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LICENSING TOPICAL REPORT

Advanced Boiling Water Reactor (ABWR) Hydrogen Recombiner Requirements Elimination

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Summary of Revision 1 Changes

Page Number	Description of Change
7	Changed revision number for Regulatory Guide 1.97 from "Revision 4" to "Revision 3"
A-2	Corrected typographical error in last paragraph
B-15	Replaced DCD markup with replacement page B-27
B-24 through B-27	Added additional Tier 1 DCD markups
C-5	Replaced DCD markup with replacement page C-166
C-6	Replaced DCD markup with replacement page C-167
C-9	Replaced DCD markup with replacement page C-168
C-10	Replaced DCD markup with replacement page C-170
C-13	Replaced DCD markup with replacement page C-173
C-30	Replaced DCD markup with replacement page C-177
C-38 through C-44	Replaced DCD markups with replacement pages C-180 through C-186
C-46	Replaced DCD markup with replacement page C-187
C-57	Replaced DCD markup with replacement page C-190
C-59	Deleted Rev. 0 strikethrough made in error
C-60	Deleted Rev. 0 strikethrough made in error
C-67 through C-69	Replaced DCD markups with replacement pages C-197 through C-199
C-70	Replaced DCD markup with replacement page C-201
C-73 through C-78	Replaced DCD markups with replacement pages C-202 through C-207
C-98	Replaced DCD markup with replacement page C-208
C-118	Replaced DCD markup with replacement page C-209
C-159	Replaced DCD markup with replacement page C-217
C-166 through C-218	Added additional Tier 2 DCD markups

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Appendices

1.0 Introduction

The purpose of this Licensing Topical Report (LTR) is to obtain US Nuclear Regulatory Commission (USNRC) approval of a generic change in the design certification for the U.S. Advanced Boiling Water Reactor (ABWR) design, in accordance with planned revisions to 10 CFR 52.63. This LTR specifically requests approval for changes to Tier 1 of the Design Control Document (DCD) and the generic Technical Specifications to reflect deletion of the hydrogen recombiners and the relaxation of the safety classification of the hydrogen and oxygen monitors from safety-related to non-safety-related.

After issuance of the design certification for the ABWR, the Nuclear Regulatory Commission approved a revision to 10 CFR 50.44 "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," to amend its standards for combustible gas control in light-water-cooled power reactors. The amended rule eliminates the requirements for hydrogen recombiners and relaxes the requirements for hydrogen and oxygen monitoring. This LTR updates the current ABWR certified design to incorporate the amended regulations.

Another purpose of this LTR is to address COL License Information Item 6.2 on providing a comparison of costs and benefits for alternate hydrogen control in accordance with DCD subsection 6.2.7.1.

NRC review of the technical content of this LTR is requested with the understanding that this LTR and subsequent discussions between GE and NRC staff may form the basis for site-specific departures requested in one or more future Combined Operating License Applications.

2.0 Description Of Design Certification

The ABWR design employs hydrogen recombiners. The hydrogen recombiners and associated equipment comprise the Flammability Control System (FCS). The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen and oxygen buildup and subsequent explosion. Hydrogen recombiners reduce the hydrogen concentration in the containment following a design basis loss-of-coolant (LOCA) or main steam line break accident. The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in the containment, thus eliminating any potential for a hydrogen explosion. The hydrogen recombiners are manually initiated since hydrogen and oxygen buildup beyond the flammability limits would not be reached until several days after a design basis accident (DBA).

Two 100% capacity independent hydrogen recombiners systems are provided as part of the certified design. Each consists of controls located in the control room, a power supply and a recombiner. Recombination is accomplished by heating the nitrogen, hydrogen, and oxygen mixture above 1150 °F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. A single recombiner is capable of maintaining the hydrogen

concentration below the 4.1 volume percent (v/p) flammability limit as defined in Regulatory Guide 1.7 for the worst case DBA. Two recombiners are provided for redundancy and independence. Each recombiner is powered from a separate engineered safety feature (ESF) bus, and is provided with a separate power panel and control panel. Operation with at least one hydrogen recombiner ensures that the post LOCA and steamline break accident hydrogen concentrations can be prevented from exceeding the flammability limit.

The Containment Atmospheric Monitoring System (CAMS) is used for post accident monitoring of the primary containment. The system monitors radiation levels as well as the hydrogen and oxygen gas concentration levels in the drywell and in the suppression chamber, displays the measurements in the main control room (MCR), and activates alarms in the MCR upon detection of high levels of radiation and/or gas concentrations.

The CAMS consists of two independent divisions and each division is composed of two radiation channels and a local oxygen and hydrogen gas monitor.

The hydrogen and oxygen monitoring equipment of each CAMS division analyze the hydrogen and oxygen gas concentration levels in the drywell or in the suppression chamber and provide separate gas concentration displays in the MCR.

Each CAMS division is powered from its respective divisional Class 1E power source. In the CAMS, independence is provided between the Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

3.0 Description Of Proposed Departure

The proposed departure eliminates the requirements for hydrogen recombiners from the ABWR certified design and reclassifies the hydrogen and oxygen monitors from safety-related to non-safety-related. Exemption from the generic Technical Specifications associated with the hydrogen recombiner elimination is requested. Tier 2 design departures associated with the Tier 1 departures are also identified.

4.0 Justification Of Proposed Departure

The proposed changes to the certified design are applicable to both the Flammability Control System and the Containment Atmospheric Monitoring System. The basis for the proposed departures are separately addressed.

The proposed change to eliminate the hydrogen recombiners is justified based on the following:

- The revised 10 CFR 50.44 does not require light water reactors which operate with inerted containments to have hydrogen recombiners. The primary containment of the ABWR certified design is inerted with nitrogen.

- The revised 10 CFR 50.44 requires that containments have a capability for ensuring a mixed atmosphere. The ABWR certified design uses the Atmospheric Control System, and containment sprays to ensure that the primary containment atmosphere is well mixed.

- The revised 10 CFR 50.44 requires that equipment be provided for monitoring hydrogen in the containment. The ABWR design provides equipment to continuously measure the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning.

- The revised 10 CFR 50.44 requires that equipment be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. The ABWR design provides equipment to continuously measure the concentration of oxygen in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning.

The proposed changes to the design certification to downgrade the hydrogen monitors from safety-related to non-safety-related are justified based on the following:

- Hydrogen recombiners, and therefore, monitors, are no longer required to mitigate design basis accidents, and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2.

- As part of the rulemaking to revise 10 CFR 50.44, the Commission concluded that hydrogen monitoring is not the primary means of monitoring of beyond design basis accidents.

- Section 4 of Attachment 2 to SECY-00-0198, "Status Report on Study of Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control for Nuclear Power Plants)" found that hydrogen monitors were not risk significant.

- The hydrogen monitoring equipment requirements no longer meet any of the four criteria in 10 CFR 50.36 (c)(2)(ii) for retention in the Technical Specifications and, therefore, may be relocated to other licensee-controlled documents.

The proposed changes to the design certification to downgrade the oxygen monitors from safety-related to non-safety-related are justified based on the following:

- Recombiners, and therefore, oxygen monitors, are no longer required to mitigate design basis accidents, and, therefore, the oxygen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2.

- As part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 2, as defined in Regulatory Guide 1.97, is an appropriate classification for the oxygen monitors because the monitors are required to verify the status of the inert containment.

- Oxygen monitoring is not the primary means of indicating a significant abnormal degradation of the inert containment.

- Oxygen monitors have not been shown in PRA to be risk significant.

- The oxygen monitoring equipment requirements no longer meet any of the four criteria in 10 CFR 50.36 (c)(2)(ii) for retention in the Technical Specifications and, therefore, may be relocated to other licensee-controlled documents.

The proposed changes revise the generic Technical Specifications in Chapter 16 of Tier 2 of the ABWR DCD to reflect changes in the applicable regulatory requirements and criteria in 10 CFR 50.44. With the elimination of the Flammability Control System, hydrogen recombiners are no longer required and the safety classification of the hydrogen and oxygen monitors is changed to non-safety-related.

COL License Information Item 6.2 states that the costs and benefits of alternate hydrogen control in accordance with DCD subsection 6.2.7.1 shall be provided. In the statement of consideration for the 2003 revision to its requirements for combustible gas control in containment in 10CFR50.44, the Commission made the following statement:

> In plants with Mark I and II containments, the containment atmosphere is required to be maintained with a low concentration of oxygen, rendering it inert to combustion. Mark I and II containments can be challenged beyond 24 hours by the long-term generation of oxygen through radiolysis. The regulatory analysis for this proposed rulemaking found the cost of maintaining the recombiners exceeded the benefit of retaining them to prevent containment failure sequences that progress to the very late time frame. The NRC believes that this conclusion would also be true for the backup hydrogen purge system even though the cost of the hydrogen purge system would be much lower because the system also is needed to inert the containment.
>
> 68 Fed. Reg. 54123, 54126 (Sept. 16, 2003). Based, in part, on this analysis, the NRC approved an amendment to 10 CFR § 52.47(a)(1)(ii) stating that in applications for design certifications the required demonstrations of compliance with the Three Mile Island requirements are not required to evaluate alternative hydrogen control systems. Although this amendment did not apply directly to the certification of the ABWR, the ABWR containment, like the Mark I and II containments, is inerted. Consequently, the

NRC's conclusion regarding the cost-benefit balance for recombiners should also apply to the ABWR. Similarly, the Commission found that combustible gas generated during a severe accident is not risk-significant for inerted containments. 68 Fed. Reg. at 54125. Therefore, deletion of the recombiners will not affect the results of the evaluation of Severe Accident Mitigation Design Alternatives (SAMDAs).

The DCD Tier 1 changes and exemption from the generic Technical Specifications have been evaluated under the criteria in Section VIII.A.4 of the ABWR design certification rule. The proposed departures and exemptions meet the criteria as stated in Section VIII of the design certification rule and 10 CFR 50.12 (a)(1) that establish the basis for NRC approval. This evaluation is summarized in Appendix A.

5.0 Nuclear Safety Review

The elimination of the hydrogen recombiner (Flammability Control System) and relaxation of the classification of the hydrogen and oxygen monitors, including the removal of these requirements from generic Technical Specifications, is acceptable in light of existing plant equipment, instrumentation, procedures, and programs that provide effective mitigation of and recovery from reactor accidents.

In addition, the ABWR certified design provides for the Atmospheric Control System to provide and maintain an inert atmosphere in the primary containment during plant operation. Atmospheric mixing of hydrogen releases is achieved by natural processes. Mixing will be enhanced by operation of the containment sprays, which are used to control pressure in the primary containment. Ensuring the containment atmosphere is well mixed is a major tenet of the revised 10 CFR 50.44.

Maintenance and surveillance/inspection requirements will be reduced because of the reduced number of active components.

The revised 10 CFR 50.44 eliminates requirements for hydrogen control systems to mitigate hydrogen releases. The installation of hydrogen recombiners and/or vent or purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design basis LOCA. The Commission has found that this hydrogen release is not risk-significant because the design basis LOCA hydrogen release does not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant accident sequences that would threaten containment integrity.

With the elimination of the hydrogen recombiners, hydrogen and oxygen monitors are no longer required to mitigate design basis accidents and, therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. The hydrogen and oxygen monitors no longer meet the definition of Category 1 in Regulatory Guide 1.97 for post

accident monitoring. As part of the rulemaking to revise 10 CFR 50.44, the Commission found that Category 3, as defined in Regulatory Guide 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are not required to diagnose the course beyond design basis accidents. The Commission also determined that Category 2, as defined in Regulatory Guide 1.97, is an appropriate categorization for the oxygen monitors, because the monitors are required to verify the status of the inert containment.

The hydrogen recombiner and hydrogen and oxygen monitoring equipment are not considered accident precursors, nor does their existence or elimination have any adverse impact on the pre-accident state of the reactor core or post accident confinement of radionuclides within the containment building.

The regulatory requirements for the hydrogen and oxygen monitors can be relaxed without degrading the plant emergency response. The emergency response in this sense, refers to the methodologies used in ascertaining the condition of the reactor core, mitigating the consequences of an accident, assessing and projecting offsite releases of radioactivity, and establishing protective action recommendations to be communicated to offsite authorities. Classification of the hydrogen monitors as Category 3, classification of the oxygen monitors as Category 2 and removal of the hydrogen and oxygen monitors from generic Technical Specifications will not prevent an accident management strategy through the use of the Severe Accident Management Guidelines (SAMGs), the Emergency Plan (EP), the Emergency Operating Procedures (EOP), and site survey monitoring that support modification of emergency plan Protective Action Recommendations (PARs).

6.0 Consistency With ABWR Design Control Document

The changes described in this LTR are to Tier 1 and Tier 2 of the ABWR DCD Revision 4. This includes the piping and instrumentation diagram (Figure 6.2-40) changes that are marked in Appendix C.

A further detailing of changes to the DCD are described in the next section.

7.0 Descriptions Of DCD Markups

Appendix B contains Tier 1 markups to the DCD to account for the changes discussed in this LTR. Appendix C contains Tier 2 markups to the DCD to account for the changes discussed in this LTR. Appendix C also contains a markup of the generic Technical Specifications.

8.0 Conclusion

The proposed change to the certified design to eliminate the Flammability Control System is based on implementation of a revision to 10 CFR 50.44 regulations for combustible gas control. The primary containment of the ABWR certified design is inerted with nitrogen. The revised

10 CFR 50.44 no longer requires light water reactors which operate with inerted containments to have hydrogen recombiners.

The proposed changes to the design certification to downgrade the Containment Atmospheric Monitoring System hydrogen and oxygen monitors from safety-related to non-safety-related are based on the elimination of the hydrogen recombiners and that hydrogen and oxygen monitors are no longer required to mitigate design basis accidents. Therefore, the hydrogen and oxygen monitors do not meet the definition of a safety-related component as defined in 10 CFR 50.2. As part of the rulemaking to revise 10 CFR 50.44, the Commission concluded that hydrogen monitoring is not the primary means of monitoring beyond design basis accidents. The Commission also determined that Category 2, as defined in Regulatory Guide 1.97, is an appropriate classification for the oxygen monitors because the monitors are required to verify the status of the inert containment. Oxygen monitoring is not the primary means of indicating a significant abnormal degradation of the inert containment. Section 4 of Attachment 2 to SECY-00-0198, "Status Report on Study of Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control for Nuclear Power Plants)" found that hydrogen monitors were not risk significant. In addition, oxygen monitors have not been shown in PRA to be risk significant. The hydrogen and oxygen monitoring equipment requirements no longer meet any of the four criteria in 10 CFR 50.36 (c)(2)(ii) for retention in the generic Technical Specifications and, therefore, may be relocated to other licensee-controlled documents.

9.0 References

(1). Regulatory Guide 1.7, Revision 3, "Control of combustible gas concentrations in containment," March 2007.

(2). 10 CFR 50.44, "Combustible gas control for nuclear power reactors," October 16, 2003.

(3). SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)," September 14, 2000.

(4). Regulatory Guide 1.97, Revision 3, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," May 1983.

Appendix A

Justification for Changes to the Generic DCD

Justification for Changes to the Generic DCD

As discussed in this Licensing Topical Report (LTR), the proposed departures pertain to certified design material in Tier 1. This LTR also includes exemptions to generic Technical Specifications (TS) changes, and associated Tier 2 changes to the ABWR DCD. The changes to Tier 1 and the exemption to generic TS require NRC approval. This LTR demonstrates that the proposed changes meet the requirements for a design certification amendment per the proposed revision to 10 CFR 52.63(a).

10 CFR 52.63(a)(1)(vi) (as proposed in SECY-06-220) allows for a change to a generic DCD if the change "Contributes to increased standardization of the certification information." As discussed below, the proposed changes to the generic DCD satisfy this criterion.

The proposed changes involve the implementation of a revision to 10 CFR 50.44 requirements and are intended to be generic and applicable to all COL applicants that reference the ABWR design certification. The changes permitted by the revision to 10 CFR 50.44 were not available during the ABWR design certification. In particular, the revised regulations permit eliminating the hydrogen recombiners and modifying requirements applicable to containment atmosphere monitoring. As discussed in this Licensing Topical Report, the proposed changes comply with the revision to 10 CFR 50.44 regulations. At least one prospective COL applicant (i.e., the COL applicant for South Texas Project Units 3 and 4) intends to implement the proposed departures from the ABWR DCD. Furthermore, it may be expected that other COL applicants will also desire to implement the proposed departures.

Given the generic nature of these proposed changes and the fact that at least one COL applicant intends to make the changes, it would contribute to increased standardization if the NRC were to make a generic change to the DCD to incorporate these proposed changes. Therefore, the proposed changes satisfy the criteria in 10 CFR 52.63(a)(1)(vi).

Appendix B

ABWR DCD Tier 1 Marked Changes

2.2.6 Remote Shutdown System

Design Description

The Remote Shutdown System (RSS) provides remote manual control of safety-related systems to bring the reactor to hot shutdown and subsequent cold shutdown conditions from outside the main control room (MCR). Figure 2.2.6 shows the basic system configuration and scope.

The RSS has two divisional panels and associated controls and indicators for interfacing with the following systems:

(1) Residual Heat Removal (RHR) System

(2) High Pressure Core Flooder (HPCF) System

(3) Nuclear Boiler System (NBS)

(4) Reactor Service Water (RSW) System

(5) Reactor Building Cooling Water (RCW) System

(6) Electrical Power Distribution (EPD) System

(7) Atmospheric Control (AC) System

(8) Emergency Diesel Generator (DG)

(9) Make-up Water System (Condensate), (MUWC)

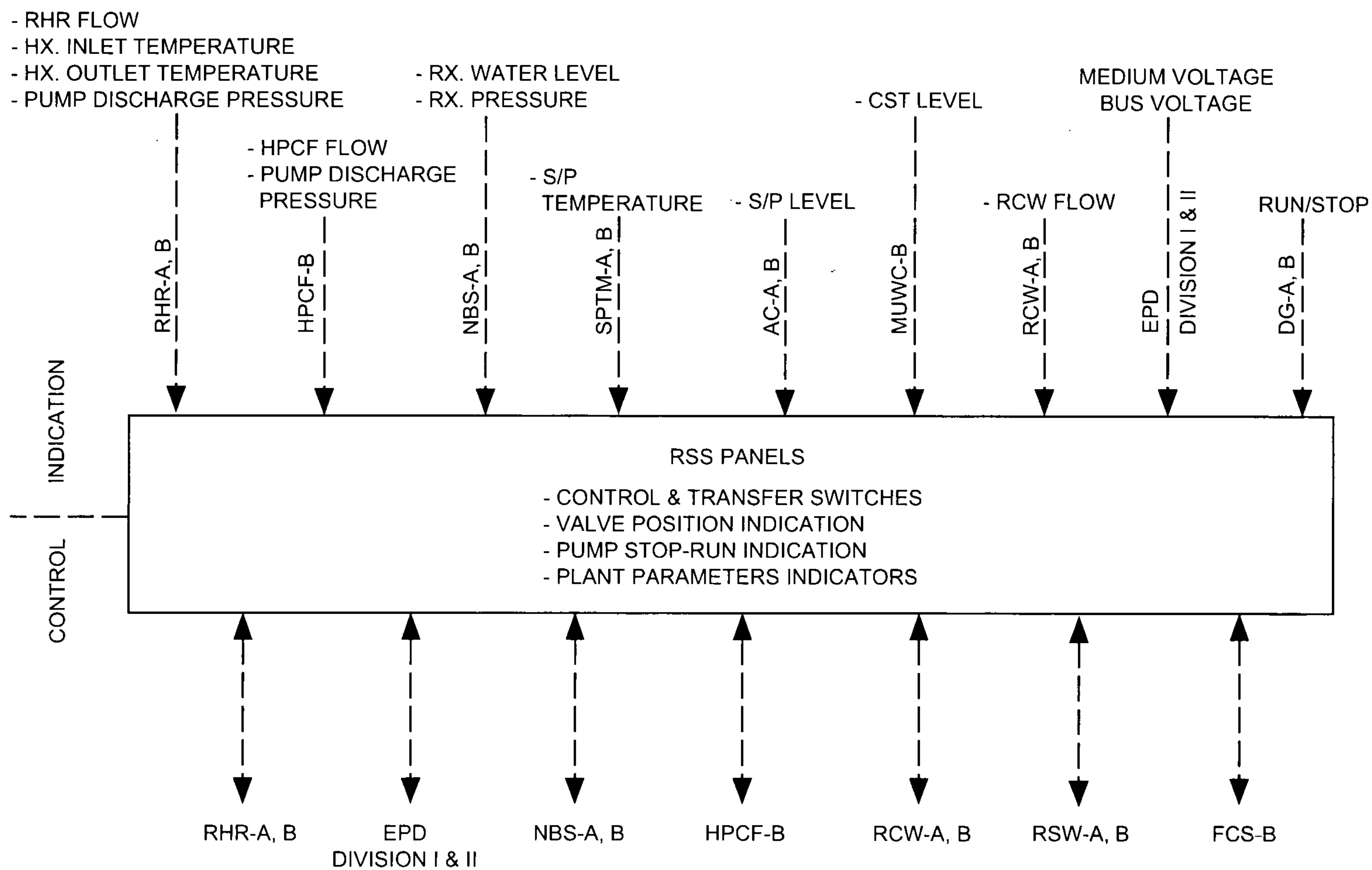
~~(10) Flammability Control System (FCS)~~

(11) Suppression Pool Temperature Monitoring (SPTM) System

RSS controls and indicators are hard-wired direct to the interfacing components and sensors.

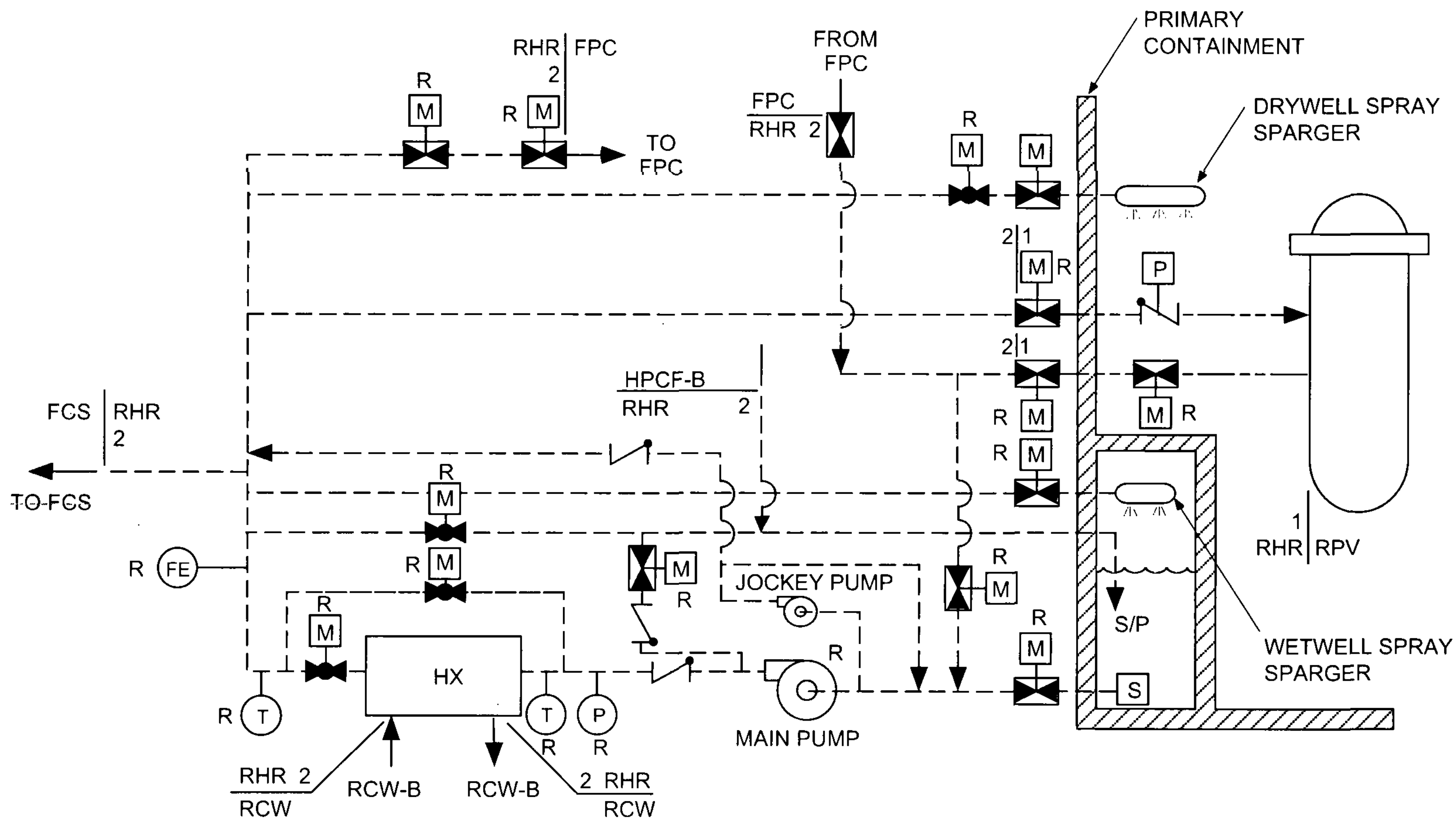
The RSS is classified as a Class 1E safety-related system.

Operation of transfer switches on the RSS panel overrides and isolates the controls from the MCR and transfers control to the RSS. Transfer switch actuation causes alarms in the MCR. Indications required for plant shutdown are provided on the RSS panels as shown on Figure 2.2.6.



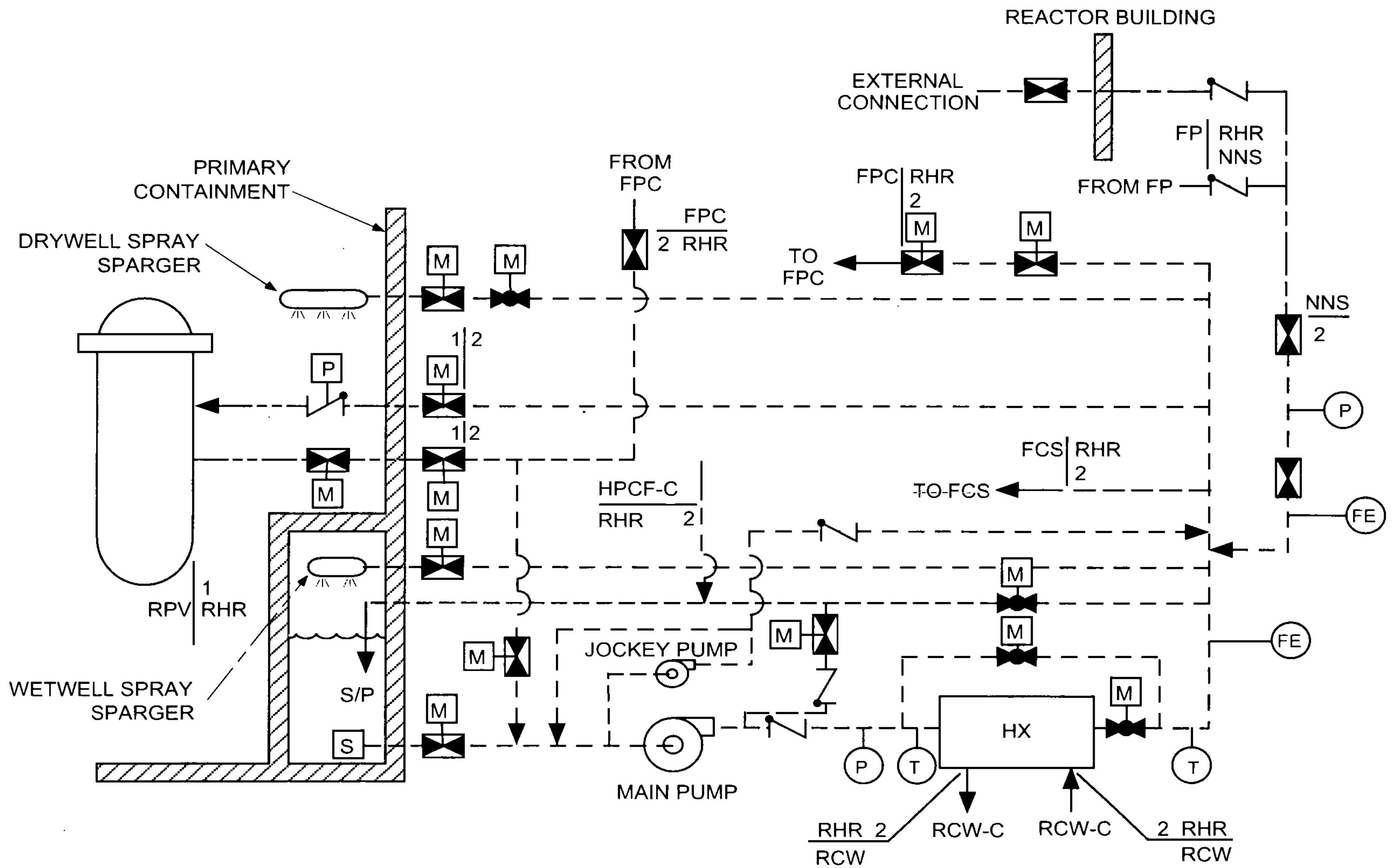
NOTES:
1. RSS PANELS A AND B INTERFACE WITH SYSTEM IN DIVISIONS A AND B (I AND II), RESPECTIVELY.

Figure 2.2.6 Remote Shutdown System



NOTES:
1. ALL ELECTRICAL POWER LOADS FOR THE CLASS 1E COMPONENTS SHOWN ON THIS FIGURE ARE POWERED FROM CLASS 1E DIVISION II EXCEPT FOR THE OUTBOARD CONTAINMENT ISOLATION VALVE OF THE SHUTDOWN COOLING SUCTION LINE, WHICH IS DIVISION III.
2. DRYWELL AND WETWELL SPRAY SPARGERS ARE COMMON TO DIVISIONS B AND C.

Figure 2.4.1b Residual Heat Removal System (RHR-B)



NOTES:
1. ALL ELECTRICAL POWER LOADS FOR THE CLASS 1E COMPONENTS SHOWN ON THIS FIGURE ARE POWERED FROM CLASS 1E DIVISION III EXCEPT FOR THE OUTBOARD CONTAINMENT ISOLATION VALVE OF THE SHUTDOWN COOLING SUCTION LINE, WHICH IS DIVISION I.
2. DRYWELL AND WETWELL SPRAY SPRAGERS ARE COMMON TO DIVISIONS B AND C.

Figure 2.4.1c Residual Heat Removal System (RHR-C)

(4) Isolation of Reactor Building Heating, Ventilating and Air Conditioning (HVAC) System on a signal indicating high drywell pressure, low reactor water level, high radiation in the secondary containment or high radiation in the fuel handling area.

(5) Isolation of containment purge and vent lines on a signal indicating high drywell pressure, low reactor water level, high radiation in the secondary containment or high radiation in the fuel handling area.

(6) Isolation of the Reactor Building Cooling Water (RCW) System and of the HVAC Normal Cooling Water (HNCW) System lines on a signal indicating high drywell pressure or low reactor water level.

(7) Isolation of the Residual Heat Removal (RHR) System shutdown cooling system loops on a signal indicating high reactor pressure or low reactor water level. Also, each RHR shutdown cooling division is individually isolated on a signal indicating high ambient temperature in its respective equipment area.

(8) Isolation of the Reactor Core Isolation Cooling (RCIC) System steamline to the RCIC turbine on a signal indicating high steam flow in the RCIC line, low steam pressure in the RCIC line, high RCIC turbine exhaust pressure, or high ambient temperature in the RCIC equipment area.

(9) Isolation of the Suppression Pool Cleanup (SPCU) System on a signal indicating high drywell pressure or low reactor water level.

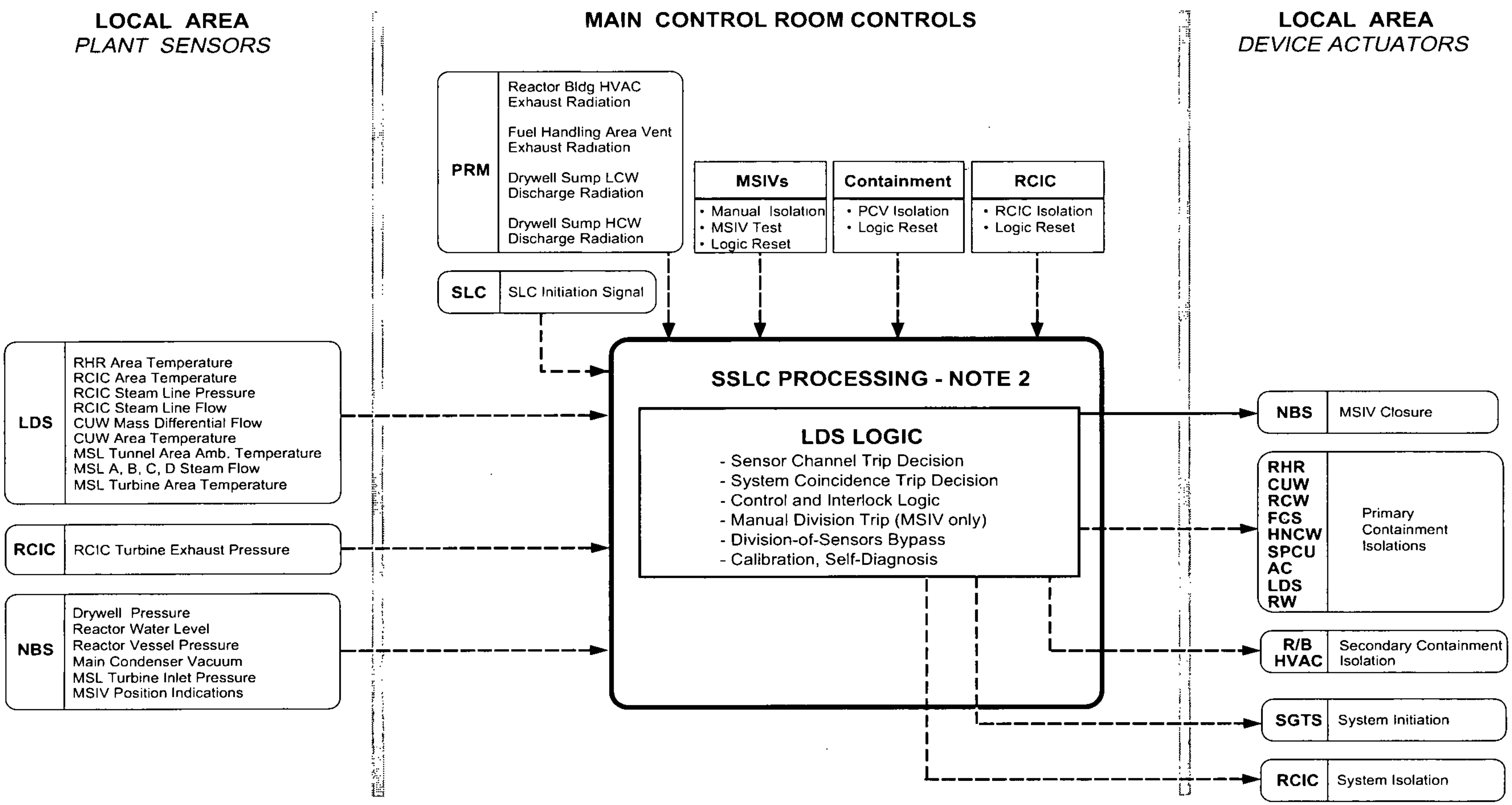
~~(10) Isolation of the Flammability Control System (FCS) on a signal indicating high drywell pressure or low reactor water level.~~

(11) Isolation of the drywell sump low conductivity waste (LCW) and high conductivity waste (HCW) discharge lines on a signal indicating high drywell pressure or low reactor water level. Also, each discharge line is individually isolated on a signal indicating high radioactivity in the discharged liquid waste; only one channel is used for this function.

(12) Isolation of the LDS fission products monitor drywell sample and return lines on a signal indicating high drywell pressure or low reactor water level.

(13) The LDS provides to the neutron monitoring system a signal indicating a high drywell pressure or low reactor water level.

Separate manual controls in the control room are provided in LDS design for logic reset, MSIV operational control, MSIV partial closure tests, and for manual isolation of primary and secondary containment.



Notes:
1. Diagram represents one of four LDS divisions.
2. See Section 3.4, Figure 3.4b for SSLC processing.

Figure 2.4.3 Leak Detection and Isolation System Interface Diagram

Table 2.7.1a Main Control Room Panels Fixed Position Alarms, Displays and Controls (Continued)

A. Fixed Position Controls (Continued)		
ARI (B) Logic Reset Switch	RHR (A) Suppression Pool Cooling Mode Initiation Switch	Div. II ADS Manual ADS Channel 2 Initiation Switch
CRD Charging Water Pressure Low Scram Bypass Switch (A)	RHR (B) Suppression Pool Cooling Mode Initiation Switch	RCIC Div. I Isolation Logic Reset Switch
CRD Charging Water Pressure Low Scram Bypass Switch (B)	RHR (C) Suppression Pool Cooling Mode Initiation Switch	RCIC Div. II Isolation Logic Reset Switch
CRD Charging Water Pressure Low Scram Bypass Switch (C)	RHR (B) Primary Containment Vessel Spray Mode Initiation Switch	RCIC Inboard Isolation Control Switch
CRD Charging Water Pressure Low Scram Bypass Switch (D)	RHR (C) Primary Containment Vessel Spray Mode Initiation Switch	RCIC Outboard Isolation Control Switch
Manual Scram Reset Switch	SGTS (B) Initiation Switch	Fire Protection System Motor Pump Control Switch
RPS Div. I Trip Reset Switch	SGTS (C) Initiation Switch	Fire Protection System Diesel Pump Control Switch
RPS Div. II Trip Reset Switch	Div. I Manual ADS Channel 1 Initiation Switch	~~FCS (B) Control Switch~~
RPS Div. III Trip Reset Switch	Div. I Manual ADS Channel 2 Initiation Switch	~~FCS (C) Control Switch~~
RPS Div. IV Trip Reset Switch	Div. II Manual ADS Channel 1 Initiation Switch	

Table 2.7.1a Main Control Room Panels Fixed Position Alarms, Displays and Controls (Continued)

B. Fixed Position Displays		
RPV Water Level	RCIC Flow	SRV Positions
RCIC Turbine Speed	RCIC Injection Valve Status	Suppression Pool Level
Wetwell Pressure	HPCF (B) Injection Valve Status	Main Steamline Flow
Suppression Pool Bulk Average Temperature	HPCF (C) Injection valve status	SLC Boron Tank Water Level
HPCF (B) Flow	RHR (A) Flow	Recirculation Pump Speeds
HPCF (C) Flow	RHR (A) Injection Valve Status	Average Drywell Temperature
RPV Pressure	RHR (B) Flow	Wetwell Hydrogen Concentration Level
Drywell Pressure	RHR (B) Injection Valve Status	Drywell Hydrogen Concentration Level
Reactor Power Level, (Neutron Flux, APRM)	RHR (C) Flow	Drywell Oxygen Concentration
Reactor Power Level (SRNM)	RHR (C) Injection Valve Status	Wetwell Oxygen Concentration
Reactor Thermal Power	Emergency Diesel Generator (A) Operating Status	~~FCS (B) Operating Status~~
MSIV Position Status (Inboard And Outboard Valves)	Emergency Diesel Generator (B) Operating Status	~~FCS (C) Operating Status~~
Reactor Mode Switch Mode Indications	Emergency Diesel Generator (C) Operating Status	Main Stack Radiation Level
Main Steamline Radiation	Primary Containment Water Level	Time
Scram Solenoid Lights (8) Status	Condensate Storage Tank Water Level	Drywell Radiation Level
Manual Scram Switch (A) Indicating Light Status	SLC Pump (A) Discharge Pressure	Wetwell Radiation Level
Manual Scram Switch (B) Indicating Light Status	SLC Pump (B) Discharge Pressure	
RPV Isolation Status Display	Main Condenser Pressure	

Table 2.11.3b Reactor Building Cooling Water Cooling Loads Division B

Operating Mode/Components*	Normal Operating Conditions	Shutdown	Hot Standby (loss of AC Power)	Emergency (LOCA)
RCW/RSW Heat Exchangers In Service	2	3	3	3
SAFETY-RELATED				
Emergency Diesel Generator B	†	†	‡	‡
RHR Heat Exchanger B	†	‡	‡	‡
Others (safety-related)*f*	‡	‡	‡	‡
NON-SAFETY-RELATED				
RWCU Heat Exchanger	‡	‡	‡	†
FPC Heat Exchanger B**	‡	‡	‡	‡
Inside Drywell	‡	‡	‡	†
Others (non-safety-related)	‡	‡	‡	‡

* Some of these cooling loads are serviced by only one or two RCW divisions. These components may be reassigned to other RCW divisions if redundancy and divisional alignment of supported and supporting systems is maintained and the design basis cooling capacity of the RCW divisions is assured.

† Equipment does not receive RCW in this mode.

‡ Equipment receives RCW in this mode. Rm. 425

f HECW refrigerators, room coolers (RHR, HPCF, SGTS, ~~FCS,~~ CAMS), RHR and HPCF motor bearing and seal coolers, and CAMS cooler.

** Includes FPC room cooler.

Table 2.11.3c Reactor Building Cooling Water Cooling Loads Division C

Operating Mode/Components*	Normal Operating Conditions	Shutdown	Hot Standby (loss of AC Power)	Emergency (LOCA)
RCW/RSW Heat Exchangers In Service	2	3	3	3
SAFETY-RELATED				
Emergency Diesel Generator C	†	†	‡	‡
RHR Heat Exchanger C	†	‡	‡	‡
Others (safety-related)*ƒ*	‡	‡	‡	‡
NON-SAFETY-RELATED				
Others (Non-safety-related)	‡	‡	‡	‡

* Some of these cooling loads are serviced by only one or two RCW divisions. These components may be reassigned to other RCW divisions if redundancy and divisional alignment of supported and supporting systems is maintained and the design basis cooling capacity of the RCW divisions is assured.

† Equipment does not receive RCW in this mode.

‡ Equipment receives RCW in this mode. Rm. 436

ƒ HECW refrigerators; SGTS and FCS room coolers; room coolers, motor bearing coolers, and mechanical seal coolers for RHR and HPCF.

~~2.14.8 Flammability Control System~~

~~*Design Description*~~

~~The Flammability Control System (FCS) is provided to control the potential buildup of hydrogen and oxygen in the containment from radiolysis of water after a design basis loss of coolant accident (LOCA). The system consists of two independent and redundant hydrogen and oxygen recombiners. Cooling water required for operation of the system after a LOCA is taken from the Residual Heat Removal (RHR) System. Figure 2.14.8 shows the basic system configuration and scope.~~

~~The FCS is classified as safety related.~~

~~After a LOCA, the system can be manually actuated from the main control room if high oxygen concentrations exist in the primary containment. Each recombiner removes gas from the drywell, recombines the oxygen with hydrogen, and returns the gas mixture, along with the condensate to the wetwell.~~

~~The system is classified as Seismic Category I. Figure 2.14.8 shows ASME Code class for the FCS piping and components.~~

~~The FCS is located in the Reactor Building.~~

~~Each of the two FCS divisions is powered from the respective Class 1E division as shown on Figure 2.14.8. In the FCS, independence is provided between Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.~~

~~Each mechanical division of the FCS (Divisions B and C) is physically separated from the other division.~~

~~The FCS has the following displays and controls in the main control room:~~

~~(1) Controls and status indication for the valves shown on Figure 2.14.8.~~

~~(2) Controls and status indication for the recombiner unit.~~

~~FCS components with display and control interfaces with the Remote Shutdown System (RSS) is shown on Figure 2.14.8.~~

~~The safety-related electrical equipment shown on Figure 2.14.8, and included in the recombiner units, is qualified for a harsh environment.~~

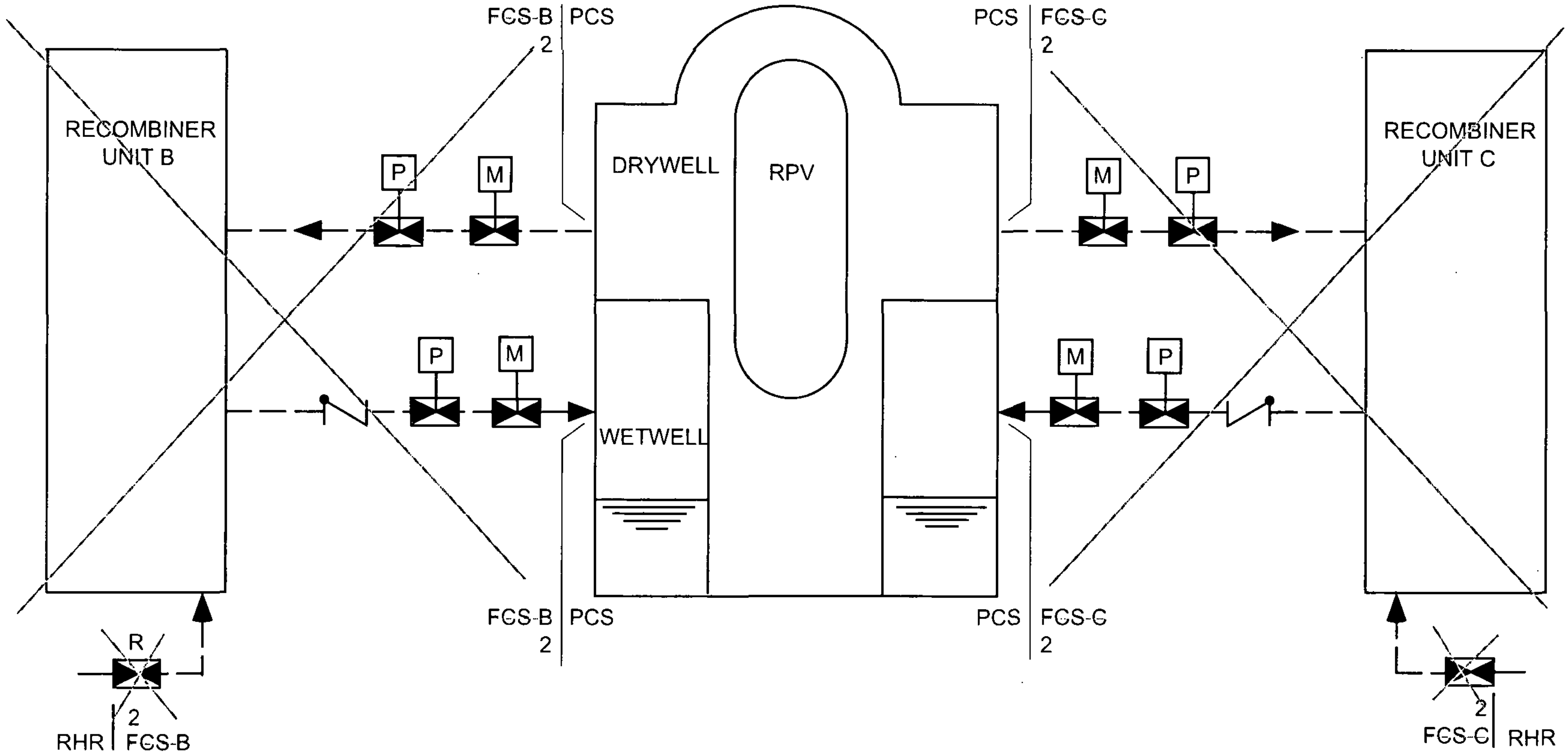
~~The motor-operated valves (MOVs) shown on Figure 2.14.8 and active safety-related MOVs in the recombiners, if any, have active safety related functions to both open and close, and perform these functions under differential pressure, fluid flow, and temperature conditions.~~

~~The check valves (CVs) shown on Figure 2.14.8 have active safety-related functions to both open and closer under system pressure, fluid flow, and temperature conditions.~~

~~The pneumatic valves shown on Figure 2.14.8 fail to the closed position in the event of loss of pneumatic pressure or loss of electrical power to the valve actuating solenoids.~~

~~*Inspections, Tests, Analyses and Acceptance Criterial*~~

~~Table 2.14.8 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the FCS.~~



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~~NOTE:~~
~~1. CLASS 1E ELECTRICAL POWER FOR FCS UNIT B IS SUPPLIED FROM DIVISION II EXCEPT FOR THE PNEUMATIC ISOLATION VALVE DUAL SOLENOIDS, WHICH IS DIVISIONS I AND III. UNIT C IS SUPPLIED FROM DIVISION III EXCEPT FOR THE OUTBOARD PNEUMATIC ISOLATION VALVE DUAL SOLENOIDS, WHICH ARE DIVISION I AND II.~~

~~Figure 2.14.8 Flammability Control System~~

~~Table 2.14.8 Flammability Control System~~

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The basic configuration for the FCS is as shown on Figure 2.14.8.	1. Inspections of the as-built system will be conducted.	1. The as-built FCS conforms with the basic configuration shown on Figure 2.14.8.
2. The ASME Code components of the FCS retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A pressure test will be conducted on those Code components of the FCS required to be pressure tested by the ASME code.	2. The results of the pressure test of the ASME code components of the FCS conform with the requirements in the ASME Code, Section III.
3. Each of the two FCS divisions is powered from the respective Class 1E division as shown on Figure 2.14.8. In the FCS, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	3. a. Tests will be performed in the FCS by providing a test signal in only one Class 1E division at a time. b. Inspection of the as-installed Class 1E divisions in the FCS will be performed.	3. a. The test signal exists only in the Class 1E division under test in the FCS. b. Physical separation or electrical isolation exists between Class 1E divisions in the FCS. Physical separation or electrical isolation exists between Class 1E divisions and non-Class 1E equipment in the FCS.
4. Each mechanical division of the FCS (Divisions B, C) is physically separated from the other divisions.	4. Inspections of the as-built FCS will be conducted.	4. Each mechanical division of the FCS is physically separated from the other mechanical divisions of FCS by structural and/or fire barriers.
5. Main control room displays and controls provided for the FCS are as defined in Section 2.14.8.	5. Inspections will be performed on the main control room displays and controls for the FCS.	5. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.14.8.
6. RSS display and control provided for the FCS are as defined in Section 2.14.8.	6. Inspections will be performed on the RSS display and control for the FCS.	6. Display and control exists on the RSS as defined in Section 2.14.8.
7. MOVs designated in Section 2.14.8 as having an active safety-related function open and close under differential pressure and fluid flow and temperature conditions.	7. Tests of installed valves for both opening and closing will be conducted under preoperational differential pressure, fluid flow, and temperature conditions.	7. Upon receipt of the actuating signal, each MOV both opens and closes, depending on the valve's safety function.

~~Table 2.14.8 Flammability Control System (Continued)~~

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
8. CVs designated in Section 2.14.8 as having an active safety-related function open and close under system pressure, fluid flow, and temperature conditions.	8. Tests of installed valves for both opening and closing will be conducted under preoperational system pressure, fluid flow, and temperature conditions.	8. Based on the direction of the differential pressure across the valve, each CV opens or closes depending upon the valve's safety functions.
9. The pneumatic valves shown on Figure 2.14.8 fail close in the event of loss of pneumatic pressure or loss of electrical power to the valve actuating solenoid.	9. Tests will be conducted on the as-built FCS pneumatic valves.	9. The pneumatic valves shown on Figure 2.14.8 fail close in the event of loss of pneumatic pressure or loss of electrical power to the valve actuating solenoid.

(3) R/B Safety-Related Diesel Generator HVAC System.

The Reactor Building HVAC System includes the following non-safety-related systems:

(1) R/B Secondary Containment HVAC System.

(2) R/B Primary Containment Supply/Exhaust System.

(3) R/B Main Steam Tunnel HVAC System.

(4) R/B Non-Safety-Related Equipment HVAC System.

(5) R/B Reactor Internal Pump (RIP) Adjustable Speed Drive (ASD) Control Panel HVAC System

R/B Safety-Related Equipment HVAC System

The R/B Safety-Related Equipment HVAC System provides cooling of safety-related equipment areas, and consists of independent fan coil units. Figure 2.15.5e shows the basic system configuration and scope.

The R/B Safety-Related Equipment HVAC System is classified as safety-related.

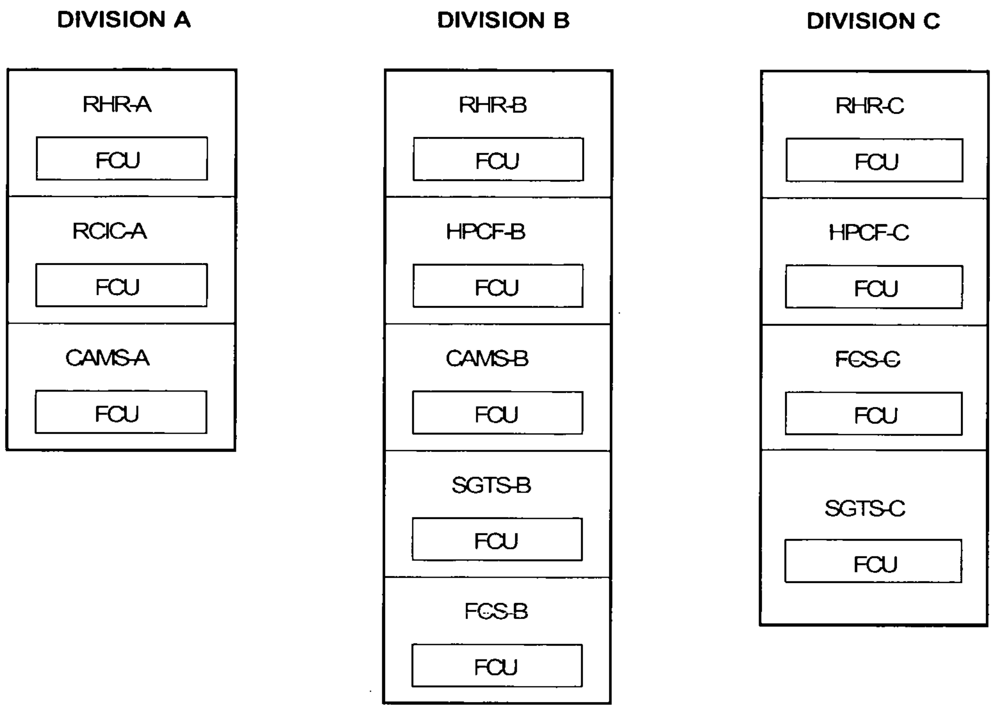
The Residual Heat Removal (RHR) System, High Pressure Core Flooder (HPCF) System and Reactor Core Isolation Cooling (RCIC) System pump room FCUs are automatically initiated upon startup of their respective room process pump. The Containment Atmospheric Monitoring System (CAMS) and Standby Gas Treatment System (SGTS) room FCUs are automatically initiated upon isolation of the Reactor Building Secondary Containment HVAC System.~~The Flammability Control System (FCS) room FCUs are also initiated upon a manual FCS start signal.~~

The temperature in the safety-related equipment areas is maintained below 40°C, except for the RHR, HPCF, and RCIC pump rooms, which are maintained below 66°C during pump operation.

The R/B Safety-Related Equipment HVAC System is classified as Seismic Category I. The R/B Safety-Related Equipment HVAC System is located in the Reactor Building.

Each of the three divisions of the R/B Safety-Related Equipment HVAC System is powered from the respective Class 1E division as shown on Figure 2.15.5e. In the R/B Safety-Related Equipment HVAC System, independence is provided between Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

Each mechanical division (Divisions A, B, C) of the R/B Safety-Related Equipment HVAC System is physically separated from the other divisions.



NOTES:

1. FCU COOLING WATER IS SUPPLIED BY THE RCW SYSTEM.
2. NORMAL VENTILATION AND SMOKE REMOVAL IS PROVIDED BY THE R/B SECONDARY CONTAINMENT HVAC SYSTEM.
3. ELECTRICAL POWER LOADS FROM DIVISIONS A, B, AND C ARE POWERED FROM CLASS 1E DIVISIONS I, II, AND III, RESPECTIVELY.

Figure 2.15.5e Reactor Building Safety-Related Equipment HVAC System

Table 2.15.5c Reactor Building Safety-Related Equipment HVAC System

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The basic configuration of the R/B Safety-Related Equipment HVAC System is as shown on Figure 2.15.5e.	1. Inspections of the as-built system will be conducted.	1. The as-built R/B Safety-Related Equipment HVAC System conforms with the basic configuration as shown on Figure 2.15.5e.
2. The RHR, HPCF, and RCIC pump room FCUs are automatically initiated upon start-up of their respective room process pumps.	2. Tests will be conducted on each pump room FCU using simulated signals indicating pump start-up.	2. Each pump room FCU starts when a signal indicates start-up of their respective room process pump.
3. The CAMS and SGTS room FCUs are automatically initiated upon isolation of the R/B Secondary Containment HVAC System.	3. Tests will be conducted on each as-built safety-related FCUs using simulated signals indicative isolation of the R/B Secondary Containment HVAC System.	3. The CAMS and SGTS room FCUs are automatically initiated upon isolation of the R/B Secondary Containment HVAC System.
~~4. The FCS room FCUs are initiated upon a manual FCS start signal.~~	~~4. Tests will be conducted on each as-built FCS room FCU using a simulated initiation signal.~~	~~4. The FCS room FCU starts upon receipt of a signal indicating FCS start.~~
5. Each of the three division of the R/B Safety-Related Equipment HVAC System is powered from the respective Class 1E division as shown on Figure 2.15.5e. In the R/B Safety-Related Equipment HVAC System, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	5. a. Tests will be performed on the R/B Safety-Related Equipment HVAC System by providing a test signal in only one Class 1E division at a time. b. Inspection of the as-built Class 1E divisions in the R/B Safety-Related Equipment HVAC System will be performed.	5. a. The test signal exists only in the Class 1E division under test in the in the R/B Safety-Related Equipment HVAC System. b. In the R/B Safety-Related Equipment HVAC System, physical separation or electrical isolation exists between Class 1E divisions. Physical separation or electrical isolation exists between these Class 1E divisions and non-class 1E equipment.

Figure 2.15.10a Reactor Building Arrangement—Section A-A

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

Figure 2.15.10j Reactor Building Arrangement, Floor 1F—Elev. 12300 mm

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

Figure 3.2i Reactor Building Radiation Zone Map for Full Power and Shutdown Operations, Floor 1F—Elevation 12300 mm

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

except the oxygen / hydrogen monitoring equipment in CAMS is non-safety-related.

2.3.3 Containment Atmospheric Monitoring System

Design Description

The Containment Atmospheric Monitoring System (CAMS) is used for post-accident monitoring of the primary containment. The purpose of the CAMS is to:

(1) Provide information on combustible levels of oxygen and hydrogen in the primary containment.

(2) Detect and measure the radiation level within the primary containment during and following an accident.

(3) Detect and measure the hydrogen concentration within the primary containment during and following an accident.

the oxygen / hydrogen monitoring equipment

The system monitors the atmospheric conditions in the drywell and in the suppression chamber for radiation levels and for hydrogen and oxygen gas concentration levels, displays the measurements in the main control room (MCR), and activates alarms in the MCR upon detection of high levels of radiation and/or gas concentrations.

The CAMS consists of two independent divisions and each division is composed of two radiation channels and oxygen/hydrogen gas monitoring equipment.

The CAMS is classified as a Class 1E safety-related system.

f radiation
hannels

Operation of ~~each CAMS division~~ can be activated manually or automatically during a post-accident condition by a signal indicating a high drywell pressure or a low reactor water level.

One radiation channel of each CAMS division monitors the radiation level in the drywell and the other channel monitors the radiation level in the suppression chamber.

quipment is

The oxygen/hydrogen monitoring equipment ~~of each CAMS division~~ analyzes the hydrogen and oxygen gas concentration levels in the drywell or in the suppression chamber and provides separate gas concentration displays in the MCR.

Each CAMS division is powered from its respective divisional Class 1E power source. In the CAMS, independence is provided between the Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

~~Both~~ CAMS ~~divisions are~~ located in the Reactor Building, except for the radiation and the gas process monitors, which are located in the Control Building.

The CAMS has the following alarms, displays, and controls in the MCR:

(1) Displays of radiation, hydrogen and oxygen levels.

or automatic

(2) Alarms for radiation levels, and for hydrogen and oxygen gas concentration levels.

(3) Manual ~~system-level~~ initiation for ~~each CAMS division~~.

of oxygen / hydrogen monitoring equipment

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.3.3 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Containment Atmospheric Monitoring System.

oxygen/hydrogen monitoring equipment

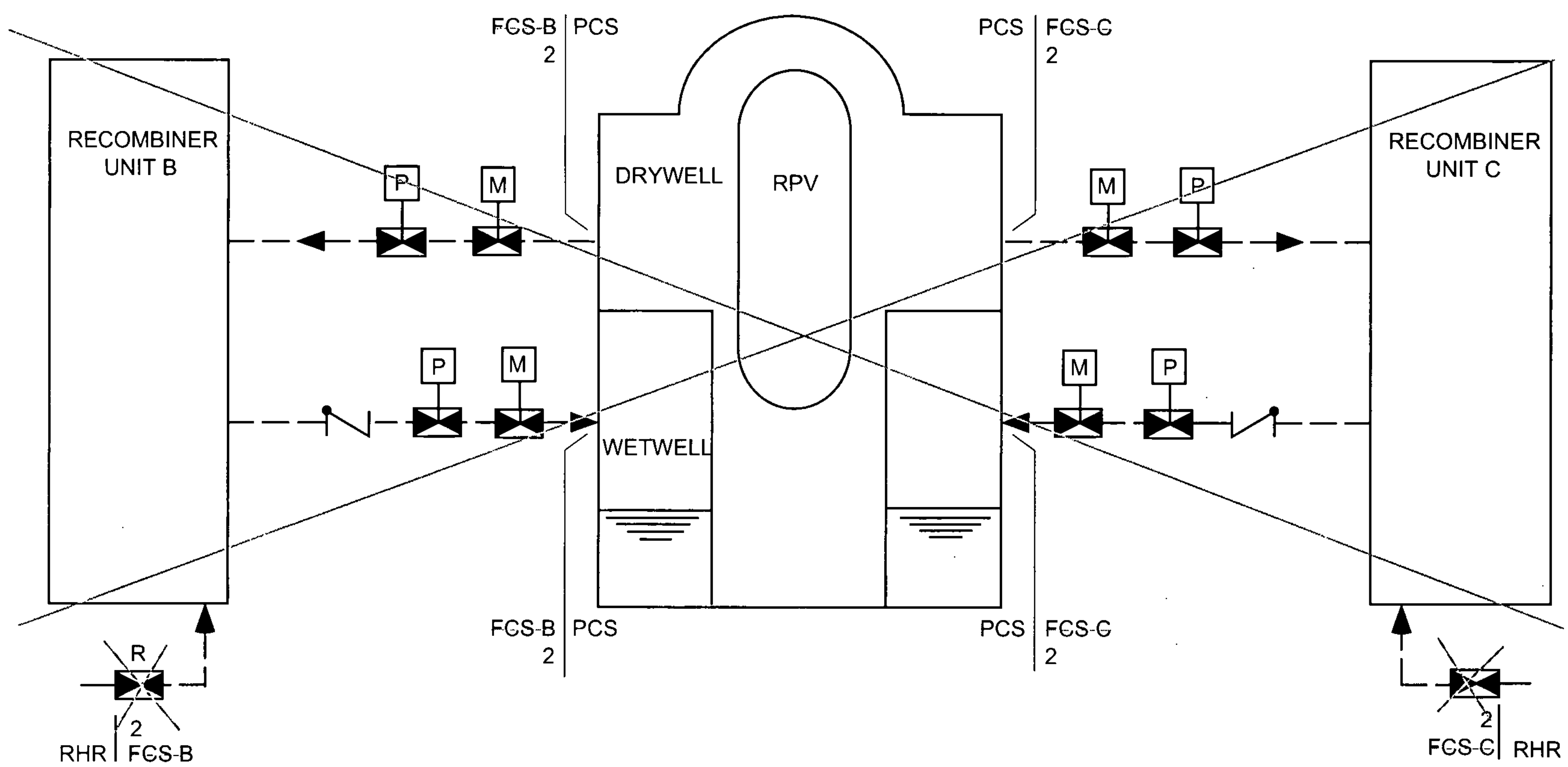
oxygen/hydrogen monitoring equipment

oxygen/hydrogen monitoring equipment

3 C... oring System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the CAMS is defined in Section 2.3.3.	1. Inspection of the as-built system will be conducted.	1. The as-built CAMS conforms with the description in Section 2.3.3.
2. Operation of ~~each~~ CAMS ~~division~~ can be activated manually by the operator or automatically.	2. Tests ~~of each division~~ of the as-built CAMS will be conducted using manual controls and simulated automatic initiation signals.	2. Each CAMS ~~division~~ is activated upon receipt of the test signals.
of radiation channels	each of	
3. Each CAMS division is powered from its respective divisional Class 1E power source. In the CAMS, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	3. a. Tests will be performed on the CAMS by providing a test signal to only one Class 1E division at a time. b. Inspection of the as-built Class 1E ~~divisions in the CAMs~~ will be performed. radiation channels	3. a. The test signal exists only in the Class 1E division under test in the CAMS. b. In the CAMS, physical separation or electrical isolation exists between Class 1E divisions. Physical separation or electrical isolation exists between these Class 1E divisions and non-Class 1E equipment.
4. Main control room alarms, displays and controls provided for the CAMS are as defined in Section 2.3.3.	4. Inspections will be performed on the main control room alarms, displays and controls for the CAMS.	4. Alarms, displays and controls exist or can be retrieved in the main control room as defined in Section 2.3.3.



~~NOTE:~~
~~1. CLASS 1E ELECTRICAL POWER FOR FCS UNIT B IS SUPPLIED FROM DIVISION II EXCEPT FOR THE PNEUMATIC ISOLATION VALVE DUAL SOLENOIDS, WHICH IS DIVISIONS I AND III. UNIT C IS SUPPLIED FROM DIVISION III EXCEPT FOR THE OUTBOARD PNEUMATIC ISOLATION VALVE DUAL SOLENOIDS, WHICH ARE DIVISION I AND II.~~

Figure 2.14.8 Not Used

Appendix C

ABWR DCD Significant Tier 2 Marked Changes

List of Acronyms (Continued)

ECLL	Electric Room Combustible Loading Limit
ECP	Electrochemical Potential or Engineering Computer Program
EDG	Emergency Diesel Generator
EDM	Electrodischarge Machining
EHC	Electrohydraulic Control
EMC	Electromagnetic Compatibility
EMI	Electromagnetic Interference
EMS	Essential Multiplexing System
EOEC	End of Equilibrium Cycle
EOF	Emergency Operations Facility
EPD	Electric Power Distribution
EPFM	Elastic-Plastic Fracture Mechanics
EPG	Emergency Procedure Guideline
EPRI	Electrical Power Research Institute
EPZ	Emergency Planning Zone
EQD	Enviromental Qualification Document
ESD	Electrostatic Discharge
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
ESS	Extraction Steam System
ESW	Essential Service Water
ETA	Event Tree Analysis
ETS	Emergency Trip System
F/D	Filter-Demineralizer
FATT	Fracture Appearance Transition Temperature
FCS	Feedwater Control System
~~FCS~~	~~Flammability Control System~~
FCU	Fan Coil Unit
FCV	Flow Control Valve
FDA	Final Design Approval
FDSA	Filled Drum Stock Area
FDWC	Feedwater Control
FHA	Fuel Handling Accident
FHB	Fuel Handling Building
FIV	Flow-Induced Vibration
FIVE	Fire Induced Vulnerability Evaluation

FCS	~~Flammability Control System~~
FCU	Fan Coil Unit
FCV	Flow Control Valve
FDA	Final Design Approval
FDSA	Filled Drum Stock Area
FDWC	Feedwater Control
FHA	Fuel Handling Accident
FHB	Fuel Handling Building
FIV	Flow-Induced Vibration
FIVE	Fire Induced Vulnerability Evaluation
FMCRD	Fine Motion Control Rod Drive
FMDC	Fine Motion Driver Cabinet
FMEA	Failure Mode and Effects Analysis
FN	Ferrite Number
FPC	Fuel Pool Cooling and Cleanup
FPS	Fire Protection System
FPS	Freeze Protection System

H2 & O2 monitors are no longer required to mitigate design-basis accidents

1.2.2.4.3 Containment Atmospheric Monitoring System

The Containment Atmospheric Monitoring System (CAMS) measures, records and alarms the radiation levels and the oxygen and hydrogen concentration levels in the primary containment under post-accident conditions. ~~It is automatically put in service upon detection of LOCA conditions.~~

1.2.2.5 Core Cooling System

In the event of a breach in the RCPB that results in a loss of reactor coolant, three independent divisions of ECCS are provided to maintain fuel cladding below the temperature limit as defined by 10CFR50.46. Each division contains one high pressure and one low pressure inventory makeup system.

1.2.2.5.1 Residual Heat Removal System

The Residual Heat Removal (RHR) System is a system of pumps, heat exchangers, and piping that fulfills the following functions:

(1) Removes decay and sensible heat during and after plant shutdown.

(2) Injects water into the reactor vessel following a LOCA to reflood the core in conjunction with other core cooling systems (Subsection 5.5.1).

(3) Removes heat from the containment following a LOCA to limit the increase in containment pressure. This is accomplished by cooling and recirculating the suppression pool water by containment sprays.

1.2.2.5.1.1 Low Pressure Flooder

Low pressure flooding is an operating mode of each RHR system, but is discussed here because the low pressure flooder (LPFL) mode acts in conjunction with other injection systems. LPFL uses the RHR pump loops to inject cooling water into the pressure vessel. LPFL operation provides the capability of core flooding at low vessel pressure following a LOCA in time to maintain the fuel cladding below the prescribed temperature limit.

1.2.2.5.1.2 Residual Heat Removal System Containment Cooling

The RHR System is placed in operation to: (1) limit the temperature of the water in the suppression pool and the atmospheres in the drywell and suppression chamber following a design basis LOCA; (2) control the pool temperature during normal operation of the safety/relief valves and the RCIC System; and (3) reduce the pool temperature following an isolation transient. In the containment cooling mode of operation, the RHR main system pumps take suction from the suppression pool and pump the water through the RHR heat exchangers, where cooling takes place by transferring heat to the service water. The fluid is then discharged back either to the

~~(10) Isolates the flammability control system lines~~

(11) Isolates the drywell sumps drain lines

(12) Isolates the fission products monitor sampling and return lines

(13) Initiates withdrawal of the automated traversing incore probe

In addition to the above functions, LDS monitors leakage inside the drywell from the following sources and annunciates the abnormal leakage levels in the control room:

(1) Fission products releases

(2) Condensate flow from the drywell air coolers

(3) Drywell sump level changes

(4) Leakages from valve stems equipped with leak-off lines

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Other leakages from the FMCRDs, the SRVs and from the reactor vessel head seal flange are monitored by their respective systems.

1.2.2.5.4 Reactor Core Isolation Cooling System

The RCIC System provides makeup water to the reactor vessel when the vessel is isolated and is also part of the emergency core cooling network. The RCIC System uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel for events defined in Section 5.4.

One division contains the RCIC System, which consists of a steam-driven turbine which drives a pump assembly and the turbine and pump accessories. The system also includes piping, valves, and instrumentation necessary to implement several flow paths. The RCIC steam supply line branches off one of the main steamlines (leaving the RPV) and goes to the RCIC turbine with drainage provision to the main condenser. The turbine exhausts to the suppression pool with vacuum breaking protection. Makeup water is supplied from the condensate storage tank (CST) or the suppression pool with preferred source being the CST. RCIC pump discharge lines include the main discharge line to the feedwater line, a test return line to the suppression pool, a minimum flow bypass line to the suppression pool and a cooling water supply line to auxiliary equipment.

Following a reactor scram, steam generation in the reactor core continues at a reduced rate due to the core fission product delay heat. The turbine condenser and the feedwater system supply the makeup water required to maintain reactor vessel inventory.

1.2.2.15.5 PCV Pressure and Leak Testing Facility

The PCV pressure and leak testing facility is a special area just outside the containment. It provides instrumentation for conducting the PCV pressure and integrated leak rate tests.

1.2.2.15.6 Atmospheric Control System

The Atmospheric Control System is designed to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during plant shutdown for refueling or maintenance.

The Atmospheric Control System is summarized in Subsection 6.2.5.2.1.

1.2.2.15.7 Drywell Cooling System

The Drywell Cooling System is summarized in Subsection 9.4.9.2.

~~1.2.2.15.8 Flammability Control System~~

~~A recombiner system is provided to control the concentration of hydrogen and oxygen produced by metal-water reaction and radiolysis following a design basis accident in the primary containment.~~

1.2.2.15.9 Suppression Pool Temperature Monitoring System

The Suppression Pool Temperature Monitoring (SPTM) System is summarized in Subsection 7.6.1.7.1.

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1.2.2.16 Structures and Servicing Systems

1.2.2.16.1 Foundation Work

The analytical design and evaluation methods for the containment and Reactor Building walls, slabs and foundation mat and foundation soil are summarized in Subsection 3.8.1.4.1.1.

1.2.2.16.2 Turbine Pedestal

The description for the turbine pedestal is the same as that for foundation work in Subsection 3.8.1.4.1.1.

1.2.2.16.3 Cranes and Hoists

The cranes and hoists are summarized in Subsection 9.1.

Figure 1.2-8 Reactor Building, Arrangement Plan at Elevation 12300mm

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Table 1.7-1 Piping and Instrumentation and Process Flow Diagrams

Tier 2 Fig. No.	Title	Type
4.6-8	CRD System	P&ID
4.6-9	CRD System	PFD
5.1-3	Nuclear Boiler System	P&ID
5.4-4	Reactor Recirculation System	P&ID
5.4-5	Reactor Recirculation System	PFD
5.4-8	Reactor Core Isolation Cooling System	P&ID
5.4-9	Reactor Core Isolation Cooling System	PFD
5.4-10	Residual Heat Removal System	P&ID
5.4-11	Residual Heat Removal System	PFD
5.4-12	Reactor Water Cleanup System	P&ID
5.4-13	Reactor Water Cleanup System	PFD
6.2-39	Atmospheric Control System	P&ID
~~6.2-40~~	~~Flammability Control System~~	~~P&ID~~
6.3-1	High Pressure Core Flooder System	PFD
6.3-7	High Pressure Core Flooder System	P&ID
6.5-1	Standby Gas Treatment System	P&ID
6.7-1	High Pressure Nitrogen Gas Supply System	P&ID
9.1-1	Fuel Pool Cooling and Cleanup System	P&ID
9.1-2	Fuel Pool Cooling and Cleanup System	PFD
9.2-1	Reactor Building Cooling Water System	P&ID
9.2-2	HVAC Normal Cooling Water System	P&ID
9.2-3	HVAC Emergency Cooling Water System	P&ID
9.2-4	Makeup Water System (Condensate)	P&ID
9.2-5	Makeup Water System (Purified)	P&ID
9.2-7	Reactor Service Water System	P&ID
9.3-1	Standby Liquid Control System	P&ID
9.3-1A	Standby Liquid Control System	PFD
9.3-6	Instrument Air System	P&ID
9.3-7	Service Air System	P&ID
9.4-1	Control Building HVAC	PFD
9.4-8	Drywell Cooling System	P&ID
9.5-1	Suppression Pool Cleanup System	P&ID

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	0	11/70	Yes	
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors	2	6/74	Yes	
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors	2	6/74	No	PWR only
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steamline Break Accident for Boiling Water Reactors	0	3/71	Yes	This page was replaced by page C-168
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	0 3	3/71 3/07	Yes	
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	2	11/78	Yes	
1.8	Personnel Selection and Training	--	--	--	See Table 17.0-1
1.9	Selection, Design, Qualification, and Testing of Emergency Diesel-Generator Units Used As Class 1E Onsite Electric Power Systems at Nuclear Plants	3	7/93	Yes	
1.11	Instrument Lines Penetrating Primary Reactor Containment	0	3/71	Yes	
1.12	Instrumentation for Earthquakes	1	4/74	No	NA
1.13	Spent Fuel Storage Facility Design Basis	1	12/75	Yes	
1.14	Reactor Coolant Pump Flywheel Integrity	1	8/75	No	PWR only
1.16	Reporting of Operating Information — Appendix A Technical Specifications	4	8/75	---	COL Applicant
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	2	5/76	Yes	

probability of steamline flooding by ECCS is extremely low. There is no high drywell pressure signal that would inhibit this logic system.

In the ABWR design, each of three RHR shutdown cooling lines has its own separate containment penetration and its own separate source of suction from the reactor vessel. Alternate shutdown using the SRV is therefore not required for the ABWR in order to meet single failure rules. Hence, the ABWR does not require SRV testing with liquid under low pressure conditions associated with this event as required in past BWRs.

1A.2.10 Relief and Safety Valve Position Indication [II.D.3]

NRC Position

Reactor Coolant System relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

Response

The ABWR Standard Plant SRVs are equipped with position sensors which are qualified as Class 1E components. These are used to monitor valve position.

In addition, the downstream pipe from each valve line is equipped with temperature elements which signal the annunciator and the plant process computer when the temperature in the tailpipe exceeds the predetermined setpoint.

These sensors are shown on Figure 5.1-3 (Nuclear Boiler System P&ID).

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1A.2.11 Systems Reliability [II.E.3.2]

This TMI action plan item superseded by USI A-45. USI A-45 is addressed in Appendix 19B.

1A.2.12 Coordinated Study of Shutdown Heat Removal Requirements [II.E.3.3]

This TMI action plan item superseded by USI A-45. USI A-45 is addressed in Appendix 19B.

~~1A.2.13 Containment Design—Dedicated Penetration [II.E.4.1]~~

~~**NRC Position**~~

~~For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere.~~

~~**Response**~~

~~A Flammability Control System is provided to control the concentration of oxygen in the primary containment. The FCS utilizes two permanently installed recombiners~~

~~located in the secondary containment. The FCS is operable in the event of a single active failure. The FCS is described in Subsection 6.2.5.~~

1A.2.14 Containment Design—Isolation Dependability [II.E.4.2]

NRC Position

(1) Containment isolation system designs shall comply with the recommendations of the Standard Review Plan, Subsection 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and non-essential systems, identify each system determined to be non-essential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

(3) All nonessential systems shall be automatically isolated by the containment isolation signal.

(4) The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

(5) The containment setpoint pressure that initiates containment isolation for non-essential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, Item II.6.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

(7) Containment purge and vent isolation valves must close on a high radiation signal.

Response

(1) The isolation provisions described in the Standard Review Plan, Subsection 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation) were reviewed in conjunction with the ABWR Standard Plant design. It was determined that the ABWR Standard Plan is designed in accordance with these recommendations of the SRP.

1AA.3.2 Vital Area and Systems

A vital area is any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident. Areas which must be considered as vital after an accident are the control room, technical support center, sampling station, sample analysis area and the HPIN nitrogen supply bottles.

The vital areas also include consideration (in accordance with NUREG- 0737, II.B.2) of the ~~post-LOCA hydrogen control system,~~ containment isolation reset control area, manual ECCS alignment area, motor control center and radwaste control panels. However, the ABWR design does not require a containment isolation reset control area or a manual ECCS alignment area, as these functions are available from the control room. Those vital areas which are normally areas of mild environment, allowing unlimited access, are not reviewed for access.

Essential systems specific to the ABWR to be considered post-accident are those for the ECCS, fission product and ~~combustible gas~~ control and the auxiliary systems necessary for their operation (i.e., instrumentation, control and monitoring, power, cooling water, and air cooling).

1AA.3.3 Post Accident Operation

Post-accident operations are those necessary to (1) maintain the reactor in a safe shutdown condition, (2) maintain adequate core cooling, (3) assure containment integrity, and (4) control radioactive releases within 10CFR100 guidelines.

Many of the safety-related systems are required for reactor protection or to achieve a safe shutdown condition. However, they are not necessarily needed once a safe shutdown condition is achieved. Thus, the systems considered herein are the engineered safety features (ESF) (Chapter 6) used to maintain the plant in a safe shutdown condition.

For purposes of this review, the plant is assumed to remain in the safe shutdown condition.

The basis for this position is that the foundation of plant safety is the provision of sufficient redundancy of systems and logic to assure that the plant is shut down and that adequate core cooling is maintained. Necessary shutdown and post-accident operations are performed from the control room, except for the post-accident sampling station, the sample analysis area, and two manual nitrogen reserve supply valves.

safety-related, or included in the systems of Table 3.2-1. It does however, represent principal components which are needed to operate, generally during post accident operations. For example, most ECCS valves are normally open, and only a pump discharge valve needs to open to direct water to the reactor. Similarly, the instrument transmitters shown are those which would provide information on long-term system performance post-accident. Control room instrumentation is not listed, since it is all in an accessible area where no irradiation degradation would be expected. Passive elements such as thermocouples and flow sensors are not listed although they are environmentally qualified. The components listed under main steam (B21) are those for ECCS function or monitoring reactor vessel level. Suppression pool level is included with the HPCF instrumentation.

for

1AA.5.1.3 Combustible Gas Control Systems and Auxiliaries

during

Flammability control in the primary containment is achieved by an inert atmosphere during all plant operating modes except operator access ~~for~~ refueling and maintenance ~~and a recombiner system to control oxygen produced by radiolysis.~~ The high pressure nitrogen (HPIN) gas supply is described in Subsection 1.2.2.12.13. The Containment Atmospheric Monitoring System (CAMS) measures and records containment oxygen/hydrogen concentrations under post-accident conditions. It is automatically initiated by detection of a LOCA (Subsection 7.6.1.6). Table 1AA-3 lists the combustible gas control principal components and their locations.

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1AA.5.1.4 Fission Product Removal and Control Systems and Auxiliaries

Engineered Safety Feature (ESF) filter systems are the Standby Gas Treatment System (SGTS) and the control building Outdoor Air Cleanup System. Both consist of redundant systems designed for accident conditions and are controlled from the control room. The SGTS filters the gaseous effluent from the primary and secondary containment when required to limit the discharge of radioactivity to the environment. The system function is described in Subsection 1.2.2.15.4.

A portion of the Control Building heating ventilating and air-conditioning (HVAC) provides detection and limits the introduction of radioactive material and smoke into the control room. This portion is described Subsection 9.4.1.1.3.

The CAMS described in the previous section also measures and records containment area radiation under post-accident conditions. A post-accident sampling system (PASS) obtains containment atmosphere and reactor water samples for chemical and radiochemical analysis in the laboratory. Delayed sampling, shielding, remote operated valves and sample transporting casks are utilized to reduce radiation exposure. The samples are manually transported between the PASS room in the Reactor Building and the analysis laboratory in the service building. The system is described in

Table 1AA-3 Post-Accident Combustible Gas Control Systems and Auxiliaries

Equipment	MPL	Location
HPIN		
Nitrogen Storage Bottles	P54-A001A Thru V	By Valve Rm (RB)
Supply Pressure	P54-PT002A, B, 004, 005	By Valve Rm (RB)
FCS		
~~Recombiner & Auxiliaries~~	~~T49-A001A,B~~	~~(PC)~~
RHR ~~Cooling/Isol. Valve~~	~~T49-F008,010; A,B~~	~~(PC)(SC)~~
~~Flow~~	~~T49-FT002,004; A,B~~	~~Inst. Rack Rm. A,B (SC)~~
~~Pressure~~	~~T49-PT003A,B~~	~~Inst. Rack Rm. A,B (SC)~~
CAMS		
Hydrogen, Oxygen Elements	D23-H_2, O_2 Rack A,B	CAMS Rm. A,B (SC)
Gas Measurement	D23-Gas Cal. Rack A,B	CAMS Rm. A,B (SC)
Gas Elements	D23-Gas Cal. Rack A,B	CAMS Rm. A,B (SC)
DW Gas Valve	D23-F004A,B	CAMS Rm. A,B (SC)
WW Gas Valve	D23-F006A,B	CAMS Rm. A,B (SC)
Essential HVH (HVAC)	U41-D113,114	CAMS Rm. A,B (SC)
Gas Supply	D23-Gas Cyl. Rack A,B	CAMS Rm. A,B (RB)

(PC)—Primary Containment
(SC)—Secondary Containment
(RB)—Reactor Building outside (Secondary Containment)

Table 3.2-1 Classification Summary (Continued)

The classification information is presented by System* in the following order:

Item No.	MPL Number†	Title
R10	R35	Electrical Wiring Penetration
R11	R40	Combustion Turbine Generator
R12	R42	Direct Current Power Supply‡
R13	R43	Emergency Diesel Generator System‡
R14	R46	Vital AC Power Supply
R15	R47	Instrument and Control Power Supply
R16	R51	Communication System
R17	R52	Lighting and Servicing Power Supply
S Power Transmission Systems		
S1	S12	Reserve Auxiliary Transformer
T Containment and Environmental Control Systems		
T0	T10	Primary Containment System
T1	T11	Primary Containment Vessel
T2	T12	Containment Internal Structures
T3	T13	Reactor Pressure Vessel Pedestal
T4	T22	Standby Gas Treatment System‡
T5	T25	PCV Pressure and Leak Testing Facility
T6	T31	Atmospheric Control System
T7	T41	Drywell Cooling System
~~T8~~	~~T49~~	~~Flammability Control System~~
T9	T53	Suppression Pool Temperature Monitoring System‡
U Structures and Servicing Systems		
U1	U21	Foundation Work

* Systems that are in and out of the ABWR Standard Plant scope are included in this table. See Subsection 1.1.2 for the identification of the site-specific elements outside the scope of the ABWR Standard Plant.

† Master Parts List Number designated for the system.

‡ These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety-related components and, conversely, some systems whose primary functions are non-safety-related contain components that have been designated safety-related.

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
~~T8 Flammability Control System~~	~~2~~	~~SC~~	~~B~~	~~B~~	~~I~~	
T9 Suppression Pool Temperature Monitoring System						
1. Electrical modules with safety-related functions	3	C,X,SC, RZ	—	B	I	
2. Cable with safety-related functions	3	C,X,SC, RZ	—	B	I	
U1 Foundation Work	2/3	C,SC,RZ	—	B	I	
U2 Turbine Pedestal	N	T	—	E	—	
U3 Cranes and Hoists						
1. Reactor Building crane	N	SC	—	E	—	(x)
2. Refueling Platform	N	SC	—	E	—	(x)
3. Upper Drywell Servicing	N	C	—	E	I	
4. Lower Drywell Servicing	N	C	—	E	I	
5. Main Steam Tunnel Servicing	N	M	—	E	—	
6. Special Service Rooms	N	SC,RZ,T, W,X	—	E	—	
U4 Elevator	N	SC,RZ,X	—	E	—	

Notes and footnotes are listed on pages 3.2-54 through 3.2-61

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F025	6	Cooling water supply line to HECW refrigerator PCV	3	B	A	S	E2	9.2-1 sh. 2,5,8
F026	6	Cooling water supply line to HECW refrigerator maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F027	6	Cooling water line to HECW refrigerator bypass line	3	B	P		E1	9.2-1 sh. 2,5,8
F028	6	Cooling water return line from HECW refrigerator	3	B	P		E1	9.2-1 sh. 2,5,8
F029	2	Cooling water supply line to FPC Hx	3	B	P		E1	9.2-1 sh. 2,5
F030	2	Cooling water return line from FPC Hx	3	B	P		E1	9.2-1 sh. 2,5
F031	2	Cooling water supply line to FPC pump room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F032	2	Cooling water return line from FPC pump room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F033	2	Cooling water line to PCV Atmospheric Monitoring System clr	3	B	P		E1	9.2-1 sh. 2,5
F034	2	Return line from PCV Atmospheric Monitoring System clr	3	B	P		E1	9.2-1 sh. 2,5
F035	2	Cooling water supply line to SGTS room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F036	2	Cooling water return line from SGTS room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F037	2	Cooling water supply line to ~~FCS~~ Rm. 425/436 room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F038	2	Cooling water return line from ~~FCS~~ room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
Rm. 425/436	3	Cooling water supply line to RHR equipment room air conditioner	3	B	P		E1	9.2-1 sh. 2,5,8
F040	3	Cooling water return line from RHR equipment room air conditioner	3	B	P		E1	9.2-1 sh. 2,5,8
F041	3	Cooling water supply line to RHR pump motor	3	B	P		E1	9.2-1 sh. 2,5,8

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
F804	2	DW/WW differential pressure instrument line valve	2	B	P		E1	6.2-39 sh. 2
F805	2	DW/WW differential pressure instrument isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
D001	1	Wetwell overpressure rupture disk	2	D	I,P	Rplc.	5 yr	6.2-39 sh. 1
D002	1	Wetwell rupture disk	2	D	I,P	Rplc.	5 yr	6.2-39 sh. 1
~~T49 Flammability Control System Valves~~								
~~F001~~	2	~~Inlet line from drywell inboard isolation valve~~	~~2~~	A	~~I,A~~	~~L,P~~ S	~~2 yr 3 mo~~	~~6.2-40~~
~~F002~~	~~2~~	~~Inlet line from drywell outboard isolation valve~~	~~2~~	A	~~I,A~~	~~L,P~~ S	~~2 yr 3 mo~~	~~6.2-40~~
~~F003~~	~~2~~	~~Flow control valve for the FCS inlet line from drywell~~	~~3~~	~~B~~	A	~~P~~ S	~~2 yr 3 mo~~	~~6.2-40~~
~~F004~~	~~2~~	~~Blower bypass line flow control valve~~	~~3~~	~~B~~	A	~~P~~ S	~~2 yr 3 mo~~	~~6.2-40~~
~~F005~~	2	~~Blower discharge line to wetwell check valve (h9)~~	~~3~~	~~C~~	A	S	~~RO~~	~~6.2-40~~
~~F006~~	~~2~~	~~Discharge line to wetwell outboard isolation valve~~	~~2~~	A	~~I,A~~	~~L,P~~ S	~~2 yr 3 mo~~	~~6.2-40~~
~~F007~~	~~2~~	~~Discharge line to wetwell inboard isolation valve~~	~~2~~	A	~~I,A~~	~~L,P~~ S	~~2 yr 3 mo~~	~~6.2-40~~
~~F008~~	~~2~~	~~Cooling water supply line from the RHR System MOV~~	~~3~~	B	A	~~P~~ S	~~2 yr 3 mo~~	~~6.2-40~~
~~F009~~	~~2~~	~~Cooling water supply line maintenance valve~~	~~3~~	B	~~P~~		~~E1~~	~~6.2-40~~
~~F010~~	2	~~Cooling water supply line admission MOV~~	~~3~~	B	A	~~P~~ S	~~2 yr 3 mo~~	~~6.2-40~~
~~F013~~	2	~~Inlet line from drywell drain line valve~~	~~3~~	B	~~P~~		~~E1~~	~~6.2-40~~
~~F014~~	~~2~~	~~Blower drain line valve~~	~~3~~	B	~~P~~		~~E1~~	~~6.2-40~~
~~F015~~	~~1~~	~~Blower discharge line to wetwell pressure relief valve~~	~~2~~	~~A,C~~	~~I,A~~	R ~~L~~	~~5 yr RO~~	~~6.2-40~~
~~F016~~	~~2~~	~~Blower discharge line to wetwell pressure relief line check valve (h3)~~	~~2~~	~~A,C~~	~~I,A~~	~~L,S~~	~~RO~~	~~6.2-40~~

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
~~F501~~	~~2~~	~~Inlet line from drywell test line valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F502~~	~~2~~	~~Discharge line to wetwell test line valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F504~~	~~2~~	~~Blower suction line test line valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F505~~	~~2~~	~~Blower discharge line test line valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F506~~	~~2~~	~~Drain line to low conductivity waste (LCW) valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F507~~	~~2~~	~~Cooling water supply line test line valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F701~~	~~2~~	~~FE-T49-FE002 upstream instrument line root valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F702~~	~~2~~	~~FE-T49-FE002 downstream instrument line root valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F703~~	~~2~~	~~Blower suction line pressure instrument line root valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F704~~	~~2~~	~~FE-T49-FE004 upstream instrument line root valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F705~~	~~2~~	~~FE-T49-FE004 downstream instrument line root valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
U41 Heating, Ventilating and Air Conditioning System Valves								
F001	2	Secondary containment supply isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F002	2	Secondary containment exhaust isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F003	3	Secondary Containment divisional supply isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F004	3	Secondary Containment divisional exhaust isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F007	4	MCR area HVAC bypass line isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2
F008	4	MCR area HVAC supply isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2
F009	4	MCR area HVAC emergency HVAC supply	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2

Figure 3H.4-7 Location of Walls Exposed to HELB, El. 12300 mm

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

Table 3I-13 Thermodynamic Environment Conditions Inside Reactor Building (Secondary Containment) Plant Accident Conditions[1] (Continued)

Plant Zone/Typical Equipment		Time[2] 1 (h)	6 (h)	12 (h)	100 (day)
FPC (cooling system, SPCU [makeup water system] valve, pump motor, heat exchanger, instrument, control electric equipment) cable sources of electricity, pipe spaces [Figs. 1.2-9/9.1-1]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
CUW (Pump, valve, non-regen. and regen. heat exchangers, pipe spaces, filter demin filter demin. valve rooms) corridor [Figs. 1.2-4/5.4-12]	Temperature (°C) Pressure (kPaG) Humidity (%)	120 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
~~FCS~~[6] ~~valves including isolation valve (recombiner instrument, controls), electrical equipment (power source cables)~~ [~~Figs. 1.2-8/6.2-40~~]	~~Temperature (°C)~~ ~~Pressure (kPaG)~~ ~~Humidity (%)~~	~~120~~ ~~102.97~~[3] Steam	~~120~~ ~~102.97~~[3] ~~Steam~~	66 ~~3.43~~ ~~100~~	66 ~~0~~ ~~90 Max~~
Mainsteam Tunnel (Outside secondary containment)					
MS isolation valve[5] MS drain isolation valve Nitrogen line isolation valve[5,6] Process water line isolation valve[5,6] [Figs 1.2-2, 1.2-3, 1.2-3a, 5.1-3]	Temperature (°C) Pressure (kPaG) Humidity (5)	171 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
Feedwater isolation valve [5] [Figs 1.2-2, 1.2-3, 1.2-3a/5.1-3]	Temperature (°C) Pressure (kPaG) Humidity (%)	171 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max
RCIC, check valve (inside MS tunnel) [Figs. 1.2-2, 1.2-3, 1.2-3a/5.4-8]	Temperature (°C) Pressure (kPaG) Humidity (%)	171 102.97[3] Steam	120 102.97[3] Steam	66 3.43 100	66 0 90 Max

1. Systems or components located in the Reactor Building outside the secondary containment or in other buildings and required to support the equipment listed in this table during accident condition will be qualified to the conditions specified in the equipment qualification design criteria table for the respective area or building.
2. Time means the time from the occurrence of LOCA.
3. The 102.97 kPaG equipment qualification pressure specified is the structural design basis for the respective rooms (see Subsection 6.2.3.3) in which this equipment is located and not the saturation pressure associated with the equipment qualification temperature.

- Makeup Water (Condensate) System upstream of the injection valve for the purpose of providing a filling and flushing water source. Another interface with MUWC is between the pair of valves to the FPC System. The MUWC System is discussed in Section 3MA.11, where it is explained how certain MUWC upgrades were made that provide an open path to the CST. The MUWC line cannot be pressurized because of the open communication to the CST, and the CST is vented to atmosphere. There is no source to pressurize the MUWC line because of closed valves in the RHR System's URS region.

- High Conductivity Waste (Radwaste) for drainage located up stream of the pump suction. HCW upgrades are discussed in the Radwaste System, Section 3MA.13.

- Low Conductivity Waste, (Radwaste) at the end of a branch off of the loop B mainline down stream of the RHR heat exchanger. The LCW upgrades are discussed in the Radwaste System, Section 3MA.13.

- Sampling System at the outlet of the RHR heat exchanger. The Sampling System's design pressure exceeds the URS design pressure without upgrade.

- Fuel Pool Cooling and Cleanup System on an RHR System discharge branch. FPC System upgrades are discussed in Section 3MA.8.

- ~~Flammability Control System branches off the main discharge line downstream of the branch that returns to the suppression pool. The FCS design pressure exceeds the URS design pressure without upgrade.~~

- The Fire Protection System and the fire truck connection provide water for the Alternating Current (AC) Independent Water Addition piping of RHR loop C upstream of the RPV injection, wetwell spray line, and drywell spray line. The Fire Protection System piping is designed for 1.37 MPaG and is protected from over pressure by two locked closed block and bleed valves, RHR-F101 and RHR-F102, and a drain pipe between these valves vented to the HCW sump in the Reactor Building. This design very effectively prevents reactor pressure from reaching the Fire Protection System. No upgrade to URS is practical or appropriate for the extensive piping of the Fire Protection System since the system function is not related to ISLOCA nor is its interconnection a normal plant operational pathway.

3MA.2.3 Upgraded Components — RHR System

A detailed listing of the components upgraded for the RHR System follows, including identification of those interfacing system components not requiring upgrade.

Table 5.2-6 LDS Control and Isolation Function vs. Monitored Process Variables

LDS Control & Isolation Functions \ Monitored Variables	Reactor Water Level Low	Turbine Inlet SL Press Low	Reactor Pressure High	MSL Flow Rate High	MSL Radiation High	MSL Tunnel Amb. Temp High	Turbine Area Amb. Temp High	Main Condenser Vacuum Low	Drywell Pressure High	RHR Equip Area Temp High	RCIC Equip Area Temp High	RCIC SL Pressure Low	RCIC SL Flow Rate High	RCIC Vent Exhaust Press High	CUW Equip Area Temp HIgh	CUW Differential Flow High	SLCS Pumps Running	LCW Drain Line Radiation High	HCW Drain Line Radiation High	R/B HVAC Exhaust Air Rad High	F/H Exhaust Air Rad High
MSIVs & MSL Drain Line Valves	L1.5	X		X	X	X	X	X													
CUW Process Lines Isolation	L2		X*			X									X	X	X				
RHR S/C PCV Valves	L3		X							X											
RCIC Steamline Isolation											X	X	X	X							
ATIP Withdrawal	L3								X												
DW RAD Sampling Isolation	L2								X												
SPCU Process Line Isolation	L3								X												
DW LCW Sump Drain Line Isolation	L3								X									X			
DW HCW Sump Drain Line Isolation	L3								X										X		
RCW PCV Valves Isolation	L1								X												
HNCW PCV Valves Isolation	L1								X												
AC System P&V Valves Isolation	L3								X											X	X
~~FCS PCV Valves Isolation~~	~~L3~~								~~X~~												
R/B HVAC Air Ducts Isolation	L3								X											X	X
SGTS Initiation	L3								X											X	X

* Head spray valve only

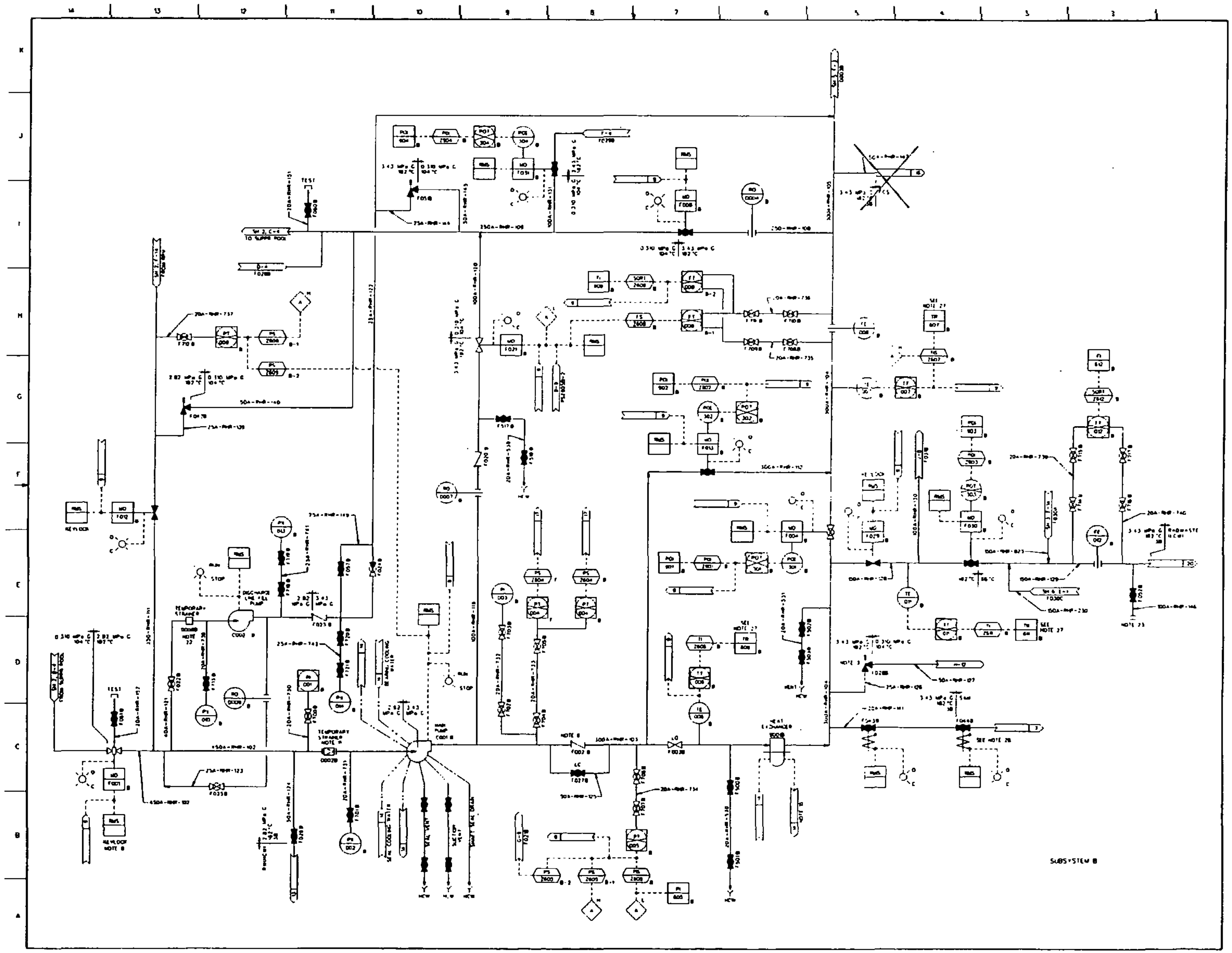

FIGURE 5.4-10 RESIDUAL HEAT REMOVAL SYSTEM P&ID (Sheet 4 of 7)
ABWR DCD/Tier 2 Rev. 0 21-106

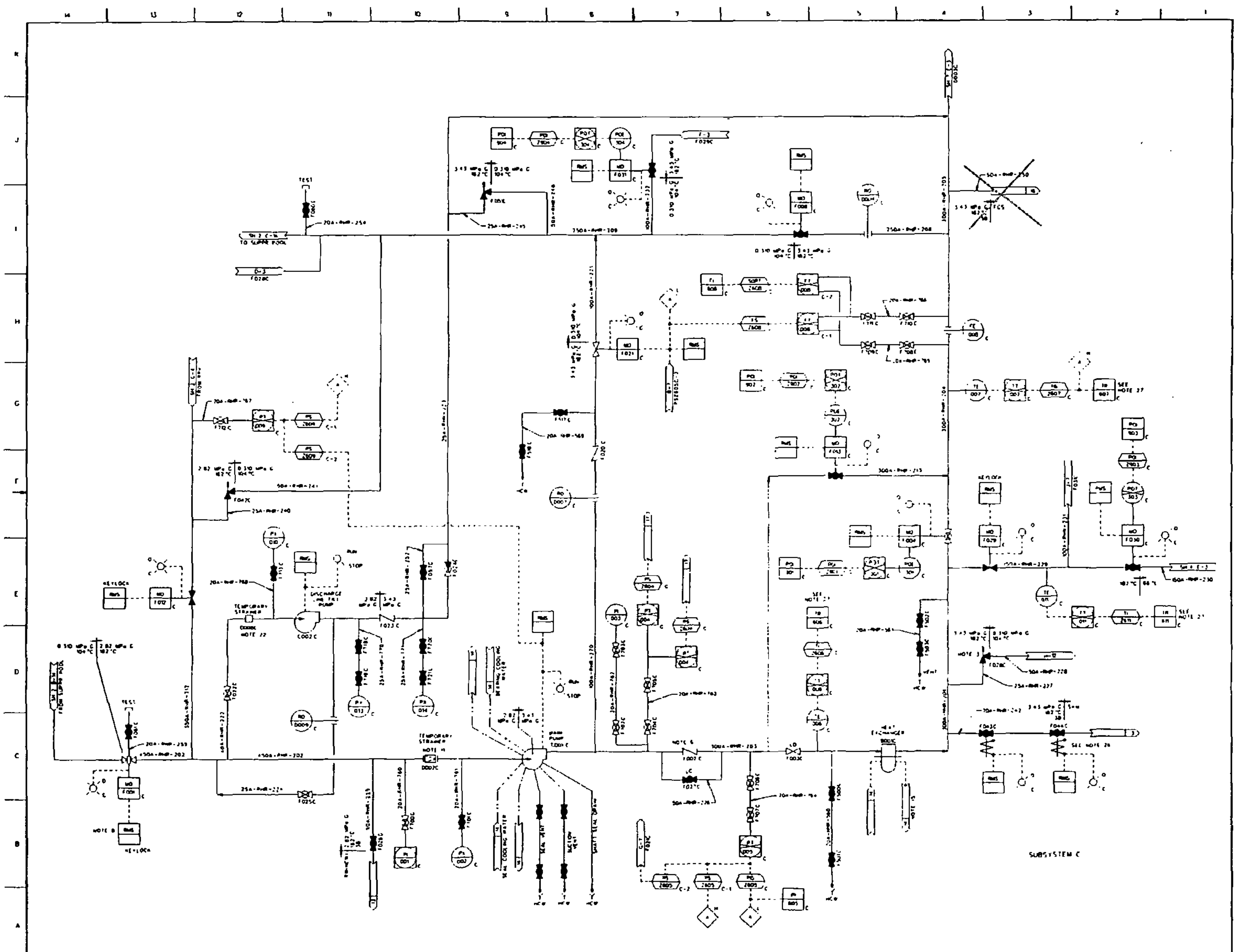

FIGURE 5.4-10 RESIDUAL HEAT REMOVAL SYSTEM P&ID (Sheet 6 of 7)

ABWR DCD/Tier 2 Rev. 0 21-108

List of Figures (Continued)

exception of the makeup valves (50A), all containment purge system CIVs are in closed position during normal reactor operation. The purge and vent valves are open only during the inerting and de-inerting modes. All containment purge system CIVs automatically close upon receipt of containment isolation signal. Also, these valves are outside containment and accessible should manual actuation be required. Since this arrangement has adequate redundancy, and independence and is not unduly vulnerable to common mode failures, it is not necessary to have redundant and independent CIVs as would be required by Criterion 54.

6.2.4.4 Test and Inspections

The Containment Isolation System is scheduled to undergo periodic testing during reactor operation. The functional capabilities of power-operated isolation valves are tested remote-manually from the control room. By observing position indicators and changes in the affected system operation, the closing ability of a particular isolation valve is demonstrated.

Air-testable check valves are provided on influent emergency core cooling lines of the HPCF and RHR Systems whose operability is relied upon to perform a safety function.

A discussion of testing and inspection of isolation valves is provided in Subsection 6.2.1.6. Instruments are periodically tested and inspected. Test and/or calibration points are supplied with each instrument. Leakage integrity tests shall be performed on the containment isolation valves with resilient material seals at least once every three months.

6.2.5 Combustible Gas Control in Containment

The Atmospheric Control System (ACS) is provided to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during shutdown for refueling or equipment maintenance and during limited periods of time to permit access for inspection at low reactor power. ~~The Flammability Control System (FCS) is provided to control the potential buildup of hydrogen and oxygen from design basis metal-water reaction and radiolysis of water. The objective of these systems is to preclude combustion of hydrogen causing damage to essential equipment and structures. The COL applicant is required to provide a comparison of costs and benefits for any optional alternate system of hydrogen control.~~

6.2.5.1 Design Bases

Since there is no design requirement for the ACS ~~or FCS~~ in the absence of a LOCA and since there is no design basis accident in the ABWR that results in core uncovery or fuel failures, the following requirements mechanistically assume that a LOCA producing the

design basis quantities of hydrogen and oxygen has occurred. Following are criteria that serve as the bases for design:

(1) The hydrogen generation from metal-water reaction is defined in Regulatory Guide 1.7.

(2) The hydrogen and oxygen generation from radiolysis is defined in Regulatory Guide 1.7.

(3) The ACS establishes an inert atmosphere throughout the primary containment following an outage or other occasions when the containment has been purged with air to an oxygen concentration greater than 3.5%.

(4) The ACS maintains the primary containment oxygen concentration below the maximum permissible limit per Regulatory Guide 1.7 during normal, abnormal, and accident conditions in order to assure an inert atmosphere.

(5) The ACS also maintains a slightly positive inert gas pressure in the primary containment during normal, abnormal and accident conditions to prevent air (oxygen) leakage into the inerted volumes from the secondary containment, and provides non-safety-related monitoring of the oxygen concentration in the primary containment to assure a breathable mixture for safe personnel access. Essential safety-related monitoring is provided by the Containment Atmospheric Monitoring System (CAMS), as described in Chapter 7.

(6) The drywell and the suppression chamber will be mixed uniformly after the design basis LOCA due to natural convection and molecular diffusion. Mixing will be further promoted by operation of the containment sprays.

~~(7) The FCS is capable of controlling combustible gas concentrations in the containment atmosphere for the design bases LOCA without relying on purging and without releasing radioactive material to the environment.~~

(8) The ACS ~~and FCS together are~~ is designed to maintain an inert primary containment after the design-bases LOCA, assuming a single-active failure. The backup purge function need not meet this criterion.

(9) Components of the ACS inside the Reactor Building are protected from postulated missiles and pipe whip, as required to assure proper action.

(10) The ACS has the capability to withstand the dynamic effects associated with a safe shutdown earthquake without loss of isolation function.

(11) The system is designed so that all components exposed to the primary containment atmosphere (i.e., inboard isolation valves) are capable of withstanding the temperature, humidity, pressure, and radiation transients resulting from a LOCA.

(12) The ACS is non-safety class except as necessary to assure primary containment integrity (penetrations, isolation valves). The ACS and ~~FCS are~~ designed and built to the requirements specified in Section 3.2. is

(13) The ACS includes a liquid nitrogen storage tank, vaporizer and heater along with the valves and piping carrying nitrogen to the containment, valves and piping from the containment to the SGTS and HVAC exhaust line, dedicated containment overpressure relief line with attached valves and rupture disk, non-safety oxygen monitoring, and all related instruments and controls. The ACS does not include any structures or housing supporting the aforementioned equipment or any ducting in the primary containment. Figure 6.2-39 shows the system P&ID.

The nitrogen supplied from the ACS shall be oil-free with a moisture content of less than 2.5 ppm. Filters are provided to remove particulates larger than 5 micrometers.

(14) The system is designed to facilitate periodic inspections and tests. The ACS can be inspected or tested during normal plant conditions.

(15) The primary containment purge system will aid in the long-term post-accident cleanup operation. The primary containment atmosphere will be purged through the SGTS to the outside environment. Nitrogen makeup will be available during the purging operation.

(16) The ACS is also designed to release containment pressure before uncontrolled containment failure could occur.

6.2.5.2 System Design

6.2.5.2.1 General

is

The ~~FCS and~~ ACS ~~are~~ systems designed to control the environment within the primary containment. ~~The FCS provides control over hydrogen and oxygen generated following a LOCA. In an inerted containment, mixing of any hydrogen generated is not required.~~ Any oxygen evolution from radiolysis is very slow such that natural convection and molecular diffusion is sufficient to provide mixing. Spray operation will provide further

assurance that the drywell or wetwell is uniformly mixed. ~~The FCS consists of the following features:~~

~~(1) The FCS has two recombiners installed in the secondary containment. The recombiners process the combustible gases drawn from the primary containment drywell.~~

~~(2) The FCS is activated when a LOCA occurs. The oxygen and hydrogen remaining in the recombiners after having been processed are transmitted to the suppression pool.~~

The ACS provides and maintains an inert atmosphere in the primary containment during plant operation. The system is not designed as a continuous containment purging system. The ACS exhaust line isolation valves are closed when an inert condition in the primary containment has been established. The nitrogen supply makeup lines, compensating for leakage, provide a makeup flow of nitrogen to the containment. If a LOCA signal is received, the ACS valves close. Nitrogen purge from the containment occurs during shutdown for personnel access. Purging is accomplished with the containment inlet and exhaust isolation valves opened to the selected exhaust path and the nitrogen supply valves closed. Nitrogen is replaced by air in the containment (see Item (3) Shutdown-Deinerting below this subsection). The system has the following features:

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(1) Atmospheric mixing is achieved by natural processes. Mixing will be enhanced by operation of the containment sprays, which are used to control pressure in the primary containment.

(2) The ACS primary containment nitrogen makeup maintains an oxygen-deficient atmosphere ($\leq$3.5% by volume) in the primary containment during normal operation.

(3) The redundant oxygen analyzer system (CAMS) measures oxygen in the drywell and suppression chamber. Oxygen concentrations are displayed in the main control room. Description of safety-related display instrumentation for containment monitoring is provided in Chapter 7. Electrical requirements for equipment associated with the combustible gas control system are in accordance with the appropriate IEEE standards as referenced in Chapter 7.

In addition, the ACS provides overpressure protection to relieve containment pressure, as required, through a pathway from the wetwell airspace to the stack. The pathway is isolated during normal operation by a rupture disk.

The following modes of ACS operation are provided:

(1) **Startup**—Inerting: Liquid nitrogen is vaporized with steam or electric heaters to a temperature greater than -7°C and is injected into the wetwell and the drywell. The nitrogen will be mixed with the primary containment atmosphere by the drywell coolers in the drywell and, if necessary, by the sprays in the wetwell.

(2) **Normal**—Maintenance of Inert Condition: A nitrogen makeup system automatically supplies nitrogen to the wetwell and upper drywell to maintain a slightly positive pressure in the drywell and wetwell to preclude air leakage from the secondary to the primary containment. An increase in containment pressure is controlled by venting through the drywell bleed line.

(3) **Shutdown**—Deinerting: Air is provided to the drywell and wetwell by the Reactor Building HVAC purge supply fan. Exhaust is through the drywell and wetwell exhaust lines to the plant vent, through the HVAC or SGTS, as required. During shutdown, purge air provides containment access ventilation.

(4) Overpressure Protection: If the wetwell pressure increases to about 617.8 kPaG(Subsection 19E.2.8.1), the rupture disk will open. The overall containment pressure decreases as venting continues. Closing the two 250A air-operated butterfly valves re-establishes containment isolation as required.

(5) ACS, except COPS, primary containment isolation valves, if open, (they are normally closed) are automatically closed if the drywell high pressure, or reactor low water level 3 setpoint is reached or if high radiation is detected in the exhaust flow. (See Table 5.2-6)

The following interfaces with other systems are provided:

(1) Residual Heat Removal System (RHR): The RHR System provides post-accident suppression pool cooling, as necessary, following heat dumps to the pool, including the exothermic heat of reaction released by the design basis metal-water reaction. This heat of reaction is very small and has no real effect on pool temperature or RHR heat exchanger sizing. The wetwell spray portion of the RHR may be activated during a LOCA help mixing by reducing pocketing. Wetwell spray would also serve to accelerate deaeration of the suppression pool water, though the impact of the dissolved oxygen on wetwell airspace oxygen concentration is very small. ~~The RHR System also provides cooling water to the exhaust flow from the FCS.~~

setpoint in a severe accident, combined with the already low core damage frequency and reliable containment heat removal, produces an extremely low probability of significant fission product release. In addition, the elapsed time to rupture disk opening is greater than 24 hours for most severe accident sequences.

The net risk reduction associated with the implementation of the COPS system in the design of the ABWR is summarized in Table 19E.2-27 and Figure 19E.2-22. All sequences which would result in COPS operation were assumed to lead to failure of the drywell head. This may slightly over predict the probability of drywell head failure since there will be somewhat more time available for the recovery of containment heat removal if the COPS system were not present. Table 19E.2-26 indicates a low probability of RHR recovery in the interval between the time of COPS initiation and the time of drywell head failure if COPS were not present. For the case with firewater addition to the containment, the probability of RHR recovery during the period of interest is 4%. Therefore, no significant error is introduced into the calculation.

Table 19E.2-27 indicates that the probability of drywell head failure increases by a factor of 50 for sequences with core damage (Class I and III) if the COPS system is not present. For Class II sequences, the loss of containment heat removal may lead to core damage for those sequences which have drywell head failure. Since the probability of drywell head failure increases by a large factor without COPS system, the core damage probability associated with Class II events also increases by the same amount. Figure 19E.2-22 shows the probability of exceedence versus whole body dose at 0.81 kilometers for the ABWR and for the ABWR without the COPS system. The offsite dose is reduced as a result of the COPS implementation into the design.

~~6.2.5.2.7 Flammability Control System~~

~~(1) The FCS consists of two permanently installed, safety-related thermal hydrogen recombiners with associated piping, valves, controls and instrumentation. The recombiner units are located in the secondary containment and controlled from the main control room. Each recombiner shown in Figure 6.2-40 removes gas from the drywell, recombines the oxygen with hydrogen, and returns the gas mixture along with the condensate to the suppression chamber. Each recombiner unit is an integral package consisting of a blower, electric heater, reaction chamber, water spray cooler, water separator, piping, valves, controls and instrumentation.~~

~~(2) During operation of the system, gas is drawn from the drywell by the blower and heated. Hydrogen and oxygen in the gas will be recombined into steam in the reaction chamber and condensed in the spray cooler. The condensate and spray water, along with some of the gas, are returned to the wetwell. The rest of the gas is recycled through the blower. Cooling water required for~~

~~operation-of-the-system-after-a-LOCA-is-taken-from-the-RHR-system. The cooling-water-is-used-to-cool-the-water-vapor-and-the-residual-gases-leaving-the recombiner-prior-to-returning-them-to-the-containment.~~

~~(3) All-pressure-containing-equipment, including-piping-between-components-is considered-an-extension-of-the-containment, and-designed-to-ASME-Section III-Safety-Class-2-requirements. Independent-drywell-and-suppression chamber-penetrations-are-provided-for-the-two-recombiners. Each penetration-has-two-normally-closed-isolation-valves; one-pneumatically operated-and-one-motor-operated. The-system-is designed-to-meet-Seismic Category-I-requirements. The-recombiners-are-in-separate-rooms-in-the secondary-containment-and-are-protected-from-damage-by-flood, fire, tornadoes-and-pipe-whip.~~

~~(4) After-a-LOCA, the-system-is-manually-actuated-from-the-control-room-when high-oxygen-levels-are-indicated-by-the-containment-atmospheric-monitoring system-(CAMS). (If-hydrogen-is-not-present, oxygen-concentrations-are controlled-by-nitrogen-makeup.) Operation-of-either-recombiner-will-provide effective-control-over-the-buildup-of-oxygen-generated-by-radiolysis-after-a design-basis-LOCA. Once-placed-in-operation-the-system-continues-to-operate until-it-is-manually-shut-down-when-an-adequate-margin-below-the-oxygen concentration-design-limit-is-reached.~~

6.2.5.3 Design Evaluation

The ACS is designed to maintain the containment in an inert condition except for nitrogen makeup needed to maintain a positive containment pressure and prevent air (0_2) leakage from the secondary into the primary containment.

The primary containment atmosphere will be inerted with nitrogen during normal operation of the plant. Oxygen concentration in the primary containment will be maintained below 3.5% by volume measured on a dry basis.

During normal operation, nitrogen makeup and containment pressure control are accomplished using only the 50A supply lines. The large valves (550A) in the containment ventilation lines are closed and flow to the plant stack through the overpressure protection line (250A) is prevented by the rupture disk.

The following conditions assure that the large (550A) containment purge and vent lines will be isolated following a LOCA:

(1) The valves remain closed at all times during normal operation and will only be opened for inerting or de-inerting at the beginning and end of a shutdown.

6.2.5.4 Tests and Inspections

Complete process systems are pressure tested to the maximum practicable extent. Piping systems will be hydrostatically tested in their entirety, utilizing available valves or temporary plugs. Hydrostatic testing of piping systems will be performed at a pressure 1.5 times the design pressure, but in no case at less than 519.8 kPaG. The test pressure will be held for a minimum of 30 minutes. Pneumatic testing may be substituted for hydrostatic testing in accordance with the applicable codes.

Preoperational testing will demonstrate the ability of the ACS to meet design requirements. Each valve will be exercised both opened and closed and position indication verified. Trip and alarm logic signals will also be checked. The tests assure correct functioning of all controls, instrumentation, compressors, recombiners, piping and valves. System reference characteristics, such as pressure differentials and flow rates, are documented during the preoperational tests and are used as base points for measurements in subsequent operational tests.

During plant operation, the ACS, its valves, instrumentation, wiring and other components outside the containment can be inspected visually at any time. Testing frequencies of the ACS components are generally correlated with testing frequencies of the associated controls and instrumentation. When a valve control is tested, the operability of that valve and its associated instrumentation are generally tested by the same action. In addition, inservice inspection and testing of all ASME Section III, Class 3 components is done in accordance with Subsections 6.6.5 and 3.9.6, respectively.

Preoperational tests of the ACS is ~~and FCS are~~ conducted during the final stages of plant construction prior to initial startup.

The overpressure protection concept was designed to minimize any adverse impact on normal operation or maintenance. Initially, several rupture disks from a batch of rupture disk could be tested to verify the opening characteristics and setpoint. The disk would be replaced every five years according to normal industry practice. The installation of the disk would not impact containment leakage tests, since disk integrity is expected to be essentially perfect.

The overpressure protection valves would be tested during preoperational testing and periodically during inservice testing (Subsection 3.9.6), to verify their normally open position and their ability to close using AC power and pneumatic air.

6.2.5.5 Instrumentation Requirements

Separate inerting flow indication to both the drywell and wetwell are provided. Drywell pressure and makeup flow are monitored and recorded in the main control room. Additional drywell pressure instrumentation, with a lower setpoint, is provided in addition to the redundant, safety-grade drywell pressure instrumentation of the Nuclear

Table 6.2-7 Containment Isolation Valve Information*

MPL	System	Page
B21	Nuclear Boiler	Page 6.2-140 thru Page 6.2-142
B31	Reactor Recirculation	Page 6.2-125
C41	Standby Liquid Control	Page 6.2-126
D23	Containment Atmospheric Monitoring	Page 6.2-127 thru Page 6.2-128
E11	Residual Heat Removal	Page 6.2-129 thru Page 6.2-136
E22	High Pressure Core Flooder	Page 6.2-137 thru Page 6.2-139
E31	Leak Detection & Isolation	Page 6.2-166
E51	Reactor Core Isolation Cooling	Page 6.2-144 thru Page 6.2-148
G31	Reactor Water Cleanup	Page 6.2-157 thru Page 6.2-158
G51	Suppression Pool Cleanup	Page 6.2-159
K17	Radwaste	Page 6.2-167
P11	Makeup Water (Purified)	Page 6.2-165
P21	Reactor Building Cooling Water	Page 6.2-160
P24	HVAC Normal Cooling Water	Page 6.2-161
P51	Service Air	Page 6.2-162
P52	Instrument Air	Page 6.2-163
P54	High Pressure Nitrogen Gas Supply	Page 6.2-164
T31	Atmospheric Control	Page 6.2-149 thru Page6.2-154
~~T49~~	~~Flammability Control~~	~~Page 6.2-155 and Page 6.2-156~~
See page 6.2-167 for notes		

* This table responds to NRC Questions 430.35, 430.50b. 430.50c, 430.50d and 430.50f regarding containment isolation provisions for fluid system lines and for fluid instrument lines penetrating containment within the scope of the ABWR Standard Plant. Locked closed isolation valves are identified on the P&IDs. The containment information is presented separately for each system for the MPL numbers given below.

~~Table 6.2-7 Containment Isolation Valve Information~~
~~Flammability Control System~~

~~Valve No.~~	~~T49-F001C~~	~~T49-F001B~~	~~T49-F002A~~	~~T49-F002E~~
~~Tier 2 Figure~~	~~6.2-40 (Sheet 2)~~	~~6.2-40 (Sheet 1)~~	~~6.2-40 (Sheet 1)~~	~~6.2-40 (Sheet 2)~~
~~Applicable Basis~~	~~GDC 56~~	~~GDC 56~~	~~GDC 56~~	~~GDC 56~~
~~Fluid~~	~~DW Atmosphere~~	~~DW Atmosphere~~	~~DW Atmosphere~~	~~DW Atmosphere~~
~~Line Size~~	~~100A~~	~~100A~~	~~100A~~	~~100A~~
~~ESF~~	~~Yes~~	~~Yes~~	~~Yes~~	~~Yes~~
~~Leakage Class~~	~~(a)~~	~~(a)~~	~~(a)~~	~~(a)~~
~~Location~~	~~O~~	~~O~~	~~O~~	~~O~~
~~Type C Leak Test~~	~~No(u)~~	~~No(u)~~	~~No(u)~~	~~No(u)~~
~~Valve Type~~	~~Gate~~	~~Gate~~	~~Gate~~	~~Gate~~
~~Operator~~	~~Motor~~	~~Motor~~	~~Pneumatic~~	~~Pneumatic~~
~~Primary Actuation~~	~~Electrical~~	~~Electrical~~	~~Electrical~~	~~Electrical~~
~~Secondary Actuation~~	~~Manual~~	~~Manual~~	~~Manual~~	~~Manual~~
~~Normal Position~~	~~Close~~	~~Close~~	~~Close~~	~~Close~~
~~Shutdown Position~~	~~Close~~	~~Close~~	~~Close~~	~~Close~~
~~Post-Accident Position~~	~~Open~~	~~Open~~	~~Open~~	~~Open~~
~~Power Fail Position~~	~~As-is~~	~~As-is~~	~~As-is~~	~~As-is~~
~~Containment Isolation Signal[(e)]~~	~~A,K~~	~~A,K~~	~~A,K~~	~~A,K~~
~~Closure Time (s)~~	~~<30~~	~~<30~~	~~<30~~	~~<30~~
~~Power Source (Div)~~	~~III~~	~~II~~	~~I, III~~	~~I, II~~

~~See page 6.2-167 for notes~~

~~Table 6.2-7 Containment Isolation Valve Information Flammability Control System~~

~~Valve No.~~	~~T49-F006A~~	~~T49-F006E~~	~~T49-F007C~~	~~T49-F007B~~
~~Tier 2 Figure~~	~~6.2-40 (Sheet 1)~~	~~6.2-40 (Sheet 2)~~	~~6.2-40 (Sheet 2)~~	~~6.2-40 Sheet 1)~~
~~Applicable Basis~~	~~GDC 56~~	~~GDC 56~~	~~GDC 56~~	~~GDC 56~~
~~Fluid~~	~~WW Atmosphere~~	~~WW Atmosphere~~	~~WW Atmosphere~~	~~WW Atmosphere~~
~~Line Size~~	~~150A~~	~~150A~~	~~150A~~	~~150A~~
~~ESF~~	~~Yes~~	~~Yes~~	~~Yes~~	~~Yes~~
~~Leakage Class~~	~~(a)~~	~~(a)~~	~~(a)~~	~~(a)~~
~~Location~~	~~O~~	~~O~~	~~O~~	~~O~~
~~Type C Leak Test~~	~~No(u)~~	~~No(u)~~	~~No(u)~~	~~No(u)~~
~~Valve Type~~	~~Gate~~	~~Gate~~	~~Gate~~	~~Gate~~
~~Operator~~	~~Pneumatic~~	~~Pneumatic~~	~~Motor~~	~~Motor~~
~~Primary Actuation~~	~~Electrical~~	~~Electrical~~	~~Electrical~~	~~Electrical~~
~~Secondary Actuation~~	~~Manual~~	~~Manual~~	~~Manual~~	~~Manual~~
~~Normal Position~~	~~Close~~	~~Close~~	~~Close~~	~~Close~~
~~Shutdown Position~~	~~Close~~	~~Close~~	~~Close~~	~~Close~~
~~Post Accident Position~~	~~Open~~	~~Open~~	~~Open~~	~~Open~~
~~Power Fail Position~~	~~As-is~~	~~As-is~~	~~As-is~~	~~As-is~~
~~Containment Isolation Signal[e]~~	~~A,K~~	~~A,K~~	~~A,K~~	~~A,K~~
~~Closure Time (s)~~	~~<30~~	~~<30~~	~~<30~~	~~<30~~
~~Power Source (Div)~~	~~I, III~~	~~I, II~~	~~III~~	~~II~~

~~See page 6.2-167 for notes~~

(o) Furthermore, these valves are subject to ASME leak rate tests as in (k) above.

(p) Rupture discs are normally closed and sealed from leakage. The opening setpoint of these rupture discs is higher than primary containment test pressures. Additionally, these rupture discs are subject to the Type A test.

(q) SPCU suction line is always filled with water, since it is located below the suppression pool water level and is sealed from the containment atmosphere.

(r) SPCU return line terminates below the suppression pool water level and is sealed from the containment atmosphere.

(s) The outboard side of these valves is always pressurized with nitrogen gas at a pressure higher than the post-accident peak containment pressure. The nitrogen supply in these lines is required for post-accident mitigating function.

(t) The outboard side of these valves is always filled with water and pressurized above 110% post-accident peak containment pressure. These lines are kept charged with cooling water for cooling emergency equipment necessary for post-accident mitigation.

(u) Line will be drained and tested with air.

~~(v)~~ ~~Flammability control is a closed loop, safety grade system required to be functional post-accident. Whatever is leaking (if any) is returned to the primary containment. In addition, during ILRT, these valves are opened and the lines are subjected to Type A test.~~

(v) ~~(w)~~ These lines terminate below the drywell sumps water level and are sealed from the containment atmosphere.

(w) ~~(x)~~ The outboard side of these valves are provided with a water leg. In addition, these valves are subject to ASME leak tests as in (k) above.

(x) ~~(y)~~ Not applicable.

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Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-37	RCIC Turbine Steam	14450	80	1200	550		A
X-38	RPV Head Spray	14450	310	1500	550		A
X-50	CUW Pump Feed	14480	310	0	600		A
X-60	MUWP Suction	13500	290	0	200		A
X-61	RCW Suction (A)	13500	45	-3000	200		A
X-62	RCW Return (A)	13500	45	-2000	200		A
X-63	RCW Suction (B)	13500	225	3400	200		A
X-64	RCW Return (B)	13500	225	2400	200		A
X-65	HNCW Suction	13500	225	250	350		
X-66	HNCW Return	13500	225	1400	350		
X-69	SA	19000	42	0	90		A
X-70	IA	9000	46	0	200		A
X-71A	ADS Accumulator (A)	19000	50	0	200		A
X-71B	ADS Accumulator (B)	19000	296.5	1000	200		A
X-72	Relief Valve Accumulator	19000	296.5	2000	200		A
X-80	Drywell Purge Suction	13700	68	0	550		A
X-81	Drywell Purge Exhaust	19000	216	0	550		A
X-82	~~FCS Suction~~ Spare	~~14850~~	~~225~~	~~-600~~	~~150~~		A
X-90	Spare	20100	46	0	400		A
X-91	Spare	20100	296.5	1000	400		A
X-92	Spare	16400	45	12700	400		A
X-93	Spare	14700	135	-500	400		A
X-100A	RIP Power	13500	55	-1100	450	O-ring	B

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* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-104C	FMCRD Position Indicator	20100	99	0	300	O-ring	B
X-104D	FMCRD Position Indicator	20100	279.5	0	300	O-ring	B
X-104E	FMCRD Position Indicator	19000	81	1350	300	O-ring	B
X-104F	FMCRD Position Indicator	19000	260.5	1350	300	O-ring	B
X-104G	FMCRD Position Indicator	19000	99	0	300	O-ring	B
X-104H	FMCRD Position Indicator	19000	279.5	0	300	O-ring	B
X-105A	Neutron Detection	20100	81	1350	300	O-ring	B
X-105B	Neutron Detection	20100	260.5	1350	300	O-ring	B
X-105C	Neutron Detection	20100	99	-5250	300	O-ring	B
X-105D	Neutron Detection	20100	279.5	1350	300	O-ring	B
X-110	Spare ~~FCS Suction~~	~~13500~~	~~55~~	~~1000~~	~~300~~	~~O-ring~~	B
X-111	Spare	13500	280	1350	300	O-ring	B
X-112	Spare	13500	180	-5250	300	O-ring	B
X-113	Spare	13500	180	1350	300	O-ring	B
X-130A	I & C	13500	45	0	300	O-ring	B
X-130B	I & C	13500	212	0	300	O-ring	B
X-130C	I & C	13500	124	0	300	O-ring	B
X-130D	I & C	13500	295	0	300	O-ring	B
X-140A	I & C	13500	45	-27000	300	O-ring	B

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

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Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-240	Wetwell Purge Suction	9200	45	1200	550		A
X-241	Wetwell Purge Exhaust	9200	230	0	550		A
X-242	~~FCS Return~~ Spare	~~1500~~	225	~~-1000~~	~~150~~		A
X-250	Spare	8500	45	0	400		A
X-251	Spare	9000	213	0	400		A
X-252	~~FCS Return~~ Spare	~~1500~~	~~50~~	0	300		B
X-253	Spare	2650	135	1000	300		B
X-254	Spare	2650	225	-1000	300		B
X-255	Spare	1200	282	0	300		B
X-300A	I & C	7300	134	0	300	O-ring	B
X-300B	I & C	7300	211	0	300	O-ring	B
X-320	I & C	8900	74	0	90	O-ring	B
X-321A	I & C	2050	97.5	0	300	O-ring	B
X-321B	I & C	6000	262.5	0	300	O-ring	B
X-322A	I & C	400	78	0	90	O-ring	B
X-322B	I & C	400	258	0	90	O-ring	B
X-322C	I & C	400	102	0	90	O-ring	B
X-322D	I & C	400	282	0	90	O-ring	B
X-322E	I & C	2000	94	0	90	O-ring	B
X-322F	I & C	2000	266	0	90	O-ring	B
X-323A	I & C	-5200	30	0	90	O-ring	B
X-323B	I & C	-5200	210	0	90	O-ring	B
X-323C	I & C	-5200	156	0	90	O-ring	B
X-323D	I & C	-5200	304	0	90	O-ring	B
X-323E	I & C	-7500	100	0	90	O-ring	B
X-323F	I & C	-7500	230	0	90	O-ring	B

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

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Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-65	HNCW Suction	350	E	E/D/H	No
X-66	HNCW Return	350	E	E/D/H	No
X-69	SA	90	E	E/D/H	No
X-70	IA	200	E	E/D/H	No
X-71A	ADS Accumulator (A)	200	S	C/K	No
X-71B	ADS Accumulator (B)	200	S	C/K	No
X-72	Relief Valve Accumulator	200	S	C/K	No
X-80	Drywell Purge Suction	550	E	E/C/J	Yes
X-81	Drywell Purge Exhaust	550	E	E/C/J	Yes
X-82	~~FCS Suction~~	~~150~~	~~S~~	E/C/H	~~No~~
X-90	Spare Spare	400	P	B/A	No
X-91	Spare	400	P	B/A	No
X-92	Spare	400	P	B/A	No
X-93	Spare	400	P	B/A	No
X-100A	IP Power	450	S	C/J	No
X-100B	IP Power	450	S	C/J	No
X-100C	IP Power	450	S	C/J	No
X-100D	IP Power	450	S	C/J	No
X-100E	IP Power	450	S	C/J	No
X-101A	LP Power	300	S	C/J	No
X-101B	LP Power	300	S	C/J	No
X-101C	LP Power	300	S	C/J	No
X-101D	FMCRD Power	300	S	C/J	No
X-101E	FMCRD Power	300	S	C/J	No
X-101F	FMCRD Power	300	S	C/J	No
X-101G	FMCRD Power	300	S	C/J	No
X-102A	I & C	300	S	C/J	No
X-102B	I & C	300	S	C/J	No
X-102C	I & C	300	S	C/J	No
X-102D	I & C	300	S	C/J	No
X-102E	I & C	300	S	C/J	No

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Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-102F	I & C	300	S	C/J	No
X-102G	I & C	300	S	C/J	No
X-102H	FMCRD Control	300	S	C/J	No
X-102J	FMCRD Control	300	S	C/J	No
X-103A	I & C	300	S	C/J	No
X-103B	I & C	300	S	C/J	No
X-103C	I & C	300	S	C/J	No
X-104A	FMCRD Pos. Indicator	300	S	C/J	No
X-104B	FMCRD Pos. Indicator	300	S	C/J	No
X-104C	FMCRD Pos. Indicator	300	S	C/J	No
X-104D	FMCRD Pos. Indicator	300	S	C/J	No
X-104E	FMCRD Pos. Indicator	300	S	C/J	No
X-104F	FMCRD Pos. Indicator	300	S	C/J	No
X-104G	FMCRD Pos. Indicator	300	S	C/J	No
X-104H	FMCRD Pos. Indicator	300	S	C/J	No
X-105A	Neutron Detection	300	S	C/J	No
X-105B	Neutron Detection	300	S	C/J	No
X-105C	Neutron Indicator	300	S	C/J	No
X-105D	Neutron Indicator	300	S	C/J	No
X-110	~~FCS Suction~~	~~100~~	~~S~~	~~E/C/H~~	~~No~~
X-111	Spare Spare	300	P	B/A	No
X-112	Spare	300	P	B/A	No
X-113	Spare	300	P	B/A	No
X-130A	I & C	300	S	C/J	No
X-130B	I & C	300	S	C/J	No
X-130C	I & C	300	S	C/J	No
X-130D	I & C	300	S	C/J	No
X-140A	I & C	300	S	C/J	No
X-140B	I & C	300	S	C/J	No
X-141A	I & C	300	S	C/J	No
X-141B	I & C	300	S	C/J	No

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Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-201	RHR Pump Suction (A)	450	S	C/H	No
X-202	RHR Pump Suction (B)	450	S	C/H	No
X-203	RHR Pump Suction (C)	450	S	C/H	No
X-204	RHR Pump Test (A)	250	S	C/H	No
X-205	RHR Pump Test (B)	250	S	C/H	No
X-206	RHR Pump Test (C)	250	S	C/H	No
X-210	HPCF Pump Suction (B)	400	S	C/H	No
X-211	HPCF Pump Suction (C)	400	S	C/H	
X-213	RCIC Turbine Exhaust	550	S	C/G	
X-214	RCIC Pump Suction	200	S	C/H	
X-215	RCIC Vacuum Pump Ex.	250	S	C/G	No
X-216	SPCU Pump Suction	200	S	C/H	No
X-217	SPCU Pump Return	250	S	C/H	No
X-220	MSIV Leakage	250	S	C/G	No
X-240	Wetwell Purge Suction	550	E	E/C/J	Yes
X-241	Wetwell Purge Exhaust	550	E	E/C/J	Yes
X-242	~~FCS-Return~~ Spare	~~150~~	S	~~E/C/H~~	No
X-250	Spare		P	B/A	No
X-251	Spare		P	B/A	No
X-252	~~FCS-Return~~ Spare	~~150~~	~~S~~	~~E/C/H~~	No
X-253	Spare	300	S	B/A	No
X-254	Spare	300	S	B/A	No
X-255	Spare	300	S	B/A	No
X-300A	I&C	300	S	C/J	No
X-300B	I&C	300	S	C/J	No
X-320	I&C	90	S	C/J	No
X-321B	I&C	300	S	C/J	No
X-322A	I&C	90	S	C/J	No
X-322B	I&C	90	S	C/J	No

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The following figures are located in Chapter 21:

Figure 6.2-38 Plant Requirements, Group Classification and Containment Isolation Diagram (Sheets 1 – 2)

Figure 6.2-39 Atmospheric Control System P&ID (Sheets 1 – 3)

~~**Figure 6.2-40 Flammability Control System P&ID (Sheets 1 — 2)**~~

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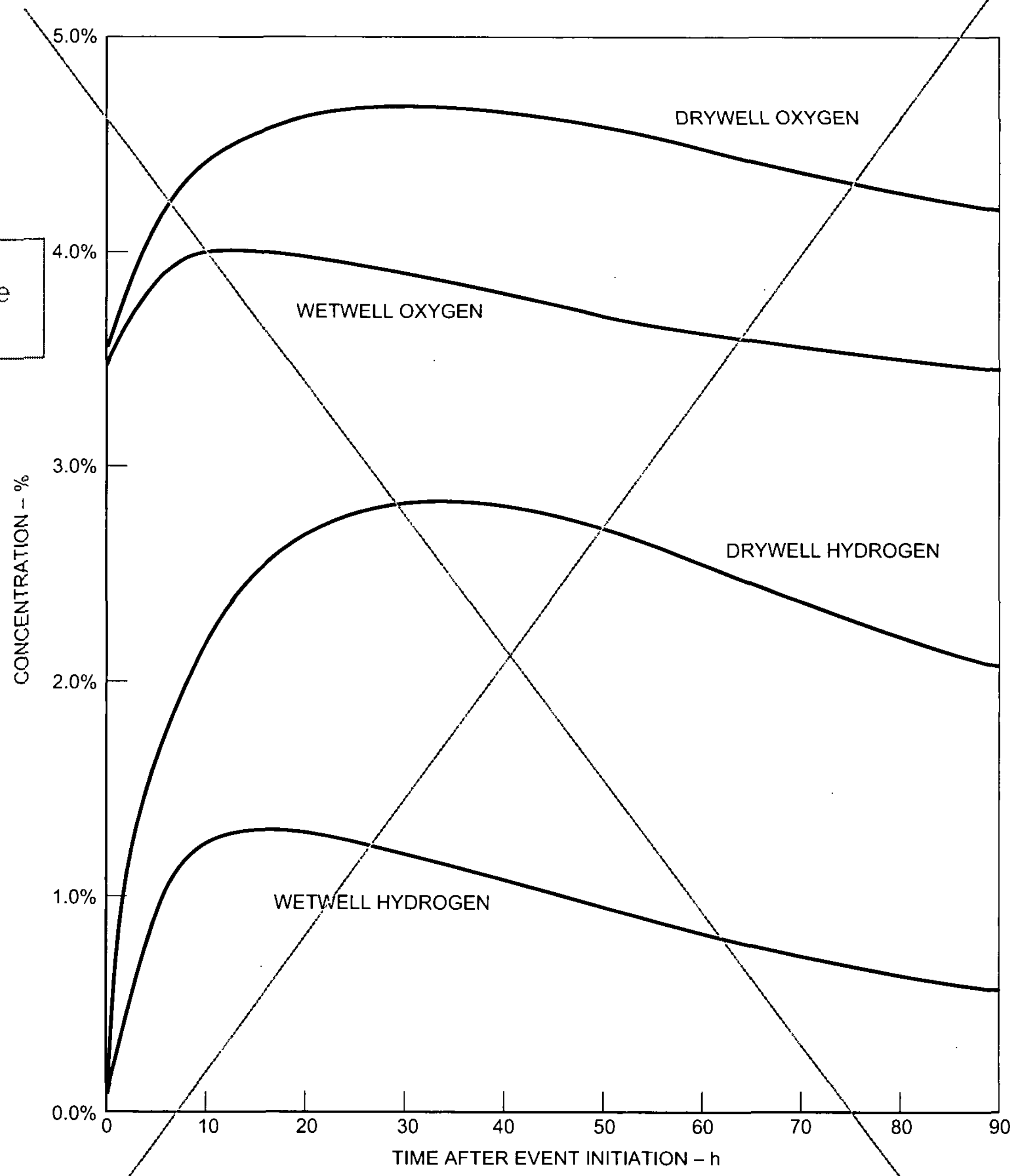

Figure 6.2-41 Hydrogen and Oxygen Concentrations in Containment After Design Basis LOCA

Figure 6.2-38 Group Classification and Containment Isolation Diagram (Sheet 1 of 2)

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

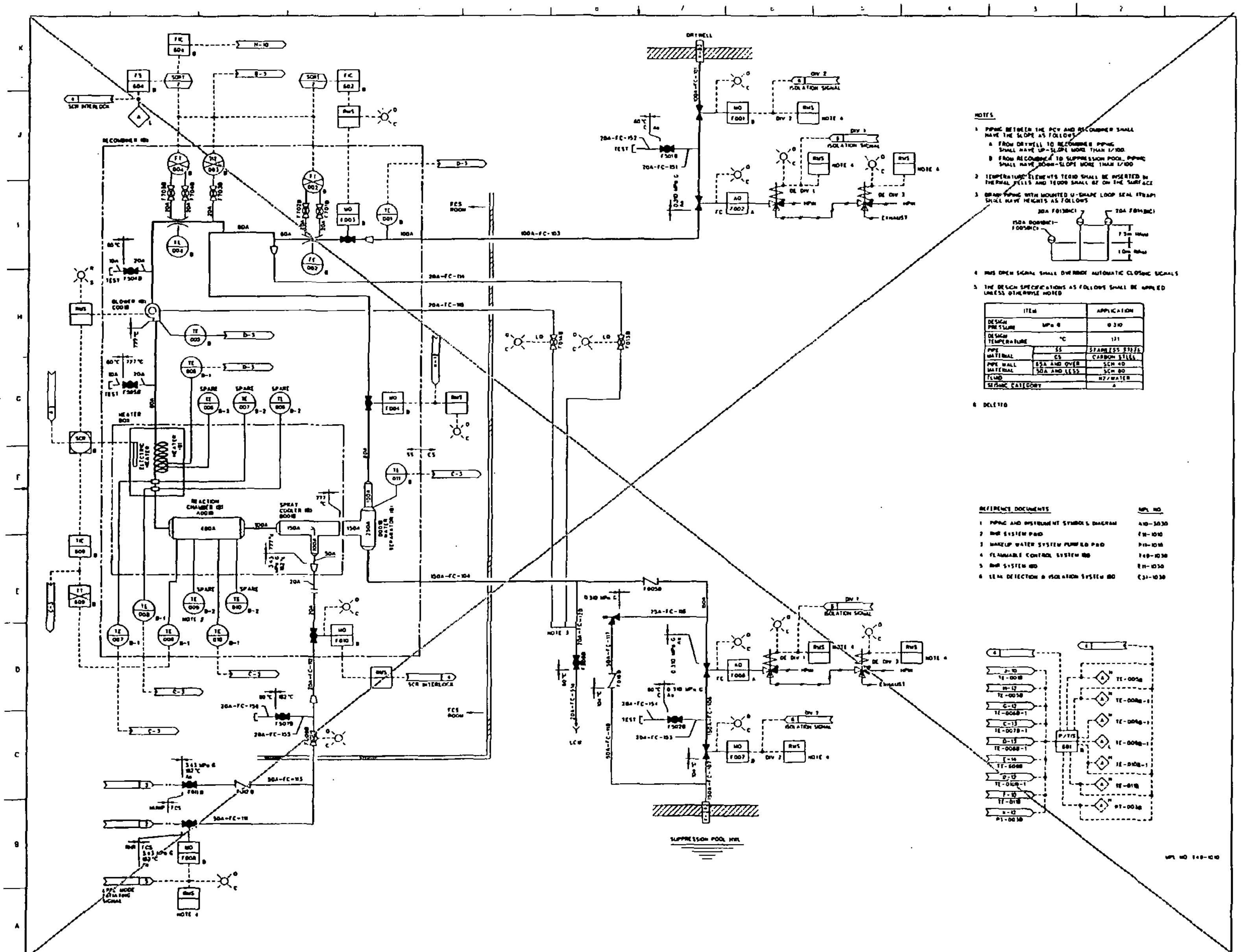

FIGURE 6.2-40 FLAMMABILITY CONTROL SYSTEM P&ID (Sheet 1 of 2)
ABWR DCD/Tier 2 Rev. 0 21-123

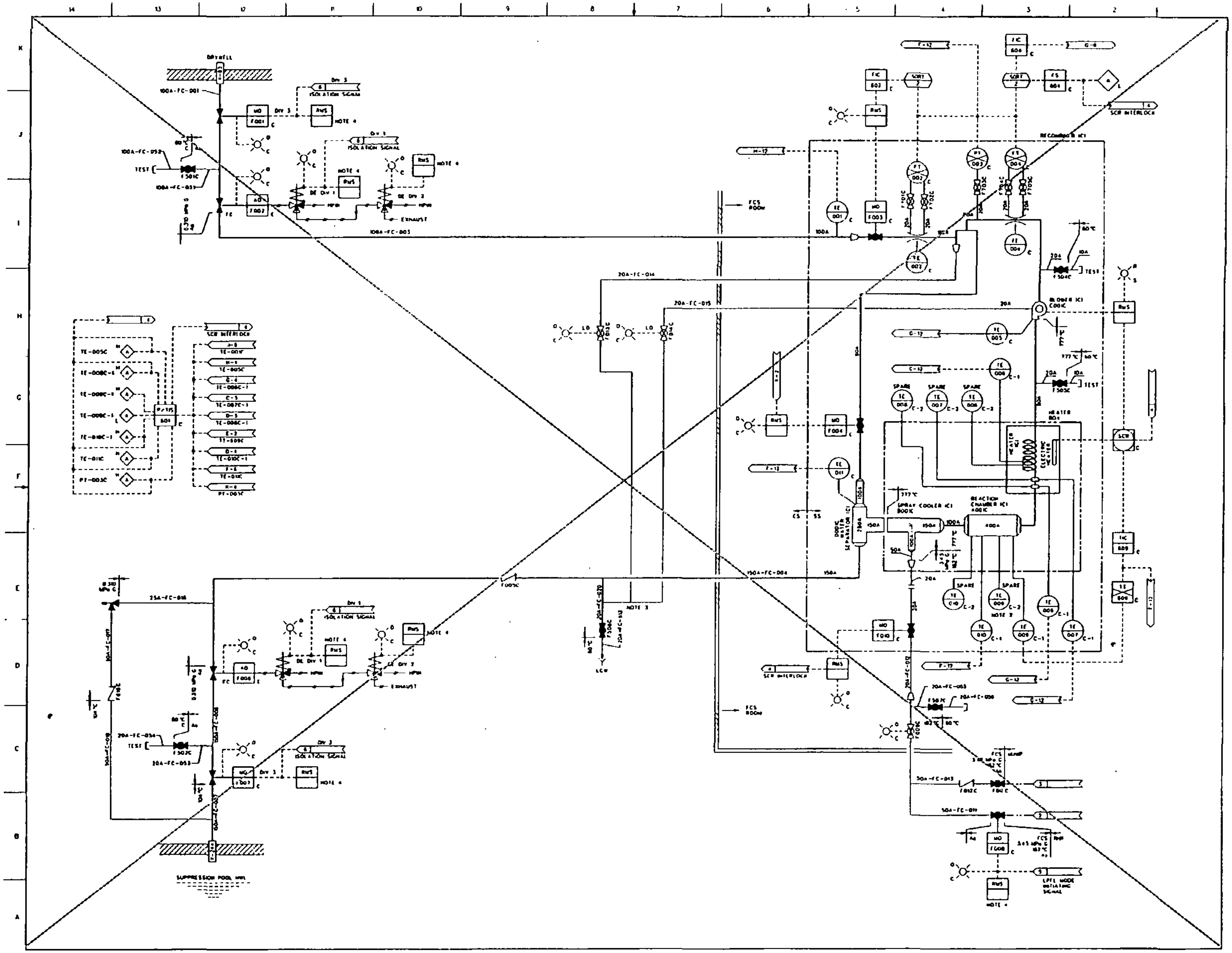

FIGURE 6.2-40 FLAMMABILITY CONTROL SYSTEM P&ID (Sheet 2 of 2)
ABWR DCD/Tier 2 Rev. 0 21-124

6.5.2 Containment Spray Systems

Credit is not taken for any fission product removal provided by the drywell/wetwell spray portions of the RHR System.

6.5.3 Fission Product Control Systems

Fission product control systems are provided in conjunction with other ESF systems to limit the release of radioactive material from the containment to the environment following postulated design basis breaks inside containment and refueling operation accident events. Dose analyses are provided in Chapter 15. The fission product control systems consist of the primary containment and the secondary containment. The following is a discussion of each fission product control system.

6.5.3.1 Primary Containment

The primary containment is a cylindrical steel-lined reinforced concrete structure forming a limited leakage boundary for fission products released to the containment atmosphere following a LOCA or other event. The containment is divided into the upper and lower drywells and the suppression chamber (wetwell) by the reinforced concrete diaphragm floor and the reactor vessel pedestal. The diaphragm floor is rigidly attached to the reactor pedestal and the containment wall. A liner is also provided as part of the diaphragm floor to prevent bypass of steam from the upper drywell to the suppression chamber air space during an accident. The primary containment is totally within the secondary containment. A test program confirms the integrity of the leakage boundary. The assumed leak rate from primary containment is 0.5% of the free containment volume per day measured at the containment design pressure.

Containment leak rate testing is described in Subsection 6.2.6. The primary containment walls, liner plate, mechanical penetrations, isolation valves, hatches, and locks function to limit release of radioactive materials, subsequent to postulated accidents, such that the resulting offsite doses are less than the guideline values of 10CFR100.

The structural design details of the primary containment are discussed in Subsection 3.8.2. Primary containment isolation valves are discussed in Subsection 6.2.4. The conditions in the containment during and after the design basis events are given in Section 6.2.

Layouts of the primary containment structure are given in the building arrangement drawings in Section 1.2.

The primary containment atmosphere is inerted with nitrogen by the Atmospheric Control System (ACS). The ACS is described in Subsection 6.2.5. ~~Following the design~~

basis ~~LOCA, the Flammability Control System (FCS) controls the concentration of oxygen in containment. Oxygen is generated by the radiolytic decomposition of water.~~

On appropriate signals, containment isolation valves close as required. The primary containment provides a passive barrier to limit the leakage of airborne radioactive material. Systems required to accomplish ECCS or other ESF functions are not isolated. See Subsection 6.2.4 for further details of isolation valve closure signals.

6.5.3.2 Secondary Containment

The secondary containment is provided so that leakage from the primary containment is collected, treated and monitored by the SGTS prior to release to the environment. Refer to Subsection 6.2.3 for a description of the secondary containment boundary and Subsection 6.5.1 for a description of the SGTS.

6.5.4 Not Used

6.5.5 COL License Information

6.5.5.1 SGTS Performance

The COL applicant will perform a SGTS dose/functional damage and drawdown analysis in accordance with Subsections 6.5.1.2.3.2 and 6.5.1.3.1(5) respectively.

6.5.5.2 SGTS Exceeding 90 Hours of Operation Per Year

The COL applicant is required to demonstrate the SGTS system is capable of performing its intended function in the event of a LOCA, if more than 90 hours of operation per year (excluding test) for either train is anticipated.

6.5.6 References

6.5-1 Thomas E. Murley (NRC) letter to Ricardo Artigas (GE), August 7, 1987, "Advanced Boiling Water Reactor Licensing Review Bases".

Table 6.6-1 Examination Categories and Methods (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam Method
C	P21	Reactor Building Cooling Water (Cont.)	All Class C branch lines 100A and smaller, i.e.: - lines to and from RHR/HPCF pumps seals, motor bearing coolers - lines to and from RCIC pump room coolers - instrument lines - lines to and from FPC, SGTS, ~~FCS~~ room coolers - lines to and from CAM System coolers and air conditioning unit - drain lines - test connections - and etc.	Figure 9.2-1 sh. 1, 2, 4, 5, 7, 8 Rm. 425 / 436	Exempted per IWD-1220		
			All pressure-retaining components and piping		D-B	External Surfaces (Note 7)	VT-2
C	P41	Reactor Service Water	From suction strainers through RSW pumps C001A, D, B, E, C, F, and through RCW HXs and into but not including the discharge canal to the ultimate heat sink.	Figure 9.2-7			
			All pressure-retaining components and piping		D-B	External Surfaces (Note 7)	VT-2
			Integral attachments		D-B	Welds (Note 8)	VT-3
			Piping and Component Supports		F-A	Supports (Note 6)	VT-3

Table 7.1-1 Comparison of GESSAR II and ABWR I&C Safety Systems (Continued)

I & C System	GESSAR II Design	ABWR Design
Leak Detection and Isolation System (LD&IS):	Leak detection system (LDS) separate from containment and reactor vessel isolation & control system (CRVICS).	Combined LDS and CRVICS to make LD&IS.
	Main steam positive leakage & control system (MSPLCS).	MSPLCS deleted.
	All inboard isolation valves powered by Division 2; all outboard isolation valves powered by Division 1.	Divisions 1, 2, and 3 are used in various combinations to obtain redundant pairs of inboard/outboard isolation valves.
RHR/Wetwell Drywell Spray Mode:	2 wetwell/drywell cooling divisions. Both automatically and manually actuated.	2 wetwell/drywell cooling divisions. Manual actuation only.
RHR/Suppression Pool Cooling Mode:	2 loops and 2 divisions. Manual initiation.	3 loops and 3 divisions. Automatic and manual initiation.
~~Flammability Control System:~~	~~Part of combustible gas control system.~~	~~Independent system.~~
Standby Gas Treatment System:	Redundant active and passive components.	Redundant active components; single filter train.
Emergency Diesel Generator System:	ESF diesels: Divisions 1 & 2. HPCS diesel: Div. 3.	ESF Diesels: Divisions I,II & III (HPCF included on Divisions II & III).
Reactor Building Cooling Water:	Open loop to ultimate heat sink. System was called "essential service water system".	Closed loop with limited quantity of water.
Containment Atmospheric Control System:	Hydrogen mixing system interface.	Dedicated hydrogen mixing not required for inerted containment.
High Pressure Nitrogen Gas Supply:	(Air supply only)	Replaces air supply to ADS and SRV accumulators. Also used for testing MSIVs.
Alternate Rod Insertion (ARI) Function:	(Not applicable)	New function provided by fine motion control rod drive (FMCRD) capability of the rod control & information system (RC&IS).
Standby Liquid Control System (SLCS):	Squib-type injection valve.	Motor-operated-type injection valve.
	Pump indication "RUN", "STOP", "TRIPPED"	Pump indication "RUN", "STOP"

(g) Testability

The HPIN System can be tested at any time by isolating the system from the normal nitrogen source and allowing the nitrogen pressure to decrease. At the proper pressure, valves will open, admitting nitrogen from the high pressure storage bottles; other valves will close, isolating the non-safety-related portions of the system.

(h) Environmental Considerations

The system safety-related equipment is selected in consideration of the normal and accident environments in which it must be operated.

(i) Operational Considerations

The HPIN System, when required for emergency conditions, is initiated automatically with no operator action required.

Running lights, valve positions, indicating lights, and alarms are available in the control room for the operator to accurately assess the HPIN System operation. Common trouble alarms are available in the main control room for the system. Isolation valves have indicating lights for full-open and full-closed positions.

~~7.3.1.1.11 Flammability Control System—Instrumentation and Controls~~

~~(See Subsection 6.2.5)~~

7.3.1.2 Design Basis Information

IEEE-279 defines the requirements for design bases. Using the IEEE 279 format, the following nine paragraphs fulfill this requirement for systems and equipment described in this section.

(1) Conditions

The plant conditions which require protective action involving the systems of this section and other sections are examined in Chapter 15.

(2) Variables

The plant variables that are monitored to provide automatic protective actions are discussed in the initiating circuits sections for each system. For additional information, see Chapter 15, where safety analysis parameters for each event are cited.

(7) Electrical Power Distribution System (EPDS)

(a) The following functions have transfer and control switches located on the Division I remote shutdown panel:

(i) 6.9 kV feeder breaker: Unit auxiliary transformer A to M/C E
(ii) 6.9 kV feeder breaker: Reserve auxiliary transformer A to M/C E
(iii) 6.9 kV feeder breaker: Emergency diesel generator A to M/C E
(iv) 6.9 kV feeder breaker: Combustion turbine generator to M/C E
(v) 6.9 kV load breaker: M/C E to P/C E20
(vi) 480V feeder breaker: TR to P/C E20

(b) The following functions have transfer and control switches located on the Division II remote shutdown panel:

(i) 6.9 kV feeder breaker: Unit auxiliary transformer B to M/C F
(ii) 6.9 kV feeder breaker: Reserve auxiliary transformer A to M/C F
(iii) 6.9 kV feeder breaker: Emergency diesel generator B to M/C F
(iv) 6.9 kV feeder breaker: Combustion turbine generator to M/C F
(v) 6.9 kV load breaker: M/C F to P/C F20
(vi) 480V feeder breaker: TR to P/C F20

(c) A 6.9 kV M/C (E,F) voltmeter is provided on RSS panels A,B, respectively.

~~(8) Flammability Control System (FCS)~~

~~(a) The following FCS equipment function has transfer and control switches located on both remote shutdown panels as indicated:~~

~~(i) Valve (cooling water inlet) B~~

(9) Atmospheric Control (AC) System

(a) Suppression pool level indication is provided on both RS panels.

(10) Makeup Water Condensate System (MUWC)

(a) Condensate storage pool level indication is provided on RS panel B.

(11) Suppression Pool Temperature Monitoring System (SPTM)

(a) Suppression pool temperature indication is provided on both RS panels.

The RSS provides instrumentation and controls outside the main control room to allow prompt hot shutdown of the reactor after a scram and to maintain safe conditions during hot shutdown. It also provides capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

7.4.2.4.2 Specific Regulatory Requirements Conformance

Table 7.1-2 identifies the Remote Shutdown System (RSS) and the associated codes and standards applied in accordance with the Standard Review Plan. The following analysis lists the applicable criteria in order of the listing on the table, and discusses the degree of conformance for each. Any exceptions or clarifications are so noted.

(1) 10CFR50.55a (IEEE-279)

The Remote Shutdown System (RSS) consists of two panels (Division I and Division II) which are located in separate rooms in the Reactor Building.

The RSS provides remote control capability as defined by the following interfaces:

System	Total Channels	RSS Interface
Residual Heat Removal	A, B, C	A, B
High Pressure Core Flooder	B, C	B
Nuclear Boiler System	A, B, C, D	A, B
Reactor Bldg. Cooling Water	A, B, C	A, B
Reactor Service Water	A, B, C	A, B
Electrical Power Distribution	I, II, III, IV	I, II
~~Flammability Control System~~	~~B, C~~	~~B~~

The RSS is designed such that it does not degrade the capability of the interfacing systems. All equipment is qualified as Class 1E, consistent with the safety-related interfaces.

Separation and isolation is preserved both mechanically and electrically in accordance with IEEE-279 and Regulatory Guide 1.75.

With regard to Paragraph 4.2 of IEEE-279, a single-failure event is assumed to have occurred to cause the evacuation of the control room. The RSS is not designed to

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temperature variable is considered a Type A variable since no credit is taken for automatic initiation in the safety analysis.

(j) Drywell Atmosphere Temperature

~~Surveillance~~ monitoring of the temperatures in the drywell is provided by multiple temperature sensors distributed throughout the drywell to detect local area "hot-spots" and to monitor the operability of the drywell cooling system. With this drywell air temperature monitoring system supplied by multiple temperature sensors throughout the drywell, the Regulatory Guide 1.97 requirements for monitoring of drywell air temperature are met and provides the ability to determine drywell bulk average temperature.

(k) Drywell/Wetwell Hydrogen/Oxygen Concentration

The Containment Atmospheric Monitoring System (CAMS) consists of two independent and redundant drywell/containment oxygen and hydrogen concentration monitoring channels. Emergency response actions regarding these variables are consistently directed toward minimizing the magnitude of these parameters (i.e., there are no safety actions which must be taken to increase the hydrogen/oxygen levels if they are low). Consequently, the two channel CAMS design provides adequate PAM indication, since, in the event that the two channels of information disagree, the operator can determine a correct and safe action based upon the higher of the two (in-range) indications.

Minimizing drywell containment oxygen and hydrogen concentrations are accomplished using manual operator actions through the use of containment venting and purging or the use of containment spray.

(l) Wetwell Atmosphere Air Temperature

Surveillance monitoring of temperatures in the wetwell is provided by multiple temperature sensors dispersed throughout the wetwell, therefore, the required indication of bulk average wetwell atmosphere temperature is satisfied.

(m) Standby Liquid Control System Flow

No flow indication is provided for the ABWR design. The positive displacement SLCS pumps are designed for constant flow. Any flow blockage or line break would be indicated by abnormal system pressure (high or low as compared to RCS pressure) following SLCS initiation. Changing neutron flux, SLCS pressure and SLCS tank level are substituted for SLCS flow and are considered adequate to verify proper system function. One channel of SLCS discharge pressure is provided in addition to the monitoring of neutron flux.

(n) Suppression Pool/Wetwell Water Level

Table 7.5-2 ABWR PAM Variable List (Continued)

Variable	Range Required	Type	Category	Discussion Section
SLCS Storage Tank Level	Top to Bottom	D	3	Subsection 7.5.2.1(2)(o)
SRV Position	Closed – Not Closed	D	2	
Feedwater Flow	0–110% Design Flow	D	3	
High Radioactivity Liquid Tank Level	Top to Bottom	D	3	
Standby Energy Status	Plant Specific	D	2	
Suppression Pool Water Temperature	4.4°C to 140°C	A, D	1	Subsection 7.5.2.1(2)(i)
Drywell Atmosphere Temperature	4.4°C to 226.7°C	D	1	Subsection 7.5.2.1(2)(j)
Drywell/Wetwell Hydrogen Concentration	0–30 Volume%	C	$\boxed{3}$ ~~1~~	Subsection 7.5.2.1(2)(k)
Drywell/Wetwell Oxygen Concentration	0–10 Volume%	C	$\boxed{2}$ ~~1~~	Subsection 7.5.2.1(2)(k)
Wetwell Atmosphere Temperature	4.4°C to 226.7°C	D	1	Subsection 7.5.2.1(2)(l)
Secondary Containment Airspace (effluent) Radiation Noble Gas	37 pBq/cm^3 to 37MBq/cm^3	C	2	
Containment Effluent Radioactivity—Noble Gas	37 pBq/cm^3 to 0.37μBq/cm^3	C	3	
Condensate Storage Tank Level	Top to Bottom	D	3	
Cooling Water Temperature to ESF System Components	4.4°C to 93.3°C	D	2	
Cooling Water Flow to ESF System Components	0–110% Design Flow	D	2	
Emergency Ventilation Damper Position	Open – Closed Status	D	2	
Service Area Radiation Exposure Rate	10^{-3} Gy/h to 10^2 Gy/h	E	3	
Purge Flows—Noble Gases and Vent Flow Rate	37 PBq/cm^3 to 0.37 Bq/cm^3 0–110% Vent Design Flow	E	2	
Identified Release Points—Particulates and Halogens	37 nBq/cm^3 to 3.7 mBq/cm^3 0–110% Vent Design Flow	E	3	

Table 7.5-6 Design Basis Accidents

Event Description	NSOA Event Figure No.	Tier 2 Section No.	Manual Action Variables*
Control Rod Ejection Accident	15A.6-28	15.4.8	None†
Control Rod Drop Accident	15A.6-29	15.4.9	P_{RPV}, L_{RPV}, ϕ
Control Rod Withdrawal Error Power Operation	15A.6-30	15.4.2	None†
Fuel Handling Accident	15A.6-31	15.7.4	R_{2C}
Loss-of-Coolant Accident Resulting from Spectrum of Postulated Piping Breaks within the RCPB Inside Containment	15A.6-32	15.6.5	H_{2C}, O_{2C}, L_{RPV}, L_{SP}, P_{RPV}, P_{DW} Ø
Small, Large, Steam and Liquid Piping Breaks Outside Containment	15A.6-33	15.6.4	T_{SP}, P_{RPV}, L_{RPV}
Abnormal Startup of Idle Reactor Internal Pump	15A.6-38	15.4.4	P_{RPV}, L_{RPV}
Recirculation Flow Control Failure—All RIPs Runout	15A.6-39	15.4.5	Ø, L_{RPV}
Recirculation Flow Control Failure—All RIPs Runback	15A.6-40	15.3.2	L_{RPV}
Trip of All RIPs	15A.6-41	15.3.1	P_{RPV}, L_{RPV}
Loss of RHR Shutdown Cooling	15A.6-42	15.2.9	T_{RPV}
RHR Shutdown Cooling Increased Cooling	15A.6-43	15.1.6	T_{RPV}
Feedwater Controller Failure Runout of Two Feedwater Pumps	15A.6-44	15.1.2	P_{RPV}, L_{RPV}
Pressure Regulatory Failure—Opening of All Bypass and Control Valves	15A.6-45	15.1.3	P_{RPV}, L_{RPV}
Pressure Regulatory Failure—Closure of All Bypass and Control Valves	15A.6-46	15.2.1	T_{SP}, P_{RPV}, L_{RPV}
Main Turbine Trip with Bypass Failure	15A.6-48	15.2.3	T_{SP}, P_{RPV}, L_{RPV}
Generator Load Rejection with Bypass Failure	15A.6-49	15.2.2	T_{SP}, P_{RPV}, L_{RPV}
Misplaced Fuel Bundle Accident	15A.6-50	15.4.7	None
Reactor Internal Pump Seizure	15A.6-51	15.3.3	P_{RPV}, L_{RPV}
Reactor Internal Pump Shaft Break	15A.6-52	15.3.4	P_{RPV}, L_{RPV}

* See Table 7.5-9 for Definition of Symbols.

† Analysis indicates not plausible.

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Table 7.5-9 Definition of Symbols for Tables 7.5-4 Through 7.5-8

T_{SP}	—	Suppression Pool Temperature
T_{DW}	—	Drywell Temperature
T_{RPV}	—	Reactor Water Temperature
P_{RPV}	—	RPV Pressure
P_{WW}	—	Wetwell Pressure
L_{RPV}	—	RPV Level
L_{SP}	—	Suppression Pool Level
~~Ø~~	~~—~~	~~Neutron Flux~~
H_{2C}	—	Drywell/Wetwell Hydrogen Concentration
O_{2C}	—	Drywell/Wetwell Oxygen Concentration
~~P_{DW}~~	~~—~~	~~Drywell Atmospheric Pressure~~
T_{2C}	—	Temperature—Secondary Containment
R_{2C}	—	Radiation Level—Secondary Containment
L_{2C}	—	Sump Level—Secondary Containment
R_E	—	Exhaust Vent Radiation Level
L_C	—	Drywell Level
R_C	—	Radiation Level-Primary Containment

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Table 7A-1 List of Equipment Interface with Essential MUX Signals (Continued)

Device	Div	Description
U41-C623C	3	MCR RECIRC SUPP FAN (C)
U41-C623G	3	MCR RECIRC SUPP FAN (G)
U41-D101	1	RCIC PUMP ROOM HVH
U41-D102	3	HPCF PUMP (C) ROOM HVH
U41-D103	1	RHR PUMP (A) ROOM HVH
U41-D104	3	RHR PUMP (C) ROOM HVH
U41-D105	2	RHR PUMP (B) ROOM HVH
U41-D106	2	HPCF PUMP (B) ROOM HVH
U41-D107	3	Rm. 436 ~~FCS ROOM (A)~~ HVH
U41-D108	2	~~FCS ROOM~~ (B) HVH
U41-D109	1	Rm. 425 FPC PUMP (A) ROOM HVH
U41-D110	2	FPC PUMP (B) ROOM HVH
U41-D111	3	SGTS ROOM HVH (C)
U41-D112	2	SGTS ROOM HVH (B)
U41-D113	1	CAMS (A) ROOM HVH
U41-D114	2	CAMS (B) ROOM HVH
U41-F001A	1	AO VLV - R/A SUP ISO VLV
U41-F001B	2	AO VLV - R/A SUP ISO VLV
U41-F002A	1	AO VLV - R/A EXH ISO (A)
U41-F002B	2	AO VLV - R/A EXH ISO (B)
U41-F003A	1	MO VALVE
U41-F003B	2	MO VALVE
U41-F003C	3	MO VALVE
U41-F004A	1	MO VALVE
U41-F004B	2	MO VALVE
U41-F004C	3	MO VALVE
U41-F005A	1	MO VALVE
U41-F005B	2	MO VALVE
U41-F005C	3	MO VALVE
U41-TE052	1	TEMP ELEMENT
U41-TE056	2	TEMP ELEMENT
U41-TE060	3	TEMP ELEMENT
U41-TE103B	2	TEMP ELEMENT
U41-TE103C	3	TEMP ELEMENT

Table 8.3-1 D/G Load Table—LOCA + LOPP

Sys. No	Load Description	Rating (kW)	Generator Connected Loads (kW) A (Div I)	B (Div II)	C (Div III)	Note*
—	Motor operated Valves	231x3	X	X	X	(2)
C12	FMCRD (@0.25 pF)	210x1 (840kV·A)	210	—	—	
C41	SLC Pump	45x2	45	45	—	
E11	RHR Pump	540x3	540	540	540	
	Fill Pump	3.7x3	X	X	X	
E22	HPCF Pump	1400x2	—	1400	1400	
P21	RCW Pump	370x4	740	740	—	
		280x2	—	—	560	
P25	HECW Pump	22x5	22	44	44	
	HECW Refrigerator	135x5	135	270	270	
P41	RSW Pump	270x6	540	540	540	
R23	P/C Transf. Loss	42.1x6	84.2	84.2	84.2	
R42	DC 125V CHGR Div. I	70x1	70	—	—	
	Div. II, III, IV	34x3	—	68	34	
	125VDC stby charger		70	—	34	(3)
R46	Vital CVCF		—	—	—	
	(Div. 1, 2, 3)	20x3	20	20	20	
	(Div. 4)	20		20		
R47	Transf. C/R Inst	20x6	40	40	40	
R52	Lighting	100x3	100	100	100	
T22	SGTS Fan	18.5x2	—	18.5	18.5	
	SGTS Heater	10x6	—	30	30	
~~T49~~	~~FCS Heater~~	~~130x2~~		~~130~~	~~130~~	
	~~FCS Blower~~	~~12x2~~		~~12~~	~~12~~	
U41	MCR HVAC Fans B-C	74.5x4	—	149	149	(5)
	MCR Recirc Fans B-C	14x4	—	28	28	(5)
	C/B Elec Equip Area HVAC Fans A-C	14x6	28	28	28	(5)
	R/B DG/Elec Equip Area HVAC Fans A-C	84x6	168	168	168	(5)
	R/B DG Room Emergency Supply Fans A-C	46.5x6	93	93	93	(5)
	R/B Equip Area Room Coolers A-C		89	107	84	(5)

Table 8.3-3 Notes for Tables 8.3-1 and 8.3-2

(1) – : shows that the load is not connected to the switchgear of this division.

X : shows that the load is not counted for D/G continuous output calculation by the reasons shown on other notes.

(2) "Motor operated valves" are operated only 30–60 seconds. Therefore they are not counted for the DG continuous output calculation.

(3) Div. IV battery charger is fed from Div. II motor control center.

(4) Load description acronyms are interpreted as follows:

C/B	Control Building	HX	Heat Exchanger
COMP	Computer	IA	Instrument Air
CRD	Control Rod Drive	MCR	Main Control Room
		MUWC	Make Up Water System (condensed)
CVCF	Constant Voltage Constant Frequency	NPSS	Nuclear Protection Safety System
DG	Diesel Generator	R/B	Reactor Building
~~FCS~~	~~Flammability Control System~~	RCW	Reactor Cooling Water (building)
		RHR	Residual Heat Removal
FMCRD	Fine Motion Control Rod Drive	RSW	Reactor Service Water
HECW	Emergency Cooling Water	SGTS	Standby Gas Treatment
HPCF	High Pressure Core Flooder	SLC	Standby Liquid Control

(5) Redundant units, one unit of a division operates and one unit is in standby in case the operating unit shuts down. Total connected load is shown on the table, but operating loads are half these amounts.

Table 8.3-4 D/G Load Sequence Diagram Major Loads

	Mode	Div.	Block 1 (20 s)	Block 2 (30 s)	Block 3 (35 s)	Block 4 (40 s)	Block 5 (45 s)	Block 6 (50 s)	Block 7 (55 s)	Block 8 (60 s)	Block 9 (After 65 s)	
Non-Accident Loads	LOPP	I	MOV Inst. Tr Lighting FMCRD*	DG HVAC	RCW Pump HECW Pump	RCW Pump	RSW Pump R/B Emer. HVAC C/B Emer. HVAC	RSW Pump	SGTS	Chargers CVCFs	SLC Pump HECW Refrig	RHR Pump
Non-Accident Loads	LOPP	II	MOV Inst. Tr Lighting	DG HVAC	RCW Pump HECW Pump	RCW Pump MCR HVAC	RSW Pump R/B Emer. HVAC C/B Emer. HVAC	RSW Pump	SGTS	Chargers CVCFs	SLC Pump HECW Refrig	RHR Pump
Non-Accident Loads	LOPP	III	MOV Inst. Tr Lighting	DG HVAC	RCW Pump HECW Pump	RCW Pump MCR HVAC	RSW Pump R/B Emer. HVAC C/B Emer. HVAC	RSW Pump		Chargers CVCFs	HECW Refrig	RHR Pump
LOCA Loads	LOCA & LOPP	I	MOV Inst. Tr Lighting FCMRD*	RHR Pump DG HVAC	RCW Pump HECW Pump	RCW Pump	RSW Pump R/B Emer. HVAC C/B Emer. HVAC	RSW Pump	SGTS	Chargers CVCFs	SLC Pump HECW Refrig	FCS
LOCA Loads	LOCA & LOPP	II	MOV HPCF Pump Inst. Tr Lighting	RHR Pump DG HVAC	RCW Pump HECW Pump	RCW Pump MCR HVAC	RSW Pump R/B Emer. HVAC C/B Emer. HVAC	RSW Pump	SGTS	Chargers CVCFs	SLC Pump HECW Refrig	FCS
LOCA Loads	LOCA & LOPP	III	MOV HPCF Pump Inst. Tr Lighting	RHR Pump DG HVAC	RCW Pump HECW Pump	RCW Pump MCR HVAC	RSW Pump R/B Emer. HVAC C/B Emer. HVAC	RSW Pump		Chargers CVCFs	HECW Refrig	

* FMCRDs are the only Non-Class 1E loads on the DG buses.

Table 9.2-4b Reactor Building Cooling Water Division B

Operating Mode/Components	Normal Operating Conditions		Shutdown at 4 Hours		Shutdown at 20 Hours		Hot Standby (No Loss of AC)		Hot Standby (Loss of AC)		Emergency (LOCA) (Suppression Pool at 97°C	
	Heat*	Flow*	Heat	Flow	Heat	Flow	Heat	Flow	Heat	Flow	Heat	Flow
Essential												
Emergency Diesel Generator B	—	—	—	—	—	—	—	—	13.40	229	13.40	229
RHR Heat Exchanger B	—	—	108.02	1,199	34.75	1,199	—	—	25.54	1,199	89.18	1,199
Others (essential)†	6.28	360	6.70	360	6.70	360	6.28	360	7.12	360	7.95	360
Non-Essential												
CUW Heat Exchanger‡	20.10	159	—	159	—	159	20.10	159	20.93	159	—	—
FPC Heat Exchanger B*f*	7.12	279	7.12	279	7.12	279	7.12	279	7.12	279	9.63	279
Inside Drywell**	5.44	279	6.28	279	5.40	279	5.40	279	2.51	279	—	—
Others (non-essential)††	2.93	159	1.47	159	1.47	159	1.47	159	0.33	9.1	—	9.1
Total Load	41.87	1,236	129.79	2,435	55.27	2,435	40.19	1,236	77.04	2,514	120.16	2,076

* Heat in GJ/h; flow in m^3/h, sums may not be equal due to rounding. Rm. 425 / 436

† HECW refrigerator, room coolers (RHR, HPCF, SGTS, ~~FCS,~~ CAMS), CAMS cooler, HPCF and RHR motor and mechanical seal coolers.

‡ The heat transferred from the CUW heat exchanger at the start of cooldown is appreciable, but during the critical last part of a cooldown, the heat removed is very little because the temperature difference between the reactor water and the RCW System is small. Sometimes, the operators may remove the CUW heat exchangers from service during cooldown. Thus, the heat removed varies from about that during normal operation at the start of cooldown to very little at the end of cooldown.

f Includes FPC room cooler.

** Drywell (B) and RIP coolers.

†† Reactor Building sampling coolers; LCW sump coolers (in drywell and reactor building), RIP MG sets and CUW pump coolers.

Table 9.2-4c Reactor Building Cooling Water Division C

Operating Mode/Components	Normal Operating Conditions		Shutdown at 4 Hours		Shutdown at 20 Hours		Hot Standby (No Loss of AC)		Hot Standby (Loss of AC)		Emergency (LOCA) (Suppression Pool at 97°C	
	Heat*	Flow*	Heat	Flow	Heat	Flow	Heat	Flow	Heat	Flow	Heat	Flow
Essential												
Emergency Diesel Generator C	—	—	—	—	—	—	—	—	13.40	229	13.40	229
RHR Heat Exchanger C	—	—	108.02	1,199	34.75	1,199	—	—	25.54	1,199	89.18	1199
Others (essential)†	6.28	360	6.70	360	6.70	360	6.28	360	6.70	360	7.12	360
Non-Essential												
Others (non-essential)‡	20.51	422	19.26	422	7.54	422	20.51	422	0.54	50	0.75	50
Total Load	26.80	782	133.98	1,981	48.57	1,981	26.80	782	46.05	1838	110.53	1838

Rm. 425 / 436

* Heat in GJ/h; flow in m^3/h, sums may not be equal due to rounding.

† HECW refrigerator, room coolers, motor coolers, and mechanical seal coolers for RHR and HPCF, FCS room cooler, SGTS room cooler.

‡ Instrument and service air coolers, CRD pump oil cooler, radwaste components, HSCR condenser, and turbine building sampling coolers.

9.4.5.2 R/B Safety-Related Equipment HVAC System

9.4.5.2.1 Design Bases

9.4.5.2.1.1 Safety Design Bases

The R/B Safety-Related Equipment HVAC System is designed to provide a controlled temperature environment to ensure the continued operation of safety-related equipment in harsh environment under accident conditions. The rooms cooled by the Safety-Related Equipment HVAC System are maintained at negative pressure relative to atmosphere by the secondary containment HVAC System during the normal operating mode, and by standby gas treatment system in isolation mode.

The systems and components are Seismic Category I and are located in the Reactor Building, separate and independent compartments of a Seismic Category I structure that is tornado-missile, and flood protected.

Fire protection has been evaluated and is described in Subsection 9.5.1.

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9.4.5.2.1.2 Power Generation Design Bases

The system is designed to provide an environment with controlled temperature and humidity to ensure both the comfort and safety of plant personnel and the integrity of Reactor Building equipment. The systems are designed to facilitate periodic inspection of the principal system components.

9.4.5.2.2 System Description

The R/B Safety-Related Equipment HVAC System consists of 12 safety-related fan coil units (FCU) of division A, B, or C. Each FCU has the responsibility to cool one safety-related equipment room in the secondary containment. The safety-related equipment HVAC (fan coil units) system P&ID is shown in Figure 9.4-3. Space temperatures are maintained less than 40°C normally and less than 66°C during pump operation:

(1) RHR(A) pump room

(2) RHR(B) pump room

(3) RHR(C) pump room

(4) HPCF(B) pump room

(5) HPCF(C) pump room

(6) RCIC pump room

(7) ~~FCS(B)~~ Rm. 425 room

Rm. 436

(8) F room

(9) SGTS(B) room

(10) SGTS(C) room

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(11) CAMS(A) room

(12) CAMS(B) room

9.4.5.2.2.1 RHR, HPCF and RCIC Pump Room HVAC Systems

The FCU's automatically start when RHR pumps, HPCF pumps, and RCIC turbine are started. These rooms are normally cooled by the Secondary Containment HVAC System. The fan coil units are open ended and recirculate cooling air within the space served. Space heat is removed by cooling water passing through the coil section. Divisional Reactor Building Cooling Water (RCW) is used as the cooling medium. The units are fed from the same divisional power as that for the equipment being served. Drain pan discharge (condensate) is routed to a floor drain located within the room.

Rm. 425 / 436

9.4.5.2.2.2 ~~FCS~~ Room HVAC System

Cooling of Rm. 425 / 436 is automatically initiated upon receipt of a secondary containment isolation signal.

Cooling of the ~~FCS~~ rooms are ~~automatically initiated upon receipt of a secondary containment isolation signal or a manual FCS start signal.~~

Rm. 425 / 436

These rooms are cooled by the Secondary Containment HVAC System during normal conditions. The units are open ended and recirculate cooling air within the space served. Space heat is removed by cooling water passing through the coil section. Divisional RCW is used as the cooling medium. The units are fed from the same divisional power as that for the ~~FCS~~ being served. Humidity is not specifically maintained at a set range, but is automatically determined by the surface temperature of the cooling coil. Drain pan discharge (condensate) is routed to a floor drain located within the room.

9.4.5.2.2.3 SGTS and CAMS HVAC Systems

Cooling of the SGTS and CAMS rooms are automatically initiated upon receipt of a secondary containment isolation signal.

These rooms are cooled by the Secondary Containment HVAC System during normal conditions. The units are open ended and recirculate cooling air within the space served. Space heat is removed by cooling water passing through the coil section. Divisional RCW is used as the cooling medium. The units are fed from the same divisional power as that for the equipment being served. Drain pan discharge (condensate) is routed to a floor drain located within the room.

Table 9.4-4d Not Used

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Table 9.4-4e HVAC System Component Descriptions — Safety-Related Fan Coil Units (Response to Question 430.243)

Safety-Related Fan Coil Units	Capacity (MJ/h)
HPCF Pump Room Div B	460.55
HPCF Pump Room Div C	460.55
RHR Pump Room Div A	307.73
RHR Pump Room Div B	307.73
RHR Pump Room Div C	307.73
RCIC Pump Room Div A	69.08
Rm. 425 Room Div B	54.85
FCS Room Div C	54.85
Rm. 436 Room Div A	83.74
CAMS Room Div B	83.74
SGTS Room Div B	16.75
SGTS Room Div C	16.75

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9A.4.1.3.21 Emergency Electrical Room B (Rm No. 326)

(1) Fire Area—F3201

(2) Equipment: See Table 9A.6-2

Safety-Related	Provides Core Cooling
Yes, D2	Yes, D2

(3) Radioactive Material Present—None.

(4) Qualifications of Fire Barriers—The wall common with the emergency electrical room C (Rm 337), the wall common with corridor C (Rm. 335), the wall common with the corridor B (Rm 321), the portion of the wall common with elevator and stair tower 3, the wall common with RIP Panel Room (320), the exterior wall, the floor and the ceiling are of 3 h fire-resistive concrete construction. Two 3 h fire-resistive double doors provide access and egress from the emergency electrical room C (Rm 337) and RIP Panel room (Rm 320). ~~Two piping spaces are entered to this room at elevation 10300 mm to facilitate the FCS piping to the next elevation.~~ The walls of these ~~piping~~ spaces are fire barrier of 3 h fire resistive concrete construction.

(5) Combustibles Present:

Fire Loading	Total Heat of Combustion (MJ)
Cable Tray	1454 MJ/m^2 ECLL (1454 MJ/m^2 maximum average) applies.

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull station at Col. 6.9-E.2 and 1.9-F.5.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel with an electrically safe nozzle.	Col. 6.9-E.2 & 1.9-F.5/Manual
ABC hand extinguishers	Col. 6.9-E.2 & 1.9-F.5/Manual

(8) Fire Protection Design Criteria Employed:

(9) Consequences of Fire—The postulated fire assumes the loss of the function. The provisions for core cooling systems backup are defined in Subsection 9A.2.5.

Smoke from a fire will be removed by the normal HVAC System operating in its smoke removal mode.

(10) Consequences of Fire Suppression—Suppression extinguishes the fire. Refer to Section 3.4, "Water Level (Flood) Design", for the drain system.

(11) Design Criteria Used for Protection Against Inadvertent Operation, Careless ~~Operation~~ or Rupture of the Suppression System:

(a) Location of the manual suppression system external to the room

(b) Provision of raised supports for the equipment

(c) Refer to Section 3.4, "Water Level (Flood) Design", for the drain system.

(d) ANSI B31.1 standpipe (rupture unlikely)

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(12) Fire Containment or Inhibiting Methods Employed:

(a) The functions are located in a separate fire-resistive enclosure.

(b) The means of fire detection, suppression and alarming are provided and accessible.

(13) Remarks—The room contains cable in conduit only.

9A.4.1.4.8 Corridor C (Equipment Entry) (Rm No. 430)

(1) Fire Area—F4301

(2) Equipment: See Table 9A.6-2

Safety-Related	Provides Core Cooling
Yes, D3	Yes, D3

(3) Radioactive Material Present—None that can be released as a result of fire.

(4) Qualifications of Fire Barriers—The walls common with the C diesel generator room (Rm 432), valve room (C) (Rm 431), corridor B (Rm 420), the ~~Flammability Control System~~ Rm. 436 room (Rm 436) and the exterior wall serve as fire barriers and are of 3 h fire-resistive concrete construction. The floor is also a fire barrier to limit the size of the fire areas below and to protect the

lower regions of the building, which contains the majority of the ESF equipment. The walls are concrete and are not rated as they are internal to fire area F4301. A section of the ceiling common to fire areas F4300, F1300 and F3300 above is of 3 h fire-resistive concrete construction. The remainder of the ceiling is not fire rated as it is internal to fire area F4310. Access to the corridor is provided from corridors A and B via 3 h fire-resistive doors. The corridor provides direct access to the electrical and instrumentation penetration room (Rm 433) through a nonrated door and valve room (C) (Rm 431) and the ~~Flammability Control System~~ Rm 436 (Rm 436) through 3 h fire-resistive doors. There is an open hatch to the floors above. A large steel non-fire-rated door provides access to the reactor building for moving in fuel and other large loads.

(5) Combustibles Present:

Fire Loading	Total Heat of Combustion (MJ)
Cable Tray	727 MJ/m^2 NCLL (727 MJ/m^2 maximum average) applies
Lubricant Fuel Oil	Could be a variable due to possible lubricant, and fuel oil leaks in transient. Deluge sprinkler system provided.

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(6) Detection Provided—Class A supervised POC in the room and manual alarm pull stations at 5.9-F.2 and 2.1-F.1.

(7) Suppression Available:

Type	Location/Actuation
Ordinary hazard deluge sprinkler having a water density of 6.1 L/min/m^2 and a coverage of 9.3 m^2 per head	Hatch Area/Manual
Standpipe and hose reel	Col. 5.9-F.2 & 2.1-F.1/Manual
ABC hand extinguishers	Col. 5.9-F.2 & 2.1-F.1/Manual

(8) Fire Protection Design Criteria Employed:

(b) The means of fire detection, suppression and alarming are provided and accessible.

(13) Remarks—Although the areas surrounding the adjacent diesel generator room are of the same safety division, the diesel generator room is designated as a separate fire area due to the relatively large amounts of lubricating and fuel oil present.

9A.4.1.4.11 ~~Flammability Control System~~ Rm. 436 Room (Div. 3) (Rm No. 436)

(1) Fire Area—F4320

(2) Equipment: See Table 9A.6-2

Safety-Related	Provides Core Cooling
Yes, D1, and D2	No

(3) Radioactive Material Present—None that can be released as a result of fire.

(4) Qualifications of Fire Barriers—The floor and interior and exterior walls are fire barriers and are of 3 h fire-resistive concrete construction. The ceiling is formed by the bottom of the spent fuel storage pool (F4301) and is a 3 h fire barrier. Personnel access is provided via a 3 h fire-resistive door from corridor C (Rm 430).

(5) Combustibles Present:

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Fire Loading	Total Heat of Combustion (MJ)
Cable Tray	727 MJ/m^2 NCLL (727 MJ/m^2 maximum average) applies

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull station at Col. 5.9-F.2 and 2.1-F.1.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel	Col. 5.9-F.2 & 2.1-F.1/Manual
ABC hand extinguishers	Col. 5.9-F.2 & 2.1-F.1/Manual

Rm. 425

(4) Qualifications of Fire Barriers—The walls common with the ~~Flammability Control System~~ Room (Rm 425), the elevator and stair well walls, the Diesel Generator B Room (Rm 423) and the ECCS Valve B Room (Rm 421) serve as fire barriers and are of 3 h fire-resistive concrete construction. The floor is also a fire barrier to limit the size of the fire areas below and to protect the lower regions of the building, which contains the majority of the ESF equipment. The walls common with the E and I Penetration Room (Rm 422) and the ceiling are fire-resistive concrete but are nonrated as they are internal to fire area F4201. Access to the corridor is provided from corridor D (Rm 445), corridor C (Rm 430) and stairs and elevator No.3. A 3 h fire damper is installed in the HVAC duct (located next to the elevator) where it passes through the fire barrier floor to the division 2 areas on the level below. This fire barrier divides the division 2 area of the building to limit the magnitude of possible damage due to a single fire.

(5) Combustibles Present:

Fire Loading	Total Heat of Combustion (MJ)
Cable Tray	727 MJ/m^2 NCLL (727 MJ/m^2 maximum average) applies

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(6) Detection Provided—Class A supervised POC in the room and manual alarm pull stations at 5.9-F.2 and 2.1-F.1.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel	Col. 5.9-F.2 & 2.1-F.1/Manual
ABC hand extinguishers	Col. 5.9-F.2 & 2.1-F.1/Manual

(8) Fire Protection Design Criteria Employed:

(a) The function is located in a separate fire resistive enclosure.

(b) Fire detection and suppression capability is provided and accessible.

(c) Fire stops are provided for cable tray and piping penetrations through rated fire barriers.

(9) Consequences of Fire—The postulated fire assumes the loss of the function. The provisions for core cooling systems backup are defined in Subsection 9A.2.5. Access is provided to the corridor from either end.

(b) The means of fire detection, suppression and alarming are provided and accessible.

(13) Remarks—None.

9A.4.1.4.14 Not Used

9A.4.1.4.15 Diesel Generator B Room (Rm No. 423)

(1) Fire Area—F4200

(2) Equipment: See Table 9A.6-2

Safety-Related	Provides Core Cooling
Yes, D2	Yes, D2

(3) Radioactive Material Present—None.

(4) Qualifications of Fire Barriers—The building exte Rm 425 ls, the walls common with Corridor B (Rm 420), the wall common with ~~FCS~~-room (Rm 425), the wall common with stair wells (Rms 193 and 329), and the floor are of 3 h fire resistive concrete construction. The interior partition walls, and ceiling are not fire rated as they are internal to fire F4200. The ceiling of the room is not a fire barrier as the fan room is located directly above this diesel generator room. The exterior wall of the room has a removable section for removal of equipment from the diesel generator room. Access to this room is provided from the Clean Area Access C/D (Rm 426) through a 3 h fire-rated door and through the removable section of the external wall.

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(5) Combustibles Present:

Fire Loading	Total Heat of Combustion (MJ)
Cable Tray Lubricating Oil Fuel Oil	Could be variable due to possible oil leaks. Foam sprinkler system provided.

(6) Detection Provided—Class A supervised rate-compensated thermal detectors and infrared detectors. The detection system is a cross-zoned system requiring two detectors, one of each in each zone. Each detector initiates a local alarm upon sensing fire. The second detector alarm provides fire confirmation, which opens the preaction valve and initiates the system alarm in the control room. There is a manual pull stations at Col. 1.4-C.8.

(a) The function is located in a separate fire resistive enclosure.

(b) Fire detection and suppression capability is provided and accessible.

(c) Fire stops are provided for cable tray and piping penetrations through rated fire barriers.

(9) Consequences of Fire—The postulated fire assumes the loss of the function. The valves are spatially separated and are designed to fail closed on loss of actuation power. The provisions for core cooling systems backup are discussed in Subsection 9A.2.5.

Smoke from a fire will be removed by the normal HVAC System operating in its smoke removal mode.

This page was replaced by page C-207

(10) Consequences of Fire Suppression—Suppression extinguishes the fire. Refer to Section 3.4, "Water Level (Flood) Design", for the drain system.

(11) Design Criteria Used for Protection Against Inadvertent Operation, Careless Operation or Rupture of the Suppression System.

(a) Location of the manual suppression system in rooms external to the rooms containing safety-related equipment

(b) Provision of raised supports for the equipment

(c) Refer to Section 3.4, "Water Level (Flood) Design", for the drain system.

(d) ANSI B31.1 standpipe (rupture unlikely)

(12) Fire Containment or Inhibiting Methods Employed:

(a) The functions are located in a separate fire-resistive enclosure.

(b) The means of fire detection, suppression and alarming are provided and accessible.

(13) Remarks—None

9A.4.1.4.27 ~~Flammability Control S~~ Rm. 425 Room (Rm No 425)

(1) Fire Area—F4230

(2) Equipment: See Table 9A.6-2

Safety-Related	Provides Core Cooling
Yes, D2	No

Figure 9A.4-4 Reactor Building Fire Protection at Elevation 12300 mm

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

Figure 9A.4-9 Reactor Building Fire Protection Section A-A

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

Nondivisional Equipment

Boron Storage Tank

Storage Tank Heating elements

Power Cabling for Storage Tank Heaters

The cabling is routed in separate conduit or trays for each division, separated from each other, to meet IEEE-384. Conduit will be embedded in concrete where feasible.

The electric drive motor and cabling for the redundant pumps are located more than 1.52 m apart. The injection valves and cabling are located more than 0.91 m apart centerline to centerline.

The control cables for Division 1 and 2 equipment are in separate conduit and separate from the power cables. The Division 1 power and control cabling is routed out of the Division 2 area to the Division 1 area by conduit embedded in the floor and wails.

Postulated fire damage to the electrical equipment in the SLC area could not inadvertently result in injection of boron because this can only be done by activation of a switch on the control room panel. Fire could damage the power cabling to the pump suction valves or to the pump motors preventing opening of valves or start of pump motors on command from the control room. However, the SLC equipment is not required for safe shutdown of the reactor, since it is redundant to the RPS.

~~9A.5.5.9 Flammability Control System~~

~~The flammability control system equipment is located in a large enclosed area at grade level at approximately 180 degrees azimuth. The rooms have a fire barrier floor and is completely surrounded by fire barrier walls and doors. There are large access doors to the outside at the centerline of the room.~~

~~The FCS is made up of two independent redundant divisions (Divisions 2 and 3), and each division is located in the fire area division 2 and 3 respectively. Each division has two suction isolation valves (inboard and outboard) and two return isolation valves (inboard and outboard). The inboard isolation valves are motor operated (MO) valves, and the outboard isolation valves are fail close (FC) air operated (AO) solenoid valves (two solenoids per valve). They are powered from division 1 and 4. Fire in either division may cause the inboard valves (Div. 2 or 3) to fail to operate, but the outboard isolation valves are still capable to isolate because they are powered from different divisions (Div. 1 and 4). Loss of a complete division is acceptable because FCS is made up of two independent redundant divisions mounted in two separated fire areas.~~

Table 9A.6-2 Fire Hazard Analysis Equipment Database Sorted by Room — Reactor Building (Continued)

Item No.	MPL No	Elect Div.	Elev. Location	Location Number Coord.	Location Alpha Coord.	Description	System Drawing	Room No.
577	C51-J001C	N	1500	3.8	A.2	ATIP DRIVE C	107E5074/0	215
578	C51-J009	N	1500	4.5	A.2	TIP PURGE EQUIPMENT	107E5074/0	215
579	C51-J011	N	1500	4.5	A.2	TIP PURGE VALVE	107E5074/0	215
580	C51-N006A	N	1500	3.8	A.2	TIP IN-CORE PROBE	107E5074/0	215
581	C51-N006B	N	1500	4.0	A.2	TIP IN-CORE PROBE	107E5074/0	215
582	C51-N006C	N	1500	4.2	A.2	TIP IN-CORE PROBE	107E5074/0	215
583	D21-RE018	N	1500	4.0	A.2	AREA RAD DETECTOR	299X701-171/0	215
584	C51-N005A	N	1500	4.0	A.8	OB PROXIMITY SWITCH	107E5074/0	216
585	C51-N005B	N	1500	4.0	A.8	OB PROXIMITY SWITCH	107E5074/0	216
586	C51-N005C	N	1500	4.0	A.8	OB PROXIMITY SWITCH	107E5074/0	216
587	D21-RE019	N	1500	4.0	A.5	AREA RAD DETECTOR	299X701-171/0	216
588	C51-J004A	N	1500	4.0	B.1	TIP BALL/SHR VLV ASM	107E5074/0	216
589	C51-J004B	N	1500	4.0	B.1	TIP BALL/SHR VLV ASM	107E5074/0	216
590	C51-J004C	N	1500	4.0	B.1	TIP BALL/SHR VLV ASM	107E5074/0	216
591	D21-RE023	N	-1700	5.0	B.1	AREA RAD DETECTOR	299X701-171/0	219
592	D21-RE022	N	-1700	2.0	E.9	AREA RAD DETECTOR	299X701-171/0	221
593	H23-P006*	N	-1700	2.6	F.0	MULTIPLEXER	----?----	221
594	H23-P007*	N	-1700	2.8	F.0	MULTIPLEXER	----?----	221
595	~~T49 F006B~~	~~2~~	~~800~~	~~2.8~~	~~E.5~~	~~MO-GATE-VALVE~~	~~107E6047/0~~	~~221~~
596	~~T49 F007B~~	~~2~~	~~800~~	~~2.8~~	~~E.5~~	~~AO-GATE-VALVE~~	~~107E6047/0~~	~~221~~
597	~~T49-F007B-1~~	~~1~~	~~800~~	~~2.8~~	~~E.5~~	~~SOLENOID-VALVE~~	~~107E6047/0~~	~~221~~
598	~~T49-F007B-2~~	~~4~~	~~800~~	~~2.8~~	~~E.5~~	~~SOLENOID-VALVE~~	~~107E6047/0~~	~~221~~
599	~~X-242~~	~~2~~	~~800~~	~~2.7~~	~~E.5~~	~~FCS RETURN~~	~~NT-1006643~~	~~221~~
600	E11-F008B	2	1200	2.1	D.2	MO GLOBE VALVE (PRET)	103E1797/1	222
601	E11-F021B	2	1200	2.1	D.2	MO GATE VALVE (RECIRC)	103E1797/1	222
602	E11-F031B	2	-1700	2.1	D.4	MO GLOBE VALVE (TEST)	103E1797/1	222
603	E11-POT304B	2	-1700	2.1	D.4	POS XMTR (FO31B)	103E1797/1	222
604	T31-F739B	2	2800	2.1	D.5	SO VALVE	107E6043/0	222
605	T31-F741B	2	-1700	2.1	D.5	SO VALVE	107E6043/0	222
606	T31-F743B	2	2800	2.1	D.5	SO VALVE	107E6043/0	222
607	T31-F745B	2	-1700	2.1	D.5	SO VALVE	107E6043/0	222
608	X-205	2	1200	2.1	D.4	RHR PUMP B TEST RET	795E880/3	222
609	X-322E	3	400	2.2	D.6	SUPP CHAMBER WATER LEV	107E6043/0	222
610	D21-RE010	N	-1700	3.8	F.1	AREA RAD DETECTOR	299X701-171/0	223

Table 9A.6-2 Fire Hazard Analysis
Equipment Database Sorted by Room — Reactor Building (Continued)

Item No.	MPL No	Elect Div.	Elev. Location	Location Number Coord.	Location Alpha Coord.	Description	System Drawing	Room No.
611	T31-TE053A	N	-1700	4.2	E.8	TEMP ELEMENT	107E6043/0	223
612	T31-TE053B	N	-1700	4.2	E.9	TEMP ELEMENT	107E6043/0	223
613	T31-TE053C	N	-1700	3.8	E.8	TEMP ELEMENT	107E6043/0	223
614	T31-TE053D	N	-1700	3.8	E.9	TEMP ELEMENT	107E6043/0	223
615	X-013B	2	1400	4.0	E.9	RIP PURGE WTR SUPPLY	795E882/4	223
616	X-020B	N	1400	4.0	E.9	CRD INSERT	796E367/3	223
617	D21-RE021	N	-1700	4.0	F.8	AREA RAD DETECTOR	299X701-171/0	225
618	T31-TE053F	N	-1700	3.7	F.5	TEMP ELEMENT	107E6043/0	225
619	T31-TE053H	N	-1700	3.7	F.3	TEMP ELEMENT	107E6043/0	225
620	B31-FIS001A	N	-1700	3.4	F.2	FLOW IND SWITCH	107E5194/0	225
621	B31-FIS001B	N	-1700	3.4	F.2	FLOW IND SWITCH	107E5194/0	225
622	B31-FIS001C	N	-1700	3.4	F.2	FLOW IND SWITCH	107E5194/0	225
623	B31-FIS001D	N	-1700	3.4	F.2	FLOW IND SWITCH	107E5194/0	225
624	B31-FIS001E	N	-1700	3.4	F.2	FLOW IND SWITCH	107E5194/0	225
625	B31-FIS001F	N	-1700	3.4	F.2	FLOW IND SWITCH	107E5194/0	225
626	B31-FIS001G	N	-1700	3.4	F.2	FLOW IND SWITCH	107E5194/0	225
627	B31-FIS001H	N	-1700	3.4	F.2	FLOW IND SWITCH	107E5194/0	225
628	B31-FIS001J	N	-1700	3.4	F.2	FLOW IND SWITCH	107E5194/0	225
629	B31-FIS001K	N	-1700	3.4	F.2	FLOW IND SWITCH	107E5194/0	225
630	B31-TI001A	N	-1700	3.4	F.2	TEMP INDICATOR	107E5194/0	225
631	B31-TI001B	N	-1700	3.4	F.2	TEMP INDICATOR	107E5194/0	225
632	U41-D131B	N	-1700	3.5	F.4	RIP/FMCRD CP RM FCU B	107E5189/0	225
633	E11-F008C	3	1200	5.9	D.2	MO GLOBE VALVE (PRET)	103E1797/1	230
634	E11-F021C	3	1200	5.9	D.2	MO GATE VALVE (RECIRC)	103E1797/1	230
635	E11-F031C	3	-1700	5.9	D.4	MO GLOBE VALVE (TEST)	103E1797/1	230
636	E11-POT304C	3	-1700	5.9	D.4	POS XMTR (FO31C)	103E1797/1	230
637	T31-F739C	3	2800	5.9	D.5	SO VALVE	107E6043/0	230
638	T31-F741C	3	-1700	5.9	D.5	SO VALVE	107E6043/0	230
639	~~T49-F006C~~	~~3~~	800	~~5.8~~	~~D.5~~	~~MO GATE VALVE~~	~~107E6047/0~~	~~230~~
640	~~T49-F007A-1~~	~~1~~	800	~~5.8~~	~~D.5~~	~~SOLENOID VALVE~~	~~107E6047/0~~	230
641	~~T49-F007A-2~~	4	800	~~5.8~~	~~D.5~~	~~SOLENOID VALVE~~	~~107E6047/0~~	230
642	~~T49 F007C~~	3	800	~~5.8~~	~~D.5~~	~~AO GATE VALVE~~	~~107E6047/0~~	230
643	X-206	3	1200	5.9	D.4	RHR PUMP C TEST RET	795E880/3	230
644	X-322D	4	400	5.8	D.6	SUPP CHAMBER WATER LEV	107E6043/0	230

Table 9A.6-2 Fire Hazard Analysis Equipment Database Sorted by Room — Reactor Building (Continued)

Item No.	MPL No	Elect Div.	Elev. Location	Location Number Coord.	Location Alpha Coord.	Description	System Drawing	Room No.
1754	C51-K002B	2	13500	3.7	E.8	SRNM PREAMPLIFIER	107E5074/0	424
1755	C51-K002F	2	13500	3.7	E.8	SRNM PREAMPLIFIER	107E5074/0	424
1756	X-100B	N	13500	3.7	E.8	INTERNAL PUMP POWER	795E882	424
1757	X-102B	2	16400	3.8	E.9	CONTROL & INSTRUMENT	795E898	424
1758	X-102F	2	16400	3.7	E.8	CONTROL & INSTRUMENT	795E898	424
1759	X-103B	2	13500	3.9	E.9	COMPENSATION/INSTR LINE	107E6043/0	424
1760	X-105B	2	13500	3.8	E.8	NEUTRON DETECTOR	795E898	424
1761	T49-A001B	2	12300	3.1	F.7	RECOMBINER	107E6047/0	425
1762	T49-B001B	2	12300	3.1	F.7	SPRAY COOLER	107E6047/0	425
1763	T49-C001B	2	12300	3.1	F.7	BLOWER	107E6047/0	425
1764	T49-D001B	2	12300	3.1	F.7	WATER SEPARATOR	107E6047/0	425
1765	T49-D002B*	2	12300	3.1	F.7	RECOMB HEATER	107E6047/0	425
1766	T49-F003B	2	12300	3.1	F.7	MO GLOBE VALVE	107E6047/0	425
1767	T49-F004B	2	12300	3.1	F.7	MO GLOBE VALVE	107E6047/0	425
1768	T49-F008B	2	12300	3.1	F.6	MO GLOBE VALVE	107E6047/0	425
1769	T49-F009B	2	12300	3.1	F.7	MAN OPER GLOBE VALVE	107E6047/0	425
1770	T49-F010B	2	12300	3.1	F.7	MO GLOBE VALVE	107E6047/0	425
1771	T49-F013B	2	12300	3.1	F.7	MAN OPER GATE VALVE	107E6047/0	425
1772	T49-F014B	2	12300	3.1	F.7	MAN OPER GATE VALVE	107E6047/0	425
1773	T49-F016B	2	12300	3.1	F.7	MAN OPER GATE VALVE	107E6047/0	425
1774	T49-FT002B	2	12300	3.1	F.7	FLOW TRANSMITTER	107E6047/0	425
1775	T49-FT004B	2	12300	3.1	F.7	FLOW TRANSMITTER	107E6047/0	425
1776	T49-LS011B	2	12300	3.1	F.7	LEVEL SWITCH	107E6047/0	425
1777	T49-LS012B	2	12300	3.1	F.7	LEVEL SWITCH	107E6047/0	425
1778	T49-LS013B	2	12300	3.1	F.7	LEVEL SWITCH	107E6047/0	425
1779	T49-PT003B	2	12300	3.1	F.7	PRESS TRANSMITTER	107E6047/0	425
1780	T49-TE001B	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1781	T49-TE005B	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1782	T49-TE006B-1	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1783	T49-TE006B-2	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1784	T49-TE007B-1	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1785	T49-TE007B-2	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425

Table 9A.6-2 Fire Hazard Analysis Equipment Database Sorted by Room — Reactor Building (Continued)

Item No.	MPL No	Elect Div.	Elev. Location	Location Number Coord.	Location Alpha Coord.	Description	System Drawing	Room No.
1786	T49-TE008B-1	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1787	T49-TE008B-2	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1788	T49-TE009B-1	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1789	T49-TE009B-2	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1790	T49-TE010B-1	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1791	T49-TE010B-2	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1792	T49-TE011B	2	12300	3.1	F.7	TEMP ELEMENT	107E6047/0	425
1793	T49-TT609B	2	12300	3.1	F.7	TEMP TRANSMITTER	107E6047/0	425
1794	U41-D108	2	12300	2.7	F.5	FCS ROOM (B) HVH	107E5189/0	425
1795	H22-P039*	N	12300	5.2	E.6	CONT VESSEL PRESS LK TEST	----?----	430
1796	E11-F005C	3	14500	6.0	D.2	MO GATE VALVE (INJ)	103E1797/1	431
1797	E11-F011C	1	14550	5.9	D.6	MO GATE VALVE (ISOL)	103E1797/1	431
1798	E11-F017C	3	14500	5.9	D.8	MO GLOBE VALVE (SPRAY)	103E1797/1	431
1799	E11-F018C	3	14500	5.8	D.8	MO GLOBE VALVE (SPRAY)	103E1797/1	431
1800	E22-F003C	3	14500	6.0	D.4	MO GATE VALVE (INJ)	107E6008/0	431
1801	T31-F735C	3	14500	6.0	D.5	SO VALVE	107E6043/0	431
1802	X-030B	3	14500	5.8	D.8	RHR C DRYWELL SPRAY	795E880/3	431
1803	X-031B	3	14500	5.9	D.3	HPCF C SUPPLY	795E876/4	431
1804	X-032B	3	14500	5.9	D.2	RHR C LPCF	795E880/3	431
1805	X-033C	3	14550	5.8	D.7	RHR C SHTDN CLG SUCT	795E880/3	431
1806	P21-DPS033C	3	12300	6.5	F.0	DP SW (EMER DG C)	107E5112/0	432
1807	P21-DPS034C	3	12300	6.5	F.0	DP SW (EMER DG C)	107E5112/0	432
1808	R43-A104C*	3	12300	6.1	F.9	AIR STORAGE	SSAR FIG 9.5-8	432
1809	R43-A204C*	3	12300	6.1	F.7	AIR STORAGE	SSAR FIG 9.5-8	432
1810	R43-A401C*	3	12300	6.2	E.7	LUBE OIL SUPPLY TANK	H:87-1137	432
1811	R43-A501C*	3	17300	6.9	F.4	EXPANSION TANK	H:87-1137	432
1812	R43-C013C*	3	12300	6.1	E.4	FUEL OIL DRAIN UNIT	H:87-1137	432
1813	R43-C401C*	3	12300	6.1	E.9	DG C LUBE OIL PUMP	SSAR FIG 9.5-9	432
1814	R43-DPS091C*	3	12300	6.9	E.4	DIFF PRESS SWITCH	H:87-1137	432
1815	R43-J001C	3	12300	6.6	F.0	DIESEL GENERATOR	796E301	432
1816	R43-LIS191C*	3	12300	6.7	E.9	LEVEL IND SWITCH	H:87-1137	432
1817	R43-LS142C*	3	12300	6.2	F.4	LEVEL SWITCH	SSAR FIG 9.5-6	432

Table 9A.6-2 Fire Hazard Analysis Equipment Database Sorted by Room — Reactor Building (Continued)

Item No.	MPL No	Elect Div.	Elev. Location	Location Number Coord.	Location Alpha Coord.	Description	System Drawing	Room No.
1818	X-177	N	15900	5.4	E.3	PCV & D/F LEAK	----?----	433
1819	X-130C	3	13500	5.6	E.2	MAIN STEAM FLOW RATE	795E877	433
1820	X-143C	3	14700	5.6	E.2	REA WATER LEV & PRESS	795E877	433
1821	X-144C	3	12650	5.6	E.2	REA WATER LEV & PRESS	795E877	433
1822	C51-K002C	3	13500	4.7	E.8	SRNM PREAMPLIFIER	107E5074/0	435
1823	C51-K002G	3	13500	4.7	E.8	SRNM PREAMPLIFIER	107E5074/0	435
1824	C51-K002L	3	13500	4.7	E.8	SRNM PREAMPLIFIER	107E5074/0	435
1825	X-100C	N	13500	4.8	E.8	INTERNAL PUMP POWER	795E882	435
1826	X-100E	N	13500	4.2	E.8	INTERNAL PUMP POWER	795E882	435
1827	X-102C	3	16400	4.3	E.9	CONTROL & INSTRUMENT	795E898	435
1828	X-102G	3	13500	4.3	E.9	CONTROL & INSTRUMENT	795E898	435
1829	X-103C	3	16400	4.7	E.8	COMPENSATION/INSTR LINE	107E6043/0	435
1830	X-105C	3	13500	4.7	E.8	NEUTRON DETECTOR	795E898	435
1831	T49-A001A	3	12300	4.0	F.7	RECOMBINER	107E6047/0	436
1832	T49-B001A	3	12300	4.0	F.7	SPRAY COOLER	107E6047/0	436
1833	T49-C001A	3	12300	4.0	F.7	BLOWER	107E6047/0	436
1834	T49-D001A	3	12300	4.0	F.7	WATER SEPARATOR	107E6047/0	436
1835	T49-D002A*	3	12300	4.0	F.7	RECOMB HEATER	107E6047/0	436
1836	T49-F003A	3	12300	4.0	F.7	MO GLOBE VALVE	107E6047/0	436
1837	T49-F004A	3	12300	4.0	F.7	MO GLOBE VALVE	107E6047/0	436
1838	T49-F008A	3	12300	4.0	F.6	MO GLOBE VALVE	107E6047/0	436
1839	T49 F009A	3	12300	4.0	F.7	MAN OPER GLOBE VALVE	107E6047/0	436
1840	T49-F010A	3	12300	4.0	F.7	MO GLOBE VALVE	107E6047/0	436
1841	T49-F013A	3	12300	4.0	F.7	MAN OPER GATE VALVE	107E6047/0	436
1842	T49-F014A	3	12300	4.0	F.7	MAN-OPER-GATE-VALVE	107E6047/0	436
1843	T49-F016A	3	12300	4.0	F.7	MAN-OPER-GATE-VALVE	107E6047/0	436
1844	T49-FT002A	3	12300	4.0	F.7	FLOW TRANSMITTER	107E6047/0	436
1845	T49-FT004A	3	12300	4.0	F.7	FLOW TRANSMITTER	107E6047/0	436
1846	T49 LS011A	3	12300	4.0	F.7	LEVEL SWITCH	107E6047/0	436
1847	T49-LS012A	3	12300	4.0	F.7	LEVEL SWITCH	107E6047/0	436
1848	T49-LS013A	3	12300	4.0	F.7	LEVEL SWITCH	107E6047/0	436
1849	T49-PT003A	3	12300	4.0	F.7	PRESS TRANSMITTER	107E6047/0	436

Table 9A.6-2 Fire Hazard Analysis Equipment Database Sorted by Room — Reactor Building (Continued)

Item No.	MPL No	Elect Div.	Elev. Location	Location Number Coord.	Location Alpha Coord.	Description	System Drawing	Room No.
1850	T49-TE001A	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1851	T49-TE005A	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1852	T49-TE006A-1	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1853	T49-TE006A-2	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1854	T49-TE007A-1	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1855	T49-TE007A-2	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1856	T49-TE008A-1	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1857	T49-TE008A-2	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1858	T49-TE009A-1	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1859	T49-TE009A-2	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1860	T49-TE010A-1	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1861	T49-TE010A-2	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1862	T49-TE011A	3	12300	4.0	F.7	TEMP ELEMENT	107E6047/0	436
1863	T49-TT609A	3	12300	4.0	F.7	TEMP TRANSMITTER	107E6047/0	436
1864	U41-D107	3	12300	4.6	F.9	FCS ROOM (C) HVH	107E5189/0	436
1865	B21-F001A	N	12300	4.3	A.2	MO GATE VALVE (FW)	103E1791/1	440
1866	B21-F001B	N	12300	3.7	A.2	MO GATE VALVE (FW)	103E1791/1	440
1867	B21-F003A	1	12300	4.3	A.9	AO CHECK VALVE	103E1791/1	440
1868	B21-F003B	2	12300	3.7	A.9	AO CHECK VALVE	103E1791/1	440
1869	B21-F007A	N	12300	3.5	B.0	MO GATE VALVE (CUWINJ)	103E1791/1	440
1870	B21-F007B	N	12300	4.5	B.0	MO GATE VALVE (CUWINJ)	103E1791/1	440
1871	B21-F009A	1	16300	4.2	B.0	NO GLOBE VALVE (MSIV)	103E1791/1	440
1872	B21-F009A	2	16300	4.2	B.0	NO GLOBE VALVE (MSIV)	103E1791/1	440
1873	B21-F009B	1	16300	4.6	B.0	NO GLOBE VALVE (MSIV)	103E1791/1	440
1874	B21-F009B	2	16300	4.6	B.0	NO GLOBE VALVE (MSIV)	103E1791/1	440
1875	B21-F009C	1	16300	3.4	B.0	NO GLOBE VALVE (MSIV)	103E1791/1	440
1876	B21-F009C	2	16300	3.4	B.0	NO GLOBE VALVE (MSIV)	103E1791/1	440
1877	B21-F009D	1	16300	3.8	B.0	NO GLOBE VALVE (MSIV)	103E1791/1	440
1878	B21-F009D	2	16300	3.8	B.0	NO GLOBE VALVE (MSIV)	103E1791/1	440
1879	B21-F012	2	12300	4.0	B.0	MO GLOBE VALVE (DR)	103E1791/1	440
1880	B21-F013	N	12300	4.0	B.0	MO GLOBE VALVE (DR)	103E1791/1	440
1881	B21-F014	N	12300	4.0	B.0	MO GLOBE VALVE (DR)	103E1791/1	440
1882	B21-F015	N	12300	4.0	B.0	AO GLOBE VALVE	103E1791/1	440

Table 9A.6-2 Fire Hazard Analysis Equipment Database Sorted by Room — Reactor Building (Continued)

Item No.	MPL No	Elect Div.	Elev. Location	Location Number Coord.	Location Alpha Coord.	Description	System Drawing	Room No.
2210	X-104A	N	19000	5.8	C.4	FMCRD POSITION DISPLAY	103E1167	518
2211	X-104E	N	19000	5.8	C.7	FMCRD POSITION DISPLAY	103E1167	518
2212	D23-F001B	2	19000	2.2	D.8	SO VALVE	107E5139/1	520
2213	D23-F004B	2	19000	2.2	D.8	MO GLOBE VALVE	107E5139/1	520
2214	D23-F005B	2	19000	2.2	D.8	MO GLOBE VALVE	107E5139/1	520
2215	D23-PT007B	2	19000	2.2	D.8	PRESS TRANSMITTER	107E5139/1	520
2216	X-101F	N	19000	2.2	D.7	FMCRD POWER	103E1167	520
2217	X-142B	2	20100	2.3	D.9	REA WATER LEV & PRESS	795E877	520
2218	X-146B	2	19000	2.3	D.9	DRYWELL PRESSURE	107E6043/0	520
2219	X-147	2	20100	2.2	D.8	REA WATER LEV WIDE RNG	795E877	520
2220	X-162B	2	19000	2.2	D.8	CAMS GAMMA DETECTOR	10R281-431	520
2221	R24 MCC SC111	N	18100	2.4	F.1	MCC SC111 - R/B	107E5072/0	520
~~2222~~	~~T49-F001C~~	3	~~20100~~	~~2.7~~	~~E.4~~	~~MO GATE VALVE~~	~~107E6047/0~~	~~521~~
2223	~~T49 F002B~~	2	20100	~~2.7~~	~~E.4~~	~~AO GATE VALVE~~	~~107E6047/0~~	521
~~2224~~	~~T49 F002B-1~~	~~1~~	20100	~~2.7~~	~~E.4~~	~~SOLENOID VALVE~~	~~107E6047/0~~	~~521~~
2225	~~T49 F002B 2~~	4	20100	~~2.7~~	~~E.4~~	~~SOLENOID VALVE~~	~~107E6047/0~~	~~521~~
2226	X-081	N	19000	2.8	E.5	DRYWELL PURGE EXHAUST	107E6043/0	521
~~2227~~	~~X-082~~	2	20100	~~2.7~~	~~E.4~~	~~FCS INTAKE~~	~~NT-1006643~~	~~521~~
2228	T31-F004	2	19000	2.8	E.6	AO VALVE	107E6043/0	521
2229	T31-F005	2	19000	2.8	E.6	AO VALVE	107E6043/0	521
2230	T31-F008	1	19000	2.6	E.6	AO VALVE	107E6043/0	521
2231	T31-F009	1	19000	2.6	E.6	AO VALVE	107E6043/0	521
2232	T31-F010	2	19000	2.8	E.6	AO VALVE	107E6043/0	521
2233	U41-C206B	2	19700	1.5	F.3	DG(B) EMER SUPP FAN B	107E5189/0	522
2234	U41-C206F	2	19700	1.5	F.8	DG(B) EMER SUPP FAN F	107E5189/0	522
2235	U41-TIS058	2	19700	1.1	F.5	TEMP IND SW EMER EXH	107E5189/0	523
2236	R43-P001B*	2	19700	1.2	E.6	DG(B) CONTROL PNL (A)	----?----	524
2237	R43-P002B*	2	19700	1.8	E.6	DG(B) SCT PANEL	----?----	524
2238	T31-F801A	1	18100	2.0	D.5	SO VALVE	107E6043/0	528
2239	T31-POS070B	N	18100	2.1	D.1	POSITION SWITCH	107E6043/0	528

Table 9A.6-2 Fire Hazard Analysis Equipment Database Sorted by Room — Reactor Building (Continued)

Item No.	MPL No	Elect Div.	Elev. Location	Location Number Coord.	Location Alpha Coord.	Description	System Drawing	Room No.
2240	X-102J	2	19000	2.2	D.3	CONTROL & INSTRUMENT	795E898	528
2241	X-104B	N	19000	2.2	D.5	FMCRD POSITION DISPLAY	103E1167	528
2242	X-104F	N	19000	2.1	D.2	FMCRD POSITION DISPLAY	103E1167	528
2243	D23-PT007A	1	19000	5.7	D.9	PRESS TRANSMITTER	107E5139/1	530
2244	D23-F001A	1	19000	5.7	D.9	SO VALVE	107E5139/1	530
2245	D23-F005A	1	19000	5.7	D.9	MO GLOBE VALVE	107E5139/1	530
2246	D23-F004A	1	19000	5.7	D.9	MO GLOBE VALVE	107E5139/1	530
2247	~~T49-F001B~~	2	20100	~~5.7~~	~~D.8~~	~~MO-GATE VALVE~~	~~107E6047/0~~	530
2248	~~T49-F002C~~	3	20100	~~5.7~~	~~D.8~~	~~AO GATE VALVE~~	~~107E6047/0~~	530
2249	~~T49-F002C-1~~	1	20100	~~5.7~~	~~D.8~~	~~SOLENOID VALVE~~	~~107E6047/0~~	530
2250	~~T49-F002C-2~~	4	20100	~~5.7~~	~~D.8~~	~~SOLENOID VALVE~~	~~107E6047/0~~	530
2251	X-142C	3	20100	5.7	D.8	REA WATER LEV & PRESS	795E877	530
2252	X-146C	3	19000	5.7	D.9	DRYWELL PRESSURE	107E6043/0	530
2253	X-162A	1	19000	5.7	D.8	CAMS GAMMA DETECTOR	10R281-431	530
2254	T31-TE052H	N	18100	5.3	E.5	TEMP ELEMENT	107E6043/0	531
2255	X-101G	N	19000	5.8	D.6	FMCRD POWER	103E1167	532
2256	X-104C	N	19000	5.8	D.3	FMCRD POSITION DISPLAY	103E1167	532
2257	X-104G	N	19000	5.8	D.5	FMCRD POSITION DISPLAY	103E1167	532
2258	X-110	3	20100	5.9	D.2	COMPENSATION/INSTR LINE	107E6043/0	532
2259	U41-C209G	3	19700	6.5	F.8	DG(C) EMER SUPP FAN G	107E5189/0	533
2260	U41-C209C	3	19700	6.5	F.3	DG(C) EMER SUPP FAN C	107E5189/0	533
2261	U41-TIS062	3	19700	6.8	F.5	TEMP IND SW EMER EXH	107E5189/0	534
2262	R43-P001C*	3	19700	6.3	E.6	DG(C) CONTROL PNL (A)	----?----	536
2263	R43-P002C*	3	19700	6.8	E.6	DG(C) SCT PANEL	----?----	536
2264	E31-TE020A	1	18100	4.0	A.5	MSL TUN AMB TEMP ELEM	103E1792/1	440
2265	E31-TE020B	2	18100	4.0	A.5	MSL TUN AMB TEMP ELEM	103E1792/1	440
2266	E31-TE020C	3	18100	4.0	A.5	MSL TUN AMB TEMP ELEM	103E1792/1	440

Figure 12.3-5 Reactor Building Radiation Zone Map for Full Power and Shutdown Operation at Elevation 12300 mm

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

Figure 12.3-10 Reactor Building Radiation Zone Map for Full Power and Shutdown Operation, Section A-A

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

Figure 12.3-16 Reactor Building Radiation Zone Map Post LOCA at Elevation 12300 mm

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

Figure 12.3-21 Reactor Building Radiation Zone Map Post LOCA, Section A-A

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

DC System

- Neutron Monitoring
  - Source Range
  - Intermediate Range
  - Power Range
  - TIP System
- Reactor Protection
- Rod Worth Minimizer
- ~~Hydrogen Recombiners~~

Procedures For Off-Normal Or Alarm Conditions.
Prepare all procedures for off-normal or alarm conditions that require operator action in the MCR and RSS. These correspond to the number of alarm annunciators. Each annunciator important to safety should have its own written procedure, which should normally contain (a) the meaning of the annunciator, (b) the source of the signal, (c) the immediate action that is to occur automatically, (d) the immediate operator action and (e) the long-range actions. If more than one annunciator applies to a given procedure, repetition of the procedure may not be required if the applicable annunciators are listed at the beginning of the procedure.

General Plant Operating Procedures.
As discussed in Section A5 of ANSI/ANS-3.2, procedures shall be prepared for the integrated operations of the plant. Typical general plant procedures are listed below:

- Cold Shutdown to Hot Standby
- Hot Standby to Minimum Load (nuclear startup)
- Recovery from Reactor Trip
- Operation at Hot Standby
- Turbine Startup and Synchronization of Generator
- Changing Load and Load Follow (if applicable)
- Power Operation and Process Monitoring
- Power Operation with Less than Full Reactor Coolant Flow

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operational for verifying the leak detection and isolation functions as indicated in the parenthesis:

(a) Neutron Monitoring System (ATIP isolation)

(b) Containment System (drywell coolant leakage)

(c) Standby Gas Treatment System (system initiation)

(d) Reactor Protection System (isolation bypass)

(e) Standby Liquid Control System (system initiation)

(f) Nuclear Boiler System (MSIV, MSL drain valves)

(g) RHR System (shutdown cooling suction isolation)

(h) CUW System (containment isolation valve)

(i) RCIC System (system isolation)

(j) SSLC (LDS logic processing)

(k) Other auxiliary systems (e.g., PRM, RD, RCW, HNCW, HVAC, ACS, ~~FCS,~~ SPCU, etc.) associated with the LDS functions

(3) General Test Methods and Acceptance Criteria

Since the LDS is comprised mostly of logic, the checks of valve response and timing and the testing of sensors will be performed as part of, or in conjunction with, the various systems with which they are associated.

Performance shall be observed and recorded during a series of individual component and integrated system tests. These tests shall demonstrate that the LDS operates properly as specified in Subsection 7.3.1.1.2 and applicable LDS design specification through the following testing:

(a) Correct implementation and operation of the LDS software-based controls and instrumentation. This test shall check the system behavior against the functional, performance and interface requirements as specified by the appropriate design documents and the Hardware/Software System Specification (HSSS).

(b) Verification of various indicators, annunciators, and alarms used to monitor system operation and status for correct functions.

(c) Proper operation of leakoff and drainage measurement functions such as those associated with the reactor vessel head flange, drywell cooler condensate, and various primary system valves.

14.2.12.1.18 Remote Shutdown System Preoperational Test

(1) Purpose

Verify the feasibility and operability of intended remote shutdown functions from the Remote Shutdown System (RSS) panel and other local and remote locations outside the main control room which will be utilized during the remote shutdown scenario.

(2) Prerequisites

The construction tests have been successfully completed, and the SCG has reviewed the test procedure and approved the initiation of testing. Communication shall be established between the RSS panel, main control room, and each system associated with the RSS. Additionally, the 480 VAC and 6.9 kVAC electrical power system shall be in operation and available and 125 VAC/125 VDC control power shall be supplied to the remote shutdown panel. The applicable portions of the RHR, HPCF, RCW, RSW, NBS, ACS, ~~FCS~~ and MUWC shall be available, as needed, to support the specified testing.

(3) General Test Methods and Acceptance Criteria

The Remote Shutdown System (RSS) consists of the control and instrumentation available at the dedicated remote shutdown panel(s) and other local and remote locations intended to be used during the remote shutdown scenario.

Much of the specified testing can be accomplished in conjunction with, or as part of, the individual system and component preoperational testing. However, the successful results of such testing shall be documented as part of this test, as applicable. Performance shall be observed and recorded during a series of individual component and integrated system tests. This test shall demonstrate that the RSS operates properly as specified in Subsection 7.4.1.4 and applicable RSS design specification through the following testing:

(a) Proper functioning of the system controls and instrumentation associated with the RSS after transfer of control to the RSS panel.

(b) Proper operation of remote shutdown system pumps and valves including establishment of system flow paths using RSS control.

(c) Proper functioning of RSS transfer switches including verification of proper override of main control room functions.

(b) The BOP scope of piping systems are as follows:

(i) Main steam piping downstream of the MSIV outside containment

(ii) Feedwater piping outside containment downstream of the isolation check valves

(iii) RPV head vent piping

(iv) CUW suction and discharge piping, including the head spray line

(v) RHR suction and discharge and injection piping in shutdown cooling mode and LPFL mode

(vi) RCIC turbine steam supply and exhaust piping

(vii) RCIC pump suction and discharge piping

(viii) SLC system piping (pump suction/discharge)

(ix) RSW suction and discharge piping

(x) RCW suction and discharge piping

(xi) HPCF suction and injection piping

(xii) Diesel generator fuel, cooling, intake and exhaust piping

~~(xiii) FCS hydrogen recombiner piping~~

(xiv) CRD system piping (pump suction/discharge)

Thermal expansion testing during the preoperational phase will consist of displacement measurements on the NSSS portion of piping during the RRS/RPV internal hot functional test (Subsection 14.2.12.1.2) and visual inspections at ambient temperature on the NSSS and BOP portions of piping. The testing will be in conformance with ANSI/ASME-OM7 as discussed in Subsection 3.9.2.1.2, and will consist of a combination of visual inspections and local and remote displacement measurements. This testing, as well as that performed during the power ascension phase per Subsection 14.2.12.2.10, includes the inspection and testing of RCPB component supports as described in Subsection 5.4.14.4. Visual inspections are performed to identify actual or potential constraints to free thermal growth prior to or between tests. Displacement measurements will be made utilizing specially installed instrumentations and also using the position of supports such as snubbers. Results of the thermal expansion testing are acceptable when all systems move as predicted and there are no observed restraints to free thermal growth or when additional analysis shows that any unexpected results will not produce unacceptable stress values.

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Vibration testing will be performed on system components and piping during preoperational function and flow testing. This testing will be in accordance with ANSI/ASME-OM3 as discussed in Subsection 3.9.2.1.1 and will include visual observation and local and remote monitoring in critical steady-state

(d) Proper operating conditions and performance capability during the following system operational tests:

(i) Placing a standby polisher unit into service

(ii) Transferring the resin inventory of any polisher vessel into the resin receiver tank

(iii) Removing operating filter from service, backwashing and restoring to service

(iv) Transferring the resin storage tank resins to any polisher vessel

(v) Transferring resin from resin receiver tank to the radwaste system

(vi) Operating the system at full condensate flow with four filters and five polisher vessels

(e) Proper operation of interlocks and equipment protective devices

(f) Proper operation of permissive, prohibit, and bypass functions

(g) Ability to perform online exchange of standby and spent filter units and polisher vessels

(h) Proper operation of filter and polisher support facilities such as those used for regeneration of resins or for handling of wastes

(i) Proper operation of the system flow bypass feature through manually operating the system flow bypass valve from the main control room

14.2.12.1.55 Reactor Water Chemistry Control Systems Preoperational Test

(1) Purpose

To verify proper operation of the various chemical addition systems designed for actively controlling the reactor water chemistry, including the oxygen injection system, the zinc injection passivation system, the iron ion injection system, and the Hydrogen Water Chemistry System (HWCS).

(2) Prerequisites

The construction tests have been successfully completed, and the SCG has reviewed the test procedure(s) and approved the initiation of testing. The ~~FCS,~~ Offgas System, appropriate electrical power, and other required interfacing systems shall be available, as needed, to support the specified testing. The appropriate vendor precautions shall be followed with regards to the operation of the affected systems and components and for the actual reactor water chemistry given the existing reactor operating state.

mode of operation. Generator hydrogen purity and leakage rate shall meet the appropriate design requirements.

(e) Proper operation of the stator cooling system to provide adequate stator cooling water flow at prescribed flow rate and maintain inlet temperature and conductivity control.

(f) Correct function of the generator runback circuits in response to simulated high stator cooling water outlet temperature and low stator cooling water pressure signals.

(g) Proper operation while powered from primary and any alternate sources, including transfers, and in degraded modes for which the system, subsystem or component is expected to remain operational.

(h) Acceptable generator clearance and vibration levels, during both transient and steady-state operation. This test can be performed during the startup test stage in conjunction with turbine testing.

(i) Acceptable differential pressure between air side and hydrogen side of generator seal oil system.

~~14.2.12.1.72 Flammability Control System Preoperational Test~~

~~(1) Purpose~~

~~To verify the ability of the Flammability Control System (FCS) to recombine hydrogen and oxygen and therefore maintain the specified inert atmosphere in the primary containment during long term post accident conditions.~~

~~(2) Prerequisites~~

~~The construction tests, including the pressure proof test, have been successfully completed, and the SCC has reviewed the test procedure and approved the initiation of testing. All system instrumentation shall be in accordance with the FCS instrument data sheets and calibrated per instrument supplier's instructions. All services, including water, electricity and communications, shall be available and performing at their rated design levels (flow, voltage, pressure, etc.). The wetwell and drywell airspace regions of the primary containment shall be intact, and all other required interfaces shall be available, as needed, to support the specified testing.~~

~~(3) General Test Methods and Acceptance Criteria~~

~~Performance shall be observed and recorded during a series of individual component and integrated system tests. This test shall demonstrate that the~~

~~FCS operates properly as specified in Subsection 6.2.5 and applicable FCS design specifications through the following testing:~~

~~(a) Proper operation of instrumentation and system controls in all combinations of logic~~

~~(b) Verification of various component alarms including alarm actuation and reset, alarm set value, alarm indication and operating logic~~

~~(c) Proper operation of all motor operated and air operated valves, including stroking using valve opening/closing switches at the control room, verification of indicator lamp, timing and isolation function, if applicable~~

~~(d) Proper system operating conditions (i.e., the system shall be operated normally without any abnormalities, vibration, or leakage in components, valves, and piping within the FCS) for the following test cases while the FCS is in accident operating mode and regular testing mode of operation as defined in the design specification:~~

~~(i) Triple heater test for inside heater box temperature during steady-state operation~~

~~(ii) Blower running test for blower flow rate, flow control valve position and each line's gas flow rate~~

~~(iii) Reaction chamber heatup test for blower flow rate, flow control valve position, each line's gas flow rate and the time for heating up the reactor chamber~~

~~(e) Proper operation of interlocks including operation of all components subject to interlocking, interlocking set value and operating logic~~

~~(f) Proper operation of permissive, prohibit, and bypass functions~~

~~(g) Proper system operation while powered from primary and alternate sources, including transfers, and in degraded modes for which the system is expected to remain operational~~

14.2.12.1.73 Loose Parts Monitoring System Preoperational Test

(1) Purpose

To verify proper functioning of Loose Parts Monitoring System (LPMS) equipment.

(2) Prerequisites

The construction tests have been successfully completed, and the SCG has reviewed the test procedure and approved the initiation of testing. Reactor

Table 14.3-9 Generic Safety Issues (Continued)

Tier 2 Entry	Parameter	Tier 2 Value
19B.2-10	A-31: Residual Heat Removal (RHR) Shutdown Requirements	
	RHR System Composed of 3 Electrically And Mechanically Independent Divisions	----
	Shutdown Cooling Can Be Manually Initiated from the Control Room	----
	RHR System Can Be powered from Both Offsite and Standby Emergency Electrical Power	----
19B.2-11	A-35: Adequacy of Offsite Power Systems	
	Equipment Qualified for Operation with Voltage up to 10% Less than Normal	----
19B.2.12	A-36: Control of Heavy Loads Near Spent Fuel	
	Equipment Handling Components Meet Single Failure Criteria	----
	Redundant Safety Interlocks and Limit Switches Prevent Heavy Loads Over Spent Fuel	----
19B.2.13	A-39: Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits	
	Each S/RV Discharge Pipe Fitted with an X-Quencher	----
19B.2-16	A-44: Station Blackout	
	Sources of Electrical Power	
	No. of Standby Turbine Generators	1
	No. of Emergency Diesel Generators	3
19B.2-17	A-47: Safety Implications of Control Systems	
	Feedwater Controller	
	Trip Feedpumps on High Water Level	----
	Fault Tolerant Through Redundant Micro-processors and Self Diagnostics	----
19B.2-18	A-48: Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	
	Containment Inerted During Normal Operation	----
	~~Permanently Installed Hydrogen Recombiners~~	----
19B.2-20	B-17: Criteria for Safety-Related Operator Actions	
	RHR Heat Exchanger in LPCI Injection Loop	----
19B.2-22	B-55: Improved Reliability of Target Rock Safety/Relief Valves	
	ABWR Uses a Direct Acting S/RV Design	----

Table 14.3-10 TMI Issues

Tier 2 Entry	Parameter	Tier 2 Value
19A.2.17	I.D.3 Safety System Status Monitoring	
	Automatic Indication of Bypassed and Inoperable Status of Safety Systems	----
19B.2.65	I.D.5(2) Plant Status and Post-Accident Monitoring	
	Post-Accident Information Available to the Operator is in Compliance with RG 1.97	----
19B.2.66	I.D.5(3) On-Line Reactor Surveillance System	
	ABWR Design Incorporates a Reactor Vessel Loose Parts Monitoring System	----
1A.2.5	II.B.1 Reactor Coolant System Vents	
	Steam-Driven RCIC	1
	Power-Operated Relief Valves	
	Number	18
	Dual Position Indication	
	Position Sensors	----
	SRV Discharge Temperature Elements	----
	Remotely Operable from the Control Room	----
1A.2.6	II.B.2 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	
	Vital Areas as per NUREG-0737 Accessible Post-LOCA	
	Continuous Occupancy	----
	Non-Continuous Occupancy	----
1A.2.7	II.B.3 Post-Accident Sampling	
	Able to Obtain Samples Under Accident Conditions	----
19A.2.21	II.B.8 Rulemaking Proceeding on Degraded Core Accidents	
	Inerted Primary Containment	----
	~~Permanently-Installed-Recombiners~~	----
1A.2.9	II.D.1 Testing Requirements	
	SRVs Qualified for Steam Discharge	----
	Redundant Logic to Respond to High Water Level Conditions	----
	RHR Shutdown Cooling Systems	
	Number	3
	Separate Vessel Penetration and Suction Lines	----

Table 14.3-10 TMI Issues (Continued)

Tier 2 Entry	Parameter	Tier 2 Value
1A.2.10	II.D.3 Relief and Safety Valve Position Indication	
	Dual Position Indication	
	Position Sensors	----
	SRV Discharge Temperature Elements	----
1A.2.13	II.E.4.1 Dedicated Penetrations	
	~~Recombiners in Secondary Containment~~	
	Number	2
	Permanently Installed	----
1A.2.14	II.E.4.2 Isolation Dependability	
	Diverse Containment Isolation Signals	----
	Non-Essential Systems	
	Automatically Isolated On Containment Isolation Signal	----
	Redundant Isolation Valves	----
	Resetting Isolation Signal Does Not Automatically Reopen Isolation Valves	----
	Containment Purge and Vent Valves	
	Close on Isolation Signals	----
	Fail Closed	----
	Close on High Radiation	----
19A.2.27	II.E.4.4 Purging	
	Drywell Has Primary Containment Supply/Exhaust System	----
1A.2.15	II.F.1 Additional Accident Monitoring Instrumentation	
	Plant Post Accident Monitoring Variables	
	Neutron Flux	----
	Control Rod Position	----
	Boron Concentration	----
	Reactor Coolant System Pressure	----
	Drywell Pressure	----
	Drywell Sump Level	----
	Coolant Level in Reactor	----
	Suppression Pool Water Level	----
	Containment Area Radiation	----

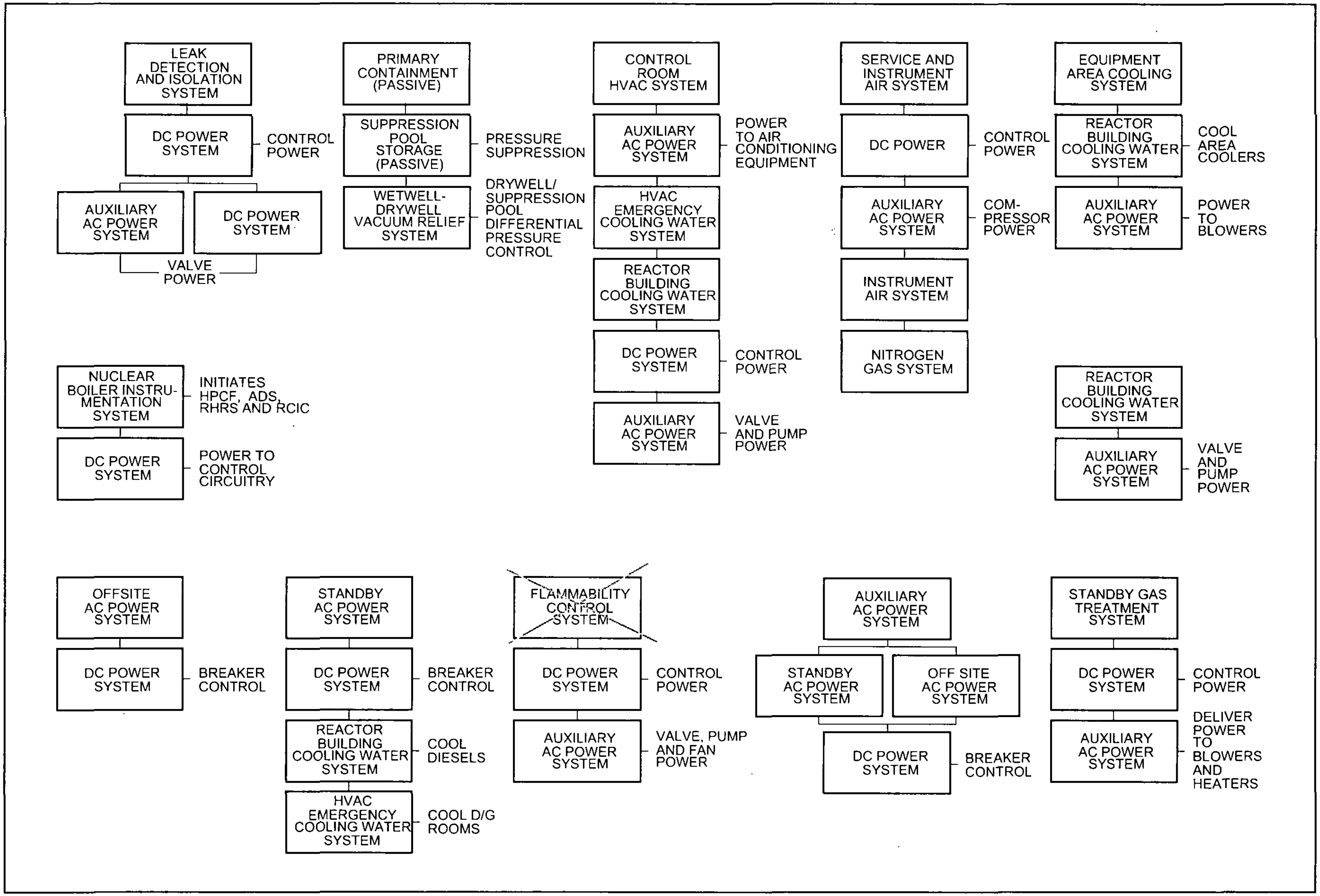

Figure 15A-7 Safety System Auxiliaries — Group 2

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(continued)

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(continued)

3.3 INSTRUMENTATION

3.3.6.1 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.6.1 The PAM instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

NOTE

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each Function.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Provide alternate method of monitoring, determine the cause of the inoperability, and submit plans and schedule for restoring the instrumentation channels of the Functions to OPERABLE status to the NRC.	14 days
C. One or more Functions with two required channels inoperable.	~~NOTE~~ ~~This Action is not applicable to Functions 11 and 12.~~	
	C.1 Restore at least one inoperable channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D. Two required hydrogen/oxygen monitor channels inoperable.~~	~~D.1 Restore one required hydrogen/oxygen monitor channel to OPERABLE status.~~	~~72 hours~~
E. Required Action and associated Completion Time of Condition C ~~or D~~ not met.	E.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
F. As required by Required Action E.1 and referenced in Table 3.3.6.1-1.	F.1 Be in MODE 3.	12 hours
G. As required by Required Action E.1 and referenced in Table 3.3.6.1-1.	G.1 Provide alternate method of monitoring, determine the cause of the inoperability, and submit plans and schedule for restoring the instrumentation channels of the Functions to OPERABLE status to the NRC.	14 days

Table 3.3.6.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION I.1
1.	Reactor Steam Dome Pressure.	2	F
2.	Reactor Vessel Water Level-Wide Range.	2	F
3.	Reactor Vessel Water Level-Fuel Zone.	2	F
4.	Suppression Pool Water Level.	2	F
5.	Containment Pressure.		
	5a. Drywell Pressure.	2	F
	5b. Wide Range Containment Pressure.	2	F
6.	Drywell Area Radiation.	2	G
7.	Wetwell Area Radiation.	2	G
8.	PCIV Position.	2 per penetration flow path[(a),(b)]	F
9.	Startup Range Neutron Monitor-Neutron Flux.	2[(c)]	F
10.	Average Power Range Monitor-Neutron Flux.	2[(d)]	F
~~11.~~	~~Containment Atmospheric Monitors-Drywell H_2 & O_2 Analyzer.~~	~~2~~	~~F~~
~~12.~~	~~Containment Atmospheric Monitors-Wetwell H_2 & O_2 Analyzer.~~	~~2~~	~~F~~
13.	Containment Water Level.	2	F
14.	Suppression Pool Water Temperature.	2[(e)]	F
15.	Drywell Atmosphere Temperature.	2	F
16.	Main Steam Line Radiation.	2	F

(a) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(b) Not required for isolation valves whose associated penetration flow path is isolated by at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(c) When power is ≤ [10]% RTP

(d) When power is > [10]% RTP

(e) Bulk average temperature.

Table 3.3.6.2-1 (page 1 of 2)
Remote Shutdown System Instrumentation

FUNCTION (INSTRUMENT OR CONTROL PARAMETER)	REQUIRED NUMBER OF DIVISIONS
1. Reactor Steam Dome Pressure.	2
2. HPCF B Flow.	1
3. HPCF B Controls.	1(c)
4. HPCF B Pump Discharge Pressure.	1
5. RHR Flow.	2(a)
6. RHR Hx Inlet Temperature.	2(a)
7. RHR Hx Outlet Temperature.	2(a)
8. RHR Hx Bypass Valve Position.	2(a)
9. RHR Hx Outlet Valve Position.	2(a)
10. RHR Pump Discharge Pressure.	2(a)
11. RHR Controls.	2(a)(c)
12. RPV Wide Range Water Level.	2
13. RPV Narrow Range Water Level.	2
14. Reactor Building Cooling Water Flow.	2
15. Reactor Building Cooling Water Controls.	2(c)
16. Reactor Building Service Water System Controls.	2(c)
~~17. Cooling Water Flow to Flammability Control System~~	~~1~~
18. Suppression Pool Water Level.	2
19. Condensate Storage Tank Water Level.	1

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.3.1 Primary Containment Hydrogen Recombiners

LCO 3.6.3.1 Two primary containment hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One primary containment hydrogen recombiner inoperable.	A.1	--------NOTE-------- LCO 3.0.4 is not applicable. -------------------- Restore primary containment hydrogen recombiner to OPERABLE status.	 30 days
B. Two primary containment hydrogen recombiners inoperable.	B.1	Verify by administrative means that the hydrogen control function is maintained.	1 hour AND Once per 12 hours thereafter
	AND		
	B.2	Restore one primary containment hydrogen recombiner to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.3.1.1	Perform a system functional test for each primary containment hydrogen recombiner.	6 months
SR 3.6.3.1.2	Perform a system functional test for each primary containment hydrogen recombiner.	18 months
SR 3.6.3.1.3	Visually examine each primary containment hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	18 months
SR 3.6.3.1.4	Perform a resistance to ground test for each heater phase.	18 months

5.5 Procedures, Programs, and Manuals

5.5.2.1 Offsite Dose Calculation Manual (ODCM) (continued)

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. a determination that the change(s) maintain the levels of radioactive effluent control required pursuant to 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after review and acceptance by plant reviews and the approval of the [Plant Superintendent]; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of, or concurrent with, the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include the Low Pressure Core Flooder, High Pressure Core Flooder, Residual Heat Removal, Reactor Core Isolation Cooling, ~~Hydrogen Recombiner,~~ Post Accident Sampling, Standby Gas Treatment, Suppression Pool Cleanup, Reactor Water Cleanup, Fuel Pool Cooling and Cleanup, Process Sampling, Containment Atmospheric Monitoring, and Fission Product Monitor. The program shall include the following:

(continued)

BASES

LCO
(Continued)

8. Primary Containment Isolation Valve (PCIV) Position (continued)

position indication for valves in an isolated penetration is not required to be OPERABLE.

Indication of the completion of the containment isolation function is provided by valve closed/not closed indications for individual valves on safety related displays. Annunciators are provided to alert the operator to any lines that may not be isolated.

[For this plant, the PCIV position PAM instrumentation consists of the following:]

9. and 10. Wide Range Neutron Flux

Wide range neutron flux is a Category I variable provided to verify reactor shutdown. The display controller uses data from four APRM channels and four SRNM channels to provide a display of neutron flux on the main control room panel with a range of 10^{-6}% to 125% RTP. These displays are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.

~~11. and 12. Containment Atmospheric Monitors-Drywell and Wetwell Hydrogen and Oxygen Analyzer~~

~~Drywell and wetwell hydrogen and oxygen analyzers are Category I instruments provided to detect high hydrogen or oxygen concentration conditions that represent a potential for containment breach. These parameters are also important in verifying the adequacy of mitigating actions. There are two divisions in the Containment Atmospheric Monitoring System analyzers with one channel of~~ $\cancel{H_2}$ ~~monitoring and one channel of~~ $\cancel{O_2}$ ~~monitoring per division. Samples of either the drywell or wetwell are drawn into the analyzers based on the position of a selector switch in the main control room. Displays and alarms are provided in the main control room. These displays are the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.~~

(continued)

ACTIONS (Continued)

C.1

~~As noted in the LCO this action does not apply to Functions 11 & 12, (hydrogen/oxygen monitors), which are addressed in Condition D.~~ When a Function has two required channels that are INOPERABLE then one channel must be restored to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

Multiple entry into the condition table causes Condition A to be invoked on completion of Action C.1 so appropriate additional action is taken.

~~D.1~~

~~When two hydrogen/oxygen monitor display channels are inoperable, at least one channel must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable, based on the backup capability of the Post Accident Sampling System to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA that would cause core damage would occur during this time.~~

E.1

are

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition referenced in the Table is Function dependent. If the required Actions and associated Completion Times for Conditions C~~, or D are~~ not met for a Function then Condition E is entered for that function and Table 3.3.6.1-1 used to transfer to the appropriate subsequent Condition.

(continued)

BASES

ACTIONS (Continued)

F.1

For the PAM Functions in Table 3.3.6.1-1, if any Required Action and associated Completion Time of Condition C or D is not met, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant condition from full power conditions in an orderly manner and without challenging plant systems.

G.1

Since alternate means of monitoring the parameters to which this Condition applies have been developed and tested, the Required Action is to submit a report to the NRC instead of requiring a plant shut down. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

SURVEILLANCE REQUIREMENTS

The following SRs apply to each PAM instrumentation Function in Table 3.3.6.1-1, except SR 3.3.6.1.1 does not apply to Function 8.

SR 3.3.6.1.1

Performance of a CHANNEL CHECK once every [31] days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one instrumentation channel to a similar parameter on other instrumentation channels. It is based on the assumption that independent displays of the same parameter should read approximately the same value. Significant deviations between displays could be an indication of excessive instrument drift or other faults in one of the channels. A

(continued)

BASES

SURVEILLANCE REQUIREMENTS (Continued)

SR 3.3.6.1.2 (continued)

address some of the same components required by the PAM displays.

REFERENCES

1. Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," ~~[Date].~~ June 2006.
2. DCD Tier 2, Section 7.5

Date shown is date current revision was issued.

This page was replaced by page C-209

B 3.3 INSTRUMENTATION

B 3.3.6.2 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the plant in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility of the control room becoming inaccessible. A safe shutdown condition is defined as MODE 3. With the plant in MODE 3, the High Pressure Core Flooder System, the safety/relief valves, and the Residual Heat Removal System Shutdown Cooling and Suppression Pool Cooling Modes can be used to remove core decay heat and meet all safety requirements. Additional systems assisting in the remote shutdown capability are portions of the Nuclear Boiler System, the Reactor Building Cooling Water System, the Reactor Building Service Water System, the Electrical Power Distribution System, ~~and the Flammability Control System.~~ The long term supply of water for the HPCF and the ability to operate shutdown cooling from outside the control room allow extended operation in MODE 3. and

In the event that the control room becomes inaccessible, the operators can establish control at either of two remote shutdown panels (Division I and Division II) and place and maintain the plant in MODE 3. The two panels have a different complement of controls and indications, but either panel may be used to achieve and maintain MODE 3. The main difference between the two panels is that one of them uses HPCF and one SRV to regulate pressure and provide the decay heat removal and inventory make up. The other panel uses 3 SRVs, the LPFL, and the shutdown cooling mode of an RHR system to provide this capability.

The postulated conditions assumed to exist when the Main Control Room becomes inaccessible are 1) the plant is operating initially at or less than design power and 2) the plant is not experiencing any transient or accident situations. Therefore, complete control of engineered safeguard feature systems from outside the main control room is not required.

Even though the loss of offsite power is considered unlikely, the remote shutdown panels are powered from Class 1E power system buses I and II so that backup AC power would

(continued)

BASES

LCO
(Continued)

5 through 11. RHR A, B Control & Indication. (continued)

division II panel) are required to be OPERABLE to provide redundant capability to achieve MODE 3 from both RSS panels.

12. and 13. RPV Wide Range/Narrow Range Water Level.

Reactor vessel water level is provided to support monitoring of core cooling, to verify operation of the make up pumps, and is needed for satisfactory operator control of the make up pumps. The wide range water level channels cover the range from the near top of the fuel to near the top of the steam separators. The narrow range provides indication from near the bottom of the separators to above the steam lines. RPV level is a necessary parameter for achieving and maintaining the reactor in MODE 3. One channel of each range is provided on each of the RSS panels. Both channels are required to be OPERABLE to provide redundant capability to achieve MODE 3 from both RSS panels.

14, 15, and 16. Reactor Building Cooling Water Flow/Controls & Reactor Building Service Water Controls.

These parameters and controls are required to monitor and control the water supply for cooling the equipment needed to achieve MODE 3 and to provide containment heat removal. The Reactor Building Cooling Water controls provided are as given in reference 4 and the Reactor Building Service Water controls provided are as given in reference 5. One channel of each Function is provided on each of the RSS panels. Both channels of each Function are required to be OPERABLE to provide redundant capability to achieve MODE 3 from both RSS panels.

~~17. Cooling Water Flow to Flammability Control System.~~

~~A control for the FCS B inlet valve is provided on the division II panel only. This control is needed in order for the operator to isolate cooling water flow to FCS. One channel is required to be OPERABLE to assure that MODE 3 can be achieved from the Division II RSS panel.~~

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Primary Containment Hydrogen Recombiners

BASES

BACKGROUND

The primary containment hydrogen recombiner eliminates the potential breach of primary containment due to a hydrogen-oxygen reaction and is part of combustible gas control required by 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2). The primary containment hydrogen recombiner is required to reduce the hydrogen concentration in the primary containment following a loss of coolant accident (LOCA). The primary containment hydrogen recombiner accomplishes this by recombining hydrogen and oxygen to form water vapor. The water vapor is returned to the primary containment, thus eliminating any discharge to the environment. The primary containment hydrogen recombiner is manually initiated, since flammability limits would not be reached until several hours after a Design Basis Accident (DBA).

The primary containment hydrogen recombiner functions to maintain the hydrogen gas concentration within the containment at or below the flammability limit of 4.0 volume percent (v/o) following a postulated LOCA. It is fully redundant and consists of two 100% capacity subsystems. Each primary containment hydrogen recombiner consists of an enclosed blower assembly, heater section, reaction chamber, direct contact water spray gas cooler, water separator, and associated piping, valves, and instruments.

The primary containment hydrogen recombiner will be manually initiated from the main control room when the hydrogen gas concentration in the primary containment reaches approximately 1 v/o. When the primary containment is inerted (oxygen concentration < 3.5 v/o), the primary containment hydrogen recombiner will only function until the oxygen is used up (2.0 v/o hydrogen combines with 1.0 v/o oxygen). Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Feature bus and is provided with separate power panel and control panel.

(continued)

BASES

BACKGROUND (continued)

The process gas circulating through the heater, the reaction chamber, and the cooler is automatically regulated to 255 m^3/h by the use of an orifice plate in-stalled in the cooler. The process gas is heated to 718°C. The hydrogen and oxygen gases are recombined into water vapor, which is then condensed in the water spray gas cooler by the associated residual heat removal subsystem and discharged with some of the effluent process gas to the wetwell. The majority of the cooled, effluent process gas is mixed with the incoming process gas to dilute the incoming gas prior to the mixture entering the heater section.

APPLICABLE SAFETY ANALYSES

The primary containment hydrogen recombiner provides the capability of controlling the bulk hydrogen concentration in primary containment to less than the lower flammable concentration of 4.0 v/o following a DBA. This control would prevent a primary containment wide hydrogen burn, thus ensuring that pressure and temperature conditions assumed in the analysis are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate in primary containment following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant; or

b. Radiolytic decomposition of water in the Reactor Coolant System.

To evaluate the potential for hydrogen accumulation in primary containment following a LOCA, the hydrogen generation is calculated as a function of time following the initiation of the accident. Assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

The calculation confirms that when the mitigating systems are actuated in accordance with emergency procedures, the peak hydrogen concentration in the primary containment is 4.0 v/o (Ref. 4).

The primary containment hydrogen recombiners satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

Two primary containment hydrogen recombiners must be OPERABLE. This ensures operation of at least one primary containment hydrogen recombiner subsystem in the event of a worst case single active failure.

Operation with at least one primary containment hydrogen recombiner subsystem ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, the two primary containment hydrogen recombiners are required to control the hydrogen concentration within primary containment below its flammability limit of 4.0 v/o following a LOCA, assuming a worst case single failure.

In MODE 3, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in this MODE, the probability of an accident requiring the primary containment hydrogen recombiner is low. Therefore, the primary containment hydrogen recombiner is not required in MODE 3.

In MODES 4 and 5, the probability and consequences of a LOCA are low due to the pressure and temperature limitations in these MODES. Therefore, the primary containment hydrogen recombiner is not required in these MODES.

ACTIONS

A.1

With one primary containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit, the amount of time available after the event for operator action

(continued)

BASES

ACTIONS (continued)

A.1 (continued)

to prevent exceeding this limit, and the low probability of failure of the OPERABLE primary containment hydrogen recombiner.

Required Action A.1 has been modified by a Note indicating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is provided because of the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit, the low probability of the failure of the OPERABLE subsystem, and the amount of time available after a postulated LOCA for operator action to prevent exceeding the flammability limit.

B.1 and B.2

With two primary containment hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by the atmospheric control system (ACS). The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. Both the initial verification and all subsequent verifications may be performed as an administrative check by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

(continued)

BASES

ACTIONS (continued)

C.1

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.1 and SR 3.6.3.1.2

Performance of a system functional test for each primary containment hydrogen recombiner ensures that the recombiners are OPERABLE and can attain and sustain the temperature necessary for hydrogen recombination. In particular, SR 3.6.3.1.1 verifies, every 6 months, that the minimum heater sheath temperature increases to ≥ [316°C] in ≤ [1.5 hours] and that it is maintained > [316°C] for ≥ [2] hours thereafter to check the ability of the recombiner to function properly (and to make sure that significant heater elements are not burned out). Additionally, SR 3.6.3.1.2 verifies, every 18 months, that the reaction chamber temperature increases to ≥ [621°C] in [2] hours and that it is maintained > [636°C] and < [662°C] for ≥ [2] hours.

Operating experience has shown that these components usually pass the Surveillance when performed at the 6 and 18 month Frequencies, respectively. Therefore, these Frequencies were concluded to be acceptable from a reliability standpoint.

SR 3.6.3.1.3

This SR ensures there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, except for the blower assemblies, they are subject to only minimal mechanical failure. The only credible failures involve loss of power or blower function, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.1.3 (continued)

abnormal conditions that could cause such failures. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.1.4

This SR requires performance of a resistance to ground test of each heater phase to make sure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ [10,000] ohms.

Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. Regulatory Guide 1.7, Revision 1.
4. DCD Tier 2, Section 6.2.5.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.2 Primary Containment Oxygen Concentration

BASES

BACKGROUND

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal water reaction in the core or due to radiolysis.

The primary method to control combustible gases is to inert the primary containment. With the primary containment inert, that is, oxygen concentration < 3.5 volume percent (v/o), a combustible mixture cannot be present in the primary containment for any hydrogen concentration. ~~The capability to inert the primary containment and maintain oxygen < 3.5 v/o works together with the hydrogen recombiners (LCO 3.6.3.1, "Primary Containment Hydrogen Recombiners") to provide redundant and diverse methods to mitigate events that produce hydrogen. For example, an event that rapidly generates hydrogen from zirconium metal water reaction will result in excessive hydrogen in primary containment, but oxygen concentration will remain < 3.5 v/o and no combustion can occur. Long term generation of both hydrogen and oxygen from radiolytic decomposition of water may eventually result in a combustible mixture in primary containment, except that the hydrogen recombiners remove hydrogen and oxygen gases faster than they can be produced from radiolysis and again no combustion can occur.~~ This LCO ensures that oxygen concentration does not exceed 3.5 v/o during operation in the applicable conditions.

APPLICABLE SAFETY ANALYSES

The Reference 1 calculations assume that the primary containment is inerted when a Design Basis Accident loss of coolant accident occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of metal water reaction in the reactor core will not produce combustible gas mixtures in the primary containment. ~~Oxygen, which is subsequently generated by radiolytic decomposition of water, is recombined by the hydrogen recombiners (LCO 3.6.3.1) more rapidly than it is produced.~~

Primary containment oxygen concentration satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES

LCO	The primary containment oxygen concentration is maintained < 3.5 v/o to ensure that an event that produces any amount of hydrogen does not result in a combustible mixture inside primary containment.
APPLICABILITY	The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is ≤ 15% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.
ACTIONS	<u>A.1</u> If oxygen concentration is ≥ 3.5 v/o at any time while operating in MODE 1, with the exception of the relaxations allowed during startup and shutdown, oxygen concentration must be restored to < 3.5 v/o within 24 hours. The 24 hour Completion Time is allowed when oxygen concentration is ≥ 3.5 v/o because of the ~~availability of other hydrogen mitigating systems (e.g., hydrogen recombiners) and the~~ low probability of an event that would generate significant amounts of hydrogen occurring during this period.

(continued)

TABLE I
ABWR EPG ABBREVIATIONS

ADS	—	Automatic Depressurization System
APRM	—	Average Power Range Monitor
ARI	—	Alternate Rod Insertion
CAMS	—	Containment Air Monitoring System
CRD	—	Control Rod Drive
CUW	—	Reactor Water Cleanup
ECCS	—	Emergency Core Cooling System
FAS	—	Firewater Addition System
F~~C~~S	—	Flammability ~~Gas Control~~ System
F/D	—	Filter/Demineralizer
FMCRD	—	Fine Motion Control Rod Drive
FPC	—	Fuel Pool Cooling
HCU	—	Hydrualic Control Unit
HPCF	—	High Pressure Core Flooder
HVAC	—	Heating, Ventilating and Air Conditioning
LCO	—	Limiting Condition for Operation
LPCF	—	Low Pressure Core Flooder mode of RHR System
MSIV	—	Main Steamline Isolation Valves
NPSH	—	Net Positive Suction Head
RBHVAC	—	Reactor Building HVAC
RCIC	—	Reactor Core Isolation System
RHR	—	Residual Heat Removal
RIP	—	Reactor Internal Pump
RPS	—	Reactor Protection System
RPV	—	Reactor Pressure Vessel
RSCS	—	Rod Sequence Control System

18A.5 PRIMARY CONTAINMENT CONTROL GUIDELINE

PC/H Monitor and control hydrogen and oxygen concentrations

If while executing the following steps:

- *Drywell or suppression chamber hydrogen concentration cannot be determined to be below 6% and drywell or suppression chamber oxygen RC-1] and execute it concurrently with this procedure; ~~secure and prevent operation of the FCS and,~~ initiate containment sprays in accordance with [Step PC/H-4] until drywell and suppression chamber hydrogen concentrations can be determined to be below 6% or drywell and suppression chamber oxygen concentrations can be etermined to be below 5%.*

PC/H-2 Monitor and control hydrogen and oxygen concentrations in the drywell.

(PC/H-2.1 Deleted, not applicable to ABWR.)

~~*When drywell hydrogen concentration reaches [0.5% (minimum hydrogen concentration for recombiner operation or minimum detectable hydrogen concentration, whichever is higher)] but only if drywell hydrogen concentration is below [6% (maximum hydrogen concentration for recombiner operation or 6%, whichever is lower)] or drywell oxygen concentration is below [5% (maximum oxygen concentration for recombiner operation or 5%, whichever is lower)], and only if suppression pool level is below [11.70 m (elevation of bottom of suppression pool to lower drywell vent)], place FCS in service and enter [procedure developed from the RPV Control Guideline] at [Step RC-1] and execute it concurrently with this procedure.*~~

(PC/H-2.2 Deleted, not applicable to ABWR.)

~~*When drywell hydrogen concentration reaches [6% (maximum hydrogen concentration for recombiner operation or 6%, whichever is lower)] and drywell oxygen concentration reaches [5% (maximum oxygen concentration for recombiner operation or 5%, whichever is lower)], secure FCS operation.*~~

PC/H-2.3 Continue in this procedure at [Step PC/H-4].

PC/H-3 Monitor and control hydrogen and oxygen concentrations in the

suppression chamber.

(PC/H-3.1 Deleted, not applicable to ABWR.)

~~*When suppression chamber hydrogen concentration reaches [0.5% (minimum hydrogen concentration for recombiner operation or minimum detectable hydrogen concentration, whichever is higher)], but only if suppression pool level is below [11.70 m (elevation of bottom of suppression pool to lower drywell vent)], and only if drywell hydrogen concentration is below [6% (maximum hydrogen concentration for recombiner operation or 6%, whichever is lower)] or drywell oxygen concentration is below [5% (maximum oxygen concentration for recombiner operation or 5%, whichever is lower)], place FCS in service and enter [procedure developed from the RPV Control Guideline] at [Step RC-1] and execute it concurrently with this procedure.*~~

(PC/H-3.2 Deleted, not applicable to ABWR.)

PC/H-4 When drywell or suppression chamber hydrogen concentration reaches 6% and drywell or suppression chamber oxygen concentration is above 5%, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.

Table 1
Operating Values of Secondary Containment Parameters
(Cont.)

Secondary Containment Parameter	Maximum Normal Operating Value	Maximum Safe Operating Value
Area Temperature	℃	℃
El. -8200 T.M.S.L. (Floor 100-B3F)		
• HCU Area - 0°	40	100
• RHR(A) Pump & Hx Room	65	100
• RCIC Room	65	100
• HPCF(C) Pump Room	65	100
• RHR(C) Pump & Hx Room	65	100
• CRD Pump Room	40	100
• HCU Area - 180°	40	100
• RHR(B) Pump & Hx Room	65	100
• HPCF(B) Pump Room	65	100
• CUW Non-Regen. Hx Room	50	100
• CUW Pump Area	40	100
El. -1700 T.M.S.L. (Floor 200-B2F)		
• TIP Area	40	100
• ECCS Pump Maintenance Area	40	100
• Valve Room A	40	100
• Valve Room C	40	100
• FMCRD & RIP Maintenance Area	40	100
• Valve Room B	40	100
• CUW Regen. Hx & Valve Area	50	100
• CUW F/D Valve Area	40	100
El. 4800 T.M.S.L. (Floor 300-B1F)		
• RPV Instrument Area 1 & 3	40	100
• RPV Instrument Area 2 & 4	40	100
• CUW/FPC F/D Area	40	100
El. 12300 T.M.S.L. (Floor 400-1F)		
• Valve Room A	40	100
• Valve Room C	40	100
~~• FCS A~~rea	65	~~100~~
• Valve Room B	40	100
• CUW Valve Area	40	100
El. 18100 T.M.S.L. (Floor 500-2F)		
• Main Steam/FW Tunnel	55	171
• FPC Area		
- Hx Area	40	100
- Pump Area	65	100
- Valve Area	40	100

Table 1
Operating Values of Secondary Containment Parameters
(Cont.)

Secondary Containment Parameter	Maximum Normal Operating Value	Maximum Safe Operating Value
HVAC Cooler Differential Temperature	℃	
RHR(A) Pump Room	Hi Alarm	N/A
RHR(B) Pump Room	Hi Alarm	N/A
RHR(C) Pump Room	Hi Alarm	N/A
HPCF(B) Pump Room	Hi Alarm	N/A
HPCF(C) Pump Room	Hi Alarm	N/A
RCIC Pump/Turbine Room	Hi Alarm	N/A
FPC(A)	Hi Alarm	N/A
FPC(B)	Hi Alarm	N/A
SGTS(B)	Hi Alarm	N/A
SGTS(C)	Hi Alarm	N/A
CAMS(A)	Hi Alarm	N/A
CAMS(B)	Hi Alarm	N/A
Main Steam/FW Tunnel	Hi Alarm	N/A
~~FCS(B)~~	~~Hi Alarm~~	~~N/A~~
~~FCS(C)~~	~~Hi Alarm~~	~~N/A~~
RIP Handling Machine Control Room A	Hi Alarm	N/A
RIP Handling Machine Control Room B	Hi Alarm	N/A
CRD Auto Exchanger Control Room A	Hi Alarm	N/A
CRD Auto Exchanger Control Room B	Hi Alarm	N/A
ISI Room A	Hi Alarm	N/A
ISI Room B	Hi Alarm	N/A
Primary Containment L/T Measurement Room	Hi Alarm	N/A
Plant Outage Workers Room	Hi Alarm	N/A
Refueling Machine Control Room	Hi Alarm	N/A
CUW Non-Regen. Hx Area	Hi Alarm	N/A
CUW Regen. Hx Area	Hi Alarm	N/A
CUW Valve Room	Hi Alarm	N/A

Table 1
Operating Values of Secondary Containment Parameters
(Cont.)

Secondary Containment Parameter	Maximum Normal Operating Value	Maximum Safe Operating Value
HVAC Exhaust Radiation Level	10^{-5} Gy/h	
Reactor building	0.1	N/A
Refuel floor (Fuel Handling Area)	1	N/A
Area Radiation Level	10^{-5} Gy/h	10^{-5} Gy/h
El. -8200 T.M.S.L. (Floor 100-B3F)		
• HCU Area - 0°	5	—
• RHR(A) Pump & Hx Room	30	—
• RCIC Room	200	—
• HPCF(C) Pump Room	5	—
• RHR(C) Pump & Hx Room	30	—
• CRD Pump Room	5	—
• HCU Area - 180°	5	—
• RHR(B) Pump & Hx Room	30	—
• HPCF(B) Pump Room	5	—
• CUW Non-Regen. Hx Room	20,000	—
• CUW Pump Area	500	—
El. -1700 T.M.S.L. (Floor 200-B2F)		
• TIP Area	5	—
• ECCS Pump Maintenance Area	5	—
• Valve Room A	5	—
• Valve Room C	5	—
• FMCRD & RIP Maintenance Area	5	—
• Valve Room B	5	—
• CUW Regen. Hx & Valve Area	20,000	—
• CUW F/D Valve Area	5	—
El. 4800 T.M.S.L. (Floor 300-B1F)		
• RPV Instrument Area 1 & 3	5	—
• RPV Instrument Area 2 & 4	5	—
• CUW/FPC F/D Area	5	—
El. 12300 T.M.S.L. (Floor 400-1F)		
• Valve Room A	5	—
• Valve Room C	5	—
~~• FCS Area~~	~~5~~	—
• Valve Room B	5	—
• CUW Valve Area	5	—

Table 1
Operating Values of Secondary Containment Parameters
(Cont.)

Secondary Containment Parameter	Maximum Normal Operating Value	Maximum Safe Operating Value
Area Water Level	cm	cm
El. 12300 T.M.S.L. (Floor 400-1F)		
• Valve Room A	5	>20
• Valve Room C	5	>20
~~• FCS Area~~	~~5~~	~~>20~~
• Valve Room B	5	>20
• CUW Valve Area	5	>20
El. 18100 T.M.S.L. (Floor 500-2F)		
• Main Steam/FW Tunnel	5	400
• FPC Area		
- Hx Area	5	>20
- Pump Area	5	100
- Valve Area	5	>20
El. 23500 T.M.S.L. (Floor 600-3F)		
• SRV/MSIV Maintenance Room	5	>20
• SLC Area	5	46
• SGTS Area		
- Fan Area	5	>20
- Filter Train Area	5	>20

18B Differences Between BWROG EPG Revision 4 and ABWR EPG

Table 18B-1 Differences Between BWROG EPG Revision 4 and ABWR EPG

ABWR EPG Step	BWROG EPG Rev. 4 Step	Differences from BWROG Rev. 4 EPG	Basis for Differences
PC/H-1 Override, second bullet item	*PC/H-1 Override, second bullet item*	• *Deleted phrase, "hydrogen mixing systems and" throughout this document.* • Deleted "secure and prevent operation of the FCS and,.."	• *The ABWR ~~Flammability Gas Control System~~ does not have hydrogen mixing systems.* • The ABWR does not have a Flamability Control System.

ABWR EPG Step	BWROG EPG Rev. 4 Step	Differences from BWROG Rev. 4 EPG	Basis for Differences
PC/H-2.1	*PC/H-2.1*	• Added new text: Deleted, not applicable to ABWR. • ~~*Replaced phrase, "place hydrogen recombiners in service taking suction directly on the drywell and operate the drywell hydrogen mixing system", with the phrase, "place FCS in service".*~~ ~~*Added phrase, "and only if suppression pool water level is below [11.70 m(elevation of the suppression pool-to-lower drywell vent)]", as a condition for initiating FCS.*~~ • ~~*Added phrase, "enter [procedure developed from the RPV Control Guideline] at [Step RC-1] and execute it concurrently with this procedure".*~~	• The FCS has been removed from the ABWR standard design. • ~~*The Flammability Gas Control System (FCS) is described in Section 6.2.5. The system equipment is located outside of the drywell. It consists of two blowers that only take suction on the drywell and two hydrogen recombiners. Flow is discharged to the wetwell. An explicit instruction to take suction directly from the drywell is redundant. Instruction is given to place the FCS into service rather than to start the hydrogen recombiners because in the FCS, pushing the FCS Start control switch will start the blower, heater, and recombiner, and align valves.*~~ • ~~*When the recombiners discharge into the wetwell, the vacuum breakers open to allow flow into the lower drywell, through the suppression pool-to-lower drywell vents to the upper drywell. This flow path can only be established if the vents are not covered by water.*~~ • ~~*Step RC-1 requires a reactor scram. Scramming the reactor at this point will make the depressurization (as required in Step PC/H-4) transient less severe and is consistent with the strategies in other sections of the Primary Containment Control Guidelines.*~~
PC/H-2.2	PC/H-2.2	• Added new text: Deleted, not applicable to ABWR. • ~~*Replaced the phrase, "secure any hydrogen recombiner taking suction on the drywell", with the phrase, "secure FCS operation".*~~	• The FCS has been removed from the ABWR standard design. • ~~*See basis for PC/H-2.1 above.*~~

ABWR EPG Step	BWROG EPG Rev. 4 Step	Differences from BWROG Rev. 4 EPG	Basis for Differences
PC/H-3.1	PC/H-3.1	• Added new text: Deleted, not applicable to ABWR. • ~~Deleted phrase, "but only if suppression chamber hydrogen ..." in the first paragraph through "...directly on the suppression chamber" in the second paragraph.~~ ~~In the second paragraph, replaced "taking suction indirectly on the suppression chamber by way of the drywell" with "place FCS in service".~~ • ~~Added instruction to operate the FCS only if suppression pool water level is below the suppression pool-to-lower-drywell vent.~~ • ~~Added phrase, "enter [procedure developed from the RPV Control Guideline] at [Step RC-1] and execute it concurrently with this procedure".~~	• The FCS has been removed from the ABWR standard design. • ~~The ABWR hydrogen recombiners only take suction directly on the drywell. They take suction indirectly on the suppression pool chamber by way of the drywell in conjunction with operation of the blowers of FCS and the vacuum breakers and through the suppression pool-to-lower-drywell vents.~~ • ~~Operation of the FCS will purge the wetwell through the vacuum breakers. The purge flow is mixed with the drywell atmosphere. Recombination takes place in the recombiners located outside of the drywell. Operation of FCS is contingent upon suppression pool water level being below the pool-to-lower drywell vents to allow mixing of the drywell and wetwell atmosphere through these vents. See also discussion of basis for step DW/T-2.~~ • ~~Step RC-1 requires a reactor scram. Scramming the reactor at this point will make the depressurization (as required in Step PC/H-4) transient less severe and is consistent with the strategies in other sections of the Primary Containment Control Guidelines.~~

Table 18F-1
Inventory of Controls Based Upon the ABWR EPGs and PRA (Continued)

No.	Fixed Position Controls
62	RHR(B) Primary Containment Vessel Spray Mode Initiation SW
63	RHR(C) Primary Containment Vessel Spray Mode Initiation SW
64	SGTS(B) Initiation SW
65	SGTS(C) Initiation SW
66	Div I Manual ADS Channel 1 Initiation SW
67	Div I Manual ADS Channel 2 Initiation SW
68	Div II Manual ADS Channel 1 Initiation SW
69	Div II ADS Manual ADS Channel 2 Initiation SW
70	RCIC Div. I Isolation Logic Reset SW
71	RCIC Div. II Isolation Logic Reset SW
72	RCIC Inboard Isolation CS
73	RCIC Outboard Isolation CS
74	RHR(C) Manual Valves For Firewater Injection (F101, F102, F103)*
75	CUW Regenerative Heat Exchanger Manual Bypass Valve*
76	Turbine Building HVAC System Controls*
77	SLC Local Controls*
78	Fire Protection System Motor Pump Control SW†
79	Fire Protection System Diesel Pump Control SW†
80	Control Rod Scram Test Switches†
81	"A" Scram Solenoid Main Power Breaker CS†
82	"B" Scram Solenoid Main Power Breaker CS†
83	RPS Div. I Trip Inhibit SW†
84	RPS Div. II Trip Inhibit SW†
85	RPS Div. III Trip Inhibit SW†
86	RPS Div. IV Trip Inhibit SW†
87	Rod Worth Minimizer Bypass,†
88	CAMS(A) Operating Mode SW†
89	CAMS(B) Operating Mode SW†
90	CAMS(A) Sample Select SW†
91	CAMS(B) Sample Select SW†
92	~~FCS(B) Control SW†~~

Table 18F-1
Inventory of Controls Based Upon the ABWR EPGs and PRA (Continued)

No.	Fixed Position Controls
~~93~~	~~FCS(C) Control SW~~†

* Provided outside the main control room.

† To be provided at main control room area panels, not at the operator control panels.

Table 18F-1
Inventory of Controls Based Upon the ABWR EPGs and PRA (Continued)

No.	Other Control Functions
1	HPCF(B) System controls for terminating system flow, injecting flow, and isolation of potential discharges to reactor building areas
2	HPCF(C) System controls for terminating system flow, injecting flow, and isolation of potential discharges to reactor building areas
3	RCIC System controls for terminating system flow, injecting flow, isolation of potential discharges to reactor building areas, and venting of the RPV to the main condenser
4	RHR(A) System controls for terminating system flow, injecting flow, suppression pool cooling, wetwell spray, drywell spray, shutdown cooling, and isolation of potential discharges to reactor building areas
5	RHR(B) System controls for terminating system flow, injecting flow, suppression pool cooling, wetwell spray, drywell spray, shutdown cooling, and isolation of potential discharges to reactor building areas
6	RHR(C) System controls for terminating system flow, injecting flow, suppression pool cooling, wetwell spray, drywell spray, shutdown cooling, and isolation of potential discharges to reactor building areas
7	Main steamline drain containment isolation valve controls
8	SRV opening and closing controls for each SRV
9	SGTS(B) System controls for venting of the primary containment, and control of secondary containment (reactor building) radiation
10	SGTS(C) System controls for venting of the primary containment, and control of secondary containment (reactor building) radiation
11	RBHVAC containment isolation valves controls
12	ACS containment isolation valves controls
13	SGTS(B) room cooler fan control
14	SGTS(C) room cooler fan control
15	CAMS(A) room cooler fan control
16	CAMS(B) room cooler fan control
17	RHR(A) pump room cooler fan control
18	RHR(B) pump room cooler fan control
19	RHR(C) pump room cooler fan control
20	HPCF(B) pump room cooler fan control
21	HPCF(C) pump room cooler fan control
22	RCIC pump room cooler fan control
~~23~~	~~FCS(B) room cooler fan control~~
~~24~~	~~FCS(C) room cooler fan control~~

Table 18F-2
Inventory of Displays Based Upon the ABWR EPGs and PRA

No.	Fixed Position Displays	No.	Fixed Position Displays
1	RPV Water Level ★★	27	RHR(C) Flow ★★
2	RCIC Turbine Speed	28	RHR(C) Injection Valve Status
3	Wetwell Pressure ★★	29	Emergency Diesel Generator (A) Operating Status ★★
4	Suppression Pool Bulk Average Temperature ★★	30	Emergency Diesel Generator (B) Operating Status ★★
5	HPCF(B) Flow ★★	31	Emergency Diesel Generator (C) Operating Status ★★
6	HPCF(C) Flow ★★	32	Primary Containment Water Level ★★
7	RPV Pressure ★★	33	Condensate Storage Tank Water Level ★★
8	Drywell Pressure ★★	34	SLC Pump(A) Discharge Pressure ★★
9	Reactor Power Level, (Neutron Flux, APRM) ★★	35	SLC Pump(B) Discharge Pressure ★★
10	Reactor Power Level (SRNM) ★★	36	Main Condenser Pressure
11	Reactor Thermal Power ★★	37	SRV Positions ★★
12	MSIV Position Status (Inboard And Outboard Valves) ★★	38	Suppression Pool Level ★★
13	Reactor Mode Switch Mode Indications	39	Main Steamline Flow ★★
14	Main Steamline Radiation ★★	40	SLC Boron Tank Water Level ★★
15	Scram Solenoid Lights(8) Status	41	Recirculation Pump Speeds
16	Manual Scram SW(A) Indicating Light Status	42	Average Drywell Temperature ★★
17	Manual Scram SW(B) Indicating Light Status	43	Wetwell Hydrogen Concentration Level ★★
18	RPV Isolation Status Display ★★	44	Drywell Hydrogen Concentration Level ★★
19	RCIC Flow ★★	45	Drywell Oxygen Concentration ★★
20	RCIC Injection Valve Status	46	Wetwell Oxygen Concentration ★★
21	HPCF(B) Injection Valve Status	~~47~~	~~FCS(B) Operating Status~~
22	HPCF(C) Injection Valve Status	48	~~FCS(C) Operating Status~~
23	RHR(A) Flow ★★	49	Main Stack Radiation Level ★★
24	RHR(A) Injection Valve Status	50	Time

★★ Denotes Regulatory Guide 1.97 Parameter

18F-2 Inventory of Displays Based Upon the ABWR EPGs and PRA (Continued)

No.	Other Displays
1	Reactor building differential pressure
2	Reactor building HVAC exhaust radiation level
3	Fuel handling area ventilation exhaust radiation level
4	RHR(A) pump room cooler operation status
5	RHR(B) pump room cooler operation status
6	RHR(C) pump room cooler operation status
7	HPCF(B) pump room cooler operation status
8	HPCF(C) pump room cooler operation status
9	RCIC pump room cooler operation status
~~10~~	~~FCS(B) room cooler operation status~~
~~11~~	~~FCS(C) room cooler operation status~~
12	FPC(A) room cooler operation status
13	FPC(B) room cooler operation status
14	SGTS(B) room cooler operation status
15	SGTS(C) room cooler operation status
16	CAMS(A) room cooler operation status
17	CAMS(B) room cooler operation status

PRIMARY CONTAINMENT HYDROGEN CONTROL PC/H

[[.........

.... [3]]]

PRIMARY CONTAINMENT HYDROGEN CONTROL PC/H (cont'd)

[[.........

.... {3}]]

[[.........

.... {3}]]

PRIMARY CONTAINMENT HYDROGEN CONTROL PC/H–2

[[.........

.... {3}]]

PRIMARY CONTAINMENT HYDROGEN CONTROL PC/H–2 (cont'd)

[[.........

.... {3}]]

[[.........

.... {3}]]

PRIMARY CONTAINMENT HYDROGEN CONTROL PC/H–2 (cont'd)

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[[.........

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PRIMARY CONTAINMENT HYDROGEN CONTROL PC/H–2 (cont'd)

[[.........

.... [3]]]

PRIMARY CONTAINMENT HYDROGEN CONTROL PC/H–3

[[.........

.... {3}]]

PRIMARY CONTAINMENT HYDROGEN CONTROL PC/H–3 (cont'd)

[[.........

.... {3}]]

[[.........

.... {3}]]

Response

The BWR Owners' Group sponsored a program to evaluate depressurization modes other than full actuation of the ADS. The results of this program were submitted to the NRC in a letter report from D. B. Waters, Chairman of BWR Owners' Group, to D. G. Eisenhut, Director (NRC), dated December 29, 1980. A summary of this evaluation follows.

The cases analyzed in the letter report above show that, based on core cooling considerations, no significant improvement can be achieved by a slower de-pressurization rate. A significantly slower depressurization will result in increased core uncovery times before ECCS injection. Furthermore, a moderate decrease in the depressurization rate necessitates an earlier action time to initiate ADS. Such an earlier actuation time has the negative impact of providing less time for the operator to start high pressure ECCS without obtaining a significant benefit to vessel fatigue usage. This earlier actuation time necessitates a higher initiation level which would result in an increased frequency of ADS actuation.

It should be noted that the ADS is not a normal core cooling system, but is a backup for the high pressure core cooling systems such as feedwater, RCIC or HPCF. If ADS operation is required, it is because normal and/or emergency core cooling is threatened. As a full ADS blowdown is well within the design basis of the RPV and the system is properly designed to minimize the threat to core cooling, no change in depressurization rate is required or appropriate.

19A.2.12 Evaluation of Alternative Hydrogen Control Systems [Item (1) (xii)]

NRC Position

Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of 10 CFR 50.34(f). As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

(1) A comparison of costs and benefits of the alternative systems considered.

(2) For the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of 10 CFR 50.34.

(3) For the selected system, preliminary design descriptions of equipment, function, and layout.

Response

The ABWR primary containment is inerted and is, therefore, protected from hydrogen combustion regardless of the amount or rate of hydrogen generation. In fact, increasing amounts of hydrogen moves the primary containment oxygen concentration further from the flammable regime. ~~The ABWR is also provided with permanently installed~~

~~recombiners which prevent the buildup of oxygen, due to radiolysis, from creating a potentially flammable mixture.~~ Radiolysis is the only potential source of oxygen in the ABWR primary containment.

e deletion of the ammability ontrol System, luding the combiners, from STP 3 & 4 sign, and the sign's capability accomodate ygen from diolysis, is scribed in ibsection 6.2.5.

See Subsection 6.2.7.1 for COL license information pertaining to alternate hydrogen control.

3 Long-Term Training Upgrade [Item (2) (i)]

NRC Position

Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs. (Applicable to construction permit applicants only.) [I.A.4.2]

Response

COL license information, see Subsection 19A.3.1. This will be addressed as part of simulator design which falls under operator training (Section 18.8.8).

19A.2.14 Long-Term Program of Upgrading of Procedures [Item (2) (ii)]

NRC Position

Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Applicable to construction permit applicants only.) [I.C.9]

Response

COL license information, see Subsection 19A.3.2.

19A.2.15 Control Room Design Reviews [Item (2) (iii)]

NRC Position

Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. [I.D.1]

Response

This item is addressed in Subsection 1A.2.2.

19A.2.16 Plant Safety Parameter Display Console (SPDS) [Item (2) (iv)]

NRC Position

Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying

Response

Per the response to Item (1)(xii), refer to Subsection 6.2.5 for a detailed description of the inerting ~~and recombiner~~ systems.

19A.2.22 Testing Requirements [Item (2) (x)]

NRC Position

Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWRs, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. [II.D.11]

Response

This item is addressed in Subsection 1A.2.9.

19A.2.23 Relief and Safety Valve Position Indication [Item (2) (xi)]

NRC Position

Provide direct indication of relief and safety valve position (open or closed) in the control room. [II.D.3]

Response

This item is addressed in Subsection 1A.2.10.

19A.2.24 Auxiliary Feedwater System Automatic Initiation and Flow Indication [Item (2) (xii)]

NRC Position

Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWRs only.) [II.E.1.2]

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type reactors.

19A.2.25 Reliability of Power Supplies for Natural Circulation [Item (2) (xiii)]

NRC Position

Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWRs only.) [II.E.3.1]

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type reactors.

(b) Subarticle NE-3220, Division 1, and Subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraphs (f) (3) (v) (A) (1) and (Q (3) (v) (B) (1) of 10 CFR 50.34, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H St., NW., Washington, D.C.

(2) (a) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subarticle CC-3720, Service Load Category.

(c) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.

Response

(1) The containment design basis accident pressure is 0.412 MPa. The peak pressure resulting from 100% fuel-clad metal water reaction is about 0.618 MPa (Subsection 19E.2.3.2). The containment is capable of withstanding 0.618 MPa internal pressure together with dead load by meeting the code requirements (Subsection 19E.2.3.2).

(2) ABWR does not employ post accident inerting; thus, item (2) does not apply.

~~19A.2.46 Dedicated Penetration [Item (3) (vi)]~~

~~NRC Position~~

~~For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. [II.E.4.1]~~

~~Response~~

~~This item is addressed in Subsection 1A.2.13.~~

19A.2.47 Organization and Staffing to Oversee Design and Construction [Item (3) (vii)]

NRC Position

Provide a description of the management plant for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) Technical resources directed by the applicant; (C) Details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) Proposed procedures for handling the transition to operation; (E) The degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. [II.J.3.1]

Response

COL license information, see Subsection 19A.3.7.

19A.3 COL License Information

19A.3.1 Long-Term Training Upgrade

Simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs shall be provided. (Subsection 19A.2.13.) COL License Information regarding operator training is in Section 18.8.8.

19A.3.2 Long-Term Program of Upgrading of Procedures

A long-term program of upgrading procedures shall be established to begin during construction and following term program of upgrading procedures shall be established to begin during construction and follow into operation for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analysis, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Subsection 19A.2.14.) COL License Information is in Section 13.5.3.1.b.

19A.3.3 Purge System Reliability

A testing program shall be provided to ensure that the large ventilation valves close within the limits assured in the radiologic design bases. (Subsection 19A.2.27.)

19A.3.4 Licensing Emergency Support Facility

The COL applicant has a requirement to provide a near site Emergency Operational Facility (EOF) (See Subsection 19A.2.37).

Table 19A-1 ABWR—CP/ML Rule Cross Reference (Continued)

CP/ML Rule Section	Item Action Plan	Appendix Section	Title	Tier 2 Reference
(xxii)	II.K.2(9)	19A.2.34	Analysis of Upgrading of Integrated Control System	Not Applicable (P&W Only)
(xxiii)	II.K.2.(10)	19A.2.35	Hand-Wired Safety-Grade Anticipatory Reactor Trips	Not Applicable (P&W Only)
(xxiv)	II.K.3(23)	19A.2.36	Central Water Level Recording	Subsection 19A.2.26
(xxv)	III.A.1.2	19A.2.37	Upgrade License Emergency Support Facility	Subsection 19A.3.4
(xxvi)	III.D.1.1	19A.2.38	Primary Coolant Sources Outside the Containment Structure	Subsection 1A.2. 34
(xxvii)	III.D.3.3	19A.2.39	In-Plant Radiation Monitoring	Subsection 19A.3.5
(xxviii)	II.D.3.4	19A.2.40	Control Room Habitibility	Subsection 1A.2. 36
(3) (i)	I.C.5	19A.2.41	Procedures for Feedback of Operating, Design and Construction Experience	Subsection 19A.3.6/13.2.3.1 /13.5.3.3.f
(ii)	I.F.1	19A.2.42	Expand QA List	Subsection 19A.2.42
(iii)	I.F.2	19A.2.43	Develop More Detailed QA Criteria	Subsection 19A.2.43
(iv)	II.B.8	19A.2.44	Dedicated Containment Penetrations, Equivalent to a Single 3-foot Diameter Opening	Subsection 19A.2.44
(v)	II.B.8	19A.2.45	Containment Integrity	Subsection 19A.2.45
~~(vi)~~	~~II.E.4.1~~	~~19A.2.46~~	~~Dedicated Penetration~~	~~Subsection 1A.2.13~~
(vii)	II.J.3.1	19A.2.47	Organization and Staffing to Oversee Design and Construction	Subsection 19A.3.7

Power Reactors" (Reference 19B.2.18-1), and General Design Criterion 41, "Containment Atmosphere Cleanup", in Appendix A to 10 CFR Part 50 (Reference 19B.2.18-2), requires that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

Paragraph (f)(2)(ix) of 10 CFR 50.34, "Contents of Applications; Technical Information" (Reference 19B.2.18-4), requires that provision be made for a hydrogen control system that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal-water reaction.

An inerted containment ~~and the provision for permanently installed hydrogen recombiners are~~ acceptable as hydrogen control measures.

is

Resolution

The issue of a large amount of hydrogen being generated and burned within containment was resolved as stated in the NRC document SECY 89-122 dated April 19, 1989 (Reference 19B.2.18-3). This issue covers hydrogen control measures for recoverable degraded core accidents for all BWRs. Extensive research in this area has led to significant revision of the Commission's hydrogen control regulations, given in 10 CFR 50.44, published ~~December 2, 1981~~.

eptember 6, 2003

The ~~ABWR~~ containment is inerted and per 10 CFR 50.34 (f)(2)(ix) can withstand the pressure and energy addition from a 100% fuel-clad metal-water reaction. However, in the ABWR, there are no design-basis events that result in core uncovery or core heatup sufficient to cause significant metal-water reaction. Section 6.2.5.3 states that this is equivalent to the reaction of the active clad to a depth of 5.842E-3 mm (0.00023 inches) or 0.72% of the active clad.

Therefore, this issue is resolved for the ABWR.

References

19B.2.18-1 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors", Office of the Federal Register, National Archives Records Administration.

19B.2.18-2 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants."

19B.2.18-3 SECY-89-122, "Resolution of Unresolved Safety Issue A-48, Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment", April 1989.

19B.2.18-4 10 CFR 50.34, "Contents of Applications; Technical Information", Office of the Federal Register, National Archives Records Administration.

The radiation loads on the penetrations are below the TID-14844 limits so radiation is not a concern.

~~(11) Recombiners~~

~~The recombiner system is needed in a long term accident (order of days) to ensure that the oxygen concentration does not reach flammability limits. The recombiners are located outside of the primary containment. Piping is used to remove and return fluid to the primary containment. Therefore, the process fluid provides the only significant impact on this system. Since the supply and return lines are isolated during the early part of an event, the recombiners are not subjected to the primary containment thermodynamic loads until days later, after accident recovery when the environment is not as severe. At this time, recovery from a postulated accident might occur in a much less severe environment. Additionally, the integrated radiation doses will be well below the design basis values. Therefore, the recombiners will survive these accident scenarios.~~

(12) Pressure and Water Level Instrumentation

This page was replaced by page C-217

The pressure sensors used to measure both water level and pressure in the vessel and in the containment are located outside of containment. The conditions in the vessel and containment are monitored via pressure taps. The pressure sensors will not see the higher vessel or primary containment temperature and radiation doses due to the significant length-to-diameter ratio of the piping used in these sensors. The integrated radiation gamma dose for the pressure sensors is slightly over the equipment qualification limit set forth in Table 3I-16. However, the radiation limits set for design basis events are extremely conservative. Therefore, there is reasonable assurance that the sensors will survive this condition. Furthermore, the sensors are capable of withstanding very high overpressure events, on the order of 14 MPa, indicating that there is no possibility of damage from high containment pressures.

(13) Temperature Instrumentation

The GE standard practice is to use thermocouples rated to 575 K and 14 MPa. These ratings are well above the drywell and wetwell thermodynamic loads experienced during a postulated severe accident. Therefore, operation of the thermocouples should not be adversely affected. Comparison to radiation qualification limits are based on two day integrated dose rates. The equipment integrated radiation doses are below the equipment qualification dose rates of 2.0E+8 R and 2.0E+9 R for gamma and beta radiation, respectively, as set forth in Table 3I-16.

Table 19E.2-1 Potential Suppression Pool Bypass Lines (Continued)

Description	Number of Lines	Pathway From	Pathway To	Size (mm) (1 in. = 25.4 mm)	Isolation Valves	Basis For Exclusion (See Notes)
LDS Instruments	9	RPV	RB	6	CK	-
SPCU Suction	1	SP	RB	200	MO, MO	2
SPCU Return	1	SP	RB	250	MO, CK	2
Cont. Atmosphere Monitor	6	DW	RB	20	MO	-
LDS Samples	2	DW	RB	30	(SO, SO)	-
Drywell Sump Drains	2	DW	RB	100	MO, MO	-
HVCW/RBCW Supply	4	DW	RB	125	CK, MO	1
HVCW/DWCW Return	4	DW	RB	125	MO, MO	1
DW Exhaust/SGTS	1	DW	RB	550	AO, AO	7
Wetwell Vent to SGTS	1	WW	RB	550	AO, AO	2
DW Purge	1	DW	RB	350	AO	-
Inerting Makeup	1	DW	WW	50	AO, AO	-
WW Inerting/Purge	1	WW	RB	550	AO, AO	2
Instrument Air (and Service Air)	2	DW	RB	50	CK, MO	1
SRV Pneumatic Supply	3	DW	RB	50	CK, MO	1
Flammability ~~Control~~	~~2~~	~~DW~~	RB	~~100~~	(~~AO, MO~~)	3
ADS/SRV Discharge	8	RPV	WW	300	RV	-
ACS Supply	2	DW	WW	550	AO, AO	-
WW/DW Vacuum Breaker	8	DW	WW	500	CK	-
Miscellaneous Leakage	1	DW	RB	---	NONE	6
Access Tunnels	2	DW	RB	---	NONE	6

NOTES:

Legends and Acronyms

Pathway

Source (From)		Termination (To)	
RPV	Reactor Pressure Vessel	WW	Wetwell
DW	Drywell	RB	Reactor Building
SP	Suppression Pool	WW	Wetwell
		ST	Steam Tunnel

Table 19E.2-29 Equipment and Instrumentation Required to Survive Severe Accident Scenarios

Equipment and Instrumentation	10CFR50.34(f)	In-Vessel Severe Accident	Ex-Vessel Severe Accident
Equipment			
RHR	+	+	+
ADS	+	+	-
ACIWA	+	+	+
Containment Structure	+	+	+
Pedestal	+	+	+
CIVs - Inboard	+	+	+
CIVs - Outboard	+	+	+
Electrical Penetrations	+	+	+
Mechanical Penetrations	+	+	+
Hatches	+	+	+
Passive Flooder	-	-	+
COPS	-	+	+
Vacuum Breakers	+	+	+
RIP Vertical Restraints	+	+	+
Recombiners	+	+	+
Instrumentation			
RPV Water Level	+	+	-
RPV Pressure	+	+	-
Suppression Pool Water Temperature	+	+	+
DW/WW Radiation Monitor	+	+	+
DW/WW H_2 Concentration	+	+	+
DW/WW O_2 Concentration	+	+	+
DW Temperature	+	+	+
DW Pressure	+	+	+
WW Pressure	+	+	+

Figure 19R-6 Reactor Building Arrangement—Elevation 12300 mm (1F)

{{{Security-Related Information - Withheld Under 10 CFR 2.390}}}

Table 20B-1 Equipment Data Base (Continued)

POWSUP	DIV	MPLNO
DC A10	1	P41-DPS004A
DC A10	1	P41-DPS004D
DC A10	1	P41-DPT004A
DC A10	1	P41-DPT004D
DC A10	1	P41-DPT004G
DC A10	1	P54-PIS001A
DC A10	1	P54-PT002A
DC A10	1	R43-DPS091A*
DC A10	1	R43-LIS191A*
DC A10	1	R43-LS142A*
DC A10	1	R43-LS395A*
DC A10	1	T31-F001
DC A10	1	T31-F008
DC A10	1	T31-F009
DC A10	1	T31-F010
DC A10	1	T31-F025
DC A10	1	T31-F039
DC A10	1	T31-LT058A
DC A10	1	T31-LT059A
DC A10	1	T31-LT100A
~~DC A10~~	~~1~~	~~T49-F002B-1~~
~~DC A10~~	~~1~~	~~T49-F002C-1~~
~~DC A10~~	~~1~~	~~T49-F007A-1~~
~~DC A10~~	~~1~~	~~T49-F007B-1~~
~~DC A10~~	~~3~~	~~T49-F013A~~
~~DC A10~~	~~3~~	~~T49-F014A~~
~~DC A10~~	~~3~~	~~T49-F016A~~
~~DC A10~~	~~3~~	~~T49-FT002A~~
~~DC A10~~	~~3~~	~~T49-FT004A~~
~~DC A10~~	~~3~~	~~T49-LS011A~~
~~DC A10~~	~~3~~	~~T49-LS012A~~
~~DC A10~~	~~3~~	~~T49-LS013A~~

Table 20B-1 Equipment Data Base (Continued)

POWSUP	DIV	MPLNO
~~DC A10~~	3	~~T49-PT003A~~
~~DC A10~~	3	~~T49-TE001A~~
~~DC A10~~	3	~~T49-TE005A~~
~~DC A10~~	3	~~T49-TE006A-1~~
~~DC A10~~	3	~~T49-TE006A-2~~
~~DC A10~~	3	~~T49-TE007A-1~~
~~DC A10~~	3	~~T49-TE007A-2~~
~~DC A10~~	3	~~T49-TE008A-1~~
~~DC A10~~	3	~~T49-TE008A-2~~
~~DC A10~~	3	~~T49-TE009A-1~~
~~DC A10~~	3	~~T49-TE009A-2~~
~~DC A10~~	3	~~T49-TE010A-1~~
~~DC A10~~	3	~~T49-TE010A-2~~
~~DC A10~~	3	~~T49-TE011A~~
~~DC A10~~	3	~~T49-TT609A~~
DC A10	1	T53-TE001A
DC A10	1	T53-TE001E
DC A10	1	T53-TE001J
DC A10	1	T53-TE001N
DC A10	1	T53-TE002A
DC A10	1	T53-TE002E
DC A10	1	T53-TE002J
DC A10	1	T53-TE002N
DC A10	1	T53-TE003A
DC A10	1	T53-TE003E
DC A10	1	T53-TE003J
DC A10	1	T53-TE003N
DC A10	1	T53-TE004A
DC A10	1	T53-TE004E
DC A10	1	T53-TE004J
DC A10	1	T53-TE004N
DC A10	1	T53-TE005A

(10) Not Used

(11) Isolates the drywell sumps drain lines

(12) Isolates the fission products monitor sampling and return lines

(13) Initiates withdrawal of the automated traversing incore probe

In addition to the above functions, LDS monitors leakage inside the drywell from the following sources and annunciates the abnormal leakage levels in the control room:

(1) Fission products releases

(2) Condensate flow from the drywell air coolers

(3) Drywell sump level changes

(4) Leakages from valve stems equipped with leak-off lines

Other leakages from the FMCRDs, the SRVs and from the reactor vessel head seal flange are monitored by their respective systems.

1.2.2.5.4 Reactor Core Isolation Cooling System

The RCIC System provides makeup water to the reactor vessel when the vessel is isolated and is also part of the emergency core cooling network. The RCIC System uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel for events defined in Section 5.4.

One division contains the RCIC System, which consists of a steam-driven turbine which drives a pump assembly and the turbine and pump accessories. The system also includes piping, valves, and instrumentation necessary to implement several flow paths. The RCIC steam supply line branches off one of the main steamlines (leaving the RPV) and goes to the RCIC turbine with drainage provision to the main condenser. The turbine exhausts to the suppression pool with vacuum breaking protection. Makeup water is supplied from the condensate storage tank (CST) or the suppression pool with preferred source being the CST. RCIC pump discharge lines include the main discharge line to the feedwater line, a test return line to the suppression pool, a minimum flow bypass line to the suppression pool and a cooling water supply line to auxiliary equipment.

Following a reactor scram, steam generation in the reactor core continues at a reduced rate due to the core fission product delay heat. The turbine condenser and the feedwater system supply the makeup water required to maintain reactor vessel inventory.

1.2.2.15.5 PCV Pressure and Leak Testing Facility

The PCV pressure and leak testing facility is a special area just outside the containment. It provides instrumentation for conducting the PCV pressure and integrated leak rate tests.

1.2.2.15.6 Atmospheric Control System

The Atmospheric Control System is designed to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during plant shutdown for refueling or maintenance.

The Atmospheric Control System is summarized in Subsection 6.2.5.2.1.

1.2.2.15.7 Drywell Cooling System

The Drywell Cooling System is summarized in Subsection 9.4.9.2.

1.2.2.15.8 Not Used

~~A recombiner system is provided to control the concentration of hydrogen and oxygen produced by metal-water reaction and radiolysis following a design basis accident in the primary containment.~~

1.2.2.15.9 Suppression Pool Temperature Monitoring System

The Suppression Pool Temperature Monitoring (SPTM) System is summarized in Subsection 7.6.1.7.1.

1.2.2.16 Structures and Servicing Systems

1.2.2.16.1 Foundation Work

The analytical design and evaluation methods for the containment and Reactor Building walls, slabs and foundation mat and foundation soil are summarized in Subsection 3.8.1.4.1.1.

1.2.2.16.2 Turbine Pedestal

The description for the turbine pedestal is the same as that for foundation work in Subsection 3.8.1.4.1.1.

1.2.2.16.3 Cranes and Hoists

The cranes and hoists are summarized in Subsection 9.1.

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	0	11/70	Yes	
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors	2	6/74	Yes	
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors	2	6/74	No	PWR only
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steamline Break Accident for Boiling Water Reactors	0	3/71	Yes	
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	0	3/71	Yes	
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	3 ~~2~~	3/07 ~~11/78~~	Yes	
1.8	Personnel Selection and Training	--	--	--	See Table 17.0-1
1.9	Selection, Design, Qualification, and Testing of Emergency Diesel-Generator Units Used As Class 1E Onsite Electric Power Systems at Nuclear Plants	3	7/93	Yes	
1.11	Instrument Lines Penetrating Primary Reactor Containment	0	3/71	Yes	
1.12	Instrumentation for Earthquakes	1	4/74	No	NA
1.13	Spent Fuel Storage Facility Design Basis	1	12/75	Yes	
1.14	Reactor Coolant Pump Flywheel Integrity	1	8/75	No	PWR only
1.16	Reporting of Operating Information — Appendix A Technical Specifications	4	8/75	---	COL Applicant
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	2	5/76	Yes	

(2) There shall be onsite capability to perform the following within the 3 hour time period:

(a) Determine the presence and amount of certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage. Meets the requirements of NUREG-0737.

) Hydrogen in containment atmosphere. Hydrogen in containment atmosphere is measured by the ontainment Atmospheric Monitoring System. Meets the requirements of NUREG-0737 with the exception that e design follows the guidance of RG 1.7 Rev. 3, which permits the hydrogen monitor to be classified as non-ıfety related.

(c) Dissolved gases, chloride and boron in liquids. Dissolved gases are discussed in item 4 below. Meets the requirements concerning chloride and boron of NUREG-0737.

(d) Inline monitoring capability is acceptable. No inline monitors are provided in PASS.

(3) Sampling need not depend upon an isolated auxiliary system being put into operation. Meets the requirements of NUREG-0737.

(4) Reactor coolant samples and analyses for total dissolved gases and hydrogen are required. During a severe core damage accident for the ABWR, the reactor water will become mixed with the suppression pool water. The pressure in the reactor vessel will decrease to approximately the pressure within the wetwell and the drywell. As a result of this decrease in pressure, dissolved gases will partially pass out of the water phase into the gas phase. Data on gases dissolved in the reactor water under these conditions will have little meaning in responding to the accident. During accidents in which only a small amount of cladding damage has occurred or in accidents in which the reactor vessel has not been depressurized, pressurized reactor water samples may be obtained from the Process Sampling System. Therefore, the ability to obtain pressurized or depressurized reactor water samples for dissolved gas analyses has not been provided.

(5) If both of the following are present:

(a) There is only a single barrier between primary containment and the cooling water.

(b) If the cooling water is sea water or brackish water, chloride analysis within 24 hours after sampling shall be provided. If both are not present, the time to complete the analyses is increased to 4 days. Analysis does not have to be done onsite. Meets the requirements of NUREG-0737. (Note that there are several barriers to prevent chloride intrusion from the power cycle cooling water into the reactor vessel. These barriers are: the

probability of steamline flooding by ECCS is extremely low. There is no high drywell pressure signal that would inhibit this logic system.

In the ABWR design, each of three RHR shutdown cooling lines has its own separate containment penetration and its own separate source of suction from the reactor vessel. Alternate shutdown using the SRV is therefore not required for the ABWR in order to meet single failure rules. Hence, the ABWR does not require SRV testing with liquid under low pressure conditions associated with this event as required in past BWRs.

1A.2.10 Relief and Safety Valve Position Indication [II.D.3]

NRC Position

Reactor Coolant System relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

Response

The ABWR Standard Plant SRVs are equipped with position sensors which are qualified as Class 1E components. These are used to monitor valve position.

In addition, the downstream pipe from each valve line is equipped with temperature elements which signal the annunciator and the plant process computer when the temperature in the tailpipe exceeds the predetermined setpoint.

These sensors are shown on Figure 5.1-3 (Nuclear Boiler System P&ID).

1A.2.11 Systems Reliability [II.E.3.2]

This TMI action plan item superseded by USI A-45. USI A-45 is addressed in Appendix 19B.

1A.2.12 Coordinated Study of Shutdown Heat Removal Requirements [II.E.3.3]

This TMI action plan item superseded by USI A-45. USI A-45 is addressed in Appendix 19B.

1A.2.13 Containment Design—Dedicated Penetration [II.E.4.1]

NRC Position

For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere.

Response

The Flammability Control System, including the recombiners, has been deleted from the ABWR design as described in Subsection 6.2.5. Accordingly, no penetrations are required for the recombiners.

The hydrogen and oxygen monitors are declassified to non-safety related, as permitted by Regulatory Guide 1.7, Rev. 3

Response

The ABWR Standard Plant is designed in accordance with Regulatory Guide 1.97 . A detailed assessment of the Regulatory Guide, including the list of instruments, is found in Section 7.5.

1A.2.18 Safety-Related Valve Position Indication [II.K.1(5)]

NRC Position

(1) Review all valve positions and positioning requirements and positive controls and all related test and maintenance procedures to assure proper ESF functioning, if required.

(2) Verify that AFW valves are in the open position.

Response

(1) The ABWR Standard Plant is equipped with status monitoring that satisfies the requirements of Regulatory Guide 1.47. See Subsection 7.1.2 for detailed information on the status monitoring equipment and capability provided in the ABWR Standard Plant design.

In addition to the status monitoring, the COL applicant plant-specific procedures (Subsection 1A.3.2) will assure that independent verification of system lineups is applied to valve and electrical lineups for all safety-related equipment, to surveillance procedures, to restoration following maintenance and to comply with IE Bulletin 79-08. Through these procedures, approval will be required for the performance of surveillance tests and maintenance, including equipment removal from service and return to service.

(2) This requirement is not applicable to the ABWR. It applies only to Babcock & Wilcox designed reactors.

1A.2.19 Review and Modify Procedures for Removing Safety-Related Systems from Service [II.K.1(10)]

NRC Position

Review and modify (as required) procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. The COL applicant must verify the operability of safety-related systems after performing maintenance or tests as part of the test to restore a system to service.

Response

See Subsection 1A.3.2 for COL license information requirements.

Response

All of the generic February 21, 1980 GE responses were reviewed and updated for the ABWR Standard Plant. The specific responses are provided in Table 1A-1.

1A.2.34 Primary Coolant Sources Outside Containment Structure [III.D.1.1(1)]

NRC Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate Leak Reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment

(b) Measure actual leakage rates with systems in operation and report them to the NRC

(2) Continuing Leak Reduction—establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Response

Leak reduction measures of the ABWR Standard Plant include a number of barriers to containment leakage in the closed systems outside the containment. These closed systems include:

(1) Residual Heat Removal

(2) High Pressure Core Flooder

(3) Low Pressure Core Flooder

(4) Reactor Core Isolation Cooling

(5) Suppression Pool Cleanup

(6) Reactor Water Cleanup

(7) Fuel Pool Cooling and Cleanup

(8) Post-Accident Sampling

(9) Process Sampling

(10) Containment Atmospheric Monitoring

(11) Fission Product Monitor (Part of LDS)

~~(12) Hydrogen Recombiner~~

(13) Standby Gas Treatment

safety-related, or included in the systems of Table 3.2-1. It does however, represent principal components which are needed to operate, generally during post accident operations. For example, most ECCS valves are normally open, and only a pump discharge valve needs to open to direct water to the reactor. Similarly, the instrument transmitters shown are those which would provide information on long-term system performance post-accident. Control room instrumentation is not listed, since it is all in an accessible area where no irradiation degradation would be expected. Passive elements such as thermocouples and flow sensors are not listed although they are environmentally qualified. The components listed under main steam (B21) are those for ECCS function or monitoring reactor vessel level. Suppression pool level is included with the HPCF instrumentation.

1AA.5.1.3 Combustible Gas Control Systems and Auxiliaries

Flammability control in the primary containment is achieved by an inert atmosphere during all plant operating modes except operator access for refueling and maintenance ~~and a recombiner system to control oxygen produced by radiolysis~~. The high pressure nitrogen (HPIN) gas supply is described in Subsection 1.2.2.12.13. The Containment Atmospheric Monitoring System (CAMS) measures and records containment oxygen/hydrogen concentrations under post-accident conditions. It is automatically initiated by detection of a LOCA (Subsection 7.6.1.6). Table 1AA-3 lists the combustible gas control principal components and their locations.

1AA.5.1.4 Fission Product Removal and Control Systems and Auxiliaries

Engineered Safety Feature (ESF) filter systems are the Standby Gas Treatment System (SGTS) and the control building Outdoor Air Cleanup System. Both consist of redundant systems designed for accident conditions and are controlled from the control room. The SGTS filters the gaseous effluent from the primary and secondary containment when required to limit the discharge of radioactivity to the environment. The system function is described in Subsection 1.2.2.15.4.

A portion of the Control Building heating ventilating and air-conditioning (HVAC) provides detection and limits the introduction of radioactive material and smoke into the control room. This portion is described Subsection 9.4.1.1.3.

The CAMS described in the previous section also measures and records containment area radiation under post-accident conditions. A post-accident sampling system (PASS) obtains containment atmosphere and reactor water samples for chemical and radiochemical analysis in the laboratory. Delayed sampling, shielding, remote operated valves and sample transporting casks are utilized to reduce radiation exposure. The samples are manually transported between the PASS room in the Reactor Building and the analysis laboratory in the service building. The system is described in

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
D1 Process Radiation Monitoring System (includes gaseous and liquid effluent monitoring)						
1. Electrical modules—with safety-related functions (including monitors)	3	SC,X,RZ	—	B	I	
2. Cable with safety-related functions	3	SC,X,RZ	—	B	I	
3. Electrical Modules, other	N	T,SC,RZ, X,W	—	E	—	(u)
4. Cables, other	N	T,SC,RZ, X,W	—	E	—	(u)
D2 Area Radiation Monitoring System	N	X,T,W, SC,RZ,H	—	E	—	
D3 Containment Atmospheric Monitoring System						
1. Component with safety-related function	3	C,SC,X RZ	—	B	I	
2. Nonsafety-related hydrogen and oxygen monitors	N	C, SC, X RZ	--	E	--	

listed on pages 3.2-54 ~~through 3.2~~-61

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves[1]

MPL	System	Pump Page No.	Valve Page No.
B21	Nuclear Boiler		3.9-99
B31	Reactor Recirculation		3.9-102
C12	Control Rod Drive		3.9-103
C41	Standby Liquid Control	3.9-99	3.9-103
C51	Neutron Monitoring (ATIP)		3.9-104
D23	Containment Atmospheric Monitoring		3.9-104
E11	Residual Heat Removal	3.9-99	3.9-105
E22	High Pressure Core Flooder	3.9-99	3.9-110
E31	Leak Detection & Isolation		3.9-112
E51	Reactor Core Isolation Cooling	3.9-99	3.9-113
G31	Reactor Water Cleanup		3.9-118
G41	Fuel Pool Cooling & Cleanup		3.9-119
G51	Suppression Pool Cleanup		3.9-121
K17	Radwaste		3.9-121
P11	Makeup Water (Purified)		3.9-121
P21	Reactor Building Cooling Water	3.9-99	3.9-121
P24	HVAC Normal Cooling Water		3.9-127
P25	HVAC Emergency Cooling Water	3.9-99	3.9-127
P41	Reactor Service Water	3.9-99	3.9-130
P51	Service Air		3.9-131
P52	Instrument Air		3.9-131
P54	High Pressure Nitrogen Gas Supply		3.9-131
T22	Standby Gas Treatment		3.9-132
T31	Atmospheric Control		3.9-134
~~T49~~	~~Flammability Control~~		~~3.9-137~~
U41	Heating, Ventilating and Air Conditioning		3.9-138
Y52	Oil Storage and Transfer	3.9-99	3.9-139
See page 3.9-139 for notes.			

1 This table responds to NRC Questions 210.47, 210.48 and 210.49 regarding provisions for inservice testing of safety-related pumps and valves within the scope of the ABWR Standard Plant in accordance with the ASME Code. The information is presented separately for each system for the MPL number.

maintain an

(6) The containment structure is protected from or designed to withstand hypothetical missiles from internal sources and uncontrolled motion of broken pipes which could endanger the integrity of the containment.

(7) The containment structure provides means to channel the flow from postulated pipe ruptures in the drywell to the pressure suppression pool.

(8) The containment system is designed to allow for periodic tests at the calculated peak or reduced test pressure to measure the leakage from individual penetrations, isolation valves and the integrated leakage rate from the structure to confirm the leaktight integrity of the containment.

(9) The Atmospheric Control System (ACS) establishes and maintains the containment atmosphere to less than 3.5% by volume oxygen during normal operating conditions to ~~assure that~~ inert atmosphere ~~operation of two permanently installed recombiners can be initiated on high levels as determined by the Containment Atmospheric Monitoring System (CAMS)~~.

6.2.1.1.2 Design Features

The containment structure consists of the following major components (shown in Figures 1.2-2 through 1.2-12):

(1) A drywell (DW), which is comprised of two volumes:

(a) An upper drywell (UD) volume surrounding the reactor pressure vessel (RPV) and housing the steam and feedwater lines and other connections of the reactor primary coolant system, safety/relief valves (SRVs) and the drywell HVAC coolers.

(b) A lower drywell (LD) volume housing the reactor internal pumps, fine motion control rod drives (FMCRD) and undervessel components and servicing equipment. The UD is a cylindrical, reinforced concrete structure with a removable steel head and a reinforced concrete diaphragm floor. The cylindrical RPV pedestal, which is connected rigidly to the diaphragm floor, separates the LD from the wetwell. It is a prefabricated steel structure filled with concrete after erection. Ten drywell connecting vents (DCVs), approximately 1m x 2m in cross-section, are built into the RPV pedestal and connect the UD and LD. The DCVs are extended downward via 1.2m inside diameter steel pipes, each of which has three horizontal 0.7m diameter vent outlets into the suppression pool.

assurance that the drywell or wetwell is uniformly mixed. ~~The FCS consists of the following features:~~

~~(1) The FCS has two recombiners installed in the secondary containment. The recombiners process the combustible gases drawn from the primary containment drywell.~~

~~(2) The FCS is activated when a LOCA occurs. The oxygen and hydrogen remaining in the recombiners after having been processed are transmitted to the suppression pool.~~

The ACS provides and maintains an inert atmosphere in the primary containment during plant operation. The system is not designed as a continuous containment purging system. The ACS exhaust line isolation valves are closed when an inert condition in the primary containment has been established. The nitrogen supply makeup lines, compensating for leakage, provide a makeup flow of nitrogen to the containment. If a LOCA signal is received, the ACS valves close. Nitrogen purge from the containment occurs during shutdown for personnel access. Purging is accomplished with the containment inlet and exhaust isolation valves opened to the selected exhaust path and the nitrogen supply valves closed. Nitrogen is replaced by air in the containment (see Item (3) Shutdown-Deinerting below this subsection). The system has the following features:

(1) Atmospheric mixing is achieved by natural processes. Mixing will be enhanced by operation of the containment sprays, which are used to control pressure in the primary containment.

(2) The ACS primary containment nitrogen makeup maintains an oxygen-deficient atmosphere ($\leq$3.5% by volume) in the primary containment during normal operation.

(3) The redundant oxygen analyzer system (CAMS) measures oxygen in the drywell and suppression chamber. Oxygen concentrations are displayed in the main control room. ~~Description of safety-related display instrumentation for containment monitoring is provided in Chapter 7. Electrical requirements for equipment associated with the combustible gas control system are in accordance with the appropriate IEEE standards as referenced in Chapter 7.~~

In addition, the ACS provides overpressure protection to relieve containment pressure, as required, through a pathway from the wetwell airspace to the stack. The pathway is isolated during normal operation by a rupture disk.

(2) Drywell Cooling System: Provides circulation to all portions of the upper and lower drywell, the drywell head area, and the vessel support skirt area to accomplish the mixing necessary for completion of either the inerting or deinerting process and provides representative oxygen samples to the CAMS oxygen sensors. Should the arrangement of the RPV insulation leave a significant gap between itself and the RPV, forced circulation will be provided to that area. Portions of the drywell will be inerted to sufficiently below 3.5% such that the bulk average oxygen concentration does not exceed 3.5% oxygen

(3) HVAC System: (1) supplies the drywell and wetwell exhaust flow during inerting, deinerting, and shutdown venting (2) accommodates drywell bleedoff flows during startup, (3) provides sufficient air flow to limit the concentration of any nitrogen leaking from the primary containment into the secondary containment, and (4) supplies air for purging the primary containment during deinerting and shutdown venting. Nitrogen leaking from the primary containment is insignificant and does not impact HVAC design.

The two outdoor air intakes of the Control Room habitability HVAC System are located far apart to protect personnel in the control room in the event of a nitrogen pipe or storage tank rupture. Similarly, intakes for all HVAC systems are located to minimize the introduction of nitrogen from such ruptures into occupied areas of the plant.

(4) High Pressure Nitrogen Gas Supply System: Serves all pneumatically-operated components in the primary containment because the containment is inerted. The pneumatic devices in the primary containment or those which could leak into the primary containment are supplied with nitrogen for the purpose of preventing oxygen addition to the inerted volumes. The High Pressure Nitrogen Gas Supply System is supplied from the ACS nitrogen storage tank and a bank of nitrogen storage cylinders.

(5) Standby Gas Treatment System: Processes any drywell bleedoff, inerting, and deinerting exhaust flows, as required by offsite release constraints.

(6) Containment Atmospheric Monitoring System: Monitors oxygen levels in the wetwell and drywell ~~during accident conditions~~ to confirm the primary containment oxygen level is kept within limits.

Radiation monitoring in the plant vent, part of Process Radiation Monitoring, detects high radiation during deinerting.

There are no potential sources of oxygen in the containment other than that resulting from radiolysis of the reactor coolant. Consideration of potential sources of leakage of

Boiler System. If drywell pressure exceeds a given setpoint, the nitrogen makeup flow is shut off as the inerting valves are closed. The temperature of the makeup and inerting vaporizers nitrogen outlet are monitored. Low makeup vaporizer nitrogen outlet temperature alarms (only) in the main control room. Auxiliary steam feeding the main inerting vaporizer(s) is controlled to regulate the inerting vaporizer nitrogen outlet temperature. Low inerting vaporizer nitrogen outlet temperature sounds a local alarm and low-low temperature isolates the main inerting line. It is intended that the local panel be attended full-time during all main inerting operations. All locally-mounted instruments are easily read from the local ACS panel. Keylocked switches in the main control room are provided to override the containment isolation signal to the valves, providing nitrogen makeup to the drywell and wetwell and the small 50A pipe size drywell vent line. Position indication in the main control room is provided for all remotely-operated valves.

Backup purge and the addition of makeup nitrogen is initiated at the operator's discretion.

Design details and logic of the instrumentation are discussed in Chapter 7.

As discussed in Subsection 6.2.5.2, ~~safety-grade~~ oxygen monitoring is provided in the wetwell and drywell by the CAMS. This monitoring function, when used during normal operation, determines when the primary containment is inert and nitrogen purging may be terminated. It also determines when primary containment is de-inerted and personnel re-enter procedures may be initiated.

The CAMS oxygen monitors assure safe personnel entry into the primary containment after shutdown. In addition, CAMS assures that the primary containment is in an inert condition during startup, normal and abnormal operation conditions. This system has a measurement range of 0 to 25% (by volume) at 100% relative humidity. The minimum and maximum inlet temperature to the oxygen monitor will be 10°C and 65°C, respectively. Two sample points are provided in both the drywell and wetwell, high and low in their respective compartments and in opposing quadrants. Each airlock can also be sampled.

The sample lines are sized and sloped to assure draining condensation to the containment.There are no loops in the sample lines which could collect water and block flow. The oxygen monitors provide indication outside of the primary containment where necessary (for example, at and in each airlock) to assure safe operator access into first the airlock and then the containment.

The CAMS oxygen analyzing system is provided to indicate the concentration of oxygen inside the containment during reactor operation, and to aid in maintaining the oxygen concentration below a safety limit prescribed in the plant Technical Specifications. The oxygen analyzing system readings are not used as a basis for determining when drywell

(o) Furthermore, these valves are subject to ASME leak rate tests as in (k) above.

(p) Rupture discs are normally closed and sealed from leakage. The opening setpoint of these rupture discs is higher than primary containment test pressures. Additionally, these rupture discs are subject to the Type A test.

(q) SPCU suction line is always filled with water, since it is located below the suppression pool water level and is sealed from the containment atmosphere.

(r) SPCU return line terminates below the suppression pool water level and is sealed from the containment atmosphere.

(s) The outboard side of these valves is always pressurized with nitrogen gas at a pressure higher than the post-accident peak containment pressure. The nitrogen supply in these lines is required for post-accident mitigating function.

(t) The outboard side of these valves is always filled with water and pressurized above 110% post-accident peak containment pressure. These lines are kept charged with cooling water for cooling emergency equipment necessary for post-accident mitigation.

(u) Line will be drained and tested with air.

Not Used

(v) ~~Flammability control is a closed loop, safety grade system required to be functional post accident. Whatever is leaking (if any) is returned to the primary containment. In addition, during ILRT, these valves are opened and the lines are subjected to Type A test.~~

(w) These lines terminate below the drywell sumps water level and are sealed from the containment atmosphere.

(x) The outboard side of these valves are provided with a water leg. In addition, these valves are subject to ASME leak tests as in (k) above.

(y) Not applicable.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-37	RCIC Turbine Steam	14450	80	1200	550		A
X-38	RPV Head Spray	14450	310	1500	550		A
X-50	CUW Pump Feed	14480	310	0	600		A
X-60	MUWP Suction	13500	290	0	200		A
X-61	RCW Suction (A)	13500	45	-3000	200		A
X-62	RCW Return (A)	13500	45	-2000	200		A
X-63	RCW Suction (B)	13500	225	3400	200		A
X-64	RCW Return (B)	13500	225	2400	200		A
X-65	HNCW Suction	13500	225	250	350		A
X-66	HNCW Return	13500	225	1400	350		A
X-69	SA	19000	42	0	90		A
X-70	IA	9000	46	0	200		A
X-71A	ADS Accumulator (A)	19000	50	0	200		A
X-71B	ADS Accumulator (B)	19000	296.5	1000	200		A
X-72	Relief Valve Accumulator	19000	296.5	2000	200		A
X-80	Drywell Purge Suction	13700	68	0	550		A
X-81	Drywell Purge Exhaust	19000	216	0	550		A
X-82	~~FCS Suction~~ Spare	14850	225	-600	150	Welded Cap	C
X-90	Spare	20100	46	0	400		A
X-91	Spare	20100	296.5	1000	400		A
X-92	Spare	16400	45	12700	400		A
X-93	Spare	14700	135	-500	400		A
X-100A	RIP Power	13500	55	-1100	450	O-ring	B

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-104C	FMCRD Position Indicator	20100	99	0	300	O-ring	B
X-104D	FMCRD Position Indicator	20100	279.5	0	300	O-ring	B
X-104E	FMCRD Position Indicator	19000	81	1350	300	O-ring	B
X-104F	FMCRD Position Indicator	19000	260.5	1350	300	O-ring	B
X-104G	FMCRD Position Indicator	19000	99	0	300	O-ring	B
X-104H	FMCRD Position Indicator	19000	279.5	0	300	O-ring	B
X-105A	Neutron Detection	20100	81	1350	300	O-ring	B
X-105B	Neutron Detection	20100	260.5	1350	300	O-ring	B
X-105C	Neutron Detection	20100	99	-5250	300	O-ring	B
X-105D	Neutron Detection	20100	279.5	1350	300	O-ring	B
X-110	Spare ~~FCS~~-Suction	13500	55	1000	300	Welded Cap	C
X-111	Spare	13500	280	1350	300	O-ring	B
X-112	Spare	13500	180	-5250	300	O-ring	B
X-113	Spare	13500	180	1350	300	O-ring	B
X-130A	I & C	13500	45	0	300	O-ring	B
X-130B	I & C	13500	212	0	300	O-ring	B
X-130C	I & C	13500	124	0	300	O-ring	B
X-130D	I & C	13500	295	0	300	O-ring	B
X-140A	I & C	13500	45	-27000	300	O-ring	B

* This table provided in response to Questions 430.49d & e.

† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.

‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-240	Wetwell Purge Suction	9200	45	1200	550		A
X-241	Wetwell Purge Exhaust	9200	230	0	550		A
X-242	~~FCS-Return~~ Spare	1500	225	-1000	150	Welded Cap	C
X-250	Spare	8500	45	0	400		A
X-251	Spare Spare	9000	213	0	400		A
X-252	~~FCS-Return~~	1500	50	0	300	Welded Cap	C
X-253	Spare	2650	135	1000	300		B
X-254	Spare	2650	225	-1000	300		B
X-255	Spare	1200	282	0	300		B
X-300A	I & C	7300	134	0	300	O-ring	B
X-300B	I & C	7300	211	0	300	O-ring	B
X-320	I & C	8900	74	0	90	O-ring	B
X-321A	I & C	2050	97.5	0	300	O-ring	B
X-321B	I & C	6000	262.5	0	300	O-ring	B
X-322A	I & C	400	78	0	90	O-ring	B
X-322B	I & C	400	258	0	90	O-ring	B
X-322C	I & C	400	102	0	90	O-ring	B
X-322D	I & C	400	282	0	90	O-ring	B
X-322E	I & C	2000	94	0	90	O-ring	B
X-322F	I & C	2000	266	0	90	O-ring	B
X-323A	I & C	-5200	30	0	90	O-ring	B
X-323B	I & C	-5200	210	0	90	O-ring	B
X-323C	I & C	-5200	156	0	90	O-ring	B
X-323D	I & C	-5200	304	0	90	O-ring	B
X-323E	I & C	-7500	100	0	90	O-ring	B
X-323F	I & C	-7500	230	0	90	O-ring	B

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-65	HNCW Suction	350	E	E/D/H	No
X-66	HNCW Return	350	E	E/D/H	No
X-69	SA	90	E	E/D/H	No
X-70	IA	200	E	E/D/H	No
X-71A	ADS Accumulator (A)	200	S	C/K	No
X-71B	ADS Accumulator (B)	200	S	C/K	No
X-72	Relief Valve Accumulator	200	S	C/K	No
X-80	Drywell Purge Suction	550	E	E/C/J	Yes
X-81	Drywell Purge Exhaust	550	E	E/C/J	Yes
~~X-82~~	~~FCS Suction~~	~~150~~	S	E/~~C~~/H	No
X-90	Spare Spare	400	P	B/A	No
X-91	Spare	400	P	B/A	No
X-92	Spare	400	P	B/A	No
X-93	Spare	400	P	B/A	No
X-100A	IP Power	450	S	C/J	No
X-100B	IP Power	450	S	C/J	No
X-100C	IP Power	450	S	C/J	No
X-100D	IP Power	450	S	C/J	No
X-100E	IP Power	450	S	C/J	No
X-101A	LP Power	300	S	C/J	No
X-101B	LP Power	300	S	C/J	No
X-101C	LP Power	300	S	C/J	No
X-101D	FMCRD Power	300	S	C/J	No
X-101E	FMCRD Power	300	S	C/J	No
X-101F	FMCRD Power	300	S	C/J	No
X-101G	FMCRD Power	300	S	C/J	No
X-102A	I & C	300	S	C/J	No
X-102B	I & C	300	S	C/J	No
X-102C	I & C	300	S	C/J	No
X-102D	I & C	300	S	C/J	No
X-102E	I & C	300	S	C/J	No

Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-102F	I & C	300	S	C/J	No
X-102G	I & C	300	S	C/J	No
X-102H	FMCRD Control	300	S	C/J	No
X-102J	FMCRD Control	300	S	C/J	No
X-103A	I & C	300	S	C/J	No
X-103B	I & C	300	S	C/J	No
X-103C	I & C	300	S	C/J	No
X-104A	FMCRD Pos. Indicator	300	S	C/J	No
X-104B	FMCRD Pos. Indicator	300	S	C/J	No
X-104C	FMCRD Pos. Indicator	300	S	C/J	No
X-104D	FMCRD Pos. Indicator	300	S	C/J	No
X-104E	FMCRD Pos. Indicator	300	S	C/J	No
X-104F	FMCRD Pos. Indicator	300	S	C/J	No
X-104G	FMCRD Pos. Indicator	300	S	C/J	No
X-104H	FMCRD Pos. Indicator	300	S	C/J	No
X-105A	Neutron Detection	300	S	C/J	No
X-105B	Neutron Detection	300	S	C/J	No
X-105C	Neutron Indicator	300	S	C/J	No
X-105D	Neutron Indicator	300	S	C/J	No
~~X-110~~	~~FCS Suction~~	~~100~~	S	E/~~C~~/H	No
X-111	Spare Spare	300	P	B/A	No
X-112	Spare	300	P	B/A	No
X-113	Spare	300	P	B/A	No
X-130A	I & C	300	S	C/J	No
X-130B	I & C	300	S	C/J	No
X-130C	I & C	300	S	C/J	No
X-130D	I & C	300	S	C/J	No
X-140A	I & C	300	S	C/J	No
X-140B	I & C	300	S	C/J	No
X-141A	I & C	300	S	C/J	No
X-141B	I & C	300	S	C/J	No

Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-201	RHR Pump Suction (A)	450	S	C/H	No
X-202	RHR Pump Suction (B)	450	S	C/H	No
X-203	RHR Pump Suction (C)	450	S	C/H	No
X-204	RHR Pump Test (A)	250	S	C/H	No
X-205	RHR Pump Test (B)	250	S	C/H	No
X-206	RHR Pump Test (C)	250	S	C/H	No
X-210	HPCF Pump Suction (B)	400	S	C/H	No
X-211	HPCF Pump Suction (C)	400	S	C/H	No
X-213	RCIC Turbine Exhaust	550	S	C/G	No
X-214	RCIC Pump Suction	200	S	C/H	No
X-215	RCIC Vacuum Pump Ex.	250	S	C/G	No
X-216	SPCU Pump Suction	200	S	C/H	No
X-217	SPCU Pump Return	250	S	C/H	No
X-220	MSIV Leakage	250	S	C/G	No
X-240	Wetwell Purge Suction	550	E	E/C/J	Yes
X-241	Wetwell Purge Exhaust	550	E	E/C/J	Yes
~~X-242~~	~~FCS Return~~ Spare	~~150~~	~~S~~	~~E/C/H~~	~~No~~
X-250	Spare		P	B/A	No
X-251	Spare		P	B/A	No
~~X-252~~	~~FCS Return~~ Spare	~~150~~	~~S~~	~~E/C/H~~	~~No~~
X-253	Spare	300	S	B/A	No
X-254	Spare	300	S	B/A	No
X-255	Spare	300	S	B/A	No
X-300A	I&C	300	S	C/J	No
X-300B	I&C	300	S	C/J	No
X-320	I&C	90	S	C/J	No
X-321B	I&C	300	S	C/J	No
X-322A	I&C	90	S	C/J	No
X-322B	I&C	90	S	C/J	No

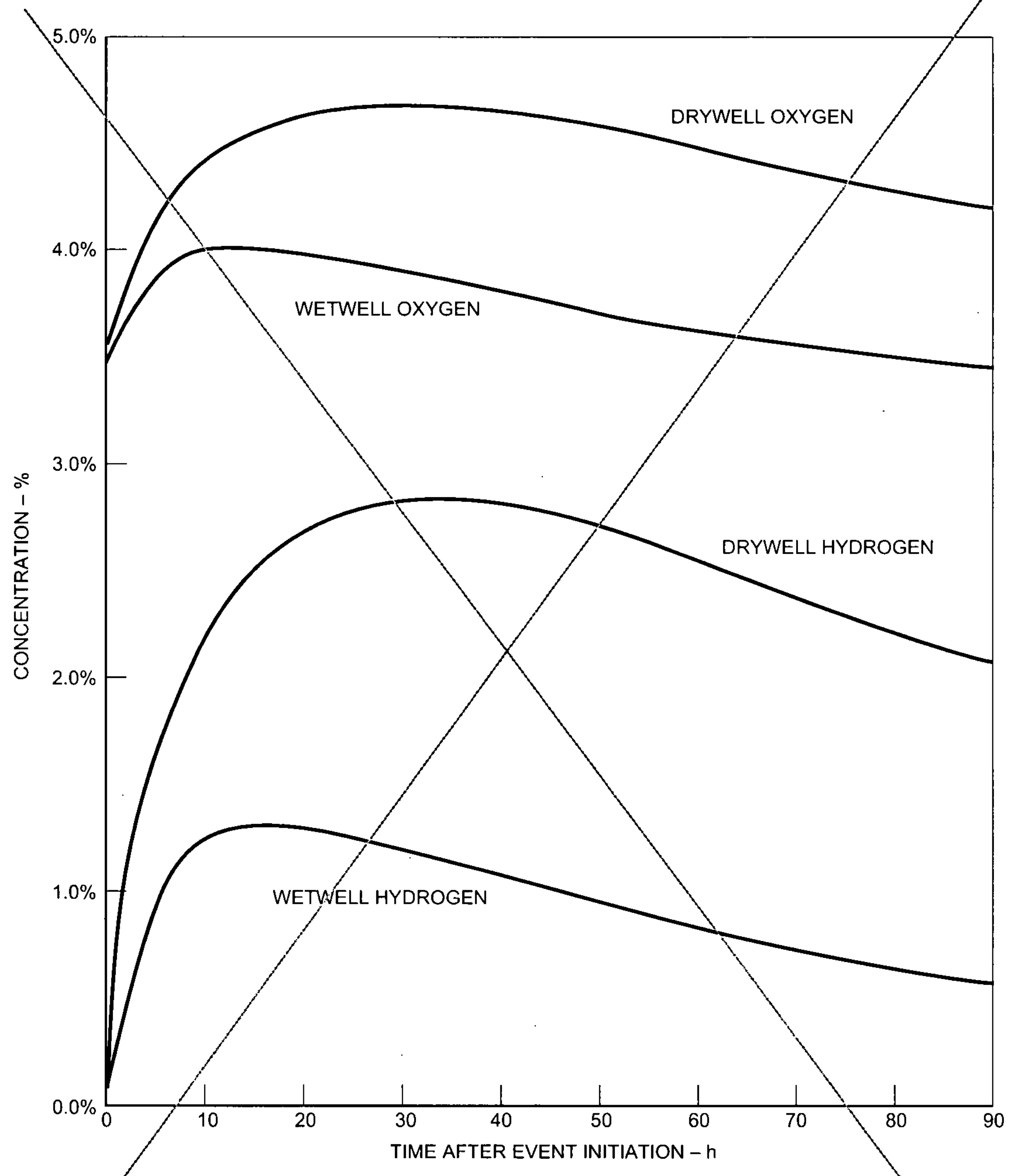


Figure 6.2-41 Hydrogen and Oxygen Concentrations in Containment After Design Basis LOCA

Table 6.6-1 Examination Categories and Methods (Continued)

Quality Group	System Number	System Title	System Description	P&ID Diagram	Sec. XI Exam Cat.	Items Examined	Exam Method
C	T49	Flammability Control	Piping from valves F006A & B up to and including the recombiner skids A & B	Figure 6.2-40			
			All pressure retaining components and piping		D-B	External Surfaces (Note 7)	VT-2
			Integral attachments		D-B	Welds (Note 8)	VT-3
			Piping and Component Supports		F-A	Supports (Note 6)	VT-3
			All Class C piping 20A, 25A, 50A, 80A and 100A in diameter, i.e.: - drain lines - test connections - SRV discharge line - instrument lines - small process lines - and etc.	Figure 6.2-40	Exempted per IWD-1220		
			All pressure retaining components and piping		D-B	External Surfaces (Note 7)	VT-2
			Integral attachments		D-B	Welds (Note 15)	VT-3
			Piping and Component Supports		F-A	Supports (Note 6)	VT-3

7.1.2.6.6 Containment Atmospheric Monitoring (CAM) Systems

(1) Safety Design Bases

General Functional Requirements:

Monitor ~~the atmosphere in the~~ inerted primary containment ~~for radiation~~ levels ~~and for concentration of~~ hydrogen and oxygen gases~~, primarily during post accident conditions.~~ Monitoring shall be ~~provided by two independent safety related divisional~~ subsystems~~.~~

/lonitor the
ıtmosphere in the
ıerted primary
:ontainment for
adiation levels and
or concentration of
ıydrogen and
ıxygen gases,
ırimarily during
ıost-accident
:onditions.

;ample and monitor
ıe oxygen and
ıydrogen
:oncentration levels
ı the drywell and
ıuppression
:hamber under
ıost-accident
onditions, and also
/hen required
luring reactor
ıperation. The
.OCA signal (low
eactor water level
ır high drywell
ıressure) shall
ıctivate the system
ınd place it into
ervice to monitor
ıe gaseous buildup
ı the primary
:ontainment
ollowing an
ıccident.

Monitor continuously the radiation environment in the drywell and suppression chamber during reactor operation and under post-accident conditions.

~~Sample and monitor~~ the oxygen and ~~hydrogen concentration levels~~ in the drywell ~~and suppression chamber~~ under post accident ~~conditions, and also when required~~ during ~~reactor operation. The LOCA~~ signal (low ~~reactor water level or high~~ drywell pressure) shall ~~activate the system and place it~~ into service ~~to monitor~~ the gaseous buildup in ~~the primary containment~~ following an ~~accident.~~

Specific Regulatory Requirements:

Specific regulatory requirements applicable to this system are listed in Table 7.1-2.

(2) Non-Safety-Related Design Bases

Separate hydrogen and oxygen gas calibration sources shall be provided for each CAM Subsystem for periodic calibration of the gas analyzers and monitors.

Suppression Pool Temperature Monitoring System—Instrumentation and Control

(1) Safety Design Bases

General Functional Requirements:

The SPTM is a Class 1E safety-related system. The general functional requirements shall be to automatically initiate suppression pool cooling or scram the reactor when high suppression pool temperatures are detected that might be caused by safety relief valve leakage or malfunction.

Specific Regulatory Requirements:

temperature variable is considered a Type A variable since no credit is taken for automatic initiation in the safety analysis.

(j) Drywell Atmosphere Temperature

Surveillance monitoring of the temperatures in the drywell is provided by multiple temperature sensors distributed throughout the drywell to detect local area "hot-spots" and to monitor the operability of the drywell cooling system. With this drywell air temperature monitoring system supplied by multiple temperature sensors throughout the drywell, the Regulatory Guide 1.97 requirements for monitoring of drywell air temperature are met and provides the ability to determine drywell bulk average temperature.

non-safety related

(k) Drywell/Wetwell Hydrogen/Oxygen Concentration

The Containment Atmospheric Monitoring System (CAMS) consists of two independent and redundant drywell/containment oxygen and hydrogen concentration monitoring channels. Emergency response actions regarding these variables are consistently directed toward minimizing the magnitude of these parameters (i.e., there are no safety actions which must be taken to increase the hydrogen/oxygen levels if they are low). Consequently, the two channel CAMS design provides adequate PAM indication, since, in the event that the two channels of information disagree, the operator can determine a correct and safe action based upon the higher of the two (in-range) indications.

(l) Wetwell Atmosphere Air Temperature

Surveillance monitoring of temperatures in the wetwell is provided by multiple temperature sensors dispersed throughout the wetwell, therefore, the required indication of bulk average wetwell atmosphere temperature is satisfied.

(m) Standby Liquid Control System Flow

No flow indication is provided for the ABWR design. The positive displacement SLCS pumps are designed for constant flow. Any flow blockage or line break would be indicated by abnormal system pressure (high or low as compared to RCS pressure) following SLCS initiation. Changing neutron flux, SLCS pressure and SLCS tank level are substituted for SLCS flow and are considered adequate to verify proper system function. One channel of SLCS discharge pressure is provided in addition to the monitoring of neutron flux.

(n) Suppression Pool/Wetwell Water Level

The status of each valve providing the HP/LP boundary is indicated in the control room. The state of the sensors is also indicated in the control room.

(14) Setpoints

See Chapter 16 for setpoints and margin.

7.6.1.4 Not Used

7.6.1.5 Wetwell-to-Drywell Vacuum Breaker System—Instrumentation and Controls

This system is described in Chapter 6.

7.6.1.6 Containment Atmospheric Monitoring (CAM) System—Instrumentation and Controls

, and a non-safety related H2/O2 monitoring system

(1) System Identification

of radiation channels

The CAM System (Figures 7.6-7 and 7.6-8) consists of two independent but redundant Class 1E divisions (I and II), which are electrically and physically separated. Each CAM ~~division~~ has the capability of monitoring the total gamma-ray dose rate ~~and~~ concentration of hydrogen and oxygen (H_2/O_2) in the drywell and~~/or~~ the suppression chamber during plant operation, and following a LOCA event.

divisional radiation channel

. CAMS also has the ability of monitoring

There are two radiation monitoring channels per division; one for monitoring the radiation level in the drywell and the other for monitoring the radiation level in the suppression chamber. Each monitoring channel consists of ~~an ion chamber~~ detector~~,~~ a digital log radiation monitor~~, and a recorder~~. Each radiation monitoring channel provides alarm indication in the control room on high radiation levels and also if the channel becomes inoperative.

and

The H2/O2 monitoring system is located within a H2/O2 analyzer panel. Each H2/O2 analyzer panel consists

~~Each divisional H_2/O_2 monitoring channel consists~~ of valves, pumps, and pipes used to extract samples of the atmosphere in the drywell or the suppression chamber and feed the extracted air sample into an analyzer and monitor for measurement, ~~recording,~~ and for alarm indication on high concentration of gas levels.

The H2/O2 monitoring system is non-safety.

The piping used for the gas extraction is made of stainless steel and utilizes heat tracing to keep the pipes dry and free of moisture condensation.

within the panels

(2) Power Sources

radiation channel

Each CAM ~~Subsystem~~ is powered from divisional 120 VAC instrument bus. ~~The same Class 1E divisional 120 VAC power source also supplies the heat tracing blanket used for the sampling lines.~~

(3) Initiating Circuits

The

Each divisional gamma radiation monitoring channel can be energized manually by the operator ~~or automatically by the LOCA signal~~. ~~For the manual mode, the~~ gamma radiation monitor is on continuously during plant operation and remains on until power is turned off by the operator.

~~In the power-off mode, the channel will be activated automatically in the presence of a LOCA (high drywell pressure or low reactor water level).~~

located within the panel and is

Each ~~divisional~~ H_2/O_2 monitoring subsystem (except for the two sampling pumps) is powered continuously during plant operation. One pump is controlled by an operator and is used during reactor operation and the other is turned on by the LOCA signal to allow measurement during an accident.

The heat tracing used in each H_2/O_2 sample line is temperature controlled to prevent moisture condensation in the pipes.

Each ~~divisional~~ H_2/O_2 analyzer and monitor can selectively measure the atmosphere in the drywell or the suppression chamber.

~~Division I and II~~ LOCA signals are provided to the CAM System from the RHR System. These signals are based on two-out-of-four logic signals for the high drywell pressure or low reactor water level.

(4) Redundancy and Diversity

radiation channels

The CAM Subsystems, Divisions I and II, are independent and are redundant to each other.

(5) Divisional Separation

radiation monitoring channel divisions

The two CAM ~~Subsystems~~ are electrically and physically separated so that no single design basis event is capable of damaging equipment in more than one CAM division. No single failure or test, calibration, or maintenance operation can prevent function of more than one division.

(6) Testability and Calibration

Each CAM Subsystem can be tested separately during plant operation to determine the operational availability of the system. Each CAM Subsystem can be tested and calibrated separately.

Gas calibration sources are provided to check the hydrogen/oxygen sensors during normal plant operation and after an accident.

(7) Environmental Consideration

The CAM System is qualified Seismic Category I and is designed for operability during normal and post-accident environments.

(8) Operational Considerations

The following information is available to the reactor ~~operator:~~ a detector, and

(a) Each gamma radiation channel consists of ~~an ion chamber,~~ a log radiation ~~monitor,~~ ~~and a recorder~~. Each channel has a range of 0.01 ~~Gy/h~~ to 10^5 ~~Gy/h~~. Each channel will initiate an alarm on high radiation level or ~~on an~~ inoperative channel. panel contains

mSv/h

(b) Each hydrogen/oxygen monitoring ~~channel uses~~ a sampling rack for extracting the atmosphere from the drywell or the suppression chamber and for analyzing the contents for both H_2/O_2 concentration. The gaseous measurements are made by volume on a wet basis after humidity correction (dry basis before humidity correction). Separate monitors are provided for oxygen and hydrogen indications. panel has

Each H_2/O_2 analyzer ~~rack has a series of~~ alarms to indicate a high concentration of hydrogen and of oxygen, and to alert the operator of any abnormal system parameter. Refer to Figure 7.6-8 for definition of these alarms.

(9) Control and Protective Functions

The CAM System does not provide control signals either to trip or to actuate other safety-related systems. However, the CAM System utilizes internal safeguards to affect system operation, alert the operator of abnormal performance, and protect equipment from damage.

7.6.1.7 Suppression Pool Temperature Monitoring System—Instrumentation and Controls

7.6.1.7.1 System Identification

The Suppression Pool Temperature Monitoring (SPTM) System is provided to monitor suppression pool temperature. Monitoring of suppression pool temperature is provided so that trends in suppression pool temperature may be established in sufficient time for proper cooling of the suppression pool water and for reactor scram due to high suppression pool temperature and for reactor power control based upon symptom-based emergency operating procedures.

The SPTM System also provides information on the post-LOCA condition of the suppression pool.

The following figures are located in Chapter 21:

Figure 7.6-5 Process Radiation Monitoring System IED (Sheets 1-11)

Figure 7.6-6 Not Used

Figure 7.6-7 Containment Atmospheric Monitoring System IED (Sheets 1-4)

Figure 7.6-8 Containment Atmospheric Monitoring System IBD (Sheets 1-10)

This figure is incorporated by reference as clarified below:

The Hydrogen and Oxygen Monitoring equipment of the CAMS system are located in the Hydrogen/Oxygen Analysis Panels. The IBDs provide only a general overview of functionality. However, this logic is internal to the panel and subject to design of vendor.

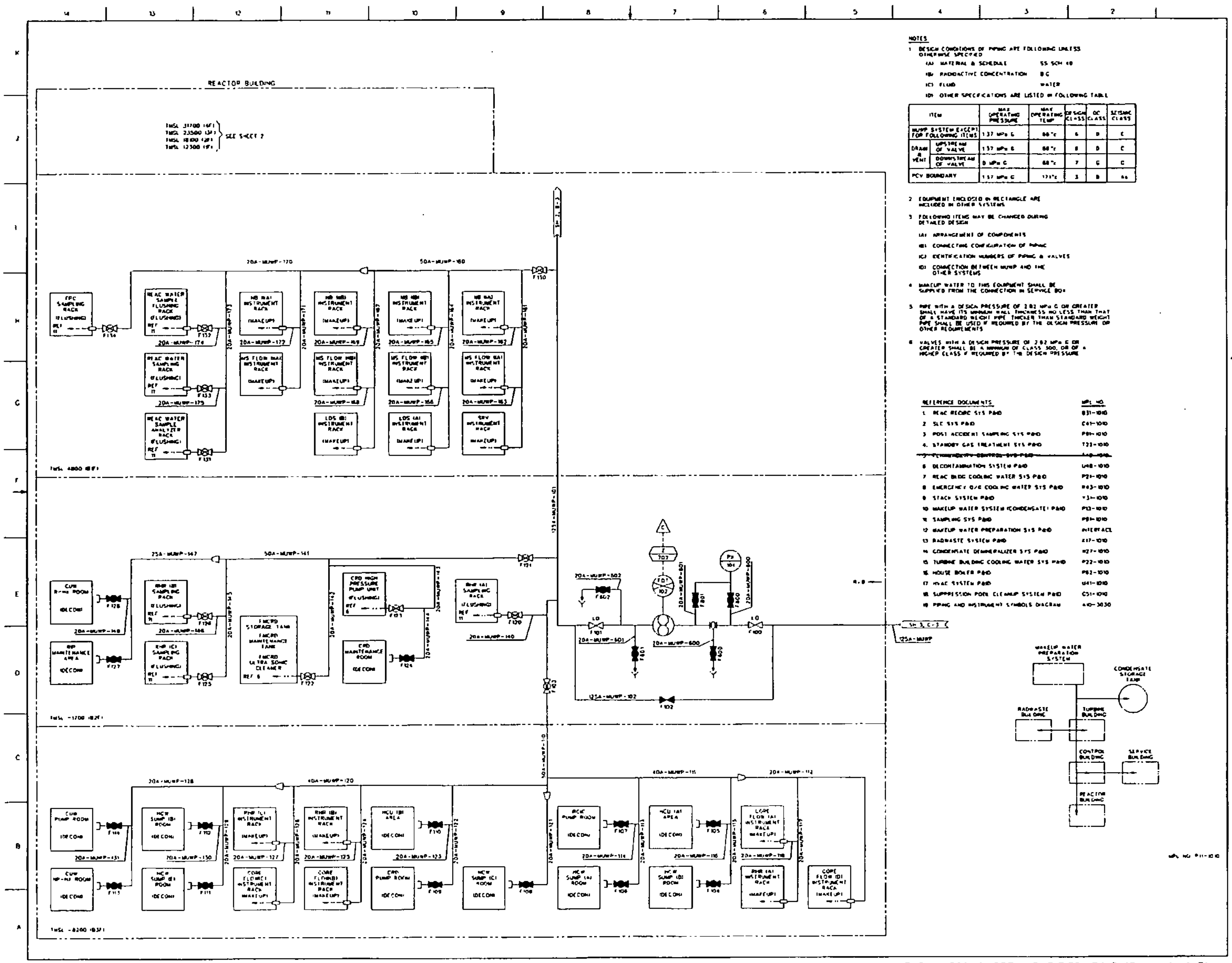


FIGURE 9.2-5 MAKEUP WATER SYSTEM (PURIFIED) P&ID (Sheet 1 of 3)

ABWR DCD/Tier 2 Rev. 0 21-523

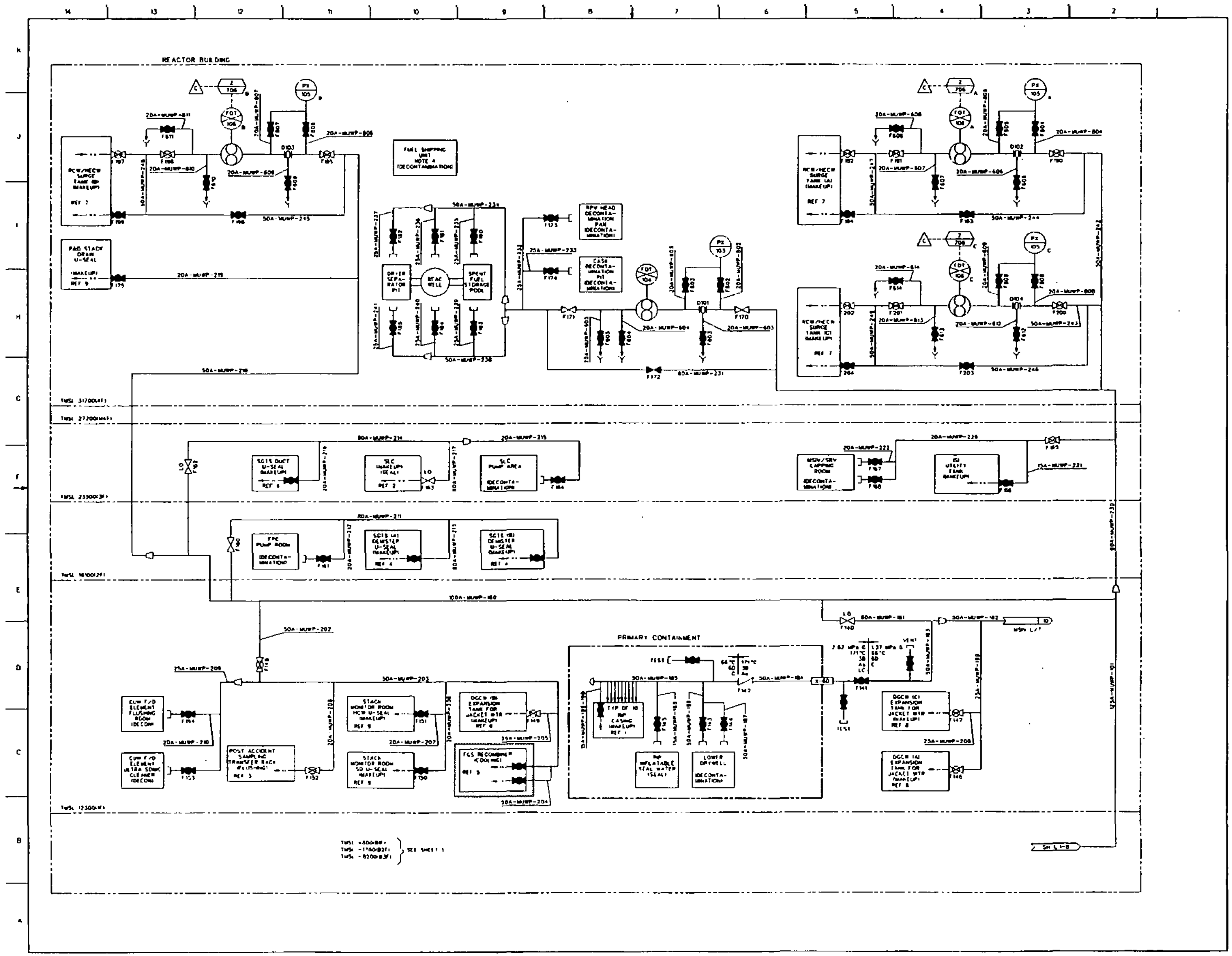


FIGURE 9.2-5 MAKEUP WATER SYSTEM (PURIFIED) P&ID (Sheet 2 of 3)

ABWR DCD/Tier 2 Rev 0 21-524

9.4.5.2 R/B Safety-Related Equipment HVAC System

9.4.5.2.1 Design Bases

9.4.5.2.1.1 Safety Design Bases

The R/B Safety-Related Equipment HVAC System is designed to provide a controlled temperature environment to ensure the continued operation of safety-related equipment in harsh environment under accident conditions. The rooms cooled by the Safety-Related Equipment HVAC System are maintained at negative pressure relative to atmosphere by the secondary containment HVAC System during the normal operating mode, and by standby gas treatment system in isolation mode.

The systems and components are Seismic Category I and are located in the Reactor Building, separate and independent compartments of a Seismic Category I structure that is tornado-missile, and flood protected.

Fire protection has been evaluated and is described in Subsection 9.5.1.

9.4.5.2.1.2 Power Generation Design Bases

The system is designed to provide an environment with controlled temperature and humidity to ensure both the comfort and safety of plant personnel and the integrity of Reactor Building equipment. The systems are designed to facilitate periodic inspection of the principal system components.

9.4.5.2.2 System Description

The R/B Safety-Related Equipment HVAC System consists of 12 safety-related fan coil units (FCU) of division A, B, or C. Each FCU has the responsibility to cool one safety-related equipment room in the secondary containment. The safety-related equipment HVAC (fan coil units) system P&ID is shown in Figure 9.4-3. Space temperatures are maintained less than 40°C normally and less than 66°C during pump operation:

(1) RHR(A) pump room

(2) RHR(B) pump room

(3) RHR(C) pump room

(4) HPCF(B) pump room

(5) HPCF(C) pump room

(6) RCIC pump room

(7) Room 425

(8) Room 436

(9) SGTS(B) room

(10) SGTS(C) room

(11) CAMS(A) room

(12) CAMS(B) room

9.4.5.2.2.1 RHR, HPCF and RCIC Pump Room HVAC Systems

The FCU's automatically start when RHR pumps, HPCF pumps, and RCIC turbine are started. These rooms are normally cooled by the Secondary Containment HVAC System. The fan coil units are open ended and recirculate cooling air within the space served. Space heat is removed by cooling water passing through the coil section. Divisional Reactor Building Cooling Water (RCW) is used as the cooling medium. The units are fed from the same divisional power as that for the equipment being served. Drain pan discharge (condensate) is routed to a floor drain located within the room.

9.4.5.2.2.2 Rm. 425 / 436 HVAC System

Cooling of Rm. 425 / 436 is automatically initiated upon receipt of a secondary containment isolation signal.

Rm. 425 / 436

These rooms are cooled by the Secondary Containment HVAC ~~System during norm~~al conditions. The units are open ended and recirculate cooling air within the space served. Space heat is removed by cooling water passing through the coil section. Divisional RCW is used as the cooling medium. The units are fed from the same divisional power as that for the ~~FCS~~ being served. Humidity is not specifically maintained at a set range, but is automatically determined by the surface temperature of the cooling coil. Drain pan discharge (condensate) is routed to a floor drain located within the room.

9.4.5.2.2.3 SGTS and CAMS HVAC Systems

Cooling of the SGTS and CAMS rooms are automatically initiated upon receipt of a secondary containment isolation signal.

These rooms are cooled by the Secondary Containment HVAC System during normal conditions. The units are open ended and recirculate cooling air within the space served. Space heat is removed by cooling water passing through the coil section. Divisional RCW is used as the cooling medium. The units are fed from the same divisional power as that for the equipment being served. Drain pan discharge (condensate) is routed to a floor drain located within the room.

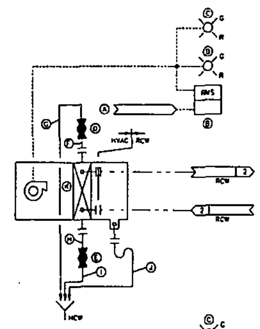
Table 9.4-4d Not Used

Table 9.4-4e HVAC System Component Descriptions — Safety-Related Fan Coil Units (Response to Question 430.243)

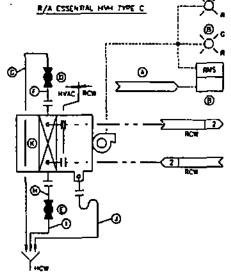
Safety-Related Fan Coil Units	Capacity (MJ/h)
HPCF Pump Room Div B	460.55
HPCF Pump Room Div C	460.55
RHR Pump Room Div A	307.73
RHR Pump Room Div B	307.73
RHR Pump Room Div C	307.73
RCIC Pump Room Div A	69.08
Rm. 425 Div B	54.85
Rm. 436 Div C	54.85
CAMS Room Div A	83.74
CAMS Room Div B	83.74
SGTS Room Div B	16.75
SGTS Room Div C	16.75

14 | 13 | 12 | 11 | 10 | 9 | 8 | 7 | 6 | 5 | 4 | 3 | 2

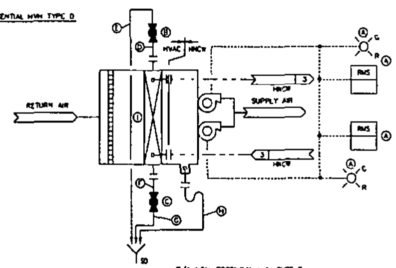
K J I H G F E D C B A



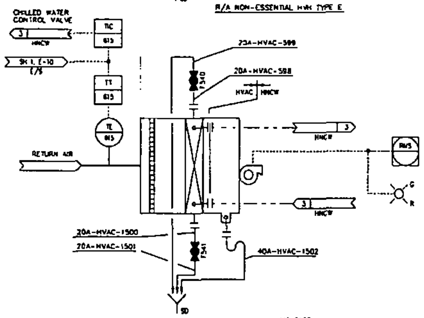
R/A ESSENTIAL HVH TYPE C



R/A ESSENTIAL HVH TYPE D



R/A NON-ESSENTIAL HVH TYPE E



U41-0123
REFUELING MACHINE CONTROL ROOM HVH

TABLE 3: ESSENTIAL HVH

TYPE	EQ. NO.	HVH NAME	HVH INITIATION SIGNAL Ⓐ	PANEL NO. Ⓑ	PANEL NO. Ⓒ	VALVE NO. Ⓓ	VALVE NO. Ⓔ	PIPING SIZE & NO. Ⓕ	PIPING SIZE & NO. Ⓖ	PIPING SIZE & NO. Ⓗ	PIPING SIZE & NO. Ⓘ	PIPING SIZE & NO. Ⓙ	COOLING COIL Ⓚ
D	U41-0101	RCIC PUMP ROOM HVH	RCIC PUMP START SIGNAL			F504	F505	20A-HVAC -508	20-HVAC -509	20-HVAC -510	20-HVAC -511	40-HVAC -512	B101
D	U41-0102	HPCF PUMP (C) ROOM HVH	HPCF PUMP (C) START SIGNAL			F506	F507	-513	-514	-515	-516	-517	B102
D	U41-0103	RHR PUMP (A) ROOM HVH	RHR PUMP (A) START SIGNAL			F508	F509	-518	-519	-520	-521	-522	B103
D	U41-0104	RHR PUMP (C) ROOM HVH	RHR PUMP (C) START SIGNAL			F510	F511	-523	-524	-525	-526	-527	B104
D	U41-0105	RHR PUMP (B) ROOM HVH	RHR PUMP (B) START SIGNAL			F512	F513	-528	-529	-530	-531	-532	B105
D	U41-0106	HPCF PUMP (B) ROOM HVH	HPCF PUMP (B) START SIGNAL			F514	F515	-533	-534	-535	-536	-537	B106
D	U41-0107	Rm 436	SECONDARY CONTAINMENT ISOLATION SIGNAL			F516	F517	-538	-539	-540	-541	-542	B107
D	U41-0108	Rm 425	SECONDARY CONTAINMENT ISOLATION SIGNAL			F518	F519	-543	-544	-545	-546	-547	B108
D	U41-0111	SGTS ROOM HVH (C)	SECONDARY CONTAINMENT ISOLATION SIGNAL			F524	F525	-558	-559	-560	-561	-562	B111
D	U41-0112	SGTS ROOM HVH (B)	SECONDARY CONTAINMENT ISOLATION SIGNAL			F526	F527	-563	-564	-565	-566	-567	B112
C	U41-0113	CAMS (A) ROOM HVH	SECONDARY CONTAINMENT ISOLATION SIGNAL			F528	F529	-568	-569	-570	-571	-572	B113
C	U41-0114	CAMS (B) ROOM HVH	SECONDARY CONTAINMENT ISOLATION SIGNAL			F530	F531	-573	-574	-575	-576	-577	B114

TABLE 4: NON-ESSENTIAL AIR HANDLING UNIT (NON-SAFETY RELATED)

TYPE	EQ. NO	HVH NAME	PANEL NO. Ⓐ	VALVE NO Ⓑ	VALVE NO. Ⓒ	PIPING SIZE & NO. Ⓓ	PIPING SIZE & NO. Ⓔ	PIPING SIZE & NO Ⓕ	PIPING SIZE & NO. Ⓖ	PIPING SIZE & NO. Ⓗ	COOLING COIL Ⓘ
E	U41-0121A	R/A MS TUNNEL HVH (A) NOTE 3		F536	F537	20A-HVAC -588	20A-HVAC -589	20A-HVAC -590	20A-HVAC -591	40A-HVAC -592	B121A
E	U41-0121B	R/A MS TUNNEL HVH (B) NOTE 3		F538	F539	20A-HVAC -593	20A-HVAC -594	20A-HVAC -595	20A-HVAC -596	40A-HVAC -597	B121B
E	U41-0115	R/A SPCU PUMP ROOM		F	F	20A-HVAC -1503	20A-HVAC -1504	20A-HVAC -1505	20A-HVAC -1506	40A-HVAC -1507	B115
-	U41-0123	R/A REFUELING MACHINE CONTROL ROOM HVH		F	F	20A-HVAC -598	20A-HVAC -599	20A-HVAC -1500	20A-HVAC -1501	40A-HVAC -1502	B123
D	U41-0109	FPC PUMP (A) ROOM HVH		F520	F521	-548	-549	-550	-551	-552	B109
D	U41-0110	FPC PUMP (B) ROOM HVH		F522	F523	-553	-554	-555	-556	-557	B110

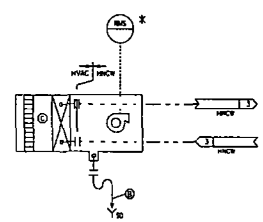
TABLE 5: NON-ESSENTIAL AIR HANDLING UNIT (NON-SAFETY RELATED)

TYPE	EQ. NO.	HVH NAME	PIPE SIZE & NO. Ⓑ	COOLING COIL Ⓒ
MV504	U41-0131A	CRD CONTROL ROOM	20A-HVAC -1514	B131A
MV504	U41-0132A	MG SET ROOM (A) NOTE 4	20A-HVAC -1517	B132A
MV504	U41-0132B	MG SET ROOM (C) NOTE 4	20A-HVAC -1518	B132B
MV504	U41-0133A	NON-DIVISIONAL ELECTRICAL EQUIPMENT ZONE NOTE 4	20A-HVAC -1519	B133
MV504	U41-0134A	ISI ROOM	20A-HVAC -1512	B134A

NOTES

1. PIPING MATERIALS FOR VENT AND DRAIN SHALL BE CS.
2. PIPING DESIGN/QUALITY CLASSIFICATION SHALL BE 7G EXCEPT OTHERWISE SPECIFIED.
3. THESE HVH'S CAN BE COMBINED INTO A COMMON COIL.
4. THESE UNITS ARE IN CONTROL BUILDING

REFERENCE DOCUMENTS	MPL NO.
1. PIPING AND INSTRUMENT SYMBOLS DIAGRAM	A10-3030
2. REACTOR BLDG COOLING WATER SYS P&ID	P21-1010
3. HVAC NORMAL COOLING WATER SYS P&ID	P24-1010



R/A NON-ESSENTIAL UNIT TYPE MV504

R/B AND C/B HVAC

U41-1010

107E5169\8065802

FIGURE 9.4-3 SECONDARY CONTAINMENT HVAC SYSTEM (Sheet 2 of 3)

Table of Contents (Continued)

(9) Consequences of Fire—The postulated fire assumes the loss of the function. The provisions for core cooling systems backup are defined in Subsection 9A.2.5.

Smoke from a fire will be removed by the normal HVAC System operating in its smoke removal mode.

(10) Consequences of Fire Suppression—Suppression extinguishes the fire. Refer to Section 3.4, "Water Level (Flood) Design", for the drain system.

(11) Design Criteria Used for Protection Against Inadvertent Operation, Careless Operation or Rupture of the Suppression System:

(a) Location of the manual suppression system external to the room

(b) Provision of raised supports for the equipment

(c) Refer to Section 3.4, "Water Level (Flood) Design", for the drain system.

(d) ANSI B31.1 standpipe (rupture unlikely)

(12) Fire Containment or Inhibiting Methods Employed:

(a) The functions are located in a separate fire-resistive enclosure.

(b) The means of fire detection, suppression and alarming are provided and accessible.

(13) Remarks—The room contains cable in conduit only.

9A.4.1.4.8 Corridor C (Equipment Entry) (Rm No. 430)

(1) Fire Area—F4301

(2) Equipment: See Table 9A.6-2

Safety-Related	Provides Core Cooling
Yes, D3	Yes, D3

(3) Radioactive Material Present—None that can be released as a result of fire.

(4) Qualifications of Fire Barriers—The walls common with the C diesel generator room (Rm 432), valve room (C) (Rm 431), corridor B (Rm 420), ~~the Flammability Control System room~~ (Rm 436) and the exterior wall serve as fire barriers and are of 3 h fire-resistive concrete construction. The floor is also a fire barrier to limit the size of the fire areas below and to protect the

lower regions of the building, which contains the majority of the ESF equipment. The walls are concrete and are not rated as they are internal to fire area F4301. A section of the ceiling common to fire areas F4300, F1300 and F3300 above is of 3 h fire-resistive concrete construction. The remainder of the ceiling is not fire rated as it is internal to fire area F4310. Access to the corridor is provided from corridors A and B via 3 h fire-resistive doors. The corridor provides direct access to the electrical and instrumentation penetration room (Rm 433) through a nonrated door and valve room
(C) (Rm 431) and ~~the Flammability Control System~~ room (Rm 436) through 3 h fire-resistive doors. There is an open hatch to the floors above. A large steel non-fire-rated door provides access to the reactor building for moving in fuel and other large loads.

(5) Combustibles Present:

Fire Loading	Total Heat of Combustion (MJ)
Cable Tray	727 MJ/m^2 NCLL (727 MJ/m^2 maximum average) applies
Lubricant Fuel Oil	Could be a variable due to possible lubricant, and fuel oil leaks in transient. Deluge sprinkler system provided.

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull stations at 5.9-F.2 and 2.1-F.1.

(7) Suppression Available:

Type	Location/Actuation
Ordinary hazard deluge sprinkler having a water density of 6.1 $L/min/m^2$ and a coverage of 9.3 m^2 per head	Hatch Area/Manual
Standpipe and hose reel	Col. 5.9-F.2 & 2.1-F.1/Manual
ABC hand extinguishers	Col. 5.9-F.2 & 2.1-F.1/Manual

(8) Fire Protection Design Criteria Employed:

(b) The means of fire detection, suppression and alarming are provided and accessible.

(13) Remarks—Although the areas surrounding the adjacent diesel generator room are of the same safety division, the diesel generator room is designated as a separate fire area due to the relatively large amounts of lubricating and fuel oil present.

9A.4.1.4.11 Room 436

(1) Fire Area—F4320

(2) Equipment: See Table 9A.6-2

Safety-Related	Provides Core Cooling
Yes, D1, and D2	No

(3) Radioactive Material Present—None that can be released as a result of fire.

(4) Qualifications of Fire Barriers—The floor and interior and exterior walls are fire barriers and are of 3 h fire-resistive concrete construction. The ceiling is formed by the bottom of the spent fuel storage pool (F4301) and is a 3 h fire barrier. Personnel access is provided via a 3 h fire-resistive door from corridor C (Rm 430).

(5) Combustibles Present:

Fire Loading	Total Heat of Combustion (MJ)
Cable Tray	727 MJ/m^2 NCLL (727 MJ/m^2 maximum average) applies

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull station at Col. 5.9-F.2 and 2.1-F.1.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel	Col. 5.9-F.2 & 2.1-F.1/Manual
ABC hand extinguishers	Col. 5.9-F.2 & 2.1-F.1/Manual

(4) Qualifications of Fire Barriers—The walls common with ~~the Flammability Control System Room~~ (Rm 425), the elevator and stair well walls, the Diesel Generator B Room (Rm 423) and the ECCS Valve B Room (Rm 421) serve as fire barriers and are of 3 h fire-resistive concrete construction. The floor is also a fire barrier to limit the size of the fire areas below and to protect the lower regions of the building, which contains the majority of the ESF equipment. The walls common with the E and I Penetration Room (Rm 422) and the ceiling are fire-resistive concrete but are nonrated as they are internal to fire area F4201. Access to the corridor is provided from corridor D (Rm 445), corridor C (Rm 430) and stairs and elevator No.3. A 3 h fire damper is installed in the HVAC duct (located next to the elevator) where it passes through the fire barrier floor to the division 2 areas on the level below. This fire barrier divides the division 2 area of the building to limit the magnitude of possible damage due to a single fire.

(5) Combustibles Present:

Fire Loading	Total Heat of Combustion (MJ)
Cable Tray	727 MJ/m^2 NCLL (727 MJ/m^2 maximum average) applies

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull stations at 5.9-F.2 and 2.1-F.1.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel	Col. 5.9-F.2 & 2.1-F.1/Manual
ABC hand extinguishers	Col. 5.9-F.2 & 2.1-F.1/Manual

(8) Fire Protection Design Criteria Employed:

(a) The function is located in a separate fire resistive enclosure.

(b) Fire detection and suppression capability is provided and accessible.

(c) Fire stops are provided for cable tray and piping penetrations through rated fire barriers.

(9) Consequences of Fire—The postulated fire assumes the loss of the function. The provisions for core cooling systems backup are defined in Subsection 9A.2.5. Access is provided to the corridor from either end.

(b) The means of fire detection, suppression and alarming are provided and accessible.

(13) Remarks—None.

9A.4.1.4.14 Not Used

9A.4.1.4.15 Diesel Generator B Room (Rm No. 423)

(1) Fire Area—F4200

(2) Equipment: See Table 9A.6-2

Safety-Related	Provides Core Cooling
Yes, D2	Yes, D2

(3) Radioactive Material Present—None.

(4) Qualifications of Fire Barriers—The building exterior walls, the walls common with Corridor B (Rm 420), the wall common with ~~FCS room~~ (Rm 425), the wall common with stair wells (Rms 193 and 329), and the floor are of 3 h fire resistive concrete construction. The interior partition walls, and ceiling are not fire rated as they are internal to fire F4200. The ceiling of the room is not a fire barrier as the fan room is located directly above this diesel generator room. The exterior wall of the room has a removable section for removal of equipment from the diesel generator room. Access to this room is provided from the Clean Area Access C/D (Rm 426) through a 3 h fire-rated door and through the removable section of the external wall.

(5) Combustibles Present:

Fire Loading	Total Heat of Combustion (MJ)
Cable Tray Lubricating Oil Fuel Oil	Could be variable due to possible oil leaks. Foam sprinkler system provided.

(6) Detection Provided—Class A supervised rate-compensated thermal detectors and infrared detectors. The detection system is a cross-zoned system requiring two detectors, one of each in each zone. Each detector initiates a local alarm upon sensing fire. The second detector alarm provides fire confirmation, which opens the preaction valve and initiates the system alarm in the control room. There is a manual pull stations at Col. 1.4-C.8.

(a) The function is located in a separate fire resistive enclosure.

(b) Fire detection and suppression capability is provided and accessible.

(c) Fire stops are provided for cable tray and piping penetrations through rated fire barriers.

(9) Consequences of Fire—The postulated fire assumes the loss of the function. The valves are spatially separated and are designed to fail closed on loss of actuation power. The provisions for core cooling systems backup are discussed in Subsection 9A.2.5.

Smoke from a fire will be removed by the normal HVAC System operating in its smoke removal mode.

(10) Consequences of Fire Suppression—Suppression extinguishes the fire. Refer to Section 3.4, "Water Level (Flood) Design", for the drain system.

(11) Design Criteria Used for Protection Against Inadvertent Operation, Careless Operation or Rupture of the Suppression System.

(a) Location of the manual suppression system in rooms external to the rooms containing safety-related equipment

(b) Provision of raised supports for the equipment

(c) Refer to Section 3.4, "Water Level (Flood) Design", for the drain system.

(d) ANSI B31.1 standpipe (rupture unlikely)

(12) Fire Containment or Inhibiting Methods Employed:

(a) The functions are located in a separate fire-resistive enclosure.

(b) The means of fire detection, suppression and alarming are provided and accessible.

(13) Remarks—None

9A.4.1.4.27 Room 425

(1) Fire Area—F4230

(2) Equipment: See Table 9A.6-2

Safety-Related	Provides Core Cooling
Yes, D2	No

(b) The BOP scope of piping systems are as follows:

(i) Main steam piping downstream of the MSIV outside containment

(ii) Feedwater piping outside containment downstream of the isolation check valves

(iii) RPV head vent piping

(iv) CUW suction and discharge piping, including the head spray line

(v) RHR suction and discharge and injection piping in shutdown cooling mode and LPFL mode

(vi) RCIC turbine steam supply and exhaust piping

(vii) RCIC pump suction and discharge piping

(viii) SLC system piping (pump suction/discharge)

(ix) RSW suction and discharge piping

(x) RCW suction and discharge piping

(xi) HPCF suction and injection piping

(xii) Diesel generator fuel, cooling, intake and exhaust piping

(xiii) ~~FCS hydrogen recombiner piping~~ Not Used

(xiv) CRD system piping (pump suction/discharge)

Thermal expansion testing during the preoperational phase will consist of displacement measurements on the NSSS portion of piping during the RRS/RPV internal hot functional test (Subsection 14.2.12.1.2) and visual inspections at ambient temperature on the NSSS and BOP portions of piping. The testing will be in conformance with ANSI/ASME-OM7 as discussed in Subsection 3.9.2.1.2, and will consist of a combination of visual inspections and local and remote displacement measurements. This testing, as well as that performed during the power ascension phase per Subsection 14.2.12.2.10, includes the inspection and testing of RCPB component supports as described in Subsection 5.4.14.4. Visual inspections are performed to identify actual or potential constraints to free thermal growth prior to or between tests. Displacement measurements will be made utilizing specially installed instrumentations and also using the position of supports such as snubbers. Results of the thermal expansion testing are acceptable when all systems move as predicted and there are no observed restraints to free thermal growth or when additional analysis shows that any unexpected results will not produce unacceptable stress values.

Vibration testing will be performed on system components and piping during preoperational function and flow testing. This testing will be in accordance with ANSI/ASME-OM3 as discussed in Subsection 3.9.2.1.1 and will include visual observation and local and remote monitoring in critical steady-state

BASES

SURVEILLANCE REQUIREMENTS (Continued)	SR 3.3.6.1.2 (continued) address some of the same components required by the PAM displays.
REFERENCES	1. Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," ~~[Date].~~ May 1983. 2. DCD Tier 2, Section 7.5

Table 18H- 4 (Cont'd)
INVENTORY BASED UPON THE ABWR EPGs
SECONDARY CONTAINMENT CONTROL

SECONDARY CONTAINMENT TEMPERATURE CONTROL SC/T–1 (cont'd)

[[.........

.... [3]]]

Table 18H- 4 (Cont'd)
INVENTORY BASED UPON THE ABWR EPGs
SECONDARY CONTAINMENT CONTROL

SECONDARY CONTAINMENT TEMPERATURE CONTROL SC/T–3 (cont'd)

[[.........

.... {3}]]

Table 18H- 4 (Cont'd)
INVENTORY BASED UPON THE ABWR EPGs
SECONDARY CONTAINMENT CONTROL

SECONDARY CONTAINMENT TEMPERATURE CONTROL SC/T–3 (cont'd)

[[.........

.... {3}]]

Table 18H- 12 (cont'd)
INVENTORY OF CONTROLS BASED UPON THE ABWR EPGs AND PRA

[[.........

.... [3]]]

Table 18H- 12 (cont'd)
INVENTORY OF CONTROLS BASED UPON THE ABWR EPGs AND PRA

[[.........

.... {3}]]

Table 18H- 13 (Cont'd)
INVENTORY OF CONTROLS BASED UPON THE ABWR EPGs AND PRA

[[.........

.... {3}]]

Table 18H- 13 (cont'd)
INVENTORY OF CONTROLS BASED UPON THE ABWR EPGs AND PRA

[[.........

.... {3}]]

The radiation loads on the penetrations are below the TID-14844 limits so radiation is not a concern.

(11) Not Used

~~The recombiner system is needed in a long term accident (order of days) to ensure that the oxygen concentration does not reach flammability limits. The recombiners are located outside of the primary containment. Piping is used to remove and return fluid to the primary containment. Therefore, the process fluid provides the only significant impact on this system. Since the supply and return lines are isolated during the early part of an event, the recombiners are not subjected to the primary containment thermodynamic loads until days later, after accident recovery when the environment is not as severe. At this time, recovery from a postulated accident might occur in a much less severe environment. Additionally, the integrated radiation doses will be well below the design basis values. Therefore, the recombiners will survive these accident scenarios.~~

(12) Pressure and Water Level Instrumentation

The pressure sensors used to measure both water level and pressure in the vessel and in the containment are located outside of containment. The conditions in the vessel and containment are monitored via pressure taps. The pressure sensors will not see the higher vessel or primary containment temperature and radiation doses due to the significant length-to-diameter ratio of the piping used in these sensors. The integrated radiation gamma dose for the pressure sensors is slightly over the equipment qualification limit set forth in Table 3I-16. However, the radiation limits set for design basis events are extremely conservative. Therefore, there is reasonable assurance that the sensors will survive this condition. Furthermore, the sensors are capable of withstanding very high overpressure events, on the order of 14 MPa, indicating that there is no possibility of damage from high containment pressures.

(13) Temperature Instrumentation

The GE standard practice is to use thermocouples rated to 575 K and 14 MPa. These ratings are well above the drywell and wetwell thermodynamic loads experienced during a postulated severe accident. Therefore, operation of the thermocouples should not be adversely affected. Comparison to radiation qualification limits are based on two day integrated dose rates. The equipment integrated radiation doses are below the equipment qualification dose rates of 2.0E+8 R and 2.0E+9 R for gamma and beta radiation, respectively, as set forth in Table 3I-16.

Table 19M-2 Weighting Factors for Adjusting Generic Location Fire Frequencies for Application to Plant-Specific Locations (References FIVE Table1.1) (Continued)

Plant Location (Table 1.1 Of Five)	Weighting Factors[1] (Wfl) (Table 1.1 Of Five)	Weighting Factor (Wfl) ABWR Analysis	WFL Value
Radwaste Area	The number of units per site and divide by the number of radwaste areas.	One reactor divided by one radwaste building per site. (Since the radwaste building is a grouping of fire areas separate from any area containing safety-related equipment, a fire in the radwaste building cannot affect safe shut- down of the plant.)	1
Transformer Yard	The number of units per site and divide by the number of switch-yards.	One reactor divided by one switchyard.	1
Plant-Wide Components (cables, transformers, elevator motors, ~~hydrogen recombiner~~/ analyzer).	The number of units per site.	One reactor per site.	1

Notes:

1. The analyst must identify the number of like locations when determining the number of building, e.g., a 480 volt load center is "like" a switchgear room.
2 Reactor building does not include containment.