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MFN 15-069

Docket number: 52-045

September 9, 2015

US Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: **GEH Proposed Resolution of Item # 26 - Fukushima Recommendation 4.2 Mitigation Strategies of NRC Suggested U.S. Advanced Boiling Water Reactor Design Changes**

References:

1. Letter from R.E. Kingston, GEH to USNRC, Subject: ABWR Standard Plant Design Certification Renewal Application Design Control Document, Revision 5, Tier 1 and Tier 2, December 7, 2010.

2. Letter from USNRC to Jerald G. Head, GEH, Subject: GE-Hitachi Nuclear Energy – United States Advanced Boiling-Water Reactor Design Certification Renewal Application, July 20, 2012.

3. Letter from Jerald G. Head, GEH, to USNRC, Subject: Response to NRC Letter: GE Hitachi Nuclear Energy – United States Advanced Boiling-Water Reactor Design Certification Renewal Application (July 20, 2012), September 17, 2012.

GEH submitted a Design Certification Renewal application for the U.S. Advanced Boiling Water Reactor (ABWR) in Reference 1 pursuant to the requirements of Subpart B, "Standard Design Certifications," of Title 10 of the Code of Federal Regulations (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

In Reference 2, the NRC suggested design changes to address issues that the agency considered to be regulatory improvements or changes that could meet the 10 CFR 52.59(b) criteria. In addition, the NRC requested that GEH implement the Fukushima Near-Term Task Force recommendations contained in SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku

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Earthquake and Tsunami," dated February 17, 2012. Collectively, these items are termed the "28-item backfit list".

In Reference 3, GEH responded to Reference 2 and committed to address the "28-item backfit list".

In public and private meetings the NRC held with GEH on May 7th, 2015, GEH reviewed the closure plan for the "28-item backfit list" for those items that GEH would not receive any additional Requests for Additional Information. During that meeting, Item #26 of the "28-item backfit list" was discussed and GEH's approach was revised taking into consideration the staff's feedback.

Please find GEH's proposed resolution to Item #26 of the "28-item backfit list" transmitted in Reference 2. Enclosure 1 contains the complete response, while Enclosure 2 contains the Design Control Document markups associated with this response.

If you have any questions concerning this letter, please contact Hugh Upton at 408-314-8499.

I declare under penalty of perjury that the foregoing information is true and correct to the best of my knowledge, information, and belief.

Sincerely,

Peter M Yandow FOR

Jerald G. Head
Senior Vice President, Regulatory Affairs

Commitments: No additional commitments are made in this response.

Enclosures:

1. GEH Response to Item #26 - Fukushima Recommendation 4.2 Mitigation Strategies

2. GEH Response to Item #26 - Fukushima Recommendation 4.2 Mitigation Strategies - ABWR DCD Revision 5 Markups

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Enclosure 1

MFN 15-069

GEH Response to Item #26 - Fukushima Recommendation 4.2 Mitigation Strategies

NRC – Fukushima Recommendation 4.2 – Mitigation Strategies

Address the design related aspects of Fukushima Recommendation 4.2 regarding mitigation strategies for beyond-design-basis external events as outlined in Attachment 2 of the Order issued on March 12, 2012 (ML12054A735).

GEH Response:

To provide plant operators with additional mitigation options for response to a Beyond Design Basis Event, GEH will make the following modifications:

- Add Alternating Current (AC) Independent Water Addition (ACIWA) subsystem to Residual Heat Removal (RHR) Loop B, including connection to the Fire Protection system (FP) and external hose connection for fire truck.
- Make FP diesel-driven fire pump fuel capacity sufficient for 72 hours of operation; include severe weather/flooding protection for ACIWA and the diesel-driven fire pump.
- Analyze Reactor Core Isolation Cooling (RCIC) system for operation with 121°C (250°F) pump suction temperature.
- Add Reactor Building (RB) external connections for Diverse and Flexible Coping Strategies (FLEX) diesel generators to 480VAC RB 1E power centers.

Summary

To address core cooling, spent fuel pool (SFP) makeup, and containment spray, GEH will amend the ABWR DCD, Tier 1, Section 2.4.1 and Tier 2, Table 1AA-2, Table 3.2-1, Table 3.9-8, Attachment 3MA.2, Section 5.4, Technical Specifications (TS) Limiting Conditions for Operation (LCOs) 3.5.1 and 3.6.2.4 Actions/Bases, and Section 19.9 to include the addition of ACIWA capability to Loop B of RHR. Tier 1, Section 2.15.6 will be amended to include wetwell spray and SFP makeup in the firewater supply to RHR description. Tier 1, Section 2.4.4 and Tier 2, Section 5.4 will also be amended to show analysis of RCIC operation at 121°C SP temperature during beyond design basis events. Tier 2, Section 19.8 will be amended to include severe weather and flooding protection, minimum fuel supply requirement for ACIWA and the diesel-driven fire pump. Tier 2, Sections 19.8 and 19.9 will be amended to include wetwell spray and spent fuel pool makeup in the ACIWA description.
GEH will amend ABWR DCD, Tier 1, Section 2.12.1 and Tier 2, Section 8.3 to describe the addition of external connection points for FLEX portable 480 VAC diesel generator(s).

Impact on the DCD

ABWR DCD rev 5, Tier 1, Section 2.4.1, third paragraph will be changed to read:

> Except for the non-ASME Code components of the alternating current (AC) power source independent water addition feature (Figures 2.4.1b and 2.4.1c), the entire RHR System shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c is classified as safety-related.

ABWR DCD rev 5, Tier 1, Section 2.4.1 RHR operating modes list item (7) will be changed to read:

(7) AC power source independent water addition (Divisions B and C)

ABWR DCD rev 5, Tier 1, Section 2.4.1 AC Independent Water Addition Mode will be changed to read:

> Divisions B and C of the RHR System will also function in an AC independent water addition mode. This mode provides a means of injecting emergency makeup water to the reactor by cross connecting the Reactor Building Fire Protection (FP) System header, or alternately utilizing additional sources of water from external connections just outside the Reactor Building. This makes the mode independent of the normal safety-related AC power distribution network. This mode is accomplished by manually opening two in-series valves on the cross-connection piping just upstream of the tie-in to the normal RHR piping. This is accomplished by local manual action at the valves. Fire Protection System water can be directed to either the RPV, the wetwell or drywell spray sparger, or the spent fuel pool by local manual opening of the Division B or C RHR injection valve, the Division B or C wetwell spray valve, the two Division B or C drywell spray valves, or the two Division B or C valves to the Fuel Pool Cooling and Cleanup System (FPC), respectively. "Local manual" as used in this paragraph means manually operating the valves at the valves.

ABWR DCD rev 5, Tier 1, Section 2.4.1 Other Provisions, last sentence of last paragraph will be changed to read:

> For RHR-B and C, the upgraded branch lines include all the paths listed for RHR-A plus the supplemental fuel pool cooling suction path from the Fuel Pool Cooling System (including the RHR isolation valve) that connects to the shutdown cooling suction line, titled "From FPC." The upgraded lines also include the pipelines and valves that are part of the AC independent water addition mode that extend from the non-code boundary indicated by "NNS" to the "external connection" outside the "reactor building" and to the Fire Protection System interfaces indicated by "FP".

ABWR DCD rev 5, Tier 1, Figure 2.4.1.b Residual Heat Removal System (RHR-B) will be updated to show the new AC independent water addition piping, valves, and connections for Loop B.

To ABWR DCD rev 5, Tier 1, Section 2.4.4, the following text will be added:

> The RCIC system is capable of injecting sufficient water to the vessel to maintain core cooling with suction aligned to the suppression pool and a suction temperature of 121°C (250°F) during beyond design basis events (e.g. Extended Station Blackout).

To ABWR DCD rev 5, Tier 1, Section 2.4.4, Table 2.4.4 the following text will be added:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. The RCIC system has the capability of injecting sufficient water to the vessel to maintain core cooling with suction aligned to the suppression pool and a suction temperature of 121°C (250°F) during beyond design basis events (e.g. Extended Station Blackout).	11. Analyses will be performed of the as-built RCIC System to assess the system capability with 121°C water at the pump suction.	11. The RCIC system is capable of injecting sufficient water to the vessel to maintain core cooling with suction aligned to the suppression pool and a suction temperature of 121°C (250°F) during beyond design basis events (e.g. Extended Station Blackout).

ABWR DCD rev 5, Tier 1, Section 2.15.6 will be modified to read:

Fire water supply connections to Loops B and C of the Residual Heat Removal System piping are provided from the portion of FPS used for the Reactor and Control Buildings. These connections are part of the AC independent water addition mode of the RHR System for reactor vessel injection, wetwell or drywell spray, or spent fuel pool makeup.

ABWR DCD rev 5, Tier 2, Table 1AA-2, page 1AA-11 will be modified to read:

FPS Supply Valve   E11-F101, 102, 103;B,C   Valve Rm. B,C (SC)

ABWR DCD rev 5, Tier 2, Table 3.2-1, E11 RHR System will be modified to read:

11. Valves to fire protection, Subsystems B and C (F100B/C, F103B/C and F104B/C)

ABWR DCD rev 5, Tier 2, Table 3.9-8, page 3.9-105 will be modified as follows:

E11-F101 and E11-F102 "Qty" will be changed to 2.
Figure 5.4-10 sh. 5 will be added to "Tier 2 Fig." column for E11-F101 and E11-F102.

ABWR DCD rev 5, Tier 2, Attachment 3MA.2.2 will be modified to reflect the addition of ACIWA to RHR Loop B.

ABWR DCD rev 5, Tier 2, Attachment 3MA.2.3 will be modified to move the "RHR Subsystem C suction piping from the suppression pool" from the middle of RHR B section to the beginning of the RHR C section and to add the new "RHR Subsystem B outdoor fire truck connection in RHR pump discharge pipe to the RPV" to the end of the RHR B section.

ABWR DCD rev 5, Tier 2, Section 5.4.6.1.1.1 will be modified to read:

The RCIC System shall initiate and discharge, within 30 seconds, a specified constant flow into the reactor vessel over a specified pressure range. The RCIC water discharge into the reactor vessel varies between a temperature of 10°C up to and including a temperature of 77°C during design conditions and up to 121°C during beyond design basis events (e.g. Extended Station Blackout). The mixture of the cool RCIC water and the hot steam does the following:

ABWR DCD rev 5, Tier 2, Section 5.4.7.1 will be modified to read:

The RHR System has two AC-independent water addition subsystems which consist of piping and manual valves connecting the fire protection system to the RHR pump discharge lines on loops B and C downstream of each main pump's discharge check valve. These flow paths allow for injection of water into the reactor vessel, wetwell or drywell spray, or the spent fuel pool during severe accident conditions in which all AC power and all ECCS pumps are unavailable. Additionally, hookups are provided external to the reactor building for connection of a fire truck pump and alternate water source to each subsystem. The hookups are on different faces of the reactor building and separated by more than 40 meters.

ABWR DCD rev 5, Tier 2, Section 5.4.7.1.1.10 will be modified to read:

The AC-independent water addition mode (Alternating Current independent) of the RHR System provides a means for introducing water from the Fire Protection System (FPS) directly into the reactor pressure vessel, to the drywell or wetwell spray header, or to the spent fuel pool under degraded plant conditions when AC power is not available from either onsite or offsite sources. The RHR System provides the piping and valves which connect the FPS piping with RHR loops B and C pump discharge piping. The manual valves in these lines permit adding water from the FPS to the RHR System if the RHR System is not operable. The primary means for supplying water through this connection is by use of the diesel-driven pump in the FPS. A backup to this pump is provided by connections on the outside of the reactor building at grade level, which allows hookup of the ACIWA to fire truck pumps.

Figure 5.4-10 shows the connections from either the diesel-driven pump or the fire truck to RHR system loops B and C. The connections to the diesel-driven pump (FPS) are adjacent to the RHR valve rooms. Opening valves F101B/C and F102B/C allows water to flow from the FPS into the RHR piping. Periodic stroke testing of these valves is required by Table 3.9-8 to ensure valve operability. The fire truck connections are located outside the reactor building at grade level, on different faces of the building. Both connections to the RHR system are protected by a check valve (F100B/C and F104B/C for the diesel-driven pump and the fire truck, respectively) to insure that RCS pressurization does not result in a breach of the injection path. Detailed procedures for the operation of the ACIWA, including operation of the FPS valve in the yard, are required to be developed by the COL applicant. See Section 19.9.7.

ABWR DCD rev 5, Tier 2, Section 5.4.7.1.1.10.4 will be modified to read:

> Operation of the AC-independent water addition mode is entirely manual. All of the valves which must be opened or closed during fire water addition are located adjacent to or within the applicable ECCS valve room. The connections to add water using fire truck pumps are located outside the reactor building at grade level on different faces of the building.

ABWR DCD rev 5, Tier 2, Section 5.4.7.2.6(8) will be modified to read:

> The RHR System is provided with piping and valves which separately connect RHR loops B and C pump discharge piping to the Fire Protection System (FPS) and to reactor building external fire truck pump hookups. These connections allow for addition of FPS water to the reactor pressure vessel, the drywell or wetwell spray header, or the spent fuel pool during events when AC power is unavailable from both onsite and offsite sources. Operation of the RHR System in the AC-independent water addition mode is entirely manual. All valves required to be opened or closed for operation are located adjacent to or within the respective loop ECCS valve room to provide ease of operation.

ABWR DCD rev 5, Tier 2, Table 5.4-2 will be modified to account for analysis of RCIC system operation at 121°C SP temperature during beyond design basis events (e.g. Extended Station Blackout). This appears in two places in the table.

ABWR DCD rev 5, Tier 2, Figure 5.4-10 Residual Heat Removal System P&ID (Sheet 5 of 7) will be updated to show the new AC independent water addition piping, valves, and connections for Loop B.

ABWR DCD rev 5, Tier 2, Figure 5.4-10 Residual Heat Removal System P&ID (Sheet 7 of 7) will be updated to show the naming changes for ACIWA valves and instruments for Loop C.

ABWR DCD rev 5, Tier 2, Chapter 16 Technical Specification LCOs 3.5.1 and 3.6.2.4 Actions will be updated to reflect the addition of ACIWA to RHR Loop B.

ABWR DCD rev 5, Tier 2, Chapter 16 Technical Specification LCOs 3.5.1 and 3.6.2.4 Bases will be updated to reflect the addition of ACIWA to RHR Loop B.

To ABWR DCD rev 5, Tier 2, Section 19.8.1.3 the following text will be added:

> The ACIWA System (including the diesel-driven fire pump fuel supply tank) will be protected against site flood and severe weather events. The ACIWA diesel fuel storage tank will have sufficient storage capacity to support 72 hours of operation.

ABWR DCD rev 5, Tier 2, Section 19.8.2.3 will be updated to include wetwell spray and spent fuel pool makeup in the ACIWA description.

ABWR DCD rev 5, Tier 2, Table 19.8-2 Seismic qualification of the ACIWA system Basis will be updated to include wetwell spray and spent fuel pool.

ABWR DCD rev 5, Tier 2 ,Table 19.8-7 Firewater Addition System Injection Locations will be updated to include wetwell spray and spent fuel pool.

ABWR DCD rev 5, Tier 2, Section 19.9.7 will be amended to include spent fuel pool makeup.

ABWR DCD rev 5, Tier 2, Section 19.9.7 will be amended to read:

> The procedures to be developed by the applicant will address operation of the ACIWA for vessel injection, drywell or wetwell spray operation, and spent fuel pool makeup. Operation of the ACIWA System in the vessel injection mode requires valves F005B/C, F101B/C, and F102B/C to be opened and valve F592B/C to be closed. Reactor depressurization to below ACIWA System operating pressure is required prior to ACIWA operation in the vessel injection mode. Operation of the ACIWA in the drywell spray mode requires valves F017B/C, F018B/C, F101B/C, and F102B/C to be opened and valve F592B/C to be closed. Operation of the ACIWA in the wetwell spray mode requires valves F019B/C, F101B/C, and F102B/C to be opened and valve F592B/C to be closed. Operation of the ACIWA in the spent fuel pool makeup mode requires valves F014B/C, F015B/C, F101B/C, and F102B/C to be opened and valve F592B/C to be closed. These valves are shown on Figure 5.4-10. The diesel fire pump will start automatically when the ACIWA is properly aligned. If the normal firewater system water supply is unavailable, the alternate water supply can be made available by opening the manual valve between the diesel driven fire pump and the alternate water supply. This valve is shown in Figure 9.5-4. If it is necessary to use a fire truck, valve F103B/C must be opened in addition to operation of the valves discussed above for ACIWA operation. The valve for operation of the ACIWA using the fire truck is also shown on Figure 5.4-10. All of the valves required for ACIWA operation are manually operable.

To ABWR DCD rev 5, Tier 1, Section 2.12.1, the following text will be added:

> External (to Reactor Building) connections are provided to all three 1E Reactor Building 480 VAC Power Centers for portable FLEX diesel generators. The connectors are isolated from the Power Centers by normally open 1E breakers.

ABWR DCD rev 5, Tier 1, Figure 2.12.1 will be updated to show the FLEX diesel generator tie-in points to the 1E divisional power centers.

To ABWR DCD rev 5, Tier 2, Section 8.3.1.1.2.1, the following text will be added:

> To deal with an Extended Loss of AC Power (ELAP), external (to the Reactor Building) connections to each 1E RB divisional power center for portable FLEX 480 VAC diesel generators (DG) are installed, normally isolated from the 1E 480 VAC divisional Power Centers by open 1E breakers. The Reactor and Control Building 480 VAC buses can be re-energized via FLEX DGs, powering the 1E battery chargers, DC buses, and vital buses through UPS. The 480 VAC bus feeder breaker from the 6.9 KV to 480 V transformer and all load breakers on the Power Center will be opened to isolate the bus. The FLEX DG is connected, started, then the bus feeder from the DG is closed, energizing the 480 VAC power center. Power center load breakers are then closed one by one to energize loads (MCCs and discrete loads).

ABWR DCD rev 5, Tier 2, Figure 8.3-1 Sheet 3 will be updated to show the FLEX diesel generator tie-in points to the 1E divisional power centers.

The ABWR DCD Rev 5 marked-up pages are provided in Enclosure 2. Pages from the beginning of the DCD section have been included even though they may not have changed to provide easier review.

Enclosure 2

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GEH Response to Item #26 - Fukushima Recommendation 4.2 Mitigation Strategies

ABWR DCD Revision 5 Markups

IMPORTANT NOTICE REGARDING CONTENTS OF THIS DOCUMENT
Please Read Carefully

2.4.1 Residual Heat Removal System

Design Description

The Residual Heat Removal (RHR) System has three separate divisions. The major functions of the RHR System are:

(1) Containment heat removal.

Except for the non-ASME Code components of the alternating current (AC) power source independent water addition feature (Figures 2.4.1b and 2.4.1c), the entire RHR System shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c is classified as safety-related.

(4) Augmented fuel pool cooling.

Figures 2.4.1a, 2.4.1b, and 2.4.1c show the basic system configuration and scope. Figure 2.4.1d shows the RHR System control interfaces.

~~Except for the non-ASME Code components of the alternating current (AC) power source independent water addition feature (Figure 2.4.1c), the entire RHR System shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c is classified as safety-related.~~

The RHR System operates in the following modes:

(1) Low pressure core flooder (LPFL) (Divisions A, B, and C)

(2) Suppression pool cooling (Divisions A, B, and C)

(3) Wetwell spray (Divisions B, and C)

(7) AC power source independent water addition (Divisions B and C)

(5) Shutdown cooling (Divisions A, B, and C)

(6) Augmented fuel pool cooling, and fuel pool makeup (Divisions B, and C)

~~(7) AC power source independent water addition (Division C)~~

(8) Full flow test (Divisions A, B, and C)

(9) Minimum flow bypass (Divisions A, B, and C)

Low Pressure Core Flooder Mode

As shown on Figure 2.4.1d, the RHR System channel measurements are provided to the Safety System Logic and Control (SSLC) for signal processing, setpoint comparisons, and generating trip signals. The RHR System is automatically initiated when either a high drywell pressure or low reactor water level condition exists (i.e., LOCA signal). A RHR initiation signal is provided

to the systems as identified on Figure 2.4.1d. The SSLC processors use a two-out-of-four voting logic for RHR System initiation. Each RHR division can also be initiated manually (LPFL mode).

Following receipt of an initiation signal, the RHR System automatically initiates and operates in the LPFL mode to provide emergency makeup to the reactor vessel. The initiation signal starts the pumps, which run in the minimum flow mode until the reactor depressurizes to less than the pump's developed head pressure. A low reactor pressure permissive signal occurs above the pump's developed head pressure, which signals the injection valve to open. As the injection valve opens, the reactor pressure is contained by the testable check valve until the reactor pressure becomes less than the pump's developed head pressure of the minimum flow mode, at which time injection flow begins. This sequence satisfies the response requirements for all potential LOCA pipe breaks when the injection valve opens within 36 seconds after receiving the low reactor pressure permissive signal. The LPFL injection flow for each division begins when the reactor vessel pressure is no less than 1.55 MPa above the drywell pressure. When the reactor vessel pressure is no less than 0.275 MPa greater than the drywell pressure, the LPFL injection flow for each division is 954 m^3/h minimum. The LPFL mode is accomplished by all three divisions of the RHR System by transferring water from the suppression pool to the reactor pressure vessel (RPV), via the RHR heat exchangers. The system automatically aligns to the LPFL mode of operation from the test mode, the suppression pool cooling, or wetwell spray modes upon receipt of an initiation signal. The wetwell spray mode is applicable for Divisions B or C. If a drywell spray valve is open in Division B or C, that RHR division automatically aligns to the LPFL mode in response to the injection valve beginning to open. The RPV injection valve in each division requires a low reactor pressure permissive signal to open, and closes automatically on receipt of a high reactor vessel pressure signal.

Suppression Pool Cooling Mode

The suppression pool cooling mode of the RHR System limits the long-term post-LOCA temperature of the suppression pool, and limits the long-term peak temperatures and pressures within the wetwell and drywell regions of the containment. In this mode, the RHR System circulates water through the RHR heat exchangers and returns it directly to the suppression pool. This mode is manually initiated by control of individual system components. In the suppression pool cooling mode, the total heat removal capacity between the RHR and ultimate heat sink is no less than 0.371 MJ/s·°C for each division. 0.371 MJ/s·°C is the limiting heat removal capacity of all the RHR modes. The heat removal path is the RHR heat exchanger, the Reactor Building Cooling Water (RCW) System, and the Reactor Service Water (RSW) System. In the suppression pool cooling mode, the RHR tube side heat exchanger (Hx) flow rate is 954 m^3/h minimum per division. The RHR pumps have sufficient net positive suction head (NPSH) available at the pump. Suction from the suppression pool is the limiting NPSH condition of all the RHR modes.

Containment Spray Mode

The containment spray mode of the RHR System is available in Divisions B and C, and consists of the wetwell spray and drywell spray operating together. In this mode, the RHR System pumps suppression pool water to a single wetwell spray header and single drywell spray header through the associated RHR heat exchanger. The containment spray mode of the RHR System is initiated manually by control of individual system components. The drywell spray inlet valves can only be opened if a high drywell pressure condition exists and if the injection valves are fully closed. The wetwell spray flow rate for either Division B or C is no less than 114 m^3/h.

Shutdown Cooling Mode

In the shutdown cooling mode of operation, the RHR System removes decay heat from the reactor core, and is used to achieve and maintain a cold shutdown condition by removing decay and sensible heat from the core and reactor vessel. This mode reduces reactor pressure and temperature to cold shutdown conditions. In this mode, each division takes suction from the RPV via its dedicated suction line, pumps the water through its respective heat exchanger tubes, and returns the cooled water to the RPV. Two divisions (B and C) discharge water back to the RPV via dedicated spargers, while the third division (A) utilizes the vessel spargers of one of the two feedwater lines (FW-A). Shutdown cooling is initiated manually once the RPV has been depressurized below the system low pressure permissive. In any division, the shutdown cooling suction valve cannot be opened unless the following valves in that division are closed:

(1) Suppression pool suction valve

(2) Suppression pool return valve

(3) Drywell spray valves

(4) Wetwell spray valve

Each shutdown cooling suction valve automatically closes on low reactor water level. The low pressure portions of the shutdown cooling piping are protected from high reactor pressure by automatic closure of the shutdown cooling suction valves on a high reactor vessel pressure. The shutdown cooling flow rate for any division is no less than 954 m^3/h.

Augmented Fuel Pool Cooling and Fuel Pool Makeup

The augmented fuel pool cooling mode of the RHR System (Divisions B and C) can supplement the Fuel Pool Cooling (FPC) System as follows: (1) directly cooling the fuel pool by circulation fuel pool water through the RHR heat exchanger and returning it to the fuel pool; and (2) while providing shutdown cooling during refueling operations, return the cooled RHR shutdown cooling flow to the fuel pool. Also, this mode provides for fuel pool emergency makeup capability by permitting the RHR pumps (Divisions B and C) to transfer suppression pool water to the fuel pool. This mode is accomplished manually by control of individual system components. In the augmented fuel pool cooling mode, the RHR tube side heat exchanger flow rate for Division B or C is no less than 350 m^3/h.

AC Independent Water Addition Mode

~~Division C of the RHR System also functions in an AC independent water addition mode. This mode provides a means of injecting emergency makeup water to the reactor by cross connecting the Reactor Building Fire Protection (FP) System header, or alternately utilizing additional sources of water from an external connection just outside the Reactor Building. This makes it independent of the normal safety-related AC power distribution network. This mode is accomplished by manually opening two in-series valves on the cross-connection piping just upstream of the tie-in to the normal RHR piping. This is accomplished by local manual action at the valves. Fire Protection System water can be directed to either the RPV or the drywell spray sparger by local manual opening of the Division C RHR injection valve or the two Division C drywell spray valves. "Local manual" as used in this paragraph means manually operating the valves at the valves.~~

Full Flow Test Mode

Each division of the RHR System has a full flow test mode to permit pump flow testing during

Divisions B and C of the RHR System will also function in an AC independent water addition mode. This mode provides a means of injecting emergency makeup water to the reactor by cross connecting the Reactor Building Fire Protection (FP) System header, or alternately utilizing additional sources of water from external connections just outside the Reactor Building. This makes the mode independent of the normal safety-related AC power distribution network. This mode is accomplished by manually opening two in-series valves on the cross-connection piping just upstream of the tie-in to the normal RHR piping. This is accomplished by local manual action at the valves. Fire Protection System water can be directed to either the RPV, the wetwell or drywell spray sparger, or the spent fuel pool by local manual opening of the Division B or C RHR injection valve, the Division B or C wetwell spray valve, the two Division B or C drywell spray valves, or the two Division B or C valves to the Fuel Pool Cooling and Clean Up System (FPC), respectively. "Local manual" as used in this paragraph means manually operating the valves at the valves.

signal.

Other Provisions

The RHR System is classified as Seismic Category I. Figures 2.4.1a, 2.4.1b, and 2.4.1c show the ASME Code Class for the RHR System. The RHR System is located in the Reactor Building.

Each of the three divisions is powered from the Class 1E division as shown on Figures 2.4.1a, 2.4.1b, 2.4.1c. In the RHR System, independence is provided between Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

Outside the primary containment, each mechanical division of the RHR System (Divisions A, B, and C) is physically separated from the other divisions.

The RHR System has the following displays and controls in the main control room:

(1) Parameter displays for the instruments shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c.

(2) Controls and status indication for the active safety-related components shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c.

(3) Manual system level initiation capability for the following modes:

(a) LPFL initiation

(b) Standby

(c) Shutdown cooling

(d) Suppression pool cooling

(e) Drywell spray

RHR System components with displays and control interfaces with the Remote Shutdown System (RSS) are shown on Figures 2.4.1a and 2.4.1b.

The safety-related electrical equipment shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c located inside the primary containment and the Reactor Building is qualified for a harsh environment.

The motor-operated valves shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c have active safety-related functions and perform these functions to open, close, or both open and close, under differential pressure, fluid flow, and temperature conditions.

The check valves (CVs) shown on Figures 2.4.1a, 2.4.1b, and 2.4.1c have safety-related functions to open, close, or both open and close under system pressure, fluid flow, and temperature conditions.

The RHR System main pumps are interlocked to prevent starting with a closed suction path.

Each RHR loop has a continuously running jockey pump to maintain the system piping

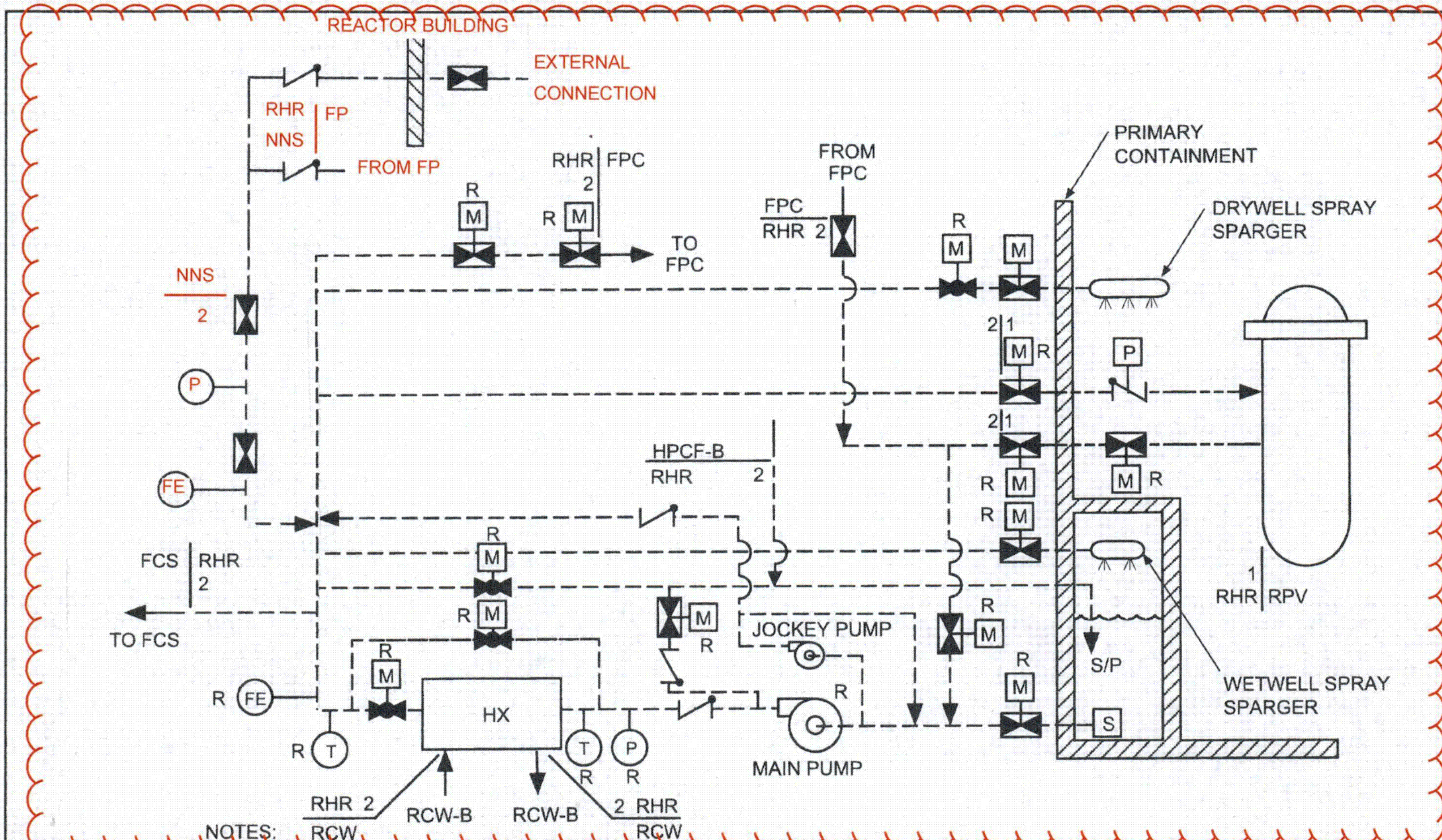
For RHR-B and C, the upgraded branch lines include all the paths listed for RHR-A plus the supplemental fuel pool cooling suction path from the Fuel Pool Cooling System (including the RHR isolation valve) that connects to the shutdown cooling suction line, titled "From FPC." The upgraded lines also include the pipelines and valves that are part of the AC independent water addition mode that extend from the non-code boundary indicated by "NNS" to the "external connection" outside the "reactor building" and to the Fire Protection System interfaces indicated by "FP."

suction side of the pump with all its branches have a design pressure of 2.82 MPaG for intersystem LOCA (ISLOCA) conditions. Refer to Figures 2.4.1a, 2.4.1b, and 2.4.1c. For RHR-A, the upgraded branch lines from the main pump suction include the path to and including the suppression pool suction valve, the path to the shutdown cooling outboard containment isolation valve, and the path to the jockey pump's discharge check valve including the jockey pump's bypass return line. ~~For RHR-B and C, the upgraded branch lines include all the paths listed for RHR-A plus the path to (and including the valve) the Fuel Pool Cooling System that branches off the shutdown cooling suction line, titled "From FPC." For RHR-C, the upgraded branch lines include all the paths listed for RHR A and B plus the pipeline and valves that are part of the AC independent water addition mode that extends from the noncode boundary~~

~~indicated by "NNS" to the "external connection" outside the "reactor building" and to the Fire Protection System interface indicated by "FP".~~

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.4.1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the RHR 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)

2.4.4 Reactor Core Isolation Cooling System

Design Description

The Reactor Core Isolation Cooling (RCIC) System consists of a turbine, pump, piping, valves, controls and instrumentation. The RCIC turbine is driven by the steam from the reactor pressure vessel (RPV) which then drives the RCIC pump. The function of the RCIC System is to provide makeup water to the RPV.

The RCIC steam supply to the turbine branches off one of the main steamlines inside containment upstream of the inboard MSIV and exhausts to the suppression pool (S/P). The primary source of RCIC pump suction is the Condensate Storage Tank (CST). The suppression pool is the secondary source of RCIC pump suction. Figure 2.4.4a shows the basic system configuration and scope. Figure 2.4.4b shows RCIC System control interfaces.

The RCIC System shown on Figure 2.4.4a is classified as safety-related.

The RCIC System operates in the following modes:

(1) RPV water makeup.

(2) Full flow test.

(3) Minimum flow bypass.

RPV Water Makeup Mode

As shown on Figure 2.4.4b, the RCIC System channel measurements are provided to the Safety System Logic and Control (SSLC) System for signal processing, setpoint comparisons, and generating trip signals. The RCIC System is automatically initiated when either a high drywell pressure or low reactor water level condition exists. RCIC System is actuated at a reactor water level higher than the High Pressure Core Flooder (HPCF) system actuation level. The SSLC processors use a two-out-of-four voting logic for system initiation and shutdown. Manual RCIC System initiation can be performed from the main control room (MCR). The RCIC System can be started by local operation of RCIC System components outside the MCR.

The RCIC System automatically shuts down when a high reactor water level condition exists. Following RCIC shutdown on high reactor water level signal, the RCIC System automatically restarts to provide RPV water makeup, if the low reactor water level initiation signal recurs.

During this mode, the primary source pump suction is the CST. Automatic transfer of pump suction from the CST to the S/P occurs when a low CST water level or a high suppression pool water level signal exists. This transfer can be manually overridden from the MCR. The CST and S/P water level signals are processed through SSLC's two-out-of four voting logic to initiate suction transfer.

The RCIC system is capable of injecting sufficient water to the vessel to maintain core cooling with suction aligned to the suppression pool and a suction temperature of 121°C (250°F) during beyond design basis events (e.g. Extended Station Blackout).

In the RPV water makeup mode, the RCIC pump delivers a flow rate of at least 182 m^3/h against a maximum differential pressure (between the RPV and the suction source) of 8.12 MPa. This flow rate is achieved within 29 seconds of receipt of the system initiation signal. The RCIC pump has sufficient net positive suction head (NPSH) available at the pump.

The RCIC System operates for a period of at least 2 hours under conditions of no AC power availability and no other simultaneous failures, accidents or other design basis conditions.

Full Flow Test Mode

The RCIC System has a full flow test mode to permit pump flow testing during plant operation. During the test, water is pumped from the suppression pool and returned to the suppression pool via the test return line. The vessel injection valve is kept closed.

If a system initiation signal occurs during the full flow test mode, the RCIC System automatically aligns to the RPV water makeup mode.

Minimum Flow Bypass Mode

The RCIC System has a minimum flow bypass mode that assures there is always flow in the RCIC pump when it is operating. This is accomplished automatically by monitoring pump discharge flow, and opening a minimum flow valve to the suppression pool when flow falls below minimum value. The minimum flow valve closes when the pump flow exceeds the minimum value. Minimum flow bypass operation is automatic based on a flow signal opening the minimum flow valve when the flow is low, with a concurrent high pump discharge pressure signal.

Other Provisions

The RCIC System shown on Figure 2.4.4a is classified as Seismic Category I. Figure 2.4.4a shows the ASME Code class for the RCIC System. The RCIC System is located inside primary containment and in the Reactor Building.

As shown on Figure 2.2.4a, the RCIC System components are powered from Class 1E Division I, except for the steam supply outboard containment isolation valve, which is powered from Class 1E Division II. All RCIC System components shown on Figure 2.2.4a except the inboard containment isolation valves are powered from DC sources. In the RCIC System, independence is provided between Class 1E divisions, and also between Class 1E divisions and non-Class 1E equipment.

Outside the primary containment, except for the piping from the CST, the RCIC System shown on Figure 2.4.4a is physically separated from the two divisions of the High Pressure Core Flooder (HPCF) System.

The RCIC System has the following displays and controls in the main control room (MCR):

(1) Parameter displays for the instruments shown on Figure 2.4.4a.

Table 2.4.4 Reactor Core Isolation Cooling System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
9.	9.	9.
a. MOVs designated in Section 2.4.4 as having active safety-related function open, close or both open and close under differential pressure, fluid flow, and temperature conditions.	a. Tests of installed valves for opening, closing, or both opening and closing will be conducted under pre-operational differential pressure, fluid flow, and temperature conditions.	a. Upon receipt of the actuating signal, each MOVs opens, closes, or both opens and closes, depending upon the valve's safety functions. The following valves open, or close, in the following time limits: Valve — Time Steam Supply Containment Isolation Valves — ≤ 30 s Close Injection Valve — ≤ 15 s Open
b. CVs designated in Section 2.4.4 as having an active safety-related function open, close, or both open and close, under system pressure, fluid flow, and temperature conditions.	b. Tests of installed valves for opening, closing, or both opening and closing, will be conducted under system preoperational pressure, fluid flow, and temperature conditions.	b. Based on the direction of the differential pressure across the valve, each CV opens, closes, or both opens and closes, depending upon the valve's safety functions.
10. The RCIC turbine is tripped if low suction pressure condition is present.	10. Test will be conducted using a simulated low suction pressure signal.	10. The turbine trip and throttle valve receives a trip signal.

Design Commitment

11. The RCIC system has the capability of injecting sufficient water to the vessel to maintain core cooling with suction aligned to the suppression pool and a suction temperature of 121°C (250°F) during beyond design basis events (e.g. Extended Station Blackout).

Inspections, Tests, Analyses

11. Analyses will be performed of the as-built RCIC System to assess the system capability with 121°C water at the pump suction.

Acceptance Criteria

11. The RCIC system is capable of injecting sufficient water to the vessel to maintain core cooling with suction aligned to the suppression pool and a suction temperature of 121°C (250°F) during beyond design basis events (e.g. Extended Station Blackout).

2.15.6 Fire Protection System

Design Description

The Fire Protection System (FPS) detects, alarms and extinguishes fires. Fire detection and alarm systems are provided in all fire areas. The FPS consists of a motor driven pump, a diesel drive pump, sprinkler systems, standpipes and hose reels, and portable extinguishers. The foam systems are also used for special applications. The basic configuration of the FPS water supply system is shown on Figure 2.15.6. The FPS provides fire protection for the Reactor Building, Control Building, Turbine Building, Radwaste Building, and other plant buildings.

Areas covered by sprinklers or foam systems are also covered by the manual hose system. Areas covered only by manual hoses can be reached from at least two hose stations. A hose reel and fire extinguisher are located no greater than 30.5m from any location within the buildings.

The FPS is classified as non-safety-related. The sprinkler systems and the standpipe systems in the Reactor and Control Buildings and portions of the FPS water supply system identified in Figure 2.15.6 remain functional following a safe shutdown earthquake (SSE). These portions of the water supply are separated from the remainder of the system by valves as shown in Figure 2.15.6.

Fire water supply connections to Loops B and C of the Residual Heat Removal System piping are provided from the portion of FPS used for the Reactor and Control Buildings. These connections are part of the AC independent water addition mode of the RHR System for reactor vessel injection, wetwell or drywell spray, or spent fuel pool makeup.

or Control Building. The two fire water pumps provide 5678 liters/min of flow each at a differential pressure of 863 kPa.

~~A fire water supply connection to the Residual Heat Removal System piping is provided from the portion of the FPS used for the Reactor and Control Buildings to provide an AC independent water addition system mode of the RHR System for reactor vessel injection or drywell sprays.~~

Automatic foam water extinguishing systems are provided for the diesel generator rooms and day tank rooms.

Fire detection and alarm systems are supplied with power from a non-Class 1E uninterruptible power supply.

The FPS has the following displays and alarms in the Main Control Room (MCR):

(1) Detection system fire alarms.

(2) Status of FPS pumps.

Table 1AA-2 Post-Accident Emergency Core Cooling Systems and Auxiliaries

Equipment	MPL	Location
ADS & Transmitters		
SR Valve	B21-F010A,C,F,H,L,N,R,T	Upper Drywell (PC)
SR Accumulator	B21-A004A,C,F,H,L,N,R,T	Upper Drywell (PC)
Rx Water Level (ADS,RHR)	B21-LT003A thru H	Instrument Rack Rm. (SC)
Rx Water Level (HPCF)	B21-LT001A,B,C,D	Instrument Rack Rm. (SC)
Rx Pressure (RHR)	B21-PT301A,B,C,D	Instrument Rack Rm. (SC)
DW Pressure (HPCF, RHR)	B21-PT025A,B,C,D	Instrument Rack Rm. (SC)
HPCF		
Pumps	E22-C001B,C	HPCF Rm. B,C (SC)
SP Suction Valve	E22-F006B,C	HPCF Rm. B,C (SC)
Rx Injection Valve	E22-F003B,C	Valve Rm. B,C (SC)
CST Suction Valve	E22-F001B,C	Valve Rm. B,C (SC)
Essential HVH (HVAC)	U41-D102,106	HPCF Rm. B,C (SC)
CST Water Level	P13-LT001A,B,C,D	HPCF Rm. B,C (SC)
Flow	E22-FT008B1,B2,C1,C2	By HPCF Rm. B,C (SC)
Suction Pressure	E22-PT002,003; B,C	By HPCF Rm. B,C (SC)
Injection Pressure	E22-PT006,007; B,C	By HPCF Rm. B,C (SC)
LPCF		
Pump	E11-C001A,B,C	RHR Rm. A,B,C (SC)
Heat Exchanger	E11-B001A,B,C	RHR Rm. A,B,C (SC)
RCW Discharge Valve	P21-F013A,B,C	RHR Rm. A,B,C (SC)
SP Suction Valve	E11-F001A,B,C	RHR Rm. A,B,C (SC)
Rx Injection Valve	E11-F005A,B,C	Valve Rm. A,B,C (SC)
Rx Return Valve	E11-F010,011,012;A,B,C	Valve Rm. A,B,C (SC)
DW Spray Valve	E11-F017,018;B,C	Valve Rm. B,C (SC)
WW Spray Valve	E11-F019B,C	Valve Rm. B,C (SC)
FPC Supply Valve	E11-F015B,C	Valve Rm. B,C (SC)
FPS Supply Valve	E11-F101,102,103;B,C	Valve Rm. B,C (SC)
Essential HVH (HVAC)	U41-D103,104,105	RHR Rm. (SC)
Flow	E11-FT008A1,B1,C1	By RHR Rm. A,B,C (SC)
Flow	E11-FT008A2,B2,C2	By RHR Rm. A,B,C (SC)

Table 3.2-1 Classification Summary (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classi-fication[d]	Quality Assur-ance Require-ment[e]	Seismic Category[f]	Notes
E1 RHR System							
1.	Heat exchangers—primary side	2	SC	B	B	I	
2.	Deleted						
3.	Piping including supports within outermost isolation valves*	1	C,SC	A	B	I	(g)
4.	Containment spray piping including supports and spargers, within and including the outer most isolation valves 2	2	C,SC	B	B	1	
4a.	Piping including supports beyond outermost isolation valves	2	SC	B	B	I	(g)
5.	Main Pumps including supports	2	SC	B	B	I	
6.	Main Pump motors	2	SC	B	B	I	
7.	Valves—isolation, (LPFL line) including shutdown suction line isolation valves	1	C,SC	A	B	I	(g)
8.	Valves—isolation, other (pool suction valves and pool test return valves)	2	SC	B	B	I	(g)
9.	Valves beyond isolation valv	2	SC	B	B	I	(g)
10.	Jockey pump motors includ supports			B	B	I	
~~11.~~	~~Valves to fire protection, Subsystem C (F100C, F103C and F104C)~~ 11. Valves to fire protection, Subsystems B and C (F100B/C, F103B/C, and F104B/C)	N	SC	—	E	—	
E2 High Pressure Core Flooder System							

Notes and footnotes are listed on pages 3.2-52 through 3.2-59

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)
F049	2	Drywell spray line vent & test line inboard valve	2	B	P		E1	5.4-10 sh. 5,7
F051	3	Fill pump discharge line relief valve	2	C	A	R	5 yr	5.4-10 sh.3,4,6
F052	1	Drain line for the suppression pool	2	B	P		E1	5.4-10 sh. 4
F101	2 ~~1~~	AC independent water addition input valve	2	B	A	S	3 mo	5.4-10 sh. ~~7~~ 5,7
F102	2 ~~1~~	AC independent water addition input valve	2	B	A	S	3 mo	5.4-10 sh. ~~7~~ 5,7
F500	3	Heat exchanger inlet drain line inboard valve	2	B	P		E1	5.4-10 sh.3,4,6
F502	3	HX outlet line drain line inboard valve	2	B	P		E1	5.4-10 sh.3,4,6
F504	3	RPV injection line vent line inboard valve	2	B	P		E1	5.4-10 sh.3,4,7
F506	1	RPV injection line drain line inboard valve	2	B	P		E1	5.4-10 sh. 3
F506	2	RPV injection line drain line inboard valve	1	B	P		E1	5.4-10 sh. 5,7
F508	3	Shutdown Cooling suction line vent line valve	2	B	P		E1	5.4-10 sh. 2
F509	2	Vent valve—FPC return line	2	B	P		E1	5.4-10 sh. 5,7
F511	2	Drywell spray line inboard drain line valve	2	B	P		E1	5.4-10 sh. 5,7
F513	2	Drywell spray line inboard drain line valve	2	B	P		E1	5.4-10 sh. 5,7
F515	2	Wetwell spray line inboard drain line valve	2	B	P		E1	5.4-10 sh. 5,7
F517	3	RHR pump min flow line drain line inboard valve	2	B	P		E1	5.4-10 sh.3,4,6
F700	3	RHR pump suction line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F701	3	RHR pump suction line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6

3MA System Evaluation For ISLOCA

3MA.1 General Comments About the Appendix

This Attachment discusses each of the systems evaluated in detail, presented in the order listed in the Appendix, and following a repetitive outline format.

The first section, “Upgrade Description,” describes the changes made to the system and the reasons for placement of the URS boundary.

The second section, “Downstream Interfaces,” discusses the systems that interface with the subject system, that could potentially be pressurized by reactor pressure passed through (downstream) the subject system. Each downstream system is dispositioned as being either not applicable for URS upgrading or applicable and the topic of another Attachment 3MA section.

The third section, “Upgraded Components,” provides a detailed listing of the components upgraded to the URS design pressure. Also, to indicate some components were not inadvertently overlooked, some components are shown as “No change.” The listings are grouped in sections that describe a particular pressure travel path. This grouping may include more than the system of the subject section to detail the path to the tank or sink in which the pressure is dissipated after crossing the last closed valve at the URS boundary.

3MA.2 Residual Heat Removal System

3MA.2.1 Upgrade Description

The RHR System pump suction piping was low pressure and has been upgraded to the URS design pressure. The RHR has two suction sources, one from the suppression pool and the other from the RPV as used for shutdown cooling. The suction piping also includes the keep-fill pump and its piping.

The URS boundary was terminated at the last valve before the suppression pool, which is valve E11-F001. The suppression pool is a large structure, designed to 0.310 MPaG and impractical to upgrade to the URS design pressure.

The other suction branch to the RPV is not a URS boundary because it interfaces to the high pressure RPV. The only portions of the RHR System that are not upgraded to the URS design pressure is unobstructed piping to the suppression pool.

3MA.2.2 Downstream Interfaces

Other systems are listed below that interface with RHR and could possibly be exposed to reactor pressure. A description of the interface location and a statement of its applicability to ISLOCA is given.

- 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

The Fire Protection System and the fire truck connections provide water for the Alternating Current (AC) Independent Water Addition piping of RHR loops B and C upstream of the RPV injection lines, wetwell and drywell spray lines, and spent fuel pool lines. The Fire Protection System piping is designed for 1.37 MPaG and each line is protected from over pressure by two locked closed block and bleed valves, RHR-F101B/C and RHR-F102B/C, and drain pipes between these valves which are vented to the HCW sump in the Reactor Building.

- 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.

RESIDUAL HEAT REMOVAL SYSTEM, Tier 2 Figure 5.4-10, Sheets 1 through 7.

RHR Subsystem A suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	RHR Pump C001A	3.43 MPaG, 182°C, 3B, As	No change
	450A-RHR-002 Pipe	2.82 MPaG, 182°C, 3B, As	Was 1.37 MPaG
	20A-RHR-701 Pipe	2.82 MPaG, 182°C, 3B, As	Was 1.37 MPaG
	20A-RHR-F701A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX002A Press.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-D002A Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-700 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F700A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PI001A Press.I	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-018 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F026A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-F001A MO Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	450A-RHR-001 Pipe	0.310 MPaG, 104°C,3B,As	No change
	450A-RHR-D001A Suct.Str.	0.310 MPaG, 104°C,3B,As	No change

RHR Subsystem A suction piping from the reactor pressure vessel.

Press./Temp./Design/ Reference	Components	Seismic Class	Remarks
Sheet 3	350A-RHR-011 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	350A-RHR-F012A MO Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-032 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-F042A Rel.Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-707 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F712A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PT009A Press.T	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	350A-RHR-011 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	* 20A-RHR-504 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	* 20A-RHR-F508A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-030 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 100A-RHR-031 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 100A-RHR-F041A Check V	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 100A-RHR-F040A Valve.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

* To LCW funnel drain to LCW Sump.
** To MUW (Condensate) Stem interface.

RHR Subsystem A discharge fill pump suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	40A-RHR-C002A Pump	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-015 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-F022A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-D008A Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-708 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F713A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX010APress.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

	25A-RHR-017 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
	25A-RHR-F025A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-D009A RO	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem A discharge from relief valves and test line valve directly to the suppression pool without restriction.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 3	250A-RHR-008 Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-025 Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-014 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-037 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-033 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-021 Pipe	0.310 MPaG, 104°C,3B,As	No change
Sheet 2	250A-RHR-008 Pipe	0.310 MPaG, 104°C,3B,As	No change
	Suppression Pool		

RHR Subsystem A flushing line interface at branch discharging to feedwater.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	100A-MUWC-134 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 2	100A-RHR -F032A Valve	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -026 Pipe	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -F033A Check V.	3.43 MPaG, 182°C,3B,As	No change

RHR Subsystem A flushing line interface at suction shutdown branch from RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	100A-MUWC-133 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 3	100A-RHR -F040A Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -031 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -F041A Check V.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem B suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 4	RHR Pump C001B	3.43 MPaG, 182°C,3B,As	No Change
	450A-RHR-102 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-731 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F701B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX002B Press.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-D002B Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-730 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F700B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PI001B Press.I	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-124 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F026B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-F001B MO Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	450A-RHR-101 Pipe	0.310 MPaG, 104°C,3B,As	No change

	450A-RHR-D001B Suct.Str.	0.310 MPaG, 104°C,3B,As	No change

RHR Subsystem B suction piping from the reactor pressure vessel.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 4	350A-RHR-111 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	350A-RHR-F012B MO Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-139 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-F042B Rel.Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-737 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F712B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PT009B Press.T	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	350A-RHR-111 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	* 20A-RHR-534 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	* 20A-RHR-F508B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-137 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 300A-RHR-114 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	** 300A-RHR-F016B Valve LC	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	*** 100A-RHR-138 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	*** 100A-RHR-F041B Check Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	*** 100A-RHR-F040B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

* To LCW funnel drain to LCW Sump. ** To FPC System interface.
*** To MUW (Condensate) System interface.

RHR Subsystem B discharge fill pump suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 4	40A-RHR-C002B Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-121 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-F022B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	40A-RHR-D008B Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-738 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F713B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX010BPress.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-123 Pipe	2.82 MPaG, 182°C182°C,3B,As	Was 1.37 MPaG
	25A-RHR-F025B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	25A-RHR-D009B RO	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem B flushing line interface at branch discharging to RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 1	100A-MUWC-137 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 5	100A-RHR -F032B Valve	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -132 Pipe	3.43 MPaG, 182°C,3B,As	No change
	100A-RHR -F033B Check V.	3.43 MPaG, 182°C,3B,As	No change

RHR Subsystem B flushing line interface at suction of shutdown branch from RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks

Sheet 1	100A-MUWC-136 Pipe	1.37 MPaG, 66°C,4D,B	No change
Sheet 2	100A-RHR -F040B Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 2	100A-RHR -138 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	100A-RHR -F041B Check V.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG

RHR Subsystem C suction piping from the suppression pool.

(Moved to RHR Subsystem C section)

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 6	RHR Pump C001C	3.43 MPaG, 182°C,3B,As	No change
	450A-RHR-202 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-761 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F701C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX002C Press.Pt	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-D002C Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-760 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F700C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PI001C Press.I	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-225 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F026C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-F001C MO Vlv	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	450A-RHR-201 Pipe	0.310 MPaG, 104°C,3B,As	No change
	450A-RHR-D001C Suct. Str.	0.310 MPaG, 104°C,3B,As	No change

RHR Subsystem B discharge from relief valves and test line valve directly to the suppression pool without restriction.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 4	250A-RHR-109 Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-131 Pipe	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-120 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-145 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-140 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-127 Pipe	0.310 MPaG, 104°C,3B,As	No change
Sheet 2	250A-RHR-109 Pipe	0.310 MPaG, 104°C,3B,As	No change
	Suppression Pool		

RHR Subsystem B interface with Radwaste System.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 4	150A-RHR-129 Pipe	3.43 MPaG, 182°C,3B,As	No change
	150A-LCW-F006 Valve	2.82 MPaG, 66°C,4D,B	Was 0.981 MPaG
	150A-LCW-CS Pipe	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-CS Pipe	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-CS Valve LO	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-CS AO Valve	0.981 MPaG, 66°C,4D,B	No change
	* LCW Collector Tank A	0 MPaG, 66°C,4D,B	No change
	200A-LCW-CS Valve LO	0.981 MPaG, 66°C,4D,B	No change
	200A-LCW-CS AO Valve	0.981 MPaG, 66°C,4D,B	No change
	*LCW Collector Tank B	0 MPaG, 66°C,4D,B	No change

* Each LCW collector tank is served by the HVAC tank vent system exhausting tank air through filter to

RW Stack.

RHR Subsystem C suction piping from the reactor pressure vessel.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks

RHR Subsystem B outdoor fire truck connection in RHR pump discharge pipe to RPV.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 7	100A-RHR -F103B Valve	2.82 MPaG, 66°C,7E,C	New Item
	100A-RHR -F104B Check V.	2.82 MPaG, 66°C,7E,C	New Item
	100A-RHR - Pipe	2.82 MPaG, 66°C,7E,C	New Item
	100A-RHR - Pipe	2.82 MPaG, 66°C,7E,C	New Item
	100A-RHR -F100B Check V.	2.82 MPaG, 66°C,7E,C	New Item
	100A-RHR -F101B Key Lock V.	3.43 MPaG, 182°C,3B,As	New Item
	100A-RHR - Pipe	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR - Pipe	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR -F790B Globe V.	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR -PI-099B Press I	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR - Pipe	3.43 MPaG, 182°C,3B,As	New Item
	* 20A-RHR -F592B Globe V. LO	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR - Pipe	3.43 MPaG, 182°C,3B,As	New Item
	** 20A-RHR -F591B Globe V. NC	3.43 MPaG, 182°C,3B,As	New Item
	100A-RHR -F102B Key Lock V.	3.43 MPaG, 182°C,3B,As	New Item
	20A-RHR -FE-100B Flow El.	3.43 MPaG, 182°C,3B,As	New Item
	*** 300A-RHR -105 Pipe	3.43 MPaG, 182°C,3B,As	No change

* Funnel drain to LCW sump in Reactor Building.

** Test valve.

*** Injection pipe to RPV at outboard isolation valve MO F-005B.

RHR Subsystem C suction piping from the suppression pool.

Reference	Components	Press./Temp./Design/ Seismic Class	Remarks
Sheet 6	RHR Pump C001C	3.43 MPaG, 182°C,3B,As	No change
	450A-RHR-202 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-761 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F701C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PX002C Press.Pt.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-D002C Temp.Str.	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-760 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-F700C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	20A-RHR-PI001C Press.I	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-225 Pipe	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	50A-RHR-F026C Valve	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
	450A-RHR-F001C MO Vlv	2.82 MPaG, 182°C,3B,As	Was 1.37 MPaG
Sheet 2	450A-RHR-201 Pipe	0.310 MPaG, 104°C,3B,As	No change
	450A-RHR-D001C Suct. Str.	0.310 MPaG, 104°C,3B,As	No change
	100A-RHR-221 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-246 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-241 Pipe	0.310 MPaG, 104°C,3B,As	No change
	50A-RHR-228 Pipe	0.310 MPaG, 104°C,3B,As	No change

conditions during a 2 hour SBO event (for which HVAC systems will not be available) will not exceed the envelope of conditions used to qualify the RCIC equipment. These evaluations will be documented in an RCIC Two Hour Station Blackout Evaluation. Auxiliaries have the capability to operate for a period of 8 hours. Analyses to demonstrate this non-design basis capability utilize realistic, best-estimate assumptions and analysis methods. See Subsection 5.4.15.2 for COL license information requirements.

During loss of AC power, the RCIC System, when started at water Level 2, is capable of preventing water level from dropping below the level which ADS mitigates (Level 1). This accounts for decay heat boiloff and primary system leakages.

Following a reactor scram, steam generation will continue at a reduced rate due to the core fission product decay heat. At this time, the turbine bypass system will divert the steam to the main condenser, and the feedwater system will supply the makeup water required to maintain reactor vessel inventory.

In the event that the reactor vessel is isolated and the feedwater supply unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat. Upon reaching a predetermined low level, the RCIC System will be initiated automatically. The turbine-driven pump will supply demineralized makeup water from (1) the condensate storage tank (CST) to the reactor vessel and (2) the suppression pool. Seismically installed level instrumentation is provided for automatic transfer of the water source with manual override from CST to suppression pool on receipt of either a low CST water level or

The RCIC System shall initiate and discharge, within 30 seconds, a specified constant flow into the reactor vessel over a specified pressure range. The RCIC water discharge into the reactor vessel varies between a temperature of 10°C up to and including a temperature of 77°C during design conditions and up to 121°C during beyond design basis events (e.g. Extended Station Blackout). The mixture of cool RCIC water and the hot steam does the following:

reactor decay heat. This will result in a rise in pool water temperature. RHR heat exchangers are used to maintain pool water temperature within acceptable limits by cooling the pool water.

5.4.6.1.1 Residual Heat and Isolation

5.4.6.1.1.1 Residual Heat

~~The RCIC System shall initiate and discharge, within 30 seconds, a specified constant flow into the reactor vessel over a specified pressure range. The RCIC water discharge into the reactor vessel varies between a temperature of 10°C up to and including a temperature of 77°C. The mixture of the cool RCIC water and the hot steam does the following:~~

(1) Quenches steam.

(2) Removes reactor residual heat.

5.4.6.2.5.3 Test Mode

A design functional test of the RCIC System may be performed during normal plant operation by drawing suction from the suppression pool and discharging through a full flow test return line back to the suppression pool. The discharge valve to the vessel remains closed during test mode operation. The system will automatically return from test to operating mode if system initiation is required and the flow will be automatically directed to the vessel.

5.4.6.2.5.4 Limiting Single Failure

The most limiting single failure with the RCIC System and its HPCF system backup is the failure of HPCF. With an HPCF failure, if the capacity of the RCIC System is adequate to maintain reactor water level, the operator shall follow Subsection 5.4.6.2.5.2. However, if the RCIC capacity is inadequate, Subsection 5.4.6.2.5.2 still applies. Additionally, the operator may initiate the ADS described in Subsection 6.3.2.2.2.

5.4.6.3 Performance Evaluation

The analytical methods and assumptions in evaluating the RCIC System are presented in Chapter 15 and Appendix 15A. The RCIC System provides the flows required from the analysis (Figure) within a 30 second interval based upon considerations noted in Subsection 5.4.6.2.4.

5.4.6.4 Preoperational Testing

The preoperational and initial startup test program for the RCIC System is presented in Chapter 14.

5.4.7 Residual Heat Removal System

Evaluations of the Residual Heat Removal (RHR) System against the applicable General Design Criteria (GDC) are provided in Subsections 3.1.2 and 5.4.7.1.4.

5.4.7.1 Design Basis

The RHR System is composed of three electrical and mechanical independent divisions designated A, B, and C. Each division contains the necessary piping, pumps, valves and heat exchangers. In the low pressure flooder mode, suction is taken from the suppression pool and injected into the vessel outside the core shroud (via the feedwater line on Division A and via the low pressure flooder subsystem discharge return line on Divisions B and C).

The RHR System provides two independent containment spray cooling systems (on loops B and C), each having a common header in the wetwell and a common spray header in the drywell and sufficient capacity for containment depressurization.

Shutdown cooling suction is taken directly from the reactor via three shutdown cooling suction nozzles on the vessel. Shutdown cooling return flow is via the feedwater line on loop A and via low pressure flooder subsystem discharge return lines on loops B and C.

Connections are provided to the upper pools on two loops to return shutdown cooling flow to the upper pools during normal refueling activities if necessary. These connections also allow the RHR System to provide additional fuel pool cooling capacity as required by the Fuel Pool Cooling System during the initial stages of the refueling outage.

~~The RHR System provides an AC-independent water addition subsystem which consists of piping and manual valves connecting the fire protection system to the RHR pump discharge line on loop C downstream of the pump's discharge check valve. This flow path allows for injection of water into the reactor vessel and the drywell spray during severe accident conditions in which all AC power and all ECCS pumps are unavailable. Additionally, an external hookup outside the reactor building for connection of a fire truck pump to an alternate water source is provided.~~

As shown in Table 5.4-4, the RHR heat exchanger primary (tube) side design pressure is 3.43 MPaG and the secondary (shell) side design pressure is 1.37 MPaG. This pressure distribution is acceptable for the following reasons:

(1) Heat exchanger primary side leakage is accommodated by the surge tank of the pump loop of the reactor building cooling water system. The inlet to the secondary side of

The RHR System has two AC-independent water addition subsystems which consist of piping and manual valves connecting the fire protection system to the RHR pump discharge lines on loops B and C downstream of each main pump's discharge check valve. These flow paths allow for injection of water into the reactor vessel, wetwell or drywell spray, or the spent fuel pool during severe accident conditions in which all AC power and all ECCS pumps are unavailable. Additionally, hookups are provided external to the reactor building for connection of a fire truck pump and alternate water source to each subsystem. The hookups are on different faces of the reactor building and separated by more than 40 meters.

tube side of the heat exchanger, whereas, prior BWRs had the reactor water flowing through the shell side. The primary purpose for the change was to reduce radiation buildup in the heat exchanger by providing a more open geometry flow path through the center of the tubes, as apposed to the shell side construction of spacers, baffles, and low flow velocity locations, which can provide places for radioactive slug to accumulate. Also, the ABWR does not have a steam condensing mode, which needed reactor water or steam on the shell side. Tubes can accommodate a higher design pressure much more easily and effectively than the shell's large cylindrical structure; therefore, the shell can take advantage of the reactor building cooling water system's lower design pressure.

5.4.7.1.1 Functional Design Basis

The RHR System provides the following four principal functions:

(1) Core cooling water supply to the reactor to compensate for water loss beyond the normal control range from any cause up to and including the design basis (LOCA).

(2) Suppression pool cooling to remove heat released to the suppression pool (wetwell), as necessary, following heat inputs to the pool.

The AC-independent water addition mode (Alternating Current independent) of the RHR System provides a means for introducing water from the Fire Protection System (FPS) directly into the reactor pressure vessel, to the drywell or wetwell spray header, or to the spent fuel pool under degraded plant conditions when AC power is not available from either onsite or offsite sources. The RHR System provides the piping and valves which connect the FPS piping with RHR loops B and C pump discharge piping. The manual valves in these lines permit adding water from the FPS to the RHR System if the RHR System is not operable. The primary means for supplying water through this connection is by use of the diesel-driven pump in the FPS. A backup to this pump is provided by connections on the outside of the reactor building at grade level, which allows hookup of the ACIWA to fire truck pumps.

Figure 5.4-10 shows the connections from either the diesel-driven pump or the fire truck to RHR system loops B and C. The connections to the diesel-driven pump (FPS) are adjacent to the RHR valve rooms. Opening valves F101B/C and F102B/C allows water to flow from the FPS into the RHR piping. Periodic stroke testing of these valves is required by Table 3.9-8 to ensure valve operability. The fire truck connections are located outside the reactor building at grade level, on different faces of the building. Both connections to the RHR system are protected by a check valve (F100B/C and F104B/C for the diesel-driven pump and the fire truck, respectively) to insure that RCS pressurization does not result in a breach of the injection path. Detailed procedures for the operation of the ACIWA, including operation of the FPS valve in the yard, are required to be developed by the COL applicant. See Section 19.9.7.

5.4.7.1.1.10 AC-Independent Water Addition (ACIWA) Mode

~~The AC-independent water addition mode (Alternating Current independent) of the RHR System provides a means for introducing water from the Fire Protection System (FPS) directly into the reactor pressure vessel, or to the drywell spray header, or to the wetwell spray header under degraded plant conditions when AC power is not available from either onsite or offsite sources. The RHR System provides the piping and valves which connect the FPS piping with the RHR loop C pump discharge piping. The manual valves in this line permit adding water from the FPS to the RHR System if the RHR System is not operable. The primary means for supplying water through this connection is by use of the diesel-driven pump in the FPS. A backup to this pump is provided by a connection on the outside of the reactor building at grade level, which allows hookup of the ACIWA to a fire truck pump.~~

~~Figure 5.4-10 shows the connections from either the diesel-driven pumps or the fire truck to the RHR system. The connections to the diesel-driven pump are in the RHR valve room. Opening valves F101 and F102 allows water to flow from the FPS into the RHR piping. Periodic stroke testing of these valves is required by Table 3.9-8 to ensure valve operability. The fire truck connection is located outside the reactor building at grade level. Both connections to the RHR system are protected by a check valve (F100 and F104 for the diesel-driven pump and the fire truck, respectively) to insure that RCS pressurization does not result in a breach of the injection path. Detailed procedures for the operation of the ACIWA, including operation of the FPS valve in the yard, are required to be developed by the COL applicant. See Section 19.9.7.~~

It is likely that elevated radiation levels may exist in the areas where the valves to align the ACIWA System for vessel injection or drywell spray are located. Preliminary calculations indicate that dose rates could range from 2 to 10R/h in these areas depending on specific piping arrangements, shielding, and SGTS operation. The COL applicant is required to perform dose rate calculations in the ACIWA operating procedures. See Section 19.9.7. If contaminated water were circulated through specific ECCS lines following core damage, the areas where the ACIWA System valves are located would not be accessible. However, it is anticipated that ACIWA System operation will not be required following core damage and subsequent ECCS operation. Under these postulated conditions, operation of the ECCS will obviate the need for ACIWA operation.

5.4.7.1.1.10.4 Containment Performance Without ACIWA

The ACIWA mode of the RHR System provides manual capability to prevent core damage when all emergency core cooling systems are lost. If core damage occurs and heat removal is not recovered, this system increases the time to COPS operation, provides cooling of the seals of the movable penetrations, and provides cooling of the seals of the drywell air space. Without ACIWA, the lower drywell would heat up after core damage and vessel failure until the passive flooder system actuates. Flooder actuation will provide water to the debris in the lower drywell in a similar manner as the ACIWA mode. However, the passive flooder does not add thermal mass to the containment, nor does it have the capability of mitigating suppression pool bypass.

~~Operation of the AC-independent water addition mode is entirely manual. All of the valves which must be opened or closed during fire water addition are located within the same ECCS valve room. The connection to add water using a fire truck pump is located outside the reactor building at grade level.~~

5.4.7.1.2 Design Basis for Isolation of RHR System from Reactor Coolant System

The low pressure portions of the RHR System are isolated from full reactor pressure whenever the primary system pressure is above the RHR System design pressure (see Subsection 5.4.7.1.3

Operation of the AC-independent water addition mode is entirely manual. All of the valves which must be opened or closed during fire water addition are located adjacent to or within the applicable ECCS valve room. The connections to add water using fire truck pumps are located outside the reactor building at grade level on different faces of the building.

Subsection 5.2.5.2.1 (12) and Table 5.4-6].

The RHR pumps are protected against damage from a closed discharge valve by means of automatic minimum flow valves which open on low mainline flow and close on high mainline flow.

5.4.7.1.3 Design Basis for Pressure Relief Capacity

The relief valves in the RHR System are sized on the basis of thermal relief and valve bypass leakage only.

Redundant interlocks prevent opening valves to the low pressure suction piping when the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure.

Overpressure protection is achieved during system operation when the system is not isolated from the reactor coolant pressure. The RHR System is operational and not isolated from the Reactor Coolant System only when the reactor is depressurized. Two modes of operation are applicable: the flooder mode and the shutdown cooling mode. For the flooder mode, the injection valve opens through interlocks only for reactor pressure less than approximately 3.45 MPaG. For the shutdown cooling mode, the suction valves can be opened through interlocks only for reactor pressures less than approximately 0.93 MPaG. Once the system is operating in

5.4.7.2.6 Manual Action

(1) Emergency Mode [Low pressure flooder (LPFL) mode]

Each loop in the subsystem is initiated automatically by a low water level in the reactor vessel or high pressure in the drywell. Each loop in the system can also be placed in operation by means of a Manual Initiation pushbutton switch.

During the LPFL mode, water is initially pumped from the suppression pool and diverted through the minimum flow lines until the injection valve in the discharge line is signalled to open on low reactor pressure. As the injection valve opens on low reactor pressure, flow to the RPV comes from the suppression pool, through the RHR heat exchanger, and the injection valve. This creates a flow signal that closes the minimum flow line.

The system remains in the operating mode until manually stopped by the operator.

(2) Test Mode

Full flow functional testing of the RHR System can be performed during normal plant operation or during plan shutdown by manual operation of the RHR System from the control room. For plant testing during normal plant operation, the pump is started and the return line to the suppression pool is opened. A reverse sequence is used to terminate this test. Upon receipt of an automatic initiation signal while in the flow testing mode, the RHR System is returned to automatic control.

(3) Suppression Pool Cooling

The suppression cooling (SPC) mode of RHR can be initiated and stopped manually. The SPC mode removes heat released into the suppression pool, as necessary, following heat additions to the pool. During this mode of operation, water is pumped from the suppression pool through the RHR heat exchangers, and back to the suppression pool. This RHR SPC mode is also initiated automatically as described in Subsection 5.4.7.1.1.5.

(4) Wetwell and Drywell Spray Cooling

Two of the RHR loops provide containment spray cooling subsystems. Each subsystem provides both wetwell and drywell spray cooling. This subsystem provides steam condensation and primary containment atmospheric cooling following a LOCA by pumping water from the suppression pool, through the heat exchangers and into the wetwell and drywell spray spargers in the primary containment. The drywell spray mode is initiated by manual operator action post-LOCA in the presence of high drywell pressure. The wetwell spray mode is initiated as required by manual operator action. If the wetwell spray is operated without

drywell spray, it will be in conjunction with suppression pool cooling to achieve rated flow through the RHR heat exchanger for containment cooling. The drywell spray mode is terminated automatically following a LOCA signal as the injection valve opens, and the wetwell spray is terminated automatically by a LOCA signal. Both drywell and wetwell spray can be terminated manually by operator action with no permissive interlocks to be satisfied.

(5) Shutdown Cooling

The Shutdown Cooling Subsystem is manually activated by the operator following insertion of the control rods and normal blowdown to the main condenser. In this mode, the RHR System removes residual heat (decay and sensible) from the reactor vessel water at a rate sufficient to cool it to 60°C within 24 hours after the control rods are inserted. The subsystem can maintain or reduce this temperature further so that the reactor can be refueled and serviced.

Reactor water is cooled by pumping it directly from the reactor shutdown cooling nozzles, through the RHR heat exchangers, and back to the vessel (via feedwater on loop A and via the LPFL Subsystem on the other two loops).

This system is initiated and shut down by manual operator action.

(6) Fuel Pool Cooling

Two of the RHR loops provide supplemental fuel pool cooling during normal refueling activities and any time the fuel pool heat load exceeds the cooling capacity of the fuel pool heat exchangers. For normal refueling activities where the reactor well is flooded and the fuel pool gates are open, water is drawn from the reactor shutdown suction lines, pumped through the RHR heat exchangers and discharged through the reactor well distribution spargers. For 100% core removal, if necessary, water is drawn from the Fuel Pool Cooling (FPC) System skimmer surge tanks, pumped through the RHR heat exchangers and returned to the fuel pool via the FPC System cooling lines. These operations are initiated and shut down by operator action.

(7) Reactor Well and Equipment Pool Drain

The RHR System provides routing and connections for emptying the reactor well and equipment pool to the suppression pool after servicing. Water is pumped or drained by gravity through the FPC System return lines to the RHR shutdown suction lines and then to the suppression pool.

(8) AC-Independent Water Addition

~~The RHR System is provided with piping and valves which connect the RHR loop C pump discharge piping to the Fire Protection System (FPS) and to a reactor building external fire truck pump hookup. These connections allow for addition of FPS water to the reactor pressure vessel, or the drywell spray header or wetwell spray header during events when AC power is unavailable from both onsite and offsite sources. Operation of the RHR System in the AC-independent water addition mode (Alternating Current-independent) is entirely manual. All valves required to be opened or closed for operation are located within the same loop C ECCS valve room to provide ease of operation.~~

5.4.7.3 Performance Evaluation

RHR System performance depends on sizing its heat exchanger and pumping flow rate characteristics with enough capacity to satisfy the most limiting events. The worst case transient

The RHR System is provided with piping and valves which separately connect RHR loops B and C pump discharge piping to the Fire Protection System (FPS) and to the reactor building external fire truck pump hookups. These connections allow for addition of FPS water to the reactor pressure vessel, the drywell or wetwell spray header, or the spent fuel pool during events when AC power is unavailable from both onsite and offsite sources. Operation of the RHR System in the AC-independent water addition mode is entirely manual. All valves required to be opened or closed for operation are located adjacent to or within the respective loop ECCS valve room to provide ease of operation.

5.4

the infinite variety of such curves that is possible due to: (1) clean steam systems that may allow the main condenser to be used as the heat sink when nuclear steam pressure is insufficient to maintain steam air ejector performance; (2) the condition of fouling of the exchangers; (3) operator use of one or two cooling loops; (4) coolant water temperature; and (5) system flushing time. Since the exchangers are designed for the fouled condition with relatively high service water temperature, the units have excess capability to cool when first cut in at high vessel temperature. Total flow and mix temperature must be controlled to avoid exceeding a 55°C/hour cooldown rate. See Subsection 5.4.7.1.1.7 for minimum shutdown time to reach 100°C.

5.4.7.3.2 Worst Case Transient

Several limiting events were considered for RHR heat exchanger sizing. Those events were:

(1) Feedwater line break (FWLB)

(2) Main steamline break

(3) Inadvertent opening of a relief valve

(4) Normal shutdown

(5) Emergency shutdown

(6) ATWS

Table 5.4-2 Design Parameters for RCIC System Components

(1) RCIC Pump Operation (C001)		
Flow rate	Injection flow – 182 m^3/h	
	Cooling water flow – 4 to 6 m^3/h	
	Total pump discharge – 188 m^3/h (includes no margin for pump wear)	
Water temperature range	10° to 60°C, continuous duty 40° to 77°C, short duty	
NPSH	7.3m minimum	
Developed head	900m at 8.22 MPaA reactor pressure	
	186 m at 1.14 MPaA reactor pressure	
Maximum pump shaft power	675 kW at 900m developed head 125 kW at 186m developed head	
Design pressure	11.77 MPaG	
(2) RCIC Turbine Operation (C002)		
	High Pressure Condition	Low Pressure Condition
Reactor pressure (saturated temperature)	8.19 MPaA	1.14 MPaA
Steam inlet pressure	8.12 MPaA, minimum	1.03 MPaA, minimum
Turbine exhaust pressure	0.11 to 0.18 MPaA, maximum	0.11 to 0.18 MPaA, maximum
Design inlet pressure	8.62 MPaG at saturated temperature	
Design exhaust pressure	8.62 MPaG at saturated temperature	
(3) RCIC leakoff orifices (D017, D018)	Sized for 3.2 mm diameter minimum to 4.8 mm diameter maximum	
Flow element (FE007)		
Flow at full meter differential pressure	250 m^3/h	
Normal temperature	10 to 77°C	
System design pressure/temperature	8.62 MPaG/302°C	
Maximum unrecoverable loss at normal flow	0.031 MPa	
Installed combined accuracy (Flow element, Flow transmitter and Flow indicator)	±2.5% at normal flow and normal	

40°C to 121°C during beyond design basis events (e.g. Extended Station Blackout)

Table 5.4-2 Design Parameters for RCIC System Components (Continued)

Vacuum breaker check valves (F054 & F055)	Full flow and open with a minimum pressure drop (less than 3.92 kPa across the valves)
Steam inlet drain pot system isolation (F040 & F041)	These valves allow for drainage of the steam inlet drain pot and must operate against a differential pressure of 8.12 kPa
Steam inlet trip bypass valve (F058)	This valve bypasses the trap D008 and must operate against a differential pressure of 8.12 kPa
Cooling loop shutoff valve (F012)	This valve allows water to be passed through the auxiliary equipment coolant loop and must operate against a differential pressure of 9.65 MPa
Pump test return valve (F009)	This valve allows water to be returned to the suppression pool during RCIC system test and must operate against a differential pressure of 9.65 MPa
Steam supply bypass valve (F045)	Open and/or close against full differential of 8.12 MPa within 5 seconds
Turbine exhaust check valve (F038)	Capable of with standing impact loads due to “flapping” during startup.
Vacuum pump discharge isolation valve (F047)	Open and/or close against 0.314 MPa differential pressure at a temperature of 170°C.
Vacuum pump discharge check valve (F046)	Located at the highest point in the line.

(5) Instrumentation – For instruments and control definition, refer to Subsection 7.4.1.1.

(6) Condensate Storage Requirements

Total reserve storage for RCIC and HPCF System is 570 m^3.

(7) Piping RCIC Water Temperature
The maximum water temperature range for continuous system operation shall not exceed 60°C; however, due to potential short-term operation at higher temperatures, piping expansion calculations were based on ~~77°C~~. 121°C

(8) Turbine Exhaust Vertical Reactor Force
The turbine exhaust sparger is capable of withstanding a vertical pressure unbalance of 0.137 MPa. This pressure unbalance is due to turbine steam discharge below the suppression pool water level.

(9) Ambient Conditions	Relative Temperature	Humidity(%)
Normal plant operation	10 to 40°C	10 to 90

(10) Suction Strainer Sizing

The suppression pool suction shall be sized so that:

(a) Pump NPSH requirements are satisfied when strainer is 50% plugged; and particles over 2.4 mm diameter are restrained from passage into the pump and feedwater sparger.

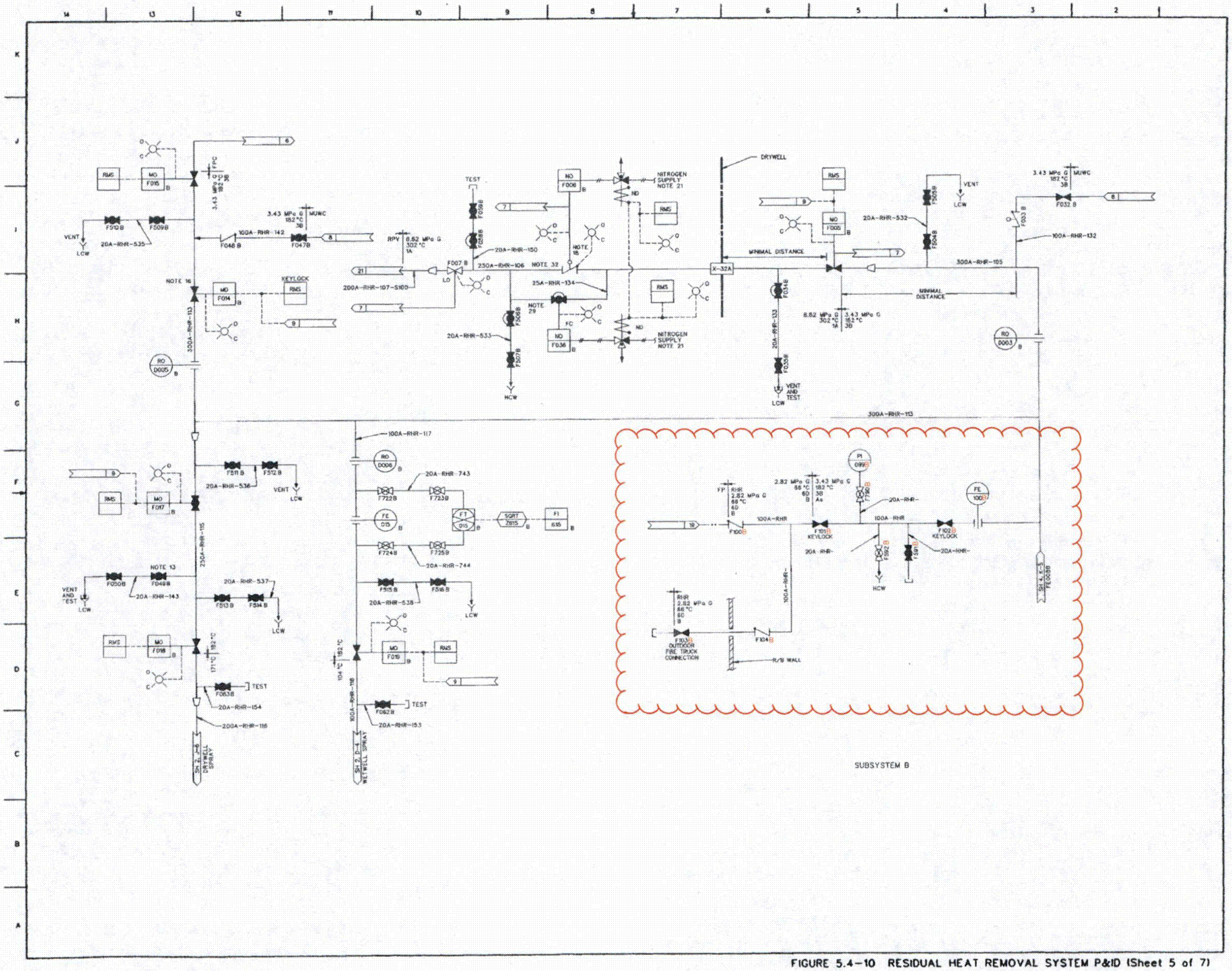


FIGURE 5.4–10 RESIDUAL HEAT REMOVAL SYSTEM P&ID (Sheet 5 of 7)

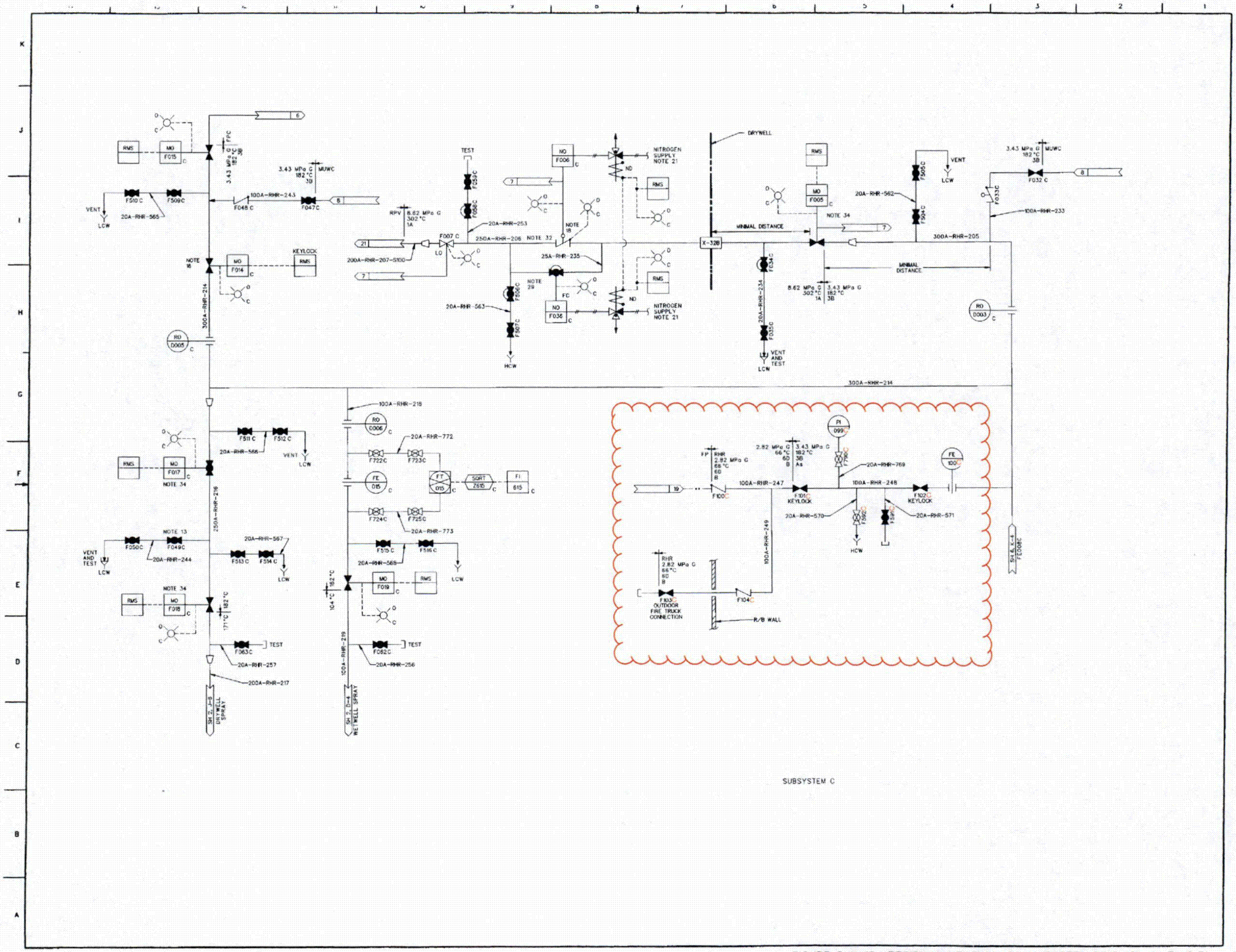


FIGURE 5.4–10 RESIDUAL HEAT REMOVAL SYSTEM P&ID (Sheet 7 of 7)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 ECCS – Operating

LCO 3.5.1 Each ECCS subsystem and the Automatic Depressurization System (ADS) function of eight safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves and RCIC are not required to be OPERABLE with reactor steam dome pressure ≤ 0.343 MPaG for ADS and ≤ 1.03 MPaG for RCIC.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two ECCS subsystems inoperable provided RCIC is OPERABLE.	A.1	Restore ECCS subsystem(s) to OPERABLE status.	14 days
B. RCIC inoperable. OR RCIC and any one other ECCS subsystem inoperable.	B.1.1	Verify the CTG is functional by verifying the CTG starts and achieves steady state voltage and frequency within 2 minutes.	7 days
	AND		
	B.1.2	Verify the CTG circuit breakers are capable of being aligned to each of the ESF buses.	7 days AND Once per 8 hours thereafter
	OR		
	B.2	Verify the ACIWA mode of RHR ~~(C)~~ B or C subsystem is functional.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	AND		
	B.3	Restore ECCS subsystem(s) to OPERABLE status.	14 days
C. RCIC and any other two ECCS subsystems inoperable provided at least one HPCF subsystem is OPERABLE.	C.1.1.1	Verify the CTG is functional by verifying the CTG starts and achieves steady state voltage and frequency within 2 minutes.	72 hours
		AND	
	C.1.1.2	Verify the CTG circuit breakers are capable of being aligned to each of the ESF buses.	72 hours AND Once per 8 hours thereafter
	OR		
	C.1.2	Verify the ACIWA mode of RHR ~~(C)~~ B or C subsystem is functional.	72 hours
	AND		
	C.2	Restore one ECCS subsystem to OPERABLE status.	7 days
D. Any three ECCS subsystems inoperable provided RCIC is OPERABLE.	D.1	Restore one ECCS subsystem to OPERABLE status.	3 days

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Containment Spray

LCO 3.6.2.4 Two RHR containment spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION		REQUIRED ACTION	COMPLETION TIME
A. One RHR containment spray subsystem inoperable.	A.1	Verify the ACIWA mode of RHR ~~(C)~~ B or C subsystem is functional.	7 days
	<u>AND</u>		
	A.2	Restore RHR containment spray subsystem to OPERABLE status.	14 days
B. Two RHR containment spray subsystems inoperable.	B.1	Restore one RHR containment spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Times not met.	C.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 ECCS – Operating

BASES

BACKGROUND

The ECCS is designed, in conjunction with the primary and secondary containment, to limit the release of radioactive materials to the environment following a loss of coolant accident (LOCA). The ECCS directs water to both inside and outside the core shroud to cool the core during a LOCA. The ECCS network is composed of the High Pressure Core Flooder (HPCF) System, the Reactor Core Isolation Cooling (RCIC) System, and the low pressure core flooder (LPFL) mode of the Residual Heat Removal (RHR) System. The ECCS also consists of the Automatic Depressurization System (ADS). The suppression pool provides the required source of water for the ECCS. Although no credit is taken in the safety analyses for the condensate storage tank (CST), it is capable of providing a source of water for both the RCIC System and the two HPCF subsystems.

On receipt of an initiation signal, ECCS pumps automatically start; simultaneously the system aligns, and the pumps inject water, taken either from the CST or suppression pool, into the Reactor Coolant System (RCS) as RCS pressure is overcome by the discharge pressure of the ECCS pumps. Although the system is initiated, ADS action is delayed, to allow time for confirmation of the initiating signal. The discharge pressure of the HPCF pumps exceeds that of the RCS, and the pumps inject coolant into the flooding sparger above the core. Once the steam driven RCIC turbine has accelerated, the RCIC pump discharge pressure exceeds that of the RCS and injects coolant into the reactor pressure vessel (RPV) via one of the feedwater lines. If the break is small, RCIC or either of the HPCF pumps will maintain coolant inventory, as well as vessel level, while the RCS is still pressurized. If the RCIC and HPCFs fail, they are backed up by ADS in combination with the LPFL. In this event, the ADS timed sequence would be allowed to time out and open the selected safety/relief valves (S/RVs), depressurizing the RCS and allowing the LPFL to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure drops rapidly, and the HPCF and LPFL subsystems cool the core.

(continued)

BACKGROUND (Continued)

Water from the break returns to the suppression pool where it is used again and again. Water in the suppression pool is circulated through a heat exchanger cooled by the Reactor Building Cooling Water (RCW) System. The ECCS network is effective in cooling the core regardless of the size or location of the piping break.

Apart from its ECCS function the RCIC System is also designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. Under these conditions, the HPCF and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 11 are satisfied.

All ECCS subsystems are designed to ensure that no single active component failure will prevent automatic initiation and successful operation of the minimum required ECCS subsystems.

The ECCS injection systems are arranged in three separate divisions each comprised of a high pressure and low pressure subsystem. ECCS Division 1 consists of the RCIC system and LPFL-A. ECCS Division 2 consists of HPCF-B and LPFL-B. ECCS Division 3 consists of HPCF-C and LPFL-C.

LPFL is an independent operating mode of the RHR System. There are three LPFL subsystems. Each LPFL subsystem (Ref. 2) consists of a motor driven pump, a heat exchanger, piping, and valves to transfer water from the suppression pool to the RPV. Each LPFL subsystem has its own suction and discharge piping. Each LPFL subsystem takes suction from the suppression pool. LPFL subsystems B and C have dedicated discharge nozzles to the RPV that connect to flooding spargers in the vessel annulus area outside the core shroud. LPFL subsystem A discharges to one of the main feedwater injection lines and thus also supplies coolant to the vessel annulus area outside the core shroud via the feedwater sparger. The LPFL subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, each LPFL pump is automatically started approximately 10 seconds after electrical power is available. When the RPV pressure drops sufficiently, LPFL flow to the RPV begins. RHR System valves in the LPFL flow

(continued)

BACKGROUND
(Continued)

path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the RPV. A discharge test line is provided to route water from and to the suppression pool to allow testing of each LPFL pump without injecting water into the RPV.

The HPCF System is comprised of two separate subsystems. Each HPCF subsystem (Ref. 1) consists of a single motor driven pump, a flooder sparger above the core, and piping and valves to transfer water from the suction source to the sparger. Suction piping is provided from the CST and the suppression pool. Pump suction is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCF System. The HPCF System is designed to provide core cooling over a wide range of RPV pressures, (0 to 8.12 MPaD), vessel to the air space of the compartment containing the water source for the pump suction. Upon receipt of an initiation signal, the HPCF pumps automatically start (when electrical power is available) and valves in the flow path begin to open. Since the HPCF System is designed to operate over the full range of RPV pressures, HPCF flow begins as soon as the necessary valves are open. A full flow test line is provided to route water from and to the CST to allow testing of the HPCF System during normal operation without injecting water into the RPV.

The RCIC System (Ref. 1) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from main steam line B, upstream of the inboard main steam line isolation valve.

(continued)

BASES

BACKGROUND (Continued)

The RCIC System is designed to provide core cooling for a wide range of reactor pressures, 1.03 MPaG to 8.12 MPaG. Upon receipt of an initiation signal, the RCIC

AC Independent water Addition (ACIWA) mode of RHR (References 13 and 14). If RCIC is inoperable, water can be injected into the RPV either by powering other ECCS subsystems from the CTG or by the Fire Protection System (FPS) using one of the loops of the ACIWA mode of RHR.

suppression pool to allow testing of the RCIC System during normal operation without injecting water into the RPV. For the station black out scenario, where all AC power from the offsite AC circuits and from the standby diesel generators are assumed to be lost, RCIC is designed to provide makeup water to the RPV. Diverse alternatives to RCIC are provided by the Combustion Turbine Generator (CTG) and the ~~AC-Independent Water Addition (ACIWA) mode of RHR(C) (References 13 and 14). If RCIC is inoperable, water can be injected into the RPV either by powering other ECCS subsystems from the CTG or by the Fire Protection System (FPS) using the ACIWA mode of RHR(C).~~

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed or RPV pressure is greater than the LPFL pump discharge pressures following system initiation. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the ECCS discharge line "keep fill" systems are designed to maintain all pump discharge lines filled with water.

The ADS (Ref. 1) consists of 8 of the 18 S/RVs. It is designed to provide depressurization of the primary system during a small break LOCA if RCIC and HPCF fail or are unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (LPFL), so that these subsystems can provide core cooling. Each ADS valve is supplied with pneumatic power from either its own dedicated accumulator located in the drywell, or from the atmospheric control system (ACS) directly when pneumatic power from the accumulators is not needed. The ACS also supplies the nitrogen (at pressure) necessary to assure the ADS accumulators remain charged for use in emergency actuation.

(continued)

BASES

ACTIONS

A.1, B.1.1, B.1.2, B.2, and B.3

With one or two ECCS subsystem(s) inoperable provided RCIC is OPERABLE (Condition A), the inoperable subsystem(s) must be restored to OPERABLE status within 14 days. If RCIC is

each of the ESF buses or one of the loops of the AC-Independent Water Addition (ACIWA) mode of RHR is verified to be functional within 7

subsystem(s) can be restored to OPERABLE status within 14 days provided the Combustion Turbine Generator (CTG) is verified, initially within 7 days and once per 8 hours thereafter, to be functional and capable of being aligned to ~~each of the ESF buses or the AC-Independent Water Addition (ACIWA) mode of RHR(C) is verified to be functional within 7~~ days. In these Conditions, the remaining OPERABLE subsystems provide more than adequate core cooling during a LOCA. However, overall ECCS reliability is reduced; and a single failure impacting one or more of the remaining OPERABLE subsystems concurrent with a LOCA would result in degraded ECCS performance and reduced margins to 10 CFR 50.46 acceptance criteria. Nevertheless, even given the worse case single failure concurrent with a LOCA initiated from this Condition, there will always be at least one ECCS subsystem available to inject water into the RPV. (For the special case of an LPFL-A line break and failure of a diesel generator, the CTG would be available to provide emergency electrical power to the ECCS pumps.) Additional analyses of limiting design basis scenarios demonstrate that in such cases 10 CFR 50.46 acceptance criteria will still be met. Furthermore, results of PRA sensitivity studies performed (References 9 and 15) for Condition A show that this

compensate for RCIC's inoperability, if one of the loops of the ACIWA mode of RHR is verified to be functional, the Fire Protection System (FPS) can be used to inject water into the RPV during a station blackout with the RPV sufficiently depressurized. Loop B(C) of ACIWA is verified to be functional by

each of the ESF buses (LCO 3.8.1), other ECCS subsystems can be powered by the CTG during a station blackout to compensate for RCIC's inoperability. Alternatively, to ~~compensate for RCIC's inoperability, if the ACIWA mode of RHR(C) subsystem is verified to be functional, the Fire Protection System (FPS) can be used to inject water into the RPV during a station blackout with the RPV sufficiently depressurized. The ACIWA is verified to be functional by~~

(continued)

ACTIONS
(Continued)

A.1, B.1.1, B.1.2, B.2, and B.3 (continued)

~~stroking one complete cycle of each of the two manual valves in the FPS connection to RHR(C) injection line, by starting the FPS diesel driven fire pump and verifying that the FPS header pressure is maintained, and by stroking one complete cycle of the RHR(C) subsystem injection valve using its handwheel~~.

stroking one complete cycle of each of the two manual valves in the FPS connection to the RHR Loop B(C) injection line, by starting the FPS diesel-driven fire pump and verifying that the FPS header pressure is maintained, and by stroking one complete cycle of the RHR Loop B(C) injection valve using its handwheel.

ECCS reliability.

C.1.1.1, C.1.1.2, C.1.2, C.2, D.1 and E.1

With RCIC and any other two ECCS subsystems inoperable, provided at least one HPCF subsystem is OPERABLE, one ECCS subsystem must be restored to OPERABLE status within 7 days. With any three ECCS subsystems inoperable, provided RCIC is OPERABLE, one ECCS subsystem must be restored to OPERABLE status within 3 days. With all three high pressure ECCS subsystems inoperable, at least one high pressure ECCS subsystem must be restored to OPERABLE status within 12 hours. In these Conditions, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA, but the single failure criterion capability for all combinations of systems out of service is not satisfied. Therefore, the Completion Times are limited to 7 days, 3 days and 12 hours, respectively, depending on the combination of ECCS subsystems that are inoperable. Additional analyses of limiting design basis scenarios demonstrate that in such cases 10 CFR 50.46 acceptance criteria will still be met (Ref. 8). Furthermore, results of PRA sensitivity studies performed (References 9 and 15) show that this situation is acceptable from an overall plant risk perspective.

Additionally, for Condition C, where RCIC is inoperable, either the CTG must be verified, within 72 hours, to be

(continued)

BASES

ACTION (continued)

C.1.1.1, C.1.1.2, C.1.2, C.2, D.1 and E.1 (continued)

functional and the circuit breakers are capable of being ~~aligned to each of the ESF buses, or, alternatively, the ACIWA mode of RHR(C) subsystem must be verified to be functional within 72 hours. If the CTG is verified to be functional and capable of being aligned to each of the ESF buses (LCO 3.8.1), other ECCS subsystems can be powered by the CTG during a station blackout to compensate for RCIC's inoperability. If the ACIWA mode of RHR(C) subsystem is verified to be functional, the Fire Protection System (FPS) can be used to inject water into the RPV during a station blackout with the RPV sufficiently depressurized. The ACIWA is verified to be functional by stroking one complete cycle of each of the two manual valves in the FPS connection to RHR(C) injection line, by starting the FPS diesel-driven fire pump and verifying that the FPS header pressure is maintained, and by stroking one complete cycle of the RHR(C) subsystem injection valve using its handwheel. If the CTG or~~ aligned to each of the ESF buses, or, alternatively, one of the loops of ACIWA mode of RHR must be verified to be functional within 72 hours. If the CTG is verified to be functional and capable of being aligned to each of the ESF buses (LCO 3.8.1), other ECCS subsystems can be powered by the CTG during a station blackout to compensate for RCIC's inoperability. If one of the loops of ACIWA mode of RHR is verified to be functional, the Fire Protection System (FPS) can be used to inject water into the RPV during a station blackout with the RPV sufficiently depressurized. Loop B(C) of ACIWA is verified to be functional by stroking one complete cycle of each of the two manual valves in the FPS connection to the RHR Loop B(C) injection line, by starting the FPS diesel-driven fire pump, and verifying that the FPS header pressure is maintained, and by stroking one complete cycle of the RHR Loop B(C) injection valve using its handwheel. If the CTG or ACIWA is not available, the Completion Time for Condition C is limited to 72 hours based on an overall risk perspective.

Since the ECCS availability is reduced relative to Conditions A and B, a more restrictive Completion Time is imposed. The 7 and 3 day Completion Times for Required Actions C.2 and D.1 are based on the low probability of a LOCA occurring during this period and the overall redundancy provided by the ECCS and its continued ability to perform the intended safety function while assuring a return towards full ECCS capability in a reasonable time so as to not significantly impact overall ECCS reliability. The

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.4 Residual Heat Removal (RHR) Containment Spray

BASES

BACKGROUND

The primary containment is designed with a suppression pool so that, in the event of a loss of coolant accident (LOCA) or a rapid depressurization of the reactor pressure vessel (RPV) through the safety/relief valves, steam released from the primary system is channeled through the suppression pool water and condensed without producing significant pressurization of the primary containment. The primary containment is designed so that with the pool initially at the minimum water volume and the worst single failure of the primary containment heat removal systems, suppression pool energy absorption combined with subsequent operator controlled pool cooling will prevent the primary containment pressure from exceeding its design value. However, the primary containment must also withstand a postulated bypass leakage pathway that allows the passage of steam from the drywell directly into the wetwell airspace, bypassing the suppression pool. Some means must be provided to condense steam from the wetwell so that the pressure inside primary containment remain within the design limit. This function is provided by two redundant RHR containment spray subsystems (only RHR subsystems B and C operate in this mode. The purpose of this LCO is to ensure that both subsystems are OPERABLE in applicable MODES.

Each of the two RHR containment spray subsystems contains a pump and a heat exchanger, which are manually initiated and independently controlled. The two subsystems perform the containment spray function by circulating water from the suppression pool through the RHR heat exchangers and returning it to the common wetwell spray sparger and the common drywell spray sparger. In addition, the ACIWA mode of RHR~~(C) subsystem~~ provides a backup drywell or wetwell spray capability. The wetwell sparger only accommodates a small portion of the total RHR pump flow; the remainder of the flow is routed to the drywell spray sparger. Reactor Building Cooling Water (RCW) circulating through the shell side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the external heat sink via the reactor service water (RSW) system. Either RHR wetwell spray subsystem is sufficient to

(continued)

BASES (continued)

APPLICABILITY (continued)

these MODES. Therefore, maintaining the RHR containment spray subsystems OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one RHR containment spray subsystem inoperable, the ~~ACIWA mode of RHR(C) using the Fire Protection System (FPS) can be used to inject water into the drywell or wetwell spray spargers. The ACIWA is verified to be functional by stroking one complete cycle of each of the two manual valves in the FPS connection to RHR(C) injection line, by verifying that the FPS header pressure is maintained, and by stroking one complete cycle of the RHR(C) subsystem injection valve.~~ The functionality of ACIWA required here is not as restrictive as that required for LCO 3.5.1 Required Action B.2 where the concern is station blackout.

ACIWA mode of RHR B or C using the Fire Protection System (FPS) can be used to inject water into the drywell or wetwell spray spargers. Loop B(C) of ACIWA is verified to be functional by stroking one complete cycle of each of the two manual valves in the FPS connection to the RHR Loop B(C) injection line, by verifying that the FPS header pressure is maintained, and by stroking one complete cycle of the RHR Loop B(C) injection valve.

If the ACIWA is verified to be functional, it compensates

Completion Time is restricted to 7 days. If the ACIWA is verified to be functional, a Completion Time of 14 days is chosen in light of the redundant containment spray capabilities afforded by the OPERABLE subsystem and ACIWA, and the low probability of a small break in the reactor coolant boundary occurring during this period.

B.1

With both RHR containment spray subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. In this Condition, there is a substantial loss of the primary containment bypass leakage mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low

(continued)

As an example, the first 20 cutsets contribute approximately three quarters of the total core damage frequency. Two-thirds of this amount is due to station blackout events, all of which involve failure or unavailability of the Reactor Core Isolation Cooling (RCIC) system. In addition, eight of the twenty basic events of greatest Fussell-Vesely importance belong to RCIC. If the RCIC were not present in the design, the calculated CDF would be approximately one decade higher. These observations highlight RCIC and its capability to operate without AC power for several hours as important features of the ABWR. They also identify the importance of station battery capability to provide RCIC control power for several hours.

As an additional example, failure of the combustion turbine generator (CTG) is included in each of the station blackout failure sequences and cutsets. It is also among the top twenty in Fussell-Vesely importance. These insights identify the diverse source of emergency power provided by the CTG as an important feature of the ABWR design.

Other systems and features which provide diversity in addition to fulfilling redundant functions were identified and their importance assessed. Following these evaluations important ABWR features and capabilities were identified.

19.8.1.3 Features Selected

The specific capabilities and features identified as being important to safety are listed in Table 19.8-1. The basis for the selection of each feature or capability is also provided in the table.

RCIC

In the unlikely event that offsite AC power is lost and the three Emergency Diesel Generators and the CTG are not available, the RCIC system can provide core cooling from a diverse power source (reactor steam) for an extended amount of time. RCIC operation for an extended period of time requires that makeup water supply be switched from the CST to the suppression pool. In addition, the station battery capability must be adequate to provide RCIC control and motive power for approximately eight hours. The capability of the RCIC to provide core cooling from a power source diverse from AC provides approximately decade reduction in the calculation of the estimated CDF. Sensitivity studies have shown that RCIC operation for two hours provides most of the benefit.

Combustion Turbine Generator

In the unlikely event that offsite AC power is lost and all three EDGs are unavailable, the CTG provides a diverse source of AC power. It is connectable to any of the three safety divisions and is capable of powering one complete set of normal safe shutdown loads. No plant support systems are needed to start or run the CTG. The CTG starts automatically (this feature is not "important" in the context of this analysis) and safety-grade loads are to be added manually. Although the probability of losing offsite power and all three EDGs at the same time is very small, the consequences of such an event is potentially very significant. The capability to

provide AC power from a diverse source substantially reduces the risk of a loss of offsite power resulting in a station blackout.

High Pressure Core Flooder (HPCF) Logic and Control

The operation of the HPCF is controlled by the digital safety system logic and control (SSLC) system. As identified in SECY 93-087, the common cause failure of digital instrumentation and

The ACIWA System (including the diesel driven fire pump fuel supply tank) will be protected against site flood and severe weather events. The ACIWA diesel fuel storage tank will have sufficient storage capacity to support 72 hours of operation.

operation through an independent and diverse "hard wired" circuit. Although the probability of a common cause failure of the SSLC is very low, an independent and diverse means of HPCF operation further reduces the risk associated with system operation through the multiplexed digital SSLC.

AC-Independent Water Addition (ACIWA) System

The ACIWA provides diverse capability to provide water to the reactor in the event that AC power or the ABWR engineered safety systems are not available. The system has a diesel driven pump with an independent water supply and all needed valves can be accessed and operated manually. In addition, support systems normally required for emergency core cooling systems are not required for ACIWA operation. Even though the ACIWA is not a first line prevention or mitigation system with respect to core damage, it is important in preventing and mitigating severe accidents in the unlikely event all other systems are unavailable.

Reactor Building Cooling Water (RCW) / Reactor Service Water (RSW)

The RCW system and the RSW system are each designed with two parallel loops in each division. Each loop (i.e., 50% of the capacity of each division) is capable of removing all of the component heat loads associated with operation of the ECCS pumps. Together, the two loops in each division are capable of removing heat from the suppression pool through the RHR heat exchangers during LOCA. The parallel loops of RSW and RCW within each division substantially reduce the calculated CDF.

Prevention of Intersystem LOCA

In SECY 90-016 and 93-087 it has been recommended that designers should reduce the possibility of a loss of coolant accident outside containment by designing (to the extent practical) all systems and subsystems connected to the Reactor Coolant System (RCS) to withstand full RCS pressure. All piping systems, major systems components (pumps and valves), and subsystems connected to the reactor coolant pressure boundary (RCPB) which extend outside the primary containment boundary are designed to the extent practicable to an ultimate rupture strength (URS) at least equal to full RCPB pressure. The design provisions provided reduce the possibility of an intersystem loss of coolant accident (ISLOCA) and consequently the probability of a loss of coolant accident outside the containment being an initiating event that could lead to core damage.

seismic event. Therefore, all of the seismic analyses assume that only emergency AC power and DC power are potentially available.

19.8.2.2 Logical Process Used to Select Important Design Features

The seismic margins analysis did not include the calculation of minimal cutsets which contribute to CDF. Therefore, there was no calculation of importance parameters such as Fussell-Vesely or Risk Achievement. Since importance parameters were not available, two alternate bases were used to select the important features. The first basis used was the identification of the functions and equipment whose failure would result in the shortest path to core damage in terms of the number of failures required and the relative seismic capacities of the components involved. The second basis used was the identification of the most sensitive functions and equipment in terms of the effect on accident sequence and accident class HCLPFs due to potential variations of component seismic capacities. Using these two bases, the seismic margins analysis was systematically reviewed to identify the “important” features.

19.8.2.3 Features Selected

Table 19.8-2 lists the features selected and the rationale for selection. These features met the criteria of either the shortest path to core damage or the most sensitive components.

Shortest Paths to Core Damage

It is assumed that the failure of any Category I structure leads directly to core damage. The structures with lowest HCLPFs are the containment and the reactor building. It is important that HCLPFs for Category 1 structures not be compromised by future modifications or additions that could affect safety equipment.

Seismic failure of DC power also is assumed to lead directly to core damage. Without DC power, all instrument and equipment control power is lost and the reactor cannot be controlled or depressurized. In the seismic margins analysis it is assumed that this results in a high pressure core melt. The limiting components for DC power are the batteries and the cable trays.

It is possible that a large seismic event could impair the ability to scram due to deformation of the channels that enclose each fuel bundle. In the event that the scram function is impaired, the only means of reactivity control would be the Standby Liquid Control (SLC) System. Seismic failure of the SLC system to insert borated solution into the reactor is controlled by the seismic capacity of the SLC pump and the SLC system boron solution tank.

Emergency AC power and plant service water were both treated as having the same effects in the seismic margins analysis. Failure of either system would require only one additional failure to result in core damage. The limiting components for seismic failure of emergency AC power are the diesel generators, transformers, motor control centers, and circuit breakers. The limiting components for seismic failure of plant service water are the service water pumps, room air conditioners, and the service water pump house.

Most Sensitive Components

The HCLPFs of the accident sequences with the lowest HCLPFs could be increased by increasing the individual HCLPFs of the AICWA pumps, the fuel channels, or the RHR heat exchangers. The HCLPFs of the appropriate accident sequences would be increased by an amount equal to the increase in the HCLPF of any of these components.

The only single item that could, by itself, decrease the HCLPF of any accident sequence below the acceptance criteria is a Category I structure having a HCLPF below 1.67 times SSE. This would also decrease the HCLPF of accident class IE; ATWS with high pressure melt due to loss of inventory.

The only system that could, by itself, result in lowering an accident sequence HCLPF below the acceptance criteria is DC power. DC power has two components that could fail the sequence—the batteries and the cable trays.

AC-Independent Water Addition (ACIWA)

The ACIWA can provide vessel injection, wetwell or drywell spray, or spent fuel pool makeup in the event all AC power is unavailable.

system has a diesel driven pump with an independent water supply and all needed valves can be accessed and operated manually. In addition, support systems normally required for ECCS operation are not required to function for ACIWA operation. ~~The ACIWA can provide either vessel injection or drywell spray in the event all AC power is unavailable.~~ Although the system pumps are housed in an external building (shed), the collapse of the building would not prevent the diesel driven pump from starting and running. The fire truck provides a backup to these pumps.

Seismic Walkdown

In addition to the above identified features, it was judged important that the seismic walkdown noted in Subsection 19.9.5 be conducted to seek seismic vulnerabilities.

19.8.3 Important Features from Fire Analyses

19.8.3.1 Summary of Analysis Results

An ABWR fire risk screening analysis based on the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology was performed to assess vulnerability to fires within the plant. Each scenario evaluated was calculated to have an acceptably low core damage frequency.

19.8.3.2 Logical Process Used to Select Important Design Features

The screening criterion for EPRI's FIVE methodology provided the primary basis for systematically evaluating important design features. The FIVE methodology provides procedures for identifying fire compartments for evaluation purposes, defining fire ignition frequencies, and performing quantitative screening analyses. The criterion for screening

Table 19.8-2 Important Features from Seismic Analyses

Feature	Basis
Seismic design of the containment (2.14.1) and reactor building (2.15.10) and assurance that future modifications or additions to internal structures meet the requirements of Subsection 3.8 if they are made in the vicinity of safety equipment.	Failure of seismic Category I structures could lead directly to core damage because of possible damage to ESF equipment. The Containment and the Reactor Building are the seismic Category I structures with the lowest HCLPFs.
Seismic qualification of the station batteries and cable trays (2.12.12).	DC power is required for all safety-related instrument and equipment control functions. Failure of the DC power system could lead directly to core damage.
Seismic qualification of the emergency AC power system diesel generators (2.12.13), 480V transformers (2.12.1), circuit breakers (2.12.1, 2.12.12, and motor control centers (2.12.1, 2.12.12).	In a severe seismic event, it is likely that offsite AC power will be lost and emergency AC power will be the only source of AC power. The components in the emergency AC power system with the lowest HCPLFs are the diesel generators, 480V transformers, circuit breakers, and motor control centers.
Seismic qualification of the plant service water system service water pumps, room air conditioners, and pump house (2.11.3).	In a severe seismic event, it is likely that offsite AC power will be lost and emergency AC power will be the only source of AC power. The plant service water system is required for diesel generator cooling and other cooling functions. The components in the service water system most ... er
Seismic qualification of SLC system boron solut[ion] tank and SLC pumps (2.2.4)	ACIWA can provide vessel injection, wetwell or drywell spray, or spent fuel pool makeup using equipment that does not require AC power. ... deformation of the fuel channels and the SLC system may be the only means of reactivity control. The most sensitive components in the SLC system are the boron solution tank and the SLC pumps.
Seismic qualification of the ACIWA system including the pumps, valves, and water supply ([2.15.6 (SSE only)]. The collapse of the ACIWA building (shed) should not prevent the pumps from starting and running [2.15.6 (SSE only)]. All needed valves for system operation can be accessed and operated manually (2.15.6, 2.4.1).	~~ACIWA can provide either vessel injection or drywell spray using equipment that does not require AC power~~. In addition, support systems normally required for ECCS operation are not required for ACIWA operation. ACIWA is an important system in preventing and mitigating severe accidents.
Seismic qualification of the RHR heat exchangers (2.4.1). See also Subsection 19K.5.	Seismic failure of RHR heat exchangers could partially drain the suppression pool and flood the RHR rooms. RHR is needed for decay heat removal and water in the suppression pool would provide fission product scrubbing in the event of core damage.

Table 19.8-7 Key Severe Accident Parameters (Continued)

Parameter Description	Value	Relates to What Feature?	Cross Reference
Lower Drywell Flooder			
Elevation	–10.5 m	Lower Drywell Flooder	
Area per valve	0.0081 m^2	Lower Drywell Flooder	
Plug Melting Temperature	533 K	Lower Drywell Flooder	
Suppression Pool Mass	3.6×10^6 kg	Containment Performance	ITAAC 2.14.1
COPS			
Equivalent Flow Diameter of Disk	0.2 m (8 in.)	COPS	
Diameter of Piping	0.25 m (10 in.)	COPS	
Setpoint	0.72 MPa	COPS	ITAAC 2.14.6
Tolerance at nom. temp.	5%	COPS	ITAAC 2.14.6
Effect of temp. on setpoint	2% per 55.6°C	COPS	
Firewater Addition System			
Injection Locations	Vessel, Wetwell, Drywell, and SFP ~~Vessel and Drywell~~	ACIWA	ITAAC 2.4.1, 2.15.6
Maximum Flow Rate	0.06 m^3/s	ACIWA	
Minimum Flow rate at COPS Setpoint	0.04 m^3/s	ACIWA	
Oxygen Concentration	<3.5% By Volume	Containment Inerting	Technical Specification LCO 3.6.3.2

(c) Service building,

(d) Pump house at the ultimate heat sink,

(e) Diesel generator fuel oil transfer pits, and

(f) Radwaste building.

(3) Close and dog all external water tight doors in the reactor and control buildings.

(4) Shut the plant down.

(5) Use power from the diesel generators or CTG if offsite power is lost.

Underground passages between buildings would not be affected because they are required to be watertight.

19.9.4 Confirmation of Seismic Capacities Beyond the Plant Design Bases

The seismic analysis assumed seismic capacities for some equipment for which information was not available. It is expected that these capacities can be achieved, but determination of actual seismic capacities must be deferred to the COL applicant when sufficient design detail is available. The actions specified in Subsection 19H.5 will be taken by the COL applicant.

19.9.5 Plant Walkdowns

The COL applicant shall develop procedures for the plant walkdown to seek seismic vulnerabilities which will be conducted by the COL applicant as outlined in Subsection 19H.5.1.

Similar walkdowns will be conducted by the COL applicant for internal fire and flooding events.

19.9.6 Confirmation of Loss of AC Power Event

The COL applicant will confirm the frequency estimate for the loss of AC power event (Subsection 19D.3.1.2.4). This review will address site-specific parameters (as indicated in the staff's licensing review basis document) such as specific causes (e.g., a severe storm) of the loss of power, and their impact on a timely recovery of AC power.

19.9.7 Procedures and Training for Use of AC-Independent Water Addition System

Specific, detailed procedures will be developed by the COL applicant for use of the AC-independent Water Addition System (including use of the fire truck) to provide vessel injection, wetwell spray, drywell spray, and ~~suppression~~ spent fuel pool makeup water, if necessary. Training will be included in the COL applicant's crew training program.

~~The procedures to be developed by the applicant will address operation of the ACIWA for vessel injection or drywell spray operation. Operation of the ACIWA System in the vessel injection mode requires valves F005, F101, and F102 to be opened and valve F592 to be closed. Reactor depressurization to below ACIWA System operating pressure is required prior to ACIWA operation in the vessel injection mode. Operation of the ACIWA in the drywell spray mode requires valves F017, F018, F101, and F102 to be opened and valve F592 to be closed. These valves are shown on Figure 5.4-10. The diesel fire pump will start automatically when the ACIWA is properly aligned for vessel injection or drywell spray. If the normal firewater system water supply is unavailable, the alternate water supply can be made available b y opening the manual valve between the diesel driven fire pump and the alternate water supply. This valve is shown in Figure 9.5-4. If it is necessary to use a fire truck for vessel injection or drywell spray, valve F103 must be opened in addition to operation of the valves discussed above for ACIWA operation. The valve for operation of the ACIWA using the fire truck is also shown on Figure 5.4-10. All of the valves required for ACIWA operation are manually operable.~~

If it is necessary to operate the ACIWA, radiation levels may be elevated in the rooms where the valves required for ACIWA operation are located. The applicant will make dose rate calculations for the specific configuration being constructed. These calculation will include the specific piping layout, shielding considerations, the potential for systems within the room to have recently been operated and thus contain radioactive coolant, and any other factors that significantly affect the dose rates. These dose rate calculations will be considered in the development of the specific plant procedures for ACIWA operation.

19.9.8 Actions to Avoid Common-Cause Failures in the Essential Multiplexing System (EMUX) and Other Common-Cause Failures

The procedures to be developed by the applicant will address operation of the ACIWA for vessel injection, drywell or wetwell spray operation, and spent fuel pool makeup. Operation of the ACIWA System in the vessel injection mode requires valves F005B/C, F101B/C, and F102B/C to be opened and valve F592B/C to be closed. Reactor depressurization to below ACIWA System operating pressure is required prior to ACIWA operation in the vessel injection mode. Operation of the ACIWA in the drywell spray mode requires valves F017B/C, F018B/C, F101B/C, and F102B/C to be opened and valve F592B/C to be closed. Operation of the ACIWA in the wetwell spray mode requires valves F019B/C, F101B/C, and F102B/C to be opened and valve F592B/C to be closed. Operation of the ACIWA in the spent fuel pool makeup mode requires valves F014B/C, F015B/C, F101B/C, and F102B/C to be opened and valve F592B/C to be closed. These valves are shown on Figure 5.4-10. The diesel fire pump will start automatically when the ACIWA is properly aligned. If the normal firewater system water supply is unavailable, the alternate water supply can be made available by opening the manual valve between the diesel driven fire pump and the alternate water supply. This valve is shown in Figure 9.5-4. If it is necessary to use a fire truck, valve F103B/C must be opened in addition to operation of the valves discussed above for ACIWA operation. The valve for operation of the ACIWA using the fire truck is also shown on Figure 5.4-10. All of the valves required for ACIWA operation are manually operable.

2.12.1 Electrical Power Distribution System

Design Description

The AC Electrical Power Distribution (EPD) System consists of the transmission network (TN), the plant switching stations, the Main Power Transformer (MPT), the Unit Auxiliary Transformers (UAT), the Reserve Auxiliary Transformer(s) (RAT(s)), the plant main generator (PMG) output circuit breaker, the medium voltage metal-clad (M/C) switchgear, the low voltage power center (P/C) switchgear, and the motor control centers (MCCs). The distribution system also includes the power, instrumentation and control cables and bus ducts to the distribution system loads, and the protection equipment provided to protect the distribution system equipment. The EPD System within the scope of the Certified Design starts at the low voltage terminals of the MPT and the low voltage terminals of the RAT(s) and ends at the distribution system loads. Interface requirements for the TN, plant switching stations, MPT, and RAT(s) are specified below.

The plant EPD System can be supplied power from multiple power sources; these are independent transmission lines from the TN, the PMG, and the combustion turbine generator (CTG). In addition, the EPD System can be supplied from three onsite Class 1E Standby Power Sources (Emergency Diesel Generators (DGs)). The Class 1E portion of the EPD System is shown in Figure 2.12.1.

During plant power operation, the PMG supplies power through the PMG output circuit breaker through the MPT to the TN, and to the UATs. When the PMG output circuit breaker is open, power is backfed from the TN through the MPT to the UATs.

The UATs can supply power to the non-Class 1E load groups of medium voltage M/C power generation (PG) and plant investment protection (PIP) switchgear, and to the three Class 1E divisions (Division I, II, and III) of medium voltage M/C switchgear.

The RAT(s) can supply power to the non-Class 1E load groups of medium voltage M/C PG and PIP switchgear, and to the three Class 1E divisions (Division I, II, and III) of medium voltage M/C switchgear.

Non-Class 1E load groups of medium voltage M/C switchgear are supplied power from a UAT with an alternate power supply from a RAT. In addition, the non-Class 1E medium voltage M/C switchgear can be supplied power from the CTG.

Class 1E medium voltage M/C switchgear are supplied power directly (not through any bus supplying non-Class 1E loads) from at least a UAT or a RAT. Class 1E medium voltage M/C switchgear can also be supplied power from their own dedicated Class 1E DG or from the non-Class 1E CTG.

The UATs are sized to supply their load requirements, during design operating modes, of their respective Class 1E divisions and non-Class 1E load groups. UATs are separated from the

RAT(s). In addition, UATs are provided with their own oil pit, drain, fire deluge system, grounding, and lightning protection system.

The PMG, its output circuit breaker, and UAT power feeders are separated from the RAT(s) power feeders. The PMG, its output circuit breaker, and UAT instrumentation and control circuits, are separated from the RAT(s) instrumentation and control circuits.

The MPT and its switching station instrumentation and control circuits, from the switchyard(s) to the main control room (MCR), are separated from the RAT(s) and its switching station instrumentation and control circuits.

The medium voltage M/C switchgear and low voltage P/C switchgear, with their respective transformers, and the low voltage MCCs are sized to supply their load requirements. M/C and P/C switchgear, with their respective transformers, and MCCs are rated to withstand fault currents for the time required to clear the fault from the power source. The PMG output circuit breaker, and power feeder and load circuit breakers for the M/C and P/C switchgear, and MCCs are sized to supply their load requirements and are rated to interrupt fault currents.

Class 1E equipment is protected from degraded voltage conditions.

EPD System interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault opens before other devices.

Instrumentation and control power for the Class 1E divisional medium voltage M/C switchgear and low voltage P/C switchgear is supplied from the Class 1E DC power system in the same division.

The PMG output circuit breaker is equipped with redundant trip devices which are supplied from separate, non-Class 1E DC power systems.

EPD System cables and bus ducts are sized to supply their load requirements and are rated to withstand fault currents for the time required to clear the fault from its power source.

For the EPD System, Class 1E power is supplied by three independent Class 1E divisions. Independence is maintained between Class 1E divisions, and also between Class 1E divisions and non-Class 1E equipment.

The only non-Class 1E loads connected to the Class 1E EPD System are the Fine Motion Control Rod Drives (FMCRDs) and the associated AC standby lighting system.

There are no automatic connections between Class 1E divisions.

Class 1E medium voltage M/C switchgear and low voltage P/C switchgear and MCCs are identified according to their Class 1E division. Class 1E M/C and P/C switchgear and MCCs

External (to Reactor Building) connections are provided to all three 1E Reactor Building 480 VAC Power Centers for portable FLEX diesel generators. The connectors are isolated from the Power Centers by normally open 1E breakers.

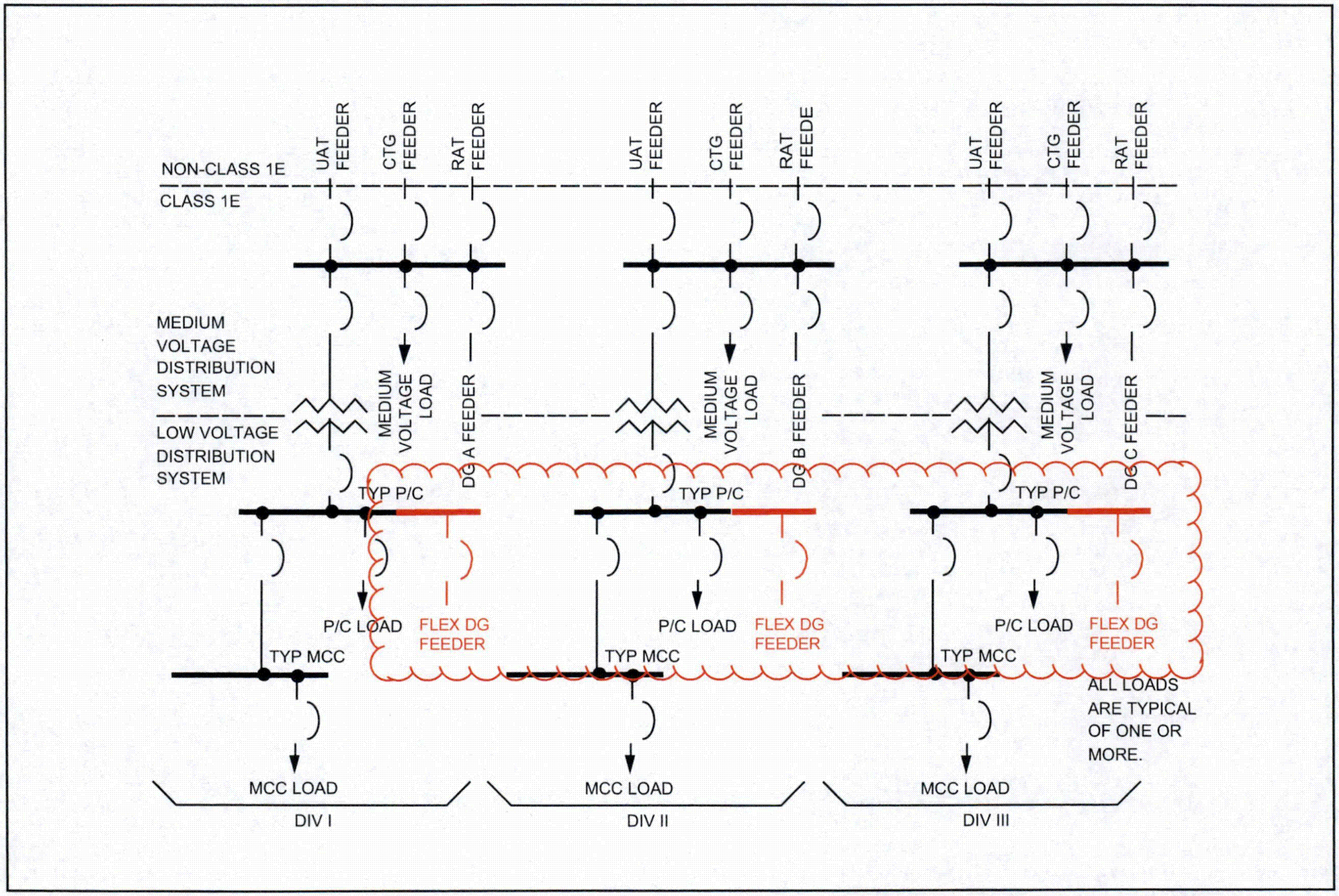


Figure 2.12.1 Class 1E Electrical Power Distribution System

Non-Class 1E loads being supplied from a Class 1E bus exists only in Division I, as described above for the FMCRDs. Except for associated AC standby and associated DC emergency lighting circuits, non-Class 1E loads are not permitted on Divisions II or III. This prevents any possibility of interconnection between Class 1E divisions.

8.3.1.1.2 Low Voltage Class 1E Power Distribution System

8.3.1.1.2.1 Power Centers

Power for 480V auxiliaries is supplied from power centers consisting of 6.9 kV/480V transformers and associated metal clad switchgear (see Figure 8.3-1). There are two power centers in each Class 1E division.

Class 1E 480V power centers supplying Class 1E loads are arranged as independent radial systems, with each 480V bus fed by its own power transformer. Each 480V Class 1E bus in a division is physically and electrically independent of the other 480V buses in other divisions and non-Class 1E load groups.

The 480V power centers are sized to supply motor control centers and motor loads greater than 100 kW, and up to and including 300 kW. Switchgear for the 480V power centers is of indoor, metal-enclosed type with draw-out circuit breakers which will interrupt maximum fault currents. Control power is from the Class 1E 125 VDC power system of the same Class 1E division.

Power centers are located in their respective divisional equipment areas.

8.3.1.1.2.2 Motor Control Centers

The Class 1E 480V MCCs are sized to supply motors 100 kW or smaller, control power transformers, process heaters, motor operated valves and other small electrically operated

To deal with an Extended Loss of AC Power (ELAP), external (to the Reactor Building) connections to each 1E RB divisional power center for portable FLEX 480 VAC diesel generators (DG) are installed, normally isolated from the 1E 480 VAC divisional Power Centers by open 1E breakers. The Reactor and Control Building 480 VAC buses can be re-energized via FLEX DGs, powering the 1E battery chargers, DC buses, and vital buses through UPS. The 480 VAC bus feeder breaker from the 6.9 KV to 480 V transformer and all load breakers on the Power Center will be opened to isolate the bus. The FLEX DG is connected, started, then the bus feeder from the DG is closed, energizing the 480 VAC power center. Power center load breakers are then closed one by one to energize loads (MCCs and discrete loads).

protection for fault currents in the penetration in the event of circuit breaker over-current fault protection failure.

8.3.1.1.3 120/240V Distribution System

Individual transformers and distribution panels are located in the vicinity of the loads requiring Class 1E 120/240V power. This power is used for emergency lighting, and other 120V Class 1E loads.

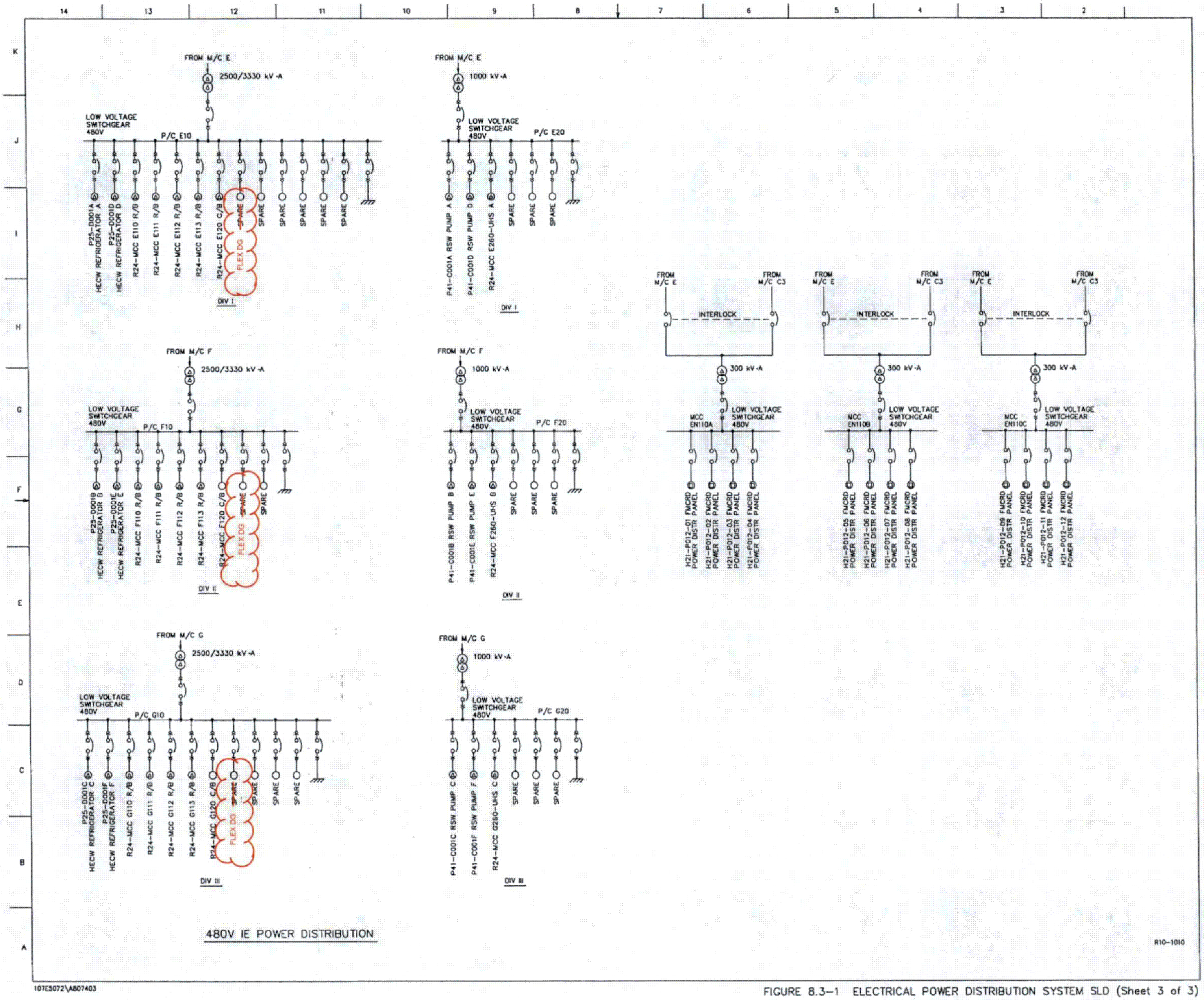


FIGURE 8.3–1 ELECTRICAL POWER DISTRIBUTION SYSTEM SLD (Sheet 3 of 3)