1D Response to NRC Post-Fukushima Recommendations

1D.1 Introduction

In response to the accident at the Fukushima Daiichi Nuclear Power Plant caused by the March 2011, Magnitude 9 Tohoku earthquake and subsequent tsunami, the Nuclear Regulatory Commission (NRC) established a Near-Term Task Force (NTTF) to review NRC processes and regulations to determine if improvements to its regulatory system were needed. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. These recommendations were issued in SECY-11-0093 (Reference 1D.5-9).

SECY-11-0124 (Reference 1D.5-13) and SECY-11-0137 (Reference 1D.5-1) provided the NRC Commissioners with the Staff's recommendations, including prioritization for implementation. Subsequently, SECY-12-0025 (Reference 1D.5-2) was issued describing proposed Orders to be issued to licensees and a draft request for information pursuant to 10 CFR 50.54(f). SECY-12-0025 stated that combined license plants under review would address the three orders and the request for information through the review process. On March 9, 2012, the Commission issued a Staff Requirements Memorandum (SRM) for SECY-12-0025 (Reference 1D.5-4) approving issuance of the orders and request for information with some modifications.

This appendix addresses the Tier 1 recommendations (recommendations that the Staff determined should be started without unnecessary delay) and Orders contained in SECY-12-0025, the Tier 2 recommendations (recommendations which could not be initiated in the near term) contained in SECY-11-0137, and the modifications documented in the SRM consistent with the as issued Orders (EA-12-049 [Reference 1D.5-5]), 050 (Reference 1D.5-6) and 51 (Reference 1D.5-7). The NRC recommendation in each of the following subsections is a summary of the recommendation from the NRC documents. The response to each recommendation discusses how ABWR addresses the recommendation. The numbers in parentheses of subsection headings correspond to the NTTF recommendation number.

1D.2.1 Seismic and Flooding Reevaluations (2.1)

1D.2.1.1 Seismic

NRC Recommendation

Perform a reevaluation of the seismic hazards using present-day NRC requirements and guidance to develop a Ground Motion Response Spectrum (GMRS). The seismic source models for the actual location of the plant (per applicable NUREG guidelines) may be used to characterize the hazards.

Response

In order to confirm seismic design adequacy of the standard plant, COL applicants shall perform a reevaluation of the seismic hazards using the current (at the time of COL) NRC requirements and guidance to develop a GMRS. The latest consensus seismic source models for Central America and the Central and Eastern United States Seismic Source Characterization (CEUS), NUREG-2115 (Reference 1D.5-8), may be used to characterize the hazards. The following COL License Information items provide additional guidance:

Item No.	Subject	Subsection
2.2	Seismic Design Parameters	2.3.1.2
19.4	Confirmation of Seismic Capacities Beyond the Plant Design Bases	19.9.4
19.5	Plant Walkdowns	19.9.5

1D.2.1.2 Flooding

NRC Recommendation

Perform a reevaluation of all appropriate external flooding sources, including the effects from local intense precipitation on the site, probable maximum flood (PMF) on streams and rivers, storm surges, seiches, tsunami, and dam failures. It is requested that the reevaluation apply present-day regulatory guidance and methodologies being used for ESP and COL reviews including current techniques, software, and methods used in present-day standard engineering practice to develop the flood hazard.

The recommendation also noted that flooding risks are of concern because the safety consequences of a flooding event may increase sharply with a small increase in flooding level.

Response

The COL applicant will use the current (at the time of COL) regulatory guidance and methodologies to evaluate flooding hazards. The site selection process will take into account the worst case predicted flood. Then grade level and flood control methods (e.g., site grading) will be determined based on this predicted flood level. The grade level floor will be at least 0.3 meters above this predicted flood level. Therefore, external flooding should not be a major concern for the ABWR.
The following COL License Information items provide additional guidance:

Item No.	Subject	Subsection
2.14	Floods	2.3.2.12
2.15	Probable Maximum Flood on Streams and Rivers	2.3.2.13
2.16	Ice Effects	2.3.2.14
2.19	Flooding Protection Requirements	2.3.2.17
19.3	Event Specific Procedures for Severe External Flooding	19.9.3
19.5	Plant Walkdowns	19.9.5

1D.2.2 Seismic and Flooding Walkdowns (2.3)

NRC Recommendations

Perform seismic walkdowns in order to identify and address plant specific degraded, nonconforming, or unanalyzed conditions and verify the adequacy of strategies, monitoring, and maintenance programs such that the nuclear power plant can respond to external events. The walkdown will verify current plant configuration with the current licensing basis, verify the adequacy of current strategies, maintenance plans, and identify degraded, non-conforming, or unanalyzed conditions. The walkdown procedure should be developed and submitted to the NRC. The procedure may incorporate current plant procedures, if appropriate. Prior to the walkdown, licensees should develop acceptance criteria, collect appropriate data, and assemble a team with relevant technical skills.

The NRC also requests that each addressee confirm they will use the industry developed, NRC endorsed, flood walkdown procedures or provide a description of plant specific walkdown procedures.

Response

Seismic and flooding plant walkdowns will be conducted after construction as documented in COL License Information Item 19.5 (Subsection 19.9.5).

1D.2.3 Station Blackout (SBO) Rulemaking (4.1)

NRC Recommendation

Strengthen the station blackout (SBO) mitigation capability at all operating and new reactors for design-basis and beyond-design-basis events. This includes (1) a minimum coping time of 8 hours for loss of all Alternating Current (AC) power, (2) establishing the equipment, procedures, and training necessary to implement an extended coping time of 72 hours for core and spent fuel cooling and for reactor coolant system (RCS) and primary containment integrity, and (3) pre-planning and pre-staging offsite resources to support uninterrupted core and spent fuel cooling, and RCS and primary containment integrity under conditions involving significant degradation of offsite transportationinfrastructure associated with a significant natural disaster. This recommendation will be implemented by rulemaking.

SECY-12-0025 adds the requirement that the loss of the Ultimate Heat Sink (UHS) should be evaluated as part of the SBO evaluation.

Response

(1) ABWR design can withstand an SBO for an extended period of time using an alternate AC power source, the Combustion Turbine Generator (CTG) with the only requirement to provide fuel for the CTG. This is described in Appendix 1C (Table 1C-3). Additionally, the ABWR design can withstand a sustained loss of all AC power, including the loss of CTG, for 70 hours while maintaining core cooling, as shown in Figure 19E.2-6c and described in Subsection 19E.2.2.3.

The ABWR design has a number of features which mitigate an SBO and extended loss of all AC power:

- The primary mitigation for an extended SBO is provided by a CTG, which is independent from the Emergency Diesel Generators (EDGs) and can be connected to the 1E buses. This CTG has black start capability and can be available for use within 10 minutes. The CTG is housed in a structure designed to the Internal Code Council International Building Code. This structure is protected from design basis floods and adverse weather conditions. (DCD Tier 1, Subsection 2.12.11 and Appendix 1C).
- The Division I batteries have a capacity of 8 hours (Subsection 8.3.2.1.3.1). The COL applicant will develop battery loading profiles and define appropriate load shedding during station blackout to confirm actual capacity (COL License Information item 19.9 Subsection 19.9.9).
- The Reactor Core Isolation Cooling system can provide core cooling for at least 8 hours during SBO conditions without reliance on AC power (Appendix 19E.2.1.2.2.1).
- The Alternating Current-Independent Water Addition (ACIWA) system is a seismically qualified system with an external permanent diesel-driven pump and water supply capable of providing water to the Residual Heat Removal (RHR) system for core and containment cooling without reliance on AC power. Operation of the ACIWA system is described in Subsection 5.4.7.1.1.10.
- The ACIWA system can be used to provide makeup water to the Spent Fuel Pool (SFP).

(2) The ABWR Design has the installed equipment (e.g., ACIWA system) to implement an extended coping time in excess of 72 hours without reliance on AC power for core, spent fuel and containment cooling as documented in Subsection 19E.2.2.3. The 72 hours of core cooling can be provided without reliance on the UHS. Relevant procedures and training will be developed per the "Plant Operational Procedure Development Plan" described in Section 13.5 (COL License Information item 13.3 Subsection 13.5.3.1).

(3) Pre-planning and pre-staging resources to support uninterrupted core and spent fuel pool cooling, and RCS and primary containment integrity under conditions involving significant degradation of the onsite facilities will be developed as part of the Emergency Procedure development (as required in COL License Information item 13.4, Subsection 13.5.3.2). Additionally, the COL Applicant will arrange for sufficient offsite resources to sustain core, containment, and spent fuel pool cooling for an extended period of time. These plans and resources will provide this capability under circumstances involving significant degradation of offsite transportation infrastructure associated with a significant natural disaster.

Detailed procedures and training associated with enhancing SBO mitigation capabilities in accordance with the SBO Rule will be developed during implementation of operational programs as described in Section 13.5 (COL License Information item 19.7, Subsection 19.9.7).

The following COL License information items provide additional guidance:

Item No.	Subject	Subsection
1.5	Emergency Procedures and Emergency Procedure Training Program	1A.3.1
1.13	Station Blackout Procedures	1C.4.1

Item No.	Subject	Subsection
5.8	Analysis of Non-Design Basis Loss of AC Coping Capability	5.4.15.2
5.8	Analysis to Demonstrate the Facility has 8 Hour Non-Design SBO Capability	5.4.15.2.1
5.8	Analysis to Demonstrate that the DC Batteries and SRV/ADS Pneumatics have Sufficient Capacity	5.4.15.2.2
9.9	Spent Fuel Firewater Makeup Procedures and Training	9.1.6.9
9.10	Protection of RHR System Connections to FPC System	9.1.6.10
9.37	Operation Procedures for Station Blackout	9.5.13.20
19.7	Procedures and Training for Use of ACIWA System	19.9.7
19.12	Procedures for Operation of RCIC from Outside the Control Room	19.9.12
19.19c	Procedures to Assure SRV Operability During Station Blackout	19.9.22

1D.2.4 Mitigating Strategies for Beyond Design Basis Events (4.2)

NRC Recommendation

NRC issued Order EA-12-049 (Reference 1D.5-5) to power reactor licensees and holders of construction permits requiring a three-phase approach for mitigating beyond-design-basis external events. The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment and spent fuel pool (SFP) cooling capabilities. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from offsite. The final phase requires obtaining sufficient offsite resources to sustain these functions indefinitely.

Response

The ABWR Design incorporates an ACIWA System. Operation of the ACIWA system is discussed in Subsection 5.4.7.1.1.10. The system can utilize either the diesel driven fire pump (described in Subsection 9.5.1.3.5) or an external pump (e.g. fire truck or trailer mounted pump) using one of two external Reactor Building connection points. Water can be directed to the Reactor Pressure Vessel (RPV), SFP or Containment Spray lines via the RHR system piping. ACIWA support equipment will be housed in a separate building. That building will be capable of withstanding the site specific external events such as seismic events, flooding, and other site specific external events (e.g. hurricanes). The capability of the building housing the ACIWA support equipment will be evaluated in the plant specific PRA (COL License Information item 19.30, Subsection 19.9.30).

Equipment procured for the ACIWA system will be procured in accordance with Diverse and Flexible coping Strategies (FLEX) guidance (Reference 1D.5-3). Equipment survivability was evaluated as part of the PRA and the results are summarized in Subsection 19E.2.1.2.3.4. The equipment that will be installed may be different than that assumed in the PRA and it will be evaluated as required by DCD Tier 1, Section 3.6 Design Reliability Assurance Program (DRAP) Design Requirements. The Design Reliability Acceptance Program (D-RAP) requires confirmation that the SSCs will function reliably when challenged.

The COL Applicant will identify additional equipment to be procured to provide defense in depth mitigation capability. Equipment to be considered includes:

- High capacity pumps which will be in addition to those described above
- Portable diesel generators to provide power to lighting and instrumentation
- Portable DC power supplies
- Handheld satellite phones
- Various hoses, fittings, cables, and jumpers necessary to connect the above equipment

The COL applicant will prepare a FLEX Integrated Plan for mitigating a Beyond Design Basis External Event (BDBEE) (Subsection 13.5.3.2). This plan will be patterned after the industry FLEX program (Reference 1D.5-3). The FLEX Integrated Plan will provide site specific guidance and strategies to restore core, containment and spent fuel cooling following a BDBEE. The strategies will be capable of mitigating a simultaneous loss of all AC power (including the CTG) and loss of normal access to the ultimate heat sink, and will be capable of being implemented in all operating modes. Subsection 19E.2.2.3 and Appendix 19Q discuss the design capabilities to be used in implementing the plan. The following COL License Information items provide additional guidance:

Item No.	Subject	Subsection
1.5	Emergency Procedures and Emergency Procedure Training Program	1A.3.1
1.13	Station Blackout Procedures	1C.4.1
9.9	Spent Fuel Firewater Makeup Procedures and Training	9.1.6.9
9.10	Protection of RHR System Connections to FPC System	9.1.6.10
9.19	Use of Communication System in Emergencies	9.5.13.2
9.31	Portable and Fixed Emergency Communication Systems	9.5.13.14
9.37	Operation Procedures for Station Blackout	9.5.13.20
13.4	Emergency Procedures Development	13.5.3.2

Item No.	Subject	Subsection
19.7	Procedures and Training for Use of ACIWA System	19.9.7
19.12	Procedures for Operation of RCIC from Outside the Control Room	19.9.12
19.14	Accident Management	19.9.14
19.19c	Procedures to Assure SRV Operability During Station Blackout	19.9.22

The equipment required to mitigate the BDBEE will be adequately protected from external events (COL License Information item 19.19b, Subsection 19.9.21). NRC Order EA-12-049 (Reference 1D.5-5) to power reactor licensees and holders of construction permits requires a three-phase approach for mitigating beyond-design-basis external events. Based on site specific evaluations of available consumables, this may be simplified to two phases if it can be shown that on site capability is sufficient to reach the point at which offsite supplies would be available. The site specific evaluation will also determine the durations of each phase based on predicted replenishment times for the consumables and this will be documented in the FLEX Plan.

The ABWR design does not take credit for the CTG that is part of the design bases for SBO. The design as described in Subsection 19E.2.2.3 credits the installed RCIC, ACIWA, and the Containment Overpressure Protection (COPS) systems to provide core, containment, and spent fuel cooling during all phases in accordance with the NRC Order.

To support the implementation of the FLEX Plan, the following system design requirements have been added to the original design:

- Added ACIWA subsystem to RHR Loop B, including connection to the Fire Protection system (FP) and external hose connection for fire truck.
- Made FP diesel-driven fire pump fuel capacity sufficient for 72 hours of operation;
- Included severe weather/flooding protection for ACIWA and the diesel-driven fire pump.
- Analyzed RCIC system for operation with 121°C (250°F) pump suction temperature.
- Added Reactor Building (RB) external connections for FLEX diesel generators to 480VAC RB 1E power centers.
- Replaced control of SRVs G, J, K, and P with Control of ADS SRVs C, H, L and R on the Remote Shutdown Panel (RSP)
- Added Wide Range RPV Water Level indication (Division I and II) (Cold Calibration) to the RSP.
- Added N2 Supply Header Pressure indication (Division I and II) to the RSP
- Added CST Water Level indication (Division I) to the RSP
- Added Containment Wide Range Pressure indication (Division I and II) to RSP
- Added Wide Range Suppression Pool Water Level indication (Division I and II) to the RSP

1D.2.5 Reliable Hardened Vents (5.1)

NRC Recommendation

NRC issued an Order to operating Boiling-Water Reactor (BWR) licensees with Mark I and Mark II containments requiring them to have a reliable hardened vent to remove decay heat and maintain control of containment pressure within acceptable limits following events that result in the loss of active containment heat removal capability or prolonged SBO. The hardened vent system is required to be accessible and operable under a range of plant conditions, including a prolonged SBO and inadequate containment cooling.

Response

This recommendation does not directly apply to the ABWR since the ABWR standard plant described in the DCD does not have a Mark I or Mark II containment. The design of the ABWR COPS vent which performs the above function was approved by the NRC in FSER Subsection 19.2.3.3.4. The design of the hardened vent that is part of the COPS is seismic category 1, qualified for accident pressures, passive, and reliable. This design is described in Subsection 6.2.5.2.6. Its use in conjunction with long term cooling without AC power to prevent fuel damage is demonstrated in Appendix 19E.2.2.

1D.2.6 Spent Fuel Pool (SFP) Instrumentation (7.1)

NRC Recommendation

NRC issued an order to power reactor licensees and holders of construction permits requiring them to have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

Response

The ABWR design includes reliable level monitors in the SFP that provide indication and annunciation via the process computer in the Main Control Room (MCR). Additionally, the SFP level indication will be provided at the remote shutdown system panel or other appropriate and accessible location. The design details of these instruments are described in Subsections 7.5.2.1. (2) (q) and 9.1.3.2. The instruments are shown in Figure 9.1-1. This design is consistent with the guidance provided in NEI 12-02, Revision 1 (Reference 1D.5-11), and JLD-ISG-2012-03 (Reference 1D.5-12) except for the labeling of the water levels. To be consistent with the RPV water level labeling the lowest water level will be identified as Level 1 and the highest water level will be identified as Level 3.

1D.2.7 Emergency Procedures Rulemaking (8.0)

NRC Recommendation

Strengthen and integrate onsite emergency response capabilities such as emergency operating procedures (EOPs), plant specific technical guidelines (PSTGs), severe accident management guidelines (SAMGs), FLEX Support Guidelines (FSGs) and extensive damage mitigation guidelines (EDMGs). This includes modification of Technical Specifications to reference the approved EOP technical guidelines and providing more realistic, hands-on training on SAMGs and EDMGs.

Response

As described in Subsection 13.5.3.2 Emergency Procedures Development, the procedure development will integrate the EOPs, PTSGs, SAMGs, FSGs and EDMGs.

ABWR Section 13.5 describes the COL License Information associated with procedure development planning.

Technical Specifications meet the requirement to reference the approved EOP Guidelines. (Technical Specifications 5.5.1.1.b)

Training development requirements are outside the scope of the ABWR (see Section 13.2). However there is a COL License Information item associated with Emergency Procedures and Emergency Procedures Training Program (COL License Information item No. 1.5, Subsection 1A.3.1).

1D.2.8 Enhanced Emergency Plan Staffing and Communication (9.3)

NRC Recommendation

Assess current communications systems and equipment used during an emergency event assuming the potential onsite and offsite damage as a result of a large scale natural event resulting in a loss of all AC power. It is also requested that consideration be given to any enhancements that may be appropriate for the emergency plan with respect to communications requirements of 10 CFR 50.47, Appendix E to 10 CFR Part 50, and the guidance in NUREG-0696 in light of the assumptions stated above. Also consider the means necessary to power the new and existing communications equipment with a loss of all AC power.

Assess current staffing levels and determine the appropriate staff to fill all necessary positions for responding to an event during a beyond design basis natural event and determine if any enhancements are appropriate given the considerations of NTTF Recommendation 9.3.

Response

The Emergency Plan will be prepared as part of the COL as required by COL License Information item 13.3.1.1 Emergency Plans. NEI 12-01 ("Guidelines for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities" [Reference 1D.5-10]) will be used in assessing staff and communications capabilities necessary to respond to a beyond design basis multi-unit event by the COL applicant as required by COL License Information item 13.3.1.2 "Staffing and Communications Capabilities".

1D.3 Tier 2 NRC Recommendations/Responses

1D.3.1 Other Natural External Hazards (2.1)

NRC Recommendation
Reevaluate and upgrade as necessary the design basis of structures, systems, and components important to safety for protection against natural external hazards other than seismic and flooding.

Response

Hazards and natural phenomena potentially affecting the ABWR site will be identified, screened and evaluated in accordance with the latest revisions of the Standard Review Plan as required in Section 2.3.2.

The following COL License Information items provide additional guidance:

Item No.	Subject	Subsection
2.3	Site Location and Description	2.3.2.1
2.6	Identification of Potential Accidents	2.3.2.4
2.7	Evaluation of Potential Accidents	2.3.2.5
2.8	External Impact Hazards	2.3.2.6
2.9	Local Meteorology	2.3.2.7
2.14	Floods	2.3.2.12
2.15	Probable Maximum Flood on Streams and Rivers	2.3.2.13
2.16	Ice Effects	2.3.2.14
2.19	Flooding Protection Requirements	2.3.2.17
3.5	Flood Elevation	3.4.3.1
3.7	Flood Protection Requirements for Other Structures	3.4.3.3
3.10	Missiles Generated by Other Natural Phenomena	3.5.4.2
3.12	Impact of Failure of Out of ABWR Standard Plant Scope Non-Safety-Related Structures, Systems and Components due to a Design Basis Tornado	3.5.4.4
19.3	Event Specific Procedures for Severe External Flooding	19.9.3

1D.3.2 Safety-related AC electrical power for the SFP makeup system (7.2)

NRC Recommendation

The NRC Is to issue an order requiring safety related AC power for the SFP makeup system. In accordance with SECY-11-0137, Recommendation 7.2 will be implemented by rulemaking to provide reliable SFP instrumentation and makeup capabilities.

Response

The ABWR design provides emergency makeup to the SFP using any of the three trains of the RHR system, which are powered by safety-related AC power (see DCD Tier 1 Subsection 2.6.2). Reliable Spent Fuel Pool indication is provided as discussed in item 1D.2.6 (above).

1D.3.3 Technical Specifications requirement for onsite emergency power (7.3)

NRC Recommendation

The NRC is to issue an order to revise technical specifications to require that one train of onsite emergency power be operable for SFP makeup and instrumentation whenever spent fuel is in the SFP, regardless of the operational mode of the reactor.

In accordance with SECY 11-0137, Recommendation 7.3 will be implemented by rulemaking to provide reliable SFP instrumentation and makeup capabilities.

Response

The ABWR Technical Specifications require at least one Emergency Diesel Generator and one RHR pump to be operable in all modes of operation. The safety related RHR system is backed by the emergency diesel generators and can also be powered by the CTG. The RHR system is capable of providing makeup to the SFP.

1D.4 Analysis of Non-Design Basis Loss of AC Coping Capability

The following COL License Information items provide guidance:

Item No.	Subject	Subsection
5.8	Analysis to Demonstrate the facility has 8 hour Non-Design SBO Capability	5.4.15.2.1
5.8	Analysis to Demonstrate That the DC Batteries and SRV/ADS Pneumatics Have Sufficient Capacity	5.4.15.2.2
19.9	Actions to Mitigate Station Blackout Events	19.9.9

1D.5 References

1D.5-1 SECY-11-0137, “Prioritization of Recommended Actions to be taken in response to Fukushima Lessons Learned” October 3, 2011.

1D.5-2 SECY-12-0025, “Proposed Orders and Requests for Information in Response to Lessons Learned From Japan’s March 11, 2011, Great Tohoku Earthquake and Tsunami”.

1D.5-3 NEI 12-06 [Revision 0] "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" August, 2012.

1D.5-4 SRM for SECY 12-0025, “Staff Requirements-SECY-12-0025 Proposed Orders and Requests for Information in Response to Lessons Learned from Japan’s March 11, 2011, Tohoku Earthquake and Tsunami” March 9, 2012.

1D.5-5 EA-12-049, “Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events,” March 12, 2012.

1D.5-6 EA-12-050, “Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents” March 12, 2012.

1D.5-7 EA-12-051, “Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,” March 12, 2012.

1D.5-8 NUREG-2115, “Central and Eastern United States Seismic Source Characterization”.

1D.5-9 SECY-11-0093, “The Near-Term Report and Recommendations for Agency Actions Following the Events in Japan,” July 12, 2011.

1D.5-10 NEI 12-01, Guidelines for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities.

1D.5-11 NEI 12-02 [Revision 1] “Industry Guidance for Compliance with NRC Order EA-12-051, “To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation” August 2012.

1D.5-12 JLD-ISG 2012-03SECY-11-0124, “Recommended Actions to be taken Without Delay from the Near Term Task Force Report”, September 9, 2011

1D.5-13 SECY-11-0124, “Recommended Actions to be taken Without Delay from the Near Term Task Force Report”, September 9, 2011

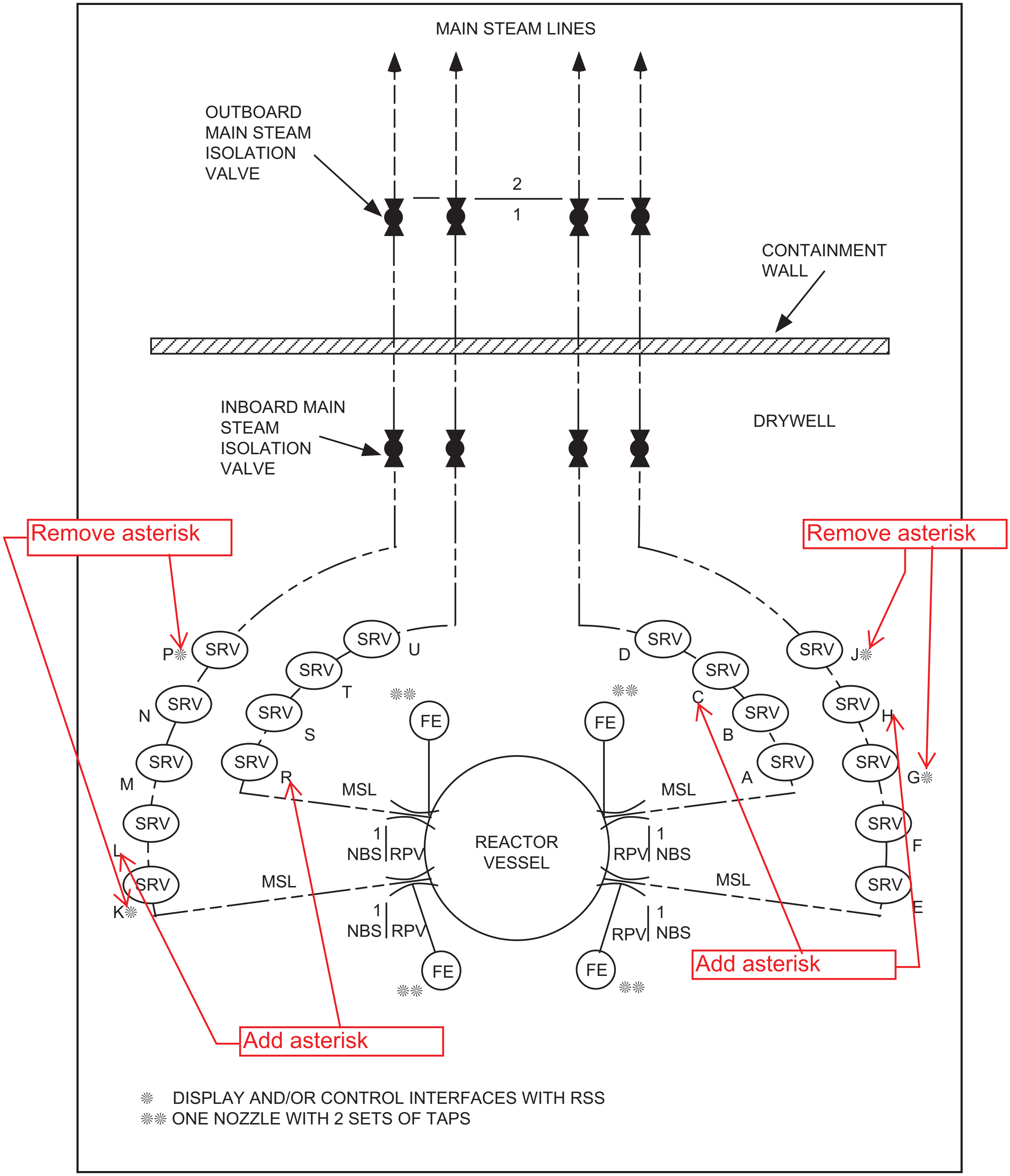
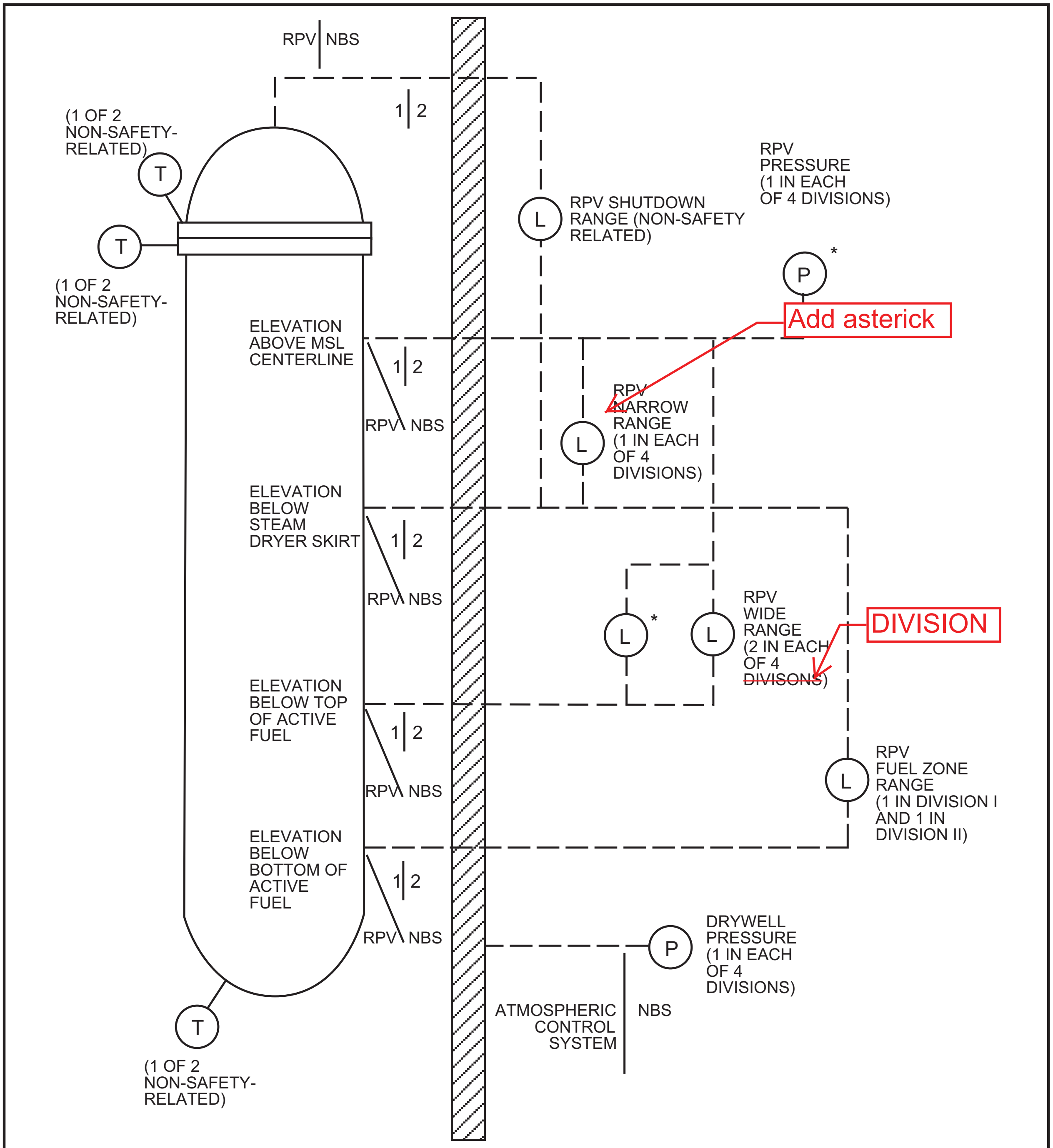


Figure 2.1.2a NBS Safety/Relief Valves and Steamline



* DISPLAY AND/OR CONTROL INTERFACES WITH RSS (2 OF 4 DIVISIONS)

NOTES:
1. EACH OF FOUR DIVISIONS OF NBS INSTRUMENTATION HAS ONE TURBINE INLET PRESSURE SENSOR AND ONE MAIN CONDENSER VACUUM SENSOR.
2. EACH DIVISION IS POWERED BY ITS RESPECTIVE CLASS 1E DIVISION.
3. THE MAIN CONDENSER VACUUM SENSORS ARE LOCATED ABOVE THE TURBINE BUILDING DESIGN BASIS FLOOD LEVEL.

Figure 2.1.2e NBS Drywell Pressure and Reactor Vessel Instrumentation

2.2.6 Remote Shutdown System

Design Description

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

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

(1) Residual Heat Removal (RHR) System

(2) High Pressure Core Flooder (HPCF) System

(3) Nuclear Boiler System (NBS)

(4) Reactor Service Water (RSW) System

(5) Reactor Building Cooling Water (RCW) System

(6) Electrical Power Distribution (EPD) System

(7) Atmospheric Control (AC) System

(8) Emergency Diesel Generator (DG)

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

(10) Flammability Control System (FCS)

(11) Suppression Pool Temperature Monitoring (SPTM) System

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

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

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

(12) High Pressure Nitrogen Gas Supply System

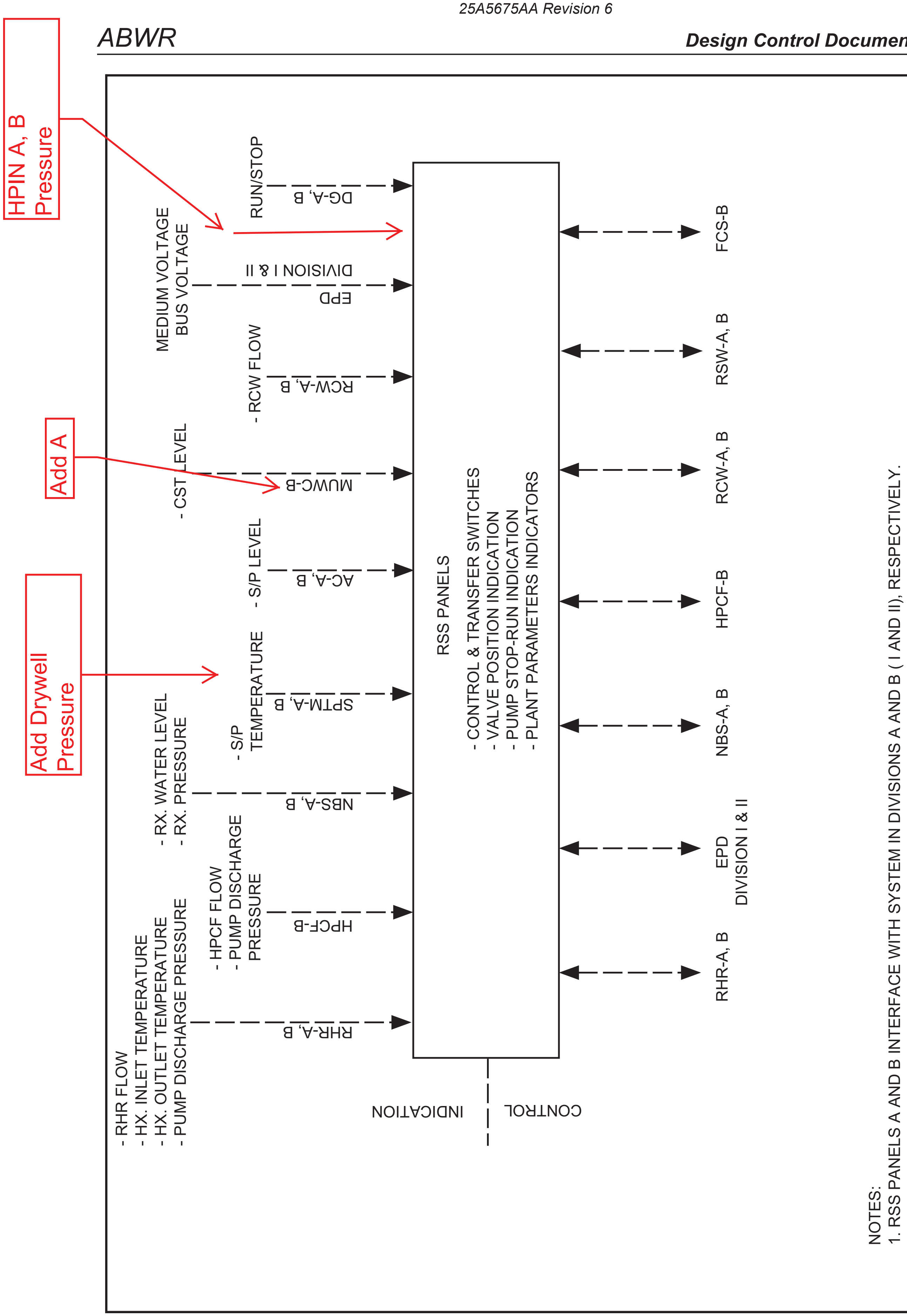
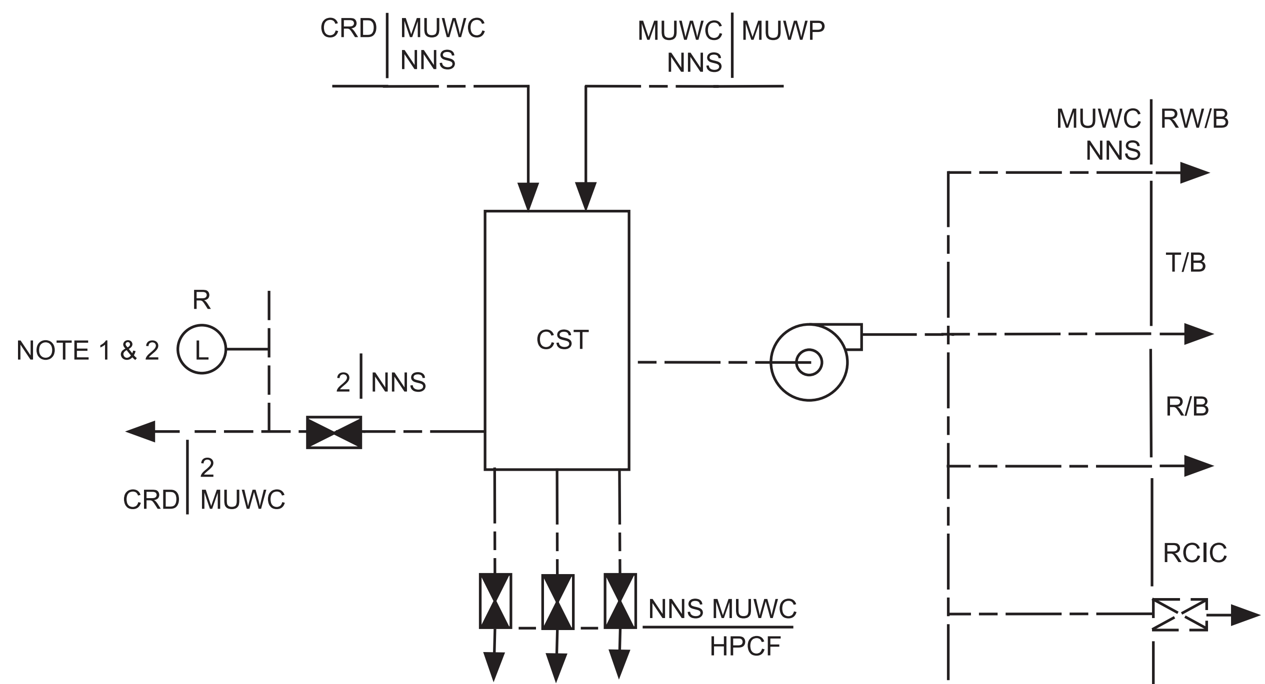


Figure 2.2.6 Remote Shutdown System



NOTES:
1. ONE SENSOR ASSIGNED TO EACH OF FOUR CLASS 1E DIVISIONS.
2. RSS INTERFACE IS ~~WITH DIVISION II LEVEL SENSOR ONLY.~~

for Division I and Division II Level Sensor

Figure 2.11.2 Makeup Water (Condensate) System

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

(1) Parameter displays for the sensors shown on Figure 2.11.13.

(2) Control and status indication for the active safety-related components shown on Figure 2.11.13.

The safety-related electrical equipment shown on Figure 2.11.13 located in the Reactor Building is qualified for a harsh environment.

The motor-operated valves (MOVs) shown on Figure 2.11.13 have active safety-related functions to open, close, or both open and close, and perform these functions under differential pressure, fluid flow, and temperature conditions.

The check valves (CVs) shown on Figure 2.11.13, have active safety-related functions to both open and close under system pressure, fluid flow, and temperature conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.13 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the HPIN System.

The HPIN System has Pressure indication for Division I and Division II on the Remote Shutdown Panel.

(v) Reactor Water level wide range indication - cold calibration (A,B)

(b) The following nuclear boiler instrumentation is provided on the remote shutdown control panels as indicated:

narrow

(i) Reactor water level wide range indication (A,B)

(ii) Reactor water level ~~shutdown~~ range indication (A,B)

(iii) Reactor pressure indication (A,B)

(iv) Indicate lights for four SRV valve open/close condition (three on Panel A, and one on Panel B

(c) The following function has transfer and control switches located at the Division 2 remote shutdown control panel: one air-operated relief valve. (The valve is 125 volt DC solenoid pilot operated.)

(5) Reactor Building Cooling Water (RCW) System

(a) The following functions have transfer and control switches located on the remote shutdown panels as indicated:

(i) RCW pumps (A,D and B,E)

(ii) RCW heat exchanger cooling water outlet valves (A,D,G and B,E,H)

(iii) RCW, RHR heat exchanger, outlet valve (A,B)

(iv) RCW, diesel generator, outlet valve (A,D and B,E)

(v) RCW separator valve between essential and non-essential loads (A,B)

(vi) RCW temperature control valves (A,B)

(b) The following RCW instrumentation is provided on the RSS control panels as indicated:

(i) RCW loop flow indication (A,B)

(ii) Indicating lights for valve positions and for pump stop/run (A,B)

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

(i) 6.9 kV feeder breaker: Unit auxiliary transformer B to M/C F

(b) Containment Wide Range Pressure on both RS panels.

(ii) 6.9 kV feeder breaker: Reserve auxiliary transformer A to M/C F

(iii) 6.9 kV feeder breaker: Emergency diesel generator B to M/C F

(iv) 6.9 kV feeder breaker: Combustion turbine generator to M/C F

(v) 6.9 kV load breaker: M/C F to P/C F20

(vi) 480V feeder breaker: TR to P/C F20

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

(8) Flammability Control System (FCS)

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

Narrow Range and Wide Range

(i) Valve (cooling water inlet) B

both RS panels.

(9) Atmospheric Control (AC) System

(13) High Pressure Instrument Nitrogen (HPIN) System
(a) Nitrogen Supply Header Pressure on both RS Panels.

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

(10) Makeup Water Condensate System (MUWC)

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

(11) Suppression Pool Temperature Monitoring System (SPTM)

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

(12) Emergency Diesel Generator (DG) System

(a) A transfer switch on each RS panel (A,B) permits DG control (start/stop) from the control room to be interrupted. There are no DG controls on the RS panels. During remote shutdown operation, the DGs can be controlled locally.

(b) Status lights provide DG status indication (run/stop) on each RS panel (A,B).

7.4.2 Analysis

7.4.2.1 Alternate Rod Insertion Function

7.4.2.1.1 General Functional Requirements Conformance

The alternate rod insertion (ARI) function is accomplished by the Rod Control and Information System (RCIS) and the Fine-Motion Control Rod Drive (FMCRD) Subsystem. This function

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

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

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

FUNCTION (INSTRUMENT OR CONTROL PARAMETER)	REQUIRED NUMBER OF DIVISIONS
20. Suppression Pool Temperature	2
21. Electric Power Distribution Controls	2(c)
22. Diesel Generator Interlock and Monitors	2
23. SRV Controls	(b)(c)

(a) RHR A for division I RSS panel, RHR B for division II RSS panel.

(b) Three on the Division I RSS, 1 on division II RSS.

(c) The specified number of channels are required to be OPERABLE for each device that can be controlled from the RSS panels.

24. N_2 Header Pressure	2
25. Drywell Pressure - Wide Range	2
26. Suppression Pool Water Level Wide Range	2
27. RPV Wide Range Water Level - Cold	2

BASES

LCO (continued)

2, 3, and 4. HPCF B Flow/Controls/Discharge Pressure

The HPCF system can be used to provide vessel inventory make up and decay heat removal while bringing the plant to MODE 3. The HPCF, in conjunction with other instruments and controls on the division II RSS panel, is sufficient to achieve and maintain MODE 3 from the Division II panel. The HPCF flow and pressure indications provide monitoring of HPCF operation. The controls provided are as given in reference 2. One channel of each Function is required to be OPERABLE to provide redundant capability to achieve MODE 3 from both RSS panels.

5 through 11. RHR A, B Control & Indication

The RHR system can be used to provide vessel inventory make up, decay heat removal, and suppression pool cooling while bringing the plant to MODE 3. The RHR, in conjunction with other instruments and controls on the RSS panels, is sufficient to achieve and maintain MODE 3 from either panel. The RHR flow indications provide monitoring of RHR operation and the heat exchanger monitors provide indication of decay heat removal. The RHR controls and monitors are adequate to place it in the shutdown cooling mode. The controls provided are as given in reference 3. Two channels of each RHR Function (RHR A on the division I panel and RHR B on the division II panel) are required to be OPERABLE to provide redundant capability to achieve MODE 3 from both RSS panels.

13. and 27.

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

of the RPV Water level conditions and ranges

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

There is an additional set of wide range instruments that have been calibrated for cold conditions and will be used when the normal instruments are off scale.

(continued)

BASES

LCO (continued)

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

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

17. Cooling Water Flow to Flammability Control System

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

and 26.

Narrow and Wide Range

18. Suppression Pool Water Level

narrow range

Suppression pool water level provides information needed to assess the status of the RCPB and to assess the status of the water supply to the ECCS. The level indicators monitor the suppression pool level from the bottom of the ECCS suction lines to five feet above the normal suppression pool level. One channel of ~~this Function is~~ provided on each of the RSS panels. Both channels are required to be OPERABLE to provide redundant capability to achieve MODE 3 from both RSS panels.

The wide range level indicators monitor the suppression pool from the centerline of the ECCS suction piping to the wetwell spargers.

both Functions are

19. Condensate Storage Tank Level

Condensate Storage Tank Level provides information needed to assess the status of the water supply to ~~the~~ HPCF. The indication is needed in order to achieve and maintain MODE 3 when using HPCF. ~~A channel of this Function is provided on the division II RSS panel. The channel is~~ required to be OPERABLE to achieve MODE 3 from ~~the Division II~~ RSS panel.

Both channels are

RCIC and

RCIC and

s

both

(continued)

BASES

LCO
(continued)

20. Suppression Pool Temperature

Suppression Pool Temperature allows the operator to detect trends in suppression pool water temperature in sufficient time to take action to prevent steam quenching vibrations in the suppression pool. This Function is required in order to maintain MODE 3. One channel of this Function is provided on each of the RSS panels. Both channels are ... PERABLE to provide redundant capability to maintain ... th RSS panels.

... er Distribution Controls

... are provided so the operator can select various AC ... r the equipment needed to achieve and maintain ... ectric Power Distribution Controls provided are as given ... nd 7. One channel of each Function is provided on each ... s. Both channels of each Function are required to be ... ovide redundant capability to achieve MODE 3 from ...

... rator System Interlock and Monitors

... provided to permit monitoring the status of the ... These monitors are required to permit the operator to ... ric power distribution. The interlock disables DG ... e control room to assure that the event that made the ... vailable will not disrupt DG operation. One channel of ... rovided on each of the RSS panels. Both channels of ... required to be OPERABLE to provide redundant capability to achieve MODE 3 from both RSS panels.

23. SRV Controls

This Function is provided to permit the operator to perform a controlled depressurization and to maintain reactor pressure within limits. Three channels are provided on the division I RSS panel and one channel is provided on the division II panel. These channels, in conjunction with other controls and indications on the panels, are sufficient to achieve and maintain MODE 3 from either panel. Three channels on the division I panel and one channel on the division II panel are required to be OPERABLE to provide redundant capability to achieve MODE 3 from both RSS panels.

24. N2 Header Pressure

This Function is provided to permit monitoring the status of the N2 Bottle Header Pressure. These monitors are required to permit the operator to manage the N2 supply to the SRVs. One channel of this Function is provided on each RSS panel. Both channels of the Function are required to be OPERABLE to provide redundant capacity to achieve MODE 3 from both RSS panels.

25.Drywell Pressure - Wide Range

This function is provided to permit monitoring the status of the drywell pressure. This will allow the operator to determine if there is a potential of operation of Containment Overpressure Protection System (COPS). One channel of this Function is provided on each RSS panel. Both channels of the Function are required to be OPERABLE to provide redundant capacity both RSS panels.

(continued)

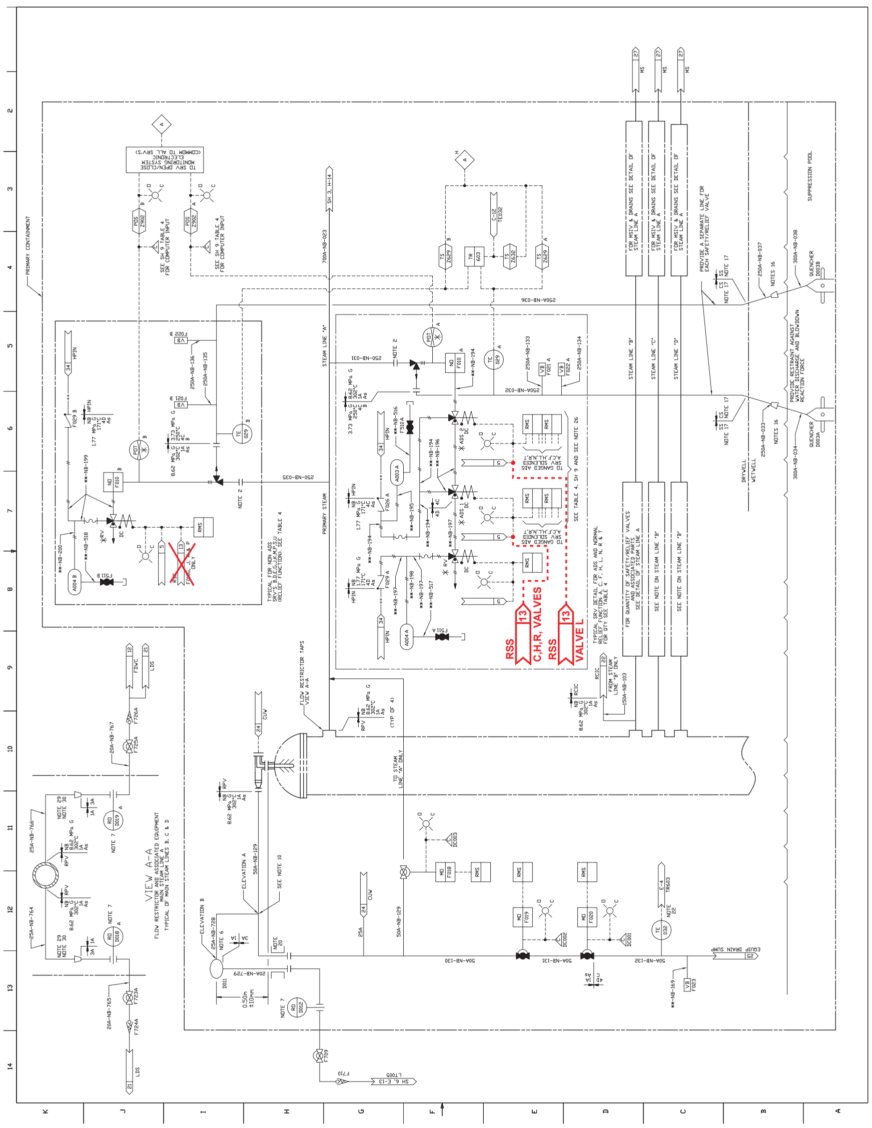


Figure 5.1-3 Nuclear Boiler System P&ID (Sheet 2 of 11)

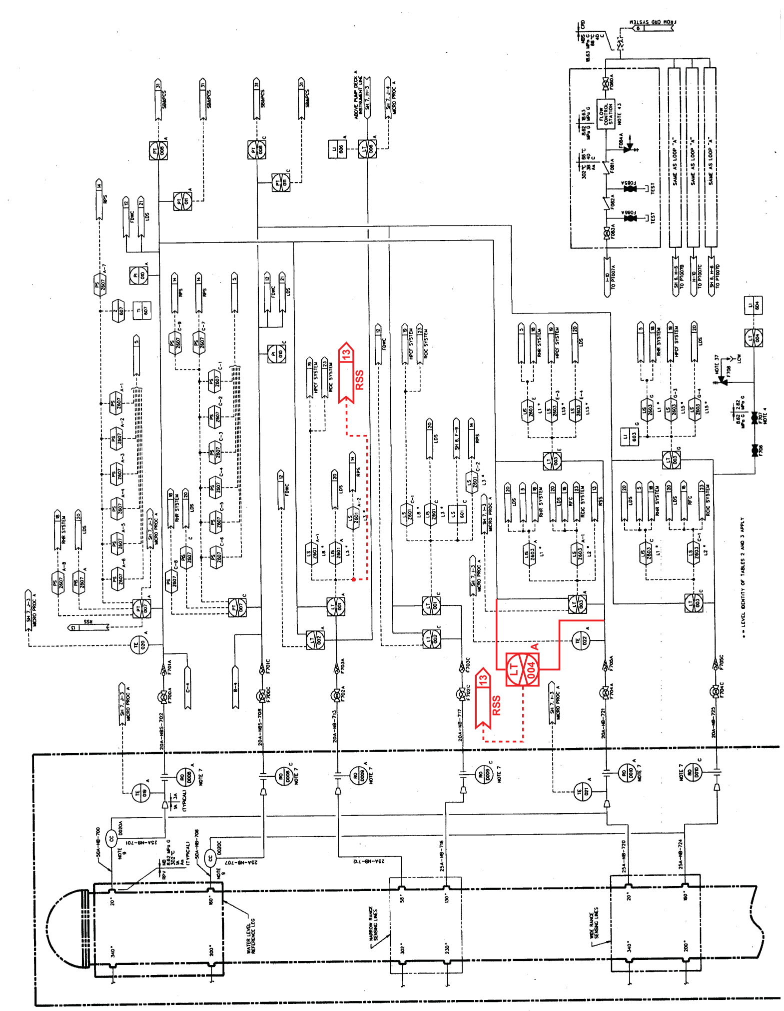


Figure 5.1-3 Nuclear Boiler System P&ID (Sheet 5 of 11)

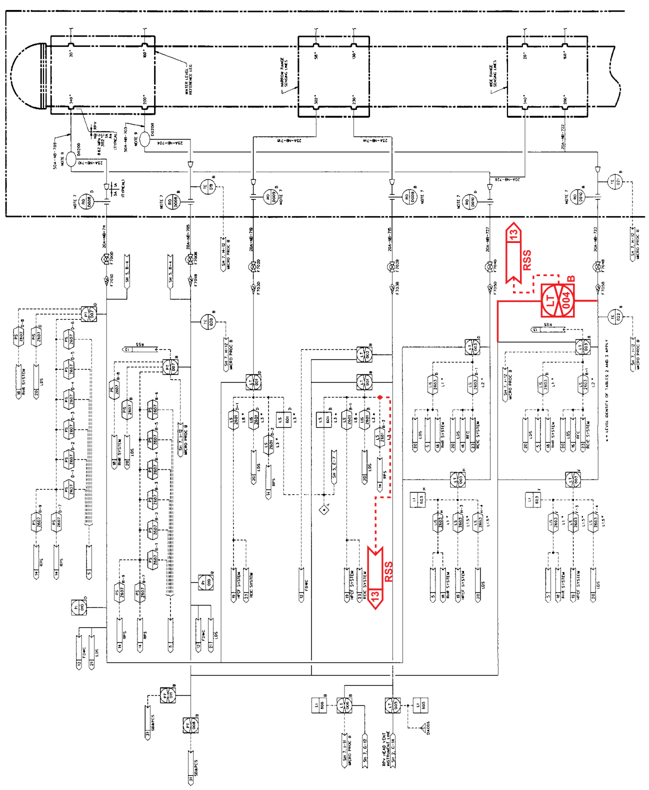


Figure 5.1-3 Nuclear Boiler System P&ID (Sheet 6 of 11)

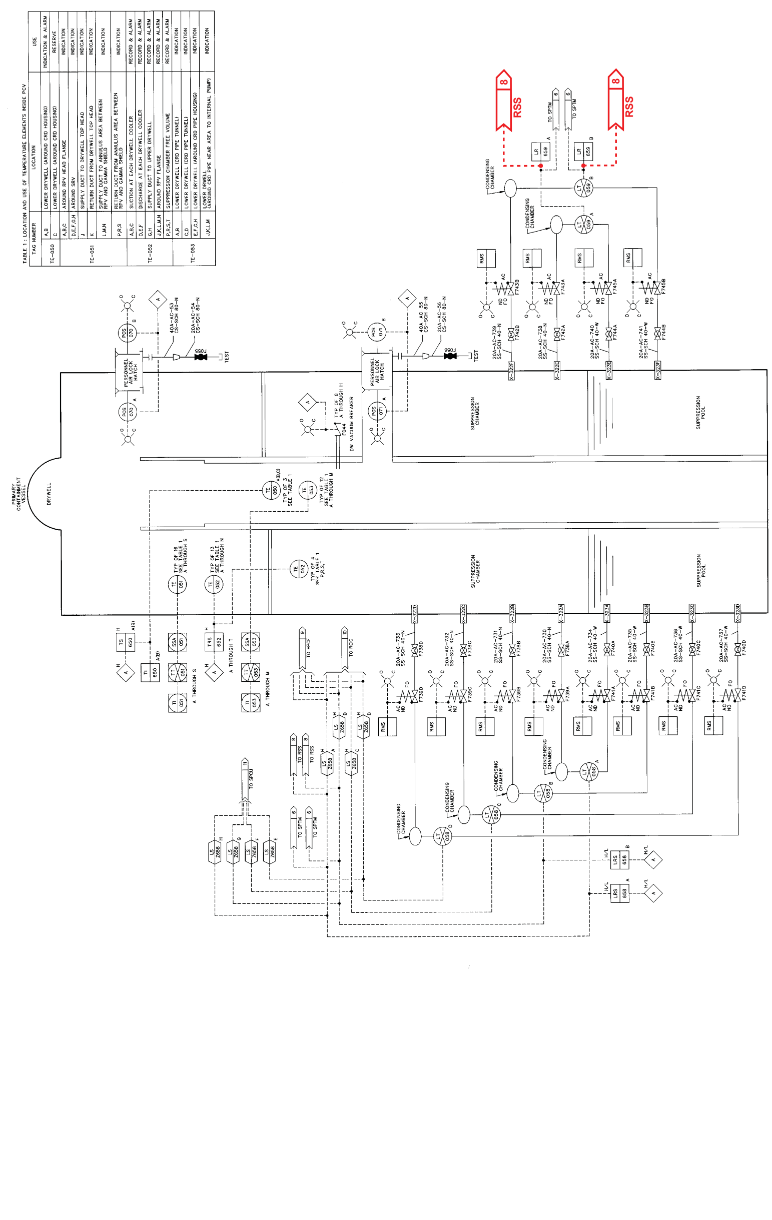


Figure 6.2-39 Atmospheric Control System P&ID (Sheet 2 of 3)

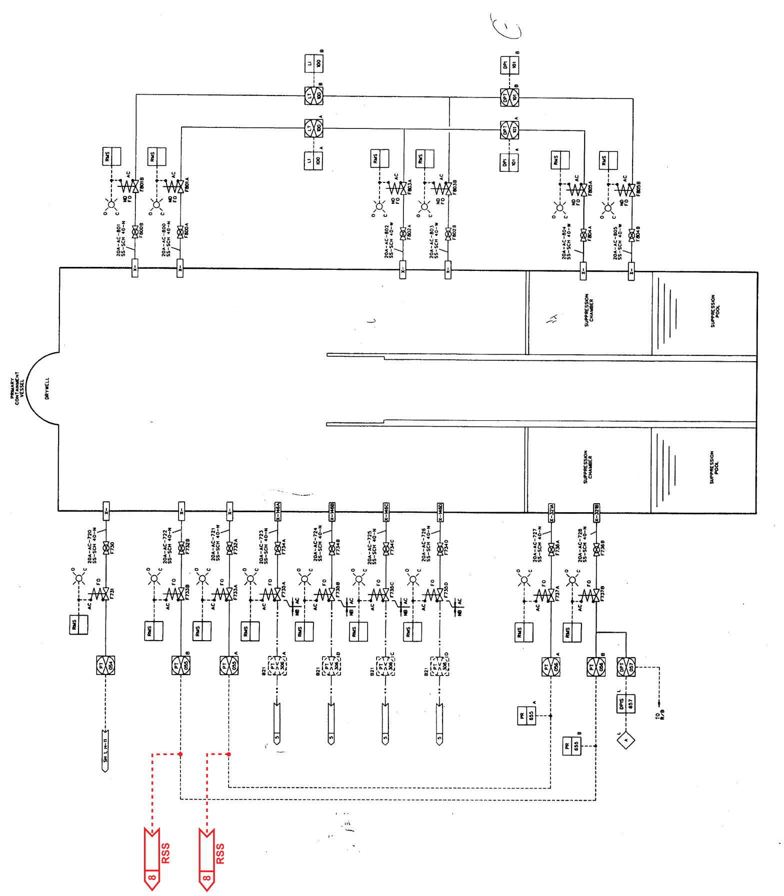


Figure 6.2-39 Atmospheric Control System P&ID (Sheet 3 of 3)

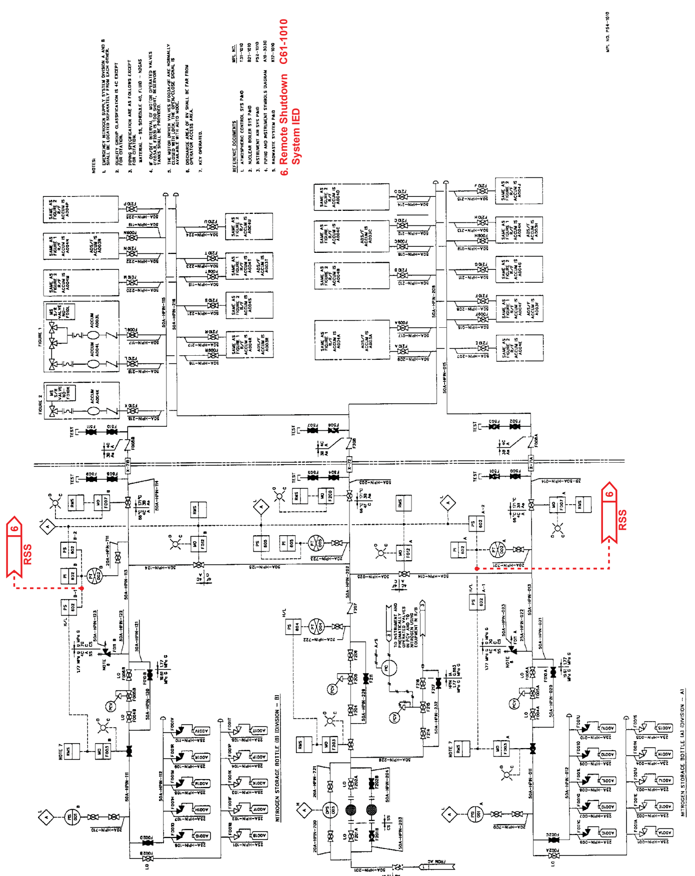


Figure 6.7-1 High Pressure Nitrogen Gas Supply System P&ID

NOTES:

1. PRESSURES SHALL BE IN MPa g.
2. THE ADS LOAD DRIVERS SHALL BE CONNECTED SO THAT IT IS NECESSARY TO ENERGIZE BOTH ADS CHANNELS TO ACTUATE THE ADS VALVES.
3. PARTS OF THE LOGIC AND CONTROL SYSTEM FOR THE OPERATION OF THE SRV'S IN THE RELIEF MODE ARE CLASSIFIED AS NON-SAFETY RELATED BUT THE TOTAL SYSTEM IS DESIGNED AS A SAFETY RELATED SYSTEM.
4. DIVISION 2 IS THE SAME AS DIVISION 1 EXCEPT THAT DIVISION 2 SUFFIX LETTERS ARE THOSE IDENTIFIED IN PARENTHESIS.
5. NUMBERS OR LETTERS IN PARENTHESIS DESIGNATE THOSE APPLICABLE TO THE SRV'S WHICH FOLLOW IN SEQUENCE AFTER THE FIRST IN THE GROUP OF 3 OR 4 SRV'S.
6. THE LOGIC SHALL INCORPORATE PROVISIONS TO REVERT 2/4 LOGIC TO 2/3 LOGIC DURING BYPASS OF A SINGLE DIVISION OF SENSORS. ALSO, THE LOGIC DIAGRAM SHALL NOT PERMIT THE BYPASS OF MORE THAN ONE DIVISION OF SENSORS AT A TIME. THE PROVISIONS ARE ILLUSTRATED IN THE SSLC BLOCK DIAGRAM, SH 36, ZONES B4 & C4.
7. ISOLATORS ARE NOT REQUIRED WHERE THE SAME DIVISIONAL LOGIC IS USED FOR BOTH INPUT SIGNALS AND LOGIC.
8. (A) FO IS "FAIL OPEN", FOR EXAMPLE, VALVE OPENS ON LOSS OF POWER AND/OR LOSS OF PNEUMATIC OR HYDRAULIC PRESSURE.

   (B) FC IS "FAIL CLOSED"; FOR EXAMPLE, VALVE CLOSES ON LOSS OF POWER AND/OR LOSS OF PNEUMATIC OR HYDRAULIC PRESSURE.
9. SEE TABLE 1 FOR ANNUNCIATOR/ALARM LIGHT INFORMATION, SH 34 & 35.
10. ALL ANNUNCIATORS ARE LOCATED IN THE MAIN CONTROL ROOM UNLESS OTHERWISE NOTED.
11. THE SRV OUTPUT LOGIC SHALL INCLUDE PROVISIONS TO BYPASS ONE OF TWO CHANNEL OUTPUTS AND TO REVERT 2/2 LOGIC TO 1/1 LOGIC WHEN BYPASSED. THE PROVISIONS ARE ILLUSTRATED IN THE SSLC BLOCK DIAGRAM, SH 36 & 37. ALSO SEE NOTE 3 ON SH 37. POWER SOURCE CONNECTIONS FOR SRV'S ARE 125V DC AND 125V DC RETURN AS SHOWN.
12. ADS OUTPUT LOGIC SHALL NOT INCLUDE PROVISIONS TO BYPASS THE DUAL OUTPUTS. SINGLE CHANNEL FAILURE IN ONE ADS DIVISION SHALL CAUSE LOSS OF OUTPUT FUNCTION IN THAT ADS DIVISION ONLY AS SHOWN ON SH 37.
13. MONITOR THE CONTINUITY OF THE SRV ADS SOLENOIDS BY APPLICATION OF A NON-ENERGIZING CURRENT TO EACH SOLENOID.
14. INTERMEDIATE PROCESSOR WHICH PREVENTS THE FAILURE OF THE NON-SAFETY RELATED DATA FROM AFFECTING THE SAFETY RELATED LOGIC.
15. THIS SIGNAL LINE SHALL BE HARDWIRED. INDICATORS REQUIRED TO BE HARDWIRED ARE SHOWN ON THIS DRAWING.
16. SIGNALS TO ANNUNCIATORS AND NON-SAFETY INDICATORS SHALL BE OPTICALLY ISOLATED FROM THE SAFETY RELATED INPUT SIGNAL.
17. Manual operation of the Division I ADS valves (R, C and H) and (Div. II) valve (L) from the Remote Shutdown Panel energizes the valve ADS solenoid of the associated division.

REFERENCE DOCUMENTS UNDER THE FOLLOWING IDENTITIES ARE TO BE USED IN CONJUNCTION WITH THIS DRAWING.

	MPL NO.
1. NUCLEAR BOILER SYSTEM, P&ID	B21-1010
2. REMOTE SHUTDOWN SYS, IBD	C61-1030
3. RESIDUAL HEAT REMOVAL SYSTEM, P&ID	E11-1010
4. HIGH PRESSURE CORE FLOODER, P&ID	E22-1010
5. LEAK DETECTION SYSTEM, IBD	E31-1030
6. REACTOR PROTECTION SYSTEM, IED	C71-1040
7. TURBINE CONTROL SYSTEM, IBD	N32-1030

SUPPORTING DOCUMENTS	MPL NO.
1. INTERLOCK BLOCK DIAGRAM (IBD) STANDARD	A10-3070

NOTES AND REFERENCES

Figure 7.3-2 Nuclear Boiler System IBD (Sheet 2 of 37)

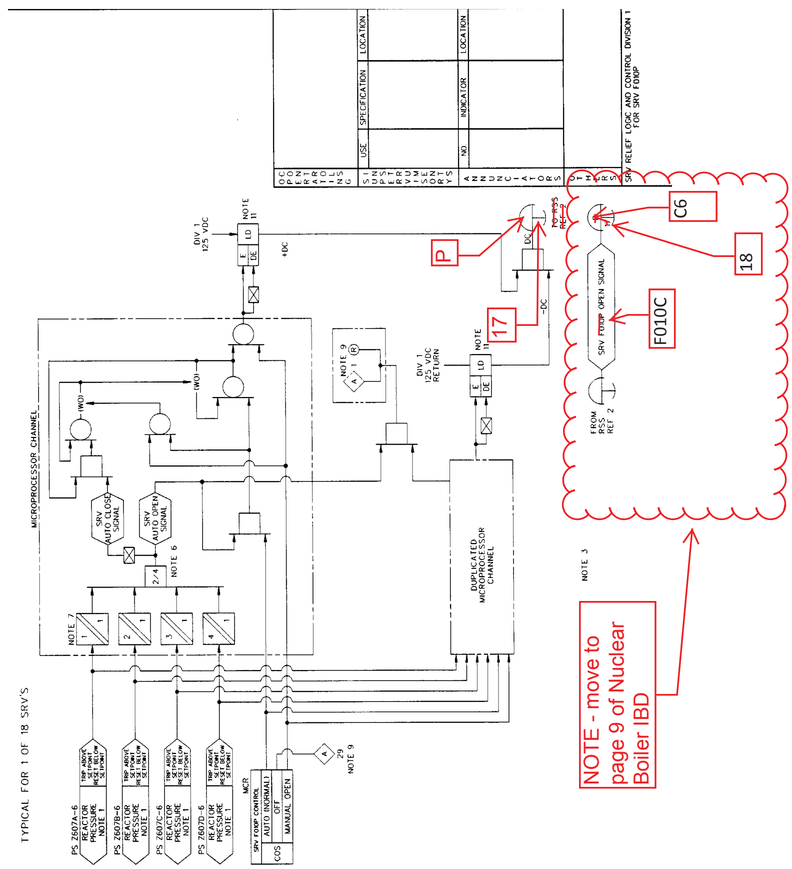


Figure 7.3-2 Nuclear Boiler System IBD (Sheet 3 of 37)

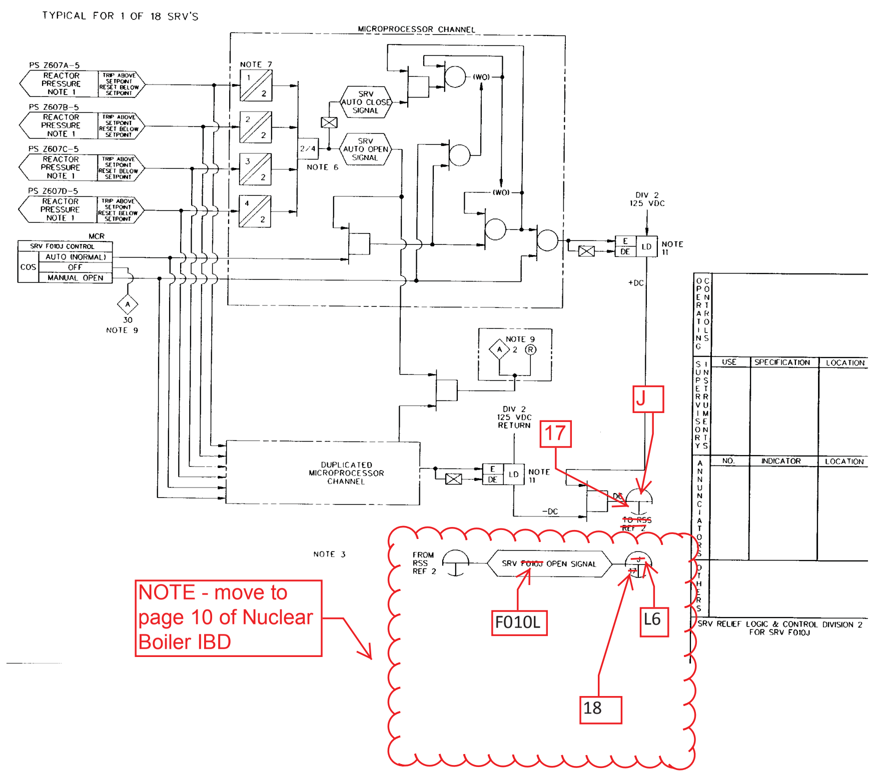


Figure 7.3-2 Nuclear Boiler System IBD (Sheet 4 of 37)

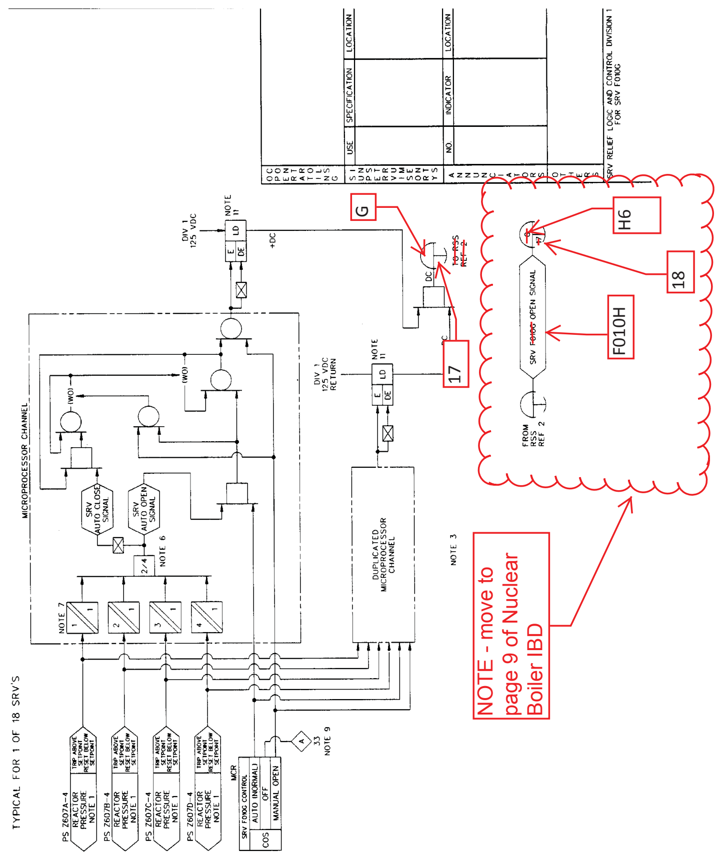


Figure 7.3-2 Nuclear Boiler System IBD (Sheet 6 of 37)

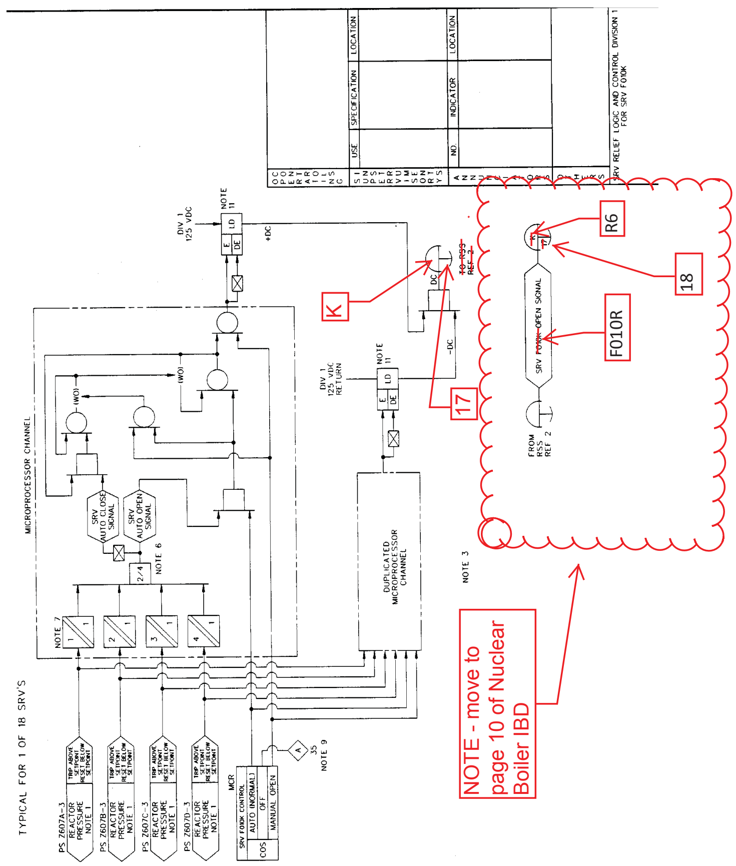


Figure 7.3-2 Nuclear Boiler System IBD (Sheet 7 of 37)

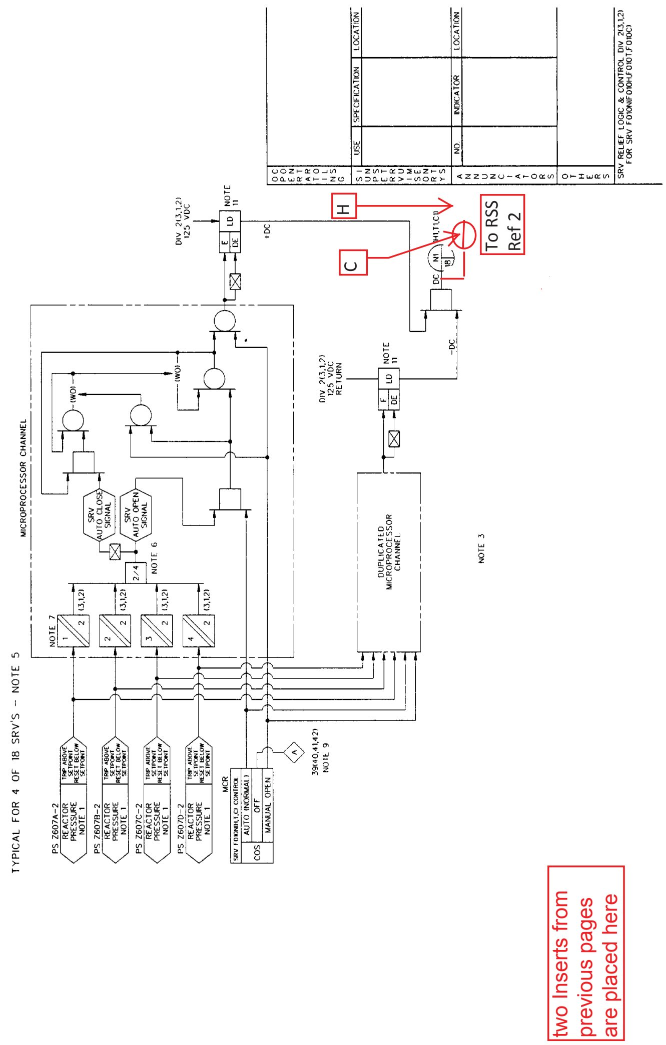


Figure 7.3-2 Nuclear Boiler System IBD (Sheet 9 of 37)

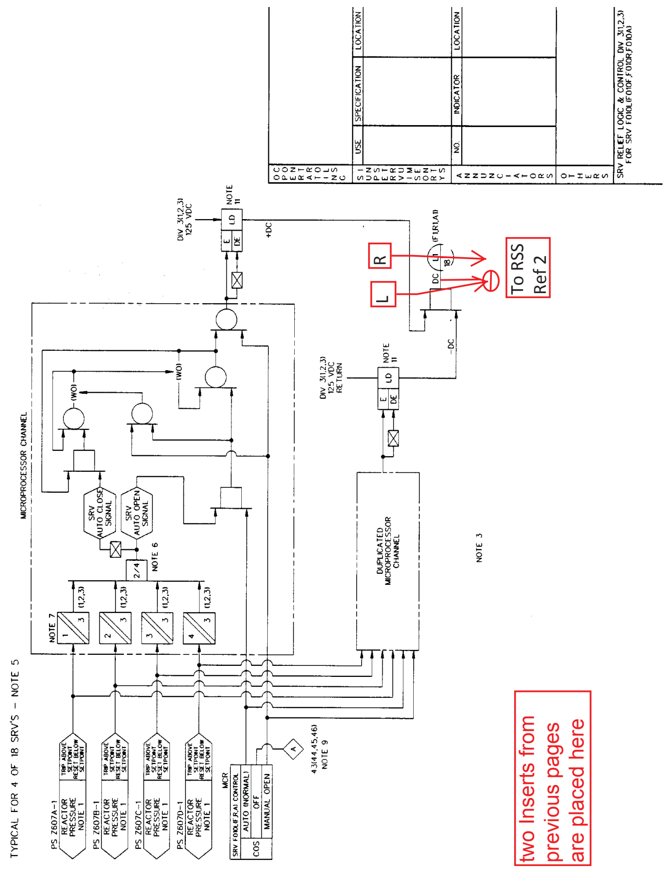


Figure 7.3-2 Nuclear Boiler System IBD (Sheet 10 of 37)

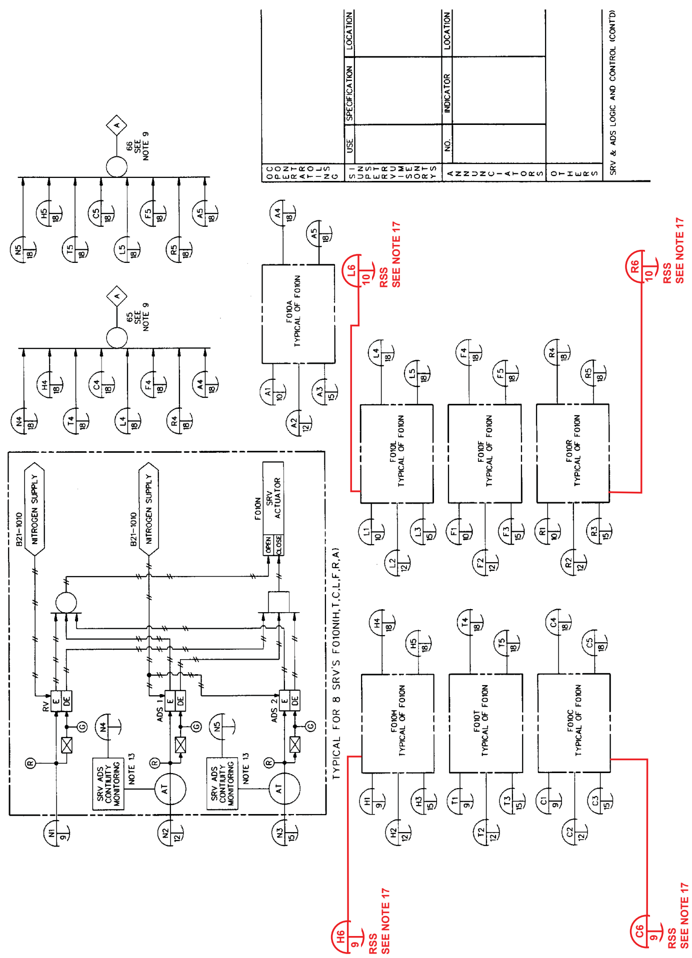


Figure 7.3-2 Nuclear Boiler System IBD (Sheet 18 of 37)

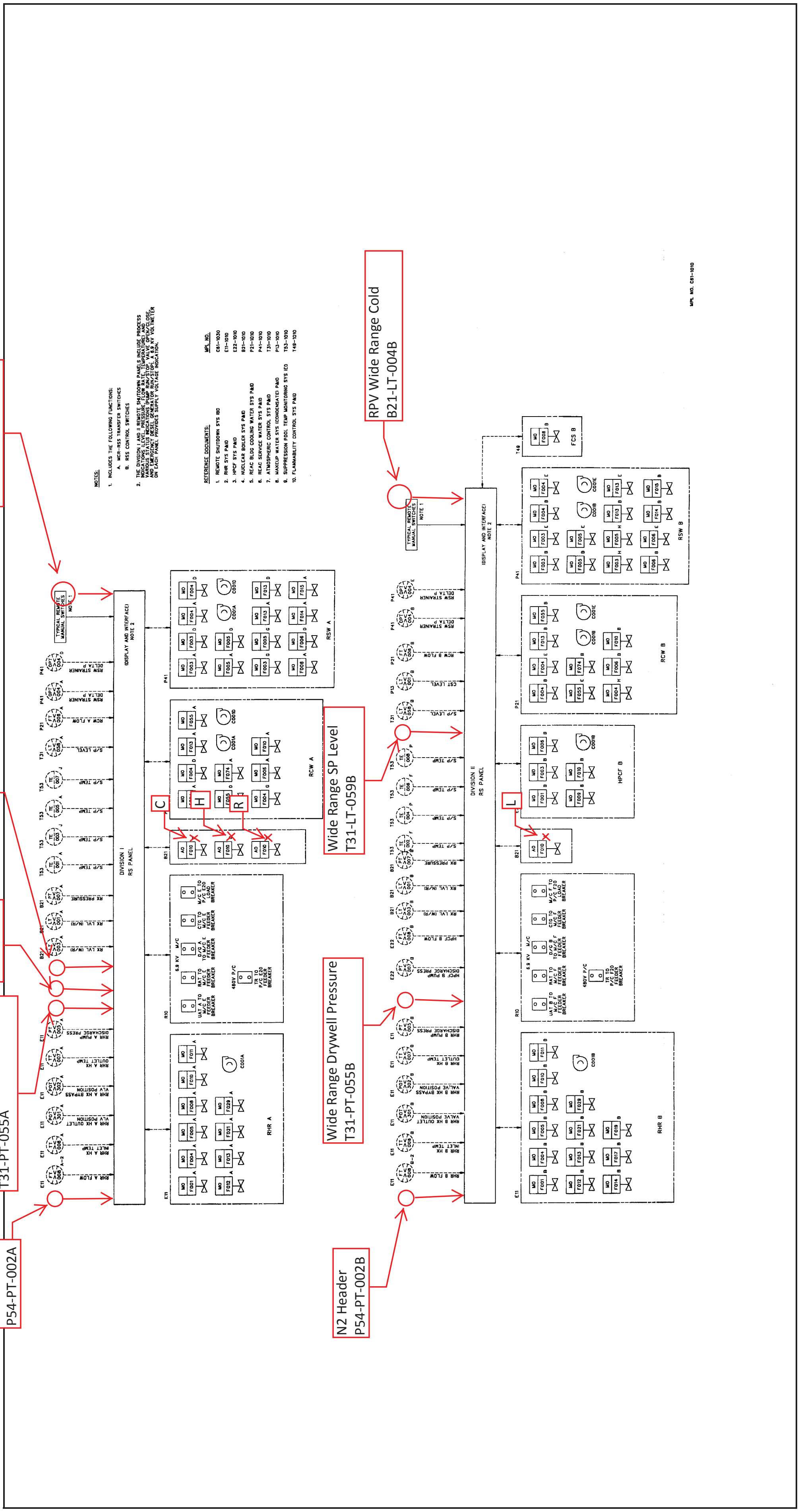


Figure 7.4-2 Remote Shutdown System IED

SYS	NAME	SH NO.	PART NO.
RHR	RHR PUMP (A)	SH 14	E11-C001A
	RHR S/P SUCTION VALVE (A)	SH 7	E11-F001A
	RHR HX BYPASS VALVE (A)	SH 6	E11-F013A
	RHR INJECTION VALVE (A)	SH 15	E11-F005A
	RHR HX OUTLET VALVE (A)	SH 6	E11-F004A
	RHR S/P RETURN VALVE (A)	SH 16	E11-F008A
	RHR SHC (INBOARD) SUCTION ISOL VALVE (A)	SH 21	E11-F010A
	RHR SHC (OUTBOARD) SUCTION ISOL VALVE (A)	SH 22	E11-F011A
	RHR SHC SUCTION VALVE (A)	SH 17	E11-F012A
	RHR MINIMUM FLOW VALVE (A)	SH 12	E11-F021A
	RHR LIQUID WASTE FLUSH ISOLATION VALVE (A)	SH 8	E11-F029A
	RHR PUMP (B)	SH 14	E11-C001B
	RHR S/P SUCTION VALVE (B)	SH 7	E11-F001B
	RHR HX BYPASS VALVE (B)	SH 6	E11-F013B
	RHR INJECTION VALVE (B)	SH 15	E11-F005B
	RHR HX OUTLET VALVE (B)	SH 6	E11-F004B
	RHR D/W SPRAY VALVE (B)	SH 8	E11-F017B
	RHR WW SPRAY VALVE (B)	SH 25	E11-F019B
	RHR S/P RETURN VALVE (B)	SH 16	E11-F008B
	RHR SHC (INBOARD) SUCTION ISOL VALVE (B)	SH 23	E11-F010B
	RHR SHC (OUTBOARD) SUCTION ISOL VALVE (B)	SH 24	E11-F011B
	RHR SHC SUCTION VALVE (B)	SH 18	E11-F012B
	RHR MINIMUM FLOW VALVE (B)	SH 12	E11-F021B
	RHR LIQUID WASTE FLUSH ISOLATION VALVE (B)	SH 8	E11-F029B
	RHR-FPC ISOLATION VALVE (B)	SH 8	E11-F014B

SYS	NAME	SH NO.	PART NO.
HPCF	HPCF PUMP (B)	SH 19	E22-C001B
	HPCF CST SUCTION VALVE (B)	SH 6	E22-F001B
	HPCF INJECTION VALVE (B)	SH 6	E22-F003B
	HPCF S/P SUCTION VALVE (B)	SH 20	E22-F006B
	HPCF TEST LINE BYPASS VALVE (B)	SH 8	E22-F009B
	HPCF MINIMUM FLOW VALVE (B)	SH 13	E22-F010B
NBS	SAFETY RELIEF VALVE (NON ADS)	SH 9	B21-~~F010P~~ F010R
	SAFETY RELIEF VALVE (NON ADS)	SH 9	B21-~~F010G~~ F010C
	SAFETY RELIEF VALVE (NON ADS)	SH 9	B21-~~F010K~~ F010H
	SAFETY RELIEF VALVE (NON ADS)	SH 9	B21-~~F010J~~ F010L
RCW	RCW PUMP (A)	SH 5	P21-C001A
	RCW PUMP (D)	SH 5	P21-C001D
	RCW HX COOLING WATER OUTLET VALVE (A)	SH 6	P21-F004A
	RCW HX COOLING WATER OUTLET VALVE (D)	SH 6	P21-F004D
	RCW HX COOLING WATER OUTLET VALVE (G)	SH 6	P21-F004G
	RCW RHR HX OUTLET VALVE (A)	SH 6	P21-F013A
	RCW D/G OUTLET VALVE (A)	SH 6	P21-F055A
	RCW D/G OUTLET VALVE (D)	SH 6	P21-F055D
	RCW SEPARATE VALVE (A)	SH 6	P21-F074A
	RCW PUMP (B)	SH 5	P21-C001B
	RCW PUMP (E)	SH 5	P21-C001E
	RCW HX COOLING WATER OUTLET VALVE (B)	SH 6	P21-F004B
	RCW HX COOLING WATER OUTLET VALVE (E)	SH 6	P21-F004E
	RCW HX COOLING WATER OUTLET VALVE (H)	SH 6	P21-F004H
	RCW RHR HX OUTLET VALVE (B)	SH 6	P21-F013B
	RCW D/G OUTLET VALVE (B)	SH 6	P21-F055B
	RCW D/G OUTLET VALVE (E)	SH 6	P21-F055E
	RCW SEPARATE VALVE (B)	SH 6	P21-F074B
	RCW COOLING WATER TEMPERATURE CONTROL VALVES	SH 27	P21-F006A(B) P21-F010A(B)

Figure 7.4-3 Remote Shutdown System IBD (Sheet 2 of 27)

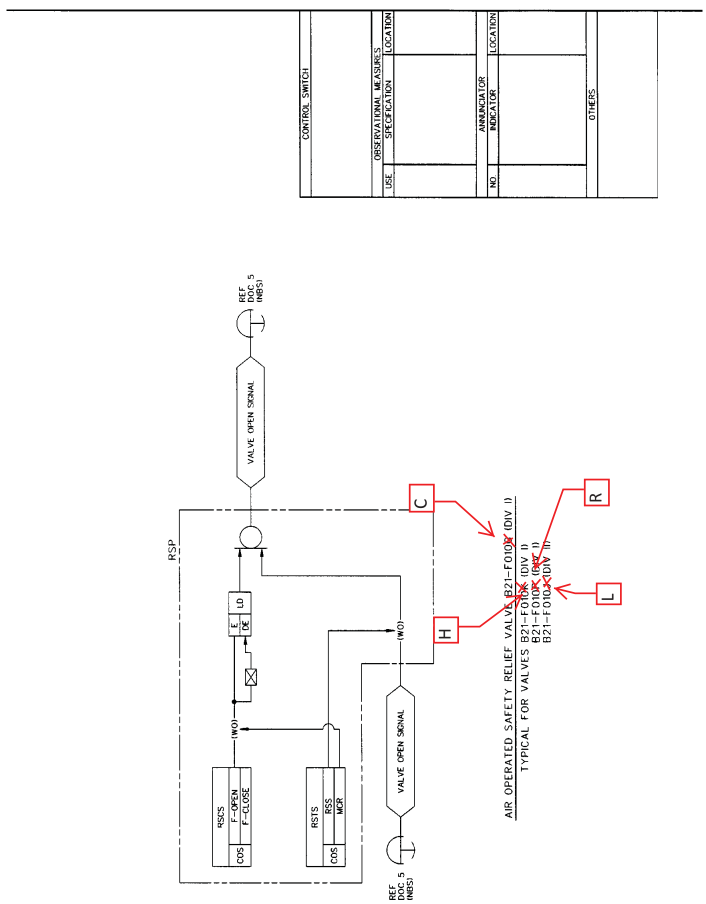


Figure 7.4-3 Remote Shutdown System IBD (Sheet 9 of 27)

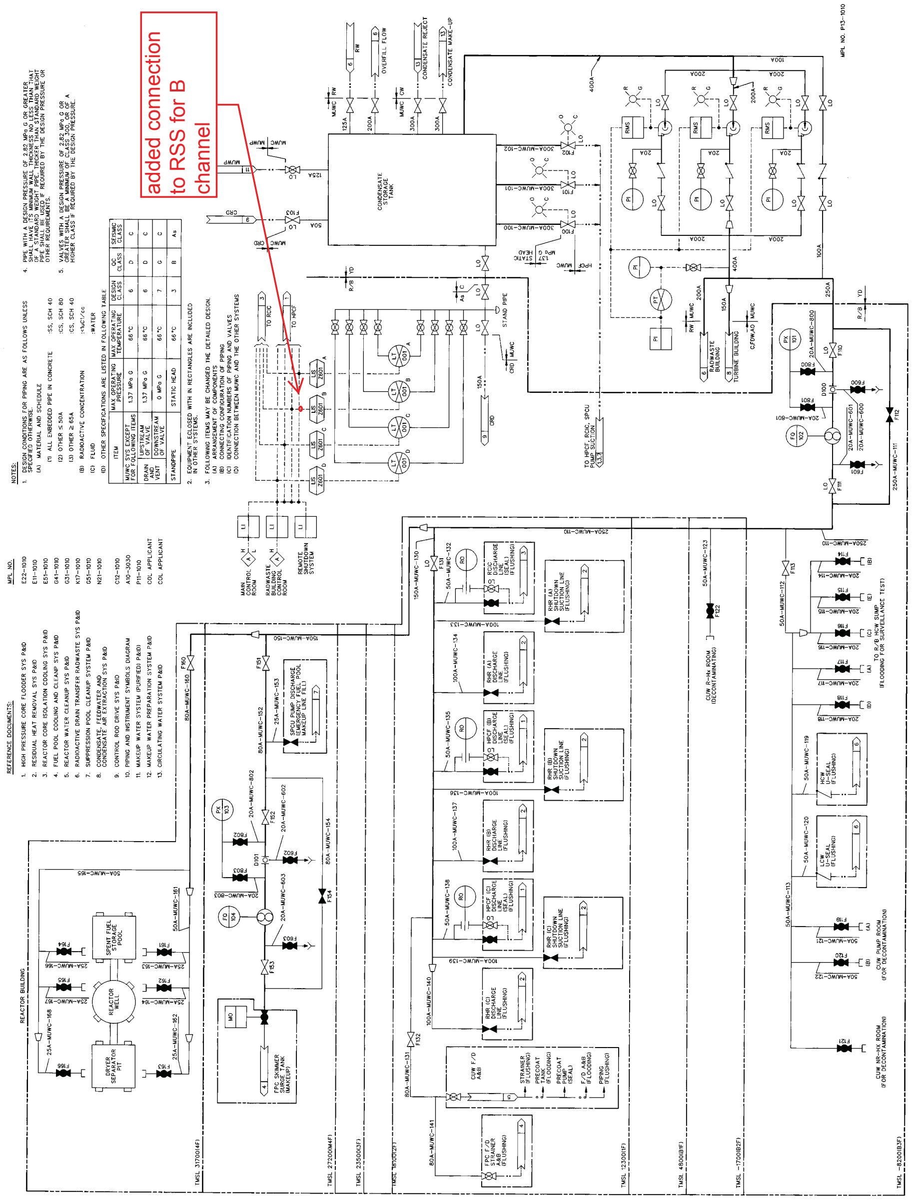


Figure 9.2-4 Makeup Water System (Condensate) P&ID