STP Nuclear Operating Company

Security-Related Information
Withhold Under 10 CFR 2.390

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South Texas Project Electric Generating Station 4000 Avenue F – Suite A Bay City, Texas 77414

June 30, 2009
U7-C-STP-NRC-090070

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTN: Document Control Desk

RE: Application to Amend the Design Certification Rule for the U.S. Advanced Boiling Water Reactor (ABWR)

Pursuant to the provisions on finality of design certification requirements in 10 C.F.R. 52.63(a)(1), STP Nuclear Operating Company (STPNOC) submits this application to amend the Design Certification Rule for the ABWR, to address the requirements of 10 C.F.R. 50.150, the Commission's new aircraft impact rule.

Background

On September 20, 2007, STPNOC submitted a combined license (COL) application to the U.S. Nuclear Regulatory Commission (NRC) for two new units to be located at its existing site in Matagorda County, Texas. The new units are designated South Texas Project Electric Generating Station (STP) Units 3 and 4. The COL application references the ABWR standard design certification. Revisions 1 and 2 of the COL application were submitted to the NRC on January 31, 2008 and September 24, 2008, respectively. The NRC is currently reviewing the COL application, and has designated September 2011 as the target date for issuing the final safety evaluation report.

The NRC published a final rule in the *Federal Register* on June 12, 2009 regarding "Consideration of Aircraft Impacts for New Nuclear Power Reactors." 74 Fed. Reg. 28,112. The new rule requires applicants for new nuclear power reactors to perform a design-specific assessment of the effects of the impact of a large, commercial aircraft. *Id.* The applicant is required to use realistic analyses to identify and incorporate design features and functional capabilities to show, with reduced use of operator actions, that either the reactor core remains cooled or the containment remains intact, and either spent fuel cooling or spent fuel pool integrity is maintained. *Id.* These requirements apply to various categories of applicants, including applicants for combined licenses that reference a standard design certification issued before the effective date of the rule which has not been amended to comply with the rule. If the NRC grants this Application, the ABWR standard design certification will be amended to address the requirements of 10 C.F.R. 50.150, so that STPNOC and other COL applicants that reference the ABWR standard design certification will meet the requirements of the aircraft impact rule.

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Aircraft Impact Assessment and Changes to the ABWR Design Control Document (DCD)

STPNOC and its contractors have performed a design-specific assessment of the effects of a beyond design basis impact of a large, commercial aircraft on the ABWR. Attachment 1 to this Application includes a new Appendix 19S, "Aircraft Impact Assessment," for Tier 2 of the ABWR DCD that provides the information required by 10 C.F.R. 50.150(b) based on the results of this assessment, along with changes to various existing pages of Tier 2 of the ABWR DCD. Appendix 19S describes the design features and functional capabilities identified in the assessment, and how the identified design features and functional capabilities show that, with reduced use of operator actions, the reactor core remains cooled or the containment remains intact, and spent fuel cooling or spent fuel pool integrity is maintained.

STPNOC requests that the NRC amend Appendix A to 10 C.F.R. Part 52 to incorporate by reference Revision 5 of the ABWR DCD, which would incorporate the changes identified in Attachment 1 to this Application. In all other respects, Revision 5 is identical to Revision 4 of the ABWR DCD, which is incorporated by reference in 10 C.F.R. Part 52, Appendix A. The necessary changes to Appendix A to 10 C.F.R. Part 52 are identified in Attachment 2 to this Application. All of the changes to the ABWR DCD are in DCD Tier 2. Attachment 1 to this Application provides annotated pages from DCD Revision 4 to identify the changes that will be incorporated in the ABWR DCD to satisfy the requirements of the new aircraft impact rule. Additionally, designated pages within Attachment 1 contain "Security-Related Information" and should be "Withheld Under 10 C.F.R. 2.390."

The final rule on aircraft impact assessment requires STPNOC, as an applicant that references a standard design certification issued before July 13, 2009 which has not been amended to address the requirements of 10 C.F.R. 50.150, either to apply for amendment of the ABWR design certification rule or to address these requirements in its COL application. STPNOC has elected to address these issues through rulemaking. Because the aircraft impact rule addresses design issues, addressing these issues through rulemaking better promotes standardization. Additionally, addressing these requirements through rulemaking is expected to minimize the impact on NRC resources, because the NRC will only need to review the information once rather than for each new COL application that references the ABWR standard design certification.

Compliance with 10 C.F.R. 52.63(a)(1)

Section VIII.B.1 of the ABWR standard design certification rule states that generic changes to Tier 2 information are governed by the requirements in 10 C.F.R. 52.63(a)(1). This Application complies with the requirements of 10 C.F.R. 52.63(a)(1). Section 52.63(a)(1) allows the Commission to modify a standard design certification rule, whether on its own motion, or in response to an application such as this one, if the Commission determines in a rulemaking that the change will meet one of seven different change criteria. As shown below, this Application satisfies at least two of the change criteria.

First, the proposed change satisfies Section 52.63(a)(1)(vi), as a change that "[s]ubstantially increases overall safety, reliability, or security of facility design, construction, or operation, and the direct and indirect costs of implementation of the rule change are justified in view of this increased safety, reliability, or security." Assessing the effects of the impact of a large, commercial aircraft and demonstrating that the specified acceptance criteria for core cooling or containment and spent fuel pool cooling or integrity are met substantially increases overall safety and security. In adopting the aircraft impact rule, the Commission determined that the direct and indirect implementation costs of compliance with the aircraft impact rule are justified in view of the increased safety and security. Satisfying the requirements of the aircraft impact rule through amending the ABWR DCD also would be less costly for both the NRC and applicants than requiring each individual COL applicant to separately satisfy the requirements.

Second, the proposed change satisfies Section 52.63(a)(1)(vii) as a change that "[c]ontributes to increased standardization of the certification information." This change criterion is satisfied because the changes would be made to the design information found in the design certification rule, which would be applied to all COL applicants referencing the ABWR design certification rule, rather than having each COL applicant address the aircraft impact rule requirements individually. Section VIII.B.2 of the ABWR design certification rule states that generic changes to Tier 2 information are applicable to all applicants. Thus, changing the ABWR DCD to address the requirements of the aircraft impact rule, rather than requiring individual COL applications to separately address these requirements, contributes to increased standardization.

The aircraft impact rulemaking documents also support the above conclusions on the change criteria in Section 52.63(a)(1). In recommending the final rule, the NRC staff evaluated whether changes to the currently approved standard design certifications would satisfy the requirements under Section 52.63(a)(1). SECY 08-0152, Enclosure 1, at 110-14. The NRC staff concluded that imposing the aircraft impact rule on the four existing design certifications meets the criteria in Section 52.63(a)(1)(vi) and (vii). *Id.* at 111. The NRC staff stated:

> The NRC notes that adoption of the aircraft impact rule may indirectly result in the applicant (or another qualified entity) of one of the four existing design certifications voluntarily requesting an amendment to the design certification, in order to address the requirements of the aircraft impact rule . . . . Such changes, which would be accomplished through rulemaking, would also be subject to the change restrictions in 10 CFR 52.63. However, the NRC's bases for determining that the aircraft impact rule meets the change criteria in 10 CFR 52.63(a)(1)(vi) and (vii) would also apply to any design certification amendment rulemaking for the purpose of complying with the aircraft impact rule. Thus, the NRC expects that it would also be able to make the necessary findings under 10 CFR 52.63(a)(1)(vi) and (vii) should it be presented with an application to amend any of the four existing design certifications for the purpose of complying with the aircraft impact rule.

Id. at 114. Thus, the NRC has already concluded that applications such as STPNOC's Application are likely to satisfy Section 52.63(a)(1).

Additional Information

As demonstrated below, this Application also provides the information specified in 10 C.F.R. 2.802(c).

First, Section 2.802(c)(1) requires a statement of "a general solution to the problem or the substance or text of any proposed regulation or amendment." Consistent with this requirement, STPNOC is seeking to amend Appendix A of Part 52 to reference Revision 5 of the ABWR DCD, which includes the changes to Revision 4 specified in Attachment 1 to this Application to comply with the new aircraft impact rule. The text of a proposed amendment of the ABWR design certification rule is provided in Attachment 2.

Second, Section 2.802(c)(2) requires a statement of the "grounds for and interest in the action requested." As discussed above, STPNOC is interested in amending the ABWR DCD to comply with NRC requirements that apply to STPNOC's pending COL application for STP Units 3 and 4, which references the ABWR standard design certification. The grounds for the change include the new aircraft impact rule, the results of STPNOC's assessment of the ABWR design, and the associated changes to the ABWR DCD provided in Attachment 1 to this Application.

Finally, Section 2.802(c)(3) requires a statement of:

> . . . the specific issues involved, the petitioner's views or arguments with respect to those issues, relevant technical, scientific or other data involved which is reasonably available to the petitioner, and such other pertinent information as the petitioner deems necessary to support the action sought.

This Application does this by explaining the issues raised by the new aircraft impact rule that STPNOC must address. The relevant technical, scientific, or other data involved are included in Attachment 1 to this Application, and the detailed assessment referenced in the new ABWR DCD Appendix 19S.

Request for Expedited Review

Because issuance of the COLs for STP Units 3 and 4 depends on approval of this Application and the associated amendment of the Design Certification Rule for the ABWR to address the new aircraft impact rule requirements, STPNOC respectfully requests that the NRC expedite consideration of the Application and the amendment of the ABWR standard design certification.

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If there are any questions regarding this Application, please contact Mr. Scott Head at (361) 972-7136 or me at (361) 972-7206.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 6/30/2009

MA McBurnett

Mark McBurnett
Vice President,
Oversight & Regulatory Affairs

fjp
Attachments:

1. Revisions to the ABWR DCD
2. Revisions to the ABWR Design Certification Rule

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ATTACHMENT 1 – Revisions to the ABWR DCD

1.2.2.12.15 House Boiler System

The House Boiler System consists of the house boilers, reboilers, feedwater components, boiler water treatment and control devices. The House Boiler System supplies turbine gland steam and heating steam, including the concentrating tanks and devices of the high conductivity waste equipment.

1.2.2.12.16 Hot Water Heating System

The Hot Water Heating System is a closed-loop hot water supply to the various heating coils of the HVAC systems. The system includes two heat exchangers, surge and chemical addition tanks and associated equipment, controls and instrumentation.

1.2.2.12.17 Hydrogen Water Chemistry System

The Hydrogen Water Chemistry System is summarized in Subsection 9.3.9.2.

1.2.2.12.18 Zinc Injection System

The Zinc Injection System is summarized in Subsection 9.3.11.1.

1.2.2.12.19 Breathing Air System

The Breathing Air System includes air compressors, dryers, purifiers and a distribution network. This network makes breathing air available in all plant areas where operations or maintenance must be performed and high radioactivity could occur in the ambient air. Special connections are provided to assure that this air is used only for breathing apparatus.

1.2.2.12.20 Sampling System (Includes PASS)

The Process Sampling System is furnished to provide process information that is required to monitor plant and equipment performance and changes to operating parameters. Representative liquid and gas samples are taken automatically and/or manually during plant operation for laboratory or online analyses.

1.2.2.12.21 Freeze Protection System

The Freeze Protection System provides insulation, steam and electrical heating for all external tanks and piping that may freeze during winter weather.

1.2.2.12.22 Iron Injection System

The Iron Injection System consists of an electrolytic iron ion solution generator and means to inject the iron solution into the feedwater system in controlled amounts.

1.2.2.12.23 Alternate Feedwater Injection System

The Alternate Feedwater Injection (AFI) System is summarized in Subsection 9.5.14.

1.2.2.16.11 Turbine Building

The Turbine Building houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.

1.2.2.16.12 Control Building

The Control Building includes the control room, the computer facility, the cable tunnels, some of the plant essential switchgear, some of the essential power, reactor building water system and the essential HVAC system.

1.2.2.16.13 Radwaste Building

The Radwaste Building houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

1.2.2.16.14 Service Building

The Service Building houses the personnel facilities and portions of the non-essential HVAC System.

1.2.2.16.15 Alternate Feedwater Injection (AFI) Pump House

The Alternate Feedwater Injection Pump House, which is located remotely from the Reactor Building, contains the additional equipment, such as the AFI pump, piping and valves and additional SRV nitrogen supply, which support the AFI function.

1.2.2.17 Yard Structures and Equipment

1.2.2.17.1 Stack

The plant stack is located on the Reactor Building and rises to an elevation of 76 meters above grade level. The stack is a steel shell construction supported by an external steel tubular frame work. The stack vents the Reactor Building, Turbine Building, Radwaste Building, and a small portion of the Control and Service buildings.

1.2.2.17.2 Oil Storage and Transfer System

The major components of this system are the fuel-oil storage tanks, pumps, and day tanks. Each diesel generator has its own individual supply components. Each storage tank is designed to supply the diesel needs during the post-LOCA period, and each day tank has capacity for 8 hours of diesel generator operation at maximum LOCA load demand. Each fuel oil pump is controlled automatically by day-tank level and feeds its day tank from the storage tank. Additional fuel oil pumps supply fuel to each diesel fuel manifold from the day tank.

1.2.2.17.3 Site Security

Site Security is summarized in Subsection 13.6.3.1.

Experiences related to identified regulatory or industry developed resolutions were eliminated to avoid repetition except for selected experiences that have a nuisance potential for reoccurring. Lead system engineers classified the more complex experiences.

Reference to the new or novel design features used in the ABWR are provided below:

Feature	**Tier 2 Section**
Fine Motion Control Rod Drive	4.6
Internal Reactor Pumps	5.4.1
Multiplexing	7A.2
Digital/Solid-State Control	7A.7
Overpressure Protection System	6.2.5.2.6, 6.2.5.3, 6.2.5.4
AC-Independent Water Addition System	5.4.7.1.1.10
Lower Drywell Flooder	9.5.1.2
Alternate Feedwater Injection	9.5.14

1.8.4 COL License Information

1.8.4.1 SRP Deviations

The SRP sections to be addressed by the COL applicant are indicated in the comments column of Table 1.8-19 as "COL Applicant". Where applicable the COL applicant will provide the information required by 10CFR50.34(g) similar to Tables 1.8-1 through 1.8-18 (see Subsection 1.8.1).

1.8.4.2 Experience Information

The experience information to be addressed by the COL applicant are indicated in the comment column of Table 1.8-22 as "COL Applicant" (see Subsection 1.8.3).

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
9.22	Vendor Specific Design of Diesel Generator Auxiliaries	9.5.13.5
9.23	Diesel Generator Cooling Water System Design Flow and Heat Removal Requirements	9.5.13.6
9.24	Fire Rating for Penetration Seals	9.5.13.7
9.25	Diesel Generator Requirements	9.5.13.8
9.26	Applicant Fire Protection Program	9.5.13.9
9.27	HVAC Pressure Calculations	9.5.13.10
9.28	Plant Security System Criteria	9.5.13.11
9.29	Not Used	9.5.13.12
9.30	Diesel Fuel Refueling Procedures	9.5.13.13
9.31	Portable and Fixed Emergency Communication Systems	9.5.13.14
9.32	Identification of Chemicals	9.5.13.15
9.33	NUREG/CR-0660 Diesel Generator Reliability Recommendations	9.5.13.16
9.34	Sound-Powered Telephone Units	9.5.13.17
9.35	Fire-Related Administrative Controls	9.5.13.18
9.36	Periodic Testing of Combustion Turbine Generator (CTG)	9.5.13.19
9.37	Operating Procedures for Station Blackout	9.5.13.20
9.38	Quality Assurance Requirements for CTG	9.5.13.21
9.39	Power Supply for Alternate Feedwater Injection Equipment	9.5.13.22
9.40	Test and Surveillance Intervals for Alternate Feedwater Injection Equipment	9.5.13.23
10.1	Low Pressure Turbine Disk Fracture Toughness	10.2.5.1
10.2	Turbine Design Overspeed	10.2.5.2
10.3	Turbine Inservice Test and Inspection	10.2.5.3
10.4	Procedures to Avoid Steam Hammer and Discharge Loads	10.3.7.1
10.5	MSIV Leakage	10.3.7.2
10.6	Radiological Analysis of the TGSS Effluents	10.4.10.1
11.1	Plant-Specific Liquid Radwaste Information	11.2.5.1
11.2	Compliance With Appendix I to 10CFR50	11.3.11.1
11.3	Plant-Specific Solid Radwaste Information	11.4.3.1
11.4	Calculation of Radiation Release Rates	11.5.6.1
11.5	Compliance with the Regulatory Shielding Design Basis	11.5.6.2

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
19.9	Action to Mitigate Station Blackout Events	19.9.9
19.10	Actions to Reduce Risk of Internal Flooding	19.9.10
19.11	Actions to Avoid Loss of Decay Heat Removal and Minimize Shutdown Risk	19.9.11
19.12	Procedures for Operation of RCIC from Outside the Control Room	19.9.12
19.13	ECCS Test and Surveillance Intervals	19.9.13
19.14	Accident Management	19.9.14
19.15	Manual Operation of MOVs	19.9.15
19.16	High Pressure Core Flooder Discharge Valve	19.9.16
19.17	Capability of Containment Isolation Valves	19.9.17
19.18	Procedures to Ensure Sample Lines and Drywell Purge Lines Remain Closed During Operation	19.9.18
19.19	Procedures for Combustion Turbine Generator to Supply Power to Condensate Pumps	19.9.19
19.19a	Actions to Assure Reliability of the Supporting RCW and Service Water Systems	19.9.20
19.19b	Housing of AICWA Equipment	19.9.21
19.19c	Procedures to Assure SRV Operability During Station Blackout	19.9.22
19.19d	Procedures for Ensuring Integrity of Freeze Seals	19.923
19.19e	Procedures for Controlling Combustibles During Shutdown	19.9.24
19.19f	Outage Planning and Control	19.9.25
19.19g	Reactor Service Water Systems Definition	19.9.26
19.19h	Capability of Vacuum Breakers	19.9.27
19.19i	Capability of the Containment Atmosphere Monitoring System	19.9.28
19.19j	Plant Specific Safety-Related Issues and Vendors Operating Guidance	19.9.29
19.19k	Procedures for Use of Alternate Feedwater Injection	19.9.31
19.19l	Procedures to Depressurize the RPV from the AFI Pump House	19.9.32
19.19m	Verification of Environmental Conditions in AFI Pump House	19.9.33
19.19n	Description of Electrical Power Supply for AFI Equipment	19.9.35
19.20	Long-Term Training Upgrade	19A.3.1

Table 3.2-1 Classification Summary (Continued)

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

Item No.	MPL Number†	Title
P8	P40	Ultimate Heat Sink
P9	P41	Reactor Service Water System
P10	P42	Turbine Service Water System
P11	P51	Station Instrument Air System
P12	P52	Instrument Air System
P13	P54	High Pressure Nitrogen Gas Supply System
P14	P61	Heating Steam and Condensate Water Return System
P15	P62	House Boiler
P16	P63	Hot Water Heating System
P17	P73	Hydrogen Water Chemistry System
P18	P74	Zinc Injection System
P19	P81	Breathing Air System
P20	P91	Sampling System (Includes PASS)
P21	P92	Freeze Protection System
P22	P95	Iron Injection System
P23	P15	Alternate Feedwater Injection System
R Station Electrical Systems		
R1	R10	Electrical Power Distribution System
R2	R11	Unit Auxiliary Transformer
R3	R13	Isolated Phase Bus
R4	R21	Non-Segregated Phase Bus
R5	R22	Metalclad Switchgear
R6	R23	Power Center
R7	R24	Motor Control Center
R8	R31	Raceway System

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

† Master Parts List Number designated for the system.

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

Table 3.2-1 Classification Summary (Continued)

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

Item No.	MPL Number†	Title
U1	U21	Foundation Work
U2	U24	Turbine Pedestal
U3	U31	Cranes and Hoists
U4	U32	Elevator
U5	U41	Heating, Ventilating and Air Conditioning‡
U5.1	U42	Potable and Sanitary Water System
U6	U43	Fire Protection System
U7	U46	Floor Leakage Detection System
U8	U47	Vacuum Sweep System
U9	U48	Decontamination System
U10	U71	Reactor Building‡
U11	U72	Turbine Building‡
U12	U73	Control Building‡
U13	U74	Radwaste Building
U14	U75	Service Building
U15	U83	Alternate Feedwater Injection Pump House
Y Yard Structures and Equipment		
Y1	Y31	Stack
Y2	Y52	Oil Storage and Transfer System
Y3	Y86	Site Security

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

† Master Parts List Number designated for the system.

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

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]		Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
	6. Other non-safety-related electrical components	N	SC,RZ,X	—	E	—	
P14	**Heating Steam and Condensate Water Return System**	N	T,SC,W	—	E	—	
P15	**House Boiler**	N	T	—	E	—	
P16	**Hot Water Heating System**	N	T	—	E	—	
P17	**Hydrogen Water Chemistry System**	N	T	—	E	—	
P18	**Zinc Injection System**	N	T	—	E	—	
P19	**Breathing Air System**	N	C,SC,T	—	E	—	
P20	**Sampling System (Includes PASS)**	N	SC,RZ,T	—	E	—	
P21	**Freeze Protection System**	N	O	—	E	—	
P22	**Iron Injection System**	N	T	—	E	—	
P23	**Alternate Feedwater Injection System**						
	1. Pumps, Valves, Piping	N	A	—	E	—	

Notes and footnotes are listed on pages 3.2-53 through 3.2-60

Table 3.2-1 Classification Summary (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
7. Cables	N	SC,X,RZ, H,T,W,F	—	E	—	(t) (u)
8. Sprinklers or deluge water	N	H,W,SC, RZ,T,O	D	E	—	(t) (u)
9. Foam, reaction or deluge	N	RZ,T	—	E	—	(t) (u)
U7 Floor Leakage Detection System	N	SC,RZ	—	E	—	
U8 Vacuum Sweep System	N	C,SC	—	E	—	
U9 Decontamination System	N	C,SC,RZ T,W,S,X	—	E	—	
U10 Reactor Building	3	C,SC,RZ, M	—	B	I	
U11 Turbine Building	N	T	—	E	—	(v)
U12 Control Building	3	X	—	B	I	
U13 Radwaste Building						
1. Structural walls and slabs above grade level (see Subsection 3H.3.3)	N	W	—	E	—	
2. Radwaste Building Substructure	3	W	—	B	I	
U14 Service Building	N	H	—	E	—	
U15 Alternate Feedwater Injection Pump House	N	A	—	E	—	

Notes and footnotes are listed on pages 3.2-53 through 3.2-60

c. C = Primary Containment
H = Service Building
M = Reactor Building steam tunnel
O = Outside onsite
RZ = Reactor Building Clean Zone (balance portion of the reactor building outside the Secondary Containment Zone)
SC = Secondary Containment portion of the reactor building
T = Turbine Building
W = Radwaste Building
X = Control Building
F = Firewater Pump House*
U = Ultimate Heat Sink Pump House*
P = Power Cycle Heat Sink Pump House*
A = Alternate Feedwater Injection Pump House

d. A,B,C,D= Quality groups defined in Regulatory Guide 1.26 and Subsection 3.2.2. The structures, systems and components are designed and constructed in accordance with the requirements identified in Tables 3.2-2 and 3.2-3.

--- = Quality Group Classification not applicable to this equipment.

e. B = The quality assurance requirements of 10CFR50, Appendix B are applied in accordance with the quality assurance program described in Chapter 17.

E = Elements of 10CFR50, Appendix B are generally applied, commensurate with the importance of the equipment's function.

f. I = The design requirements of Seismic Category I structures and equipment are applied as described in Section 3.7, Seismic Design.

--- = The seismic design requirements for the safe shutdown earthquake (SSE) are not applicable to the equipment. However, the equipment that is not safety-related but which could damage Seismic Category I equipment if its structural integrity failed is checked analytically and designed to assure its integrity under seismic loading resulting from the SSE.

g. 1. Lines 25A and smaller which are part of the reactor coolant pressure boundary and are ASME Code Section III, Class 2 and Seismic Category I.

* Pump House structures are out of the ABWR Standard Plant scope.

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)
F010	2	Bypass line around the N_2 bottle supply line PCV	3	B	P		E1	6.7-1
F011	2	N_2 bottle supply line relief valve	3	C	A	R	5 yr	6.7-1
F012	2	MOV at safety/non-safety boundary	3	A	A	P S	2 yr 3 mo	6.7-1
F200	1	Non-safety N2 supply line isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.7-1
F209	1	Non-safety N2 supply line isolation check valve	2	A,C	I,A	L,S	RO	6.7-1
F301	1	Non-safety N2 supply line isolation check valve	2	A,C	I,A	L,S	RO	6.7-1
F302	1	Non-safety N2 supply line isolation check valve	2	A,C	I,A	L,S	RO	6.7-1
T22 Standby Gas Treatment System Valves								
F001	2	Fuel handling floor inlet butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 1
F002	2	Filter train inlet butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 1
F003	2	Filter train exhaust gravity damper	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 2,3
F004	2	Filter train exhaust butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 2,3
F005	2	Cooling fan butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 2,3
F006	2	Filter train R112 injection line valve	3	B	P		E1	6.5-1 sh. 2,3
F007	2	Filter train DOP injection line valve to pre HEPA filter	3	B	P		E1	6.5-1 sh. 2,3
F008	2	Filter train DOP sampling line valve downstream of pre HEPA	3	B	P		E1	6.5-1 sh. 2,3
F009	2	Filter train DOP sampling line valve downstream of pre HEPA	3	B	P		E1	6.5-1 sh. 2,3
F010	2	Filter train DOP injection line valve downstream of charcoal absorbent	3	B	P		E1	6.5-1 sh. 2,3
F011	2	Filter train DOP sampling line valve downstream of charcoal absorbent	3	B	P		E1	6.5-1 sh. 2,3

The second part of the SRV discharge piping extends from the diaphragm floor penetration to the SRV quencher in the suppression pool. Because the diaphragm floor acts as an anchor on this part of the line, it is physically decoupled from the main steam header.

As a part of the preoperational and startup testing of the main steamlines, movement of the SRV discharge lines will be monitored.

The SRV discharge piping is designed to limit valve outlet pressure to approximately 40% of maximum valve inlet pressure with the valve wide open. Water in the line more than about 1/2 of a meter above suppression pool water level would cause excessive pressure at the valve discharge when the valve is again opened. For this reason, two vacuum relief valves are provided on each SRV discharge line to prevent drawing an excessive amount of water into the line as a result of steam condensation following termination of relief operation. The SRVs are located on the main steamline piping rather than on the reactor vessel top head, primarily to simplify the discharge piping to the pool and to avoid the necessity of having to remove sections of this piping when the reactor head is removed for refueling. In addition, valves located on the steamlines are more accessible during a shutdown for valve maintenance.

The ADS automatically depressurizes the nuclear system sufficiently to permit the LPFL mode of the RHR System to operate as a backup for the HPCF. Further descriptions of the operation of the automatic depressurization feature are presented in Section 6.3 and Subsection 7.3.1.

In addition to playing a major role in preventing core damage, depressurization of the RPV (either manually, automatically, or as a result of a LOCA) can help mitigate the consequences of severe accidents in which fuel melting and vessel failure occur. If the RPV were to fail at an elevated pressure (greater than approximately 1.37 MPaG) high pressure melt injection could occur resulting in fragmented core debris being transported into the upper drywell. The resulting heatup of the upper drywell could pressurize and fail the drywell. This failure mechanism is eliminated if the RPV is depressurized. The opening of a single SRV is capable of depressurizing the vessel sufficiently to prevent high pressure melt ejection.

One of the non-ADS safety/relief valves is provided with an additional solenoid valve and a nitrogen supply line which can supply nitrogen from the AFI Pump House. A nitrogen supply connection in the Pump House allows the use of a portable nitrogen bottle located in the Pump House. This provides a separate and diverse means of depressurizing the RPV that is independent from the nitrogen supply in the Reactor Building to the safety/relief valves. The additional solenoid is normally de-energized and may be energized with DC power supplied from the AFI Pump House.

5.2.2.4.2 Design Parameters

The specified operating transients for components within the RCPB are presented in Subsection 3.9.1. Subsection 3.7.1 provides a discussion of the input criteria for design of Seismic Category I structures, systems, and components. The design requirements established to protect

Table 6.2-7 Containment Isolation Valve Information High Pressure Nitrogen Gas Supply System

Valve No.	**P54-F007A/F008A**	**P54-F007B/F008B**	**P54-F200/F209**	**P54-F301/F302**
Tier 2 Figure	6.7-1	6.7-1	6.7-1	6.7-1
Applicable Basis	GDC 57	GDC 57	GDC 57	GDC 57
Fluid	N_2	N_2	N_2	N_2
Line Size	50A	50A	50A	50A
ESF	Yes	Yes	Yes	Yes
Leakage Class	(b)	(b)	(b)	(b)
Location	O/I	O/I	O/I	O/I
Type C Leak Test	No(r)	No(r)	Yes	Yes
Valve Type	Globe/Check	Globe/Check	Globe/Check	Spring Check/Check
Operator	Motor/Self	Motor/Self	Motor/Self	Pneumatic/Self
Primary Actuation	Electrical/N/A	Electrical/N/A	Electrical/N/A	N2 to open/N/A
Secondary Actuation	HW/N/A	HW/N/A	HW/N/A	N/A
Normal Position	Open	Open	Open	Close/Close
Shutdown Position	Open	Open	Open	Close/Close
Post-Accident Position	Close	Close	Close	Close/Close
Power Fail Position	As Is/N/A	As Is/N/A	As Is/N/A	N/A
Containment Isolation Signal[(c)]	GG (Y)	GG(Y)	GG(Y)	N/A
Closure Time (s)	30 / Instantaneous	30 / Instantaneous	30 / Instantaneous	Instantaneous
Power Source (Div)	I/N/A	II/N/A	I/N/A	N/A

See page 6.2-167 for notes

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

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-37	RCIC Turbine Steam	14450	80	1200	550		A
X-38	RPV Head Spray	14450	310	1500	550		A
X-50	CUW Pump Feed	14480	310	0	600		A
X-60	MUWP Suction	13500	290	0	200		A
X-61	RCW Suction (A)	13500	45	-3000	200		A
X-62	RCW Return (A)	13500	45	-2000	200		A
X-63	RCW Suction (B)	13500	225	3400	200		A
X-64	RCW Return (B)	13500	225	2400	200		A
X-65	HNCW Suction	13500	225	250	350		A
X-66	HNCW Return	13500	225	1400	350		A
X-69	SA	19000	42	0	90		A
X-70	IA	9000	46	0	200		A
X-71A	ADS Accumulator (A)	19000	50	0	200		A
X-71B	ADS Accumulator (B)	19000	296.5	1000	200		A
X-72	Relief Valve Accumulator	19000	296.5	2000	200		A
X-73	HPIN	13500	0	-4550	200		A
X-80	Drywell Purge Suction	13700	68	0	550		A
X-81	Drywell Purge Exhaust	19000	216	0	550		A
X-82	FCS Suction	14850	225	-600	150		A
X-90	Spare	20100	46	0	400		A
X-91	Spare	20100	296.5	1000	400		A
X-92	Spare	16400	45	12700	400		A
X-93	Spare	14700	135	-500	400		A

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

Table 6.2-9 Secondary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Diameter (mm)
60	HPIN	-1700	50
61	AFI	-1700	150
62	AFI (Drain Line)**	-8200	20

* This table is provided in response to Question 430.34.

† These HVAC openings have safety-related isolation valves with both local monitoring and remote (in control room) monitoring.

‡ These doors are monitored in the control room as per Subsection 13.6.3.4.

** Only required if two normally closed MOVs located outside Reactor Building.

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

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-65	HNCW Suction	350	E	E/D/H	No
X-66	HNCW Return	350	E	E/D/H	No
X-69	SA	90	E	E/D/H	No
X-70	IA	200	E	E/D/H	No
X-71A	ADS Accumulator (A)	200	S	C/K	No
X-71B	ADS Accumulator (B)	200	S	C/K	No
X-72	Relief Valve Accumulator	200	S	C/K	No
X-73	HPIN	200	A	E/D/H	No
X-80	Drywell Purge Suction	550	E	E/C/J	Yes
X-81	Drywell Purge Exhaust	550	E	E/C/J	Yes
X-82	FCS Suction	150	S	E/C/H	No
X-90	Spare	400	P	B/A	No
X-91	Spare	400	P	B/A	No
X-92	Spare	400	P	B/A	No
X-93	Spare	400	P	B/A	No
X-100A	IP Power	450	S	C/J	No
X-100B	IP Power	450	S	C/J	No
X-100C	IP Power	450	S	C/J	No
X-100D	IP Power	450	S	C/J	No
X-100E	IP Power	450	S	C/J	No
X-101A	LP Power	300	S	C/J	No
X-101B	LP Power	300	S	C/J	No
X-101C	LP Power	300	S	C/J	No
X-101D	FMCRD Power	300	S	C/J	No
X-101E	FMCRD Power	300	S	C/J	No
X-101F	FMCRD Power	300	S	C/J	No
X-101G	FMCRD Power	300	S	C/J	No
X-102A	I & C	300	S	C/J	No
X-102B	I & C	300	S	C/J	No
X-102C	I & C	300	S	C/J	No
X-102D	I & C	300	S	C/J	No

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

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-660B	TIP Drive	50	S	C/J	No
X-660C	TIP Drive	50	S	C/J	No
X-660D	TIP Drive Purge	50	S	C/K	No
X-680A	Spare	40	S	C/J	No
X-680B	Spare	40	S	C/J	No
X-700A	RIP Purge Water Supply	35	S	C/H	No
X-700B	RIP Purge Water Supply	35	S	C/H	No
X-700C	RIP Purge Water Supply	35	S	C/H	No
X-700D	RIP Purge Water Supply	35	S	C/H	No
X-700E	RIP Purge Water Supply	35	S	C/H	No
X-700F	RIP Purge Water Supply	35	S	C/H	No
X-700G	RIP Purge Water Supply	35	S	C/H	No
X-700H	RIP Purge Water Supply	35	S	C/H	No
X-700J	RIP Purge Water Supply	35	S	C/H	No
X-700K	RIP Purge Water Supply	35	S	C/H	No
X-750A	I&C (Core Diff Press.)	180	S	C/J	No
X-750B	I&C (Core Diff Press.)	180	S	C/J	No
X-750C	I&C (Core Diff Press.)	180	S	C/J	No
X-750D	I&C (Core Diff Press.)	180	S	C/J	No
X-751A	I&C (RIP Diff Press.)	180	S	C/J	No
X-751B	I&C (RIP Diff Press)	180	S	C/J	No
X-751C	I&C (RIP Diff Press)	180	S	C/J	No
X-751D	I&C (RIP Diff Press)	180	S	C/J	No
X-780A	Spare	180	S	B/A	No
X-780B	Spare	180	S	B/A	No

* This table is provided in response to Question 430.52b.

† E - Environment
P - Primary containment
S - Secondary containment
A - AFI pump house

6.7 High Pressure Nitrogen Gas Supply System

6.7.1 Functions

The High Pressure Nitrogen Gas Supply (HPIN) System is divided into two independent divisions, with each division containing a safety-related emergency stored nitrogen supply. The safety-related stored nitrogen supply is Safety Class 3, Seismic Category I, designed for operation of the main steam SRV ADS function accumulators.

The functions of the non-safety-related, makeup nitrogen gas supply system include providing nitrogen for:

(1) Relief function accumulators of main steam SRVs

(2) Pneumatically-operated valves and instruments inside the PCV

(3) Leak detection system radiation monitor calibration

(4) ADS function accumulators to compensate for the leakage from main steam SRV solenoid valves during normal operation

6.7.2 System Description

Normally, nitrogen gas for both safety-related and non-safety-related makeup systems is supplied from the nitrogen gas evaporator via the makeup line to the Atmospheric Control System (ACS). The nitrogen supply system shall supply nitrogen which is oil-free with a moisture content of less than 2.5 ppm. This nitrogen is filtered in the HPIN System to remove particles larger than 5 µm. All equipment using this nitrogen shall be capable of operating with nitrogen of the quality listed above. If nitrogen is not available from the ACS, nitrogen is supplied from high pressure nitrogen gas storage bottles. An additional non-safety related nitrogen gas storage bottle capable of supplying nitrogen to one of the non-ADS safety/relief valves from the AFI Pump House is added to allow system depressurization in the event of loss of nitrogen supply in the Reactor Building. The safety-related system is separated into two divisions. There are tielines between the non-safety-related and each division of the safety-related system. Each tieline has a motor-operated shutoff valve (See Figure 6.7-1 and Table 6.7-1 for details).

During operation, all SRV accumulators are supplied from the non-divisional system. If the pressure sensor in either of the safety-related systems indicates low pressure, the valve between that system and the non-divisional system closes and the supply valve to the bottled nitrogen supply in that division opens. If the pressure sensor in the non-divisional system indicates a low pressure, the valves between the non-divisional and the divisional systems close. (See Figure 7.3-10)

Each division of the safety-related system has ten bottles. Normally, outlet valves from five of the ten bottles are kept open. Each division has a pressure control valve to depressurize the

(d) Fuel Zone Water Level Range

This range uses the RPV taps at the elevation near the bottom of the dryer skirt and the taps below the top of the active fuel (above the pump deck). The zero of the instrument is the top of the active fuel and the instruments are calibrated to be accurate at 0 PaG and saturated condition. The water level measurement design is the condensate reference type and uses differential pressure devices as its primary element.

(e) Reactor Well Water Level Range

This range uses the RPV tap below the top of the active fuel. The zero of the instrument is the top of the active fuel. The temperature and pressure condition that is used for the calibration is 0 MPaG and 48.9°C water in the vessel. The water level measurement design is the pressure device which measures static water pressure inside the vessel and converts to a water level indication. This range is used to monitor the reactor water level when the reactor vessel head is removed and the reactor system is flooded during the refueling outage.

The condensate reference chamber for the narrow range and wide range water level range is common as discussed in Section 7.3

The concern that non-condensable gasses may build-up in the water column in the reactor vessel reference leg water level instrument lines, i.e., the reactor vessel instrument lines at the elevation near the main steam line nozzles, has been addressed by continually flushing these instrument lines with water supplied by the Control Rod Drive (CRD) System.

Reactor water level instrumentation that initiates safety systems and engineered safeguards systems is discussed in Subsections 7.2.1 and 7.3.1. Reactor water level instrumentation that is used as part of the Feedwater Control System is discussed in Subsection 7.7.1.4.

Reactor water level instrumentation that is provided in the alternate feedwater injection (AFI) Pump House is discussed in Subsection 9.5.14.

(7) Reactor Core Hydraulics

A differential pressure transmitter indicates core plate pressure drop by measuring the core inlet plenum and the space just above the core support assembly. The instrument sensing line used to determine the pressure below the core support assembly attaches to the same reactor vessel tap that is used for the injection of the liquid from the Standby Liquid Control System (SLCS). An instrument sensing line is provided for measuring pressure above the core support assembly. The differential pressure of the core plate is indicated locally and recorded in the main control room.

Another differential pressure device indicates the reactor internal pump developed head by measuring the pressure difference between the pressure above and below the pump deck.

(8) Reactor Vessel Pressure

Pressure indicators and transmitters detect reactor vessel internal pressure from the same instrument lines used for measuring reactor vessel water level.

The following list shows the subsection in which the reactor vessel pressure measuring instruments are discussed.

(a) Pressure transmitters and trip actuators for initiating scram, and pressure transmitters and trip actuators for bypassing the MSIV closure scram, are discussed in Subsection 7.2.1.1.

(b) Pressure transmitters and trip actuators used for RCIC and LPFL are discussed in Subsection 7.3.1.1.

(c) Pressure transmitters and recorders used for feedwater control are discussed in Subsection 7.7.1.4.

(d) Pressure transmitters that are used for pressure recording are discussed in Section 7.5.

(e) The pressure transmitter that is used for providing reactor vessel pressure indication in the AFI Pump House is discussed in Subsection 9.5.14.

(9) Pressure between the inner and outer reactor vessel head seal ring is sensed by a pressure transmitter. If the inner seal fails, the pressure at the pressure transmitter is the vessel pressure, and the associated trip actuator will trip and actuate an alarm. The plant will continue to operate with the outer seal as a backup, and the inner seal can be repaired at the next outage when the head is removed. If both the inner and outer head seals fail, the leak will be detected by an increase in drywell temperature and pressure.

(10) Safety/Relief Valve Seal Leak Detection

Thermocouples are located in the discharge exhaust pipe of the safety/relief valve. The temperature signal goes to a multipoint recorder with an alarm and will be activated by any temperature in excess of a set temperature signaling that one of the SRV seats has started to leak.

(11) Other Instruments

The feedwater temperature is measured and transmitted to the main control room.

(e) Environment Considerations

Environmental conditions are the same for the normal condition and the accident condition because there are no high-energy systems in the area (Section 3.11).

(f) Operational Considerations

There are no special operating considerations.

7.7.1.11 Other Non-Safety-Related Control Systems

The following non-safety-related control systems are described in other Tier 2 subsections as indicated.

System	Subsection
Fire Protection	9.5.1
Offgas/Radwaste	11.2, 11.3, 11.4
Drywell Cooling	9.4.8
Sampling	9.3.2
Instrument Air	9.3.6
Makeup Water	9.2.3
Atmospheric Control	6.2.5
Alternate Feedwater Injection	9.5.14

7.7.2 Analysis

The purpose of this subsection is to:

(1) Demonstrate by direct or referenced analysis that the subject-described systems are not required for any plant safety function.

(2) Demonstrate by direct or referenced analysis that the plant protection systems described elsewhere are capable of coping with all failure modes of the subject control system.

In response to item (1) above, the following is cited: upon considering the design basis, descriptions, and evaluations presented here and elsewhere throughout the document relative to

(b) **Conformance:** The FPC System is in compliance with these GDCs, in part, or as a whole, as applicable. The GDCs are generally addressed in subsection 3.1.2. Instrumentation and controls are provided in the control room. The filter/demineralizer portion is controllable from the local panels. Since the system is not associated with reactor shutdown, there are no controls needed nor provided in the remote shutdown facility.

(2) Regulatory Guide (RGs)

In accordance with the Standard Review Plan for Section 7.7 and with Table 7.1-2, only Regulatory Guide 1.151 ("Instrument Sensing Lines") need be addressed for the ABWR. The FPC instrument lines are not exposed to cold temperatures and are designed to meet the ASME code requirements of RG 1.151 and ISA S67.02. The FPC System is thus in full compliance with these criteria.

7.7.2.11 Other Non-Safety-Related Control Systems

The following non-safety-related control systems are described in other subsections of the SSAR as indicated.

System	Subsection
Fire Protection	9.5.1
Offgas/Radwaste	11.2, 11.3, 11.4
Drywell Cooling	9.4.8
Sampling	9.3.2
Instrument Air	9.3.6
Makeup Water	9.2.3
Atmospheric Control	6.2.5
Reactor Water Cleanup	5.4.8
Alternate Feedwater Injection	9.5.14

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

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

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

(b) Provision of raised supports for the equipment

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

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

(12) Fire Containment or Inhibiting Methods Employed:

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

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

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

(13) Remarks—The corridor contains piping and cable trays in its upper elevation.

9A.4.1.2.4 RHR (A)/RCIC Pipe Space (Rm No. 212)

(1) Fire Area—F1100

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

Safety--Related	Provides Core Cooling
Yes, D1	Yes, D1

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

(4) Qualifications of Fire Barriers—The room is within division 1 fire area F1100. The wall common with pipe space C (Rm 230) serves as a fire barrier and is of 3 h fire-resistive concrete construction. The ceiling and floor are concrete but are not fire rated as they are internal to fire area F1100. The containment serves as one wall of the room. Access and egress from the room is provided through a ~~nonrated shield~~ 3-hour rated fire door to the division 1 corridor (Rm 210).

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

Safety-Related	Provides Core Cooling
Yes, D1	Yes, D1

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

(4) Qualifications of Fire Barriers—The wall common with the CUW filter demineralizer area (Rm 347), the wall common with Emergency Electrical Room A (Rm 310), the wall common with the RIP Panel (Rm 315), the wall common with the Elevator (Rm 192) and stair-well (Rm 292) serve as fire barriers between adjacent fire areas and are of 3 h fire-resistive concrete construction. The remainder of the walls, the ceiling and the floor are concrete and are not rated as they are internal to fire area F1100. The containment serves as a portion of one wall of the corridor. Access to the corridor is provided from stair and elevator No.1, corridor C (Rm 335) and corridor D (Rm 344) via 3 h fire-resistive doors. The corridor provides direct access to the suppression pool personnel entry room (Rm 312)~~,~~ through a non-rated door and to Pipe Space A (Rm 313) and RPV instrument rack room (I) (Rm 314) through ~~nonrated~~ 3-hour rated fire doors.

(5) Combustibles Present:

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

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull station at Col. 5.5-B.2 and 6.2-C.8.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel	Col. 5.5-B.2 & 6.2-C.8/Manual
ABC hand extinguishers	Col. 5.5-B.2 & 6.2-C.8/Manual

(8) Fire Protection Design Criteria Employed:

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

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

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

(d) Provision of raised supports for equipment

(12) Fire Containment or Inhibiting Methods Employed:

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

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

(13) Remarks—None.

9A.4.1.3.6 Pipe Space A (Rm No. 313)

(1) Fire Area—F1100

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

Safety-Related	Provides Core Cooling
Yes, D1,D2	Yes, D1, D2

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

(4) Qualification of Fire Barriers—The walls and the floor are concrete and are not rated as they are internal to fire area F1100. The ceiling is common to fire area F4101 and is of 3 h fire-resistive concrete construction. The containment serves as one wall of the room. Access to the room is provided from Corridor A (Rm 311) via a ~~non-rated~~ 3-hour rated fire door. The room provides access to the metal grating pipe space area, and the Rm 318 at elevation 8500 mm via the stairs.

(5) Combustibles Present:

Fire Loading	Total Heat of Combustion (MJ)
None	727 MJ/m^2 NCLL (727 MJ/m^2 maximum average) applies.

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull station at Col. 5.5-B.2 and 6.2-C.8.

(13) Remarks—MO valve E51-F039 of the RCIC, and solenoid valves T31-720A,B of the Atmospheric Control System are all mounted in this room. Section 9A.5, Special Cases provides justification for locating equipment from multiple safety divisions in this room.

9A.4.1.3.7 Instrument Rack (I) (Rm No. 314)

(1) Fire Area—F1100

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

Safety-Related	Provides Core Cooling
Yes, D1	Yes, D1

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

(4) Qualifications of Fire Barriers—The wall common with the RPV instrument rack (III) room (Rm 332) serves as a fire barrier between fire areas F1100 and F1300 and is of 3 h fire-resistive concrete construction. The remainder of the walls, the ceiling and the floor are concrete and are not rated as they are internal to fire area F1100. The containment serves as one wall of the room. Access to the room is provided from corridor A (Rm 311) through a ~~nonrated~~ 3-hour rated fire door.

(5) Combustibles Present:

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

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull station at Col. 6.2-C.8 and 5.5-B.2.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel	Col. 6.2-C.8 & 5.5-B.2/Manual
ABC hand extinguishers	Col. 6.2-C.8 & 5.5-B.2/Manual

(8) Fire Protection Design Criteria Employed:

fire barrier. The remainder of the walls and the ceiling are concrete and are not rated as they are internal to fire area F4101. Access to the corridor is provided from the controlled entry room, the stairs and the elevator, and corridor C (Rm 430). The door to corridor C is ~~a 3 h fire-resistive door~~either a 5-psid door or two 3-hour rated fire doors. The corridor provides direct access to the electrical and instrumentation penetration room (Rm 411) through ~~a non-fire-rated door~~ either a 5-psid or two 3-hour rated fire doors and to ECCS valve room A through a 3 h fire rated door.

(5) Combustibles Present:

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

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull stations at 5.4-B.1 and 5.9-F.2.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel	Col. 5.4-B.1& 5.9-F.2/Manual
ABC hand extinguishers	Col. 5.4-B.1& 5.9-F.2/Manual

(8) Fire Protection Design Criteria Employed:

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

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

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

(9) Consequences of Fire—The postulated fire assumes the loss of the function. The provisions for core cooling systems backup are defined in Subsection 9A.2.5. Alternate access is provided by South Controlled Access Entry (Rm No. 193) Access is provided to the corridor from either end.

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

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

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

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

(b) Provision of raised supports for the equipment

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

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

(12) Fire Containment or Inhibiting Methods Employed:

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

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

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

9A.4.1.4.3 E and I Penetration Room (Div 1)(Rm No. 411)

(1) Fire Area—F4101

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

Safety-Related	Provides Core Cooling
Yes, D1	Yes, D1

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

(4) Qualifications of Fire Barriers—The floor is a fire barrier and is of 3 h fire-resistive concrete construction. The walls common to the Steam Tunnel (Rm 440) and the ECCS Valve A Room (Rm 414) are fire barriers and are of 3 h fire-resistive concrete construction. The other walls and the ceiling are concrete but are not rated as they are internal to fire area F4101. The containment serves as one wall of the room. Access to the room is provided from Corridor A (Rm 410) through an entry vestibule with ~~a nonrated door~~either a 5 psid door or two 3-hour rated fire doors.

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

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

(12) Fire Containment or Inhibiting Methods Employed:

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

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

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

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

(1) Fire Area—F4301

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

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

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

(4) Qualifications of Fire Barriers—The walls common with the C diesel generator room (Rm 432), valve room (C) (Rm 431), corridor B (Rm 420), the Flammability Control System room (Rm 436) and the exterior wall serve as fire barriers and are of 3 h fire-resistive concrete construction. The floor is also a fire barrier to limit the size of the fire areas below and to protect the lower regions of the building, which contains the majority of the ESF equipment. The walls are concrete and are not rated as they are internal to fire area F4301. A section of the ceiling common to fire areas F4300, F1300 and F3300 above is of 3 h fire-resistive concrete construction. The remainder of the ceiling is not fire rated as it is internal to fire area F4310. Access to the corridor is provided from corridors A via either a 5 psid or two 3-hour rated fire doors and corridor B via a 3 h fire-resistive doors. The corridor provides direct access to the electrical and instrumentation penetration room (Rm 433) through a nonrated door and valve room (C) (Rm 431) and the Flammability Control System room (Rm 436) through 3 h fire-resistive doors. There is an open hatch to the floors above. A large steel non-fire-rated door provides access to the reactor building for moving in fuel and other large loads.

9A.4.1.5 Building—Reactor Bldg El 18100 mm

9A.4.1.5.1 Corridor A (Rm No. 510)

(1) Fire Area—F4101

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

Safety-Related	Provides Core Cooling
Yes, D1	Yes, D1

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

(4) Qualifications of Fire Barriers—The walls common with the steam tunnel (Rm 440), stairwell (Rm 195), elevator (Rm 192), D/G HVAC and fan A room (Rm 514), D/G A control panel room (Rm 516), division 1 electrical penetration room (518) and the clean area access room (517) serve as fire barriers between adjacent fire areas and are of 3 h fire-resistive concrete construction. The remainder of the walls and the floor are concrete and are not rated as they are internal to fire area F4101. The ceiling is fire resistant and part of the wall is formed by the containment. Also, part of the wall in common with the steam tunnel is a blow out panel for pressure relief in the event of pressurization of secondary containment. Access to the corridor is provided from the stair and elevator via 3 h fire-resistive doors. The corridor provides direct access to the steam tunnel entry room (Rm 512) via a vestibule and non fire rated door. A ~~three hour fire-resistive door~~ 5-psid door or two 3-hour rated fire doors provides entry to and egress from corridor C (Rm No 530). The room is divided into two compartments by a non rated wall and door at row A.5.

(5) Combustibles Present:

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

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull stations at 5.5-A.9

(7) Suppression Available:

Type	Location/Actuation

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

(13) Remarks—None

9A.4.1.5.5 Steam Tunnel Entry Room (Rm No. 512)

(1) Fire Area—F4101

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

Safety-Related	Provides Core Cooling
No	No

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

(4) Qualifications of Fire Barriers—The ceiling and wall common to steam tunnel room (Rm 440) are fire barriers and are of 3 h fire-resistive concrete construction. The floor and remaining walls are internal to fire area F4101 and are not fire rated. There is a hatch in the ceiling for removal of equipment. Access is from corridor A, through a vestibule and ~~non-fire rated door~~either a 5-psid door or two 3-hour rated fire doors. Access to the steam tunnel from this room is provided via a 3 h fire-rated door. The room is also the access passage to the Division 1 E and I Penetration Room (Rm 512) via a non rated door.

(5) Combustibles Present:

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

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull stations at 5.5-A.9.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel	Col. 5.5-A.9/Manual
ABC hand extinguishers	Col. 5.5-A.9/ Manual

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

Safety-Related	Provides Core Cooling
Yes, D3	No

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

(4) Qualifications of Fire Barriers—One wall of the room is formed by the containment. The walls common to room 531 are internal to fire area F4301 and therefore are not fire rated. The remaining walls between corridor C and rooms 532 (division 3 electrical penetration room), 517 (access area A/C), 533 (D/G C fan room) and 536 (D/G C control panel room) serve as fire barriers and are of 3 h fire-resistive concrete construction. A section of the floor and ceiling are common to fire areas F1300 below and F3300 above and are of 3 h fire-resistive concrete construction. The remainder of the floor and ceiling are concrete and not rated because they are internal to fire area F4301. Access to corridor C is provided from corridor A via ~~a 3 h fire-resistive door~~either a 5-psid door or two 3-hour rated fire doors. Room 530 also contains a large equipment hatch open to the floor above and below.

(5) Combustibles Present:

Fire Loading	Total Heat of Combustion (MJ)
Lubricating Oil and Fuel Oil	Could be variable due to possible lubricant, and fuel oil leaks in transient. Deluge sprinkler system provided.

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull station at Col. 5.5-A.9 and 5.9-F.2.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel	Col. 5.5-A.9 & 5.9-F.2/Manual
ABC hand extinguishers	Col. 5.5-A.9 & 5.9-F.2/ Manual

(8) Fire Protection Design Criteria Employed:

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

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

(12) Fire Containment or Inhibiting Methods Employed:

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

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

(13) Remarks—None.

9A.4.1.5.37 Upper Drywell (Rm No.591)

(1) Fire Area—F4901

(2) Equipment: See Table 9A.6-2 for this elevation. Devices within the upper drywell are also listed at floor elevation 12300 mm.

Note: Section 9A.4.1.4.1 applies for the remainder of the information for the upper drywell. See that section for additional information.

9A.4.1.6 Building—Reactor Bldg El 23500 mm and 27200 mm

9A.4.1.6.1 Cross Corridor A (Rm No. 614)

(1) Fire Area—F4100

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

Safety-Related	Provides Core Cooling
Yes, D1	Yes, D1

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

(4) Qualifications of Fire Barriers—The exterior wall, inside wall, ceiling and floor of this corridor are of 3 h fire-resistive construction. This corridor extends across the reactor building. ~~At the south end of the corridor, a 3 h fire-resistive door opens to the electrical equipment room (Rm 640).~~ There are two 3-hour rated fire doors along this corridor. The first door is located between column lines R3/R4 and the second door is located between column lines R4/R5 on Figure 1.2-9. At the other end of the corridor, a nonrated door opens into D/G (A) exhaust fan area (Rm 613).

9A.4.1.6.43 Electrical Equipment Room (Rm No. 640)

(1) Fire Area—F6200

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

Safety-Related	Provides Core Cooling
No	No

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

(4) Qualifications of Fire Barriers—All walls and the floor are of 3 h fire-resistive concrete construction. A section of the ceiling is common to the FMCRD room (Rm 681) above and is of 3 h fire-resistive concrete construction. The remainder of the ceiling is internal to fire area F6200 and is not fire rated. Access is provided from room~~s~~ 625 through one 3-hour rated fire door and room 614 through ~~3 h fire-resistive doors~~two 3-hour rated fire doors.

(5) Combustibles Present:

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

(6) Detection Provided—Class A supervised POC in the room and manual alarm pull stations at 1.0-B.2 and 1.4-D.7.

(7) Suppression Available:

Type	Location/Actuation
Standpipe and hose reel	Col. 1.0-B.2 & 1.4-D.7/Manual
ABC hand extinguishers	Col. 1.0-B.2 & 1.4-D.7/Manual

(8) Fire Protection Design Criteria Employed:

(a) The function is located in a room separate from the rooms which contain safety-related equipment.

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

(a) Location of the manual suppression system at the perimeter of the area

(b) Provision of raised supports for the equipment

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

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

(12) Fire Containment or Inhibiting Methods Employed:

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

(13) Remarks—The area contains electrical cables in conduit. Cable insulation in conduit is discussed in Subsection 9A.3.4.

The control of the permanent and transitory combustible loads introduced through normal and maintenance operations is the responsibility of the applicant.

9A.4.1.7.2 RIP (A) Supply Fan and RCW (C) Surge Tank (Rm No. 715)

(1) Fire Area—F3300

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

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

(3) Radioactive Material Present—None.

(4) Qualifications of Fire Barriers—The walls common with the operating floor (Rm 716), the RCW A surge tank room (Rm 710), the D/G C exhaust fan room (Rm 730), the stairwell and elevator (Rms 316 and 317 respectively), and the ceiling are of 3 h fire-resistive concrete construction. The exterior wall is constructed of concrete but has ventilation openings to the outside and therefore is not fire rated. Sections of the floor common to fire areas F4100 and F4300 below (Rms 653 and 673 respectively) are also of 3 h fire-resistive concrete construction. The remainder of the floor is internal to fire area F3300 and is not fire rated.

Access to room 715 is provided by the stairwell and elevator through 3 h fire-resistive doors. Room 715 provides access to rooms~~ 710 and~~ 730 via a 3 h fire-resistive door~~s~~. Room 715 provides access to room 710 via two 5-psid doors.

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

Safety-Related	**Provides Core Cooling**
Yes, D1	Yes, D1

(3) Radioactive Material Present—None.

(4) Qualifications of Fire Barriers—Both internal walls, one exterior wall, the floor and the ceiling are 3 h fire-resistive concrete construction. The remaining exterior wall has an opening for the normal HVAC input to the reactor secondary containment and therefore is not fire rated. Access to room 710 is provided from the RIP A supply fan and RCW C surge tank room (Rm 715) via two 5-psid doors. Access to the other side of the reactor building is provided by an interconnecting corridor from this room. A 3 h rated fire door is located in the corridor.

(5) Combustibles Present—No significant quantities of exposed combustibles. 727 MJ/m^2 NCLL (727 MJ/m^2 maximum average) applies.

(6) Detection Provided—Class A supervised POC detection system and alarm pull stations room at 6.6-C.0 and 6.3-E.9.

(7) Suppression Available:

Type	**Location/Actuation**
Standpipe and hose reel	Col. 6.6-C.0/Manual
ABC hand extinguishers	Col. 6.3-E.9 /Manual

(8) Fire Protection Design Criteria Employed:

(a) The function is located in a fire area which is separate from the fire areas containing equipment which provides alternate means of performing the safety or shutdown function.

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

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

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

9.5.13.21 Quality Assurance Requirements for CTG

Quality assurance standards and practices shall be developed to assure continued operational reliability of the CTG as an AAC power source for SBO events, in accordance with Regulatory Guide 1.155 and 10CFR50.63.

9.5.13.22 Power Supply for Alternate Feedwater Injection (AFI) Equipment

The COL applicant will identify the power supply for the equipment used to support alternate feedwater injection. The power supply for the pump and motor-operated valves will be a non-safety- related power supply and independent of the emergency power supplies. The power supply will be physically separated from the emergency power supplies such that a simultaneous loss due to beyond design basis events is unlikely.

9.5.13.23 Test and Surveillance Intervals for AFI Equipment

The COL applicant will develop test and surveillance intervals for the equipment required for alternate feedwater injection.

9.5.14 Alternate Feedwater Injection System

9.5.14.1 System Description

An alternate feedwater injection (AFI) system, capable of injecting into the Reactor Pressure Vessel (RPV) at operating pressure ($\geq$ 800 g.p.m. at a pressure approximately at the lift setpoint of the first group of safety/relief valves) and located outside of the Reactor Building (R/B) is available. The system is capable of providing sufficient core cooling in the unlikely event that all normal and emergency core cooling systems are unavailable. It is comparable to the High Pressure Core Flooder (HPCF) system capacity and discharge pressure (at rated pressure). The AFI Pump House which contains this system is located such that a simultaneous loss of the non-seismic AFI Pump House and the Reactor Building is unlikely. The height and location of the AFI Pump House precludes a direct line of sight of the AFI Pump House from a Reactor Building supplied by the AFI system. A schematic of the AFI system is shown in Figure 9.5-6.

The system takes suction from an existing water source which is located near the AFI Pump House. There is a minimum of 300,000 gallons of useable water at the AFI Pump suction line while the AFI is in standby. The AFI system discharges through three normally closed motor-operated valves (MOV). The system discharge piping is routed underground or is otherwise protected from physical impact. The injection is provided through the non-safety-related portion of the CUW tie-in lines to the feedwater system. The tie-in is in the R/B portion of the Steam Tunnel. A single AFI system may be used with the injection configured to support more than one unit of a multiple unit site. The system and power supplies are non-safety grade.

The power supply for the pump and motor-operated valves is a non-safety-related power supply and independent of the emergency power supplies. The power supply is physically separated from the emergency power supplies such that a simultaneous loss due to beyond design basis events is unlikely. The specific power supply will be defined by the COL applicant.

The system can be operated from the AFI Pump House. This will ensure that the injection can be initiated within 30 minutes after the loss of normal makeup systems to provide sufficient core cooling. In addition, the operator is provided with the capability to control flow from the AFI Pump House by throttling a motor- operated valve located in the Pump House.

9.5.14.2 Safety Evaluation

This system does not degrade safety for normal operation and provides enhanced safety during and after beyond design basis events. The ability to maintain core cooling is improved by the addition of this separate and diverse means of providing cooling water to the core when all normal and emergency cooling systems are unavailable. The piping and components that interface with the CUW system are the same quality as that system up to and including the second check valve.

9.5.14.3 Testing and Inspection Requirements

Preoperational testing requirements for the AFI system are prepared as described in Subsection 14.2.12.

The COL applicant will develop test and surveillance intervals for the equipment in the AFI Pump House.

9.5.14.4 Instrumentation Requirements

The following indications are provided in the AFI Pump House:

- RPV water level
- RPV pressure
- Wetwell WR pressure
- Suppression pool water level

In addition, the following AFI-related instrumentation is provided in the AFI Pump House:

- AFI pump flow and discharge pressure
- Dedicated water storage tanks water level indication

The instrument lines, instrument rack and cables in the Reactor Building are protected from fire and shock. Fire protection of instrument lines is achieved by protecting the instrument penetration room. Fire resistant cabling with at least 3 hour fire rating is used. The instrument lines to be used for monitoring the alternate feedwater injection are branched from the existing line and are connected to new level and pressure transmitters. The additional transmitters for RPV water level, RPV pressure and wetwell WR pressure as well as the suppression pool water level transmitter are installed in a room which is protected from fire effects. The room protection is achieved by additional fire doors or modification of fire doors to water-tight doors. In the unlikely event of an instrument line break, the break flow is limited by the small size of the instrument line orifice and is accounted for in the specified capacity for the AFI pump.

9.5.15 ~~9.5.14~~ Reference

9.5-1 Stello, Victor, Jr., "Design Requirements Related To The Evolutionary Advanced Light Water Reactors (ALWRS)", Policy Issue, SECY-89-013, The Commissioners, United States Nuclear Regulatory Commission, January 19, 1989.

9.5-2 Cote, Arthur E., "NFPA Fire Protection Handbook", National Fire Protection Association, Sixteenth Edition.

9.5-3 "Design of Smoke Control Systems for Buildings", American Society of Heating, Refrigerating, and Air Conditioning Engineers, Inc., September 1983.

9.5-4 "Recommended Practice for Smoke Control Systems", NFPA 92A, National Fire Protection Association, 1988.

9.5-5 Life Safety Code, NFPA 101, National Fire Protection Association.

9.5-6 "Reliability of Emergency Diesel Generators at U.S. Nuclear Power Plants", Electric Power Research Institute, NSAC-108, September 1986.

9.5-7 Loss of All Alternating Current Power, 10CFR50.63.

9.5-8 Regulatory Guide 1.155—Station Blackout.

9.5-9 "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors", NUMARC-87-00.

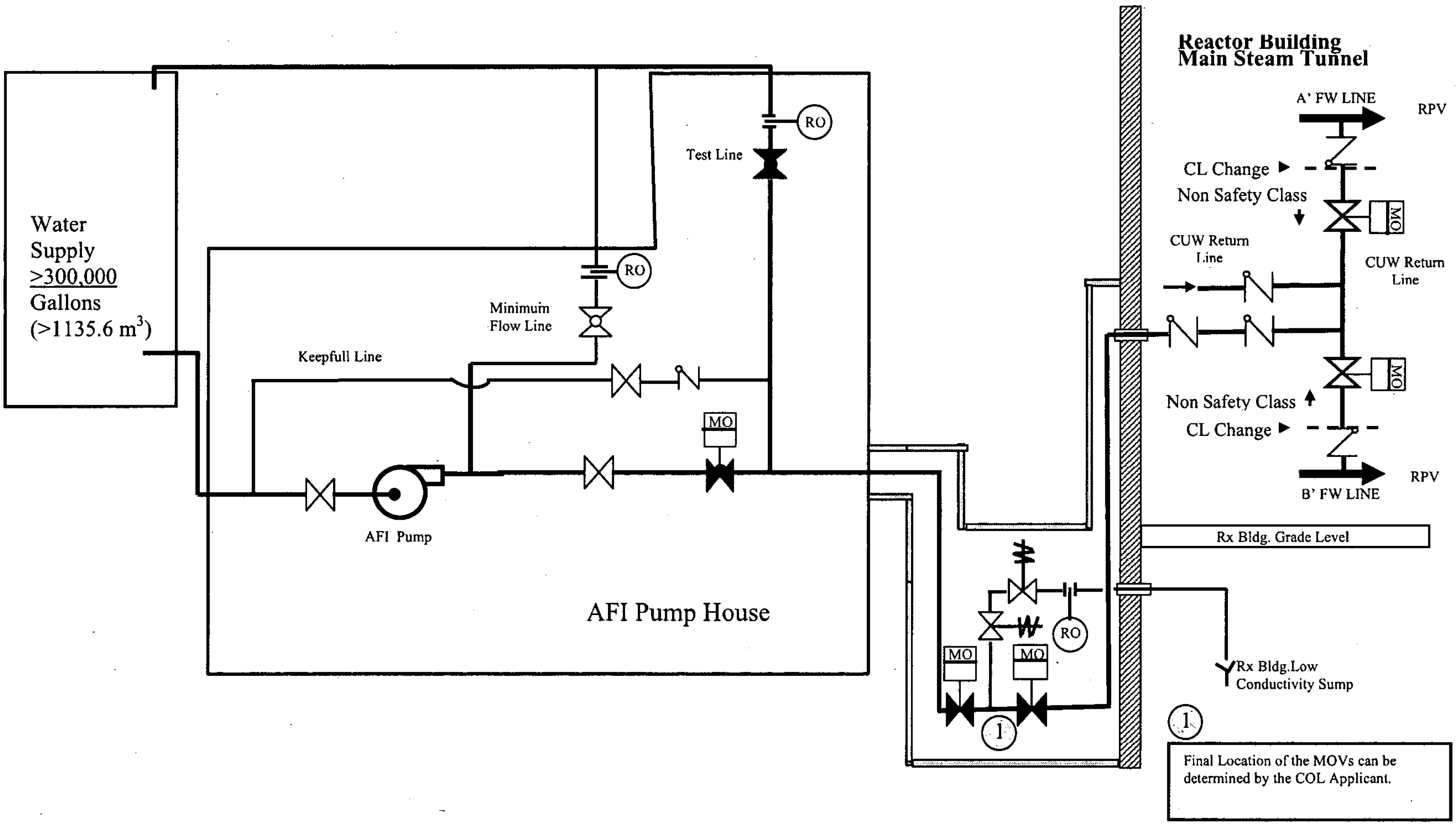

Figure 9.5-6 Alternate Feedwater Injection System Schematic

— Nitrogen gas bottles are used as nitrogen source.

(iv) HPIN is operated with hook-up of a nitrogen gas bottle from the remotely located AFI Pump House.

(f) Proper operation of interlocks and equipment protective devices including operation of all components subject to interlocking, interlocking set value and operating logic.

(g) Proper operation of permissive, prohibit, and bypass functions.

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

(i) Acceptable vibration levels and system piping movements during both transient and steady-state operation.

(j) Ability of the nitrogen gas to meet end use cleanliness requirements with respect to oil, water, and particulate matter content.

(k) Proper operation of the HPIN system during a loss of nitrogen gas testing. This test is done by shutting off the nitrogen gas supply system in a manner that will simulate a sudden nitrogen gas supply pipe break and a gradual loss of pressure (plugging or freezing) as required by Regulatory Guide 1.68.3.

14.2.12.1.29 Reactor Building Cooling Water System Preoperational Test

(1) Purpose

To verify the ability of the Reactor Building Cooling Water (RCW) System, including its ability to supply design quantities of cooling water, to essential and nonessential loads, as appropriate, during normal, abnormal, and accident conditions.

(2) Prerequisites

The construction tests have been successfully completed, and the SCG has reviewed the test procedure and approved the initiation of testing. Primary and backup power, reactor Service Water, Instrument Air, MUWP System, and other required supporting systems shall be available, as needed, for the specified testing configurations. The cooled components shall be operational and operating to the extent practicable during heat exchanger performance evaluation.

(3) General Test Methods and Acceptance Criteria

Performance shall be observed and recorded during a series of individual component and integrated system tests. These tests shall demonstrate that the RCW System and its auxiliary equipment operate properly as specified in Subsections 9.2.11 and the required interfacing systems shall be available, as needed, to support the specified testing.

(3) General Test Methods and Acceptance Criteria

Performance shall be observed and recorded during a series of component and system testing. This test shall demonstrate that the UHS operates properly as specified in Subsection 9.2.5 and applicable UHS design specifications through the following testing:

(a) Proper operation of instrumentation and various components alarms used to monitor system operation and status, including indications for UHS water level, temperature and blowdown volumes, etc., as specified in Subsection 9.2.5.9.

(b) Proper operating conditions and performance capability of the UHS spray networks during all anticipated modes of the RSW System operations as specified in Subsection 9.2.5.4.1.

(c) Proper operating conditions and performance capability of the UHS in cold weather mode of operation through the bypass line as specified in Subsection 9.2.5.4.2.

(d) Proper operation of the makeup water valve to maintain proper water level in the UHS spray pond through makeup line and maintain water quality in conjunction with the blowdown operation as specified in Subsection 9.2.5.3.4.

(e) Proper operation of blowdown from the UHS spray pond to remove excess water and maintain water quality control through the blowdown line as specified in Subsection 9.2.5.3.4.

14.2.12.1.78 Alternate Feedwater Injection System Preoperational Test

(1) Purpose

To verify the operation of the Alternate Feedwater Injection (AFI) System, including related auxiliary equipment, pumps, valves, instrumentation and controls, is as specified.

(2) Prerequisites

The construction tests have been successfully completed, and the SCG has reviewed the test procedure and approved the initiation of testing. A water source shall be available as the AFI pump suction source and the reactor vessel and feedwater lines A and B shall be sufficiently intact to receive AFI injection flow. The appropriate electrical power sources shall be available as needed, to support the specified testing and the appropriate system configurations.

(3) General Test Methods and Acceptance Criteria

Performance shall be observed and recorded during a series of individual component and integrated system tests. This test shall demonstrate that the AFI System operates properly as specified by Subsection 9.5.14 and the applicable AFI System design specification through the following testing:

(a) Correct implementation and operation of the AFI System controls and instrumentation. This test shall check the system behavior against the functional, performance and interface requirements as specified by the appropriate design documents.

(b) Verification of various component alarms for proper alarm actuation by practically operating the detector of the alarm generating source or using the simulated signal and alarm reset.

(c) Proper operation of all motor-operated valves including opening and closing with the operating switch, valve status indication and travel timing, if applicable.

(d) Proper operation of AFI pump and motor during continuous run tests.

(e) Acceptable pump NPSH under the most limiting design flow conditions.

(f) Verification that the AFI System can be operated normally at each mode and satisfy the NPSH requirement by combining all components, piping and instruments constituting this system through the following testing:

(i) Minimum flow operational test—operate the AFI pump manually using flow path from water source to water source through the minimum flow line until the temperature of the pump and motor bearing is stabilized.

(ii) Rated flow operational test—operate the AFI System at rated flow using the test line to the water source. This test shall be performed continuously from the pump motor start sequence and the minimum flow operating condition.

(iii) Reactor injection test to FW Line A—operate the AFI System to FW Line A at near rated pressure using the injection line to confirm that the pump flow operation can be verified. For this test, the motor-operated valve to FW Line A will be open and the motor-operated valve to FW Line B will be closed.

(iv) Reactor injection test to FW Line B —operate the AFI System to FW Line B at near rated pressure using the injection line to confirm that the pump flow operation can be verified. For this test, the motor-operated valve to FW Line B will be open and the motor-operated valve to FW Line A will be closed.

(g) Proper AFI pump motor start sequence and actuation of protective devices.

(h) Proper operation of interlocks including operation of all components subject to interlocking.

(i) Proper operation of permissive, prohibit, and bypass functions.

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

(k) Acceptable pump/motor vibration levels and system piping movements during both transient and steady-state operation. This test can be performed in conjunction with expansion, vibration and dynamic effects preoperational test (Subsection 14.2.12.1.51).

(l) Proper operation of the pump discharge line keep-fill system and its ability to prevent damaging water hammer during system transients.

14.2.12.2 General Discussion of Startup Tests

Those tests proposed and expected to compromise the startup test phase are discussed in this subsection. For each test a general description is provided for test purpose, test prerequisites, test description and test acceptance criteria, where applicable.

Since additions, deletions, and changes to these discussions are expected to occur as the test program is developed and implemented, the descriptions remain general in scope. In describing a test, however, an attempt is made to identify those operating and safety-oriented characteristics of the plant which are being explored and evaluated.

Where applicable, a definition of the relevant acceptance criteria for the test is given and is designated either Level 1 or Level 2. A Level 1 criterion normally relates to the value of process variables assigned in the design or analysis of the plant, component systems, or associated equipment. If a Level 1 criterion is not satisfied, the plant will be placed in a suitable hold condition until resolution is obtained. Tests compatible with this hold condition may be continued. Following resolution, applicable tests may be repeated to verify that the requirements of the Level 1 criterion are ultimately satisfied. A Level 2 criterion is associated with expectations relating to the performance of sytems. If a Level 2 criterion is not satisfied, operating and testing plans would not necessarily be altered. However, an engineering evaluation, such as an investigation of the measurements and of the analytical techniques used for the predictions, would be started. If a certain Level 2 criterion is not satisfied after a reasonable effort, then the cognizant engineering organization may choose to document the results with a full explanation of their recommendations. Thus, all Level 2 requirements may not be satisfied provided that the overall system performance is evaluated to be acceptable based on engineering's recommendations. The specific actions required for dealing with

19.0 Response to Severe Accident Policy Statement

19.1 Purpose and Summary

19.1.1 Purpose

This chapter documents the Advanced Boiling Water Reactor (ABWR) capability in response to the NRC Policy Statement on Severe Accidents (Reference 19.1-1) and in response to the ABWR Licensing Review Bases (Reference 19.1-2) which would be used for NRC review of the ABWR Standard Plant design. Response to the CP/ML (Construction Permit/Manufacturing License) Rule (Reference 19.1-3) is provided in Appendix 19A. Resolution of applicable unresolved safety issues and generic safety issues is contained in Appendix 19B. For the most part, the ABWR capability is documented by probabilistic risk assessment techniques in Appendix 19D as outlined by Reference 19.1-2. Appendices 19E and 19F support the probabilistic risk assessment and provide the deterministic assessment of the ABWR capability to withstand a severe accident.

Appendices 19H and 19I consider the ABWR response to very large seismic events. Appendix 19K identifies appropriate additional reliability and maintenance actions that are required throughout the life of the plant so that the PRA remains an adequate basis for quantifying plant safety. Shutdown risk is addressed in Appendix 19L and 19Q. A fire protection probabilistic risk assessment is given in Appendix 19M. Detailed information about common-cause failure of multiplex equipment is provided in Appendix 19N. Appendix 19P provides information about the consideration of additional design modifications to reduce the residual risk of severe accidents. ~~Finally,~~ Appendix 19R contains a screening analysis for the potential for flooding to lead to core damage. Finally, Appendix 19S provides the response to the aircraft impact assessment rule (Reference 19.1-4).

19.1.2 Summary

This analysis indicates that ABWR satisfies the severe accident related goals identified in Reference 19.1-2. The individual goals are listed in Section 19.6 where the specific manner in which the goals are satisfied is described. For the purposes of this subsection, this information is further summarized and is organized into three major areas: prevention of core damage, maintenance of containment integrity and minimizing off-site consequences.

Core damage is prevented by three divisions of the Emergency Core Cooling System (ECCS) including the Reactor Core Isolation Cooling System which can function for several hours without AC power. It also includes a reliable and proven reactor depressurization system. Feedwater and condensate pumps also provide protection against core damage. A gas turbine is also available as an alternate supply to key electrical loads. Although an AC-independent Firewater Addition System is incorporated in the design, no credit is taken for it in the calculation of core damage frequency. The calculated core damage frequency is extremely low.

Containment integrity is protected by inerting the containment volume with nitrogen and by providing a three-division heat removal system, many components of which are operated routinely and thus have very high reliability. In addition, the containment design incorporates a containment overpressure protection system. The probability of containment failure resulting from loss of heat removal is extremely small.

In response to the aircraft impact rulemaking in Reference 19.1-4, an analysis of the ABWR plant design was performed. This analysis addressed the ability of the ABWR to either cool the core or maintain primary containment intact and to either cool the spent fuel pool or provide spent fuel pool integrity. The analysis results, provided in Appendix 19S, demonstrate that the core cooling is maintained and spent fuel pool integrity is maintained.

19.1.3 References

19.1-1 50FR32138, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants", August 8, 1985.

19.1-2 Thomas E. Murley (NRC) letter to Ricardo Artigas (GE), August 7, 1987, "Advanced Boiling Water Reactor Licensing Review Bases."

19.1-3 Title 10, Code of Federal Regulations, Part 50, Section 50.34(f).

19.1-4 Title 10, Code of Federal Regulations, Part 50, Section 50.150.

configurations and performance capabilities against those assumed and modeled in Subsection 19D.6.4.2 and assess the impact of any differences on the ABWR PRA results.

19.9.27 Capability of Vacuum Breakers

The vacuum breaker seating material will be demonstrated to withstand the temperature profiles associated with the equipment survivability requirements specified in Subsection 19E.2.1.2.3.

19.9.28 Capability of the Containment Atmospheric Monitoring System

The COL applicant will demonstrate that the portion of the CAMS System which can be exposed to containment pressure can withstand the loading associated with the equipment survivability requirements specified in Subsection 19E.2.1.2.3.

19.9.29 Plant Specific Safety-Related Issues and Vendors Operating Guidance

The COL applicant shall address and incorporate plant-specific safety-related issues and the vendor's operating guidance on safe operations during shutdown (See Subsection 19Q.10 under "Shutdown Safety Issues").

19.9.30 PRA Update

A COL applicant referencing the ABWR certified design will review and, if necessary, update the design PRA to ensure that it bounds the site specific design (e.g. the ultimate heat sink) and that interface requirements of the standard design are satisfied. In addition, site characteristics such as river flooding, wind loadings, etc., will be compared to those assumed in the design PRA to ensure it is bounding. If the existing PRA is not bounding for site characteristics, then a risk based evaluation should be performed.

19.9.31 <u>Procedures for Use of Alternate Feedwater Injection (AFI)</u>

<u>Specific, detailed procedures will be developed by the COL applicant for the use of the AFI System as a source of core cooling in the event that all normal and emergency cooling is unavailable. Training will be included in the COL applicant's crew training program. Procedures and training will cover such items as identification of conditions requiring system operation, pump start and flow monitoring, correct injection valve alignment and monitoring of critical parameters such as RPV water level and pressure from the AFI Pump House.</u>

19.9.32 <u>Procedures to Depressurize the RPV from the AFI Pump House</u>

<u>Specific, detailed procedures will be developed by the COL applicant for assembling and operating the additional nitrogen supply system from the AFI Pump House to allow depressurization of the RPV through one of the safety/relief valves.</u>

19.9.33 Housing of Equipment in the AFI Pump House

The equipment for the AFI system as well as the spare nitrogen supply for operation of one safety/relief valve are housed in a separate Pump House located remotely from the Reactor Building and Turbine Building. Although the Pump House is not required to be seismic Category I and the equipment contained therein is non-safety-related, the COL applicant shall ensure that the environmental conditions within the building are within the manufacturer's recommended conditions, particularly for electrical equipment supporting those systems.

19.9.34 Test and Surveillance Intervals for Equipment in AFI Pump House

The COL applicant will develop test and surveillance intervals for the equipment in the AFI Pump House as described in Section 9.5.14.3.

19.9.35 Electrical Power Supply Description for AFI Pump House Equipment

The COL applicant will provide drawings which describe the electrical power supply to the equipment in the AFI Pump House. Provisions shall be made for unavailability of off-site power and unprotected on-site emergency power in the Reactor Building and Turbine Building for beyond design basis events.

19S Aircraft Impact Assessment

19S.1 Introduction and Background

A design-specific assessment of the effects on the ABWR of the beyond design basis impact of a large, commercial aircraft has been performed in accordance with 10 CFR 50.150(a) to identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions: (i) The reactor core remains cooled, or the containment remains intact; and (ii) spent fuel cooling or spent fuel pool integrity is maintained. The specific assumptions regarding the aircraft impact were based on guidance provided by the NRC and the Nuclear Energy Institute (NEI 07-13 Rev. 7), including the loading function derived from the aircraft impact characteristics for use in assessments of aircraft impact effects.

This appendix describes those design features and functional capabilities identified in the assessment, and discusses how the identified design features and functional capabilities show that, with reduced use of operator actions, the reactor core remains cooled or the containment remains intact, and spent fuel cooling or spent fuel pool integrity is maintained. In the following discussion the identified design features are designated as "key design features."

19S.2 Scope of the Assessment

The evaluation of plant damage caused by the impact of a large, commercial aircraft is a complex analysis problem involving phenomena associated with structural impact, shock-induced vibration, and fire effects. The analysis of the aircraft impact considers structural damage, taking into account:

- An assessment of the effects of aircraft fuselage and wing structure;
- An assessment of the effects of shock-induced vibration on systems, structures, and components;
- An assessment of the penetration of hardened aircraft components, such as engine rotors and landing gear.

The results of the assessment predict that the spent fuel pool and primary containment vessel is not perforated; therefore, further assessment of the damage to the corresponding internal systems, structures, and components caused by 1) burning aviation fuel and 2) secondary impacts is not required.

The results of the assessment predict that the Reactor Building (R/B) and Control Building (C/B) is perforated; therefore, realistic assessments of the damage to the corresponding internal systems, structures and components caused by 1) burning aviation fuel and 2) secondary impacts are performed.

19S.3 Assessment Methodology

Methods described in NEI 07-13 were followed to assess the effects on the structural integrity of the primary containment and spent fuel pool, and to assess the physical, fire and vibration effects of the aircraft impact on the core cooling capability of the existing and enhanced design.

19S.4 Results of Assessment

The following key design features and functional capabilities ensure that the ABWR design can maintain core cooling and spent fuel pool integrity following the impact of a large, commercial aircraft.

19S4.1 Primary Containment

The primary containment, as described in Tier 2 Sections 3.8 and 3H.1, is a key design feature that would protect the safety systems located inside primary containment from the impact of a large, commercial aircraft. The assessment concludes that a strike upon the primary containment would not result in the perforation of the primary containment, and would not cause direct damage to the systems within the primary containment or expose them to jet fuel.

The assessment also finds that safety-related components inside primary containment, including the reactor pressure vessel and associated ECCS piping are unaffected by shock-induced vibrations resulting from the impact of a large commercial aircraft.

19S4.2 Site Arrangement and Plant Structural Design

The design and arrangement of major structures associated with the ABWR as described in Tier 2 Section 1.2 and Figure 1.2-1 are key design features. Specifically, the assessment credited the arrangement and design of the following building features to limit the location and effects of potential aircraft impacts on the R/B, primary containment and C/B in the following locations:

(1) The location and design of the C/B structure as described in Tier 2 Sections 3.8.4 and 3H.2 are key design features that protect portions of the north wall of the R/B from the impact of a large, commercial aircraft.

(2) The location and design of the Turbine Building structure as described in Tier 1 Section 2.15.11 and Tier 2 Figures 1.2-24 through 1.2-31 are key design features that protect portions of the north wall of the C/B and R/B from the impact of a large, commercial aircraft.

(3) The location and design of the R/B structure as described in Tier 2 Sections 3.8.4 and 3H.1 are key design features that protect portions of the primary containment and the south wall of the C/B from the impact of a large, commercial aircraft. This includes the protection provided by exterior walls, interior walls, intervening structures and barriers on the large openings in the R/B exterior walls.

(4) The location and design of the spent fuel pool and its supporting structure as described in Tier 2 Section 9.1 and Figure 1.2-12 are key design features that protect the spent fuel pool from the impact of a large, commercial aircraft.

(5) The physical separation of the Class 1E emergency diesel generators and an independent power supply as described in Tier 2 Section 9.5.14 is a key design feature that prevents the loss of all electrical power to core cooling systems.

19S4.3 Fire Barriers and Fire Protection Features

The design and location of 3-hour fire barriers, including fire doors and watertight fire doors that separate the safety divisions within the R/B and C/B are key design features for the protection of core cooling equipment within these buildings from the impact of a large, commercial aircraft. The assessment credited the design and location of fire barriers (including doors) as described in Tier 2 Sections 9.5.1 and 9A.4 for the R/B and the C/B to limit the effects of internal fires created by the impact of a large, commercial aircraft.

19S4.4 Core Cooling Features

The design and physical separation of the emergency core cooling systems described in Tier 2 Section 6.3, the alternate feedwater injection system described in Tier 2 Section 9.5.14, and the containment overpressure protection system described in Tier 2 Section 6.2.5 are key design features for assuring core cooling.

19S.5 Conclusions of Assessment

This assessment based upon NEI 07-13 concludes that the ABWR can continue to provide adequate protection of the public health and safety in the event of an impact of a large, commercial aircraft, as defined by the NRC. The aircraft impact would not inhibit the ABWR's core cooling capability and spent fuel pool integrity based on best estimate calculations. The assessment resulted in the identification of the key design features and functional capabilities described in Section 19.S.4, changes to which are required to be controlled in accordance with 10 CFR 50.150(c).

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Figure 1.2-8 REACTOR BUILDING, ARRANGEMENT PLAN AT ELEVATION 12300 mm

NOT FOR PUBLIC RELEASE

Figure 1.2-9 REACTOR BUILDING, ARRANGEMENT PLAN AT ELEVATION 18100mm

NOT FOR PUBLIC RELEASE

FIGURE 1.2-10 REACTOR BUILDING, ARRANGEMENT PLAN AT ELEVATION 23500 mm

NOT FOR PUBLIC RELEASE

Figure 1.2-12 REACTOR BUILDING, ARRANGEMENT PLAN AT ELEVATION 31700/38200 mm

NOTES

1. ENCLOSED EQUIPMENT AND COMPONENTS ARE TYPICAL FOR THE OTHER STEAMLINES AND HAVE THE SAME PART NUMBERS UNLESS OTHERWISE NOTED.
   EXAMPLE: XXXB IS ON LINE "B"
   EXAMPLE: XXXC IS ON LINE "C"

2. SEE REFERENCE DOCUMENT 40 FOR THE SIZE OF THE INLET FLANGE FOR THE SAFETY RELIEF VALVES (SRVS).

3. PIPE SIZES SHOWN ON THIS DRAWING ARE APPROXIMATE EXCEPT AT POINTS OF CONNECTION WITH THE SUPPLIED EQUIPMENT OR PIPING. THE PIPING DESIGNER SHALL CHECK AND ADJUST PIPING SIZE IN ACCORDANCE WITH HIS PIPING LAYOUT FOR CONFORMANCE WITH THE NUCLEAR BOILER SYSTEM DESIGN SPECIFICATION (B21-4010) AND PROCESS DIAGRAM (B21-1020).

4. THE GLOBE VALVE F707 MAY NOT BE PROVIDED IF A SHUTOFF VALVE IS SUPPLIED WITH THE LEVEL TRANSMITTER LT004. IF THE SECOND SHUT-OFF VALVE IS PART OF LT004, THE DESIGN PRESSURE FOR THIS INSTRUMENT LINE IS 8.62 MPa G ALL THE WAY TO LT004.

5. PX TO BE CONNECTED INTO THE STRAIGHT RUN OF PIPE DOWNSTREAM OF F009 WITH UPSTREAM AND DOWNSTREAM STRAIGHT LENGTH FROM THE TAP TO GIVE AS ACCURATE A PRESSURE MEASUREMENT AS FEASIBLE. TAPS TO MEET ASME PTC 6 1964 "STEAM TURBINE" PARAGRAPH 4.74.

6. AN EXPANSION LEG SHALL BE PROVIDED IN THE INSTRUMENT LINE BETWEEN THE CONDENSING POT 0011 AND THE WATER TIGHT PENETRATION IN THE REFUELING BELLOWS. THE EXPANSION LEG AND PIPING INSTALLATION SHALL BE DESIGNED TO ALLOW FOR THE MAXIMUM CHANGE OF VESSEL LENGTH WITH TEMPERATURE TO AVOID OVERSTRESSING THE PIPING OR THE SEAL OR DAMAGE TO THE INSULATION AROUND THE VESSEL. ELEVATION "A" SHALL BE AT OR ABOVE THE CENTERLINE OF THE RPV HEAD VENT LINE. THE INSTRUMENT LINE FROM ELEVATION "A" TO ELEVATION "B" SHALL BE SLOPED CONTINUOUSLY UPWARD ≥ 1/24 AND BE KEPT AS SHORT AS PRACTICAL. THE INSTRUMENT LINE SHALL BE KEPT INSULATED FROM THE CONNECTION WITH THE RPV HEAD VENT LINE TO THE CONDENSING POT. THE INSULATION SHALL HAVE A MAXIMUM CONDUCTANCE OF 4.103E-01 J/h.cm ².°C.

7. PROVISIONS FOR INSTRUMENT LINE ISOLATION SHALL BE IN ACCORDANCE WITH SUPPORTING DOCUMENTS 3 & 4. ONE ORIFICE SHALL BE INSTALLED IN EACH INSTRUMENT LINE CONNECTED TO THE REACTOR COOLANT PRESSURE BOUNDARY (RCPB). ORIFICE SIZE IS 6.4mm AND MAXIMUM NUMBER OF ORIFICES PER LINE IS ONE.

8. VALVE MOTOR OPERATORS AND PILOT SOLENOIDS ARE AC OPERATED UNLESS OTHERWISE SPECIFIED.

9. THE CONDENSING CHAMBER SHALL BE CLOSE COUPLED TO THE 50A RPV INSTRUMENT LINE NOZZLE BY A 50A PIPE. THE 50A PIPE FROM THE REACTOR VESSEL INSTRUMENT LINE NOZZLE SHALL BE LEVEL IN THE HORIZONTAL PLANE FOR ALL CONDITIONS WITHIN 3 mm.

   THE INSTRUMENT LINE CONNECTED TO THE BOTTOM OF THE CONDENSING CHAMBER SHALL HAVE A DOWNWARD SLOPE ≥ 1/25. INSIDE THE PRIMARY CONTAINMENT, THE TOTAL VERTICAL DROP FROM THE CONDENSING CHAMBER TO THE CONTAINMENT WALL SHALL NOT EXCEED 0.9 METERS. AT THE BOTTOM CONNECTION TO THE CONDENSING CHAMBER, THE INSTRUMENT LINE SHALL BE 25A PIPE, BUT PRIOR TO PENETRATING THE PRIMARY CONTAINMENT, SHALL BE REDUCED TO 20A PIPE.
   IN ADDITION, THE CONDENSING CHAMBER HAS A 25A DRAIN LINE WHICH DRAINS THE EXCESS CONDENSATE OR WATER TO THE VARIABLE LEG INSTRUMENT LINES FOR THE RPV WIDE RANGE WATER LEVEL INSTRUMENTATION. FLEXIBILITY SHALL BE PROVIDED IN THE INSTRUMENT LINE AND CONDENSING CHAMBER DRAIN LINE SUCH THAT THE CONDENSING CHAMBER IS FREE TO MOVE WITH THE REACTOR VESSEL AS IT THERMALLY EXPANDS AND CONTRACTS. THERMAL EXPANSION SHALL NOT CHANGE THE ELEVATION OF THE CONDENSING CHAMBER WITH RESPECT TO THE RPV INSTRUMENT LINE NOZZLE BY MORE THAN 3 mm.
   INSULATE THE 50A PIPE WHICH ATTACHES THE RPV INSTRUMENT LINE NOZZLE TO THE CONDENSING CHAMBER WITH INSULATION WHICH HAS A MAXIMUM CONDUCTANCE OF 4.103E-01 J/h.cm ².°C. THE INSULATION SHALL EXTEND FROM THE RPV INSTRUMENT LINE NOZZLE TO THE CONDENSING CHAMBER. THE CONDENSING CHAMBER AND THE INSTRUMENT LINE FROM THE CONDENSING CHAMBER TO THE CONTAINMENT PENETRATION SHALL NOT BE INSULATED. THE 25A DRAIN LINE SHALL BE INSULATED.

10. LOCATE THE TEE AS CLOSE AS POSSIBLE TO REACTOR VESSEL.

11. THE MAXIMUM OPERATING PRESSURE OF AT LEAST 2.82 MPa G. AND TEMPERATURE FOR THE PORTION OF THE MAIN STEAM LINE DRAIN LINE HEADER DOWNSTREAM OF THE MOTOR OPERATED VALVES (MOVS) F014 AND F016 AND THE RESTRICTING ORIFICE D005 AND D007 TO BE DETERMINED BY THE DESIGNERS OF THE MAIN CONDENSER SYSTEM.

12. LOCATE THE DRAIN LINE 20A-NB-542 AND ASSOCIATED EQUIPMENT FOR DETECTING THE LEAKAGE AS CLOSE AS PRACTICAL TO THE RPV.

13. FOR DETAILS. SEE B11-D021 & D025.

14. THERMAL SLEEVE IS SHOWN AS ONE POSSIBLE METHOD OF ACCOMODATING ΔT BETWEEN RCIC/RHR/CUW AND FEEDWATER STREAMS. OTHER METHODS WHICH MEET APPLICABLE CODE REQUIREMENTS MAY BE USED.

15. SPRING CLOSING CHECK VALVE. SPRING ACTUATOR HELD IN OPEN POSITION BY AIR PRESSURE DURING NORMAL OPERATION. IF NO OTHER CHECK VALVE BETWEEN THE REACTOR AND THE FEEDWATER PUMPS IS DESIGNED TO CLOSE PRIOR TO APPRECIABLE FLOW REVERSAL. F003 SHOULD BE INTERLOCKED TO DUMP AIR PRESSURE AUTOMATICALLY IN THE EVENT ALL FEEDWATER PUMPS TRIP.

16. TRANSITION FROM 250A TO 300A PIPING TO BE DETERMINED BY THE PLANT ARRANGEMENT OF THE SRV DISCHARGE LINES.

17. SRV DISCHARGE LINE PIPING TO THE QUENCHER SHALL BE QUALITY GROUP C. IN ADDITION ALL WELDS IN THE SRV DISCHARGE LINE PIPING IN THE WETWELL ABOVE THE SURFACE OF THE SUPPRESSION POOL SHALL BE NON-DESTRUCTIVELY EXAMINED TO THE REQUIREMENTS OF ASME BOILER AND PRESSURE VESSEL CODE. SECTION III, CLASS 2.

18. NOT USED

19. WHEN ALL FEEDWATER FLOW IS THROUGH A LOW FLOW FEEDWATER CONTROL VALVE. ONE FEEDWATER LINE TO THE REACTOR VESSEL IS TO BE SHUT-OFF TO MINIMIZE THERMAL CYCLING OF THE FEEDWATER NOZZLES ON THE RPV. WITH BOTH FEEDWATER LINES OPEN. FLOW MAY OSCILLATE BETWEEN THE TWO LINES DUE TO THE PARTIALLY OPEN CHECK VALVE.

20. ROUTE THE PIPE THROUGH THE MANHOLE IN BETWEEN THE DRYWELL AND THE RPV FLANGE.

21. WATER LEVEL INSTRUMENTS FOR VARIOUS RANGES ARE CALIBRATED AS STATED BELOW. ALL WATER LEVEL SWITCH SETPOINTS ARE NOMINAL, I.E., THE ANALYSES ARE PERFORMED WITH THE SWITCH TRIP UNCERTAINTY INCLUDED. THE CONTAINMENT BUILDING TEMPERATURE ASSUMED TO BE 26.7°C.

   A. FUEL ZONE: THE INSTRUMENTS ARE CALIBRATED FOR SATURATED WATER AND STEAM CONDITIONS AT 0 kPa G IN THE VESSEL AND DRYWELL WITH NO PUMP FLOW.

   B. WIDE RANGE: THE INSTRUMENTS ARE CALIBRATED FOR 7.07 MPa G IN THE VESSEL. 57.2°C IN THE DRYWELL AND 46.47 kJ/kg SUB-COOLING BELOW THE MIDDLE WATER LEVEL NOZZLE.

   C. NARROW RANGE: (SAFEGUARDS AND FEEDWATER) THE INSTRUMENTS ARE CALIBRATED FOR SATURATED WATER AND STEAM CONDITIONS AT 7.07 MPa G IN THE VESSEL AND 57.2°C IN THE DRYWELL.

   D. SHUTDOWN: THE INSTRUMENT IS CALIBRATED FOR 48.9°C WATER AT 0 MPa IN THE VESSEL AND 26.7°C IN THE DRYWELL.

22. THE TEMPERATURE ELEMENT MPL B21-TE032 MAY BE LOCATED ON THE RPV HEAD VENT LINE BETWEEN THE MOTOR-OPERATED VALVES MPL B21-F019 AND B21-F020 PROVIDING THE FOLLOWING CONDITION IS SATISFIED. THE TEMPERATURE ELEMENT MPL B21-TE032 SHALL NOT BE INFLUENCED BY THE POTENTIALLY HIGH TEMPERATURES UPSTREAM OF THE MOTOR-OPERATED VALVE MPL B21-F019 WHEN THERE IS ZERO LEAKAGE THROUGH THE MOTOR-OPERATED VALVES B21-F019 & B21-F020.

23. UNLESS OTHERWISE INDICATED, ALL REFERENCED MPL ARE PREFIXED BY B21-.

24. SEE MAIN STEAM PIPING DESIGN SPECIFICATION (B21-C001) FOR THE SPECIAL DESIGN REQUIREMENTS WHICH ARE APPLICABLE TO THE PIPING BETWEEN THE STEAM LINE INBOARD AND OUTBOARD CONTAINMENT ISOLATION VALVES.

25. SEE FEEDWATER PIPING DESIGN SPECIFICATION (B21-C010) FOR THE SPECIAL DESIGN REQUIREMENTS WHICH ARE APPLICABLE TO THE PIPING BETWEEN THE FEEDWATER LINE INBOARD AND OUTBOARD CONTAINMENT ISOLATION VALVES.

26. OPERATION OF 2 OF 2 MANUAL SWITCHES IS REQUIRED FOR GANGED OPERATION OF THE 8 SRVS USED FOR THE ADS.

27. SEE SUPPORTING DOCUMENT 1 FOR SYSTEM IDENTIFICATION AT INTERCONNECTIONS.

28. PNEUMATIC SUPPLY FROM REFERENCE DOCUMENT 33.

29. THE CONDENSING CHAMBER SHALL CONSIST OF A 25A X 25A PIPE ELBOW WHICH IS CLOSE-COUPLED TO THE MAIN STEAM LINE FLOW RESTRICTOR INSTRUMENT LINE TAPS. THE RUN OF 25A INSTRUMENT LINE FROM THE TAP ON THE MAIN STEAM LINE FLOW RESTRICTOR TO THE ELBOW SHALL BE LEVEL IN THE HORIZONTAL PLANE WITHIN 0.50 CM FOR ALL CONDITIONS. THE 25A INSTRUMENT LINE CONNECTION TO THE ELBOW SHALL BE LOCAED VERTICALLY AND EXTEND DOWNWARD. THE 25A PIPE SHALL HAVE A DOWNWARD SLOPE ≥ 1/24. PRIOR TO PENETRATING THE CONTAINMENT WALL. THE 25A PIPE SHALL BE REDUCED TO 20A. FLEXIBILITY SHALL BE PROVIDED IN THE 25A AND 20A PIPE SUCH THAT THE CONDENSING CHAMBER MOVES WITH THE RPV AS IT THERMALLY EXPANDS AND CONTRACTS.

30. INSULATE WITH INSULATION THAT HAS A MAXIMUM CONDUCTANCE OF 4.103E-01 J/h.cm ².°C. THE INSULATION SHALL EXTEND FROM THE INSTRUMENT LINE TAP TO WITHIN 5.1 CM OF THE DOWNWARD EXTENDING INSTRUMENT LINE. THE HORIZONTAL INSTRUMENT LINE TAP TO THE JUNCTION WITH THE DOWNWARD EXTENDING INSTRUMENT LINE SHOULD BE LESS THAN ONE (1) METER.

31. THE MOTOR-OPERATED VALVES (MOVS) MPL B21-F007A&B IN THE CLEAN-UP WATER (CUW) SYSTEM INJECTION LINES TO THE FEEDWATER LINES MAY BE DELETED IF THE RPV FEEDWATER NOZZLE FATIGUE USAGE IS ≤ 1.0 WITHOUT THE CUW SYS FEEDWATER LINE SELECTION FEATURE.

32. THE THERMAL SLEEVE SHOWN MAY BE DELETED IF THE STRESS ANALYSIS SHOWS THAT IT IS NOT REQUIRED.

33. SEE REFERENCE DOCUMENT 3 FOR THE INSTRUMENT SETPOINT REQUIREMENTS.

34. FOR INTERFACE CONNECTIONS. SEE THE MSIV EQUIPMENT REQUIREMENTS. SPECIFICATION SUPPORT DRAWING MPL F008 AND F009.

35. FOR VENT AND DRAIN LINES OPEN TO THE ATMOSPHERE. DOWNSTREAM OF THE OUTBOARD SHUT-OFF VALVE. THE FOLLOWING BOUNDARY CONDITIONS APPLY:

    MAXIMUM OPERATING PRESSURE - 0 MPa G
    MAXIMUM OPERATING TEMPERATURE - 66 °C
    DESIGN CLASS AND QA CLASS - 7G
    SEISMIC CLASS - C

36. PIPING DESIGN SPECIFICATION AS FOLLOWS:
    A. MAXIMUM OPERATING PRESSURE - SEE SPECIFIC BOUNDARIES ON DRAWING
    B. MAXIMUM OPERATING TEMPERATURE - SEE SPECIFIC BOUNDARIES ON DRAWING
    C. MATERIAL - SEE TABLE 5
    D. PIPING THICKNESS - SEE TABLE 5
    E. ASME CLASS - SEE SPECIFIC BOUNDARIES ON DRAWING
    F. QUALITY CLASS - SEE SPECIFIC BOUNDARIES ON DRAWING
    G. SEISMIC CLASS - SEE SPECIFIC BOUNDARIES ON DRAWING
    H. FLUID - SEE TABLE 5

37. THE RELIEF VALVE F708 IS NOT REQUIRED IF INTERNAL PROTECTION IS PROVIDED WITHIN LT004 TO LIMIT THE DIFFERENTIAL PRESSURE ACROSS THE SENSING ELEMENT.

38. THE INSTRUMENT LINES SHALL BE SEISMIC CLASS As FROM THE LAST ANCHOR POINT TO THE PRESSURE TRANSMITTERS PT028 AND PT301.

39. PROVIDES INTREFACE BETWEEN SEISMIC CATEGORY I AND NON-SEISMIC CATEGORY PIPING.

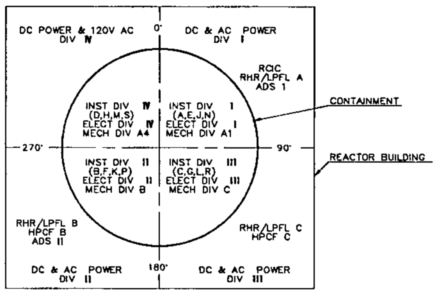
40. THE MAXIMUM OPERATING TEMPERATURE OF THE FEEDWATER LINE FROM THE SEISMIC INTERFACE TO THE UPSTREAM SIDE OF THE MOTOR OPERATED VALVE MPL B21-F001 SHALL BE DETERMINED BY THE DESIGNER OF THE FEEDWATER SYSTEM.

41. PIPE WITH A DESIGN PRESSURE OF 2.82 MPa G OR GREATER SHALL HAVE ITS MINIMUM WALL THICKNESS NO LESS THAN THAT OF A STANDARD WEIGHT PIPE. THICKER THAN STANDARD WEIGHT PIPE SHALL BE USED IF REQUIRED BY THE DESIGN PRESSURE OR OTHER REQUIREMENTS.

42. VALVES WITH A DESIGN PRESSURE OF 2.82 MPa G OR GREATER SHALL BE A MINIMUM OF CLASS 300. OR OF A HIGHER CLASS IF REQUIRED BY THE DESIGN PRESSURE.

43. THE FLOW CONTROL STATION CONTAINS THE EQUIPMENT NECESSARY FOR LOCAL FLOW INDICATION AND LOCAL FLOW CONTROL.

44. PROVIDED FOR SRV E ONLY.



ORIENTATION OF THE NUCLEAR BOILER SYSTEM INSTRUMENTS TO MEET SEPARATION REQUIREMENTS (INCLUDES ASSOCIATED INSTRUMENT SUFFIX AND ECCS LOCATION)

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

	MPL NO.
1. WATER QUALITY REQUIREMENTS	A11-3040
2. REACTOR PRESSURE VESSEL SYSTEM. ICD	B11-2020
3. NUCLEAR BOILER SYS. P&ID DATA	B21-1010
4. NUCLEAR BOILER SYSTEM. PFD	B21-1020
5. NUCLEAR BOILER SYSTEM. IBD	B21-1030
6. NOT USED	- - - -
7. NOT USED	- - - -
8. NOT USED	- - - -
9. CONTROL ROD DRIVE SYSTEM. P&ID	C12-1010
10. REACTOR RECIRCULATION SYSTEM. P&ID	B31-1010
11. FEEDWATER CONTROL SYSTEM. IBD	C31-1030
12. FEEDWATER CONTROL SYSTEM. IED	C31-1040
13. REMOTE SHUTDOWN SYSTEM. IED	C61-1040
14. REACTOR PROTECTION SYSTEM. IED	C71-1040
15. RECIRCULATION FLOW CONTROL SYS. IBD	C81-1030
16. RECIRCULATION FLOW CONTROL SYS. IED	C81-1040
17. RESIDUAL HEAT REMOVAL SYSTEM. P&ID	E11-1010
18. RESIDUAL HEAT REMOVAL SYSTEM. IBD	E11-1030
19. HIGH PRESSURE CORE FLOODER SYSTEM. IBD	E22-1030
20. LEAK DETECTION AND ISOLATION SYSTEM. IBD	E31-1030
21. LEAK DETECTION AND ISOLATION SYSTEM. IED	E31-1040
22. REACTOR CORE ISOLATION COOLING SYSTEM. P&ID	E51-1010
23. REACTOR CORE ISOLATION COOLING SYSTEM. IBD	E51-1030
24. REACTOR WATER CLEANUP SYS. P&ID	G31-1010
25. LIQUID WASTE, RADWASTE SYSTEM. P&ID	K17-1010
26. LOW CONDUCTIVITY WASTE. RADWASTE SYSTEM. P&ID	K17-1010
27. TURBINE MAIN STEAM SYSTEM. P&ID	N11-1010
28. CONDENSATE AND FEEDWATER SYSTEM. P&ID	N21-1010
29. TURBINE CONTROL SYSTEM. IBD	N32-1030
30. TURBINE CONTROL SYSTEM. IED	N32-1040
31. STEAM BYPASS & PRESSURE CONTROL SYS. IED	C85-1040
32. MAIN CONDENSER	N61-1010
33. INSTRUMENT AIR SYSTEM. P&ID	P52-1010
34. HIGH PRESS NITROGEN GAS SUPPLY SYS. P&ID	P54-1010
35. VALVE GLAND LEAKAGE TREATMENT. RADWASTE SYS. P&ID	K17-1010
36. SAMPLING SYSTEM. P&ID	P91-1010
37. NOT USED	- - - -
38. NOT USED	- - - -
39. ATMOSPHERIC CONTROL SYSTEM P&ID	T31-1010
40. MAIN STEAM PIPING EQUIPMENT REQUIREMENTS SPECIFICATION SUPPORT DRAWING	B21-C001

* DENOTES THAT THIS COMPONENT IS PART OF AN ASSEMBLY WHERE THE ENTIRE ASSEMBLY HAS ONE MPL NUMBER.

** REFERENCE INFORMATION TO BE PROVIDED AS: INTERFACE DOES NOT AFFECT THE DESIGN INFORMATION SHOWN ON THIS DRAWING OR ITS VERIFICATION.

SUPPORTING DOCUMENTS	MPL NO.
1. NUCLER PLANT SYSTEM STRUCTURE	A10-3010
2. PIPING AND INSTRUMENT SYMBOLS	A10-3030
3. GROUP CLASSIFICATION AND CONTAINMENT ISOLATION DIAGRAM	A11-1030
4. PROCESS INSTRUMENTATION REQUIREMENT SPEC	A11-3030

MPL NO. B21-1010

FIGURE 5.1-3 NUCLEAR BOILER SYSTEM P&ID (Sheet 1 of 11)

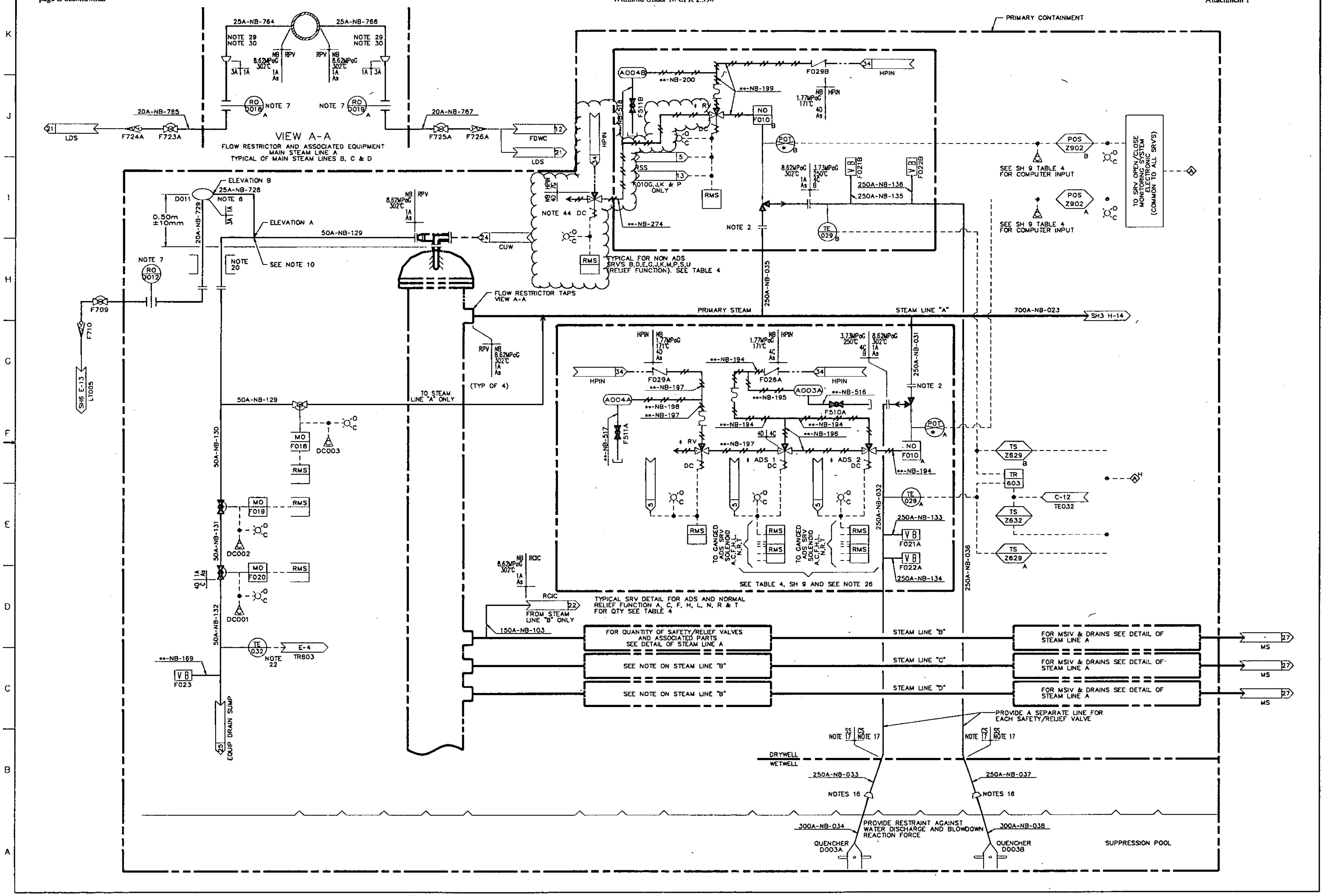


FIGURE 5.1-3 NUCLEAR BOILER SYSTEM P&ID (Sheet 2 of 11)

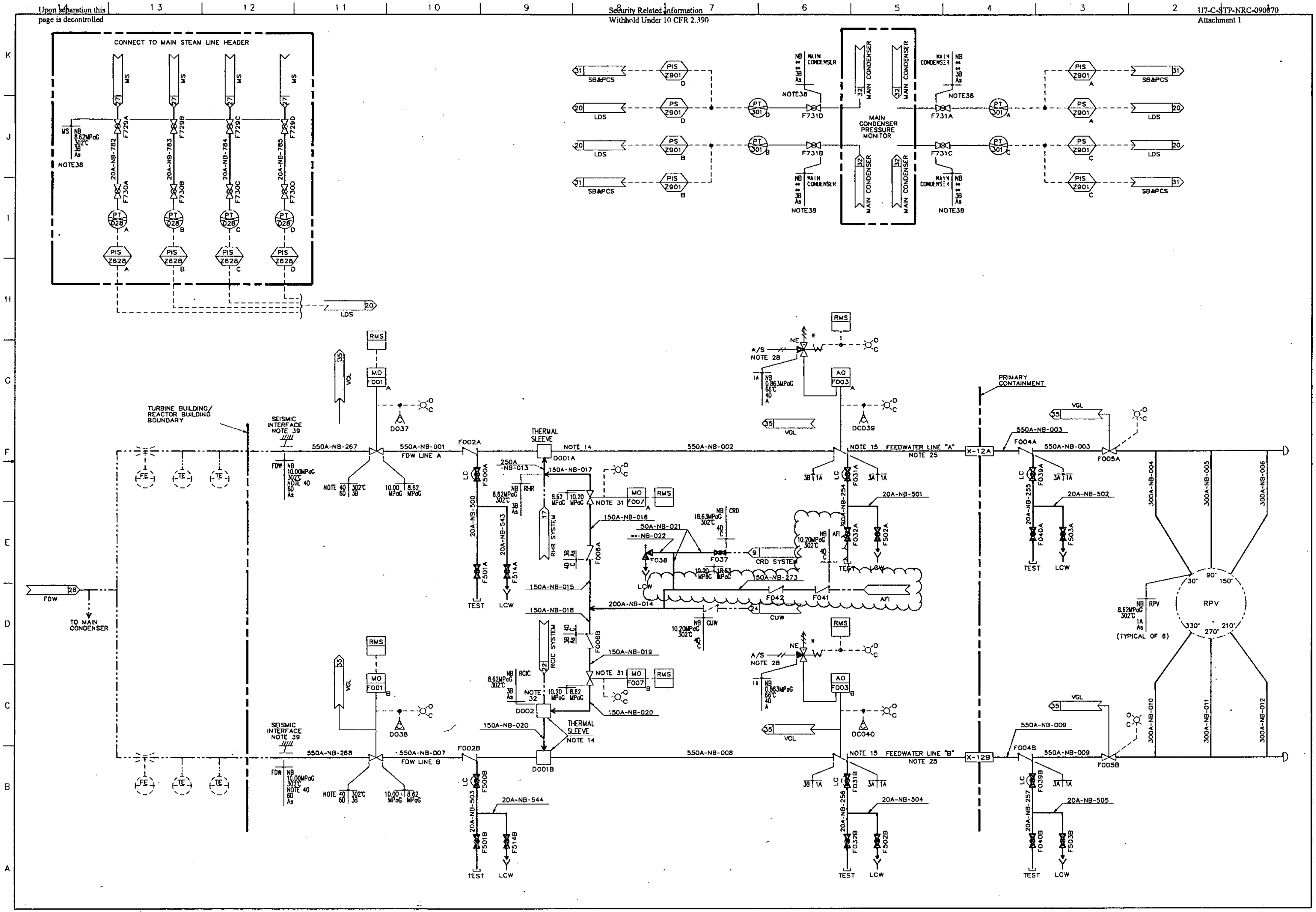


FIGURE 5.1-3 NUCLEAR BOILER SYSTEM P&ID (Sheet 4 of 11)

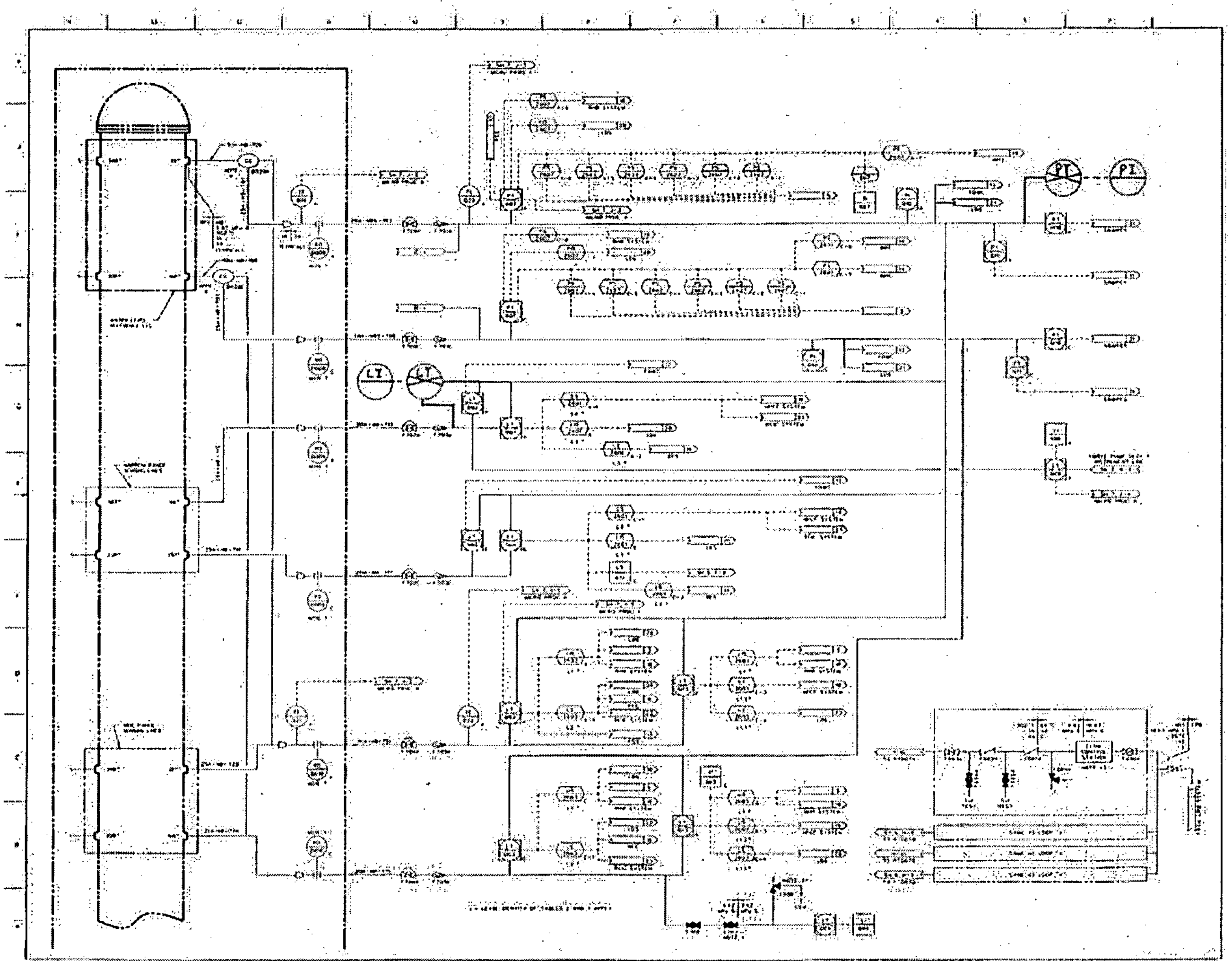


FIGURE 5.1-3 NUCLEAR BOILER SYSTEM P&ID (Sheet 5 of 11)

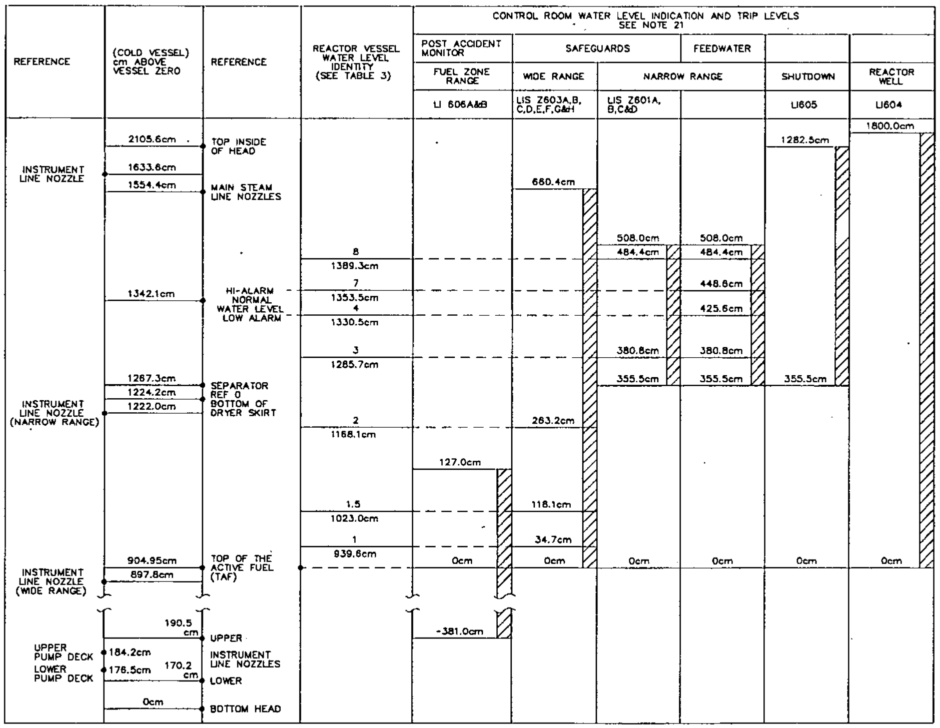
ABWR DCD/Tier 2 Rev. 5 21-74

TABLE 1. SUFFIX LETTER ASSIGNMENTS FOR SAFETY/RELIEF VALVES AND ASSOCIATED EQUIPMENT

SAFETY/RELIEF VALVE F010		P	J	M	G	S	B	K	E	U	D	N	H	T	C	L	F	R	A
TEMPERATURE ELEMENT TE029		P	J	M	G	S	B	K	E	U	D	N	H	T	C	L	F	R	A
SOLENOID VALVE	ADS 1											N	H	T	C	L	F	R	A
	ADS 2											N	H	T	C	L	F	R	A
	RV	P	J	M	G	S	B	K	E	U	D	N	H	T	C	L	F	R	A
	ALTERNATE								E										
ACCUMULATOR	A003											N	H	T	C	L	F	R	A
	A004	P	J	M	G	S	B	K	E	U	D	N	H	T	C	L	F	R	A
CHECK VALVE	F028											N	H	T	C	L	F	R	A
	F029	P	J	M	G	S	B	K	E	U	D	N	H	T	C	L	F	R	A
SRV OPEN CLOSE MONITORING SYSTEM		P	J	M	G	S	B	K	E	U	D	N	H	T	C	L	F	R	A
	COMPUTER INPUT △	NC041	NC042	NC043	NC044	NC045	NC046	NC047	NC048	NC049	NC050	NC051	NC052	NC053	NC054	NC055	NC056	NC057	NC058
* COMPUTER INPUT △		B21 NC023	B21 NC024	B21 NC025	B21 NC026	B21 NC027	B21 NC028	B21 NC029	B21 NC030	B21 NC031	B21 NC032	B21 NC033	B21 NC034	B21 NC035	B21 NC036	B21 NC037	B21 NC038	B21 NC039	B21 NC040
SPRING SET PRESSURE (MPa G)		7.92	7.92	7.99	7.99	7.99	7.99	8.06	8.06	8.06	8.06	8.13	8.13	8.13	8.13	8.20	8.20	8.20	8.20
SPRING RESEAT PRESSURE (MPa G)		7.37	7.37	7.44	7.44	7.44	7.44	7.50	7.50	7.50	7.50	7.56	7.56	7.56	7.56	7.63	7.63	7.63	7.63
RELIEF SET PRESSURE (MPa G)		7.51	7.58	7.65	7.65	7.65	7.65	7.72	7.72	7.72	7.72	7.79	7.79	7.79	7.79	7.86	7.86	7.86	7.86
RELIEF RESEAT PRESSURE (MPa G)		7.00	7.07	7.14	7.14	7.14	7.14	7.21	7.21	7.21	7.21	7.28	7.28	7.28	7.28	7.35	7.35	7.35	7.35
	PT007A THRU D/ PS-Z607A-6 THRU D-6	22 OF A,B,C,D																	
	PT007A THRU D/ PS-Z607A-5 THRU D-5		22 OF A,B,C,D																
	PT007A THRU D/ PS-Z607A-4 THRU D-4			← 22 OF A,B,C,D →															
	PT007A THRU D/ PS-Z607A-3 THRU D-3							← 22 OF A,B,C,D →											
	PT007A THRU D/ PS-Z607A-2 THRU D-2											← 22 OF A,B,C,D →							
	PT007A THRU D/ PS-Z607A-1 THRU D-1															← 22 OF A,B,C,D →			

* COMPUTER INPUTS FOR SRV POSITION SEE PERFORMANCE MONITORING AND CONTROL SYSTEM C91-4010

TABLE 2 ELEVATION CORRELATION CHART



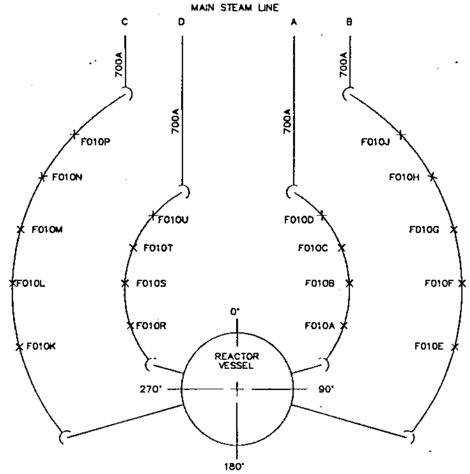


FIG. 3

SAFETY/RELIEF VALVE ORIENTATION AND STEAM PIPING LINE SIZES

TABLE 3 WATER LEVEL TRIP FUNCTION

REACTOR VESSEL WATER LEVEL	DESCRIPTION OF TRIPS	INSTRUMENT PROVIDING TRIP SIGNAL	NOTES
8	TRIPS RCIC TURBINE	LS-Z601A-1 THRU D-1	NARROW RANGE
	TRIP HPCF INJECTION VALVES	LS-Z601A-1 THRU D-1	NARROW RANGE
	CLOSE MAIN TURBINE STOP VALVES TRIPS FEEDWATER PUMPS	SEE REFERENCE DOCUMENT 12	NARROW RANGE
7	HIGH LEVEL ALARM	SEE REFERENCE DOCUMENT 12	NARROW RANGE
4	LOW LEVEL ALARM (RRS FLOW RUN BACK ON TRIP OF FEED PUMP)		NARROW RANGE
3	SCRAMS REACTOR	LIS-Z601A THRU D	NARROW RANGE
	CLOSE RHR SHUTDOWN COOLING ISOLATION VALVES	LIS-Z601A THRU D	NARROW RANGE
	CLOSE CONTAINMENT ISOL VALVES LDS EXCEPT DW COOLING AND CUW ISOL VALVES AND MSIVS	LIS-Z601A THRU D	NARROW RANGE
	TRIP 4 OF RRS PUMPS	SEE REFERENCE DOCUMENT 12	NARROW RANGE
2	INITIATES RCIC	LS-Z603A-1 THRU D-1	WIDE RANGE
	TRIP REMAINING 6 RRS PUMPS	LS-Z603A-1 THRU D-1	WIDE RANGE
	CLOSE CUW ISOL VALVES	LS-Z603A-2 THRU D-2	WIDE RANGE
1.5	INITIATES HPCF B & C	LS-Z603E-3 THRU H-3	WIDE RANGE
	CLOSE MSIV'S & DW COOLING SYSTEM ISOL VALVES	LS-Z603E-4 THRU H-4	WIDE RANGE
1	INITIATES ADS	LIS-Z603A THRU H	WIDE RANGE
	INITIATES RHR/LPFL MODE	LIS-Z603A THRU H	WIDE RANGE

TABLE 5: PIPING SPECIFICATIONS

PIPE NO.	SCHEDULE	MATERIAL	FLUID
001	**	CS	W
002	**	CS	W
003	100	CS	W
004	100	CS	W
005	100	CS	W
006	100	CS	W
007	**	CS	W
008	**	CS	W
009	100	CS	W
010	100	CS	W
011	100	CS	W
012	100	CS	W
013	**	CS	W
014	**	CS	W
015	**	CS	W
016	**	CS	W
017	**	CS	W
018	**	CS	W
019	**	CS	W
020	**	CS	W
021	**	CS	W
022	**	CS	W
023	80	CS	S
024	**	CS	S
025	80	CS	S
026	**	CS	S
027	80	CS	S
028	**	CS	S
029	80	CS	S
030	**	CS	S
031	N/A	CS	S
032	80	CS	S
033	60	SS	S
034	60	SS	S
035	N/A	CS	S
036	80	CS	S
037	60	SS	S
038	60	SS	S
039	N/A	CS	S
040	80	CS	S
041	60	SS	S
042	60	SS	S
043	N/A	CS	S
044	80	CS	S
045	60	SS	S
046	60	SS	S
047	N/A	CS	S
048	80	CS	S
049	60	SS	S
050	60	SS	S
051	N/A	CS	S
052	80	CS	S
053	60	SS	S
054	60	SS	S
055	N/A	CS	S
056	80	CS	S
057	60	SS	S
058	60	SS	S
059	N/A	CS	S
060	80	CS	S
061	60	SS	S
062	60	SS	S
063	N/A	CS	S
064	80	CS	S
065	60	SS	S
066	60	SS	S
067	N/A	CS	S
068	80	CS	S
069	60	SS	S
070	60	SS	S
071	N/A	CS	S
072	80	CS	S
073	60	SS	S
074	60	SS	S
075	N/A	CS	S
076	80	CS	S
077	60	SS	S
078	60	SS	S
079	N/A	CS	S
080	80	CS	S

TABLE 5: PIPING SPECIFICATIONS (CONT'D)

PIPE NO.	SCHEDULE	MATERIAL	FLUID
081	60	SS	S
082	60	SS	S
083	N/A	CS	S
084	80	CS	S
085	60	SS	S
086	60	SS	S
087	N/A	CS	S
088	80	CS	S
089	60	SS	S
090	60	SS	S
091	N/A	CS	S
092	80	CS	S
093	60	SS	S
094	60	SS	S
095	N/A	CS	S
096	80	CS	S
097	60	SS	S
098	60	SS	S
099	N/A	CS	S
100	80	CS	S
101	60	SS	S
102	60	SS	S
103	120	CS	S
104	160	CS	S
105	160	CS	S
106	160	CS	S
107	160	CS	S
108	160	CS	S
109	**	CS	S
110	**	CS	S
111	**	CS	S
112	**	CS	S
113	**	CS	S
114	**	CS	S
115	**	CS	S
116	**	CS	S
117	**	CS	S
118	**	CS	S
119	**	CS	S
120	**	CS	S
121	**	CS	S
122	**	CS	S
123	**	CS	S
124	**	CS	S
125	**	CS	S
126	**	CS	S
127	**	CS	S
128	**	CS	S
129	80	CS	S
130	80	CS	S
131	80	CS	S
132	**	CS	S
133	80	CS	S
134	80	CS	S
135	80	CS	S
136	80	CS	S
137	80	CS	S
138	80	CS	S
139	80	CS	S
140	80	CS	S
141	80	CS	S
142	80	CS	S
143	80	CS	S
144	80	CS	S
145	80	CS	S
146	80	CS	S
147	80	CS	S
148	80	CS	S
149	80	CS	S
150	80	CS	S
151	80	CS	S
152	80	CS	S
153	80	CS	S
154	80	CS	S
155	80	CS	S
156	80	CS	S
157	80	CS	S
158	80	CS	S
159	80	CS	S
160	80	CS	S

TABLE 5: PIPING SPECIFICATIONS (CONT'D)

PIPE NO.	SCHEDULE	MATERIAL	FLUID
161	80	CS	S
162	80	CS	S
163	80	CS	S
164	80	CS	S
165	80	CS	S
166	80	CS	S
167	80	CS	S
168	80	CS	S
169	**	CS	S
170	**	SS	N
171	**	SS	N
172	**	SS	N
173	**	SS	N
174	**	SS	N
175	**	SS	N
176	**	SS	N
177	**	SS	N
178	**	SS	N
179	**	SS	N
180	**	SS	N
181	**	SS	N
182	**	SS	A
183	**	SS	A
184	**	SS	A
185	**	SS	A
186	**	SS	A
187	**	SS	A
188	**	SS	A
189	**	SS	A
190	**	SS	A
191	**	SS	A
192	**	SS	A
193	**	SS	A
194	**	SS	N
195	**	SS	N
196	**	SS	N
197	**	SS	N
198	**	SS	N
199	**	SS	N
200	**	SS	N
201	**	SS	N
202	**	SS	N
203	**	SS	N
204	**	SS	N
205	**	SS	N
206	**	SS	N
207	**	SS	N
208	**	SS	N
209	**	SS	N
210	**	SS	N
211	**	SS	N
212	**	SS	N
213	**	SS	N
214	**	SS	N
215	**	SS	N
216	**	SS	N
217	**	SS	N
218	**	SS	N
219	**	SS	N
220	**	SS	N
221	**	SS	N
222	**	SS	N
223	**	SS	N
224	**	SS	N
225	**	SS	N
226	**	SS	N
227	**	SS	N
228	**	SS	N
229	**	SS	N
230	**	SS	N
231	**	SS	N
232	**	SS	N
233	**	SS	N
234	**	SS	N
235	**	SS	N
236	**	SS	N
237	**	SS	N
238	**	SS	N
239	**	SS	N
240	**	SS	N

TABLE 5: PIPING SPECIFICATIONS (CONT'D)

PIPE NO.	SCHEDULE	MATERIAL	FLUID
241	**	SS	N
242	**	SS	N
243	**	SS	N
244	**	SS	N
245	**	SS	N
246	**	SS	N
247	**	SS	N
248	**	SS	N
249	**	SS	N
250	**	SS	N
251	**	SS	N
252	**	SS	N
253	**	SS	N
254	160	CS	W
255	160	CS	W
256	160	CS	W
257	160	CS	W
258	160	CS	S
259	160	CS	S
260	160	CS	S
261	160	CS	S
262	160	CS	S
263	160	CS	S
264	160	CS	S
265	160	CS	S
266	**	CS	S
267	**	CS	W
268	**	CS	W
269	**	CS	S
270	**	CS	S
271	**	CS	S
272	**	CS	S
273	**	CS	W
274	**	SS	N
500	**	CS	W
501	160	CS	W
502	160	CS	W
503	**	CS	W
504	160	CS	W
505	160	CS	W
506	**	CS	S
507	**	CS	S
508	**	SS	N
509	**	SS	A
510	**	SS	N
511	**	SS	A
512	**	SS	N
513	**	SS	A
514	**	SS	N
515	**	SS	A
516	**	SS	N
517	**	SS	N
518	**	SS	N
519	**	SS	N
520	**	SS	N
521	**	SS	N
522	**	SS	N
523	**	SS	N
524	**	SS	N
525	**	SS	N
526	**	SS	N
527	**	SS	N
528	**	SS	N
529	**	SS	N
530	**	SS	N
531	**	SS	N
532	**	SS	N
533	**	SS	N
534	**	SS	N
535	**	SS	N
536	**	SS	N
537	**	SS	N
538	**	SS	N
539	**	SS	N
540	**	SS	N
541	**	SS	N
542	**	CS	S
543	160	CS	W
544	160	CS	W

TABLE 5: PIPING SPECIFICATIONS (CONT'D)

PIPE NO.	SCHEDULE	MATERIAL	FLUID
700	80	SS	S
701	**	SS	W
702	**	SS	W
703	80	SS	S
704	**	SS	W
705	**	SS	W
706	80	SS	S
707	**	SS	W
708	**	SS	W
709	80	SS	S
710	**	SS	W
711	**	SS	W
712	80	SS	W
713	**	SS	W
714	80	SS	W
715	**	SS	W
716	80	SS	W
717	**	SS	W
718	80	SS	W
719	**	SS	W
720	80	SS	W
721	**	SS	W
722	80	SS	W
723	**	SS	W
724	80	SS	W
725	**	SS	W
726	80	SS	W
727	**	SS	W
728	**	CS	S
729	**	SS	W
730	**	SS	S
731	**	SS	S
732	80	SS	W
733	**	SS	W
734	80	SS	W
735	**	SS	W
736	80	SS	W
737	**	SS	W
738	80	SS	W
739	**	SS	W
740	80	SS	W
741	**	SS	W
742	80	SS	W
743	**	SS	W
744	80	SS	W
745	**	SS	W
746	80	SS	W
747	**	SS	W
748	80	SS	W
749	**	SS	W
750	80	SS	W
751	**	SS	W
752	80	SS	W
753	**	SS	W
754	80	SS	W
755	**	SS	W
756	80	SS	W
757	**	SS	W
758	80	SS	W
759	**	SS	W
760	80	SS	W
761	**	SS	W
762	80	SS	W
763	**	SS	W
764	**	SS	S
765	**	SS	W
766	**	SS	S
767	**	SS	W
768	**	SS	S
769	**	SS	W
770	**	SS	S
771	**	SS	W
772	**	SS	S
773	**	SS	W
774	**	SS	S
775	**	SS	W
776	**	SS	S
777	**	SS	W
778	**	SS	S
779	**	SS	W

TABLE 5: PIPING SPECIFICATIONS (CONT'D)

PIPE NO.	SCHEDULE	MATERIAL	FLUID
780	**	CS	S
781	**	CS	S
782	**	SS	S
783	**	SS	S
784	**	SS	S
785	**	SS	S

FIGURE 5.1-3 NUCLEAR BOILER SYSTEM P&ID (Sheet 10 of 11)

TABLE 6 : PIPE NUMBERS FOR THE MAIN STEAM LINES

MAIN STEAM LINE	RPV TO THE OUTBOARD MSIV	OUTBOARD MSIV TO SEISMIC INTERFACE	SEISMIC INTERFACE TO MAIN STEAM SYSTEM	OUTBOARD MSIV TEST LINE	
				OUTBOARD MSIV TO REDUCER	DOWNSTREAM OF REDUCER
A	700A-NB-023	700A-NB-024	700A-NB-269	50A-NB-258	20A-NB-259
B	700A-NB-025	700A-NB-026	700A-NB-270	50A-NB-260	20A-NB-261
C	700A-NB-027	700A-NB-028	700A-NB-271	50A-NB-262	20A-NB-263
D	700A-NB-029	700A-NB-030	700A-NB-272	50A-NB-264	20A-NB-265

TABLE 7 : PIPE NUMBERS FOR THE SAFETY/RELIEF VALVE (SRV) DISCHARGE LINES

SRV	SRV DISCHARGE LINE				VACUUM BREAKER LINES	
	MSL TO SRV NOTE 2	SRV TO DIAPHRAGM FLOOR	DIAPHRAGM FLOOR TO REDUCER	REDUCER TO QUENCHER	UPSTREAM	DOWNSTREAM
F010A	250A-NB-031	250A-NB-032	250A-NB-033	300A-NB-034	250A-NB-133	250A-NB-134
F010B	250A-NB-035	250A-NB-036	250A-NB-037	300A-NB-038	250A-NB-135	250A-NB-136
F010C	250A-NB-039	250A-NB-040	250A-NB-041	300A-NB-042	250A-NB-137	250A-NB-138
F010D	250A-NB-043	250A-NB-044	250A-NB-045	300A-NB-046	250A-NB-139	250A-NB-140
F010E	250A-NB-047	250A-NB-048	250A-NB-049	300A-NB-050	250A-NB-141	250A-NB-142
F010F	250A-NB-051	250A-NB-052	250A-NB-053	300A-NB-054	250A-NB-143	250A-NB-144
F010G	250A-NB-055	250A-NB-056	250A-NB-057	300A-NB-058	250A-NB-145	250A-NB-146
F010H	250A-NB-059	250A-NB-060	250A-NB-061	300A-NB-062	250A-NB-147	250A-NB-148
F010J	250A-NB-063	250A-NB-064	250A-NB-065	300A-NB-066	250A-NB-149	250A-NB-150
F010K	250A-NB-067	250A-NB-068	250A-NB-069	300A-NB-070	250A-NB-151	250A-NB-152
F010L	250A-NB-071	250A-NB-072	250A-NB-073	300A-NB-074	250A-NB-153	250A-NB-154
F010M	250A-NB-075	250A-NB-076	250A-NB-077	300A-NB-078	250A-NB-155	250A-NB-156
F010N	250A-NB-079	250A-NB-080	250A-NB-081	300A-NB-082	250A-NB-157	250A-NB-158
F010P	250A-NB-083	250A-NB-084	250A-NB-085	300A-NB-086	250A-NB-159	250A-NB-160
F010R	250A-NB-087	250A-NB-088	250A-NB-089	300A-NB-090	250A-NB-161	250A-NB-162
F010S	250A-NB-091	250A-NB-092	250A-NB-093	300A-NB-094	250A-NB-163	250A-NB-164
F010T	250A-NB-095	250A-NB-096	250A-NB-097	300A-NB-098	250A-NB-165	250A-NB-166
F010U	250A-NB-099	250A-NB-100	250A-NB-101	300A-NB-102	250A-NB-167	250A-NB-168

TABLE 8 : PIPE NUMBERS FOR THE MAIN STEAM ISOLATION VALVE (MSIV) PNEUMATIC LINES

MSIV	OPENING-CHECK VALVE TO MSIV	OPENING-FROM ACCUMULATOR	CLOSING-VALVE CONTROL PANEL TO MSIV	DRAIN LINE
F008A	**-NB-170	**-NB-171	**-NB-172	**-NB-508
F008B	**-NB-173	**-NB-174	**-NB-175	**-NB-509
F008C	**-NB-176	**-NB-177	**-NB-178	**-NB-510
F008D	**-NB-179	**-NB-180	**-NB-181	**-NB-511
F009A	**-NB-182	**-NB-183	**-NB-184	**-NB-512
F009B	**-NB-185	**-NB-186	**-NB-187	**-NB-513
F009C	**-NB-188	**-NB-189	**-NB-190	**-NB-514
F009D	**-NB-191	**-NB-192	**-NB-193	**-NB-515

TABLE 10 : PIPE NUMBERS FOR THE MAIN STEAM LINE (MSL) INSTRUMENT LINES

MAIN STEAM LINE	MSL FLOW RESTRICTOR INSTRUMENT LINES				MSL PRESSURE TEST POINT
	INSTRUMENT LINE TO LDS		INSTRUMENT LINE TO LDS & FDWC		
	MSL TO REDUCER	REDUCER TO EXCESS FLOW CHECK VALVE	MSL TO REDUCER	REDUCER TO EXCESS FLOW CHECK VALVE	
A	25A-NB-764	20A-NB-765	25A-NB-766	20A-NB-767	20A-NB-780
B	25A-NB-768	20A-NB-769	25A-NB-770	20A-NB-771	20A-NB-781
C	25A-NB-772	20A-NB-773	25A-NB-774	20A-NB-775	-
D	25A-NB-776	20A-NB-777	25A-NB-778	20A-NB-779	-

TABLE 9 : PIPE NUMBERS FOR THE SAFETY/RELIEF VALVE (SRV) PNEUMATIC LINES

SRV	ADS PNEUMATIC LINES				PNEUMATIC LINES FOR POWER-ACTUATED RELIEF			PNEUMATIC LINE FROM AFI PUMP HOUSE
	CHECK VALVE TO SOV "ADS 2" TO SRV	BRANCH LINE FROM ACCUMULATOR	BRANCH LINE THRU SOV "ADS 1" TO SOV "ADS 2"	ACCUMULATOR DRAIN LINE	CHECK VALVE TO SOV "ADS 1" (ADS SRV) OR SRV (NON-ADS SRV)	BRANCH LINE FROM ACCUMULATOR	ACCUMULATOR DRAIN LINE	MANUAL SOV TO RV SOV
F010A	**-NB-194	**-NB-195	**-NB-196	**-NB-516	**-NB-197	**-NB-198	**-NB-517	
F010B					**-NB-199	**-NB-200	**-NB-518	
F010C	**-NB-201	**-NB-202	**-NB-203	**-NB-519	**-NB-204	**-NB-205	**-NB-520	
F010D					**-NB-206	**-NB-207	**-NB-521	
F010E					**-NB-208	**-NB-209	**-NB-522	**-NB-274
F010F	**-NB-210	**-NB-211	**-NB-212	**-NB-523	**-NB-213	**-NB-214	**-NB-524	
F010G					**-NB-215	**-NB-216	**-NB-525	
F010H	**-NB-217	**-NB-218	**-NB-219	**-NB-526	**-NB-220	**-NB-221	**-NB-527	
F010J					**-NB-222	**-NB-223	**-NB-528	
F010K					**-NB-224	**-NB-225	**-NB-529	
F010L	**-NB-226	**-NB-227	**-NB-228	**-NB-530	**-NB-229	**-NB-230	**-NB-531	
F010M					**-NB-231	**-NB-232	**-NB-532	
F010N	**-NB-233	**-NB-234	**-NB-235	**-NB-533	**-NB-236	**-NB-237	**-NB-534	
F010P					**-NB-238	**-NB-239	**-NB-535	
F010R	**-NB-240	**-NB-241	**-NB-242	**-NB-536	**-NB-243	**-NB-244	**-NB-537	
F010S					**-NB-245	**-NB-246	**-NB-538	
F010T	**-NB-247	**-NB-248	**-NB-249	**-NB-539	**-NB-250	**-NB-251	**-NB-540	
F010U					**-NB-252	**-NB-253	**-NB-541	

FIGURE 5.1-3 NUCLEAR BOILER SYSTEM P&ID (Sheet 11 of 11)

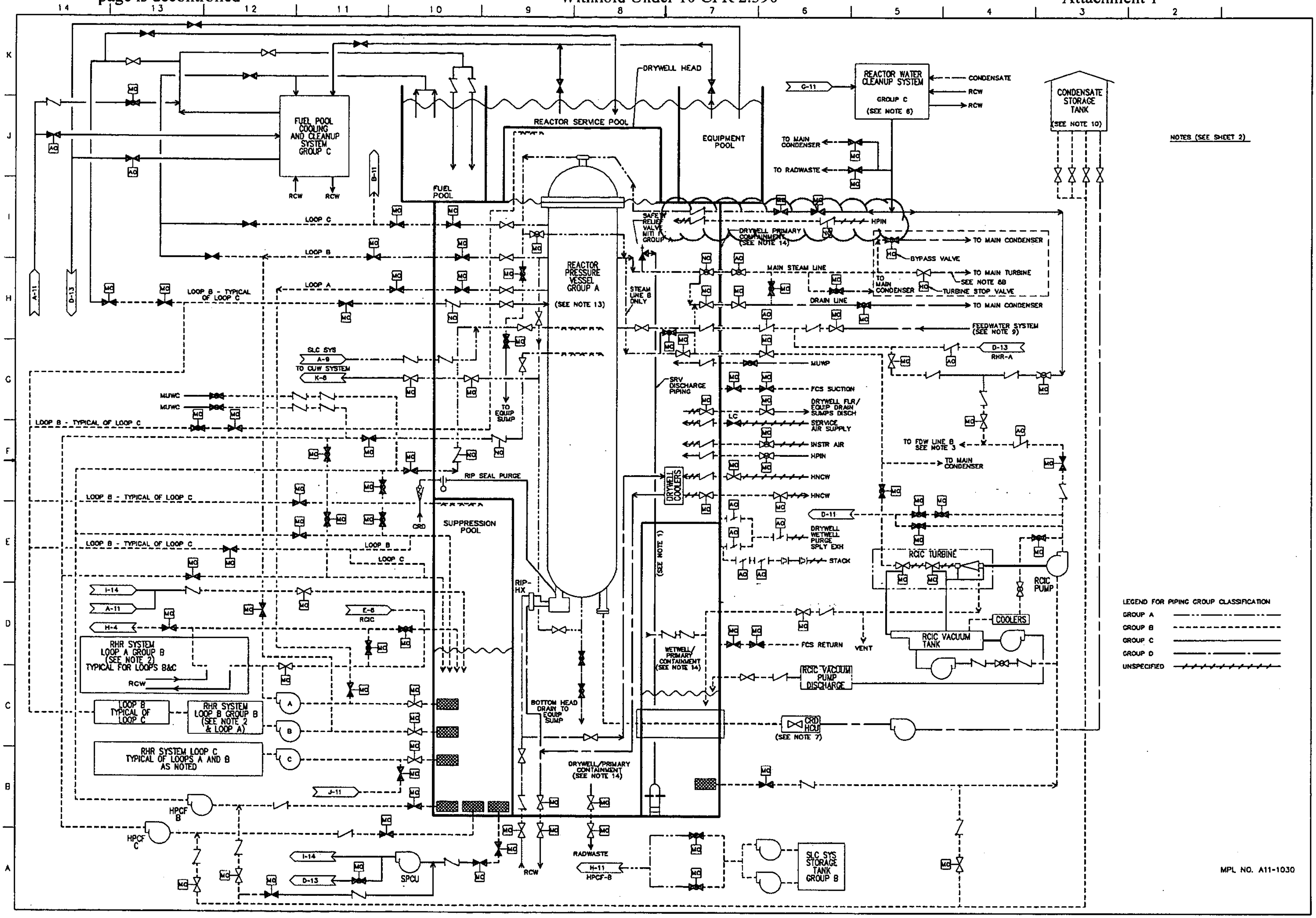


FIGURE 6.2-38 GROUP CLASSIFICATION AND CONTAINMENT ISOLATION DIAGRAM (Sheet 1 of 2)

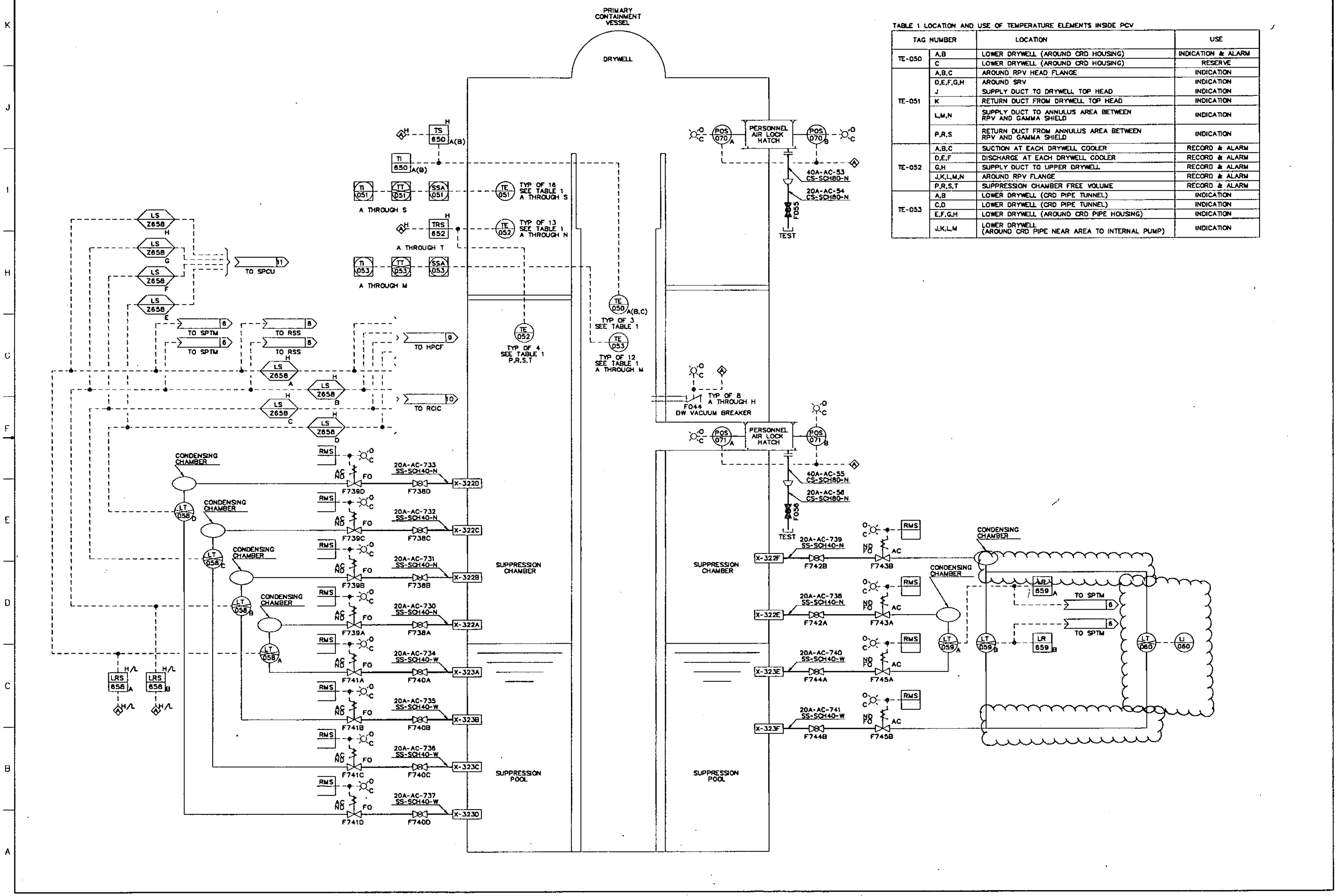


TABLE 1 LOCATION AND USE OF TEMPERATURE ELEMENTS INSIDE PCV

TAG NUMBER		LOCATION	USE
TE-050	A,B	LOWER DRYWELL (AROUND CRD HOUSING)	INDICATION & ALARM
	C	LOWER DRYWELL (AROUND CRD HOUSING)	RESERVE
TE-051	A,B,C	AROUND RPV HEAD FLANGE	INDICATION
	D,E,F,G,H	AROUND SRV	INDICATION
	J	SUPPLY DUCT TO DRYWELL TOP HEAD	INDICATION
	K	RETURN DUCT FROM DRYWELL TOP HEAD	INDICATION
	L,M,N	SUPPLY DUCT TO ANNULUS AREA BETWEEN RPV AND GAMMA SHIELD	INDICATION
	P,R,S	RETURN DUCT FROM ANNULUS AREA BETWEEN RPV AND GAMMA SHIELD	INDICATION
TE-052	A,B,C	SUCTION AT EACH DRYWELL COOLER	RECORD & ALARM
	D,E,F	DISCHARGE AT EACH DRYWELL COOLER	RECORD & ALARM
	G,H	SUPPLY DUCT TO UPPER DRYWELL	RECORD & ALARM
	J,K,L,M,N	AROUND RPV FLANGE	RECORD & ALARM
	P,R,S,T	SUPPRESSION CHAMBER FREE VOLUME	RECORD & ALARM
TE-053	A,B	LOWER DRYWELL (CRD PIPE TUNNEL)	INDICATION
	C,D	LOWER DRYWELL (CRD PIPE TUNNEL)	INDICATION
	E,F,G,H	LOWER DRYWELL (AROUND CRD PIPE HOUSING)	INDICATION
	J,K,L,M	LOWER DRYWELL (AROUND CRD PIPE NEAR AREA TO INTERNAL PUMP)	INDICATION

FIGURE 6.2-39 ATMOSPHERIC CONTROL SYSTEM P&ID (Sheet 2 of 3)

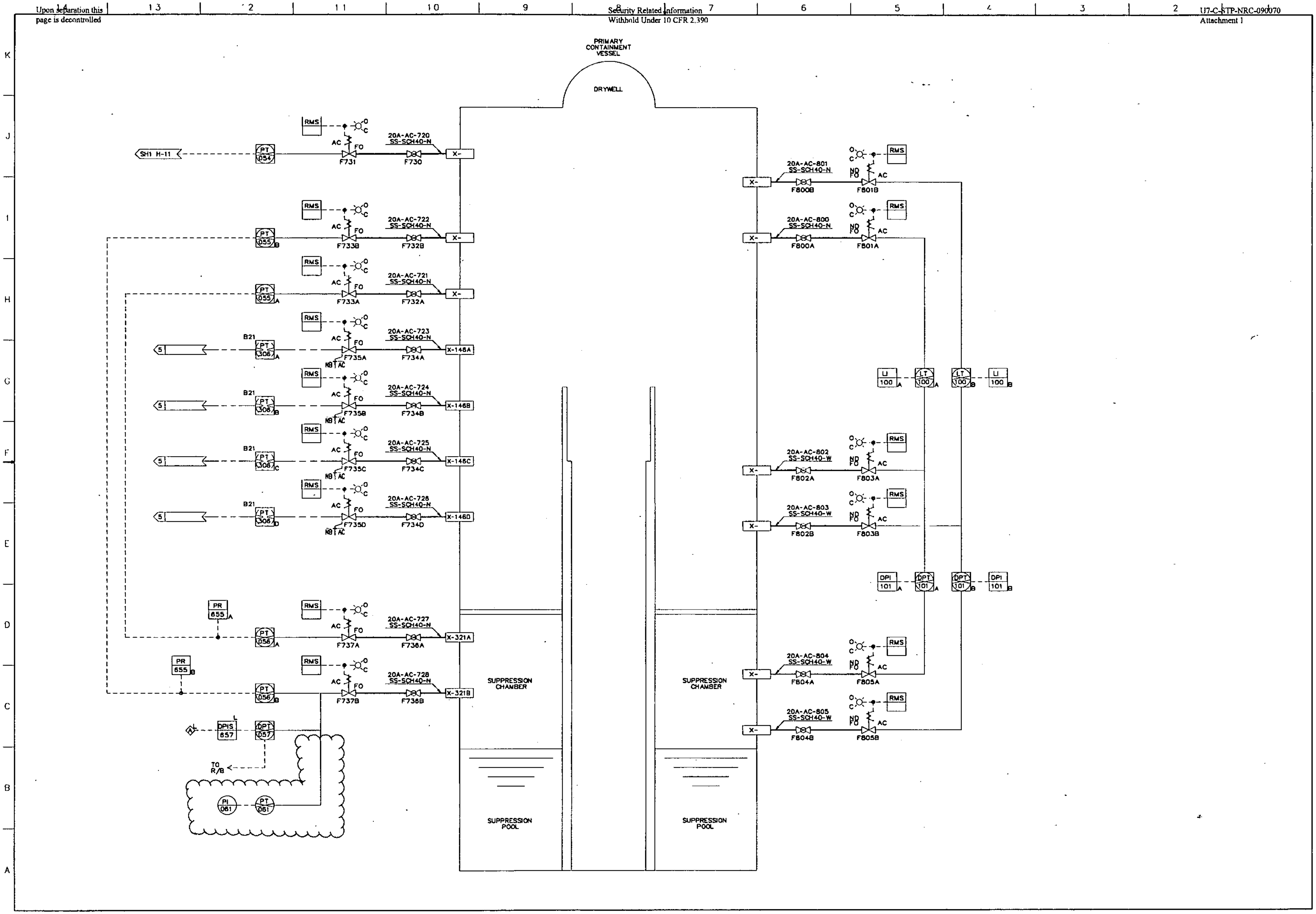
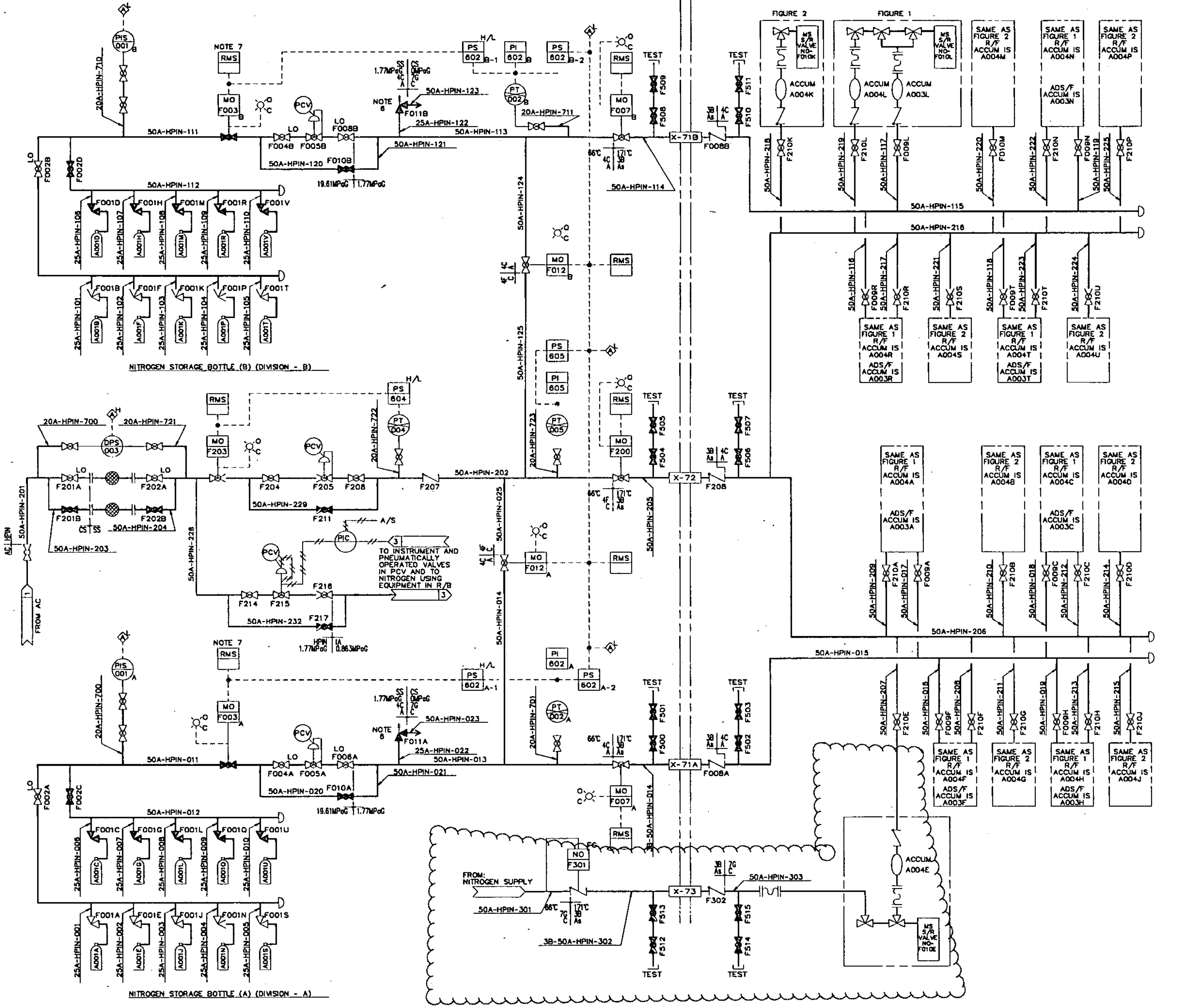


FIGURE 6.2-39 ATMOSPHERIC CONTROL SYSTEM P&ID (Sheet 3 of 3)



NOTES:

1. EMERGENCY NITROGEN SUPPLY SYSTEM DIVISION A AND B SHALL BE LOCATED SEPARATELY FROM EACH OTHER.
2. QUALITY GROUP CLASSIFICATION IS 4C EXCEPT FOR CITATION.
3. PIPING SPECIFICATION ARE AS FOLLOWS EXCEPT FOR CITATION.
   MATERIAL - SS.SCHEDULE 40.FLUID - N2GAS
4. IF ON/OFF INTERVAL OF MOTOR OPERATED VALVES (F003A,B, F203) IS TOO SHORT, RESERVOIR TANKS SHALL BE PROVIDED.
5. THE MOTOR DRIVEN VALVES (F003A/B) ARE NORMALLY CLOSED WITH LOCK. THE OPEN/CLOSE SIGNAL IS AVAILABLE WITH AUTO MODE.
6. DISCHARGE AREA OF RV SHALL BE FAR FROM OPERATOR ACCESS AREA.
7. KEY OPERATED

REFERENCE DOCUMENTS	MPL NO.
1. ATMOSPHERIC CONTROL SYS P&ID	T31-1010
2. NUCLEAR BOILER SYS P&ID	B21-1010
3. INSTRUMENT AIR SYS P&ID	P52-1010
4. PIPING AND INSTRUMENT. SYMBOLS DIAGRAM	A10-3030
5. RADWASTE SYSTEM P&ID	K17-1010

MPL NO.P54-1010

FIGURE 6.7-1 HIGH PRESSURE NITROGEN GAS SUPPLY SYSTEM P&ID (Sheet 1 of 1)

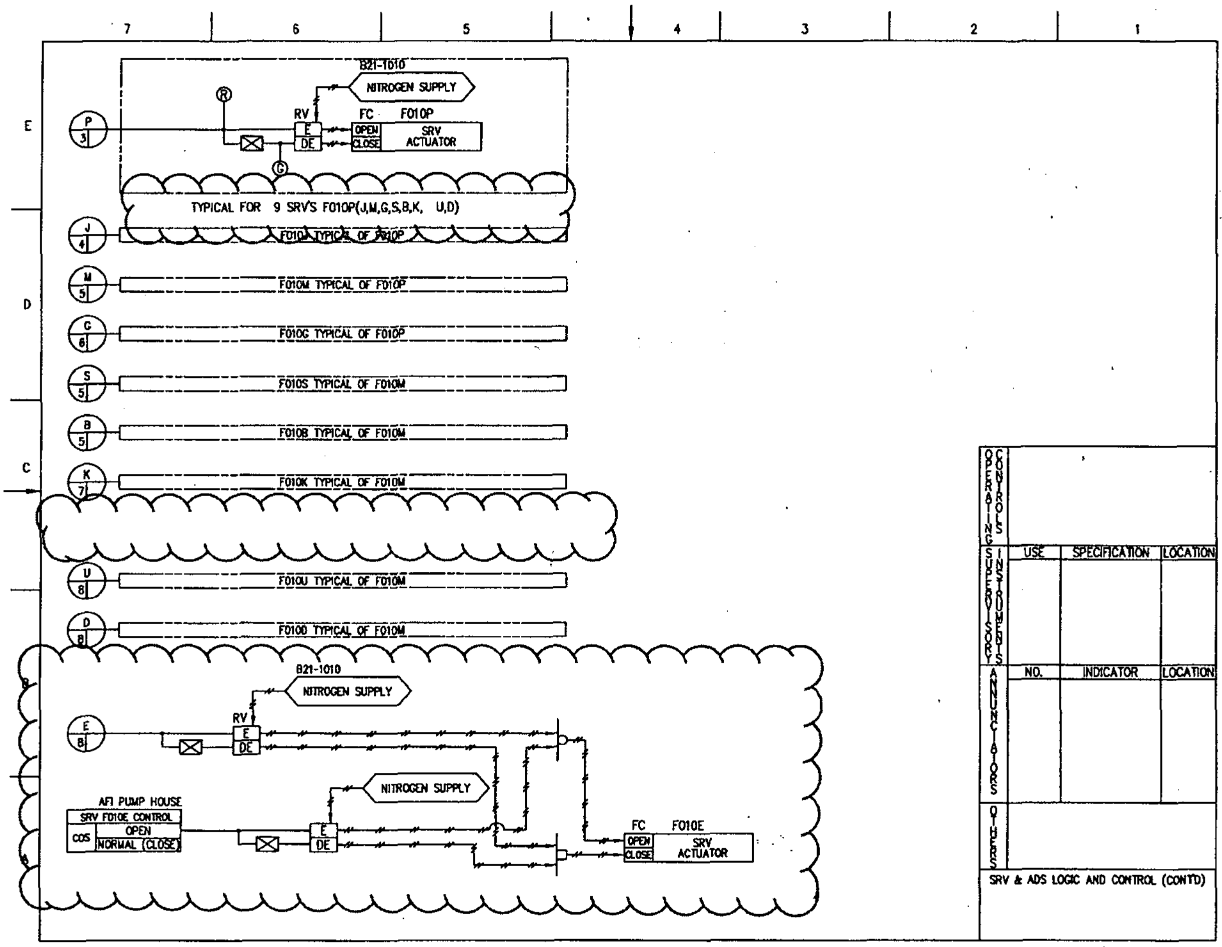


FIGURE 7.3-2 NUCLEAR BOILER SYSTEM IBD (Sheet 17 of 37)
ABWR DCD/Tier 2

NOT FOR PUBLIC RELEASE

Figure 9A.4-4 REACTOR BUILDING FIRE PROTECTION AT ELEVATION 12300mm

NOT FOR PUBLIC RELEASE

Figure 9A.4–5 REACTOR BUILDING FIRE PROTECTION AT ELEVATION 18100mm

NOT FOR PUBLIC RELEASE

FIGURE 9A.4-6 REACTOR BUILDING FIRE PROTECTION AT ELEVATION 23500 mm

ABWR DCD/Tier 2 Rev 5 21-560

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Figure 9A.4-8 REACTOR BUILDING FIRE PROTECTION AT ELEVATION 31700/38200mm

21-562

ATTACHMENT 2 – Revisions to the ABWR Design Certification Rule

In Appendix A to 10 C.F.R. Part 52, the first sentence of Section III.A is revised to read as follows:

Tier 1, Tier 2, and the generic technical specifications in the U.S. ABWR Design Control Document, Revision 5, dated June 2009, are approved for incorporation by reference by the Director of the Office of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR Part 51.