NUREG–1150 for Peach Bottom that relate the time of vessel breach to the time of containment failure or containment venting. Pressure at the time of vessel breach and containment failure location are also included. Bins are also defined for accidents where containment failure, vessel breach, or core damage does not occur. Further discussion of the PDSs and accident progression bins is found in Appendix A.

3.2 Definition of Thermal Hydraulic Conditions

Thermal hydraulic conditions for the PDSs and accident progression bins presented in Section 3.1 are defined based on the BMI–2104 and NUREG/CR–4624 analyses. Of the many thermal–hydraulic parameters calculated, the parameters of interest for evaluating instrument availability is the temperature and pressure in the areas in which instrument components are located. These areas include the reactor coolant system, containment (drywell and torus), and the reactor building for all sequences. A tabulation of the maximum value of various thermal hydraulic parameters reached during the important accident phases is presented in Table 1 based on the information presented in Appendix B. A tabulation of the timing of each accident phase is presented in Table 2.

3.3 Discussion of Accident Sequence Results

A review of the NUREG/CR–1150 PDSs and accident progression bins and the thermal hydraulic results from BMI–2104 and NUREG/CR–4624 shows that if pressure and temperature inside the primary containment are approaching the point where containment failure is possible, then instrument availability and the ability to meet information needs will be affected. Instruments that are located in the drywell and torus could be subjected to pressure and temperature conditions outside of their qualification limit as containment temperature and pressure increases. If duct failure occurs when the containment is vented or if containment failure occurs, instruments located in the reactor building could be subjected to conditions outside their qualification range due to the introduction of high

Table 1. Maximum value of key parameters during each phase for all accident sequences.

Parameter	Initiation to core uncovery	Uncovery to start of melt	Meltdown to core slump	Core slump to head failure	After head failure
Within the reactor vessel					
Average core temperature (°F)	1277 (AE–γ)	2233 (AE–γ)	3368 (TC1)	3784 (AE–γ)	N/A
Core exit gas temperature (°F)	N/A	1750 (AE–γ)	3250 (AE–γ)	3250 (AE–γ)	N/A
Maximum RPV structure temperature (°F)	N/A	550 (TC–γ)	2500 (TC–γ)	2500 (TC–γ)	N/A
Maximum Reactor System Pressure (psia)	1202 (TC1)	1090 (TC2)	1150 (TC2)	1168 (TC2)	N/A
Primary containment					
Pressure (psia)	129 (TC1)	129 (TC1)	30 (TB1)	131 (AE–γ)	132 (TB1)
Temperature (°F)	324 (TC1)	324 (TC1)	276 (TC1)	2031 (AE–γ)	1391 (TC1)
Pool temperature (°F)	349 (TC1)	349 (TC1)	212 (TC1)	212 (TC1)	212 (TC1)
Reactor building					
Temperature (°F)	250 (TC1)	250 (TC1)	1200 (TC2)	250 (TC1)	2500 (TB2)

Notes:

1. The accident sequences for each parameter is given in parentheses.
2. N/A – not applicable

Table 2. Time range of key events for BMI–2104 and NUREG/CR–4624 accident sequences (minutes).

Accident sequence	Initiation to core uncovery	Core uncovery to start of meltdown	Core meltdown to core slump	Core slump to lower head failure	After head failure	Containment failure time
TB1	Up to 528.5 (528.5)	528.5 – 642.4 (113.9)	642.4 – 694.8 (52.4)	694.8 – 733.5 (38.7)	733.5 – 1333.5 (600.0)	914.5
TB2	Up to 526.9 (526.9)	526.9 – 615.5 (88.6)	615.5 – 693.7 (78.2)	693.7 – 735.8 (42.1)	735.8 – 1333.5 (597.7)	735.8
AE–γ	Up to 1.5 (1.5)	1.5 – 11.5 (10.0)	11.5 – 26.8 (15.3)	26.8 – 33.9 (7.1)	33.9 – 126.2 (92.3)	33.9
TC1	Up to 93.8 (93.8)	93.8 – 134.0 (40.2)	134.0 – 166.8 (32.8)	166.8 – 230.5 (63.7)	230.5 – 1333.5 (1103.0)	85.3
TC2	Up to 33.8 (33.8)	33.8 – 58.3 (24.5)	58.3 – 88.3 (30.0)	88.3 – 126.3 (38.0)	126.3 – 736.3 (610.0)	126.3
TC3	Up to 33.8 (33.8)	33.8 – 58.3 (24.5)	58.3 – 88.3 (30.0)	88.3 – 126.3 (38.0)	126.3 – 736.3 (610.0)	CV @ 96.3
TW–γ	Up to 2619.6 (2619.6)	2619.6 – 2747.9 (128.3)	2747.9 – 2817.1 (69.2)	2817.1 – 3055.2 (238.1)	3055.2 – 3655.4 (600.2)	1756.2

Notes:

1. Number in parenthesis is the elapsed time.
2. The value of the upper limit of the range is the accident time at which the MARCH case was terminated in the after head failure column.
3. For TC3, CV denotes containment vent.

temperature steam and noncondensible gases into the reactor building. Hydrogen burns in the reactor building are also possible if core damage has occurred. Containment failure can occur at any time during a severe accident depending on the accident initiator and the system failures that have occurred, including prior to core uncovery before any core damage has occurred.

From the reviews of NUREG–1150, BMI–2104, and NUREG/CR–4624 analyses, the types of events that can lead to containment failure before core damage has occurred are ATWS initiated accident sequences involving failure of the SLCS and transient initiated accident sequences involving failure of the containment heat removal systems. If the accident is initiated by an ATWS and efforts to reduce power to the capacity of the containment heat removal systems are unsuccessful, continued containment pressurization and containment failure is possible before core uncovery occurs. Containment venting could be initiated to avoid containment failure in an ATWS initiated accident with SLCS failure. For transients where the high and low pressure injection systems are functioning, but failure of the containment heat removal systems have occurred, the continued heat rejection to the suppression pool will again cause containment pressurization and possible failure before core uncovery occurs. Again, containment venting could be initiated to avoid containment failure. For accidents initiated by a non–ATWS transient or a loss–of–coolant accident (LOCA) and where the containment heat removal systems are functioning, the need for containment venting or the possibility of containment failure should not exist until after vessel failure when large amounts of noncondensible gas can be generated due to core concrete interaction.

Survivability of the ducts used for containment venting would be a concern although many utilities are installing a hardened system for containment venting in response to Generic Letter 89–16.[12] These hardened vent systems are typically being designed for decay heat loads. Use of a hardened vent system would prevent severe conditions from developing in the reactor building during non–ATWS accidents and alleviate concerns on the availability of instruments located in the reactor building.

During an ATWS with SLCS failure, use of the hardened vent system, (designed for decay heat loads), would decrease the rate of containment pressurization and prolong the time to containment failure during an ATWS, but does not eliminate the prospect of containment failure. If the decision was made to vent the containment through vents other than the hardened system to achieve a greater containment depressurization rate during an ATWS or if a hardened vent system is not installed, duct failure could occur. At Peach Bottom, there are a total of nine vent paths, including four 18 in. vents from either the drywell or torus as discussed in Section 2.1 of NUREG/CR–4551.[13] During an ATWS, the energy generation rate will require three or four of the 18 in. vents to reduce containment pressure, assuming power levels of about 15%. Ducts in these vents would likely fail, releasing the steam to the reactor building. High temperature conditions will result in much of the reactor building, thus affecting instrument availability. Personnel access to the reactor building will also be impeded in this situation.

The path of the steam and noncondensible gases in the reactor building will affect instrument availability. As explained in Section 2.1 of NUREG/CR–4551, the reactor building completely encloses the primary containment and consists of several floors that are generally isolated from each other except for a large open hatch that extends to the refueling floor. Blowout panels are located in the refueling bay that vent to the environment. Steam released to the reactor building will, for the most part, pass through the open hatch to the refueling floor and out through the blowout panels. A steam vent path exists from the reactor building to the turbine building through blowout panels located in the steam tunnel. Any venting using the 18 in. lines will likely open all of the blowout panels. Most of the steam is expected to exit through the refueling floor because of the larger flow area in the path to the refueling floor compared to the steam tunnel.

The possible temperature in the reactor building if the containment fails or if duct failure occurs when the containment is vented will affect the availability of instruments located in the reactor building. The NUREG/CR–4624 document presents the results of accident sequence analyses that account for conditions in the reactor building. Sequences where containment fails before core damage and where containment failure occurs after core damage were analyzed. The analysis shows that temperature conditions above the qualification limits of many instruments can be expected due to the large amount of steam and noncondensible gases released on the reactor building as the containment depressurizes. Temperatures of 210 to 230°F are predicted in the reactor building. If core damage has occurred, then the possibility of hydrogen burns in the reactor building exist. If hydrogen burns occur, local temperature spikes in excess of 2000°F in the reactor building are possible (see Appendix B).

Oak Ridge National Laboratory (ORNL) evaluated conditions in the reactor building during an ATWS with SLCS failure assuming duct failure during containment venting.[14] The CONTAIN computer program was used for these analyses. These analyses were done to evaluate containment venting as a mitigation strategy during an ATWS with SLCS failure at a BWR plant with a Mark 1 containment. Unit 1 of Browns Ferry was considered in the evaluation. These results are of interest because of the possible effects of containment venting on reactor building conditions during an ATWS.

In the CONTAIN analysis, the reactor building was subdivided into four control volumes representing the rooms at 565, 593, 621, and 639 ft of the Browns Ferry reactor building. Steam from two 18 in. vent lines is released to 565 ft. The effect of fire protection sprays at 565 and 593 ft is factored into the evaluation. Initially, the steam flow rate is 164 lb/s and after 1 hour decreases to 110 lb/s.

The temperature results from the CONTAIN analysis performed by ORNL shows a rapid rise in temperature from an ambient temperature of 100°F in all volumes to 180°F at 621 ft, to 210°F at 593 ft, and to 225°F at 565 ft after 30 minutes. The atmospheric composition in the reactor building after 60 minutes of containment venting ranges from almost pure water vapor at 565 ft to about 5% water (95% air) at 639 ft. ORNL notes that the combination of temperature and atmospheric composition conditions will impact equipment availability and will restrict the access of plant personnel to the reactor building.

Design differences among plants must be recognized in applying the NUREG/CR–4624 and the ORNL results generically to BWR plants with Mark I containment designs. ORNL noted that there are design differences between Browns Ferry and Peach Bottom. This includes no floor wide system of fire protection sprays on any floor at Peach Bottom, as well as differences in building arrangement. These differences would not significantly affect the temperature predictions or the prediction of a steam environment in the reactor building. Among a larger group of plants, design differences would include the hardened vent design and configuration and location of the blowout panels in the reactor building. Vent designs that can withstand the loads resulting from containment venting would alleviate concerns on the availability of instruments in the reactor building. If a blowout panel is located near the point of containment or duct failure, steam exiting the failure location would flow through the panel to the environment, thus bypassing most of the reactor building. In this case, there may be less of a problem with instrument availability.

4. INSTRUMENT AVAILABILITY DURING SEVERE ACCIDENTS

Instrument availability during a severe accident (Step 3 of the methodology discussed in Section 2.1) is assessed in this section. From the review of NUREG–1150 PDSs and the thermal hydraulic data from BMI–2104 and NUREG/CR–4624, it was found that the conditions that will have the greatest impact on instrument availability are the following:

- Severe pressure and temperature environments in the vicinity of the components of the instrument system causing instrument performance to degrade (severe conditions means that environmental conditions in the vicinity of the instrument components have exceeded the qualification limits)
- Electric power failure resulting from station blackout, loss of a direct current (dc) bus, or other power interruptions that cause instruments to be unavailable
- High radiation fields in the reactor building following reactor vessel rupture and containment failure cause instrument performance to degrade in the long term (days or weeks).

Instrument availability is evaluated based on the pressure and temperature conditions at the instrument location relative to the qualification conditions and the source of backup power.

Table 3 presents a list of instruments that are included in the Regulatory Guide 1.97 measurements for Peach Bottom. Table 4 lists a description of several instruments that were not listed in the Regulatory Guide. The tables include the measurement range, specified qualification conditions, sensor location, and source of power for each instrument. The data are based on qualification information from Regulatory Guide 1.97 review for the Peach Bottom Atomic Power Station and the Technical Evaluation Report for equipment qualification.

The instruments listed in Table 3 are grouped by the three electric power source categories defined in Regulatory Guide 1.97. Category 1 provides for full qualification, redundancy, and continuous real time display and requires onsite (standby) power. Onsite (standby) power does not necessarily mean that the power source has a battery backup. Category 2 provides for qualification, but is less stringent in that it does not (of itself) include seismic qualification, redundancy of continuous display, and requires only a high reliability power source (not necessarily standby power). Category 1 and Category 2 instruments are required by Regulatory Guide 1.97 to have battery backup power only when momentary interruption of the instrumentation is not tolerable. Category 3 is the least stringent. It provides for high–quality commercial grade equipment that requires only offsite power. These categories are used since the power source is an important factor in determining instrument availability during a station blackout sequence. Battery–backed power sources and their instrument loads vary widely depending on individual plant design.

Conditions that are expected to significantly affect instrument availability are the pressure and temperature conditions in the primary containment and the temperature conditions in the reactor building resulting from containment or duct failures during containment venting in the event of an ATWS with SLCS failure. The possibility of high steam concentrations in the reactor building may also affect instrument availability due to steam condensation on instrument components. The evaluation of instrument availability focuses on the location of the sensors with consideration given to electronics, cabling, splices, and other components.

Results from the instrument availability evaluation are intended to provide scoping information that can be used to understand the limits of its availability for a wide range of plant conditions during severe accidents. The results are conservative in that less availability is

Table 3. Summary of Peach Bottom Regulatory Guide 1.97 measurements.

Plant instrumentation	Range	Qualification conditions	Location of sensor	Power supply
Category 1				
Reactor pressure	0–1500 psig	178°F, 1.5 psig 100% RH 3.5 x 10^4 rads	Reactor building	Uninterruptible power (on site source backed by station batteries)
Reactor pressure	0–1200 psig	178°F, 1.5 psig 100% RH 3.5 x 10^4 rads	Reactor building	Uninterruptible power (on site source backed by station batteries)
Reactor water level	–325 to 0 in. –165 to +50 in.	250°F, 0 psig 100% RH 3.5 x 10^4 rads	Reactor building	Class 1E
Suppression pool water temperature	30 to 230°F	317°F, 49 psig 100 RH 3.5 x 10^7 rads	On torus shell	Class 1E
Suppression pool water level	1–21 ft	183°F, 0 psig 100% RH 3.5 x 10^4 rads	Reactor building	Class 1E
Drywell pressure	5 to 25 psia 0 to 225 psig	207°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Class 1E
Containment and drywell oxygen concentration	0 to 10 vol%	141°F, 0 psig 100% RH 3.5 x 10^4 rads	Reactor building	Class 1E

Table 3. (continued).

Plant instrumentation	Range	Qualification conditions	Location of sensor	Power supply
Category 1 (continued)				
Source range monitors (SRM)	Low range – less than 10^{-6}% power (inserted into core), Upper range – 2% power (not inserted into core).	317°F, 49 psig 100% RH 4.44 x 10^7 rads (Sensor qualified to ATWS conditions)	Drywell (detectors in core)	Batteries for electronics, uninterruptible power for (battery backed) recorders, onsite sources for drive motors
Intermediate range monitors (IRM)	Low range 10^{-4}% power Upper range approximately 20% power	317°F, 49 psig 100% RH 4.44 x 10^7 rads (Sensor qualified to ATWS conditions)	Drywell (detectors in core)	Batteries for electronics, uninterrupible power for (battery backed) recorders, onsite sources for drive motors
Average power range monitors (APRM)	Low range – 1.0% power. Upper range approximately 125% power	317°F, 49 psig 100% RH 4.44 x 10^7 rads (Sensor qualified to ATWS conditions)	Drywell (detectors in core)	RPS motor–generator set for electronics, uninterruptible power for (battery backed) recorders
BWR core temperature	Not installed	N/A	–	–
Drywell sump level	High–high level	317°F, 49 psig 100% RH 4.44 x 10^7 rads	Drywell	Onsite
Radioactivity concentration or radiation level in circulating primary coolant	Not installed	N/A	–	–

Table 3. (continued).

Plant instrumentation	Range	Qualification conditions	Location of sensor	Power supply
Category 1 (continued)				
Containment and drywell hydrogen concentration	0 to 20 volume %	141°F, 0 psig 100% RH 3.5 x 10^4 rads	Reactor building	Class 1E
Primary containment area radiation—high range	1 to 10^8 R/h	317°F, 49 psig 100% RH 4.44 x 10^7 rads	Drywell	Class 1E
Primary containment isolation valve position (excluding check valves)	Closed or not closed.	Reactor building 141–250°F, 0–2 psig 100% RH 3.5 x 10^4 rads Drywell – 317°F 49 psig, 100% RH 4.4 x 10^7 rads	For valves with direct position indication: The valve limit switches are located in the drywell and reactor building. For valves with indirect position indication: The valve control circuit location is outside drywell.	Varies–Class 1E or onsite
Category 2				
Vent stack effluent radioactivity	Low range: 10^{-7} to 1.6 μCi/cm^3 High range: 1.4 x 10^{-2} to 1.4 x 10^4 μCi/cm^3	150°F, 1 psig 100 RH 3.5 x 10^4 rads	Reactor building	Onsite for low range Onsite for high range sensors Offsite for the high range recorders
Suppression chamber spray flow	0 to 25,000 gpm	150°F, 0 psig 100% RH 3.5 x 10^4 rads	Reactor building	Onsite power
Drywell atmosphere temperature	40 to + 440°F	317°F, 49 psig 100% RH 4.44 x 10^7 rads	Drywell	Class 1E

Table 3. (continued).

Plant instrumentation	Range	Qualification conditions	Location of sensor	Power supply
Category 2 (continued)				
Drywell spray flow	0 to 25,000 gpm	150°F, 1 psig 100% RH 3.5 x 10^4 rads	Reactor building	Onsite power
Main steamline isolation valves leakage control system pressure	Not installed	N/A	N/A	N/A
Primary system safety relief valve position, including ADS or flow through or pressure in valve lines	Open, closed, open previously	317°F, 49 psig 100% RH 4.44 x 10^7 rads	Drywell	Class 1E
RCIC flow	0–700 gpm	120°F, 1 psig 100% RH 3.5 x 10^4 rads	Reactor building	Station batteries
HPCI flow	0–6,000 gpm	120°F, 1 psig 100% RH 3.5 x 10^4 rads	Reactor building	Station batteries
Core spray system flow	0–10,000 gpm	120°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Onsite power
LPCI system flow	0–50,000 gpm	120°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Onsite power
Standby liquid control system flow (pressure)	0–1680 psig	140°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Onsite

Table 3. (continued).

Plant instrumentation	Range	Qualification conditions	Location of sensor	Power supply
Category 2 (continued)				
SLCS storage tank level	7 – 131.25 in.	140°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Onsite power
RHR system flow	0–50,000 gpm	120°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Onsite power
RHR heat exchanger outlet temperature	0 – 600°F	120°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Uninterruptible power, battery backup
RCIC room temperature	0 – 600°F	120°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Station batteries
HPCI room temperature	0 – 600°F	120°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Station batteries
Emergency ventilation damper position	Open/closed	120°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building or Radwaste building	Onsite
Status of standby power and other energy sources important to safety (electronic, hydraulic, pneumatic) (voltages, currents, pressures)	Various ranges	250°F, 2–0 psig 100% RH 3.5 x 10^4 rads	Reactor, Turbine, Diesel generator, and Radwaste buildings	Station batteries and onsite sources

Table 3. (continued).

Plant instrumentation	Range	Qualification conditions	Location of sensor	Power supply
Category 2 (continued)				
Common plant vent or multipurpose vent discharging any of the above releases. (Offgas stack radioactivity–noble gases.)	At 20,000 CFM: the low range is 10^{-7} to 1.6 $\mu Ci/cm^3$; the high range is 1.4 x 10^{-2} to 1.4 x 10^4 $\mu Ci/cm^3$	Not found	Offgas stack equipment building	A combination of uninterruptible power (Onsite source backed by station batteries), onsite sources and station batteries for the low range; offsite for high range recorder (only)
Common plant vent or multipurpose vent discharging any of the above releases. (Unit vent stack flow).	0 to 600 KCFM	120°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Onsite
Common plant vent or multipurpose vent discharging any of the above releases. (Offgas stack flow).	0 to 40 KCFM	Not found	Offgas stack equipment building	Onsite
Category 3				
Control rod position indicator	Full–in or not full–in	317°F, 49 psig 100% RH 4.44 x 10^7 rads	Drywell	Uninterruptible power, battery backed
RCS soluble boron concentration (grab sample)	50 to 1100 ppm	Not found	Radwaste building	Onsite and offsite sources
Analysis of primary coolant (gamma spectrum)	1 $\mu Ci/cm^3$ to 10 Ci/cm^3.	Not found	Radwaste building	Onsite and offsite sources

Table 3. (continued).

Plant instrumentation	Range	Qualification conditions	Location of sensor	Power supply
Category 3 (continued)				
Offgas stack radioactivity	At 20,000 CFM: Low range – 10^{-5} to 5.0 μCi/cm^3 High range – 1.4 x 10^{-2} to 1.4 x 10^4 μCi/cm^3	Not found	Offgas stack equipment building	A combination of uninterruptible power, onsite sources and station batteries for the low range; and onsite sources for high range sensors, offsite for high range recorder
Main feedwater flow	0 to 7 x 10^6 lb/h	Not found	Turbine building	Uninterruptible power, battery backed
Condensate storage tank level	0–42 ft	140°F, 0 psig 99% RH	Turbine building	Onsite power
Turbine bypass valve position	0–100%, and open/close	140°F, 0 psig 99% RH	Turbine building	Batteries, uninterruptible power and onsite sources
Condenser hotwell level	0–32 in.	140°F, 0 psig 99% RH	Turbine building	Onsite power
Condenser vacuum	0–30 in. hg vacuum	140°F, 0 psig 99% RH	Turbine building	Onsite power
Condenser cooling water flow. (Pump discharge pressure)	0–30 psig	Not found	Circulating water pump structure	Onsite power
Primary loop recirculation flow	0–70,000 gpm	120°F, 2 psig 100% RH 3.5 x 10^4 rads	Reactor building	Onsite power

Table 3. (continued).

Plant instrumentation	Range	Qualification conditions	Location of sensor	Power supply
Category 3 (continued)				
Radiation exposure rate (inside buildings or areas where access is required to service equipment important to safety)	Varies, 0.01 to 10^4 mR/h	Not found	Main control room, vent stack effluent, spent fuel pool area, also portable monitors	Onsite, batteries for portable monitors
Particulates and halogens. All identified release points. (Sampling with onsite analysis capability)	Requirement met with proper sampling volume and counting time	Not found	N/A	N/A
Airborne radiohalogens and particulates (portable sampling with onsite analysis capability)	Requirement met with proper sampling volume and counting time	Not found	N/A	Battery for portable samplers
Reactor building or secondary contain–ment area radiation	1 sensor: 1 to 10^6 mR/h. Balance of sensors: 0.01 to 10^4 mR/h	141–250°F, 0–2 psig 100% RH 3.5 x 10^4 rads	Various locations in reactor building	Onsite
Plant and environs radiation (portable instrumentation)	0 to 2 x 10^4 R/h, gamma and beta radiations	Not found	N/A	Battery
Primary coolant and sump (grab sample)	1. Gross Activity: 1 µCi/mL to 10 Ci/mL 2. Boron Content: 50–1100 ppm 3. Dissolved Hydrogen 0 – 2000 cc/kg 4. Dissolved Oxygen 0 – 20 ppm	Not found	Radwaste building	Onsite or offsite sources

Table 3. (continued).

Plant instrumentation	Range	Qualification conditions	Location of sensor	Power supply
Category 3 (continued)				
Containment air (grab sample)	1. Hydrogen Content: 0.1 to 30% 2. Oxygen Content: 0.1 to 30% 3. Gamma Spectrum (Isotopic analysis)	Not found	Radwaste building	Onsite or offsite sources
High radioactivity liquid tank level	0 – 100% level (top to bottom)	Not found	Radwaste building	Uninterruptible power

Table 4. Summary of Peach Bottom measurements not listed in Regulatory Guide 1.97.

Plant instrumentation	Range	Qualification conditions	Location of sensor	Power supply
Reactor building temperature	–20 to 200°F	Not found	Reactor building	Onsite or offsite sources
Reactor building pressure	0 – 50 psig	Not found	Reactor building	Onsite or offsite sources

predicted in this study than would be predicted by a more detailed, plant–specific study for the following reasons:

- Specified instrument qualification conditions were used rather than actual qualification conditions. The actual conditions may exceed the specified qualification conditions because most instruments are tested to more severe environments than those specified by the licensee. This difference could increase instrument availability for some accident sequences.

- More detailed analysis of the environmental conditions at the location of instrument components in the containment and reactor building would tend to increase availability. Location of instrument components varies widely from plant to plant and specific instrument components may be relatively protected from severe conditions expected during an accident.

- Degraded performance of instruments is likely influenced by the length of time and the magnitude of the difference between the environmental conditions and the qualification conditions. If the environmental conditions exceed the specified qualification conditions by small amounts or for short periods of time, the instruments would likely remain available. In this analysis degraded performance is assumed when the qualification limits are exceeded, regardless of the magnitude of the difference or length of time.

Plant–specific evaluations of instrument availability would be necessary to eliminate conservatism from the results. These evaluations would need to include an assessment of the relationship between the instrument uncertainties and the timing and degree to which the qualification conditions are exceeded, based on a detailed study of basic instrument capabilities and failure modes. These plant–specific evaluations are beyond the scope of this study.

The results of the study should provide an understanding of conditions where an instrumentation system response may become unreliable, thus affecting the ability of licensees to effectively diagnose and manage severe accidents. When coupled with the results of a proposed study to evaluate the actual response characteristics of selected representative systems when operated beyond their qualification or design limits, the NRC staff should have a strong technical basis to evaluate the accident management claims of licensees, as well as the evidence to justify the adequacy of their instrument systems for implementing appropriate accident management procedures and guidance.

It should be mentioned that operators may not recognize that instrument performance is degraded. An instrument reading could appear to be normal or the trends plausible when the plant conditions and trends are different than indicated. As a result, the operators could be misled about plant conditions and pursue inappropriate operation strategies. A more detailed evaluation on the expected accuracy and reliability of the instruments for conditions where the qualification limit is exceeded is recommended. Ways that erroneous instrument readings can be recognized by operators are also needed. This evaluation should consider the entire instrument loop, including transducers, transmitters, amplifiers, cabling, electronics, and other instrument system components.

4.1 Evaluation of Instrument Availability During Severe Accidents

The principal environmental challenge to any instrument is the occurrence of severe pressure and temperature conditions in the vicinity of the instrument. These conditions can result in degraded instrument performance as defined in Section 2.1. As used in this evaluation, severe conditions means that conditions in the vicinity of the instrument have exceeded the specified qualification limits. Severe conditions will occur within the reactor coolant system for any accident resulting in core meltdown. Severe conditions can also occur in the containment (drywell and torus) and in the reactor building prior to the occurrence of core damage for accidents initiated either by an ATWS with SLCS failure or for transient initiated accidents with successful actuation of core cooling systems but where containment heat removal systems have failed. In either case, continued heat rejection to the suppression pool will cause drywell and torus pressurization. Severe conditions can also occur in the containment after lower head failure due to generation of noncondensible gases. If containment failure occurs or if duct failure occurs after the containment is vented, then release of the steam and noncondensible gases can cause severe conditions in the reactor building.

Radiation could affect instrument availability in the longer term (days or weeks) if core melt occurs. Instrument components located in the reactor building could be particularly susceptible since these instruments are generally qualified to a integrated dose limit of 3.5×10^4 rads. This integrated dose could be exceeded in a few hours in a core melt accident where containment failure occurs. For instruments located in the containment, the radiation qualification limit is generally 4.4×10^7 rads. The length of time required to exceed this dose is on the order of a few weeks, assuming a realistic amount of fission product retention in the suppression pool. In either case, the principal challenge to instrument availability is judged to be pressure and temperature conditions in the vicinity of the instrument components. Degradation of instrument components that is induced by radiation will probably occur only if the instrument survives the severe pressure and temperature conditions. The long–term effect of radiation on instrument availability is discussed further in Appendix D.

Table 5 presents a summary of the instrument availability evaluation grouped in the three electrical power categories defined in Regulatory Guide 1.97. Table 6 lists several instruments that were not included in the Guide. Availability is described for the following situations:

- Severe conditions only in reactor system
- Severe containment conditions before core damage
- Severe containment conditions after core damage
- Severe reactor building conditions before core damage
- Severe reactor building conditions after core damage.

This approach is used because of the possibility of severe conditions in the containment and reactor building prior to core damage during an ATWS or accidents where the containment heat removal systems have failed.

4.1.1 Instruments Located in the Reactor Coolant System. The only instruments located in the reactor coolant system used for accident management are the detectors for the source range monitor system, the intermediate range monitor system, the local power range monitor system and the average power range monitor system. These systems would provide important information during a severe accident because they would be used to monitor the reactivity safety function.

Severe conditions will develop in the reactor coolant system if core uncovery occurs and core damage starts. Degraded performance and the failure of detectors for the systems mentioned above, will occur as temperatures approach core meltdown for any severe accident since temperatures approaching 3500°F or more would occur. As discussed in the following section, there is the possibility that the performance of these systems would degrade before core damage occurs as a result of severe conditions in the containment or reactor building.

4.1.2 Instruments Located in the Containment (Drywell or Torus). Instrument sensors located in the drywell, as listed in Table 3, include the drywell sump level, primary containment area radiation monitor, and drywell atmosphere temperature. Instrument sensors to monitor suppression pool temperature are located on the torus shell. The motorized drives for the movable

Table 5. Summary of instrument availability.

Category 1

A	Instrument Available
/	Degraded Performance Possible

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Reactor pressure	V1, V2, V3, V4, C2	A	A	A	/	/
Reactor water level	V2, V3, V4, C1	A	A	A	/	/
Source range monitor	V2	/	/	/	/	/
Intermediate range monitor	V2	/	/	/	/	/
Average power range monitor	V2	/	/	/	/	/

Table 5. (continued).

Category 1 (continued)

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Suppression pool water temperature	V1, V2, C1, C2	A				
Suppression pool water level	V1, C1	A	A	A		
Drywell pressure	V1, V2, V4, C1, C2, C3, F3	A	A	A		
Drywell sump level	V2, V3 V4, C3	A				
Primary containment isolation valve position (drywell)	C3	A				
Isolation valve position (reactor building)	C3	A	A	A		
Containment and drywell oxygen level	V1, C1, C3	A	A	A		

Table 5. (continued).

Category 1 (continued)

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Containment and drywell hydrogen concentration	V3	A	A	A		
Containment area radiation — high range	V1, V3, V4, C1, C2, C3, F1, F2, F3	A				
Main steam isolation valve position	V1, V4, C3	A				

Table 5. (continued).

Category 2

A Instrument Available

Degraded Performance Possible

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Standby liquid control system flow (pressure)	V2	A	A	A		
Standby liquid control system storage tank level	V2	A	A	A		
Primary system safety relief valve position (or flow)	V1, V2, C2, C3	A				
RCIC flow	V1, V4, C1	A	A	A		
HPCI flow	V1, V4, C1	A	A	A		

Table 5. (continued).

Category 2 (continued)

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Core spray flow	V1, V4	A	A	A		
LPCI flow	V1, V4, C2	A	A	A		
RHR system flow	V1, C1	A	A	A		
RCIC room temperature	V1, C1	A	A	A		
HPCI room temperature	V1, C1	A	A	A		
RHR heat exchanger outlet temperature	V1, C1	A	A	A		

Table 5. (continued).

Category 2 (continued)

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Suppression chamber spray flow	V4, C2	A	A	A		
Drywell atmosphere temperature	V4, C1, C2	A				
Drywell spray flow rate	V4, C1, C2	A	A	A		
Vent stack effluent (radioactivity)	V1, V4, C3, F1	A	A	A		
Emergency ventilation damper position	C1	A	A	A		
Common plant vent or multipurpose vent release (unit vent)	V1, V4, C3, F1	A	A	A		

Table 5. (continued).

Category 2 (continued)

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Common plant vent or multipurpose vent release (offgas)	V1, V4 C3, F1	A	A	A	A	A
Status of power (electrical and other energy sources)	V1	A	A	A	A[b]	A[b]

Table 5. (continued).

Category 3

A — Instrument Available

/ — Degraded Performance Possible

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Control rod position indicator	V2	A	/	/	/	/
RCS soluble boron concentration (grab sample)	V2	A	A	A	A	A
Main feedwater flow rate	V1	A	A	A	A	A
Primary loop recirculation flow	V1	A	A	A	/	/
Analysis of primary coolant (gamma spectrum)	V1, V3, V4, F1	A	A	A	A	A

Table 5. (continued).

Category 3 (continued)

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Reactor building or secondary containment area radiation monitor	C3, F2	A	A	A		
Turbine bypass valve position indicator	V1	A	A	A	A	A
Condenser vacuum	V1	A	A	A	A	A
Condenser cooling water flow	V3	A	A	A	A	A
Condensate storage tank level	V3	A	A	A	A	A
Containment gases, H_2, O_2, gamma (grab sample)	V1, V3, V4, C1, C2, C3, F1, F2, F3	A	A	A	A	A

Table 5. (continued).

Category 3 (continued)

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Primary coolant activity, boron, H_2, O_2, (grab sample)	V1, V3, V4, F1	A	A	A	A	A

a. Abbreviations for safety function identification (from Figures 1, 2, and 3):
- V1 Maintain Heat Sink
- V2 Maintain Reactivity Control
- V3 Maintain Core Heat Removal
- V4 Maintain Vessel Boundary
- C1 Maintain Pressure Control
- C2 Maintain Temperature Control
- C3 Maintain Integrity
- F1 Control Fission Products in Primary Containment
- F2 Control Fission Products in Secondary Containment
- F3 Control Fission Products in Water

b. Portions of this system are located in the reactor building, turbine building, radwaste building, and diesel generator building. All systems would be available except those with components located in the reactor building, which could experience degraded performance.

Table 6. Summary of Peach Bottom measurements not listed in Regulatory Guide 1.97.

A — Instrument Available

Degraded Performance Possible

Plant instrumentation	Safety[a] functions	Severe conditions only in reactor system	Severe containment conditions before core damage	Severe containment conditions after core damage	Severe reactor building conditions before core damage	Severe reactor building conditions after core damage
Reactor building pressure	C3	A[b]	A[b]	A[b]		
Reactor building temperature	C3	A[b]	A[b]	A[b]		

a. Abbreviations for safety function identification (from Figures 1, 2, and 3):
C3 Maintain Integrity

b. Qualification conditions not found in the available literature.

detectors used in the source range monitors (SRM) and intermediate range monitors (IRM) are also located in the drywell. Some BWR plants may have additional equipment located in the drywell, such as reference legs for the reactor vessel level system.

Severe Conditions in the Containment Before Core Damage

Degraded performance of instruments in the drywell and torus is possible during accidents where containment pressurization occurs prior to core damage. The principal challenge to instrument availability is high pressure conditions generated from continued heat rejection to the suppression pool during an ATWS or resulting from failure of the containment heat removal systems as explained earlier. Pressurization resulting from these postulated accidents would reach 100 to 115 psia before containment venting is initiated. For Peach Bottom, this is almost twice the instrument pressure qualification limit of 64 psia. If the containment is not vented, then higher pressures approaching the mean failure pressure of 165 psia are possible. The mean failure pressure of 165 psia was used in the NUREG–1150 evaluation of Peach Bottom. The primary containment area radiation monitor may be particularly affected by pressures above the qualification limit since a gas filled detector tube is used which could be affected by pressure changes. Containment temperature would also rise above the instrument qualification limit as the mean failure pressure is approached.

Temperature conditions in the containment resulting from an ATWS or from the failure of containment heat removal systems will principally affect the suppression pool water temperature indication since the upper limit of the range of this instrument will be exceeded. In the case of suppression pool temperature, the upper limit of the instrument range is 230°F. This limit would be exceeded by 100°F or more during an ATWS with SLCS failure.

Degraded performance of the motorized drives used for the SRM and IRM systems is also possible due to severe conditions in the drywell. As a result, the ability to monitor core power during an ATWS could be affected.

Severe Conditions in the Containment After Core Damage

A review of the BMI–2104 and NUREG/CR–4624 results show that drywell and torus pressure and temperature spikes are predicted to occur suddenly at normal reactor coolant system pressure due to the release of the steam and non–condensible gases from the vessel upon vessel failure. These conditions are illustrated by the results of the TB2 analysis presented in Appendix B. A drywell temperature spike of 900°F is predicted for the TB2 analysis at the time of lower head failure with a corresponding rise in pressure to 100 psia. Instruments located in the drywell could experience temperatures and pressures well above the qualification limits for brief periods of time. Exposure to these conditions could result in degraded instrument performance. Containment hydrogen burns are not considered in this evaluation since the containment is inerted with nitrogen.

Both pressure and temperature in the containment will rise after vessel failure due to generation of hot non–condensible gases from concrete decomposition. The result is illustrated by the TB1 analysis presented in Appendix B. In this case, temperature and pressure increases gradually until either the containment is vented or containment failure occurs. Degraded performance of the drywell atmosphere temperature, suppression pool temperature, or containment area radiation monitor instruments would not be expected until the temperature or pressure increased beyond the qualification limit in the containment.

4.1.3 Instruments Located in the Reactor Building. Severe conditions in the reactor building will have the greatest effect on instrument availability during a severe accident. The principal reasons are because components of many instrument systems are located in the reactor building as seen from Table 3 and because the qualification limits are generally lower when compared to instruments located in the containment.

Severe Conditions in the Reactor Building Before Core Damage

The principal challenge to availability of instrument located in the reactor building is the flow of high temperature steam that would be released to the reactor building if the containment fails. Containment venting could also release high temperature steam to the reactor building if vents other than the hardened vent system are used during an ATWS with SLCS failure or if a hardened vent system is not installed. This steam could cause the temperatures in many reactor building locations to approach 250°F, which is above the temperature qualification limit for most instruments located in the reactor building. As a result, degraded performance of these instruments is expected.

An additional challenge to instrument availability in the reactor building is the effect of steam condensation on instrument components, particularly electronic components. Condensation on component surfaces could cause failure due to electrical shorts.

Severe Conditions in the Reactor Building After Core Damage

If both core damage and containment failure occurs, severe temperatures and high steam concentrations will occur in some areas of the reactor building causing degraded performance of the instruments in those areas. In addition, there is the possibility of hydrogen burns in the reactor building. These hydrogen burns can cause temperature spikes in excess of 2000°F. It is noted that the reactor building is compartmentalized and that the effect of hydrogen burns on instrument performance could be localized.

Some testing has been conducted to assess the effects of hydrogen burns on typical nuclear reactor instrumentation system components.[15,16] Results from these tests indicate that a single hydrogen burn would not fail either the transducers or cabling of the tested systems. However, multiple hydrogen burns would cause temperatures to exceed the qualification limits. Both transducers and cabling failed when multiple hydrogen burns were used in the tests. Based on these results, degraded performance of the instrument systems in the reactor building is assumed when multiple hydrogen burns were predicted. It is recognized that the general assumption that multiple hydrogen burns will degrade performance of all instruments is conservative since the extent of the failures would be dependent on the building design, the amount of hydrogen released, and instrument system hardware.

4.1.4 Instruments Located in the Turbine and Radwaste Buildings. Instruments located in the turbine building at Peach Bottom may be exposed to steam from the reactor building if duct failure occurs during containment venting or if containment failure occurs. As discussed in Section 3.3, there are blowout panels in the steam tunnel at Peach Bottom that provide a path from the reactor building to the turbine building if the panels are opened. However, most of the steam and hydrogen would be vented through the refueling bay to the environment at Peach Bottom. As a result, instruments located in the turbine building should remain available for all accident sequences. At other BWR plants, differences in blowout panel location could cause more or less steam to be vented to the turbine building, which could affect instrument availability.

Instruments located in the radwaste building should not be affected by any release of steam or hydrogen from the reactor building.

4.2 Evaluation of Instrument Availability During a Station Blackout or Loss of a dc Bus

Table 3 presents a summary of the backup power sources available for each instrument at Peach Bottom. The backup power sources identified on Table 3 are listed below:

- Class 1E – these power sources meet the requirements of IEEE Standard 308 and are typically backed up by diesel generators and batteries, but not necessarily both.

- Batteries – station batteries
- Uninterruptable power – onsite or offsite power sources backed up by station batteries
- Onsite power – alternating current (ac) power sources backed up by diesel generators
- Offsite power – offsite ac power sources.

Class 1E power sources that have a battery backup typically have diesel generator power charging the battery. No indication on which Class 1E power sources are backed up by batteries is given in the Regulatory Guide 1.97 review for Peach Bottom.

The availability of instrumentation during a station blackout or loss of a dc bus is dependent on the plant design. Instrument availability would not be uniform among all plants. If the licensee has provided a battery backup for all Category 1 or Class 1E equipment, then these instruments would be available at the beginning of the station blackout. The duration of the instrument availability depends on the battery design, size, load, and load shedding. Generally, Category 2 or Category 3 instruments would not be available, although some plants have some Category 2 or 3 equipment on battery backup.

During a station blackout, systems that are used to obtain and analyze samples of reactor coolant, drywell or torus atmosphere, and suppression pool water may not be available. As a result, information needs requiring sampling information may not be met.

If a severe accident sequence is initiated by a loss of a dc bus, then Category 1 instruments that are powered from another dc bus would be available since Regulatory Guide 1.97 provides for redundancy for Category 1 instruments. As the dc bus is backup to a primary ac source, or alternately the dc bus is the primary power source with the ac source as a backup, in all probability, no instrument will be lost in this event. Loss of an instrument ac bus is, however, another matter and is addressed in I&E Bulletin 79–27[17] along with loss of a dc bus. Some of the Category 2 or 3 equipment would be unavailable since there are two dc buses and presumably Category 2 and 3 instruments are powered from one of the two buses. Instrument availability during a severe accident initiated by a loss of a dc bus must be evaluated for a specific plant due to differences in instrumentation design.

5. EVALUATION OF INSTRUMENTATION NEEDS AND INSTRUMENT AVAILABILITY

Information needs for each of the safety functions presented on the safety objective trees are reviewed in this section. This review utilizes the information presented in the safety objective trees presented as Figures 1, 2 and 3 of this report, and the tabulated data in Appendix A of NUREG/CR–5702.

5.1 Summary of Safety Function Evaluation

A review of the information needs based on instrument availability for the safety functions listed in Figures 1, 2, and 3 is presented in Appendix C of this report. Table 5 lists the affected safety functions for each instrument. Note that the safety function identifiers (V1, V2, etc.) are also shown in Figures 1, 2, and 3.

Important findings from the safety function review are presented in the following sections.

Severe Containment/Reactor Building Conditions Before Core Damage

If severe conditions develop in the containment (drywell and torus), performance of key instruments used to monitor the Maintain Heat Sink (V1) and Maintain Reactivity Control (V2) safety functions may degrade, limiting the ability to monitor these safety functions. The possible effect on the source, intermediate power, and average power range monitors that are located in the drywell is particularly important. In the event of an ATWS with failure of SLCS, performance of these instruments will degrade, affecting the ability to monitor core power.

Severe pressure conditions in the containment affect the ability to monitor the containment safety functions. Important instruments that monitor these conditions are the drywell temperature instruments. They are qualified to 64 psia. The ability to monitor the Maintain Temperature Control (C2) safety function, which relies on the drywell temperature instruments, could be affected when the design pressure is exceeded.

If severe conditions develop in the reactor building because of containment failure or because of duct failure during containment venting during an ATWS, then the performance of instruments in the reactor building will degrade. Affected instruments would include the reactor vessel water level monitor, reactor coolant system pressure, and drywell pressure instruments. In the event of an ATWS, or an accident with successful ECCS function, but with loss of containment heat removal, the ability to monitor vessel inventory and reactor coolant system pressure could be limited. The reactor coolant system pressure is a key parameter in monitoring the Maintain Heat Sink (V1), Maintain Reactivity Control (V2), Maintain Core Heat Removal (V3), and Maintain Vessel Boundary (V4) safety functions. The vessel water level is needed to monitor the Maintain Core Heat Removal (V3) safety function.

One possible outcome of the accident progression for ATWS is that power reduction strategies will be successful before core damage occurs. If duct failure has occurred during containment venting, then severe conditions will have occurred in both the containment and reactor building before core damage. Many of the instruments needed to monitor conditions in the reactor coolant system and containment may have degraded in performance as a result of these severe conditions, increasing the difficulty of determining that a given strategy is successful.

Severe Containment/Reactor Building Conditions After Core Damage

If containment cooling is maintained, severe conditions will probably not develop in the containment until after failure of the lower head. At that time, the Maintain Temperature Control (C1), Maintain Pressure Control (C2) and Maintain Integrity (C3) safety functions will be challenged. The principal challenge to the instruments located in the drywell and torus used to monitor these safety functions will be increasing temperature due to hot noncondensible gas generation due to core–concrete interaction.

If containment failure occurs, severe temperatures and high steam concentrations will occur in some areas of the reactor building, causing degraded performance of the instruments in those areas. There is also the possibility of hydrogen burns occurring in the reactor building that could also degrade instrument performance.

5.2 Installation of Core Temperature Measurements to Meet Information Needs

From the perspective of accident management, one possible limitation in the instrumentation in a BWR is the inability to directly monitor core temperature. During normal operation and design basis accident conditions, a BWR operates at or near saturation conditions and the temperature in the core region can be inferred from the system pressure. Once core uncovery and heatup begins, core temperature becomes unknown.

Core temperature measurement is listed in Regulatory Guide 1.97. However, the decision to require core temperature measurements was not finalized at the time Regulatory Guide 1.97 was published. Regulatory Guide 1.97 review for Peach Bottom states that BWR core temperature instrumentation is not installed based on justification provided in report SLI–8211[18] from the BWR Owner's Group. Section 6.1.b of Supplement No. 1 of NUREG–0737[19] excludes BWR core thermocouples from Regulatory Guide 1.97 instrumentation requirements.

It is recommended that the need for core temperature indication for severe accident management be evaluated. This evaluation should include alternate methods to obtain core temperature indication in lieu of measurements. The need for core temperature measurement should also be evaluated from a cost–benefit perspective since installation of these instruments will likely exceed several million dollars.

5.3 Identification of Analysis Aids to Meet Information Needs

Analysis aids are analytical tools used by the technical support staff to provide current plant status information and projections of expected behavior during a severe accident. These aids can be based on numerical first principle methodologies, artificial intelligence techniques, or a combination of both.

Two analysis aids that would be useful to the technical support staff during a severe accident include the following:

- An analysis aid to project core power level as a function of ECCS flow and other relevant parameters in the event of an ATWS
- An analysis aid to determine plant status from interpretation of results of the sampling and analysis of the contents of the reactor coolant system and containment.

As part of the accident management strategy during an ATWS, proposals to reduce core power by throttling ECCS flow have been made. An analysis aid to project the effect of throttling ECCS flow on core power could be developed to give technical support teams greater flexibility in managing an ATWS. This type of aid has been proposed in NUREG/CR–5736.[20] It is anticipated that this analysis aid would be developed from a detailed data base of coupled thermal hydraulic and neutronic calculations for the range of thermal hydraulic conditions and a representative set of control rod positions possible during an ATWS. It is expected that the aid itself could be a relatively simple set of formulations to estimate core power as a function of reactor coolant system parameters. Most of the work in developing this analysis aid would be in performing the analyses necessary for developing the set of formulations.

Analysis aids can also be developed to assess plant status through the interpretation of results of sampling and analysis of the contents of the reactor coolant system and containment. Sampling and analysis will be an important source of information on plant status, particularly if core melting or vessel failure occurs. During a severe accident, samples of the contents of the reactor coolant system and the containment are taken and analyzed for radionuclide composition and hydrogen content. The degree of core damage in later stages of the accident can be inferred from the radionuclide and hydrogen content of the samples based on experimental data on radionuclide release from the fuel during core meltdown, core–concrete interaction and other severe accident phenomena.

An approach that could be used to infer the plant state will be to establish criteria for the occurrence of a particular plant mechanism that challenges the plant safety functions based on the radionuclide and hydrogen content of the sample. Radionuclides will be categorized based on expected chemical behavior during the different phases of a severe accident. Radionuclides expected to evolve from the fuel during the different phases of a severe accident have been evaluated through various research programs sponsored by the NRC. As a result, a data base exists which can aid in estimating such parameters as the extent of core damage or if the core concrete interaction is in progress. The occurrence of a given mechanism challenging the plant safety functions will be inferred from the concentration of radionuclide and hydrogen in a sample and the trend of that concentration based on all samples taken.

The estimated cost of developing the analysis aid to project core power as a function of ECCS flow would be in excess of $1 million dollars, principally because of the large number of neutronic and thermal hydraulic calculations that would have to be done to encompass the range of conditions possible during an ATWS. The cost of developing an analysis aid to determine plant status from interpretation of results of the sampling and analysis of the contents of the reactor coolant system, containment atmosphere, and suppression pool is expected to be in the range of $100,000 to $500,000. The costs includes development and testing.

6. ENVELOPE OF SEVERE ACCIDENT PLANT CONDITIONS AND EVENT TIMING

This section presents an envelope of severe accident conditions based on the BMI–2104 and NUREG/CR–4624 analyses and provides a discussion of uncertainties in the results. The thermal hydraulic and timing data are intended to provide an indication of the conditions to be expected for a broad range of severe accidents. However, it is not recommended that this data be used for establishing qualification conditions. Qualification conditions should be developed based on plant–specific evaluations. The envelopes are most useful in assessing the adequacy of envelopes proposed for plant specific applications.

6.1 Envelope Definition

The envelope for severe accident plant conditions is defined as an upper limit which covers the expected pressure and temperature for each accident phase for any sequence. The data in Table 1 presents a summary of the maximum value of the key thermal hydraulic parameters from the BMI–2104 and NUREG/CR–4624 analyses. These data represent a bounding envelope based on these analyses and show the magnitude of the pressure and temperatures expected during a severe accident.

A summary of the elapsed time for each accident phase for each of the BMI–2104 and NUREG/CR–4624 accident sequences is presented in Table 2. These data give an indication of the timing of the accident progression and could be used to estimate the time that instruments would be available.

6.2 Envelope Uncertainty

The uncertainty of the prediction of temperature and pressure varies with the type of sequence being considered. For sequences where severe conditions develop before core damage during an ATWS with SLCS failure or a transient with ECCS function and failure of containment heat removal systems, there is little uncertainty that continued heat rejection to the suppression pool will cause the pressure and temperature in the containment (drywell and torus) to increase. If the accident is postulated to continue, the pressure will increase to the point where either containment venting is required or containment failure occurs. Under these conditions, there is little uncertainty that instrument performance could be affected.

Severe conditions occurring in the reactor building before core damage severe enough to degrade instrument performance is more uncertain. Again, there is a strong dependence on the type of accident being postulated. In the case of an ATWS with SLCS failure, the major uncertainty is the power generation rate. The effect of containment venting on reactor building conditions is also an uncertainty during an ATWS although, from the perspective of instrument performance, there is little uncertainty that release of high temperature steam to the reactor building as a result of duct failure during containment venting could impact instrument performance. For a transient with ECCS function and failure of the containment heat removal systems, severe accident phenomena such as the rate of noncondensible gas generation during core concrete interaction are the major uncertainties.

There is more uncertainty in assessing the performance of instruments located in the drywell and torus because of the generation of steam and hot gases at different times during the accident. The uncertainty here is the temperature predictions which are sensitive to the analytical assumptions made. An example is the rapid containment pressure increase predicted in the AE–γ sequence leading to containment failure which is due to the assumed core slumping scenario (See Section B.1.2 of Appendix B).

If severe conditions develop after core damage, there is little uncertainty in the conclusion of degraded performance or failure of instruments located in the reactor vessel if exposed to the temperatures expected during a core melt, which are well in excess of the qualification temperatures. Whether the core melting temperature is 4000 or 4500°F does not alter this conclusion. However, with the exception of the detectors for the neutron monitoring system, there are no Regulatory Guide 1.97 instruments located near the BWR core.

The event timing presented in Table 2 is judged to be conservative, meaning that the time until major severe accident events occur such as the start of core melt will be longer. Analysis using the current generation of severe accident simulation programs such as SCDAP/RELAP5 or MELCOR should support this judgment.

7. CONCLUSIONS AND RECOMMENDATIONS

Based on the review of thermal hydraulic results from BMI–2104 and NUREG/CR–4624, it is concluded that instrument availability for BWRs with MARK I containments is strongly influenced by (a) the severe accident phenomena, (b) the design features of the reactor vessel, containment, and reactor building including the containment vent design, and (c) the sequence of events that occur during the severe accident. Severe accident phenomena important to the availability of instruments located in the reactor coolant system or containment include the cladding and fuel melting temperature, the amount of energy transferred to the containment atmosphere upon vessel failure, and the energy released and amount of noncondensible gas generated as a result of molten core–concrete interactions. In the reactor building, important phenomena includes the energy transfer from gases escaping a ruptured containment and the energy released during combustible gas burns.

Design features of the plant have a direct influence on the conditions that can occur in the reactor coolant system, the containment, and the reactor building and affects instrument availability. The failure pressure of the containment boundary is an important example because it is significantly higher than the containment design pressure, which was used to establish the qualification conditions for the instrumentation. Conditions in the reactor building will be affected by the availability and design of containment vent systems as well as the design and location of blowout panels.

Severe conditions can develop in the containment (drywell and torus) and in the reactor building either before or after core damage has occurred. Accidents where severe conditions can develop before core damage include sequences initiated by an ATWS followed by standby liquid control system failure. Severe containment conditions can also develop during accident sequences initiated by a transient with successful ECCS function but with failure of the containment cooling system. For these sequences, management of the accident could be much more difficult because the performance of instruments located in the containment and reactor building could degrade before any core damage occurs. Conditions in the containment and reactor building could also be severe following core degradation and vessel failure as a result of core concrete interaction or contact of the melted material with the drywell shell.

The principal challenge to instrument availability during severe accidents in a BWR with a Mark I containment is concluded to be severe pressure and temperature environments in the containment and the reactor building. These severe conditions can develop either before or after core damage, depending on the sequence. Radiation is a longer term effect and would become important when instrumentation is required for monitoring that may extend weeks or months beyond the initiating event. Steam condensation in the reactor building could also affect instrument availability.

The ability to monitor reactor coolant system and containment heat removal is essential to long term recovery, particularly if core cooling is reestablished before core meltdown progresses to a noncoolable state. If the performance of the reactor system and containment pressure and temperature monitoring instruments has degraded as a result of high system temperatures or high containment or reactor building pressures and temperatures, it would be difficult to reliably monitor the core heat removal safety function and to accurately determine the vessel water level and plant pressures and temperatures. Even if instrument qualification limits are not exceeded, it is probable that some of the instruments that monitor temperature, for example those located in the suppression pool, will be exposed to temperature conditions above their measurement range resulting in degraded performance.

In the event of a station blackout, instruments which have a battery backup will be operational until power is recovered or until the battery power is depleted. Because of differences in the power source configuration at different plants, it is not possible to generally evaluate instrument performance for a station blackout. It is noted that many plants provide battery backup for all Regulatory Guide 1.97 Category 1 instrumentation although this is not a requirement. If battery backup is provided, then most of the information required to monitor the status of the reactor coolant system and containment will be available until environmental conditions challenge instrument performance. Systems used to obtain and monitor samples of reactor coolant, containment atmosphere and suppression pool water may not be available in the event of a station blackout. As a result, information needs requiring sampling information may not be met.

As discussed in Section 5.2, no direct indication of core temperature is available on a BWR. Previous evaluations have been performed on the need for core temperature measurements and it was concluded that they were not needed. It is recommended that the need for core temperature indication be reevaluated for management of severe accidents. This evaluation should include alternate methods for obtaining core temperature indication in lieu of measurements. The importance of direct core temperature measurements to accident management at a BWR must also be evaluated from a cost–benefit perspective.

Two analysis aids which can potentially provide assistance to personnel in the technical support center in managing severe accidents are

- An analysis aid to project core power level as a function of ECCS flow and other relevant parameters in the event of an ATWS

- An analysis aid to determine plant status from interpretation of results of the sampling and analysis of the contents of the reactor coolant system and containment.

As with core temperature measurements, the importance of these aids to accident management must be evaluated from a cost–benefit perspective.

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Appendix A

Discussion of Peach Bottom Plant Damage States and Accident Progression Bins

Appendix A

Discussion of Peach Bottom Plant Damage States and Accident Progression Bins

A-1. REVIEW OF THE PEACH BOTTOM PLANT DAMAGE STATES

In the NUREG-1150[A-1] analysis, accident sequences for the Peach Bottom plant are divided into four summary plant damage states. They are:

- Station blackout
- Large and small break LOCA
- Anticipated transients without scram (ATWS)
- All other transients except station blackout and ATWS.

Each PDS is defined by a group of accident sequences that has similar characteristics with respect to accident progression and containment engineered safety feature operability. These plant damage states are defined from seven indicators identified in NUREG/CR-4550[A-2] which are (a) the initiating event, (b) status of electric power, (c) status of safety relief valves (i.e. any stuck open valves), (d) status of high pressure injection systems, (e) status of RCS depressurization, (f) status of low pressure injection and decay heat removal systems, and (g) status of containment venting and containment isolation systems.

The four summary plant damage states presented above are developed from nine plant damage states for internally initiated events that are described in NUREG-1150,[A-1] NUREG/CR-4550,[A-2] and NUREG/CR-4551.[A-3] These plant damage states are reviewed so that the types of accidents that affect instrument availability are factored into the evaluation. The plant damage states included in each summary PDS and the types of accident sequences that characterize each PDS are described below.

Station Blackout

The station blackout summary plant damage state includes PDS-4 and PDS-5.

PDS-4 results from a station blackout with failure of dc power. Two sequences are included in PDS-4. One involves a stuck open safety relief valve (SRV). In these sequences, dc power failure has also occurred. Early core damage results from the immediate loss of the HPCI and the reactor core isolation cooling (RCIC) systems, both of which require dc power. Core damage would occur in about 1 hour in this case. Containment venting is not possible because of the loss of ac power.

PDS-5 involves a long-term station blackout and includes three sequences. One involves a stuck open safety relief valve SRV. The HPCI system is initially working since this system is independent of ac power. If the recovery of ac power does not occur, then the following outcomes will result:

- Depletion of the batteries occurs, resulting in injection failure, reclosure of the automatic depressurization system (ADS) valves, and repressurization of the RPV (in those cases where an SRV is not stuck open), followed by boiloff of the primary coolant and core damage. Batteries are expected to be depleted in about 10 hours in this situation.

- Failure of HPCI and RCIC systems occurs due to high suppression pool temperature or high containment pressure. Vessel boiloff and core damage occurs at low RPV pressure. The vessel is assumed to be depressurized since either the automatic depressurization system is functioning as dc power is available or a SRV is assumed to be stuck open. The containment is at high pressure but less than the saturation pressure corresponding to the temperature at which HPCI will fail (i.e., about 40 psig at the start of core damage).

Core damage results in about 13 hours as a result of coolant boiloff in either case. Containment venting is not possible because of the loss of ac power.

Loss-of-Coolant Accidents (LOCA)

The LOCA summary plant damage state includes large and medium break LOCA sequences in PDS-1. PDS-1 consists of two accident sequences:

1. A large LOCA followed by immediate failure of all high and low pressure injection systems.

2. A medium LOCA with initial HPCI success but almost immediate failure as the vessel depressurizes below HPCI working pressure. The low pressure injection systems are assumed to have failed.

For either sequence, early core damage occurs approximately 1 to 2 hours following the initiating event. The control rod drive and containment heat removal systems are functioning. Containment venting is available but is not needed.

Anticipated Transient Without Scram (ATWS)

The ATWS summary plant damage state includes PDS-6, PDS-7, PDS-8, and PDS-9. Note that containment venting occurs in PDS-7, PDS-8, and PDS-9.

PDS-6 is an ATWS where the SLCS functions. The HPCI system also functions and initially provides core cooling. However, high suppression pool temperatures causes HPCI failure, resulting in early core damage. Containment venting is available but is not done before core damage occurs.

PDS-7 is an ATWS involving the failure of the SLCS due to a stuck open relief valve. Otherwise, it is the same as PDS-8 described below.

PDS-8 is an ATWS sequence with loss of either the ac bus or the Power Conversion System (PCS) followed by failure to scram. The HPCI system fails due to high suppression pool temperature. There are two possible outcomes, which are:

1. The operator does not manually depressurize the reactor.

2. The operator depressurizes the reactor and uses the low pressure coolant injection (LPCI) system until the injection valves fail due to excessive cycling, containment failure occurs, or the containment is vented. Containment failure or venting results in failure of the LPCI system due to severe environments in the reactor building.

Early core damage ensues in about 15 minutes after initiation of the event in cases where the operator does not manually depressurize the reactor. The time to core damage ranges from 20 minutes to several hours in cases where failure of the LPCI system occurs. The time to core damage depends on the LPCI failure mode. It is noted that containment will be vented before core damage occurs.

PDS-9 is an ATWS with failure of the SLCS initiated by a loss of offsite power. However, onsite ac power sources are available. Otherwise, PDS-9 is the same as PDS-8.

Transients

The transient summary plant damage state includes PDS-2 and PDS-3.

PDS-2 consists of four transient initiated sequences. Two SRVs are stuck open in each sequence (the equivalent of an intermediate LOCA). The HPCI system functions initially, but fails when the vessel depressurizes below the HPCI working pressure. All other injection systems have failed and early core damage results. The control rod driveline (CRD) and containment heat removal systems are working as in PDS-1 but steam is directed through the SRVs to the suppression pool not to the drywell as in PDS-1. Venting is available but is not done before core damage occurs.

PDS-3 is similar to PDS-2 except that the containment heat removal is not working. The control rod drive system is also not functioning for some of the sequences included in this PDS.

A-2. REVIEW OF ACCIDENT PROGRESSION BINS

Accident progression bins were developed by considering the possible outcome of the various plant damage states. Accident progression bins were developed from the quantification of the accident progression event tree. The development of this tree requires the answers to key question regarding accident progression. Section 2.3 of NUREG-CR-4551[A-3] presents a list of 145 questions that were considered for the Peach Bottom accident progression analysis. Although the list of questions is intended for use in determining the probability of different events and system availability for the plant damage states, the questions are germane to accident management because the technical support center and operations personnel will be asking the same types of questions in a severe accident situation.

A set of bins was developed to categorize the accident progression outcome of each plant damage state based on the answers to questions on accident progression from NUREG/CR-4551.[A-3] These bins are summarized below:

- Vessel breach at a pressure >200 psia with early containment failure in the wetwell
- Vessel breach at a pressure <200 psia with early containment failure in the wetwell
- Vessel breach at a pressure >200 psia with early containment failure in the drywell
- Vessel breach at a pressure <200 psia with early containment failure in the drywell
- Vessel breach with late containment failure in the wetwell
- Vessel breach with late containment failure in the drywell
- Vessel breach with containment venting
- Vessel breach with no containment failure
- No vessel breach
- No core damage.

Early containment failure refers to containment failure before or slightly after vessel breach (lower head failure). Each of the PDSs can follow an accident progression represented by one of these bins. The possibility of arresting the core damage process before core slump and collapse occurs is also considered in the NUREG-1150[A-1] analysis.

There is a relationship between the NUREG-1150[A-1] PDSs and accident progression bins and the safety objective trees presented in Figures 1, 2, and 3 of the main document. The PDS will determine which of the challenges and mechanisms are important from among the (a) loss of flow path, (b) scram failure, (c) recriticality, (d) inadequate inventory, (e) and flow blockage challenges from the Prevent Core Dispersal From Vessel safety objective tree. The accident progression will determine which challenges and mechanisms are important from the Vessel Overtemperature and Overpressure challenges and the challenges presented on the Prevent Containment Failure and Mitigate Fission Product Release From Containment safety objective trees (Figures 2 and 3).

A-3. REFERENCES

A-1. U.S. Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five Nuclear Power Plants*, NUREG-1150, Volumes 1 and 2, December 1990.

A-2. U.S. Nuclear Regulatory Commission, *Analysis of Core Damage Frequency: Peach Bottom, Unit 2 Internal Events*, NUREG/CR-4550, Volume 4, Revision 1, Part 1, August 1989.

A-3. U.S. Nuclear Regulatory Commission, *Evaluation of Severe Accident Risks: Peach Bottom, Unit 2*, NUREG/CR-4551, December 1990.

Appendix B

Discussion of Peach Bottom Thermal Hydraulic Results

Appendix B

Discussion of Peach Bottom Thermal Hydraulic Results

Appendix B presents the results from BMI-2104[B-1] and NUREG/CR-4624[B-2] accident sequences that are representative of the NUREG-1150[B-3] PDSs and accident progression bins discussed in Appendix A for the Peach Bottom plant. Table B-1 presents a matrix showing the relationship of the BMI-2104[B-1] and NUREG/CR-4624[B-2] accident sequences analyzed to these PDSs and accident progression bins for Peach Bottom. Note that the wetwell failure and drywell failure accident progression bins are combined since containment failure location will have little impact on instrument availability.

Data from the BMI-2104[B-1] and NUREG/CR-4624[B-2] analyses for core average temperature, reactor system pressure, reactor system gas and structure temperature, and containment pressure and temperature are presented graphically in the figures in this appendix. Severe accident phases are identified on each plot and are defined as follows:

- Phase 1 - from accident initiation to start of core uncovery
- Phase 2 - from start of core uncovery to start of core melt
- Phase 3 - from start of core melt to core slump
- Phase 4 - from core slump to lower head failure
- Phase 5 - from lower head failure to end of analysis.

B-1. PEACH BOTTOM THERMAL HYDRAULIC RESULTS

The severe accident sequences from BMI-2104[B-1] and NUREG/CR-4624[B-2] that were selected for the PDS and accident progression bin matrix are presented in Table B-1. These sequences are discussed in the following sections.

B-1.1 Station Blackout

The TB1 and TB2 sequences from NUREG/CR-4624[B-2] are used to represent the station blackout sequence described in NUREG-1150. The TB1 and TB2 sequences are characterized by

Table B-1. Assignment of the BMI-2104 and NUREG/CR-4624 results to the NUREG-1150 plant damage states/accident progression bins.

Accident progression bin	Station blackout	LOCA	ATWS	Transients
VB >200 psia, early WWF or DWF	TB2	—[a]	TC2	—[a]
VB <200 psia, early WWF or DWF	—[a]	AE-γ	TC1	—[a]
VB, late WWF or DWF	TB1	—[a]	—[a]	TW-γ
VB, CV	—[a]	—[a]	TC3	—[a]
No CF	—[a]	—[a]	—[a]	—[a]
No VB	—[a]	—[a]	—[a]	—[a]
No core damage	—[a]	—[a]	—[a]	—[a]

VB - Vessel Breach
WWF - Wetwell Failure
DWF - Drywell Failure
CV - Containment Venting
CF - Containment Failure

a. No analysis was found in NUREG/CR-4624 or BMI-2104 that corresponds to this plant damage state or accident progression bin. See Section B-2.

a loss of all onsite and offsite ac power resulting in a loss of core cooling except for HPCI and RCIC, since battery power is assumed to be available. In NUREG/CR-4624, the batteries were estimated to be depleted six hours after accident initiation as compared to 10 hours in NUREG-1150[B-3] causing loss of HPCI and RCIC and resulting in core melt for both TB1 and TB2. In the TB1 case, the containment is assumed to fail late in the accident sequence due to accumulation of noncondensible gases as a result of concrete decomposition. In the case of TB2, it is assumed that containment failure occurs due to rapid pressurization following reactor vessel failure. The failure occurs in the drywell for both the TB1 and TB2 sequences.

Tables B-2 and B-3 present the timing of key events for the TB1 and TB2 sequences, respectively. The timing of key events between both sequences is essentially the same up to the point of bottom head failure, which occurs at about 730 minutes. Containment failure is assumed to occur at about this time for the TB2 analysis.

Table B-2. Key accident event times for Peach Bottom TB1 sequence.

Event	Time (min)
ECC off	360.1
Core uncovery	528.5
Core melt starts	642.4
Core slump occurs	694.8
Core collapse occurs	695.3
Lower head dryout	704.8
Lower head failure	733.5
Start of concrete attack	733.5
Containment failure	914.5
Hydrogen burn	914.9
Hydrogen burn	915.3
Hydrogen burn	919.1
Corium layers invert	928.0
End calculation	1333.5

Note: Data from Table 4.1 of NUREG/CR-4624, Volume 1.

Table B-3. Key accident event times for Peach Bottom TB2 sequence.

Event	Time (min)
ECC off	360.1
Core uncovery	526.9
Core melt starts	615.5
Core slump occurs	693.7
Core collapse occurs	694.3
Lower head dryout	705.1
Lower head failure	735.8
Containment failure occurs	735.8
Start of concrete attack	735.9
Hydrogen burn	736.9
Hydrogen burn	739.8
Hydrogen burn	934.4
Corium layers invert	934.9
Hydrogen burn	935.0
Hydrogen burn	935.4
Hydrogen burn	946.4
Hydrogen burn	946.9
Hydrogen burn	963.9
Hydrogen burn	996.5
Hydrogen burn	1058.7
Hydrogen burn	1105.9
Hydrogen burn	1138.6
End calculation	1333.5

Note: Data from Table 4.1 of NUREG/CR-4624, Volume 1

Figure B-1 presents the pressure and temperature conditions in the reactor coolant system up to vessel failure for the TB1 sequence. Figure B-2 presents the pressure and temperature conditions in the drywell for the TB1 sequence. This figure shows a relatively slow pressure increase to about 30 psia just before vessel failure. A pressure spike to about 110 - 120 psia occurs due to the release of steam from the reactor system upon vessel failure. The pressure remains at a relatively high 80 psia after the pressure spike since heat removal from the suppression pool is lost due to the station blackout. Containment failure occurs at about 914 minutes due to the high steam pressure and the addition of noncondensible gases due to core concrete interaction. The suppression pool and wetwell air space temperatures for the TB1 sequence is presented on Figure B-3. The temperature conditions in the reactor building outside of the primary containment are presented in Figure B-4. The temperature spike at 915 minutes is caused by a hydrogen burn. Pressure spikes of about 18 psia are predicted in the reactor building at the time of the hydrogen burn.

Figure B-5 presents the pressure and temperature conditions in the reactor coolant system up to vessel failure for the TB2 sequence. Figure B-6 presents the pressure and temperature conditions in the drywell for the TB2 sequence. The suppression pool temperature for the TB2 sequence is presented on Figure B-7. The wetwell air space temperature is the same as the pool temperature for this sequence. The temperature conditions in the reactor building outside of the primary containment are presented in Figure B-8. Containment failure occurs at the time of lower head failure followed by a number of hydrogen burns in the reactor building or in the refueling bay resulting in the pressure and temperature spikes shown in Figure B-8. Pressure spikes of about 24 psia are predicted in the reactor building during hydrogen burns. Temperature data for the dryers and separators and other structures within the reactor vessel were not presented in NUREG/CR-4624 for either the TB1 or TB2 sequence.

B-1.2 Loss-of-Coolant Accidents

The accident sequence selected to represent the LOCA sequences for this evaluation is the AE-γ sequence from BMI-2104.[B-1] The AE-γ sequence is the only LOCA sequence analyzed in BMI-2104 and NUREG/CR-4624.[B-2]

The AE-γ sequence is characterized by a large break (equivalent diameter >6 inches) in a recirculation line. All ECCS are assumed to fail. The suppression pool remains subcooled throughout the accident due to continued operation of the Residual Heat Removal (RHR) system operating in the suppression pool cooling mode. The containment failure scenario for the AE sequence involves an early failure due to overpressurization from generation of noncondensible gases produced from steam-cladding reactions and core-concrete interaction.

The timing of key events for AE-γ is presented in Table B-4. The reactor coolant system pressure and core average temperature is presented in Figure B-9. The pressure and temperature conditions in the drywell during this accident sequence is presented on Figure B-10. The suppression

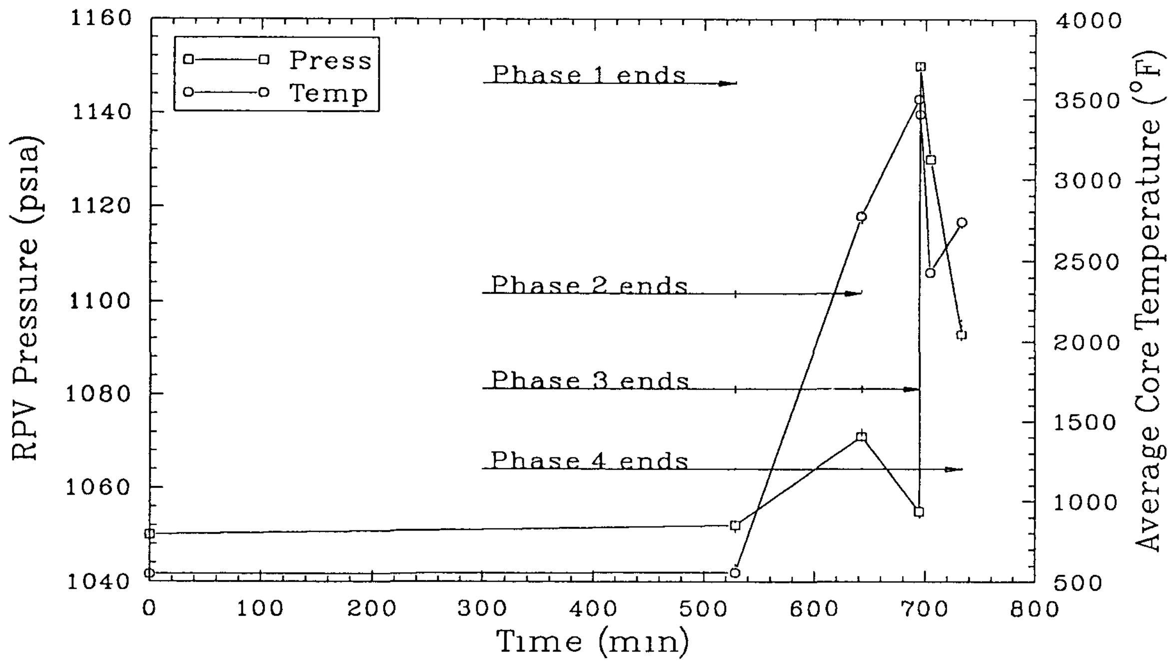

Figure B-1. Peach Bottom TB1 reactor system data.

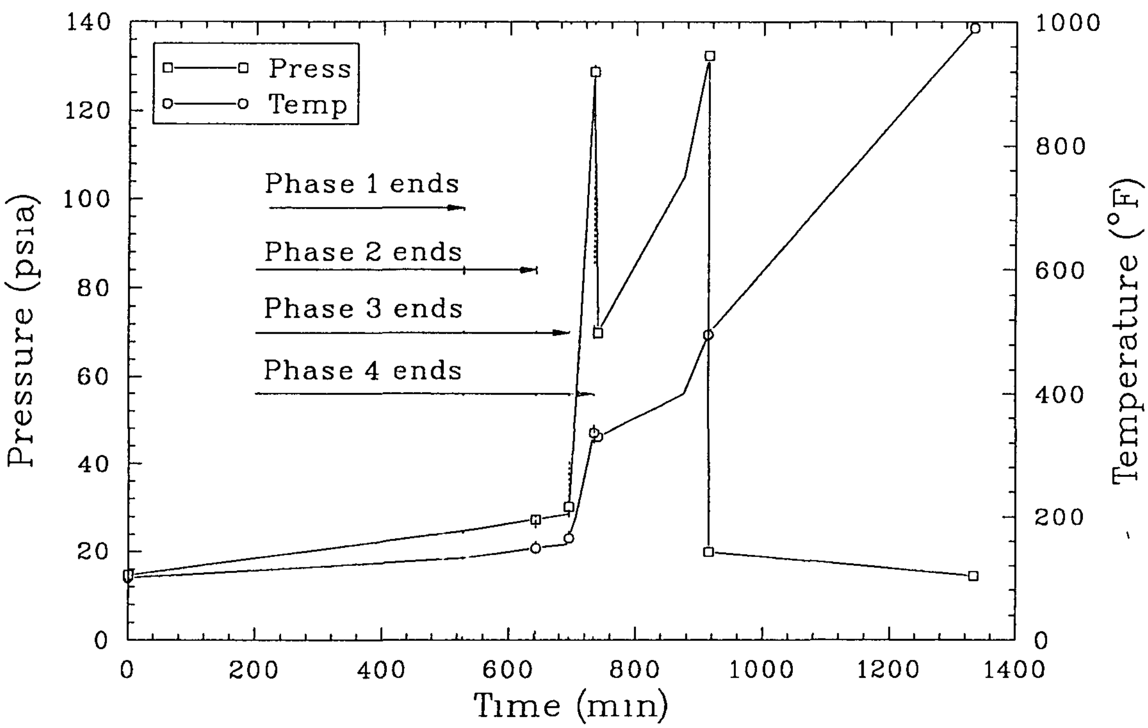

Figure B-2. Peach Bottom TB1 drywell data.

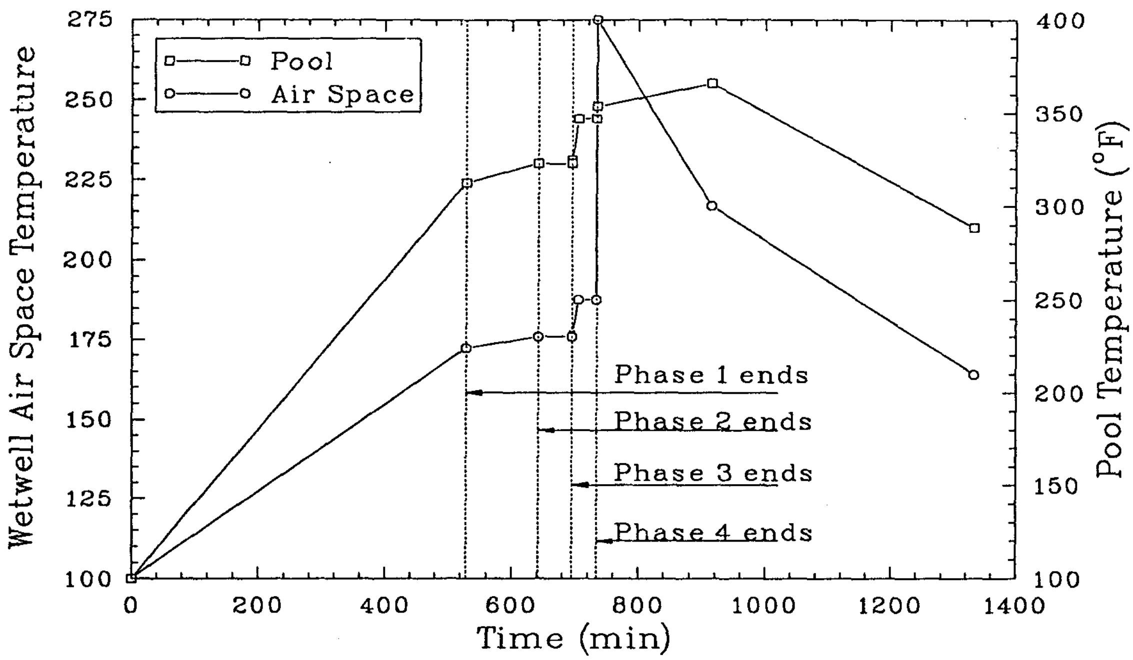

Figure B-3. Peach Bottom TB1 wetwell temperature data.

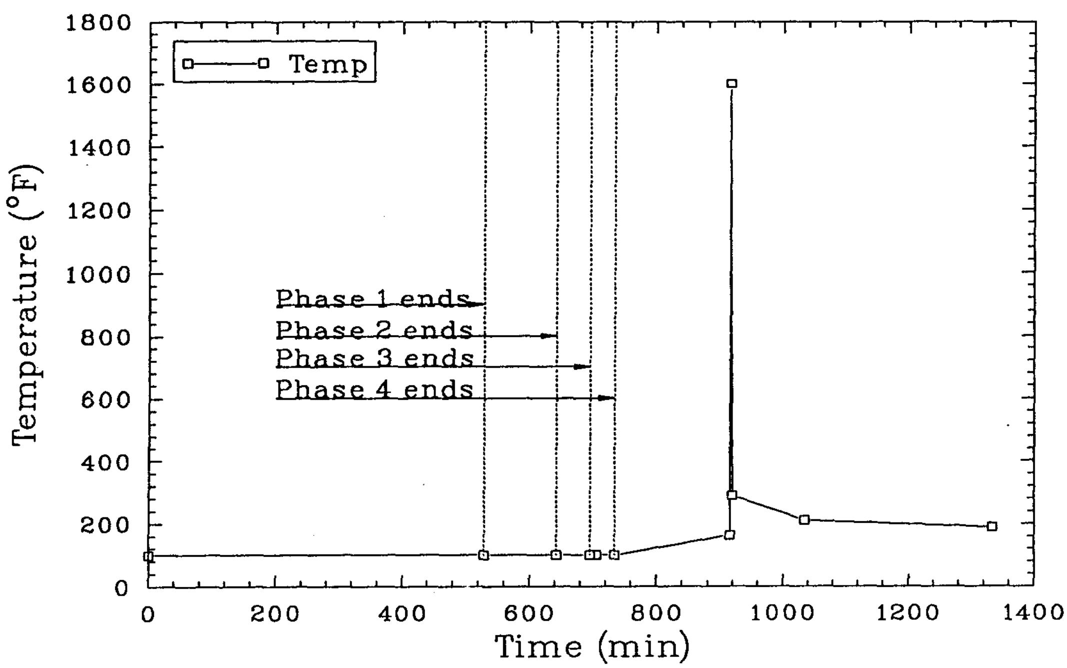

Figure B-4. Peach Bottom TB1 reactor building temperature.

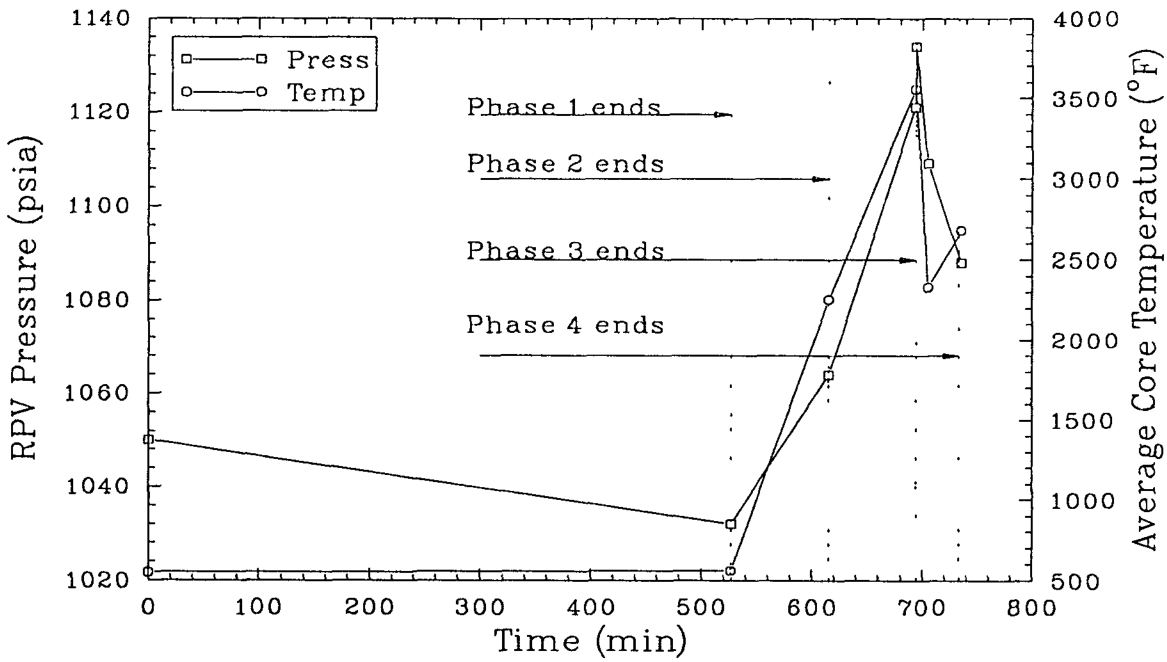

Figure B-5. Peach Bottom TB2 reactor system data.

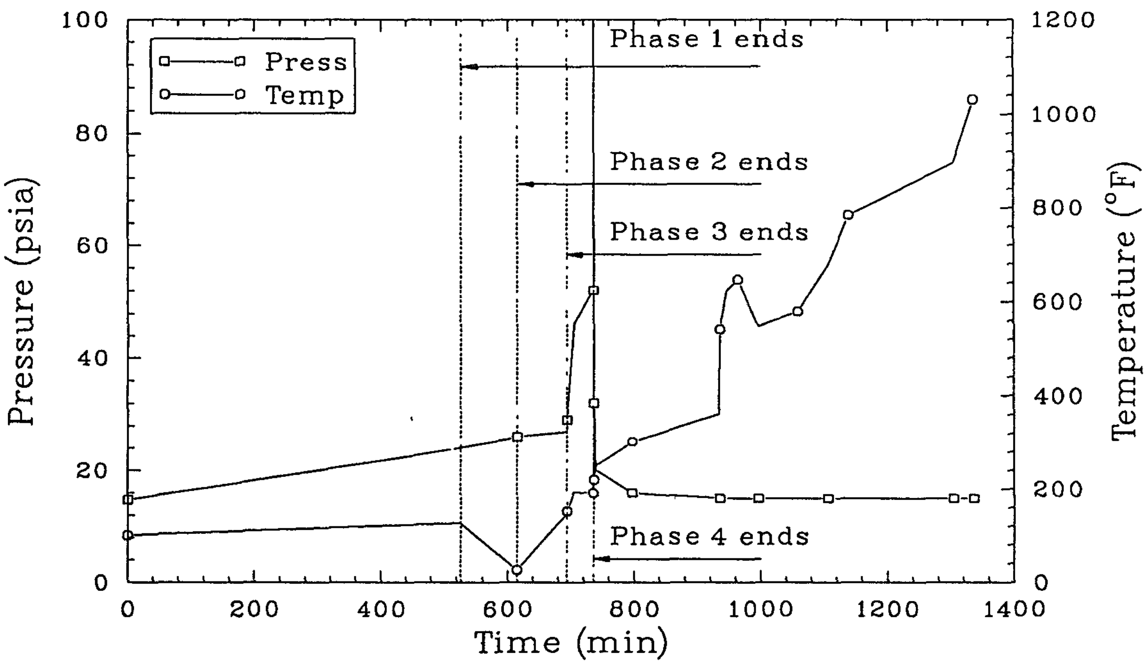

Figure B-6. Peach Bottom TB2 drywell data.

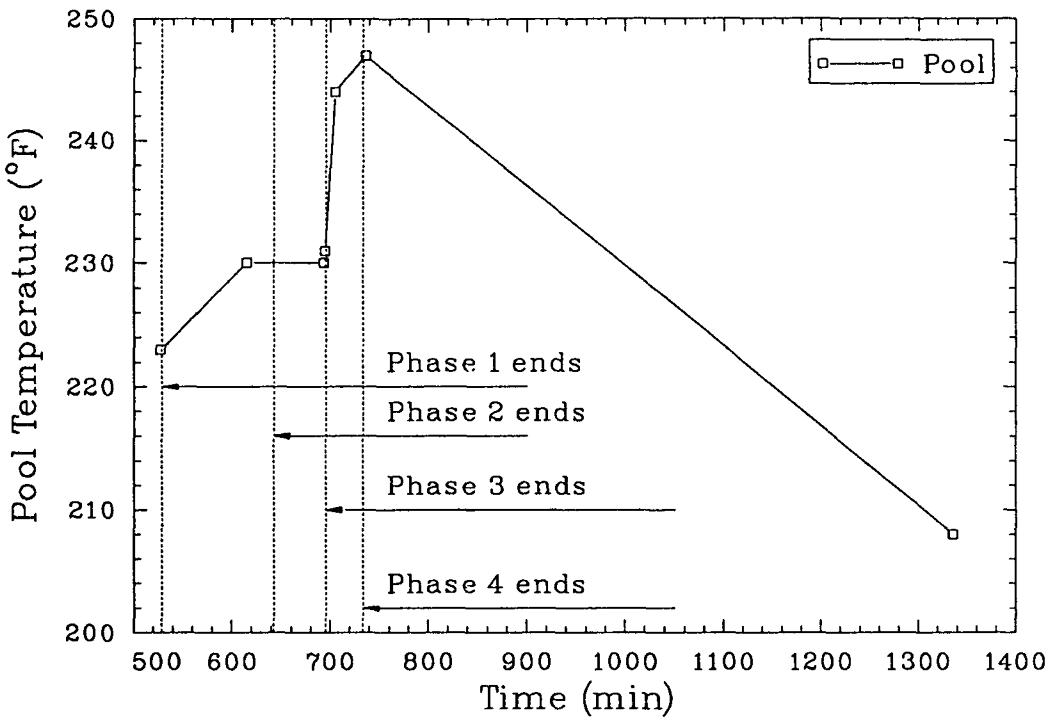

Figure B-7. Peach Bottom TB2 wetwell temperature.

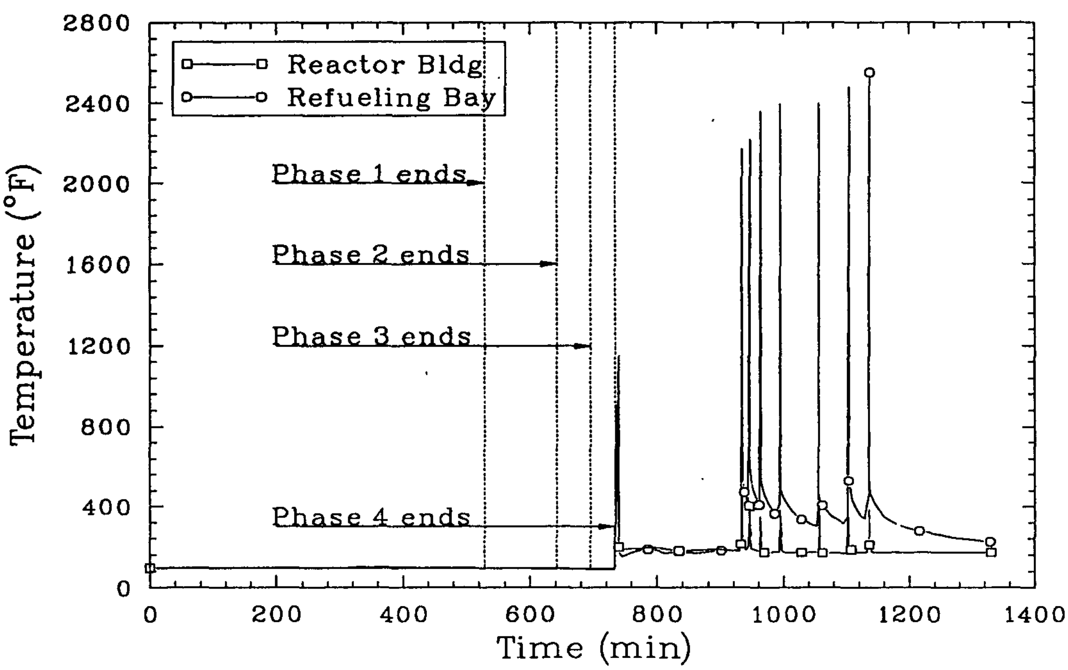

Figure B-8. Peach Bottom TB2 reactor building temperature data.

Table B-4. Key accident event times for Peach Bottom AE sequence.

Event	Time (min)
Core uncovery	1.5
Suppression pool cooling on	10.0
Core melt starts	11.5
Core slump occurs	26.8
Containment failure occurs	33.9
Lower head dryout occurs	40.0
Core collapse occurs	65.2
Lower head failure	126.2
Start of concrete attack	126.3
End calculation	727.0

Note: Data from Table 6.2 of BMI-2104, Volume 2.

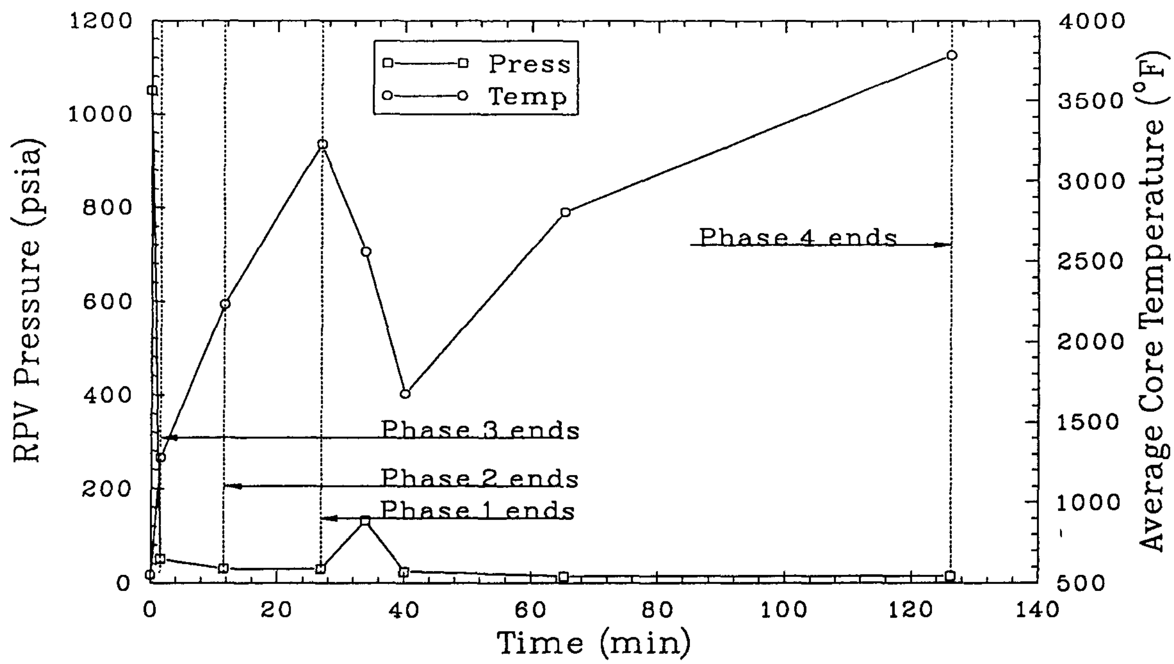

Figure B-9. Peach Bottom AE-γ reactor system data.

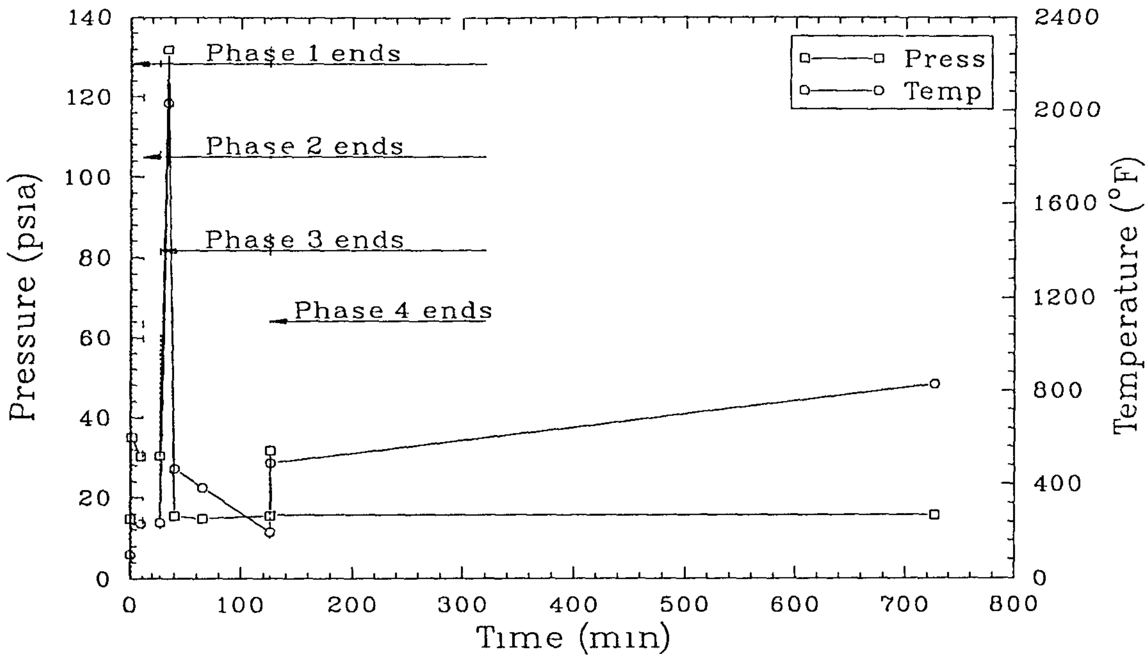

Figure B-10. Peach Bottom AE-γ drywell data.

pool and wetwell air space temperatures are presented in Figure B-11. A rapid containment pressure increase from about 27 to 34 minutes resulting in containment failure is attributed to the production of hydrogen and the transport of the noncondensibles into the wetwell in BMI-2104. Containment failure is predicted to occur before lower head failure in this sequence. BMI-2104 notes that the prediction of the occurrence of containment failure at this point in the accident sequence is sensitive to the core slumping scenario used in the analysis. Lower head failure (end of phase 4) is predicted to occur at about 126 minutes after accident initiation. Since suppression pool cooling is maintained throughout the accident, the pool water temperature remains constant and the wetwell airspace temperature remains relatively low. No data for the temperature in the reactor building outside the primary containment is given in BMI-2104.

Figure B-12 presents the gas temperature at the core exit and the lower annulus and the temperatures of the separator and lower annulus structures. The reactor system flowpath considered in the MARCH/MERGE analysis is from the core through the separators, outer annulus, jet pumps and out through the assumed recirculation line break to containment.

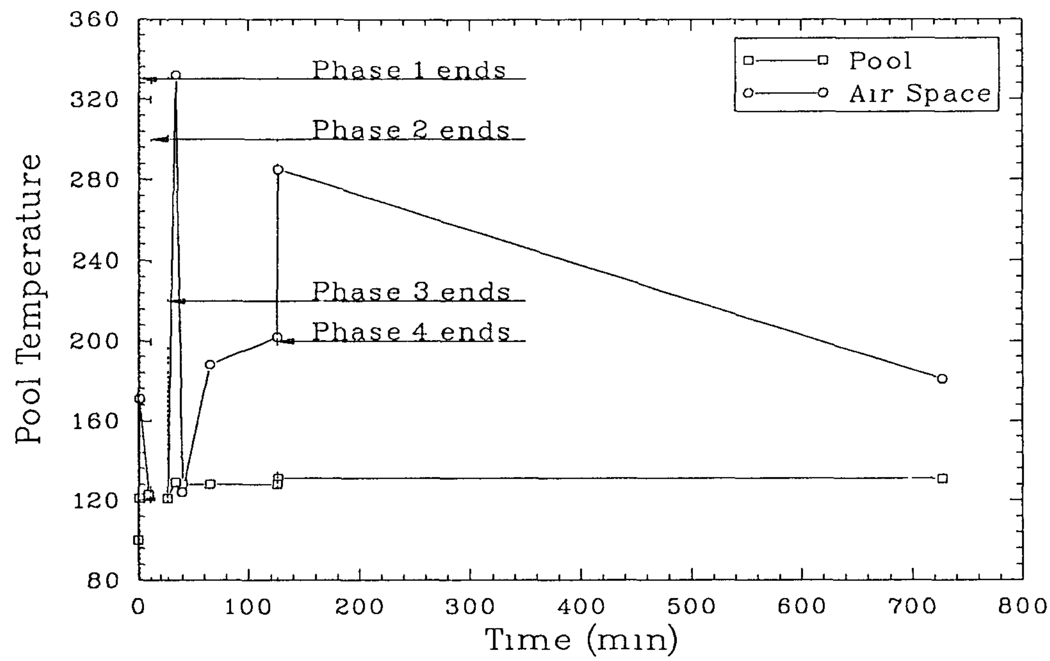

Figure B-11. Peach Bottom AE-γ wetwell temperature.

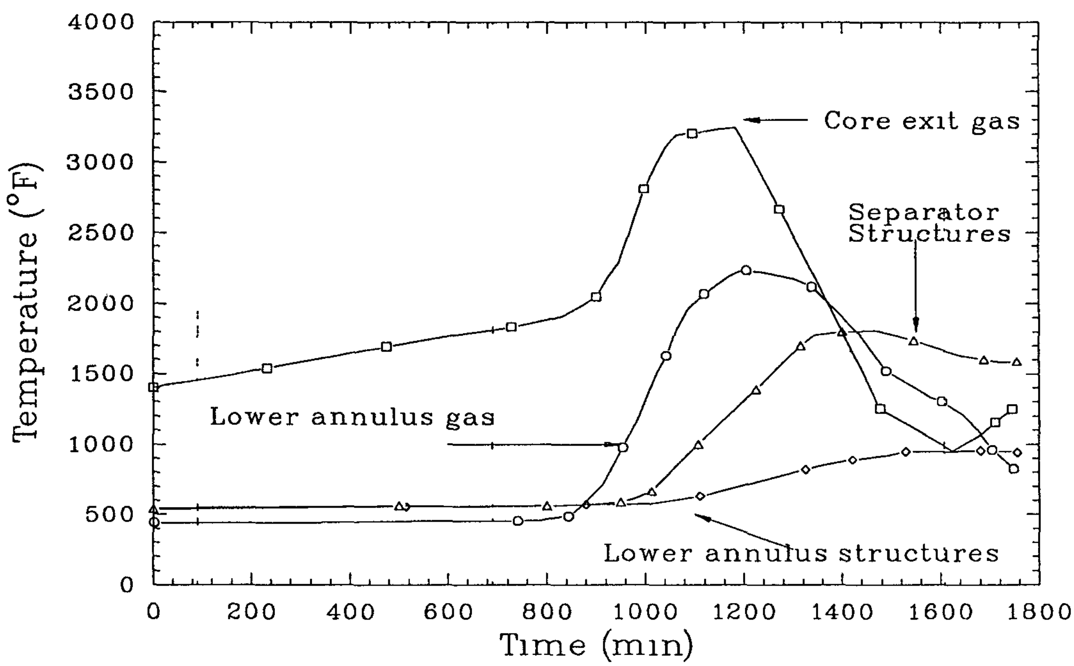

Figure B-12. Peach Bottom AE-γ gas and structure temperatures.

B-1.3 Anticipated Transient Without Scram Sequences

The result of three ATWS sequences presented in NUREG/CR-4624[B-2] are used for this evaluation. These ATWS sequences are denoted as TC1, TC2 and TC3. In addition, the reactor primary system gas and structure temperature results from the TC-γ sequence in BMI-2104[B-1] is also used for this evaluation.

The TC1 sequence is initiated by a transient with a failure to scram. The main steam isolation valves close. The operators are not successful in initiating early power reduction but are successful in depressurizing the primary system. Suppression pool heatup and containment pressurization results from continued reactor operation at an elevated power level greater than the capacity of the RHR system operating in the suppression pool cooling mode. Containment failure occurs as a result of the pool heatup and containment pressurization. Upon containment failure, flashing of the saturated suppression pool is assumed to lead to failure of the emergency core cooling pumps due to cavitation, causing core melt. The primary system is assumed to remain depressurized throughout the sequence although NUREG/CR-4624[B-2] alludes to the possibility that control air pressure may not be sufficient to keep the safety/relief valves open as the containment pressurizes.

Table B-5 presents the timing of key events for the TC1 sequence. Containment failure is predicted to occur at about 85 minutes, which is about 50 minutes before core melt is predicted to start.

Figure B-13 presents the pressure and temperature conditions in the reactor coolant system up to vessel failure for the TC1 sequence. Figure B-14 presents the pressure and temperature conditions in the drywell for the TC1 sequence. This figure shows that containment failure is predicted to occur at 85.3 minutes, before predicted core uncovery. Containment failure occurs because the rate of heat rejection to the suppression pool is greater than the RHR system operating in the suppression pool cooling mode. The suppression pool temperature for the TC1 sequence is presented in Figure B-15. The pool temperature remains at saturation conditions throughout the sequence once pool boiling occurs. The temperature conditions in the reactor building outside of the primary containment are presented in Figure B-16. Several hydrogen burns are predicted to occur in the reactor building and the refueling bay starting at about 380 minutes after accident initiation as shown on Figure B-16. Otherwise, the average temperature in the reactor building is predicted to be 240°F. Pressure spikes of approximately 22 psia are predicted in the reactor building during the hydrogen burns.

The TC2 sequence is initiated by a transient with a failure to scram. The main steam isolation valves close. The operators are not successful in initiating early power reduction or in depressurizing the primary system. Primary coolant inventory is maintained by the combination of the HPCI, RCIC and the CRD systems. Suppression pool heatup and containment pressurization results from continued reactor operation at a power level greater than the capacity of the RHR system operating in the suppression pool cooling mode. The HPCI is assumed to fail when the pool reaches 200°F due

Table B-5. Key accident event times for Peach Bottom TC1 sequence.

Event	Time (min)
Containment heat removal on	10.0
Containment failure	85.3
ECC off	86.7
Core uncovery	93.8
Core melt starts	134.0
Core slump occurs	166.8
Core collapse occurs	172.0
Lower head dryout	201.9
Lower head failure	230.5
Start of concrete attack	230.5
Hydrogen burn	383.3
Hydrogen burn	388.0
Hydrogen burn	388.5
Hydrogen burn	399.6
Hydrogen burn	400.3
Corium layers invert	400.5
Hydrogen burn	400.7
Hydrogen burn	421.7
Hydrogen burn	422.2
Hydrogen burn	446.0
Hydrogen burn	469.3
Hydrogen burn	510.3
End calculation	1333.5

Note: Data from Table 4.1 of NUREG/CR-4624, Volume 1.

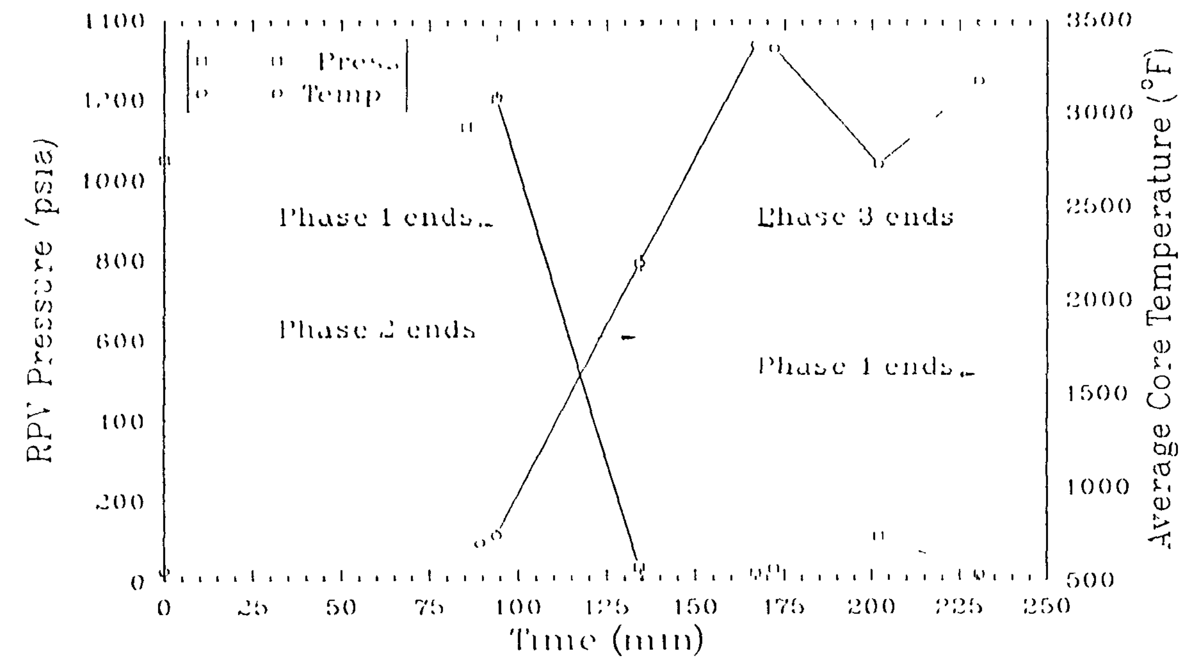

Figure B-13. Peach Bottom TC1 reactor system data.

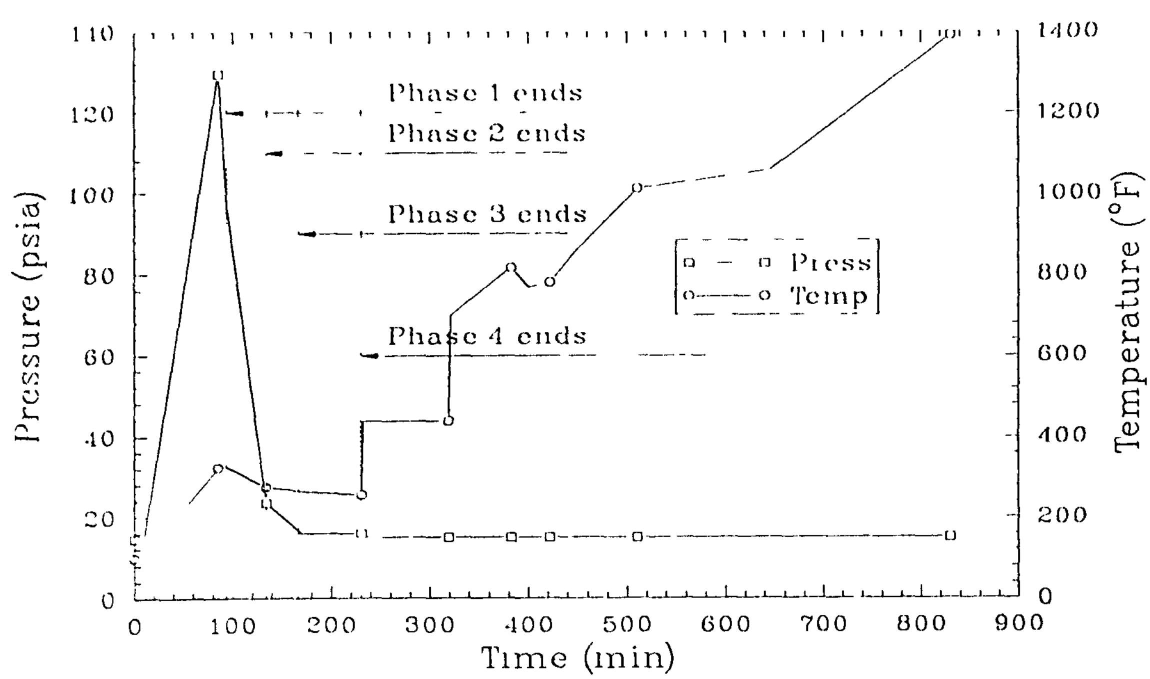

Figure B-14. Peach Bottom TC1 drywell data.

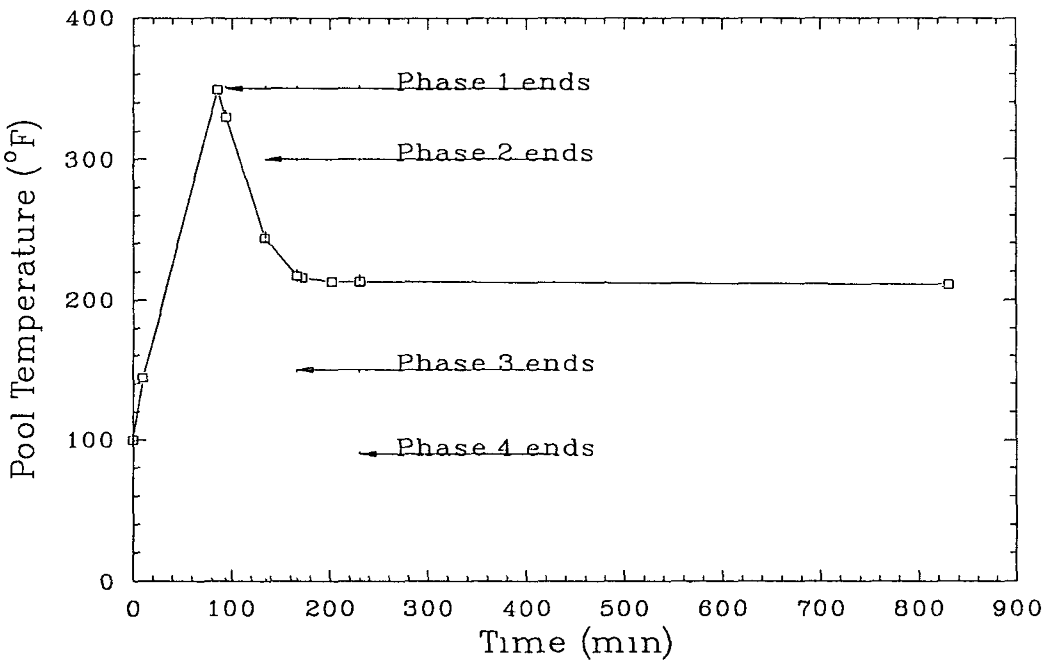

Figure B-15. Peach Bottom TC1 suppression pool temperature.

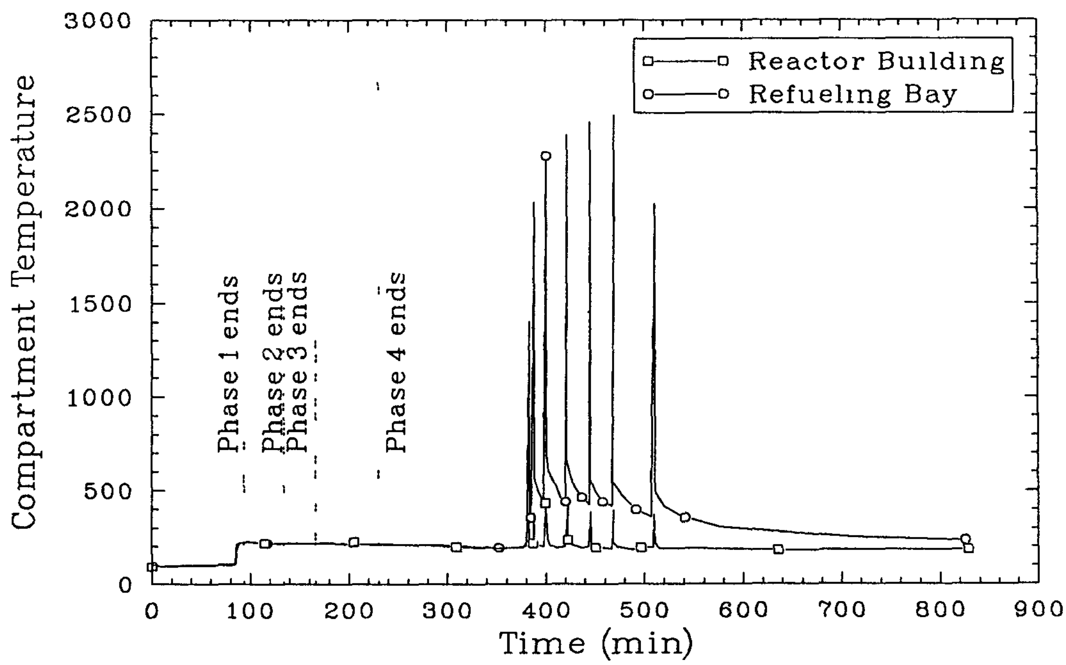

Figure B-16. Peach Bottom TC1 reactor building.

to a mechanical failure (loss of lubrication oil cooling or seal overheating). The RCIC is assumed to fail at a containment of 25 psia due to high turbine exhaust back pressure. Flow from the CRD system is insufficient to maintain core cooling under ATWS conditions. In the TC2 scenario, the containment remains intact during core meltdown with failure occurring at about the time of lower head failure.

Table B-6 presents the timing of key events for the TC2 sequence. Containment failure is predicted to occur at the time of bottom head failure. It is approximately 126 minutes after accident initiation.

Figure B-17 presents the pressure and temperature conditions in the reactor coolant system up to vessel failure for the TC2 sequence. Figure B-18 presents the pressure and temperature conditions in the drywell for the TC2 sequence. The suppression pool temperature for the TC2 sequence is presented in Figure B-19. The pool temperature remains at saturation conditions throughout the sequence once pool boiling occurs. The temperature conditions in the reactor building outside the primary containment are presented in Figure B-20. Hydrogen burns are predicted to occur in the reactor building and the refueling bay at about 126 minutes, which is the time of lower head failure. A hydrogen burn is the cause of the temperature spike shown in Figure B-20. Otherwise, the average temperature in the reactor building is predicted to be about 200°F. Pressure spikes of about 23 psia are predicted in the reactor building during the hydrogen burns.

The TC3 sequence was analyzed to investigate the effects of containment venting. The initiating events and primary system response for the TC3 scenario is similar to the TC2 scenario. To simulate containment venting, it was assumed that an 18 in. diameter vent in the wetwell air space would be opened when the containment pressure reached 10% above the design level. This pressure corresponds to about 77 psia given a design pressure of 71 psia for Peach Bottom. This pressure is lower than the 115 psia pressure at which venting is assumed to take place in the NUREG-1150[B-3] analysis.

Table B-7 presents the timing of key events for the TC3 sequence. Containment venting is initiated at about 96 minutes after accident initiation.

Figure B-21 presents the pressure and temperature conditions in the reactor coolant system up to vessel failure for the TC3 sequence. Figure B-22 presents the pressure and temperature conditions in the drywell for the TC3 sequence. The suppression pool temperature for the TC3 sequence is presented on Figure B-23. The pool temperature remains at saturation conditions throughout the sequence once pool boiling occurs. The temperature conditions in the reactor building outside of the primary containment are presented in Figure B-24. Hydrogen burns are predicted to occur in the reactor building and the refueling bay at about the time of containment venting as shown in Figure B-24. Otherwise, the average temperature in the reactor building is

Table B-6. Key accident event times for Peach Bottom TC2 sequence.

Event	Time (min)
Containment heat removal on	10.0
HPCI fails	25.2
RCIC fails	32.6
Core uncovery	33.8
Core melt starts	58.3
Core slump occurs	86.2
Core collapse occurs	88.3
Lower head dryout	114.9
Lower head failure	126.3
Containment failure	126.3
Hydrogen burn	126.9
Hydrogen burn	128.8
Start of concrete attack	136.3
Corium layers invert	282.8
End calculation	736.3

Note: Data from Table 4.1 of NUREG/CR-4624, Volume 1.

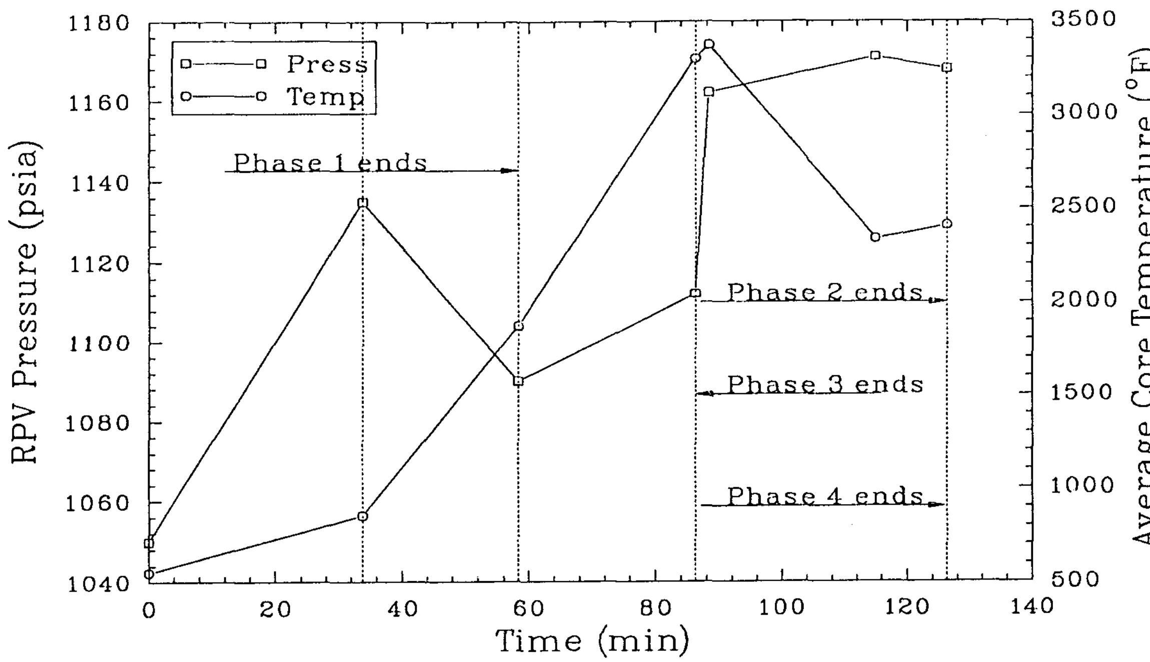

Figure B-17. Peach Bottom TC2 reactor system data.

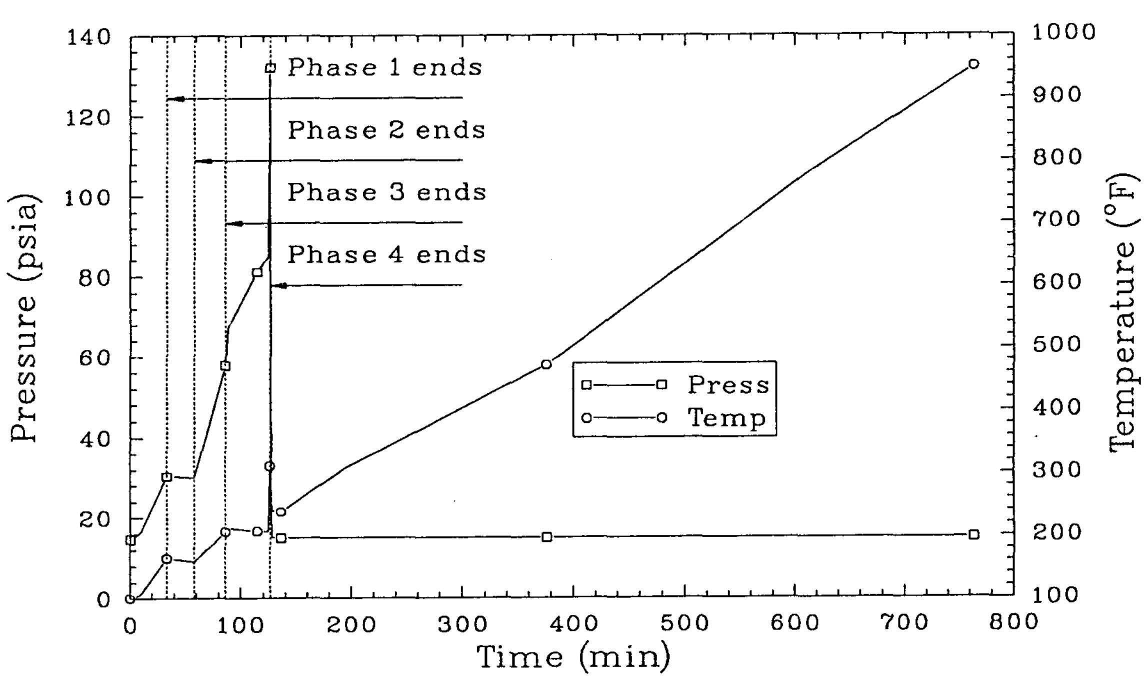

Figure B-18. Peach Bottom TC2 drywell data.

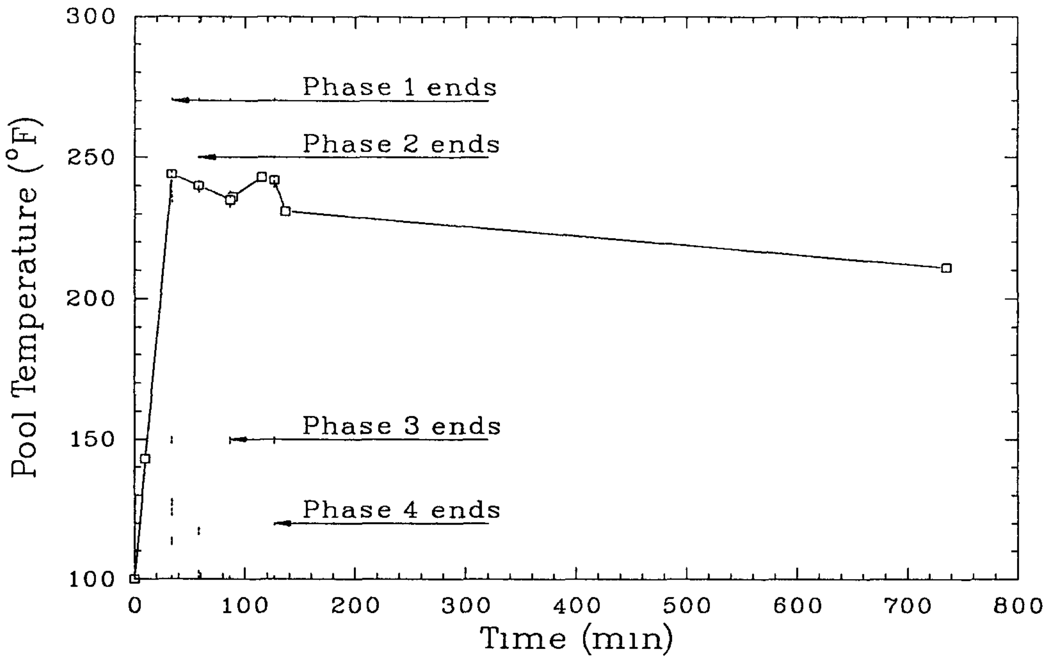

Figure B-19. Peach Bottom TC2 suppression pool temperature.

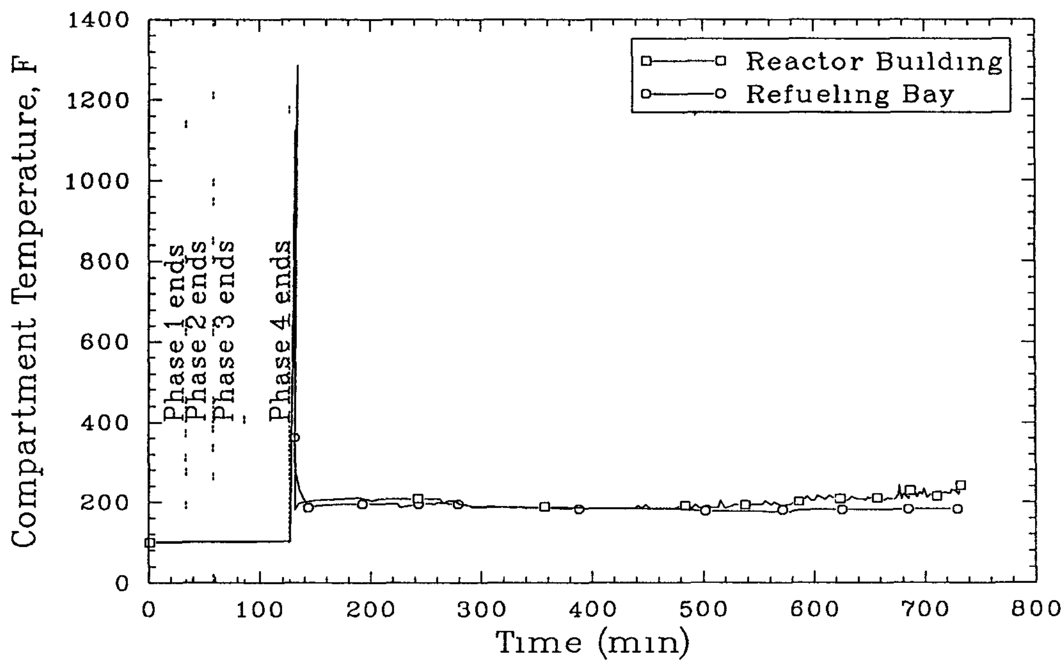

Figure B-20. Peach Bottom TC2 reactor building data.

Table B-7. Key accident event times for Peach Bottom TC3 sequence.

Event	Time (min)
Containment heat removal on	10.0
HPCI fails	25.2
RCIC fails	32.6
Core uncovery	33.8
Core melt starts	58.3
Core slump occurs	86.2
Core collapse occurs	88.3
Containment vent	96.3
Hydrogen burn	97.0
Hydrogen burn	97.8
Hydrogen burn	99.4
Hydrogen burn	104.2
Lower head dryout	118.1
Lower head failure	127.0
Start of concrete attack	136.8
Corium layers invert	282.8
End calculation	736.3

Note: Data from Table 4.1 of NUREG/CR-4624, Volume 1.

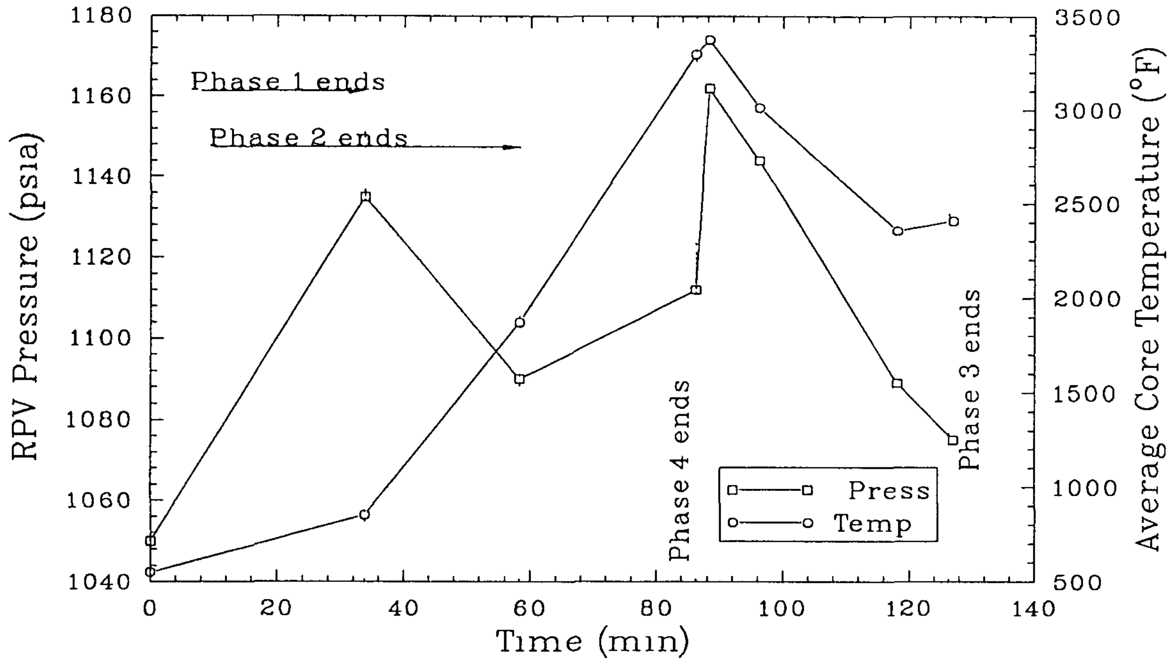

Figure B-21. Peach Bottom TC3 reactor system data.

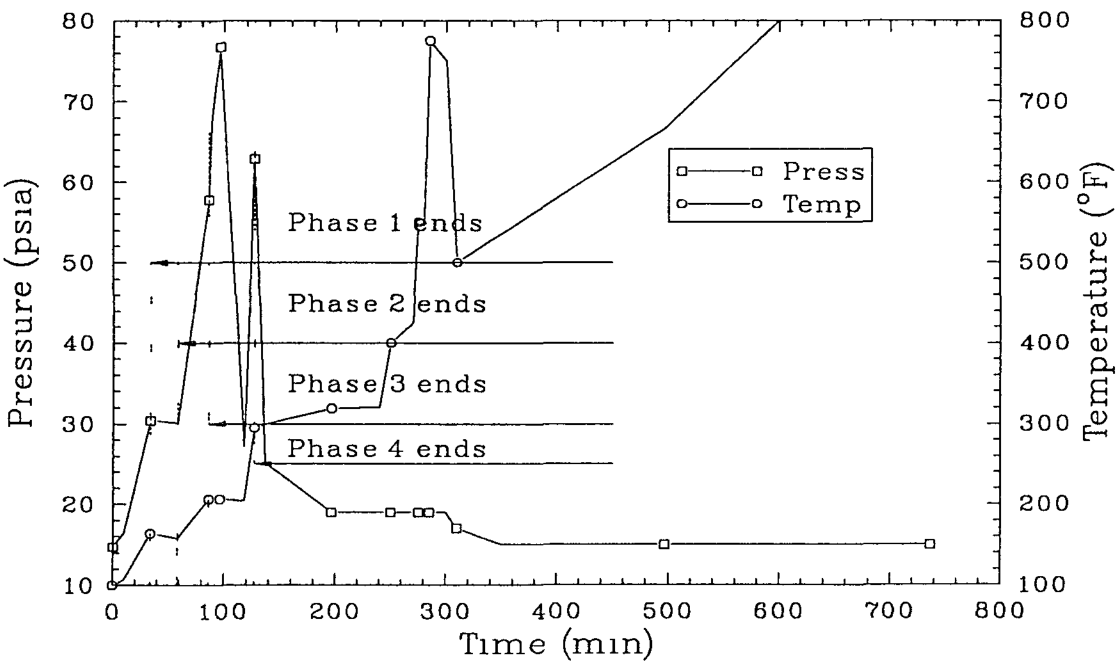

Figure B-22. Peach Bottom TC3 drywell data.

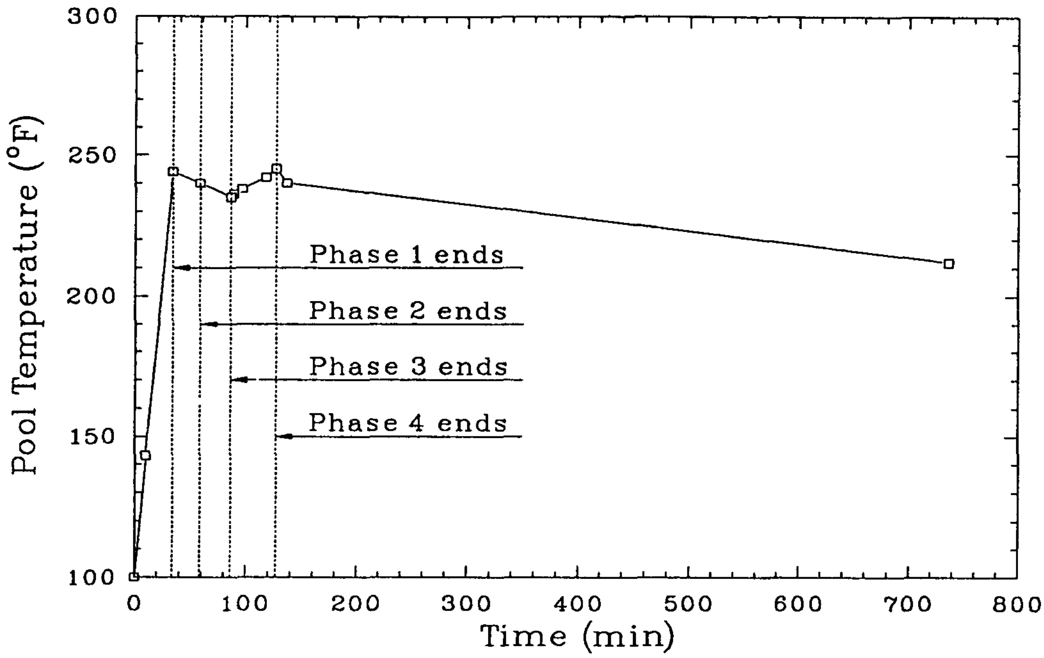

Figure B-23. Peach Bottom TC3 suppression pool temperature.

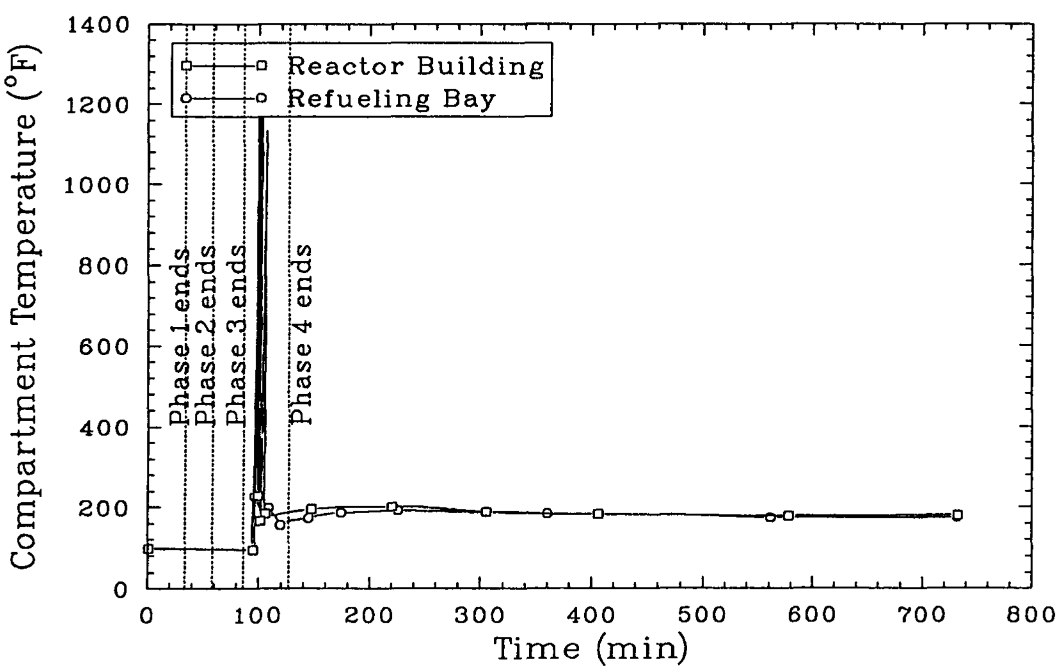

Figure B-24. Peach Bottom TC3 reactor building data.

predicted to be 200° F. Pressure spikes to approximately 25 psia are predicted in the reactor building during the hydrogen burns.

Temperature data for the dryers and separators and other structures within the reactor vessel were not presented in NUREG/CR-4624[B-2] for the TC1, TC2 or TC3 sequences in NUREG/CR-4624. Dryer and separator temperature data for ATWS sequences is taken from BMI-2104[B-1] for the TC-γ sequence and is presented In Figure B-25. The TC-γ sequence is similar to TC1 sequence. The time to core melt, vessel failure, and containment failure is somewhat faster for TC-γ than for TC1, probably because the assumed power level of 30% in TC-γ is higher than that assumed for TC1. The structure temperatures may be somewhat higher because of the power level assumption. However, more realistic gas and structure temperatures will still be well above the temperature qualification limit for any in-vessel instrumentation.

B-1.4 Transient Sequences

The accident sequence selected to represent the transient sequences for this evaluation is the TW-γ sequence from BMI-2104.[B-1] The TW-γ sequence is the only non-ATWS sequence analyzed in BMI-2104 and NUREG/CR-4624.[B-2]

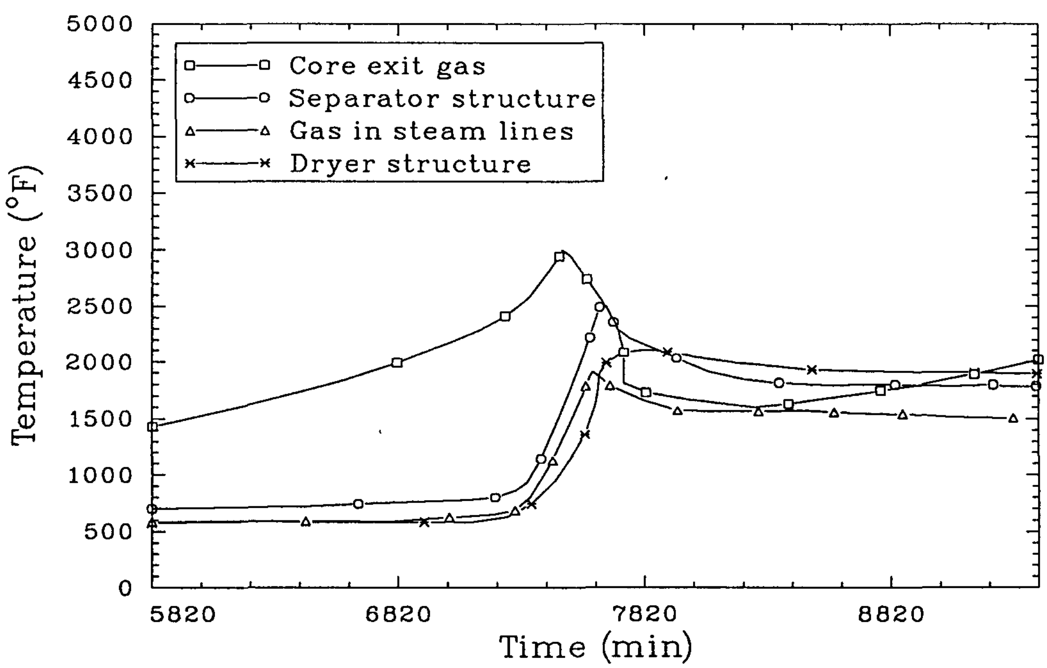

Figure B-25. Peach Bottom TC-γ gas and structure temperatures.

The TW-γ sequence is initiated by a transient with a successful reactor scram. All ECCSs function successfully but the suppression pool cooling systems fail. As a result of this failure, the suppression pool heats up causing pressurization and eventual failure of the containment. Upon containment failure, flashing of the saturated suppression pool is assumed to lead to failure of the emergency core cooling pumps due to cavitation, causing core melt.

Table B-8 presents the timing of key events for the TW-γ sequence. Containment failure is predicted to occur at about 1756 minutes after accident initiation, which is about 991 minutes before core melt is predicted to start.

Figure B-26 presents the pressure and temperature conditions in the reactor coolant system up to vessel failure for the TW-γ sequence. Figure B-27 presents the pressure and temperature conditions in the drywell for the TW-γ sequence. The suppression pool temperature is presented in Figure B-28. The pool temperature remains at saturation conditions throughout the sequence once pool boiling occurs. No data for the temperature in the reactor building outside the primary containment is given in BMI-2104.[B-1]

Figure B-29 presents the gas temperature at the core exit and in the steamlines and the temperatures of the separator and lower annulus structures. The reactor system flowpath considered in the MARCH/MERGE analysis is from the core through the separators and dryers and out through the steamlines.

B-2. REVIEW OF RESULTS

In reviewing the matrix of results presented in Table B-1, a number of the PDSs and accident progression bins were not covered by the accident sequences presented in BMI-2104[B-1] and NUREG/CR-4624.[B-2] These PDSs include sequences where containment failure (or venting) does not occur, where vessel breach does not occur, and where core damage does not occur. Challenges to instrument availability could still occur in these cases although they would likely be less severe than in accidents where events such as containment failure or vessel breach occur.

There are other types of sequences that are of interest in evaluating instrument availability, including the following:

- Sequences initiated by a medium and small break LOCA with failure of high and low pressure ECCSs. Containment heat removal systems are functioning

- Sequences initiated by a transient with failure of the high pressure coolant injection system. The operator depressurizes the system using the automatic depressurization system, but failure of the low pressure coolant injection and low pressure core spray systems leads to core melt at low reactor coolant system pressure. Containment heat removal systems are functioning.

Table B-8. Key accident event times for Peach Bottom TW-γ sequence.

Event	Time (min)
Containment failure occurs	1756.2
Core uncovery	2619.6
Core melt starts	2747.9
Core slump occurs	2817.1
Core collapse occurs	2818.9
Lower head dryout	2829.3
Lower head failure	3055.2
Start of concrete attack	3055.2
End calculation	3655.4

Note: Data from Table 6.2 of BMI-2104, Volume 2.

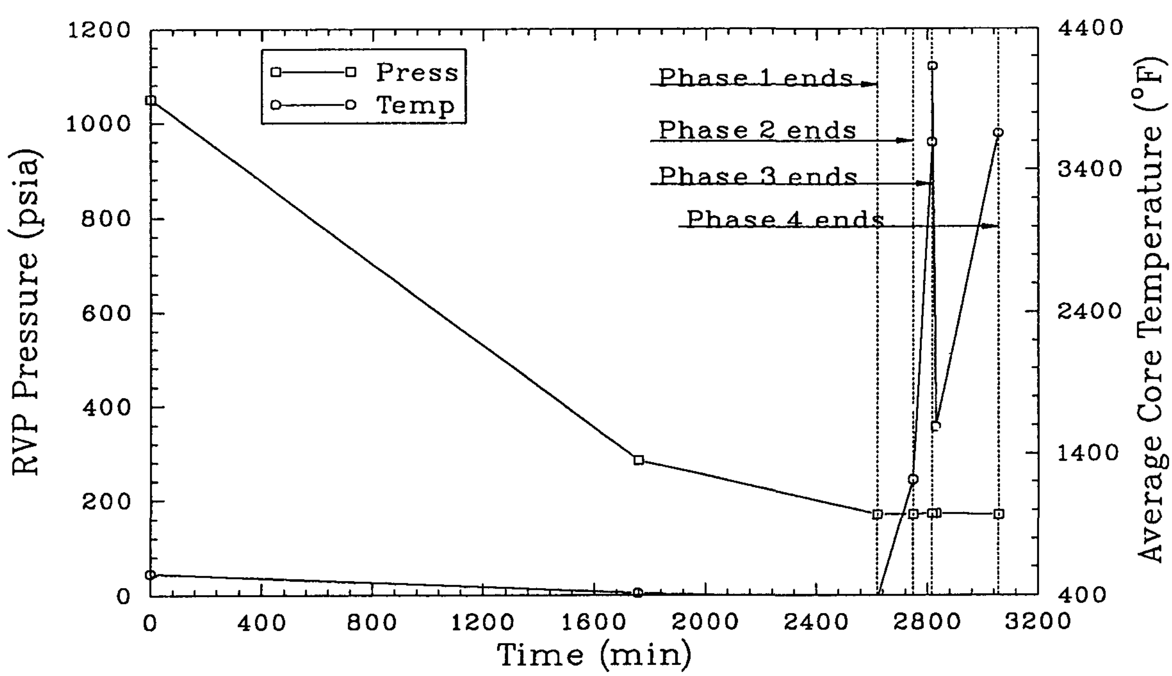

Figure B-26. Peach Bottom TW-γ reactor system data.

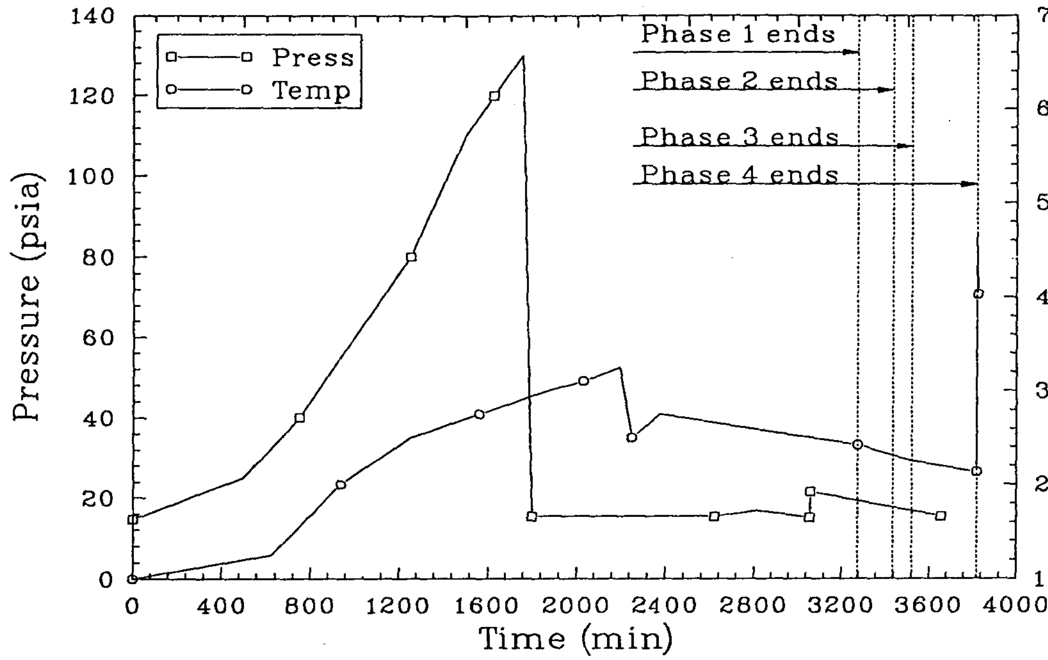

Figure B-27. Peach Bottom TW-γ drywell data.

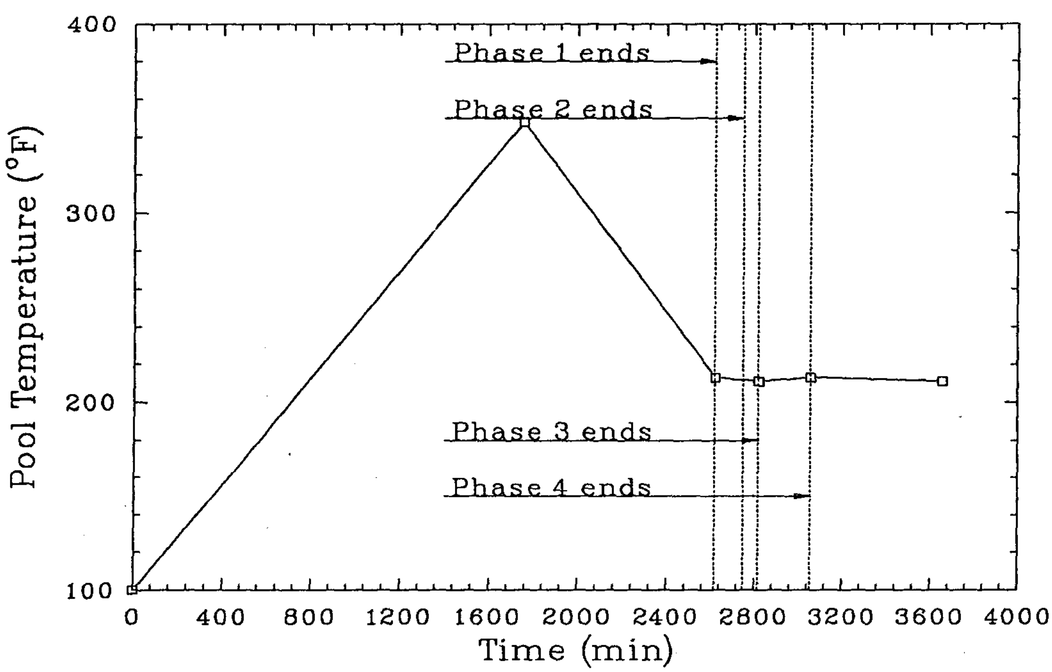

Figure B-28. Peach Bottom TW-γ suppression pool temperature.

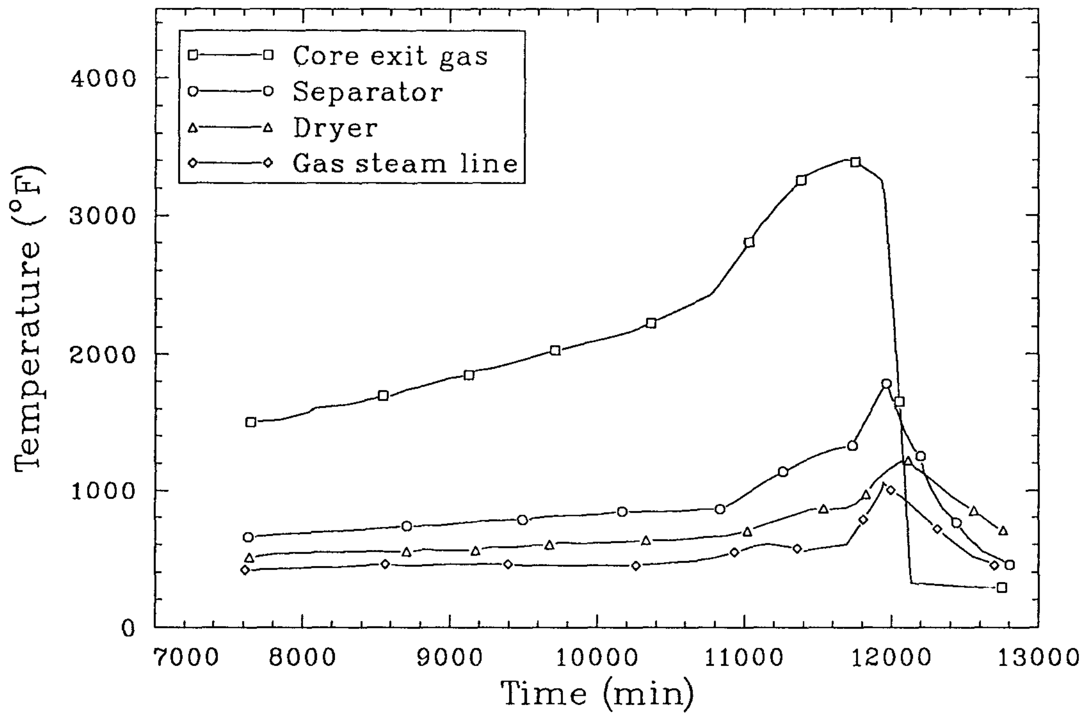

Figure B-29. Peach Bottom TW-γ gas and structure temperatures.

- Sequences initiated by a transient with failure of the high pressure core injection system and failure of the automatic depressurization system. Core melt occurs at high reactor coolant system pressure. Containment heat removal systems are functioning.

These sequences are evaluated against those presented in Table B-1 to determine if the possible range of plant conditions in the reactor coolant system are covered in the BMI-2104[B-1] and NUREG/CR-4624[B-2] analyses. Also, a bypass or V sequence is analyzed in NUREG/CR-4624. Bypass sequences are not listed on Table B-1 because of their low probability relative to the other PDSs.

For the LOCA and transient sequences listed above, it is judged that the results presented in BMI-2104[B-1] and NUREG/CR-4624[B-2] bound the possible conditions expected in the reactor coolant system, containment (drywell and wetwell), and reactor building that affect instrument availability.

Conditions in the reactor coolant system do not change appreciably for different sequences. In general, it is the assumed fuel melting temperature that determines the temperature reached in the vessel. In any situation, temperatures sufficiently hot to melt fuel will cause failure of any instrument or instrument tap near the core region.

Transient and LOCA initiated sequences with ECCS failure, but where containment heat removal systems are functioning will result in a core melt. The conditions in containment will remain relatively cool until after lower head failure where the generation of noncondensible gases from core-concrete interaction could cause heatup and pressurization of the containment. Containment venting would become necessary or containment failure would occur, given a sufficiently high gas production rate. The pressures and temperatures reached would not be much different than those shown for the station blackout sequences TB1 and TB2, although the time at which the peak temperatures would be reached would be different.

Severe conditions could exist in the reactor building if the containment is vented or if containment failure occurs. For transient or LOCA initiated sequences with ECCS failure, containment venting may be needed to avert containment failure after lower head failure. Survivability of the ducts used for containment venting would be a concern although many utilities with BWR plants with Mark I containments are installing a hardened system for containment venting in response to generic letter 89-16.[B-4] These vent systems are typically designed for decay heat loads as opposed to the heat loads possible during an ATWS with SLCS failure. Use of the hardened vent system would prevent severe conditions from developing in the reactor building during non-ATWS accidents and alleviate concerns on availability of instruments located in the reactor building. During an ATWS, use of the hardened vent system would decrease the rate of containment pressurization and prolong the time to containment failure. If the decision was made to vent the containment through vents other than the hardened system to achieve a greater depressurization rate during an ATWS or if a hardened vent system is not installed, duct failure could occur. This failure would result in severe conditions in the reactor building, affecting instrument availability.

The bypass or V sequence analyzed in NUREG/CR-4624[B-2] assumes a rupture of a 6 in. line in the low pressure ECCS in the reactor building. This rupture could cause severe conditions in the reactor building although not to the extent caused by an ATWS. Conditions in the reactor coolant system and containment are bounded by the results from other sequences.

B-3. REFERENCES

B-1. J. A. Gieseke et al., *Radionuclide Release Under Specific LWR Accident Conditions - BWR - Mark 1 Design*, BMI-2104, Volume II, July 1984.

B-2. R. S. Denning et al., *Radionuclide Release Calculations for Selected Severe Accident Scenarios - BWR, Mark 1 Design*, NUREG/CR-4624, Volume 1, July 1986.

B-3. U.S. Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five Nuclear Power Plants*, NUREG-1150, Volumes 1 and 2, December 1990.

B-4. U.S. Nuclear Regulatory Commission, *Installation of a Hardened Wetwell Vent*, Generic Letter 89-16, September 1, 1989.

Appendix C

Review of NUREG/CR-5720 Safety Function Information Needs

Appendix C

Review of NUREG/CR-5702 Safety Function Information Needs

The appendix presents a review of the information needs for each of the safety functions shown in Figures 1, 2, and 3 of the main report based on the data in Appendix A to NUREG/CR-5702.[C-1] The availability of instruments needed to meet the information needs is also assessed.

Appendix A of NUREG/CR-5702[C-1] consists of a set of tables listing the information needs to meet each of the safety objectives and safety functions presented in Figures 1, 2, and 3. Each table in Appendix A corresponds to a mechanism on the safety objective tree. The instruments required to monitor the safety function of interest are presented in these tables.

The instruments identified in Appendix A of NUREG/CR-5702[C-1] needed to verify the status of each safety function is presented in the following sections. Following each instrument name is three items in parenthesis that are the instrument location, Regulatory Guide 1.97[C-2] category, and backup power source. The data is taken from Table 3 of the main report. An explanation of the backup power source categories is provided in Section 4.2 of the main report. Instrument availability is assessed for ATWS initiated events, for other than ATWS initiated events, and for station blackout. Note that severe conditions means that environmental conditions in the vicinity of the instrument have exceeded the qualification limits.

C-1. MAINTAIN HEAT SINK SAFETY FUNCTION

The Maintain Heat Sink safety function will be challenged if: (a) the condenser becomes isolated from the reactor or the condenser vacuum is lost, or (b) the suppression pool has a high water temperature or an abnormal water level. Instruments used to identify challenges to this safety function are the suppression pool water level (reactor building, Category 1, Class 1E), suppression pool water temperature (torus shell; Category 1, Class 1E), main steam isolation valve position indicator (drywell, Category 1, Class 1E), bypass valve position indicator (turbine building, Category 3, uninterruptable power), condenser vacuum (turbine building, Category 3, onsite power), reactor pressure (reactor building, Category 1, uninterruptable power), and containment (drywell) pressure (reactor building, Category 1, Class 1E). The main steam line flowrate is also identified for this safety function, but is not included in the Regulatory Guide 1.97 review for Peach Bottom.

C-1.1 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for ATWS Initiated Events

Severe conditions will develop in the drywell affecting the ability to monitor the main steam isolation valve position. Severe conditions in the reactor building may also result from duct failure due to venting through nonhardened ducts or due to containment failure and can occur at any time during the accident, including prior to core damage. Severe conditions in the reactor building will probably result in degraded instrument performance limiting the ability to reliably monitor suppression pool water level suppression pool temperature, reactor pressure and containment pressure. The suppression pool temperature instrument will likely be operating in conditions outside of its range (30 to 230°F) during an ATWS. Instruments located in the turbine building used to monitor bypass valve position and condenser vacuum should remain available throughout the event.

C-1.2 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for Other Than ATWS Initiated Events

Severe conditions will develop in the drywell for transient sequences with successful ECCS function but with loss of containment heat removal. These conditions could affect the ability to monitor the main steam isolation valve position. The suppression pool temperature instrument may be operating in conditions outside of its range (30 to 230°F) during these types of sequences. Instruments located in the turbine building used to monitor secondary side parameters should remain available. Assuming the availability of a hardened system for containment venting, instruments located in the reactor building should remain available.

For accident sequences involving vessel failure after core melt, degraded performance of the instruments used to monitor the Maintain Heat Sink safety function becomes irrelevant.

C-1.3 Information Needs and Instrument Availability Assessment During a Station Blackout or Loss of dc Power

All instruments needed for the Maintain Heat Sink safety function are on either a Class 1E or uninterruptable power source. If battery backup is provided for the Class 1E instrument power supplies, then these instruments would be initially available.

C-2. MAINTAIN REACTIVITY CONTROL SAFETY FUNCTION

The Maintain Reactivity Control safety function will be challenged if: (a) the control rods fail to insert, or (b) recriticality occurs during the accident. Instruments used to identify challenges to this safety function are the neutron monitoring instruments including the source range monitor (drywell, Category 1, various sources), intermediate range monitor (drywell, Category 1, various sources), average power range monitor (drywell, Category 1, various sources), control rod position indicator (drywell, Category 3, uninterruptable power), reactor pressure (reactor building, Category 1, uninterruptable power). Various sources indicate that different power sources are used for different components of the source range monitors, intermediate range monitors, and average power range monitors as shown on Table 3 of the main report.

The main steamline flowrate and safety relief valve acoustic monitors are also identified for this safety function but are not included in the Regulatory Guide 1.97 review for Peach Bottom.

C-2.1 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for ATWS Initiated Events

Severe conditions in the drywell could occur at any time during the accident, including prior to core damage due to pressurization of the containment. Degraded performance of the source range monitors, intermediate range monitors, average power range monitors, and control rod position indicators could result, increasing the difficulty of monitoring power during the ATWS. If severe conditions in the reactor building develop due to duct failure due to venting through nonhardened ducts or due to containment failure, degraded performance of the reactor pressure instruments could also result.

C-2.2 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for Other Than ATWS Initiated Events

The ability to reliably monitor core power could be lost due to severe conditions in the drywell prior to core damage due to degraded performance of the source range monitors, intermediate range monitors, average power range monitors, and control rod position indicators. Severe conditions in the containment could occur prior to core damage if containment heat removal capability has failed during a transient with successful ECCS function. If core melt occurs, degraded performance of the instruments used to monitor the Maintain Reactivity Control safety function becomes irrelevant.

C-2.3 Information Needs and Instrument Availability Assessment During a Station Blackout

The source range monitors and intermediate range monitors have drive motors that are powered by onsite sources. Degraded performance of these instruments would result since sources would be unavailable during a station blackout. The average power range monitors may be available if the reactor protection system motor generator set is unaffected by the station blackout. The control rod position indicators and reactor pressure indication are powered by uninterruptable (battery backed) sources and should be available during a station blackout.

C-3. MAINTAIN CORE HEAT REMOVAL SAFETY FUNCTION

The Maintain Core Heat Removal Safety Function will be challenged if (a) an adequate inventory of cooling water is not available, or (b) flow blockages occur in the core restricting fluid flow. Instruments used to identify challenges to this safety function are the reactor vessel water level (reactor building, Category 1, Class 1E), reactor pressure (reactor building, Category 1, Class 1E), containment area radiation monitor (drywell, Category 1, Class 1E), and containment atmosphere hydrogen monitor (reactor building, Category 1, Class 1E). The post accident sampling system (outside the reactor building, Category 3, onsite or offsite sources) is also identified as an information source.

C-3.1 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for ATWS Initiated Events

Degraded performance of the containment area radiation monitoring instruments is possible before core damage when containment conditions exceeds the instrument qualification limits. Severe conditions in the reactor building may also result from duct failure due to venting through nonhardened ducts or due to containment failure and can occur at any time during the accident, including prior to core damage. Severe conditions in the reactor building will result in the inability to reliably monitor vessel water level, reactor pressure, and containment atmosphere hydrogen content. Containment conditions can also affect the reactor vessel level instruments, as compensation elements are located inside the drywell.

Postaccident sampling capability should remain available during an ATWS since sampling equipment is typically located outside the reactor building.

C-3.2 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for Other Than ATWS Initiated Events

The ability to reliably monitor containment atmosphere radiation level could be degraded due to severe conditions in the containment before core damage. Severe conditions in the containment could occur prior to core damage if containment heat removal capability has failed during a transient with successful ECCS function. Assuming the availability of a hardened vent system for containment venting, instruments located in the reactor building should remain available.

For accident sequences involving vessel failure after core melt, degraded performance of the instruments used to monitor the Maintain Core Heat Removal safety function becomes irrelevant.

Postaccident sampling capability should remain available for other than ATWS initiated events since sampling equipment is typically located outside the reactor building.

C-3.3 Information Needs and Instrument Availability Assessment During a Station Blackout

All instruments needed for the Maintain Heat Sink safety function are on either a Class 1E or uninterruptable power source. If battery backup is provided for the Class 1E instrument power supplies, then these instruments would be initially available.

Postaccident sampling capability will be unavailable during the station blackout since the power supply to the sampling equipment is from both onsite and offsite ac power sources.

C-4. MAINTAIN VESSEL BOUNDARY SAFETY FUNCTION

The Maintain Vessel Boundary safety function is challenged if: (a) vessel overtemperature occurs as core debris accumulates in the lower head as the core melts or (b) vessel overpressure occurs due to a steam explosion when molten core material mixes rapidly with water in the core region or in the lower plenum. Instruments used to identify challenges to this safety function are the reactor pressure (reactor building, Category 1, uninterruptable power), and drywell sump level (drywell, Category 1, onsite power). The reactor vessel temperature recorder is also identified for this safety function but is not included in the Regulatory Guide 1.97 review for Peach Bottom.

C-4.1 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for ATWS Initiated Events

Degraded performance of the drywell sump level instruments is possible before core damage when containment conditions exceeds the instrument qualification limits. Severe conditions in the reactor building may result from duct failure due to venting through nonhardened ducts or due to containment failure and can occur at any time during the accident, including prior to core damage. Severe conditions in the reactor building will result in the inability to reliably monitor reactor pressure.

C-4.2 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for Other Than ATWS Initiated Events

The ability to reliably monitor drywell sump level could be lost due to severe conditions in the containment before core damage. Severe conditions in the containment could occur prior to core damage if containment heat removal capability has failed during a transient with successful ECCS function. Assuming the availability of a hardened vent system for containment venting, instruments located in the reactor building should remain available.

For accident sequences involving vessel failure, degraded performance of the instruments used to monitor the Maintain Vessel Boundary safety function becomes irrelevant.

C-4.3 Information Needs and Instrument Availability Assessment During a Station Blackout

The reactor pressure instruments are powered by an uninterruptable power source and would be initially available during a station blackout. The drywell sump level is powered by an onsite source and will not be available during a station blackout.

C-5. MAINTAIN CONTAINMENT PRESSURE CONTROL SAFETY FUNCTION

The Maintain Containment Pressure Control safety function is challenged if: (a) rapid steam condensation in the containment causes negative pressure in the containment, or (b) containment overpressure due to insufficient energy removal, insufficient pool level or other causes. The instruments used to identify challenges to this safety function is the drywell pressure (reactor building, Category 1, Class 1E). The suppression chamber pressure is also identified for this safety function but is not included in the Regulatory Guide 1.97 review for Peach Bottom.

C-5.1 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for ATWS Initiated Events

Severe conditions in the reactor building may result from duct failure due to venting through nonhardened ducts or from containment failure and can occur at any time during the accident, including prior to core damage. Severe conditions in the reactor building could result in the inability to reliably monitor drywell pressure.

C-5.2 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for Other Than ATWS Initiated Events

Assuming the availability of a hardened system for containment venting, the drywell pressure instruments located in the reactor building should remain available during transient sequences with successful ECCS function but where containment heat removal is lost.

If the containment heat removal systems are functioning, then the instruments located in the containment used to monitor this safety function should remain available until after vessel failure. Instrument located in the reactor building should remain available until containment failure occurs.

C-5.3 Information Needs and Instrument Availability Assessment During a Station Blackout

The drywell pressure instruments are powered by a Class 1E power source. If battery backup is provided for the Class 1E source, then these instruments would be available.

C-6. MAINTAIN CONTAINMENT TEMPERATURE CONTROL SAFETY FUNCTION

The Maintain Containment Temperature Control safety function is challenged if: (a) loss of adequate containment heat removal occurs, or (b) molten core material comes in contact with the containment shell or melts through the basemat. Instruments used to identify challenges to this safety function are the drywell temperature (drywell, Category 2, Class 1E), drywell spray flow (reactor building, Category 2, onsite power), suppression pool temperature (torus shell, Category 1, Class 1E), and suppression chamber spray flow rate (reactor building, Category 2, onsite power). The drywell unit cooler status and vent and purge flow meter are also identified for this safety function but is not included in the Regulatory Guide 1.97 review for Peach Bottom.

C-6.1 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for ATWS Initiated Events

Severe conditions in the drywell could occur during an ATWS resulting in degraded performance of the drywell temperature and suppression pool temperature instruments. It is likely that the suppression pool temperature will be above the upper temperature limit of the instrument during an ATWS. Severe conditions in the reactor building may result from duct failure due to venting through nonhardened ducts or from containment failure and can occur at any time during the accident, including prior to core damage. Severe conditions in the reactor building could result in degraded performance of the drywell spray flow and suppression pool spray flow instrumentation.

C-6.2 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for Other Than ATWS Initiated Events

The ability to reliably monitor drywell temperature and suppression pool temperature could be lost due to severe conditions in the containment before core damage. Severe conditions in the containment could occur prior to core damage if containment heat removal capability has failed during a transient with successful ECCS function. It is possible that the suppression pool temperature will be above the upper temperature limit of the instrument during an accident when containment heat removal systems are not functioning. Assuming the availability of a hardened vent system for containment venting, instruments located in the reactor building should remain available.

If the containment heat removal systems are functioning, then the instruments located in the containment used to monitor this safety function should remain available until after vessel failure. Instrument located in the reactor building should remain available until containment failure occurs.

C-6.3 Information Needs and Instrument Availability Assessment During a Station Blackout

The drywell temperature and suppression pool temperature instruments are powered by a Class 1E power source. If battery backup is provided for the Class 1E instruments, then these instruments would be available during a station blackout.

The drywell spray and suppression pool spray flow instruments will be unavailable during the station blackout since the power supply is from onsite ac power sources. However, the spray systems will also be unavailable so that unavailability of these instruments is inconsequential.

C-7. MAINTAIN CONTAINMENT INTEGRITY SAFETY FUNCTION

The maintain containment integrity safety function would be challenged if equipment used for containment isolation failed to either prevent the initiation of containment isolation or to prevent its continuation. Three challenges are identified which are isolation failure, bypass failure or internally generated missiles. Instruments used to identify challenges to this safety function are the isolation valve position indication (drywell and reactor building, Category 1, varies - Class 1E or onsite), drywell pressure (reactor building, Category 1, Class 1E), reactor building area radiation monitoring system (reactor building, Category 1, onsite power), drywell sump and floor water levels (drywell, Category 1, onsite power), and containment oxygen level (reactor building, Category 1, Class 1E). The reactor building temperature is also identified for this safety function but is not included in the Regulatory Guide 1.97 review for Peach Bottom.

C-7.1 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for ATWS Initiated Events

Severe conditions in the drywell will occur during an ATWS resulting in degraded performance of the drywell sump and floor level instruments. Isolation valve position indication instruments in the drywell could also be affected. Severe conditions in the reactor building may result from duct failure due to venting through nonhardened ducts or due to containment failure and can occur at any time during the accident, including prior to core damage. The ability to reliably monitor drywell pressure, reactor building radiation level, and containment oxygen level could be lost. Isolation valve position indication instruments in the reactor building could also be affected.

C-7.2 Information Needs and Instrument Availability Assessment Based on Environmental Condition for Other Than ATWS Initiated Events

The ability to reliably monitor drywell sump and floor water levels could be lost due to severe conditions in the drywell before core damage. Isolation valve position indicators located in the drywell could also be affected. Severe conditions in the drywell could occur prior to core damage if containment heat removal capability has failed during a transient with successful ECCS function. Assuming the availability of a hardened vent system for containment venting, instruments located in the reactor building should remain available.

If the containment heat removal systems are functioning, then the instruments used to monitor this safety function located in the containment should remain available until after vessel failure. Instrument located in the reactor building should remain available until containment failure occurs.

C-7.3 Information Needs and Instrument Availability Assessment During a Station Blackout

The drywell pressure instruments and the containment oxygen level monitor are powered by Class 1E power sources. If battery backup is provided for the Class 1E power sources, then these instruments would be available. The reactor building area radiation monitoring system and the drywell sump level are powered by onsite sources and would be unavailable during a station blackout.

Most of the valve position indicators are powered by Class 1E sources. If battery backup is provided for the Class 1E power sources, then these instruments would be available.

C-8. CONTROL FISSION PRODUCTS IN PRIMARY CONTAINMENT SAFETY FUNCTION

The Control Fission Products in Primary Containment safety function is concerned with reducing the concentration of fission products in the containment atmosphere. Instruments used to identify challenges to this safety function are the primary containment area radiation monitoring system (drywell, Category 1, Class 1E). The postaccident sampling system would also be used to determine fission product levels in the containment atmosphere (outside the reactor building, Category 3, onsite or offsite sources).

C-8.1 Information Needs and Instrument Availability Assessment Based on Environmental Condition for ATWS Initiated Events

Degraded performance of the primary containment area radiation monitoring instruments is possible before core damage if drywell pressure or temperature exceeds the qualification limits.

Postaccident sampling capability should remain available during an ATWS since sampling equipment is typically located outside the reactor building.

C-8.2 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for Other Than ATWS Initiated Events

Degraded performance of the primary containment area radiation monitoring instruments is possible if drywell pressure or temperature exceeds the instrument qualification limits during the event.

Postaccident sampling capability should remain available for other than ATWS initiated events since sampling equipment is typically located outside the reactor building.

C-8.3 Information Needs and Instrument Availability Assessment During a Station Blackout

The primary containment area radiation monitor is powered by a Class 1E supply and should be available if battery backup is provided.

Postaccident sampling capability will be unavailable during the station blackout since the power supply to the sampling equipment is from onsite and offsite ac power sources.

C-9. CONTROL FISSION PRODUCTS IN SECONDARY CONTAINMENT SAFETY FUNCTION

The Control Fission Products in Secondary Containment safety function is concerned with reducing the concentration of fission products in the secondary containment atmosphere. Instruments used to identify challenges to this safety function are the reactor building area radiation monitoring instruments (reactor building, Category 1, onsite power). Sump water sampling is also identified as an information source on the level of fission products in the secondary containment (reactor building).

C-9.1 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for ATWS Initiated Events

Degraded performance of the reactor building area radiation monitoring instruments is possible after core damage due to venting through non-hardened ducts of due to containment failure. In addition, some of the sensors may be operating outside of their range if core damage has occurred, particularly those with an upper limit of 10^4 mR/h.

Postaccident sampling capability should remain available during an ATWS since sampling equipment is typically located outside of the reactor building.

C-9.2 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for Other Than ATWS Initiated Events

Assuming the availability of a hardened vent system for containment venting, reactor building area radiation monitoring instruments should remain available.

Postaccident sampling capability should remain available for other than ATWS initiated events since sampling equipment is typically located outside the reactor building.

C-9.3 Information Needs and Instrument Availability Assessment During a Station Blackout

Reactor building radiation monitoring capability will be unavailable during the station blackout since the power supply to the required equipment is from onsite and offsite ac power sources. Postaccident sampling capability will be unavailable during the station blackout since the power supply to the sampling equipment is from onsite and offsite ac power sources.

C-10. CONTROL FISSION PRODUCTS IN WATER SAFETY FUNCTION

The Control Fission Products in water safety function is concerned with reducing the concentration of fission products in the water present in the containment, particularly the suppression pool. The postaccident sampling system (outside the reactor building, Category 3, onsite or offsite sources) is identified as an information source on the level of fission products in the suppression pool.

C-10.1 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for ATWS Initiated Events

Postaccident sampling capability should remain available during an ATWS since sampling equipment is typically located outside the reactor building.

C-10.2 Information Needs and Instrument Availability Assessment Based on Environmental Conditions for Other Than ATWS Initiated Events

Postaccident sampling capability should remain available since sampling equipment is typically located outside the reactor building.

C-10.3 Information Needs and Instrument Availability Assessment During a Station Blackout

Postaccident sampling capability will be unavailable during the station blackout since the power supply to the sampling equipment is from onsite and offsite ac power sources.

C-11. REFERENCES

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Appendix D

Impact of Radiation Levels During Severe Accident On Instrument Availability

Appendix D

Impact of Radiation Levels During Severe Accidents On Instrument Availability

Radiation levels potentially can challenge the availability of instruments during a severe accident in the long term plant recovery phase of an accident. In particular, instruments which have components made from synthetic organic materials may be particularly susceptible. These materials are typically used in electrical insulators, gaskets, and seals as described in the first draft of NUREG-1150.[D-1]

Thresholds for radiation damage is expressed in terms of integrated dose in rads. As a result, the impact of radiation on instrument availability is not immediate, but is cumulative over time. This difference is in contrast with the effects of temperature or pressure which can impact instrument reliability if some threshold temperature or pressure value is reached.

This appendix provides estimates of integrated dose at typical instrument locations near the reactor coolant system and within the containment for radionuclide levels expected during a severe accident.

D-1. METHODS USED TO ESTIMATE DOSE

Instrument are typically qualified to design basis radiation conditions specified in NUREG-0737.[D-2] These conditions are based on the assumed release of 100% of the noble gas, 50% of the halogen and 1% of the solid radioisotopes for LOCA events which depressurize the reactor coolant system. This release is assumed to be to either the reactor coolant system or containment for a particular piece of equipment, whichever is limiting. For non-LOCA events, a release of 10% of the noble gas and iodine isotopes and no particulates is assumed. Integrated dose to equipment were estimated as part of the overall equipment qualification evaluation based on these assumptions by utilities in response to the post Three Mile Island requirements.

The BMI-2104[D-3] and NUREG/CR-4624[D-4] reports present estimates of the releases of the fission product and other aerosols from the fuel during core melt. Fission products releases are presented for chemical groups based on chemical characteristics. Estimates of the fission product and aerosol distribution for these accident sequences are also presented. The magnitude of the iodine and particulate releases is the principal difference between the BMI-2104[D-3] and NUREG/CR-4624[D-4] reports compared to the NUREG-0737[D-2] data.

Table D-1 presents a comparison of the releases during core melt predicted for the AE and TC2 sequences against those predicted in NUREG-0737.[D-2] The releases for these sequences are typical of the releases predicted during core melt. Almost 100% of the iodine is predicted to be

Table D-1. Comparison of fission product releases between BMI-2104 and NUREG/CR-4624 to NUREG-0737.

Chemical element	Release during core melt (percent)		
	NUREG-0737	BMI-2104	NUREG/CR-4624
		AE	TC2
Xe	100	100	96
I	50	100	96
Cs	1	100	96
Te	1	22	69
Sr	1	7	0
Ba	1	17	1
Ru	1	1	0
La	1	0	14
Ce	1	0	0

Note:

1. Data from Tables 6.9 of BMI-2104, Volume II is presented for the core melt releases for AE up to the time of vessel failure.

2. Data from Tables 4.6 and 4.7 of NUREG/CR-4624 used to determine the core melt releases for TC2 up to the time of vessel failure.

released from the fuel during core melt as opposed to 50% assumed in NUREG-0737. The releases of cesium and tellurium are predicted to be higher than the 1% assumed in NUREG-0737.

The inventory of fission products in the reactor coolant system will increase as a result of the higher core melt releases. In addition, core-concrete interaction is predicted to release additional fission products to the containment. A significant fraction of these fission products can be released to the reactor building if containment failure occurs.

Increased fission product releases to the reactor coolant system and containment will result in increases in the integrated dose to equipment. In particular, instruments in the reactor building were originally designed for the small radiation exposure that would occur during a design basis accident. This exposure is much smaller than the exposure that could occur during a severe accident. The impact of increased fission product releases is evaluated by estimating the integrated dose in the drywell, torus, and reactor building considering the increased fission product release expected during a severe accident. The integrated dose resulting from normal plant operation is not included in these estimates.

A computer program was developed for performing the dose estimates presented in this appendix. The models used in this program are as follows:

- Fission product energy release rate and integrated energy release utilizing the Perkins and King data set,[D-5, D-6] chemical[D-2] grouping data from the Reactor Safety Study,[D-7] BMI-2104,[D-2] and NUREG/CR-4624,[D-3] and radionuclide distribution data from BMI-2104 and NUREG/CR-4624

- Dose computation models for rectangular plane, spherical, and cylindrical geometries from the Engineering Compendium on Radiation Shielding[D-8]

- Dose conversion factor data from ANS-6.1.1.[D-9]

The Perkins and King data set[D-5, D-6] is used to determine the fission product energy release rate and integrated energy release in the reactor coolant system and containment. Data is given for 124 fission products. The fission product inventory is determined given a reactor power level of 3293 MW_{th} with an assumed 100% plant capacity factor. A one year refueling cycle with one-third of the core being refueled each cycle is assumed.

The radionuclides from the Perkins and King data set are divided into nine groups by chemical characteristics generally following the scheme used in the Reactor Safety Study,[D-7] but treating the barium and cerium radionuclides as separate groups as done in NUREG/CR-4624.[D-3] The chemical grouping is presented on Table D-2.

The dose estimates performed for the drywell, torus, and reactor building use the fission product source term data presented in BMI-2104[D-3] or NUREG/CR-4624[D-4] for the sequences presented in Appendix B.

The dose estimate for the drywell utilizes models for a spherical source. To estimate the dose due to airborne radiation, the dose point is assumed to be immersed in a spherical volume source with a volume of 159,000 ft^3 based on Peach Bottom, given that most of the free volume in the drywell is in the lower spherical portion. The receptor is located at the center of the sphere. The dose due to radionuclides that have settled is estimated assuming that the dose point is located at the center of a hemispherically shaped surface source with a diameter of 67 ft. It is assumed that 90% of the radionuclides released to the drywell settle, with the exception of the noble gas nuclides. The total estimated dose is the sum of the dose contribution for airborne and settled radionuclides.

The dose estimate for the pool is performed by representing the suppression pool as a half filled cylindrical volume source of water which is infinitely long. The source diameter is 31 ft and the

Table D-2. Chemical grouping of radionuclides for dose evaluation.

Chemical group name	Radionuclide species included
Xe	Xe, Kr
I	I, Br
Cs	Cs, Rb
Te	Te, Se, Sb
Sr	Sr
Ba	Ba
Ru	Ru, Mo, Pd, Rh and Tc
La	La, Nd, Eu, Y, Pr, Pm, Sm, Zr, Nb (Sn assumed)
Ce	Ce

Note: Based on chemical grouping given in Appendix V, Table 4 of the Reactor Safety Study.[D-7]

water volume is 135,000 ft^3 which are typical for the torus in a Mark I containment. The receptor is assumed to be at the surface of the cylinder along a line bisecting the source.

The dose estimate for the reactor building utilizes models for a infinite slab and rectangular surface sources. To estimate the dose due to airborne radiation, the dose point is assumed to be immersed in an infinite slab source with a width of 50 ft. The dose due to radionuclides that have settled is estimated assuming that the receptor is located inside a five sided rectangular box (no ceiling). Each side is treated as a rectangular surface source. The volume of this box is assumed to be 450,000 ft^3 which is about one-third of the volume of a typical BWR Mark I reactor building. The dimensions of the box are 120 (l) x 50 (w) x 75 (h). The reason for using only one-third of the reactor building volume is because most of the fission products will follow a path through the open hatch to the refueling floor after containment failure or duct failure after containment venting as explained in Section 3.3 of the main report. This path is assumed to intercept one-third of the reactor building volume. It is assumed that 90% of the radionuclides released to the drywell settle, with the exception of the noble gas nuclides. Half of the settled radionuclides are distributed on the floor with the remainder being distributed uniformly on the walls. The receptor is assumed to be

10 ft above the center of the floor. The total estimated dose is the sum of the dose contribution for airborne and settled radionuclides.

The analytical fit of the gamma ray flux to dose conversion factors presented in ANS-6.1.1-1977[D-10] is utilized. Implicit in the use of this data is the assumption that the dose response of the most sensitive components of any instrument is similar to that for tissue.

It is emphasized that the purpose of this evaluation is to estimate the impact of increased radionuclide releases during severe accidents have on total instrument dose on a comparative basis. This evaluation is not intended to be a dose assessment of any particular instrument. Actual dose assessments for particular instruments must be done on a plant specific basis.

D-2. DOSE RESULTS

Figure D-1 presents a dose comparison for the drywell for four cases representing a spectrum of source terms. A brief description of each case is presented below:

- NUREG-0737—based on the assumed release of 100% of the noble gases, 50% of the halogens and 1% of the solid particulates as specified in NUREG-0737. One-half of the halogens and solids are assumed to remain in the drywell.

- AE-drywell—based on the results in BMI-2104[D-3] for the AE large break LOCA.

- Intact Containment—based on the assumed release of 100% of the noble gases, 90% of the radionuclides in the iodine and cesium groups, 20% of the radionuclides in the tellurium group, and 1% of the strontium and barium from the fuel. All of the noble gas radionuclides and one-half of the non noble gas radionuclides remain in the drywell.

- TC3-drywell—based on the results in NUREG/CR-4624 for the TC3 sequence. The TC3 sequence is an ATWS where containment venting occurs.

The fission product source terms in the drywell for each of these cases is presented in Table D-3.

The results shown in Figure D-1 shows that the intact containment produces the highest dose. However, the dose results assuming NUREG-0737 releases is within a factor of two of the intact containment case. The AE-drywell case is lower than the NUREG-0737 case because containment failure occurred before vessel failure, resulting in the release of all noble gases and a substantial fraction of the other fission products to the environment. The TC3-drywell evaluates the dose in the drywell given a high degree of fission product retention in the suppression pool which reduces the drywell source term. An additional TC3-drywell case with no noble gases (no NG) evaluates the

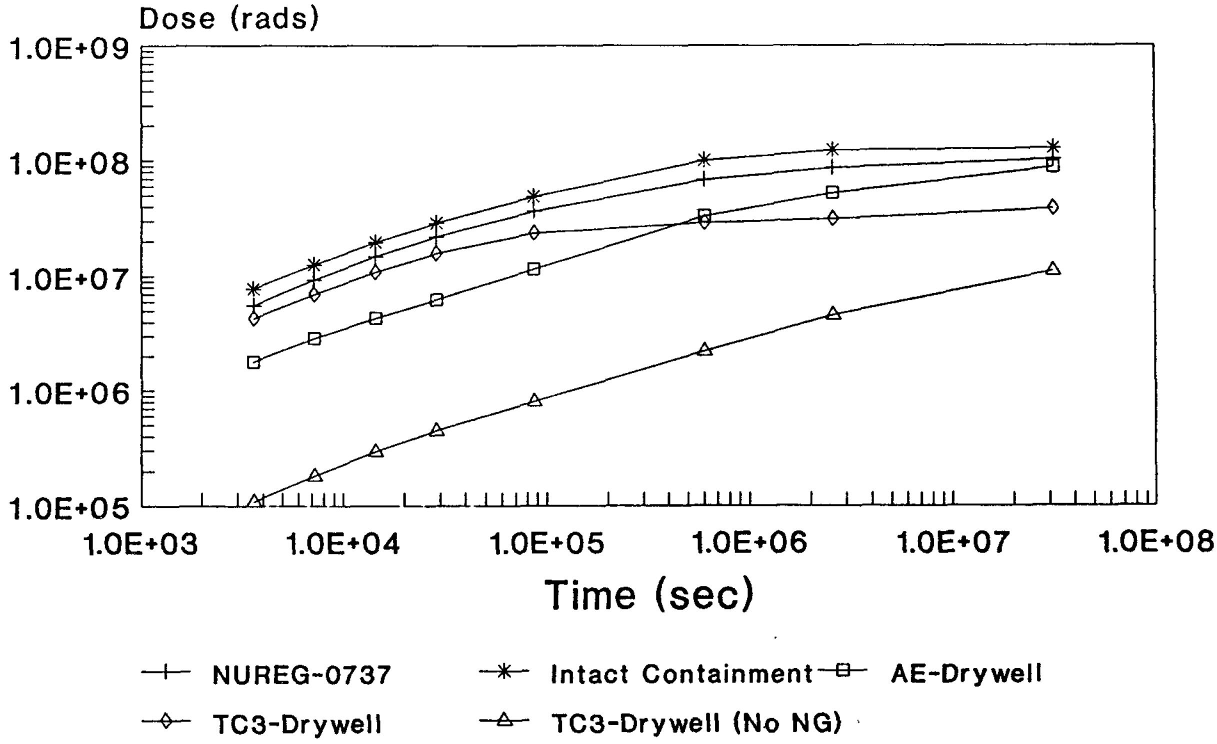

Figure D-1. Dose comparison for drywell.

Table D-3. Fission product source terms used for drywell dose comparison.

Chemical species	AE	Fraction of core inventory TC3	Intact containment	NUREG-0737
Xe	0.0	0.0	1.0	1.0
I	1.2×10^{-1}	2.4×10^{-3}	4.5×10^{-1}	2.5×10^{-1}
Cs	1.4×10^{-1}	2.5×10^{-3}	4.5×10^{-1}	5.0×10^{-3}
Te	3.2×10^{-1}	1.0×10^{-2}	1.0×10^{-1}	5.0×10^{-3}
Sr	6.9×10^{-2}	3.5×10^{-2}	5.0×10^{-3}	5.0×10^{-3}
Ba	1.8×10^{-1}	2.4×10^{-2}	5.0×10^{-3}	5.0×10^{-3}
Ru	6.0×10^{-3}	1.3×10^{-8}	0.0	5.0×10^{-3}
La	3.0×10^{-3}	1.2×10^{-3}	0.0	5.0×10^{-3}
Ce	3.0×10^{-3}	1.9×10^{-3}	0.0	5.0×10^{-3}

Note: Case where all noble gases are retained in the containment also analyzed for TC3.

effect of the noble gases on dose when compared to the TC3-drywell with noble gases case. As shown on Figure D-1, the noble gases add substantially to the integrated dose, depending on the time after accident initiation. It is noted that noble gases are generally released to the environment during a severe accident where containment failure occurs. In the TC3-drywell case, the noble gases are predicted to be released since containment venting after core melt was assumed in the NUREG/CR-4624 analysis.

Figure D-2 presents a dose comparison for the suppression pool for two cases that are described below

- TC3-Pool—based on the results in NUREG/CR-4624[D-4] for the TC3 sequence. The pool radionuclide source term for TC3 is the highest predicted for any of the sequences presented in NUREG/CR-4624.

- NUREG-0737 - based on the assumed release of 100% of the noble gases, 50% of the halogens, and 1% of the solid particulates as specified in NUREG-0737. One-half of the halogens and solids are assumed trapped in the pool water.

The fission product source terms in the suppression pool for each of these cases is presented in Table D-4. The dose predicted for the TC3 sequence is about a factor of 10 greater than that predicted for NUREG-0737 as shown on Figure D-2.

Figure D-3 presents the dose estimates for the TB1, TC2, and TC3 sequences from NUREG/CR-4624[D-4] for the reactor building. The fission product source terms in the reactor building for each of these cases is presented in Table D-5. These cases represent the range of fission product source terms in the reactor building given in NUREG/CR-4624. In the case of TC3, the dose is relatively low because of the high degree of fission product retention in the suppression pool. The dose results for TB1 and TC2 are substantially higher principally due to the high barium and strontium source term predicted in NUREG/CR-4624.

The dose results computed for the intact containment and NUREG-0737 drywell cases are compared to results presented in IDCOR Technical Report 17[D-10] on Figure D-4. The IDCOR report provides estimates of the integrated gamma radiation dose within the containment for both PWR and BWR designs. These dose estimates are based on the design basis accident extended to account for greater releases of solid fission products. No detailed information is presented in the IDCOR report on how the dose results were determined although the method used is probably similar to that presented in this section with more consideration given to specific plant geometry. The IDCOR results are comparable to the results computed for the aforementioned drywell cases.

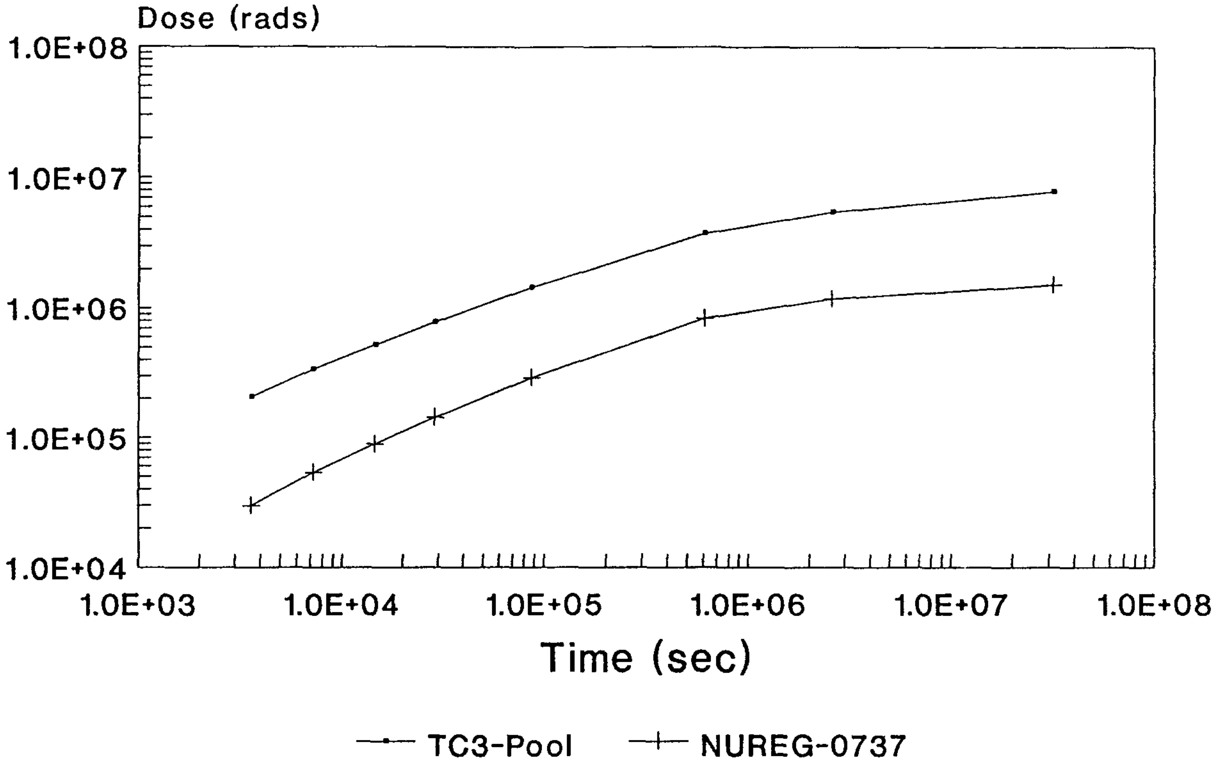

Figure D-2. Dose comparison for suppression pool.

Table D-4. Fission product source terms used for pool dose comparison.

Chemical species	Fraction of core inventory	
	TC3	NUREG-0737
Xe	0.0	0.0
I	9.6×10^{-1}	2.5×10^{-1}
Cs	8.3×10^{-1}	5.0×10^{-3}
Te	3.7×10^{-1}	5.0×10^{-3}
Sr	6.3×10^{-1}	5.0×10^{-3}
Ba	4.2×10^{-1}	5.0×10^{-3}
Ru	1.2×10^{-6}	5.0×10^{-3}
La	2.1×10^{-2}	5.0×10^{-3}
Ce	3.3×10^{-2}	5.0×10^{-3}

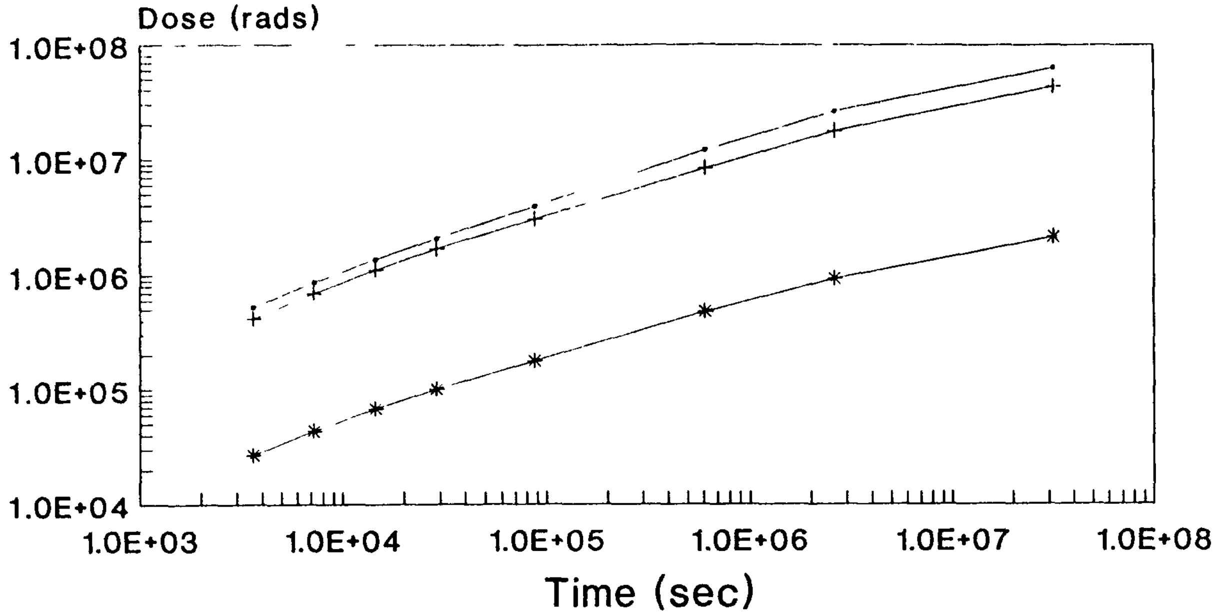

Figure D-3. Dose comparison for reactor building.

Table D-5. Fission product source terms used for reactor building dose comparison.

Chemical species	Fraction of core inventory TB1	TC2	TC3
Xe	0.0	0.0	0.0
I	4.2×10^{-2}	2.3×10^{-2}	5.6×10^{-4}
Cs	4.9×10^{-2}	2.6×10^{-2}	9.7×10^{-4}
Te	1.9×10^{-1}	9.7×10^{-2}	4.7×10^{-3}
Sr	2.6×10^{-1}	3.4×10^{-1}	6.0×10^{-3}
Ba	2.2×10^{-1}	2.5×10^{-1}	4.2×10^{-3}
Ru	5.3×10^{-7}	4.1×10^{-7}	2.8×10^{-7}
La	2.4×10^{-2}	1.2×10^{-2}	2.3×10^{-4}
Ce	3.8×10^{-2}	1.9×10^{-2}	3.3×10^{-4}

Note: Case where all noble gases are retained in the containment also analyzed for TC3.

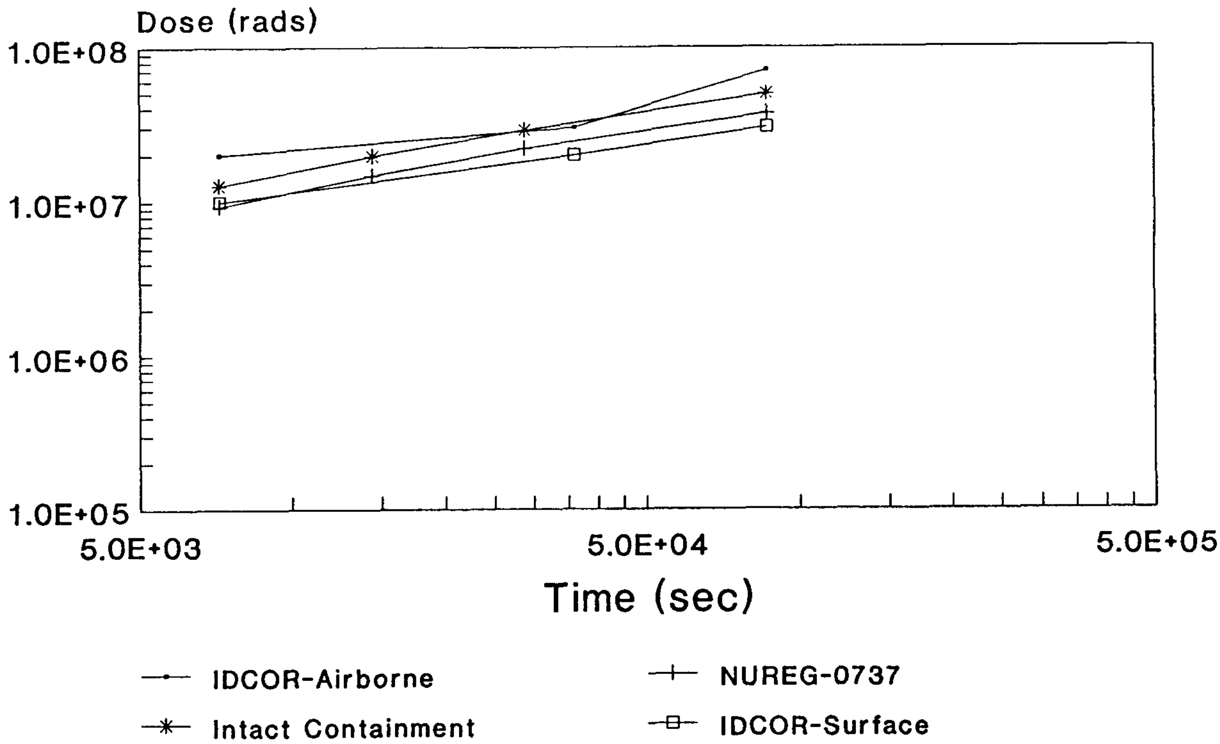

Figure D-4. Dose estimate comparison with IDCOR results.

D-3. ASSESSMENT OF INSTRUMENT AVAILABILITY

Based on data presented in Table 3 of the main report, instruments located within containment (drywell or torus) are generally qualified to an integrated dose of 4.4×10^7 rads. For instruments located in the reactor building outside containment, the radiation qualification limit is generally 3.5×10^4 rads.

Based on the results presented on Figure D-1, the integrated dose to instruments located in the drywell can approach 4.4×10^7 rads about a day after accident initiation for cases where suppression pool retention of fission products is relatively low. The dose for the NUREG-0737 case is also predicted to reach 4.4×10^7 rads after about 1 day. Relatively low pool retention would occur for sequences where the fission products are released directly to the drywell, as would be the case for the AE sequence. For sequences involving a high degree of fission product retention in the pool, which is the case for transient initiated accidents such as TC3, the integrated dose would approach 4.4×10^7 rads after several weeks if the noble gases are not released from the containment. If the noble gases are released from the containment due to containment failure or venting, the dose should never reach 4.4×10^7 rads. These results suggest that radiation exposure to instruments in the drywell should not exceed the qualification limits for most sequences.

Based on the results presented in Figure D-2, the integrated dose to instruments located in the suppression pool should not exceed the qualification limit of 4.4×10^7 rads. The dose due to airborne radionuclides in the torus is negligible based on the results presented in NUREG/CR-4624.[D-4]

The results shown on Figure D-3 suggest that the integrated dose to instruments located in the reactor building is predicted to exceed the qualification limit of 3.5×10^4 rads within a few hours of accident initiation if containment failure occurs during or after core melt. These results also suggest that the qualification limit may be exceeded if the containment is vented during or after core melt and duct failure occurs. These results would apply to any sequence where fission products are released to the reactor building during a severe accident. It should be emphasized that the availability of instruments in the reactor building will be challenged by the temperature conditions for any sequence involving containment failure or duct failure during containment failure (See Section 3.3 of the main report). Degradation of instrument performance that is radiation induced will probably occur only if the instrument can survive the temperatures possible in its location.

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10 SUPPLEMENTARY NOTES

11 ABSTRACT *(200 words or less)*
In support of the U.S. Nuclear Regulatory Commission Accident Management Research Program, the availability of instruments to supply accident management information during a broad range of severe accidents is evaluated for a Boiling Water Reactor with a Mark I containment. Results from this evaluation include: (a) the identification of plant conditions that would impact instrument performance and information needs during severe accidents, (b) the definition of envelopes of parameters that would be important in assessing the performance of plant instrumentation for a broad range of severe accident sequences, and (c) assessment of the availability of plant instrumentation during severe accidents. A similar evaluation for a pressurized water reactor with a large, dry containment design is presented in NUREG/CR-5691.

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