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SSAR MODIFICATION PAGES

1.3 Comparison Tables

This section highlights the principal design features of the plant and compares its major features with those of other BWR facilities. The design of this facility is based on proven technology obtained during the development, design, construction, and operation of BWRs of similar types. The data, performance characteristics, and other information presented here represent a current, firm design.

1.3.1 Nuclear Steam Supply System Design Characteristics

Table 1.3-1 summarizes the design and operating characteristics for the nuclear steam supply systems. Parameters are related to power output for a single plant unless otherwise noted.

1.3.2 Engineered Safety Features Design Characteristics

Table 1.3-2 compares the engineered safety features design characteristics.

1.3.3 Containment Design Characteristics

Table 1.3-3 compares the containment design characteristics.

1.3.4 Structural Design Characteristics

Table 1.3-4 compares the structural design characteristics.

1.3.5 Instrumentation and Electrical Systems Design Characteristics

Table 7.1-1 compares the instrumentation and electrical systems design characteristics.

Table 1.3-1 Comparison of Nuclear Steam Supply System Design Characteristics

Design[1]	This Plant ABWR 278-872[2]	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Thermal and Hydraulic (Section 4.4)				
Rated power (MWt)	3926	3579	3323	3833
Design power (MWt) (ECCS design basis)	4005	3729	3463	4025
Steam flow rate, Mlb/hr at 420°F (FW Temp)	16.843	15.40	14.263	16.491
Core coolant flow rate (Mlb/hr)	115.1	104.0	108.5	112.5
Feedwater flow rate (Mlb/hr)	16.807	15.372	14.564	16.455
System pressure, nominal in steam dome (psia)	1040	1040	1020	1040
Average power density (kW/l)	50.6	54.1	49.15	54.1
Maximum linear heat generation rate (kW/ft)	13.4	13.4	13.4	13.4
Average linear heat generation rate (kW/ft)	5.97	5.9	5.40	5.93
Maximum heat flux (Btu/hr/ft^2)	361,600	361,600	354,255	361,600
Average Heat flux (Btu/hr/ft^2)	161,100	159,500	144,032	160,300
Maximum UO_2 temperature (°F)	3365	3435	3325	3435
Average volumetric fuel temperature (°F)	2150	2185	2130	2185
Average cladding surface temperature (°F)	566	565	566	565
Minimum critical power ratio (MCPR)	1.17	1.20	1.24	1.20
Coolant enthalpy at core inlet (Btu/lb)	527.7	527.6	527.5	527.9
Core maximum voids within assemblies	75	79	76.2	76
Core average exit quality (% steam)	14.5	14.7	13.1	14.6
Feedwater temperature (°F)	420	420	420	420
Design power peaking factor				
Maximum relative assemble power	1.40	1.40	1.40	1.40
Local peaking factor	1.25	1.13	1.24	1.13
Axial peaking factor	1.40	1.40	1.40	1.40
Total peaking factor	2.43	2.26	2.43	2.26

Table 1.7-2 Instrument Engineering, Interlock Block and Single-Line Diagrams (Continued)

SSAR Fig. No.	Title	Type
7.7-9	Feedwater Control System	IBD
7.7-12	Steam Bypass and Pressure Control System	IED
7.7-13	Steam Bypass and Pressure Control System	IBD
7.7-14	Fuel Pool Cooling and Cleanup System	IBD
8.3-1	Electrical Power Distribution System	SLD
8.3-2	Instrument and Control Power Supply System	SLD
8.3-3	Plant Vital AC Power Supply System	SLD
8.3-4	Plant Vital DC Power Supply System	SLD

Table 1.7-3 Conversion to ASME Standard Units

	From	To convert to	Divide by
(1)	**Pressure/Stress**		
	kilopascal	1 Pound/Square Inch	6.894757
	kilopascal	1 Atmosphere (STD)	101.325
	kilopascal	1 Foot of Water (39.2°F)	2.98898
	kilopascal	1 Inch of Water (60°F)	0.24884
	kilopascal	1 Inch of HG (32°F)	3.38638
(2)	**Force/Weight**		
	newton	1 Pound - force	4.448222
	kilogram	1 Ton (Short)	907.1847
	kilogram	1 Tons (Long)	1016.047
(3)	**Heat/Energy/Power**		
	joule	1 Btu	1055.056
	joule	1 Calorie	4.1868
	kilowatt-hour	1 Btu	0.0002930711
	kilowatt	1 Horsepower(U.K.)	0.7456999
	kilowatt-hour	1 Horsepower-Hour	0.7456999
	kilowatt	1 Btu/Min	0.0175725
	joule/gram	1 Btu/Pound	2.326
(4)	**Length**		
	millimeter	1 Inch	25.4
	centimeter	1 Inch	2.54
	meter	1 Inch	0.0254
	meter	1 Foot	0.3048
	centimeter	1 Foot	30.48
	meter	1 Mile	1609.344
	kilometer	1 Mile	1.609344
(5)	**Volume**		
	liter	1 Cubic Inch	0.01638706
	cubic centimeter	1 Cubic Inch	16.38706
	cubic meter	1 Cubic Foot	0.02831685

Table 1.7-3 Conversion to ASME Standard Units (Continued)

	From	To convert to	Divide by
	cubic centimeter	1 Cubic Foot	28316.85
	liter	1 Cubic Foot	28.31685
	cubic meter	1 Cubic Yard	0.7645549
	liter	1 Gallon (US)	3.785412
	cubic centimeter	1 Gallon (US)	3785.412
	E-03 cubic centimeter	1 Gallon (US)	3.785412
(6)	**Volume Per Unit Time**		
	cubic centimeter/s	1 Cubic Foot/Min	471.9474
	cubic meter/h	1 Cubic Foot/Min	1.69901
	liter/s	1 Cubic Foot/Min	0.4719474
	cubic meter/s	1 Cubic Foot/Sec	0.02831685
	E-05 cubic meter/s	1 Gallon/Min (US)	6.30902
	cubic meter/h	1 Gallon/Min (US)	0.22712
	liter/s (101.325 kPaA,15.56°C)	1 STD CFM (14.696 psia, 60°F)	0.4474
	cubic meter/h (101.325 kPaA,15.56°C)	1 STD CFM (14.696 psia, 60°F)	1.608
(7)	**Velocity**		
	centimeter/s	1 Foot/Sec	30.48
	centimeter/s	1 Foot/Min	0.508
	meter/s	1 Foot/Min	0.00508
	meter/min	1 Foot/Min	0.3048
	centimeter/s	1 Inches/Sec	2.54
(8)	**Area**		
	square centimeter	1 Square Inch	6.4516
	E-04 square meter	1 Square Inch	6.4516
	square centimeter	1 Square Foot	929.0304
	E-02 square meter	1 Square Foot	9.290304
(9)	**Torque**		
	newton-meter	1 Foot Pound	1.355818
(10)	**Mass Per Unit Time**		
	kilogram/s	1 Pound/Sec	0.4535924

Table 1.7-3 Conversion to ASME Standard Units (Continued)

	From	To Convert to	Divide by
	kilogram/min	1 Pound/Min	0.4535924
	kilogram/h	1 Pound/Min	27.215544
(11)	**Mass Per Unit Volume**		
	kilogram/cubic meter	1 Pound/Cubic Inch	27679.90
	kilogram/cubic meter	1 Pound/Cubic Foot	16.01846
	kilogram/cubic centimeter	1 Pound/Cubic Inch	0.0276799
	liter/s	1 Gallon/Min	0.0630902
(12)	**Dynamic Viscosity**		
	Pa•s	1 Pound-Sec/Sq Ft	47.88026
(13)	**Specific Heat/Heat Transfer**		
	joule/kilogram kelvin	1 Btu/Pound-Deg F	4186.8
	watt/square meter kelvin	1 Btu/Hr-Sq Ft-Deg F	5.678263
	watt/square meter kelvin	1 Btu/Sec-Sq Ft-Deg F	2.044175E+4
	watt/square meter	1 Btu/Hr-Sq Ft	3.154591
(14)	**Temperature**		
	degree celsius	Degrees Fahrenheit	$T_{°F} = T_{°C}x1.8+32$
	degree C increment	1 Degree F Increment	0.555556
(15)	**Electricity**		
	coulomb	1 ampere hour	3600
	siemens/meter	1 mho/centimeter	100
(16)	**Light**		
	candels/square meter	1 candela/square inch	1550.003
	lux	1 footcandle	10.76391
(17)	**Radiation**		
	megabequerel	1 curie	37,000
	gray	1 rad	0.01
	sievert	1 rem	0.01

Note:

Rounding of Calculated values per Appendix C of ANSI/IEEE Std. 268.

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR

Code or Standard Number	Year	Title
American Concrete Institute (ACI)		
211.1†	1981	Practice for Selecting Proportions for Normal, Heavy Weight, and Mass Concrete.
212	1981	Guide for Admixtures in Concrete
214	1977	Recommended Practice for Evaluation of Strength Test Results of Concrete
301†	1984	Specifications for Structural Concrete for Buildings
304	1973	Practice for Measuring, Mixing, Transporting, and Placing of Concrete
305	1977	Recommended Practice for Hot Weather Concreting
306	1978	Recommended Practice for Cold Weather Concreting
307	1979	Specification for the Design and Construction of Reinforced Concrete Chimneys
308†	1981	Practice for Curing Concrete
309	1972	Practice for Consolidation of Concrete
311.1R	1981	ACI Manual of Concrete Inspection
311.4R	1981	Guide for Concrete Inspection
315†	1980	Details and Detailing of Concrete Reinforcement
318†	1983	Building Code Requirements for Reinforced Concrete
349†	1980	Code Requirements for Nuclear Safety-Related Concrete Structures
359		(See ASME BPVC Section III)
American Institute of Steel Construction (AISC)		
N690†	1984	Specifications for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities
--	--	Manual of Steel Construction
American Iron and Steel Institute		
SG-673	1986	Cold-Formed Steel Design Manual

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
American Nuclear Society (ANS)		
2.3†	1983	Standard for Estimating Tornado and Other Extreme Wind Characteristics at Nuclear Power Sites
2.8†	1981	Determining Design Basis Flooding at Power Reactor Sites
4.5†	1988	Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors
5.1†	1979	Decay Heat Power in LWRs
7-4.3.2†	1982	Application Criteria for Programmable Digital Computer Systems in Safety Systems of NPGS
18.1 (ANSI N237)	1984	Radioactive Source Term for Normal Operation of LWRs
52.1†	1983	Nuclear Safety Design Criteria for the Design of Stationary Boiling Water Reactor Plants
55.4	1979	Gaseous Radioactive Waste Processing Systems for Light Water Reactors
56.5	1979	PWR and BWR Containment Spray System Design Criteria
56.11†	1988	Standard Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants
57.1†(ANSI N208)	1980	Design Requirements for LWR Fuel Handling Systems
57.2(ANSI N210)	1976	Design Requirements for LWR Spent Fuel Storage Facilities at NPP
57.3	1983	Design Requirements for New Fuel Storage Facilities at LWR Plants
57.5†	1981	Light Water Reactor Fuel Assembly Mechanical Design and Evaluation
58.2†	1988	Design Basis for Protection of Light Water NPP Against Effects of Postulated Pipe Rupture
58.8†	1984	Time Response Design Criteria for Nuclear Safety Related Operator Actions
59.51 (ANSI N195)	1976	Fuel Oil Systems for Standby Diesel-Generators
American National Standards Institute (ANSI)‡		
A40	1993	National Plumbing Code
A58.1	1982	Design Loads for Buildings and other Structures, Minimum renumbered in 1988 as ANSI/ASCE 7-1988
AG-1		(See ASME AG-1)

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
519†	1981	IEEE Standard Recommended Practices and Requirements for Harmonic Control in Electrical Power Systems
603†	1980	IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations
622†	1987	Recommended Practice for the Design and Installation of Electric Heat Tracing Systems in Nuclear Power Generating Stations
622A†	1984	Recommended Practice for the Design and Installation of Electric Pipe Heating Control and Alarm Systems in Nuclear Power Generating Stations
	1984	Standard for Software Quality Assurance Plans
741†	1986	Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations
765†	1983	Standard for Preferred Power Supply for Nuclear Power Generating Stations
802.2†	1985	Standards for Local Area Networks: Logic Link Control
802.5†	1985	Token Ring Access Method and Physical Layer Specifications
828†	1983	Standard for Software Configuration Management Plans
829†	1983	Standard for Software Test Documentation
830†	1984	Standard for Software Requirements Specifications
845†	1988	Guide to Evaluation of Man-Machine Performance in Nuclear Power Generating Station Control Rooms and Other Peripheries
944†	1986	Recommended Practice for the Application and Testing of Uninterruptable Power Supplies for Power Generating Station
946†	1985	Recommended Practice for the Design of Safety-Related DC Auxiliary Power Systems for Nuclear Power Generating Stations
1012†	1986	Standard for Software Verification and Validation
1023†	1988	IEEE Guide to the Application of Human Factors Engineering to Systems, Equipment and Facilities of Nuclear Power Generating Stations
1033†	1985	Recommended Practice of Application of IEEE-828 to Nuclear Power Generation Stations
	1987	Guide to Software Configuration Management
Instrument Society of America (ISA)		
S7.3†	1981	Quality Standard for Instrument Air

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
S67.02-80	1980	Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants
National Electrical Manufacturers Association (NEMA)		
AB 1	1986	Molded Case Circuit Breakers
FB 1	1977	Fittings and Support for Conduit and Cable Assemblies
ICS 1†	1983	General Standards for Industrial Control
ICS 2†	1988	Standards for Industrial Control Devices, Controllers and Assemblies
MG 1	1987	Motors and Generators
WC-5		(See ICEA S-61-402)
WC 7		(See ICEA S-66-524)
WC 51		(See ICEA P-54-440)
National Fire Protection Association (NFPA)		
10†	1981	Portable Fire Extinguishers - Installation
10A	1973	Portable Fire Extinguishers - Maintenance and Use
11†	1988	Low Expansion Foam and Combined Agent Systems-Foam Extinguishing System
12†	1985	Carbon Dioxide Extinguishing Systems
13†	1985	Installation of Sprinklers Systems
14†	1986	Installation of Standpipe and Hose Systems
15†	1985	Standard for Water Spray Fixed Systems
16†	1991	Deluge Foam-Water Sprinkler and Foam-Water Spray Systems
16A†	1988	Recommended Practice for the Installation of Closed Head Foam-Water Sprinkler Systems
20†	1990	Standard for the Installation of Centrifugal Fire Pumps
24†	1984	Private Service Mains and their Appurtenances
26†	1988	Recommended Practice for the Supervision of Valves Controlling Water Supplies for Fire Protection
37†	1984	Stationary Combustion Engines and Gas Turbines
70†	1987	National Electrical Code-Handbook 1987
72†	1990	Protective Signaling Systems

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
UL-845	1988	Standard for Safety Motor Control Centers - Low Voltage Circuit Breakers
--	--	Crane Manufacturers Association of America, Specification No. 70
--	--	Aluminum Construction Manual by Aluminum Association
NCIG-01	Rev. 2	Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants
UBC	1991	Uniform Building Code

* The listing of a code or standard does not necessarily mean that it is applicable in its entirety.

† Also an ANSI code (i.e. ANSI/ASME, ANSI/ANS, ANSI/IEEE etc.).

‡ ANSI, ANSI/ANS, ANSI/ASME, and ANSI/IEEE codes are included here. Other codes that approved by ANSI and another organization are listed under the latter.

ƒ As modified by NRC accepted alternate positions to the related Regulatory Guide and identified in Table 2-1 of Reference 1 to Chapter 17.

Table 1.8-22 Experience Information Applicable to ABWR

No.	Issue Date	Title	Comment
Type: Generic Letters			
80-06	4/25/80	Clarification of NRC Requirement for Emergency Response Facilities at Each Site	
80-30	12/15/80	Periodic Updating of Final Safety Analysis Reports (FSARs)	COL Applicant
80-31	12/22/80	Control of Heavy Loads	
81-03	2/26/81	Implementation of NUREG-0313m, Rev. 1	
81-04	2/25/81	Emergency Procedures and Training for Station Blackout Events	COL Applicant
81-07	2/3/81	Control of Heavy Loads	
81-10	2/18/81	Post-TMI Requirements for the Emergency Operations Facility	
81-11	2/22/81	Error in NUREG-0619	
81-20	4/1/81	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	
81-37	12/29/81	ODYN Code Reanalysis Requirements	
81-38	11/10/81	Storage of Low-Level Radioactive Wastes at Power Reactor Sites	COL Applicant
82-09	4/20/82	Environmental Qualification of Safety-Related Electrical Equipment	
82-21	10/6/82	Technical Specifications for Fire Protection Audits	COL Applicant
82-22	10/30/82	Inconsistency Between Requirements of 10CFR73.40(d) and Standard Technical Specifications for Performing Audits of Safeguard Contingency Plans	
82-27	11/15/82	Transmittal of NUREG-0763, "Guidelines for Confirmatory In-Plant Tests of Safety-Relief Valve Discharges for BWR Plants," and NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."	
82-33	12/17/82	Supplement 1 to NUREG-0737	
82-39	12/22/82	Problems with the Submittals of 10CFR73.21 Safeguards Information Licensing Review	COL Applicant
83-05	2/83	Safety Evaluation of "Emergency Procedure Guidelines," Revision 2, NEDO-24934, June 1982	COL Applicant
83-07	2/16/83	The Nuclear Waste Policy Act of 1982	COL Applicant

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

Item No.	Subject	Subsection
8.7	Deleted	8.3.4.1
8.8	Diesel Generator Design Details	8.3.4.2
8.9	Deleted	8.3.4.3
8.10	Protective Devices for Electrical Penetration Assemblies	8.3.4.4
8.11	Deleted	8.3.4.5
8.12	Deleted	8.3.4.6
8.13	Deleted	8.3.4.7
8.14	Deleted	8.3.4.8
8.15	Offsite Power Supply Arrangements	8.3.4.9
8.16	Deleted	8.3.4.10
8.17	Deleted	8.3.4.11
8.18	Deleted	8.3.4.12
8.19	Load Testing of Class 1E Switchgear and Motor Control Centers	8.3.4.13
8.20	Administrative Controls for Bus Grounding Circuit Breakers	8.3.4.14
8.21	Administrative Controls for Manual Interconnections	8.3.4.15
8.22	Deleted	8.3.4.16
8.23	Common Industrial Standards Referenced in Purchase Specifications	8.3.4.17
8.24	Administrative Control for Switching 125Vdc Standby Charger	8.3.4.18
8.25	Control of Access to Class 1E Power Equipment	8.3.4.19
8.26	Periodic Testing of Voltage Protection Equipment	8.3.4.20
8.27	Diesel Generator Parallel Test Mode	8.3.4.21
8.28	Periodic Testing of Diesel Generator Protective Relaying	8.3.4.22
8.29	Periodic Testing of Diesel Generator Synchronizing Interlocks	8.3.4.23
8.30	Periodic Testing of Thermal Overloads and Bypass Circuitry	8.3.4.24
8.31	Periodic Inspection/Testing of Lighting System	8.3.4.25
8.32	Controls for Limiting Potential Hazards Into Cable Chases	8.3.4.26
8.33	Periodic Testing of Class 1E Equipment Protective Relaying	8.3.4.27

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

Item No.	Subject	Subsection
8.34	Periodic Testing of CVCF Power Supplies and EPAs	8.3.4.28
8.35	Periodic Testing of Class 1E Circuit Breakers	8.3.4.29
8.36	Periodic Testing of Electrical Systems & Equipment	8.3.4.30
8.37	Deleted	8.3.4.31
8.38	Class 1E Battery Installation and Maintenance Requirements	8.3.4.32
8.39	Periodic Testing of Class 1E Batteries	8.3.4.33
8.40	Periodic Testing of Class 1E CVCF Power Supplies	8.3.4.34
8.41	Periodic Testing of Class 1E Battery Chargers	8.3.4.35
8.42	Periodic Testing of Class 1E Diesel Generators	8.3.4.36
9.1	New Fuel Storage Racks Criticality Analysis	9.1.6.1
9.2	Dynamic and Impact Analysis of New Fuel Storage Racks	9.1.6.2
9.3	Spent Fuel Storage Racks Criticality Analysis	9.1.6.3
9.4	Spent Fuel Rack Load Drop Analysis	9.1.6.4
9.5	New Fuel Inspection Stand Seismic Capability	9.1.6.5
9.6	Overhead Load Handling System Information	9.1.6.6
9.7	Spent Fuel Racks Structural Evaluation	9.1.6.7
9.8	Spent Fuel Racks Thermal-Hydraulic Analysis	9.1.6.8
9.9	Spent Fuel Firewater Makeup Procedures and Training	9.1.6.9
9.10	Protection of RHR System Connections to FPC System	9.1.6.10
9.11	HECW System Refrigerator Requirements	9.2.17.1
9.12	Reactor Service Water System Requirements	9.2.17.2
9.12a	Deleted	9.3.12.1
9.13	Deleted	9.3.12.2
9.14	Deleted	9.3.12.3
9.15	Radioactive Drain Transfer System	9.3.12.4
9.16	Service Building HVAC System	9.4.10.1
9.17	Radwaste Building HVAC System	9.4.10.2
9.18	Contamination of DG Combustion Air Intake	9.5.13.1

1A.2.33.1 Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure [II.K.3 (44)]

NRC Position

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which occur in a stuck-open relief valve should be included in this category. The results of the evaluation are due January 1, 1981.

Response

GE and the BWROG have concluded, based on a representative BWR/6 plant study, that all anticipated transients in Regulatory Guide 1.70, Revision 3, combined with the worst single failure, the reactor core remains covered with water until stable conditions are achieved. Furthermore, even with more degraded conditions involving a stuck-open relief valve in addition to the worst transient (loss of feedwater) and worst single failure (of high pressure core spray), studies show that the core remains covered and adequate core cooling is available during the whole course of the transient (NEDO-24708, March 31, 1980). The conclusion is applicable to the ABWR. Since the ABWR has more high pressure makeup systems (2HPCFs and 1 RCIC), the core covering is further assured.

Other discussions of transients with single failure is presented in the response to NRC Question 440.111.

1A.2.33.2 Evaluate Depressurization Other Than Full ADS [II.K.3 (45)]

NRC Position

Provide an evaluation of depressurization methods, other than by full actuation of the Automatic Depressurization System, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown (Applicable to BWRs only).

Response

This response is provided in Subsection 19A.2.11.

1A.2.33.3 Responding to Michelson Concerns [II.K.3 (46)]

NRC Position

General Electric should provide a response to the Michelson concerns as they relate to boiling water reactors.

Clarification: General Electric provided a response to the Michelson concerns as they relate to boiling water reactors by letter dated February 21, 1980. Licensees and applicants should assess applicability and adequacy of this response to their plants.

Response

All of the generic February 21, 1980 GE responses were reviewed and updated for the ABWR Standard Plant. The specific responses are provided in Table 1A-1.

1A.2.34 Primary Coolant Sources Outside Containment Structure [III.D.1.1(1)]

NRC Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate Leak Reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment

(b) Measure actual leakage rates with systems in operation and report them to the NRC

(2) Continuing Leak Reduction—establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Response

Leak reduction measures of the ABWR Standard Plant include a number of barriers to containment leakage in the closed systems outside the containment. These closed systems include:

(1) Residual Heat Removal

(2) High Pressure Core Flooder

(3) Low Pressure Core Flooder

(4) Reactor Core Isolation Cooling

(5) Suppression Pool Cleanup

(6) Reactor Water Cleanup

(7) Fuel Pool Cooling and Cleanup

(8) Post-Accident Sampling

(9) Process Sampling

(10) Containment Atmospheric Monitoring

(11) Fission Product Monitor (Part of LDS)

(12) Hydrogen Recombiner

(13) Standby Gas Treatment

3.3 Wind and Tornado Loadings

ABWR Standard Plant structures which are Seismic Category I are designed for tornado and extreme wind phenomena.

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

Seismic Category I structures are designed to withstand a design wind velocity of 177 km/h with a recurrence interval of 50 years and 197 km/h with a recurrence interval of 100 years at an elevation of 10m above grade (see Subsection 3.3.3.1 and 3.3.3.3 for COL license information requirements).

3.3.1.2 Determination of Applied Forces

The design wind velocity is converted to velocity pressure in accordance with Reference 3.3-1 using the formula:

$$q_z = 4.94 \times 10^{-5} K_z (IV)^2$$

where K_z = The velocity pressure exposure coefficient which depends upon the type of exposure and height (z) above ground per Table 6 of Reference 3.3-1

I = The importance factor which depends on the type of structure; appropriate values of I are listed in Table 3.3-1

V = Design wind velocity with a recurrence interval of 50 years, in km/h, and

q_z = Velocity pressure in kPa

The design wind pressures and forces for buildings, components and cladding, and other structures at various heights above the ground are obtained, in accordance with Table 4 of Reference 3.3-1 by multiplying the velocity pressure by the appropriate pressure coefficients and gust factors. Gust factors are in accordance with Table 8 of Reference 3.3-1. Appropriate pressure coefficients are in accordance with Figures 2, 3a, 3b, 4, and Tables 9 and 11 through 16 of Reference 3.3-1. Reference 3.3-2 is used to obtain the effective wind pressures for cases which Reference 3.3-1 does not cover. Since the Seismic Category I structures are not slender or flexible, vortex-shedding analysis is not required and the above wind loading is applied as a static load.

Applied forces for the Reactor, Control and Radwaste Buildings are found in Appendices 3H.1, 3H.2 and 3H.3, respectively.

3.3.2 Tornado Loadings

3.3.2.1 Applicable Design Parameters

The design basis tornado is described by the following parameters:

(1) A maximum tornado wind speed of 483 km/h at a radius of 45.7m from the center of the tornado.

(2) A maximum translational velocity of 97 km/h.

(3) A maximum tangential velocity of 386 km/h, based on the translational velocity of 97 km/h.

(4) A maximum atmospheric pressure drop of 13.8 kPa with a rate of the pressure change of 8.3 kPa/s.

(5) The spectrum of tornado-generated missiles and their pertinent characteristics as given in Table 2.0-1.

See Subsection 3.3.3.2 for COL license information.

3.3.2.2 Determination of Forces on Structures

The procedures of transforming the tornado loading into effective loads and the distribution across the structures are in accordance with Reference 3.3-3. The procedure for transforming the tornado-generated missile impact into an effective or equivalent static load on structures is given in Subsection 3.5.3.1. The loading combinations of the individual tornado loading components and the load factors are in accordance with Reference 3.3-3.

The reactor building and control building are not vented structures. The exposed exterior roofs and walls of these structures are designed for the 13.8 kPa pressure drop. Tornado dampers are provided on all air intake and exhaust openings. These dampers are designed to withstand a negative13.8 kPa pressure.

3.3.2.3 Effect of Failure of Structures, Systems or Components Not Designed for Tornado Loads

All safety-related systems and components are protected within tornado-resistant structures.

See Subsection 3.3.3.4 for COL license information requirements.

3.3.3 COL License Information

3.3.3.1 Site-Specific Design Basis Wind

The site-specific design basis wind shall not exceed the design basis wind given in Table 2.0-1 (Subsection 2.2.1).

3.3.3.2 Site-Specific Design Basis Tornado

The site-specific design basis tornado shall not exceed the design basis tornado given in Table 2.0-1 (Subsection 2.2.1).

3.3.3.3 Effect of Remainder of Plant Structures, Systems and Components Not Designed for Wind Loads

All remainder of plant structures, systems and components not designed for wind loads shall be analyzed using the 1.11 importance factor or shall be checked that their mode of failure will not effect the ability of safety-related structures, systems or components performing their intended safety functions.

3.3.3.4 Effect of Remainder of Plant Structures, Systems, and Components not Designed for Tornado Loads

All remainder of plant structures, systems, and components not designed for tornado loads shall be analyzed for the site-specific loadings to ensure that their mode of failure will not effect the ability of the Seismic Category I ABWR Standard Plant structures, systems, and components to perform their intended safety functions. (See Subsection 3.3.2.3)

3.3.4 References

3.3-1 ANSI/ASCE 7-88, *Minimum Design Loads for Buildings and Other Structures,* November 27, 1990.

3.3-2 ASCE Paper No. 3269, *Wind Forces on Structures,* Transactions of the American Society of Civil Engineers, Vol. 126, Part II, 1961.

3.3-3 Bechtel Topical Report BC-TOP-3-A, Revision 3, *Tornado and Extreme Wind Design Criteria for Nuclear Power Plants.*

Table 3.3-1 Importance Factor (I) for Wind Loads

Non-Safety-Related	Safety-Related
1.00	1.11

Notes:

(1) These values of (I) are based on Table 5 of Reference 3.3-1.

3.5.4.2 Missiles Generated by Other Natural Phenomena

The COL applicant shall identify missiles generated by other site-specific natural phenomena that may be more limiting than those considered in the ABWR design and shall provide protection for the structures, systems, and components against such missiles. The COL applicant will provide this information to the NRC (Subsection 3.5.1.4).

3.5.4.3 Site Proximity Missiles and Aircraft Hazards

Analyses shall be provided that demonstrate that the probability of site proximity missiles (including aircraft) impacting the ABWR Standard Plant and causing consequences greater than 10CFR100 exposure guidelines is $\leq 10^{-7}$ per year (Subsection 3.5.1.6).

3.5.4.4 Impact of Failure of Out of ABWR Standard Plant Scope Non-Safety-Related Structures, Systems, and Components Due to a Design Basis Tornado

An evaluation of all out of ABWR Standard Plant Scope non-safety-related structures, systems, and components (not housed in a tornado structure) whose failure due to a design basis tornado missile that could adversely impact the safety function of safety-related systems and components will be provided to the NRC by the COL applicant (Subsection 3.5.1.4).

3.5.4.5 Turbine System Maintenance Program

A turbine system maintenance program, including probability calculations of turbine missile generation meeting the minimum requirement for the probability of missile generation, shall be provided to the NRC (Subsection 3.5.1.1.1.3).

3.5.4.6 Maintenance Equipment Missile Prevention Inside Containment

The COL applicant will provide procedures to ensure that all equipment inside containment, such as hoists, that is required during maintenance will either be removed prior to operation, moved to a location where it is not a potential hazard to safety-related equipment, or seismically restrained to prevent it from becoming a missile [Subsection 3.5.1.2.3 (3)].

3.5.4.7 Failure of Structures, Systems, and Components Outside ABWR Standard Plant Scope

Any failure of structures, systems and components outside ABWR Standard Plant scope which may result in external missile generation shall not prevent safety-related structures, systems and components from performing their intended safety function. The COL applicant will provide an evaluation of the adequacy of these designs for external missile protection for NRC review (Subsection 3.5.2).

3.5.5 References

3.5-1 K. Karim-Panahi et. al, *Recirculation MG Set Missile Generation Study*, PED-18-0389, March 1989. (Proprietary).

3.5-2 F. J. Moody, *Prediction of Blowdown Thrust and Jet Forces*, ASME Publication 69-HT-31, August 1969.

3.5-3 A. Amirikan, *Design of Protective Structures*, Bureau of Yards and Docks, Publication No. NAVDOCKS P-51, Department of the Navy, Washington, D.C., August 1960.

3.5-4 US Department of Army, Fundamentals of Protective Design for Conventional Weapons, TM 5-855-1, November 1986.

3.5-5 W. B. Cottrell and A. W. Savolainen, *U. S. Reactor Containment Technology*, ORNL- NSIC-5, Vol. 1, Chapter 6, Oak Ridge National Laboratory.

3.5-6 R. A. Williamson and R. R. Alvy, *Impact Effect of Fragments Striking Structural Elements*, Holmes and Narver, Inc., Revised November 1973.

3.5-7 J. D. Riera, *On the Stress Analysis of Structures Subjected to Aircraft Impact Forces*, Nuclear Engineering and Design, North Holland Publishing Co., Vol. 8, 1968.

3.5-8 *River Bend Station Updated Safety Analysis Report*, Docket No. 50-458, Volume 6, pp. 3.5-4 and 3.5-5, August 1987.

3.5-9 NUREG-1048, *Safety Evaluation Report Related to the Operation of Hope Creek Generating Station*, Supplement No. 6, July 1986.

(b) Regulatory Guide 1.15 - *Testing of Reinforcing Bars for Category I Concrete Structures*

(c) Regulatory Guide 1.28 - *Quality Assurance Program Requirements* (Design and Construction)

(d) Regulatory Guide 1.29 - *Seismic Design Classification*

(e) Regulatory Guide 1.31 - *Control of Stainless Steel Welding*

(f) Regulatory Guide 1.44 - *Control of the Use of Sensitized Stainless Steel*

(g) Regulatory Guide 1.55 - *Concrete Placement in Category I Structures*

(h) Regulatory Guide 1.60 - *Design Response Spectra for Seismic Design of Nuclear Power Plants*

(i) Regulatory Guide 1.61 - *Quality Assurance Requirements for the Design of Nuclear Power Plants*

(j) Regulatory Guide 1.69 - *Concrete Radiation-Shields for Nuclear Power Plants*

(k) Regulatory Guide 1.76 - *Design Basis Tornado*

(l) Regulatory Guide 1.142 - *Safety-Related Concrete Structures for Nuclear Power Plants* (Other than Reactor Vessels and Containment)

(m) Regulatory Guide 1.94 - *Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants*

(9) ANSI:

(a) ANSI/ASCE 7-88 - *Minimum Design Loads for Buildings and Other Structures*

(b) ANSI N5.12 - *Protective Coatings (Paint) for the Nuclear Industry*

(c) NQA-1 - *Quality Assurance Program Requirements for Nuclear Facilities and NQA-1A, Addenda to ANSI/ASME NQA-1*

(d) Deleted

(e) Deleted

(f) ANSI N45.4 - *Leakage-Rate Testing of Containment Structures for Nuclear Reactors*

(g) ANSI N101.2 - *Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities*

(h) ANSI N101.4 - *Quality Assurance for Protective Coatings Applied to Nuclear Facilities*

(10) Steel Structures Painting Council Standards

(a) SSPC-PA-1 - *Shop, Field and Maintenance Painting*

(b) SSPC-PA-2 - *Measurement of Paint Film Thickness with Magnetic Gages*

(c) SSPC-SP-1 - *Solvent Cleaning*

(d) SSPC-SP-5 - *White Metal Blast Cleaning*

(e) SSPC-SP-6 - *Commercial Blast Cleaning*

(f) SSPC-SP-10 - *Near-White Blast Cleaning*

(11) ACI-ASCE Committee 326 - *Shear and Diagonal Tension, ACI Manual of Concrete Practice, Part 2.*

(12) Applicable ASTM Specifications for Materials and Standards.

(13) *AASHTO Standard Specifications for Highway Bridges for truck loading area.*

3.8.4.2.2 Control Building

Refer to Subsection 3.8.4.2.1.

Add NRC Rules and Regulations Title 10, Chapter 1, Code of Federal Regulations, Part 73.2 and 73.55.

3.8.4.2.3 Radwaste Building Substructure

The RWB Substructure shall be designed using the same codes and standards as the reactor building. Refer to Subsection 3.8.4.2.1 for a complete list.

In addition, the non-Seismic Category I reinforced concrete portion of the superstructure is designed according to the seismic provisions of the uniform building code.

3.8.4.2.4 Seismic Category I Cable Tray, Cable Tray Supports, Conduit and Conduit Supports

(1) ANSI/AISC-N690, *Specification for Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facility.*

(2) AISI SG-673 - *Cold Formed Steel Design Manual.*

(3) *ANSI/NEMA FB1, Fittings and Supports for Conduit and Cable Assemblies.*

3.8.4.2.5 Seismic Category I HVAC Ducts and Supports

(1) ASME/ANSI AG-1, *Code on Nuclear Air and Gas Treatment.*

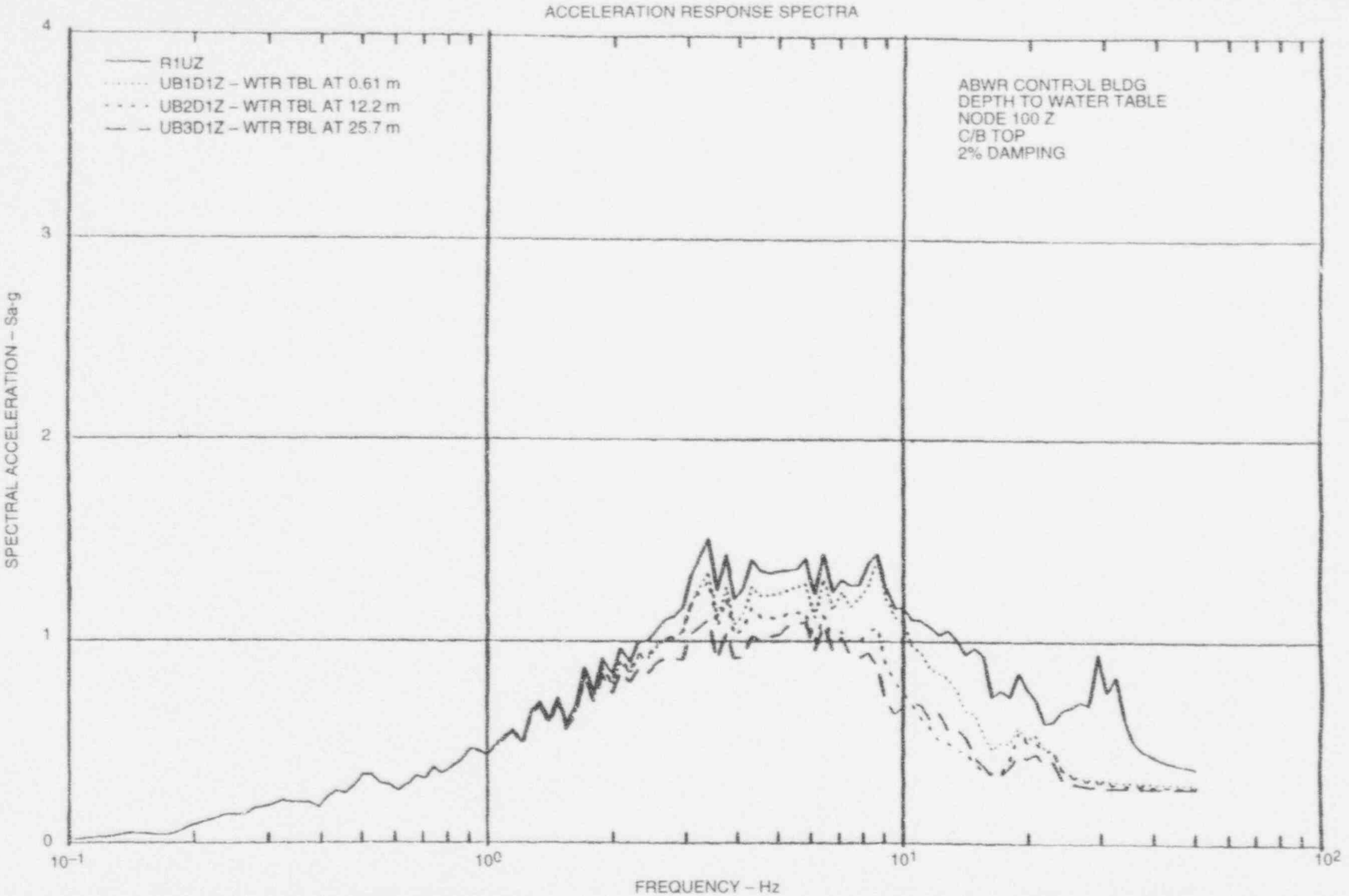

Figure 3A-83 ABWR Control Bldg. Depth to Water Table, Node 108 Z C/B Top, 2% Damping

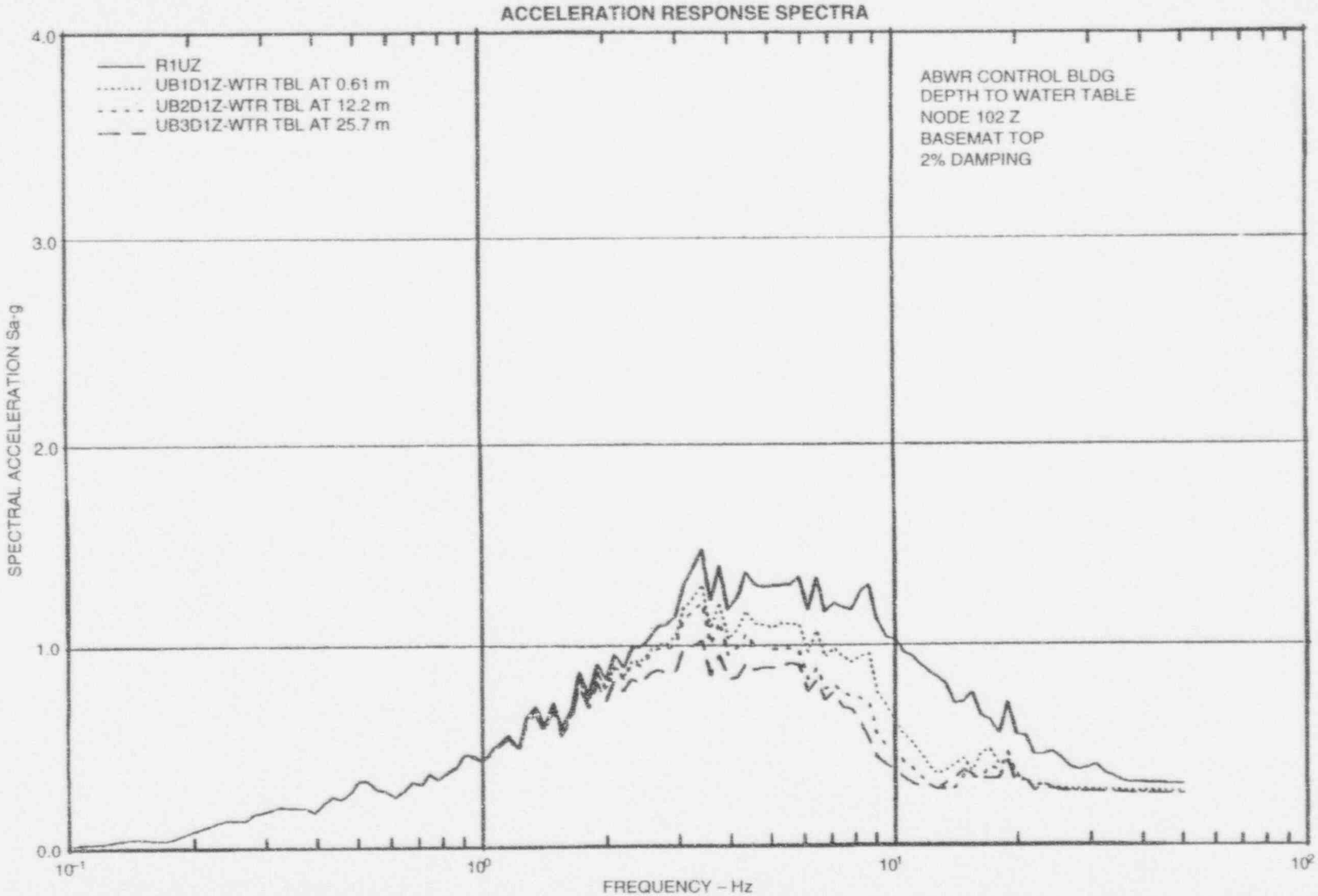

Figure 3A-84 ABWR Control Bldg. Depth to Water Table, Node 102 Z Basemat Top, 2% Damping

Table 3H.1-4 Maximum Vertical Acceleration

Reactor Building Outside RCCV		RCCV		Reactor Shield Wall & RPV Pedestal	
Node No.	Max. Vert. Accln. (g)	Node No.	Max. Vert. Accln. (g)	Node No.	Max. Vert. Accln. (g)
95	0.63	89	0.72	70	0.43
96	0.52	90	0.67	78	0.43
98	0.47	91	0.64	79	0.42
100	0.43	92	0.58	80	0.41
102	0.39	93	0.43	81	0.40
103	0.34	94	0.34	82	0.39
104	0.32	88	0.31	71	0.38
105	0.31	106	0.31	83	0.37
88	0.31			84	0.34
106	0.31			73	0.34
107	1.22			85	0.33
108	1.56			86	0.32
109	1.88			87	0.31
110	1.04				
111	0.54				
112	0.47				

Note: See Figure 3A-8 for Locations and Node Numbers.

Table 3H.1-5 Load Combinations, Load Factors and Acceptance Criteria for the Reinforced Concrete Containment

Notes:

1. For any load combination, if the effect of any load component, other than D, reduces the combined load, then the load component is deleted from the load combination.
2. Since P_a, P_i, P_s, T_a, SRV and LOCA are time dependent loads, their effects will be superimposed accordingly.
3. S = Allowable stress as in ASME Section III, Division 2, Subarticle CC-3430 for Service Load Combinations.

   U = Allowable stress as in ASME Sections III, Division 2, Subarticle CC-3420 for Factored Load Combinations.

Table 3H.1-5a Selected Load Combinations for the RCCV

No.	Event	Load Combination	Acceptance Criteria
1	SIT (1)	$D + L + P_t$ (1)	S
1*	SIT(2)	$D + L + P_t$ (2)	S
8	LBL (30 min)	$D + L + 1.5 (P_a + CO) + T_a$	U
8a,8b	IBL/SBL (6 h)	$D + L + 1.5 (P_a + CO) + 1.25\ SRV + T_a$	U
15	LBL (30 min) + SSE	$D + L + P_a + CO + T_a + SSE + R_a + Y$	U
15a, 15b	IBL/SBL (6 h) + SSE	$D + L + P_a + CO + SRV + T_a + SSE$	U

P_t = Test Pressure During SIT; P_t (1) indicates test pressures of 358 kPaG in the drywell and wetwell; and P_t (2) indicates test pressures of 310 kPaG and 138 kPaG in the drywell and wetwell, respectively.

P_a, T_a, R_a = Containment Pressure, Temperature and Pipe Support reaction loads associated with the LOCA.

Y = Local effects on the containment due to DBA. These include Y_r (restraint reaction), Y_J (jet impingement) and Y_m (missile impact). These are to be considered only for those events which are not eliminated through application of Leak-Before-Break (LBB).

S = Allowable stress as in ASME Section III, Division 2, Subarticle CC-3430 for Service Load Combinations.

U = Allowable stress as in ASME Section III, Division 2, Subarticle CC-3420 for Factored Load Combinations.

The calc[illegible] values of k_{eff} with the strongest rod withdrawn at BOC and of R are reporte[illegible] [illegible]e 4.3-2. For completeness, the uncontrolled k_{eff} and fully controlled k_{eff} values are also reported in Table 4.3-2.

4.3.2.4.2 Reactivity Variations

The excess reactivity designed into the core is controlled by the control rod system supplemented by gadolinia-urania fuel rods. Control rods are used during the cycle partly to compensate for burnup and partly to control the power distribution.

4.3.2.4.3 Standby Liquid Control System

The Standby Liquid Control System (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, from a full power and minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state. The requirements of this system are dependent primarily on the reactor power level and on the reactivity effects of voids and temperature between full-power and cold, xenon-free condition. The SLCS is discussed in Subsection 9.3.5.

4.3.2.5 Criticality of Reactor During Refueling

The core is subcritical at all times.

4.3.2.6 Stability

4.3.2.6.1 Xenon Transients

Boiling water reactors do not have instability problems due to xenon. This has been demonstrated by:

(1) Never having observed xenon instabilities in operating BWRs

(2) Special tests which have been conducted on operating BWRs in an attempt to force the reactor into xenon instability

(3) Calculations

All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient.

Analysis and experiments conducted in this area are reported in Reference 4.3-2.

4.3.2.6.2 Thermal Hydraulic Stability

The compliance of GE fuel designs to the criteria set forth in General Design Criterion 12 is demonstrated provided that the following stability compliance criteria are satisfied using approved methods:

(1) Neutron flux limit cycles, which oscillate up to 120% APRM high neutron flux scram setpoint or up to the LPRM upscale alarm trip (without initiating scram) prior to operator mitigating action shall not result in exceeding specified acceptable fuel design limits.

(2) The individual channels shall be designed and operated to be hydrodynamically stable or more stable than the reactor core for all expected operating conditions (analytically demonstrated).

The GE methodology for demonstrating the above has been reviewed and approved by the NRC in Reference 4.3-4. The stability compliance of the fuel designs described in Subsection 4.2.2.1 has been demonstrated on a generic basis and approved by the NRC in Reference 4.3-4. Should a different fuel design be chosen by the COL applicant, the methodology for demonstrating stability will be that approved by the NRC. See Subsection 4.3.5.1 for COL license information. See Subsection 4.4.3.7 for specific thermal hydraulic stability performance information.

4.3.3 Analytical Methods

The nuclear evaluations of all General Electric cores are performed using the analytical tools and methods described in Reference 4.3-2.

The lattice analyses are performed during the bundle design process. The results of these single bundle calculations are reduced to "libraries" of lattice reactivities, relative rod powers, and few group cross-sections as functions of instantaneous void, exposure, exposure-void history, control state, and fuel and moderator temperature, for use in the core analysis. These analyses are dependent upon fuel lattice parameters only and are, therefore, valid for all plants and cycles to which they are applied.

The core analysis is unique for each cycle. It is performed in the months preceding the cycle loading to demonstrate that the core meets all applicable safety limits. The principal tool used in the core analysis is the three-dimensional BWR simulator code, which computes power distributions, exposure, and reactor thermal-hydraulic characteristics, with spatially varying voids, control rods, burnable poisons and other variables.

4.3.4 Changes

Not applicable.

4.6.1.2.2.5 Scram Position Indication

Scram position indication is provided by a series of magnetic reed switches to allow for measurement of adequate drive performance during scram. The magnetic switches are located at intermediate intervals over 60% of the drive stroke. They are mounted in a probe exterior to the drive housing. A magnet in the hollow piston trips each reed switch in turn as it passes by.

As the bottom of the hollow piston contacts and enters the buffer, a magnet is lifted which operates a reed switch, indicating scram completion. This continuous full-in indicating switch is shown conceptually in Figure 4.6-3. It provides indication whenever the drive is at the full-in latched position or above.

4.6.1.2.2.6 Control Rod Separation Detection

Two redundant and separate Class 1E switches are provided to detect the separation of the hollow piston from the ball-nut. This means two sets of reed switches physically separated from one another with their cabling run through separate conduits. The separation switch is classified Class 1E, because its function detects a detached control rod and causes a rod block, thereby preventing a rod drop accident. Actuation of either switch also initiates an alarm in the control room

The principle of operation of the control rod separation mechanism is illustrated in Figure 4.6-4. During normal operation, the weight of the control rod and hollow piston resting on the ball-nut causes the spindle assembly to compress a spring on which the lower half of the splined coupling between the drive shaft and spindle assembly rests (the lower half of the splined coupling is also known as the "weighing table"). When the hollow piston separates from the ball-nut, or when the control rod separates from the hollow piston, the spring is unloaded and pushes the weighing table and spindle assembly upward. This action causes a magnet in the weighing table to operate the Class 1E reed switches located in a probe outside the lower housing.

4.6.1.2.2.7 Bayonet Couplings

There are two bayonet couplings associated with the FMCRD. The first is at the FMCRD/control rod guide tube/housing interface as illustrated in Figure 4.6-7. This bayonet locks the FMCRD and the base of the control rod guide tube to the CRD housing and functions to retain the control rod guide tube during normal operation and dynamic loading events. The bayonet also holds the FMCRD against ejection in the event of a hypothetical failure of the CRD housing weld. The locating pin on the core plate that engages the flange of the control rod guide tube and the bolt pattern on the FMCRD/housing flange assure proper orientation between the control rod guide tube and FMCRD to assure that the bayonet is properly engaged.

The second bayonet coupling is located between the control rod and FMCRD, as shown on Figure 4.6-5. The coupling spud at the top end of the FMCRD hollow piston engages and locks into a mating socket at the base of the control rod. The coupling requires a 45° rotation for engaging or disengaging. Once locked, the drive and rod form an integral unit that can only be unlocked manually by specific procedures before the components can be separated.

The FMCRD design allows the coupling integrity of this second bayonet to be checked by driving the ball-nut down into an overtravel-out position. After the weighing spring has raised the spindle to the limit of its travel, further rotation of the spindle in the withdraw direction will drive the ball-nut down away from the hollow piston (assuming the coupling is engaged). Piston movement, if any, can then be detected by a reed switch at the overtravel position. If the hollow piston and control rod are properly coupled the overtravel reed switch will not be activated, thus confirming the coupling integrity. If the hollow piston is uncoupled from the control rod the piston will follow the ball-nut to the overtravel position. The overtravel reed switch will be then be actuated by a magnet in the hollow piston, thereby indicating an uncoupled condition.

4.6.1.2.2.8 FMCRD Brake and Ball Check Valve

The FMCRD design incorporates an electromechanical brake (Figure 4.6-6) keyed to the motor shaft. The brake is normally engaged by spring force when the FMCRD is stationary. It is disengaged for normal rod movements by signals from the RCIS. Disengagement is caused by the energized magnetic force overcoming the spring load force. The braking torque of 49 N·m (minimum)between the motor shaft and the CRD spool piece is sufficient to prevent control rod ejection in the event of failure in the pressure-retaining parts of the drive mechanism. The brake is designed so that its failure will not prevent the control rod from rapid insertion (scram).

The electromechanical brake is located between the stepping motor and the synchronizing signal generators. The stationary spring-loaded disk and coil assembly are contained within the brake mounting bolted to the bottom of the stepping motor. The rotating disk is keyed to the stepping motor shaft and synchro shaft.

The brake is classified as passive safety-related because it performs its holding function when it is in its normally de-energized condition.

A ball check valve is located in the middle flange of the drive at the scram inlet port. The check valve is classified as safety-related because it actuates to close the scram inlet port under conditions of reverse flow caused by a break of the scram line. This prevents the loss of pressure to the underside of the hollow piston and the generation of loads on the drive that could cause a rod ejection.

(6) The main steam and feedwater piping and smaller connected lines are designed in accordance with the requirements of Table 3.2-1.

5.4.9.2 Power Generation Design Bases

(1) The main steamlines are designed to conduct steam from the reactor vessel over the full range of reactor power operation.

(2) The feedwater lines are designed to conduct water to the reactor vessel over the full range of reactor power operation.

5.4.9.3 Description

The main steam piping is described in Section 10.3. The main steam and feedwater piping from the reactor through the containment isolation interfaces is diagrammed in Figure 5.1-3.

As discussed in Table 3.2-1 and shown in Figure 5.1-3, the main steamlines are Quality Group A from the reactor vessel out to and including the outboard MSIV and Quality Group B from the outboard MSIVs to the turbine stop valve. They are also Seismic Category I only from the reactor pressure vessel out to the seismic interface restraint.

The feedwater piping consists of two 550A diameter lines from the feedwater supply header to the reactor. On each of the feedwater lines from the common feedwater supply header, there shall be a seismic interface restraint. The seismic interface restraint shall serve as the boundary between the Seismic Category I piping and the non-seismic piping. Downstream of the seismic restraint, there is a remote manual, motor-operated valve powered by a non-safety-grade bus. These motor-operated valves serve as the shutoff valves for the feedwater lines. Isolation of each line is accomplished by two containment isolation valves, consisting of one check valve inside the drywell and one positive closing check valve outside the containment (Figure 5.1-3). The closing check valve outside the containment is a spring-closing check valve that is held open by air. These check valves will be qualified to withstand the dynamic effects of a feedwater line break outside containment. Inside the containment, downstream of the inboard FW line check valve, there is a manual maintenance valve (B21-F005).

The design temperature and pressure of the feedwater line is the same as that of the reactor inlet nozzle (i.e., 8.62 MPa and 302°C) for turbine-driven feedwater pumps.

As discussed in Table 3.2-1 and shown in Figure 5.1-3, the feedwater piping is Quality Group A from the reactor pressure vessel out to, and including, the outboard isolation valve, Quality Group B from the outboard isolation valve to and including the shutoff valve, and Quality Group D beyond the shutoff valve. The feedwater piping and all connected piping of 65A and larger size is Seismic Category I only from the reactor pressure vessel out to, and including, the seismic interface restraint.

The materials used in the piping are in accordance with the applicable design code and supplementary requirements described in Section 3.2. The valve between the outboard isolation valve and the shutoff valve upstream of the RHR entry to the feedwater line is to effect a closed loop outside containment (CLOC) for containment bypass leakage control (Subsections 6.2.6 and 6.5.3).

The general requirements of the feedwater system are described in Subsections 7.7.1.1, 7.7.1.4, 7.7.2.4, and 10.4.7.

5.4.9.4 Safety Evaluation

Differential pressure on reactor internals under the assumed accident condition of a ruptured steamline is limited by the use of flow restrictors and by the use of four main steamlines. All main steam and feedwater piping will be designed in accordance with the requirements defined in Section 3.2. Design of the piping in accordance with these requirements ensures meeting the safety design bases.

5.4.9.5 Inspection and Testing

Testing is carried out in accordance with Subsection 3.9.6 and Chapter 14. Inservice inspection is considered in the design of the main steam and feedwater piping. This consideration assures adequate working space and access for the inspection of selected components.

5.4.10 Pressurizer

Not applicable to BWR.

5.4.11 Pressurizer Relief Discharge System

Not applicable to BWR.

5.4.12 Valves

5.4.12.1 Safety Design Bases

Line valves, such as gate, globe, and check valves, are located in the fluid systems to perform a mechanical function. Valves are components of the system pressure boundary and, having moving parts, are designed to operate efficiently to maintain the integrity of this boundary.

The valves operate under the internal pressure/temperature loading as well as the external loading experienced during the various system transient operating conditions. The design criteria, the design loading, and acceptability criteria are as specified in

The wetwell chamber design pressure is 309.9 kPaG and design temperature is 103.9°C.

Performance of the pressure suppression pool concept in condensing steam under water (main steamlines through the SRVs) has been demonstrated by the horizontal vent system tests as described in Appendix 3B.

The SRVs discharge steam from the relief valves through their exhaust piping and quenchers into the suppression pool. The quencher locations within the suppression pool are identified in Figures 1.2-3c, 1.2-13i and 3B-3. Operation of the SRVs is intermittent and closure of the valves with subsequent condensation of steam in the exhaust piping can produce a partial vacuum, thereby sucking suppression pool water into the exhaust pipes. Vacuum relief valves are provided on the exhaust piping to control the maximum SRV discharge bubble pressure resulting from high water levels in the SRV discharge pipe.

Under normal plant operating conditions, the maximum suppression pool water and wetwell airspace temperature is 35°C or less. Under blowdown conditions following an isolation event or LOCA, the initial pool water temperature may rise to a maximum of 76.7°C. The continued release of decay heat after the initial blowdown may result in suppression pool temperatures as high as 97.2°C. The Residual Heat Removal (RHR) System is available in the Suppression Pool Cooling mode to control the pool temperature. Heat is removed via the RHR heat exchanger(s) to the Reactor Building Cooling Water (RCW) System and finally to the Reactor Service Water (RSW) System. The RHR System is described in Subsection 5.4.7.

6.2.1.1.3 Design Evaluation

6.2.1.1.3.1 Summary Evaluation

The key design parameter and the maximum calculated accident parameters for the pressure suppression containment are shown in Table 6.2-1.

The maximum drywell pressure would occur during a feedwater line break. The maximum drywell temperature condition would result from a main steamline break. All of the analyses assume that the primary system and containment system are initially at the nominal operating conditions.

6.2.1.1.3.2 Containment Design Parameters

Table 6.2-2 provides a listing of the key design parameters of the primary containment system including the design characteristics of the drywell, suppression pool and the pressure suppression vent system.

Table 6.2-2a provides the performance parameters of the related ESF systems which supplement the design conditions of Table 6.2-2 for containment cooling purposes during post-blowdown long-term accident operation. Performance parameters given include those applicable to full capacity operation and reduced capacities assumed for containment analyses. Analyses calculating long-term containment response following a feedwater line break and main steamline break used containment cooling system only, and containment sprays were not used.

6.2.1.1.3.3 Accident Response Analysis

The containment functional evaluation is based upon the consideration of several postulated accident conditions which would result in the release of reactor coolant to the containment. These accidents include:

(1) An instantaneous guillotine rupture of a feedwater line

(2) An instantaneous guillotine rupture of a main steamline

(3) Small break accidents

The containment design pressure and temperature were established based on enveloping the results of this range of analyses plus providing NRC prescribed margins.

For the ABWR pressure suppression containment system, the peak containment pressure following a LOCA is very insensitive to variations in the size of the assumed primary system rupture. This is because the peak occurs late in the blowdown and is determined in very large part by the transfer of the noncondensible gases from the drywell to the wetwell air space. This process is not significantly influenced by the size of the break. In addition, there is a 15% margin between the peak calculated value and the containment design pressure that will easily accomodate small variations in the calculated maximum value.

Tolerances associated with fabrication and installation may result in the as-built size of the postulated break areas being 5% greater than the values presented in this chapter. Based on the above, these as-built variations would not invalidate the plant safety analysis presented in this chapter and Chapter 15.

6.2.1.1.3.3.1 Feedwater Line Break

Immediately following a double-ended rupture in one of the two main feedwater lines just outside the vessel (Figure 6.2-1), the flow from both sides of the break will be limited to the maximum allowed by critical flow considerations. The effective flow area on the RPV side is given in Figure 6.2-2. Reverse RPV flow in the second FW line is prevented by check valves shown in Figure 6.2-1. During the inventory depletion period, subcooled blowdown occurs and the effective flow area at saturated condition is much less than the actual break area. The detailed calculational method is provided in Reference 6.2-1.

The feedwater system side of the feedwater line break (FWLB) was modeled by adding a time variant feedwater mass flow rate and enthalpy directly to the drywell airspace. The time histories of the mass flow and enthalpy were determined from the operating characteristics of a typical feedwater system.

The maximum possible feedwater flow rate was calculated to be 164% of nuclear boiler rated (NBR), based on the response of the feedwater pumps to an instantaneous loss of discharge pressure. Since the Feedwater Control System will respond to decreasing RPV water level by demanding increased feedwater flow, and there is no FWLB sensor in the design, this maximum feedwater flow was conservatively assumed to continue for 120 seconds (Figure 6.2-3). This is very conservative because:

(1) All feedwater system flow is assumed to go directly to the drywell.

(2) Flashing in the broken feedwater line was ignored.

(3) Initial feedwater flow was assumed to be 105% NBR.

(4) The feedwater pump discharge flow will coastdown as the feedwater system pumps trip due to low suction pressure. During the inventory depletion period, the flow rate is less than 164% because of the highly subcooled blowdown. A feedwater line length of 100m was assumed on the feedwater system side.

The specific enthalpy time history, assuming the break flow of Figure 6.2-3, is shown in Figure 6.2-4.

6.2.1.1.3.3.1.1 Assumptions for Short-Term Response Analysis

The response of the Reactor Coolant System and the Containment System during the short-term blowdown period of the accident has been analyzed using the following assumptions:

(1) The initial conditions for the FWLB accident are such that system energy is maximized and the system mass is minimized. That is:

  (a) The reactor is operating at 102% of the rated thermal power, which maximizes the post-accident decay heat.

  (b) The initial suppression pool mass is at the low water level.

  (c) The initial wetwell air space volume is at the high water level.

  (d) The suppression pool temperature is the operating maximum temperature.

(2) The feedwater line is considered to be severed instantaneously. This results in the most rapid coolant loss and depressurization of the vessel, with coolant being discharged from both ends of the break.

(3) Scram occurs in less than one second from receipt of the high drywell pressure signal.

(4) The main steam isolation valves (MSIVs) start closing at 0.5 s after the accident. They are fully closed in the shortest possible time (at 3.5 s) following closure initiation. By assuming rapid closure of these valves, the RPV is maintained at a high pressure, which maximizes the calculated discharge of high energy water into the drywell.

(5) The vessel depressurization flow rates are calculated using Moody's homogeneous equilibrium model (HEM) for the critical break flow (Reference 6.2-2). The break area on the RPV side for this study is shown in Figure 6.2-2. During the inventory depletion period, subcooled blowdown occurs and the effective break area at saturated conditions is much less than the actual area. The detailed calculational method is provided in Reference 6.2-1.

Reactor vessel internal heat transfer is modeled by dividing the vessel and internals into six metal nodes. A seventh node depends on the fluid (saturated or subcooled liquid, saturated steam) covering the node at the time. The assumptions include:

(a) The center of gravity of each node is specified as the elevation of that node.

(b) Mass of water in system piping (except for HPCF and feedwater) is included in initial vessel inventory.

(c) Initial thermal power is 102% of rated power at steady-state conditions with corresponding heat balance parameters which correspond to turbine control valve constant pressure of 6.75 MPaA.

(d) Pump heat, fuel relaxation, and metal-water reaction heat are added to the ANSI/ANS-5.1 decay heat curve plus 20% margin.

(e) Initial vessel pressure is 7.31 MPaA.

(6) There are two HPCF Systems, one RCIC System, and three RHR Systems in the ABWR. One HPCF System, one RCIC System and two RHR Systems are assumed to be available. HPCF flow cannot begin until 36 seconds after a break, and then the flow rate is a function of the vessel-to-wetwell differential pressure. Rated HPCF flow is 182 m^3/h per system at 8.12 MPaD and 727 m^3/h, per system at 0.69 MPaD. Rated RHR flow is 954 m^3/h at 0.28 MPaD with shutoff head of 1.55 MPaD. Rated RCIC flow is 182 m^3/h with reactor pressure between 8.12 MPaG and 1.04 MPaG, and system shuts down at 0.34 MPaG.

(7) Drywell and wetwell airspaces are homogeneous mixtures of inert atmosphere, vapor and liquid water.

6.2.1.1.5.8.2 Vacuum Valves Operability Tests

As described in 6.2.1.1.5.7.4, the vacuum relief valves will be tested for free movement during each refueling outage. There will be no operability tests at monthly intervals, see Subsection 6.2.1.1.5.6.3 for justification.

6.2.1.1.6 Suppression Pool Dynamic Loads

During a LOCA and events such as SRV actuation, steam released from the primary system is channeled into the suppression pool where it is condensed. These actuation events impose hydrodynamic loading conditions on the containment system structures. The containment and its internal structures are designed to withstand all loading conditions associated with these events. These hydrodynamic loads are combined with those from the postulated seismic events in the load combinations specified Subsections 3.8.1.3 and 3.8.3.3. A detailed description and definition of hydrodynamic loading conditions for structure design is provided in Appendix 3B. These loading conditions are briefly summarized in the following paragraphs.

6.2.1.1.6.1 LOCA Loads

During a postulated loss-of-coolant accident (LOCA) inside the drywell, wetwell region will be subjected to the following three sequential hydrodynamic loading conditions of significance to structure design:

- Pool Swell loads
- Condensation Oscillation (CO) loads
- Chugging (CH) loads

Following a postulated LOCA and after the water is cleared from the vents, air/steam mixture from the drywell flows into the suppression pool creating a large bubble at vent exit as it exits into the pool. Bubble at vent exit expands to suppression pool hydrostatic pressure, as the air/steam mixture flow continues from the pressurized drywell. The water ligament above the expanding bubble is accelerated upward which gives rise to pool swell phenomena lasting, typically for a couple of seconds. During this pool swell phase, the wetwell region is subjected to:

(a) loads on suppression pool boundary and drag loads on structures initially submerged in the pool

(b) loads on wetwell gas space

(c) impact and drag loads on structures above the initial pool surface

The CO period of a postulated LOCA follows the pool swell transient period. During the CO period the steam condensation process at the vent exit induces periodic transient loads on the suppression pool boundary and structures initially submerged in the pool. Figure 6.2-43 shows a typical CO loading condition.

The CH period of a postulated LOCA follows the CO period, and it occurs during periods of low vent steam mass flux. During the chugging period the steam condensation process at the vent exit imposes loads on the suppression pool boundary and structures initially submerged in the pool. Figure 6.2-44 shows a typical CH loading condition.

6.2.1.1.6.2 SRV Actuation Loads

During the actuation of SRV, air (initially contained in the SRV discharge line) after it exits into the suppression pool and oscillates as Rayleigh bubble while rising to the pool free surface. The oscillating air bubble produces hydrodynamic loads on the pool boundary and drag loads on structures submerged in the pool. After the air has been expelled, steam exits steadily and condenses in the pool. This condensing steady SRV steam flow has been found to produce negligible loading on the pool boundary. Figure 6.2-45 shows a typical graphical representation of the dynamic loading due to SRV actuation.

6.2.1.1.7 Asymmetric Loading Conditions

Asymmetric loads are included in the load combination specified in Subsections 3.8.1.3, 3.8.2.3 and 3.8.3.3. The containment and internal structures are designed for these loads within the acceptance criteria specified in Subsections 3.8.1.5, 3.8.2.5 and 3.8.3.5. Since internal structures are not subject to external design or tornado winds, they are not designed for these loads.

Localized pipe forces, pool swell and SRV actuation are asymmetric pressure loads which act on the containment and internal structure (see Subsection 6.2.1.1.5 for magnitudes of pool swell and SRV loads).

The loads associated with embedded plates are concentrated forces and moments which differ according to the type of structure or equipment being supported. Earthquake loads are inertial loads caused by seismic accelerations. The magnitude of these loads is discussed in Section 3.7.

6.2.1.1.8 Containment Environment Control

The drywell ventilation system maintains temperature, pressure and humidity in the containment and its subcompartments at the normal design conditions. The safety-related containment heat removal systems described in Subsection 6.2.2 maintain required containment atmosphere conditions during accidents. Since the loss of the drywell ventilation system does not result in exceeding the design environmental conditions for the safety-related equipment inside the containment, the drywell system is not classified as safety-related.

Table 6.3-3 Single Failure Evaluation*

Assumed Failure	Systems Remaining†
Emergency Diesel Generator A	All ADS, RCIC, 2 HPCF, 2 RHR/LPFL
Emergency Diesel Generator B or C	All ADS, RCIC, 1 HPCF, 2 RHR/LPFL
RCIC Injection Valve	All ADS, 2 HPCF, 3 RHR/LPFL
One ADS Valve	All ADS minus one, RCIC, 2 HPCF, 3 RHR/LPFL

* Single, active failures are considered in the ECCS performance evaluation. Other postulated failures are not specifically considered because they all result in at least as much ECCS capacity as one of the above designed failures.

† Systems remaining, as identified in this table, are applicable to all non-ECCS line breaks. For the LOCA from an ECCS line break, the systems remaining are those listed, less the ECCS system in which the break is assumed.

Table 6.3-4 Summary of Results of LOCA Analysis

Break Location	Break Size* (cm²)	Systems Available	PCT (°C)	Maximum Local Oxidation
Based on Appendix K evaluation models:				
Steamline Inside Containment	985	1HPCF + RCIC +2 RHR/LPFL + 8 ADS	552	0.03%
Feedwater Line	839	1 HPCF + 2 RHR/LPFL + 8 ADS	542	0.03%
RHR Shutdown Cooling Suction Line	792	1 HPCF + RCIC + 2 RHR/LPFL+ 8 ADS	542	0.03%
RHR/LPFL Injection Line	205	1 HPCF + RCIC + 1RHR/LPFL + 8 ADS	542	0.03%
High Pressure Core Flooder	92	RCIC+2RHR/ LPFL + 8 ADS	542	0.03%
Bottom Head Drain Line	20.3	1HPCF + RCIC + 2 RHR/LPFL + 8 ADS	542	0.03%
Steamline Outside Containment	3939	1 HPCF + RCIC + 2 RHR/LPFL + 8 ADS	621	0.03%
Based on bounding values:				
Steamline Outside Containment	3939	1 HPCF + RCIC + 2 RHR/LPFL + 8 ADS	619	0.03%

* The most severe ABWR desgin basis LOCA calculations (Subsection 6.3.3.7.8) invlove use of bounding worst-case values for key plant parameters — including an arbitrary 20% increase in the break flow rate. Even with these bounding assumptions, the LOCA analyses demonstrate that the ABWR design still retains large margins between predicted peak fuel clad temperatures and the criteria of 10 CFR 50, Appendix K.

Tolerances associated with fabrication and installation may result as-built break areas that could be 5% greater than these values. Based on the above conservatisms in the LOCA analyses, these as-built variations would not invalidate the plant safety analysis presented in Chapter 6 and Chapter 15.

Note: The core-wide metal-water reaction for this analysis has been calculated using method 1 described in Reference 6.3-1. This results in a core-wide metal-water reaction of 0.03%.

containment leakage to the environment was assumed for the first 20 minutes after LOCA event initiation (in addition to the leakage through the MSIVs to the main turbine condenser). Each SGTS fan was sized to individually establish a continuously negative differential pressure (considering the effect of wind) within 10 minutes after SGTS initiation. The dose analysis therefore assumes direct leakage from the containment to the environs for twice the required period. In addition, it should be recognized that fission product release on the order of that specified in Regulatory Guide 1.3 and used in the LOCA dose analyses (Subsection 15.6.5) realistically requires significant core damage and most likely more than 10 or 20 minutes for transport to and leakage from the primary containment.

The calculation accounted for all expected heat sources in secondary containment after a LOCA inside primary containment. Where appropriately conservative, a realistic basis was used to determine the heat loads. For example, no single failure of a diesel was assumed, since it is likely that all divisions of power would be available. Failure of one SGTS fan to start was assumed as the single failure. Therefore, heat loads from all divisions of ECCS motors and piping were used in the calculation.

Per SRP 6.2.3, II.3(b) and SRP 6.5.3, II.2, secondary containment should be held below –6.4 mm w.g. under all wind conditions up to the wind speed at which diffusion becomes great enough to assure site boundary exposures less than those calculated for DBAs, even if ex-filtration occurs (i.e., no credit for SGTS is taken). For the ABWR, dispersion factors were calculated for each stability class over a range of wind speeds. Above 8.0 m/s, stability class D predominates and conservatively bounds observed meteorological conditions. At 8.9 m/s, above the 8.0 m/s stability class D transition, the dispersion from the increased wind speed results in offsite doses equal to or lower than the design basis calculation, which assumes the most stable, F-class stability and a 1 m/s wind speed. Therefore, the ABWR SGTS was designed to establish and maintain a negative pressure in secondary containment within 10 minutes for any wind speed up to and including 8.9 m/s.

6.5.1.3.3 SGTS Filter Train

The SGTS filter train, consisting of a demister, process heater, pre-filter, two HEPA filters, and an iodine adsorber, is considered active, and in practice provides the reliability associated with a passive component. Furthermore, the ABWR SGTS has incorporated design features to eliminate potential failures or improper operation. These features include:

(1) The advanced design of the filter housing and flow pattern virtually eliminates any untreated bypass of the filter. In addition, the all-welded design is such that degradation of filter housing integrity is not likely to occur during system standby or operation.

(2) A number of operating plant events (during normal plant operation) have ocurred causing the inadvertent deluge wetting of the charcoal. These events have rendered the filter train unavailable for safety service. These events have been observed to warrant an improved deluge design concept. These unintended deluge operations have been caused by personnel error and by failures in mechanical or electrical components. In the ABWR design, the deluge piping is not connected permanently from the fire protection system to the filter housing nozzle. Instead, a normally disconnected hose from the fire protection system is provided to act as a "spool piece" for connection by operating personnel to the filter housing, as required.

(3) Decay heat is not sufficient to cause a fire in the charcoal adsorber or HEPA filter. Calculations indicate that air flow from the process fan is more than enough to remove the heat from decay of the radioactive iodine on the charcoal or filters. Heating does not occur sufficient to cause iodine desorption or ignition of the charcoal. With the reduced source term expected for most sequences [Subsection 6.5.1.3.3(4)], any heating of the charcoal is even further reduced. Tripping or failure of the process fan will result in the auto operation of the cooling fan and the operation of the other SGTS train. The cooling fan operation will preclude charcoal heatup. No other mechanism for starting a fire in the filter housing during an accident has been identified. Other possible sequences for starting a fire in the filter train could occur during normal plant operation or plant shutdown. These sequences would involve an unspecified maintenance or operating personnel activity or an incredible malfunction of the space heaters. In this case, a fire in the SGTS charcoal, like in the Offgas System, would be a matter of plant availability and not of plant safety. The space heaters, located inside the SGTS filter housing, are powered only during SGTS standby and not during system operation. Therefore, the space heaters are not a potential cause of fire (and SGTS unavailability) when the SGTS is required to meet the licensing-basis release limits (and presumably inaccessible for repair).

Note that the space heaters each have a small fan which better distributes the heat and minimizes local warming by providing a more uniform temperature throughout the filter housing. This uniform heating further reduces the risk of fire by lowering local temperatures around the space heater and by improving the accuracy of the temperature measurements (used to detect high temperature) taken at necessarily discrete points within the filter housing.

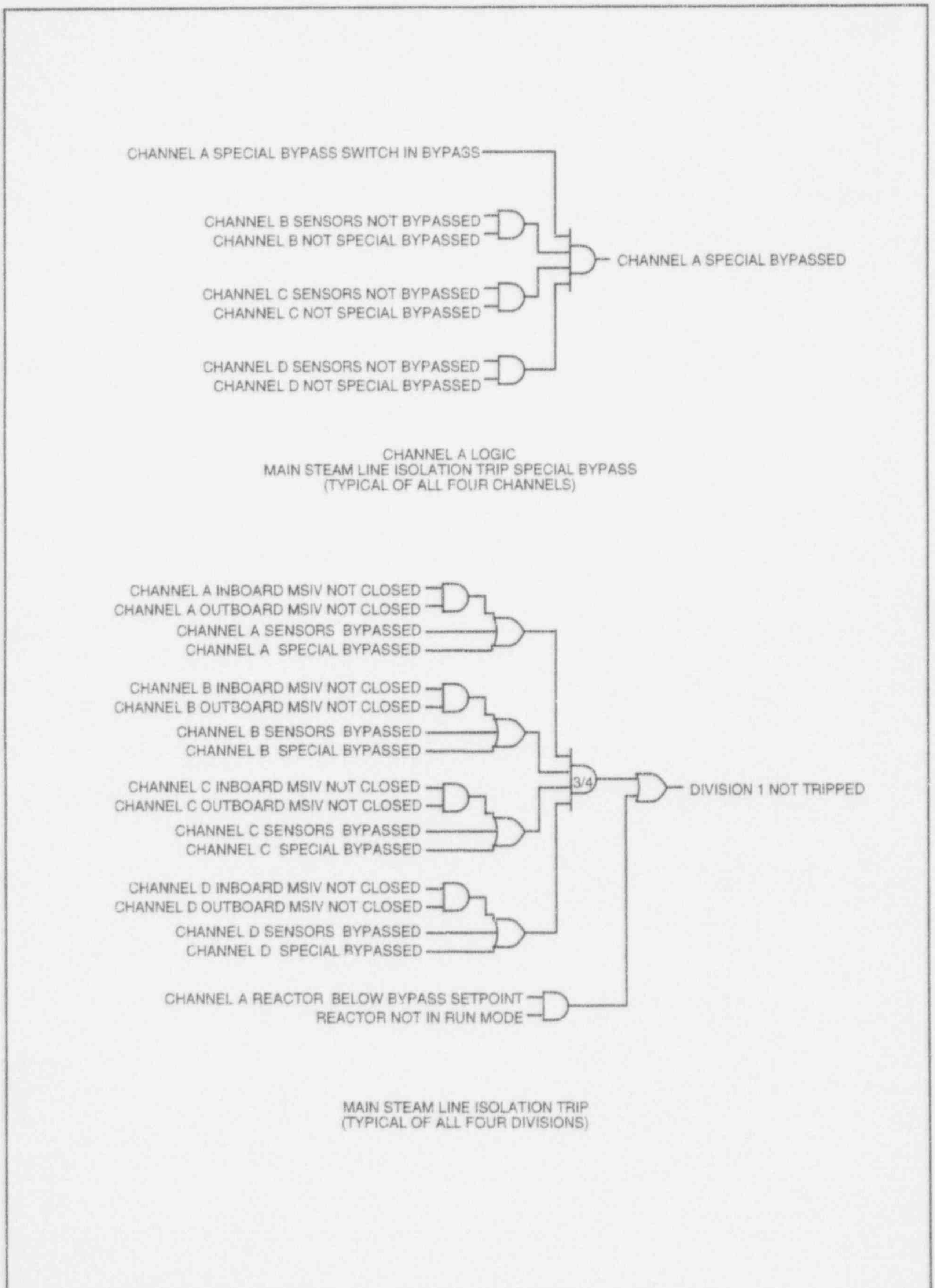

Figure 7.2-4 Division 1 Trip Logic

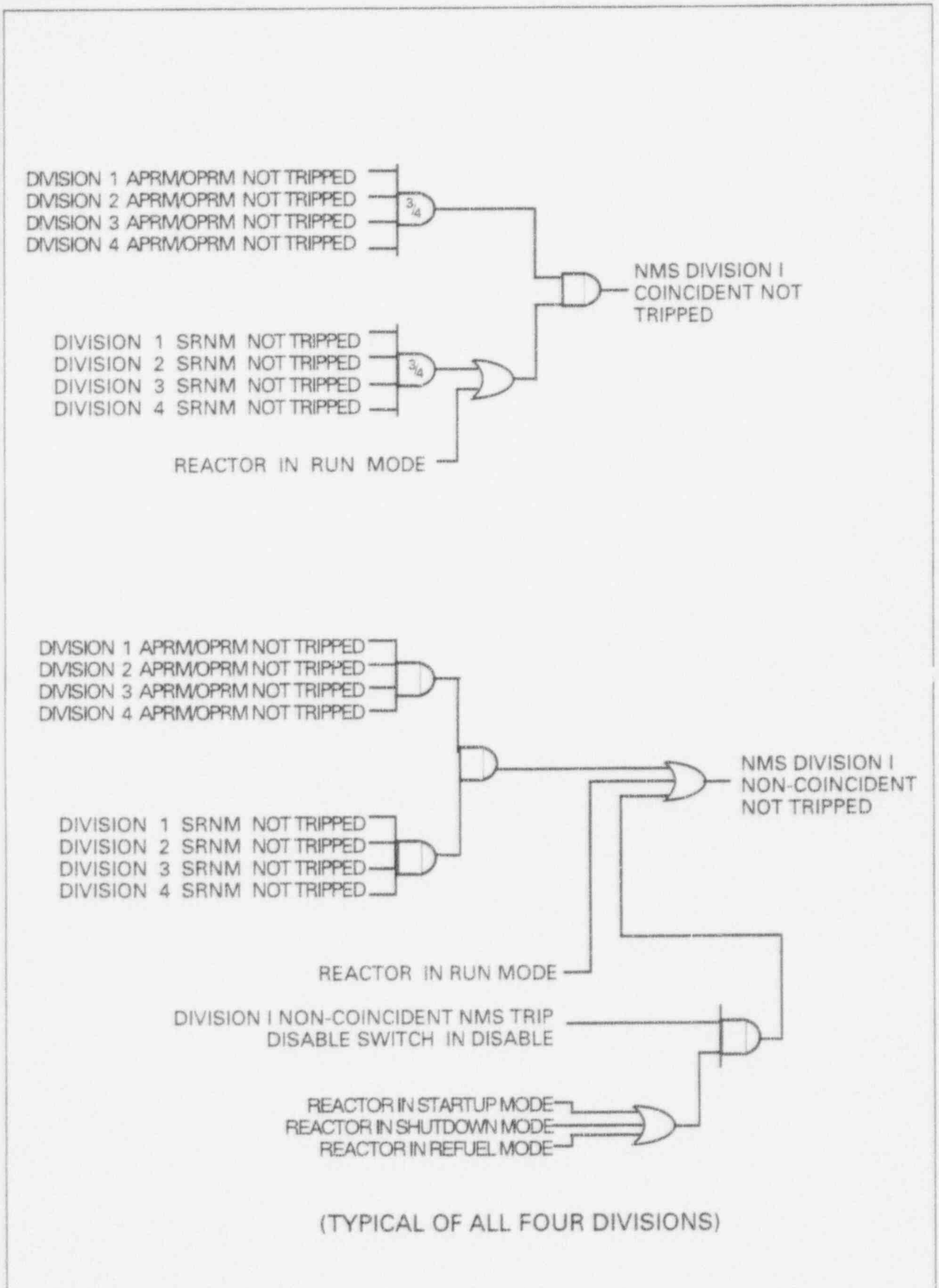

Figure 7.2-5 Division 1 Trip Logic Coincident and Non-Coincident NMS Trips

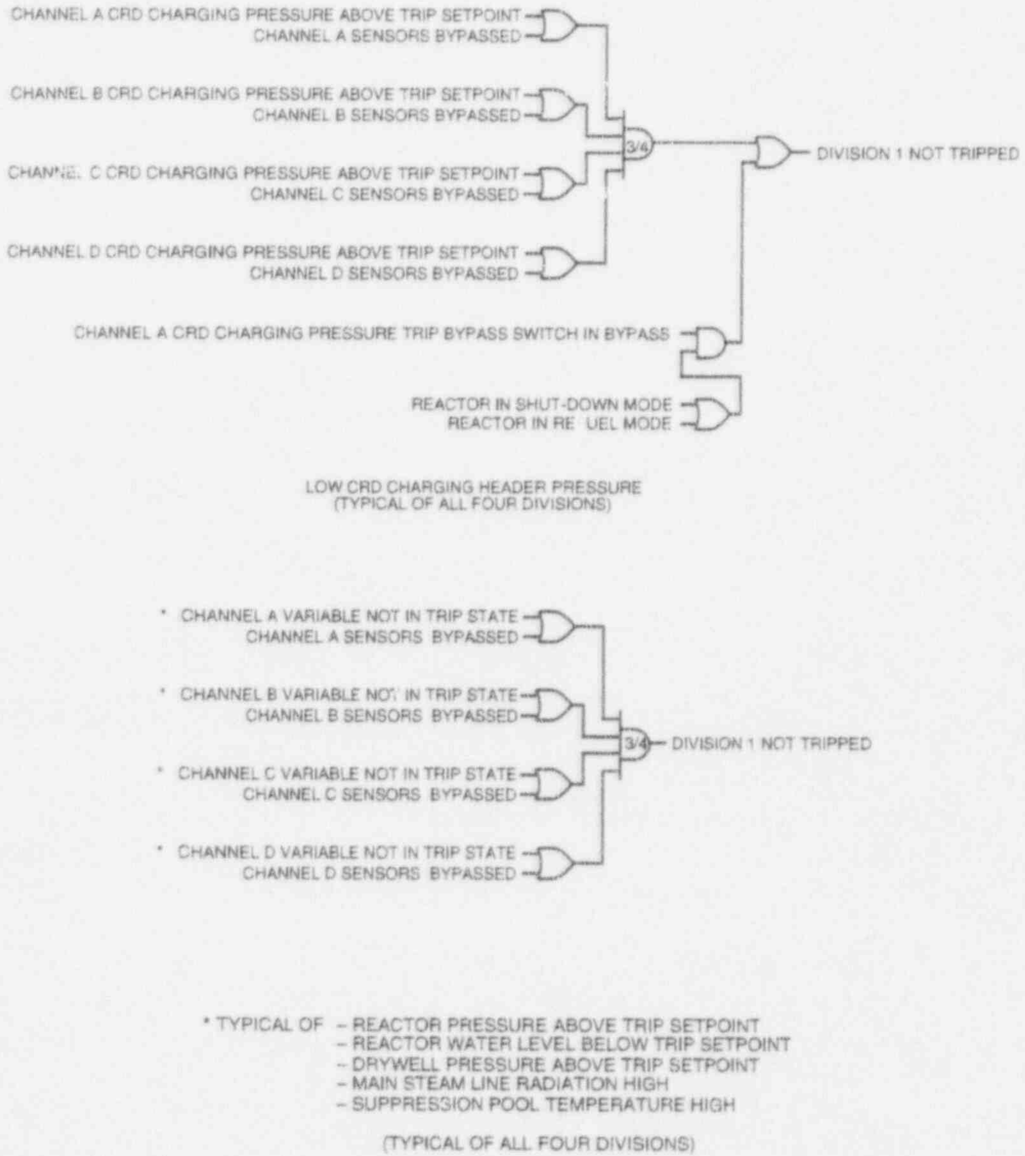

Figure 7.2-6 Division 1 Trip Logic

Figure 7.2-7 Deleted

The expected system availability during its 60-year life exceeds 0.99. The expected reliability is based upon the expected frequency of an inadvertent movement of more than one control rod. The expected frequency of an inadvertent movement of more than one control rod, due to failure, is less than or equal to once in 100 reactor operating years.

The RCIS design assures that no credible single failure or single operator error can cause or require a scram or require a plant shutdown. The RCIS design preferentially fails in a manner which results in no further normal rod movement.

(9) RCIS Bypass Capabilities

The RCIS provides the capability to bypass synchro A, if it is bad, and select synchro B for providing rod position data to both channels of the RCIS. The number and distribution of bypassed synchros are procedurally controlled by applicable plant Technical Specifications.

The RCIS allows the operator to completely bypass up to eight control rods by declaring them "Inoperable" and placing them in a bypass condition. Through operator action, an update in the status of the control rods placed into "inoperable" bypassed condition is available at the CRT display. At the display, the operator can request the data to be downloaded into the memory of the RAPI Subsystem logic with confirmation of a successful download completion signal being sent back to the CRT display.

Download of a new RCIS "Inoperable Bypass Status" to the RAPI Subsystem is only allowed when the RCIS is in a manual rod movement mode and when both keylock permissive switches are activated at the RCIS panels.

The operator can substitute a position for the rod that has been placed in a bypass state into both channels of the RCIS, if the substitute position feature is used. The substituted rod position value entered by the operator is used as the effective measured rod position that is stored in both rod action control channels and sent to other systems (e.g., the Process Computer System).

For purposes of conducting periodical inspections on FMCRD components, RCIS allows placing up to 21 control rods in "inoperable" bypass condition, only when the reactor mode switch is in REFUEL mode.

The RCIS enforces rod movement blocks when the control rod has been placed in an inoperative bypass status. This is accomplished by the RCIS logic by not sending any rod movement pulses to the FMCRD.

In response to activation of special insertion functions, such as ARI, control rods in bypass condition do not receive movement pulses.

(10) Single/Dual Rod Sequence Restriction Override (S/DRSRO) Bypass

The RCIS single/dual rod sequence restriction override bypass feature allows the operator to perform special dual or single rod scram time surveillance testing at any power level of the reactor. In order to perform this test, it is often necessary to perform single rod movements that are not allowed normally by the sequence restrictions of the RCIS.

When a control rod is placed in a S/DRSRO bypass condition, that control rod is no longer used in determining compliance to the RCIS sequence restrictions (e.g., the ganged withdrawal sequence and RRPS).

The operator can only perform manual rod movements of control rods in the S/DRSRO bypass condition. The logic of the RCIS allows this manual single/dual rod withdrawals for special scram time surveillance testing.

The operator can place up to two control rods associated with the same hydraulic control unit (HCU) in the S/DRSRO bypass condition.

The dedicated RCIS operator interface panel contains status indication of control rods in a S/DRSRO bypass condition.

The RCIS ensures that S/DRSRO bypass logic conditions have no effect on special insertion functions for an ARI or SCRAM following condition and also no effect on other rod block functions, such as MRBM, APRM, or SRNM period.

The drive insertion following a dual/single rod scram test occurs automatically. The operator makes the necessary adjustment of control rods in the system prior to the start of test for insertions, and restores the control rod to the desired positions after test completion.

The RCIS is a dual channel system and the logic of the system provides a capability for the operator to invoke bypass conditions that affect only one channel of the RCIS. The interlock logic prevents the operator from placing both channels in bypass. Logic enforces bypass conditions to ensure that the capability to perform any special function (such as an ARI, scram following, and SCRRI) is not prevented.

The RCIS logic ensures that any special restrictions that are placed on the plant operation are enforced as specified in the applicable plant Technical Specifications for invoked bypass conditions.

(2) Classification

This system is a power generation system and is classified as not required for safety.

(3) Power Sources

(a) Normal

Each processing channel of the triply redundant digital processor receives its respective power input from an uninterruptible, independent source of the instrument and control power supply system. Other system equipments such as the transmitters, input conditioners, voters, output device drivers, control room displays, etc., will also derive their required power sources from the same redundant uninterruptible power supply system.

Variable voltage, variable frequency electrical power is generated by the adjustable speed drives (ASDs) for use by the induction motors in the RIPs. Four medium voltage power buses are used to provide input power to the ten ASDs. These buses are fed from the unit auxiliary transformers connecting to the main turbine-generator. Two of the buses each provide power directly to a pair of ASDs. The other two buses each provide power to a motor-generator (M-G) set which, in turn, supplies power to three ASDs operating in parallel (see one-line diagram for AC power distribution provided as Figure 8.3-2).

The allocation of the RIP equipment on the four power buses is such that on loss of any single power bus, a maximum of three RIPs are affected. At least one circuit breaker is provided along each circuit path to protect power equipment from being damaged by overcurrent.

(b) Alternate and Startup

During the plant startup, or on loss of normal auxiliary power, reserve auxiliary transformer provides backup power to the medium voltage normal auxiliary power systems. The M-G set flywheels provide sufficient inertia for six of the RIPs to extend core flow coastdown time, thereby reducing the change in MCPR during the momentary voltage drop transient.

(4) Normal Operation

Reactor recirculation flow is varied by modulating the recirculation internal pump speeds through the voltage and frequency modulation of the adjustable speed drive output. By properly controlling the operating speed of the RIPs, the recirculation system can automatically change the reactor power level.

Control of core flow is such that, at various control rod patterns, different power level changes can be automatically accommodated. For a rod pattern where rated power accompanies 100% flow, power can be reduced to 70% of full power by full automatic or manual flow variation. At other rod patterns, automatic or manual power control is possible over a range of approximately 30% from the maximum operating power level for that rod pattern. Below 70% power level, only manual control of power (i.e., by means of manual flow setpoint control) is available.

An increase in recirculation flow temporarily reduces the void content of the moderator by increasing the flow of coolant through the core. The additional neutron moderation increases reactivity of the core, which causes reactor power level to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and a new (higher) steady-state power level is established. When recirculation flow is reduced, the power level is reduced in the reverse manner. The RFC System, operating in conjunction with the main turbine pressure regulator control, provides fully automatic load following.

The RFC System is designed to allow both automatic and manual operation. In the automatic mode, either total automatic or semi-automatic operation is possible. Fully automatic, called "Master Auto" mode, refers to the automatic load following (ALF) operation in which the master controller receives a load demand error signal from the main turbine pressure regulator. The load demand error signal is then applied to a cascade of lead/lag and proportional-integral (PI) dynamic elements in the master controller to generate a flow demand signal for balancing out the load demand error to zero. The flow demand signal is forwarded to the flow controller for comparing with the sensed core flow. The resulting flow demand error is used to generate a suitable gang speed demand to the ASDs. The speed demand to the individual ASDs causes adjustment of RIP motor power input, which changes the operating speed of the RIP and, hence, core flow and core power. This process continues until both the errors existing at the input of the flow controller and master controller are driven to zero. Fully automatic control is provided by the master controller when in the automatic mode. The flow controller can remain in automatic even though the master controller is in manual.

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"quality requirements" that identify features which require specific QA verification of compliance to drawing requirements.

For components classified as American Society of Mechanical Engineers (ASME) Section III, the shop operation must secure and maintain an ASME "N" stamp, which requires the submittal of an acceptable ASME quality plan and a corresponding procedural manual.

Additionally, the shop operation must submit to frequent ASME audits and component inspections by resident state code inspectors. Prior to shipment, every component inspection item is reviewed by QA supervisory personnel and combined into a summary product quality checklist (PQL). By issuance of the PQL, verification is made that all quality requirements have been confirmed and are on record in the product's historical file.

9.1.4.4.2 Testing

Qualification testing is performed on refueling and servicing equipment prior to multi-unit production. Test specifications are defined by the responsible design engineer and may include a sequence of operations, load capacity and life cycles tests. These test activities are performed by an independent test engineering group and, in many cases, a full design review of the product is conducted before and after the qualification testing cycle. Any design changes affecting function, that are made after the completion of qualification testing, are requalified by test or calculation.

Functional tests are performed in the shop prior to the shipment of production units and generally include electrical tests, leak tests, and sequence of operations tests.

When the unit is received at the site, it is inspected to ensure no damage has occurred during transit or storage. Prior to use and at periodic intervals, each piece of equipment is again tested to ensure the electrical and/or mechanical functions are operational.

Passive units, such as the fuel storage racks, are visually inspected prior to use.

Fuel-handling and vessel servicing equipment preoperational tests are described in Subsection 14.2.12.

9.1.4.5 Instrumentation Requirements

9.1.4.5.1 Refueling Machine

The refueling machine has a X-Y-Z position indicator system that informs the operator which core fuel cell the fuel grapple is accessing. Interlocks and a control room monitor are provided to prevent the fuel grapple from operating in a fuel cell where the control rod is not in the proper orientation for refueling.

Additionally, there is a series of mechanically activated switches and relays that provides monitor indications on the operator's console for grapple limits, hoist and cable load conditions, and confirmation that the grapple's hook is either engaged or released.

A series of load cells is installed to provide automatic shutdown whenever threshold limits are exceeded for either the fuel grapple or the auxiliary hoist units.

9.1.4.5.2 Fuel Support Grapple

Although the fuel support grapple is not essential to safety, it has an instrumentation system consisting of mechanical switches and indicator lights. This system provides the operator with a positive indication that the grapple is properly aligned and oriented and that the grappling mechanism is either extended or retracted.

9.1.4.5.3 Other

Refer to Table 9.1-5 for additional refueling and servicing equipment not requiring instrumentation.

9.1.4.5.4 Radiation Monitoring

The fuel area ventilation exhaust radiation monitoring is discussed in Subsection 11.5.2.1.3.

9.1.5 Overhead Heavy Load Handling Systems (OHLH)

9.1.5.1 Design Bases

The equipment covered by this subsection concerns items considered as heavy loads that are handled under conditions that mandate critical handling compliance.

Critical load handling conditions include loads, equipment, and operations which, if inadvertent operations or equipment malfunctions either separately or in combination, could cause:

(1) A release of radioactivity.

(2) A criticality accident.

(3) The inability to cool fuel within reactor vessel or spent fuel pool.

(4) Prevent safe shutdown of the reactor. This includes risk assessments to spent fuel and storage pool water levels, cooling of fuel pool water, new fuel criticality. This includes all components and equipment used in moving any load weighing more than one fuel assembly, including the weight of its associated handling devices (i.e. 4.45 kN).

The R/B crane as designed shall provide a safe and effective means for transporting heavy loads, including the handling of new and spent fuel, plant equipment and service tools. Safe handling includes design considerations for maintaining occupational radiation exposure as low as practicable during transportation and handling.

Where applicable, the appropriate seismic category, safety class quality group, ASME, ANSI, industrial and electrical codes have been identified (Tables 3.2-1 and 9.1-6). The designs will conform to the relevant requirements of General Design Criteria 2, 4 and 61 of 10CFR50 Appendix A.

The lifting capacity of each crane or hoist is designed to at least the maximum actual or anticipated weight of equipment and handling devices in a given area serviced. The hoists, cranes, or other lifting devices shall comply with the requirements of ANSI N14.6, ANSI B30.9, ANSI B30.10 and NUREG-0612, Subsection 5.1.1(4) or 5.1.1(5). Cranes and hoists are also designed to criteria and guidelines of NUREG-0612, Subsection 5.1.1(7), ANSI B30.2 and CMAA-70 specifications for electrical overhead traveling cranes, including ANSI B30.11, ANSI B30.16, and NUREG-0554 as applicable.

9.1.5.2 System Description

9.1.5.2.1 Reactor Building Crane

The Reactor Building (R/B) is a reinforced concrete structure which encloses the reinforced concrete containment vessel, the refueling floor, new-fuel storage vault, the storage pools for spent-fuel and the dryer and separator and other equipment. The R/B crane provides heavy load lifting capability for the refueling floor. The main hook 1.471 MN will be used to lift the concrete shield blocks, drywell head, reactor pressure vessel (RPV) head insulation, RPV head, dryer, separator strongback, RPV head strongback carousel, new-fuel shipping containers, and spent-fuel shipping cask. The orderly placement and movement paths of these components by the R/B crane precludes transport of these heavy loads over the spent fuel storage pool or over the new-fuel storage vault.

The R/B crane will be used during refueling/servicing as well as when the plant is online. During refueling/servicing, the crane handles the shield plugs, drywell and reactor vessel heads, steam dryer and separators, etc. (Table 9.1-7). Minimum crane coverage includes R/B refueling floor laydown areas, and R/B equipment storage pit. During normal plant operation, the crane will be used to handle new-fuel shipping containers and the spent-fuel shipping casks. Minimum crane coverage must include the new-fuel vault, the R/B equipment hatches, and the spent-fuel cask loading and washdown pits. A description of the refueling procedure can be found in Section 9.1.4.

The R/B crane will be interlocked to prevent movement of heavy loads over the spent-fuel storage portion of the spent-fuel storage pool. Since the crane is used for handling large heavy objects over the open reactor, the crane is of Type I design. The

R/B crane shall be designed to meet the single-failure-proof requirements of NUREG-0554.

9.1.5.2.2 Other Overhead Load Handling System

9.1.5.2.2.1 Upper Drywell Servicing Equipment

The upper drywell arrangement provides servicing access for the main steam isolation valves (MSIVs), feedwater isolation valves, safety/relief valves (SRVs), emergency core cooling systems (ECCS) isolation valves, and drywell cooling coils, fans and motors. Access to the space is via the R/B through either the upper drywell personnel lock or equipment hatch. All equipment is removed through the upper drywell equipment hatch. Platforms are provided for servicing the feedwater and MSIVs, SRVs, and drywell cooling equipment with the object of reducing maintenance time and operator exposure. The MSIVs, SRVs, and feedwater isolation valves all weigh in excess of 4.45 kN. Thus, they are considered heavy loads.

With maintenance activity only being performed during a refueling outage, only safe shutdown ECCS piping and valves need be protected from any inadvertent load drops. Since only one division of ECCS is required to maintain the safe shutdown condition and the ECCS divisions are spatially separated, an inadvertent load drop that breaks more than one division of ECCS is not credible. In addition, two levels of piping support structures and equipment platforms separate and shield the ECCS piping from heavy loads transport path.

This protection is adequate such that no credible load drop can cause either:

(1) A release of radioactivity.

(2) A criticality accident.

(3) The inability to cool fuel within reactor vessel or spent fuel pool.

9.1.5.2.2.2 Lower Drywell Servicing Equipment

The lower drywell (L/D) arrangement provides for servicing, handling and transportation operations for the RIP and FMCRD. The lower drywell OHLHS consists of a rotating equipment service platform, chain hoists, FMCRD removal machine, a RIP removal machine, and other special purpose tools.

The rotating equipment platform provides a work surface under the reactor vessel to support the weight of personnel, tools, and equipment and to facilitate transportation moves and heavy load handling operations. The platform rotates 360° in either direction from its stored or "idle" position. The platform is designed to accommodate

(d) SPCU pumps

(e) MUWC transfer pumps (three 149 m^3/h at 0.971 MPa head)

(3) Water can be sent to the CST from the following sources:

(a) MWP pumps

(b) CRD system

(c) Radwaste disposal system

(d) Condensate demineralizer system effluent (main condenser high level relief)

(4) Associated receiving and distribution piping valves, instruments, and controls shall be provided.

(5) Overflow and drain from the CST shall be sent to the radwaste system for treatment.

(6) Any outdoor piping shall be protected from freezing.

(7) All surfaces coming in contact with the condensate shall be made of corrosion-resistant materials.

(8) All of the pumps mentioned in (2) above shall be located at an elevation such that adequate suction head is present at all water levels in the CST.

(9) Instrumentation shall be provided to indicate CST water level in the main control room, Radwaste Building control room and Remote Shutdown System. High water level shall be alarmed both in the Radwaste Building control room and in the main control room (Subsection 11.2.1.2). Low water level shall be alarmed in the main control room.

(10) Potential flooding is discussed in Subsection 3.4. Potential flooding from lines within the Reactor Building and the Control Building are evaluated in Subsection 3.4.1.1.1.

9.2.9.3 Safety Evaluation

Operation of the MUWC System is not required to assure any of the following conditions:

(1) Integrity of the reactor coolant pressure boundary.

(2) Capability to shut down the reactor and maintain it in a safe shutdown condition.

(3) Ability to prevent or mitigate the consequences of events that could result in potential offsite exposures.

The MUWC System is not safety-related. However, the system incorporates features that assure reliable operation over the full range of normal plant operations.

9.2.9.4 Tests and Inspections

The MUWC System is proved operable by its use during normal plant operation. Portions of the system normally closed to flow can be tested to ensure operability and the integrity of the system.

9.2.10 Makeup Water Purified System

9.2.10.1 Design Bases

(1) The Makeup W .... Purified (MUWP) System shall provide makeup water purified for makeup to the reactor coolant system and plant auxiliary systems.

(2) The MUWP System shall provide purified water to the uses shown in Table 9.2-2.

(3) The MUWP System shall provide water of the quality shown in Table 9.2-2a. If these water quality requirements are not met, the water shall not be used in any safety-related system. The out-of-spec water shall be reprocessed or discharged.

(4) The MUWP System is not safety-related.

(5) All piping and other equipment shall be made of corrosion-resistant materials.

(6) The system shall be designed to prevent any radioactive contamination of the purified water.

(7) The interfaces between the MUWP System and all safety-related systems are located either in the Control Building or Reactor Building, which are Seismic Category I, tornado-missile resistant and flood protected structures. The interfaces with safety-related systems are safety-related valves which are part of the safety-related systems. The portions of the MUWP System, which upon their failure during a seismic event can adversely impact structures, systems, or components important to safety, shall be designed to assure their integrity under seismic loading resulting from a safe shutdown earthquake.

(8) Safety-related equipment located by portions of the MUWP System are in Seismic Category I structures and protected from all system impact.

9.2.10.2 System Description

The MUWP System P&ID is shown in Figure 9.2-5. This system includes the following:

(1) Distribution piping, valves, instruments and controls shall be provided.

(2) Any outdoor piping shall be protected from freezing.

(3) All surfaces coming in contact with the purified water shall be made of corrosion-resistant materials.

(4) Continuous analyzers are located at the MUWP System. These are supplemented as needed by grab samples. Allowance is made in the water quality specifications for some pickup of carbon dioxide and air in any demineralized water storage tank. The pickup of corrosion products should be minimal because the MUWP piping is stainless steel.

(5) Intrusion of radioactivity into the MUWP System from other potentially radioactive systems are prevented by one or more of the following:

 (a) Check valves in the MUWP lines.

 (b) Air (or siphon) breaks in the MUWP lines.

 (c) The MUWP System lines are pressurized while the receiving system is at essentially atmospheric pressure.

 (d) Piping to the user is dead ended.

(6) There are no automatic valves in the MUWP System. During a LOCA, the safety-related systems are isolated from the MUWP System by automatic valves in the safety-related system.

(7) The outboard primary containment isolation valve is locked closed during standby, hot standby and power operation.

9.2.10.3 Safety Evaluation

Operation of the MUWP System is not required to assure any of the following conditions:

(1) Integrity of the reactor coolant pressure boundary.

(2) Capability to shut down the reactor and maintain it in a safe shutdown condition.

(3) Ability to prevent or mitigate the consequences of events which could result in potential offsite exposures.

The MUWP System is not safety-related. However, the systems incorporate features that assure reliable operation over the full range of normal plant operations.

9.2.10.4 Tests and Inspections

The MUWP System is proved operable by its use during normal plant operation. Portions of the system normally closed to flow can be tested to ensure operability and integrity of the system.

Flow to the various systems is balanced by means of manual valves at the individual takeoff points.

9.2.11 Reactor Building Cooling Water System

9.2.11.1 Design Bases

9.2.11.1.1 Safety Design Bases

(1) The Reactor Building Cooling Water (RCW) System shall be designed to remove heat from plant auxiliaries which are required for a safe reactor shutdown, as well as those auxiliaries whose operation is desired following a LOCA, but not essential to safe shutdown.

The heat removal capacity is based on the heat removal requirement during a LOCA with the maximum UHS temperature (37.8°C). As shown in Table 9.2-4a, the heat removal requirement is higher during other plant operation modes, such as shutdown at 4 hours. However, the RCW System is designed to remove this larger amount of heat to meet the requirements in Subsection 5.4.7.1.1.7.

(2) The RCW System shall be designed to perform its required cooling functions following a LOCA, assuming a single active or passive failure.

(3) The safety-related portions and valves isolating the non-safety-related portions of the RCW System shall be designed to Seismic Category I and the ASME Code, Section III, Class 3, Quality Assurance B, Quality Group C, IEEE-279 and IEEE-308 requirements.

(4) The RCW System shall be designed to limit leakage to the environment of radioactive contamination that may enter the RCW System from the RHR System.

(5) Safety-related portions of the RCW System shall be protected from flooding, spraying, steam impingement, pipe whip, jet forces, missiles, fire, and the effect of failure of any non-Seismic Category I equipment, as required.

(4) The system shall be powered from Class 1E buses. Power shall be available from the Alternate AC (AAC) power source when required.

(5) The HECW System shall be protected from missiles in accordance with Subsection 3.5.1.

(6) Design features to preclude the adverse effects of water hammer are in accordance with the SRP section addressing the resolution of USI A-1 discussed in NUREG-0927.

These features shall include:

(a) An elevated surge tank to keep the system filled.

(b) Vents provided at all high points in the system.

(c) After any system drainage, venting is assured by personnel training and procedures.

(d) System valves are slow acting.

(7) The HECW System shall be protected from failures of high and medium energy lines as discussed in Section 3.6.

(8) The design operation of the HECW compressors will take into account power or operational perturbations which could result in a) frequent immediate or elongated restarts, b) in unacceptable compressor coolant and lubrication oil interactions, and c) compressor coolant leaks or releases.

(9) The system piping design will take into account unacceptable nil-ductility-temperature conditions associated with normal and transient operation.

9.2.13.2 System Description

The HECW System consists of subsystems in three divisions. Division A has one refrigerator and pump, and Divisions B and C have two refrigerator units, two pumps, instrumentation and distribution piping and valves to corresponding cooling coils. A chemical addition tank is shared by all HECW divisions. Each HECW division shares a surge tank with the corresponding division of the RCW System. The refrigerator capacity is designed to cool the Reactor Building safety-related electrical equipment HVAC Systems and Control Building safety-related equipment area HVAC Systems.

The system is shown in Figure 9.2-3. The refrigerators are located in the Control Building as shown in Figures 1.2-20 and 1.2-21. Each refrigerator unit consists of a evaporator, a centrifugal compressor, refrigerant, piping, and package chiller controls. This system shares the RCW surge tanks which are in the Reactor Building (Figure 1.2-12). Equipment is listed in Table 9.2-8. Each cooling coil is controlled by a room thermostat. Alternately, flow may be controlled by a temperature control valve. Condenser cooling is from the corresponding division of the RCW System.

Piping and valves for the HECW System, as well as the cooling water lines from the RCW System, designed entirely to ASME Code, Section III, Class 3, Quality Group C, Quality Assurance B requirements. The extent of this classification is up to and including drainage block valves. There are no primary or secondary containment penetrations within the system. The HECW System is not expected to contain radioactivity.

High temperature of the returned cooling water causes the standby refrigerator unit to start automatically. Makeup water is supplied from the MUWP System, at the surge tank. Each surge tank has the capacity to replace system water losses for more than 100 days during an emergency. The only non-safety-related portions of the HECW divisions are the chemical addition tank and the piping from the tank to the safety-related valves which isolate the safety-related portions of the system.

Also, see Subsection 9.2.17.1 for COL license information requirements.

9.2.13.3 Safety Evaluation

The HECW System is a Seismic Category I system, protected from flooding and tornado missiles. All components of the system are designed to be operable during a loss of normal power by connection to the ESF buses (Tables 8.3-1 and 8.3-2). Redundant components are provided to ensure that any single component failure does not preclude system operation in Divisions B and C. The system is designed to meet the requirements of Criterion 19 of 10CFR50. The refrigerators of each division are in separate rooms.

During a Station Blackout (SBO), the HECW refrigerators, pumps and instrumentation will be powered by the AAC System which will become available in ten minutes. Provisions will be made to ensure prompt and reliable restart of the chiller units. COL license information requirements are provided in Subsection 9.2.17.1.

The response to SBO is discussed in Chapter 1, Appendix 1C. During the SBO, little heat will be generated in the areas cooled by HECW because only battery powered equipment will be operating. These areas are the main control room, the Control Building essential electrical equipment rooms and the Reactor Building essential electrical equipment rooms. The HVAC fans in these areas are powered by Class 1E buses. When AAC power becomes available, these fans will be powered and will start supplying outside air and exhausting any hot air from these areas. When chilled water becomes available, cooled air will be circulated in these areas to restore normal temperature.

If a LOPP event occurs, there are provisions for a stop signal to the HECW pumps to trip the breakers or for sequencing the HECW pumps back onto the emergency bus

9.2.16.1.3.2 Component Description

The TSW heat exchangers are shown on Figure 9.2-6a and are described in Subsection 9.2.14.2.

9.2.16.1.3.3 System Operation

The system is operated from the main control room.

9.2.16.1.4 Safety Evaluation

The TSW System is not interconnected with any safety-related system.

9.2.16.1.5 Instrumentation Application

Pressure and temperature indicators are provided where required for testing the system.

9.2.16.1.6 Tests and Inspections

All major components are tested and inspected as separate components prior to installation, and as an integrated system after installation to ensure design performance. The systems are preoperationally tested in accordance with the requirements of Chapter 14.

The components of the TSW System and associated instrumentation are accessible during plant operation for visual examination. Periodic inspections during normal operation are made to ensure operability and integrity of the system. Inspections include measurement of the TSW System flow, temperatures, pressures, differential pressures and valve positions to verify the system condition.

9.2.16.2 Portions Outside Scope of ABWR Standard Plant

All portions of the TSW System that are outside the Turbine Building are not in the scope of the ABWR Standard Plant. Subsections 9.2.16.2.1 and 9.2.16.2.2 provide a conceptual design of these portions of the TSW System as required by 10CFR52. The interface requirements for this system are part of the design certification.

The site-dependent portions of the TSW System shall meet all requirements in Subsections 9.2.16.1.1 through 9.2.16.1.5 and following requirements. This subsection provides a conceptual design and interface requirements for those portions of the TSW System which are site dependent and are a part of the design certification.

9.2.16.2.1 Safety Design Bases (Interface Requirement)

There are none.

9.2.16.2.2 Power Generation Design Bases (Interface Requirements)

The COL applicant shall provide the following system design features and additional information which are site dependent:

(1) The temperature increase and pressure drop across the heat exchangers.

(2) The required and available net positive suction head for the TSW pumps at pump suction locations considering anticipated low water levels.

(3) The location of the TSW pump house.

(4) The heat removal requirements from the TCW System are in Subsection 9.2.14.2.

(5) System low point drains and high point vents are provided as required. All components are maintained full of water (to prevent waterhammer) when not in service except when undergoing maintenance.

9.2.16.2.3 System Description

9.2.16.2.3.1 General Description (Conceptual Design)

The TSW System consists of three 50% capacity vertical wet pit pumps located at the intake structure. Two pumps are in operation during normal operation with one pump in standby.

The TSW pumps supply cooling water to the three TCW heat exchangers (two are normally in service and one is on standby).

9.2.16.2.3.2 Component Description (Conceptual Design)

Three strainers are provided (one for each TSW pump). Debris collected in the strainer is sluiced to a disposal collection area.

Piping and valves in the TSW System are protected from interior corrosion with suitable corrosion resistant material as required by site specific soil and water conditions.

9.2.16.2.3.3 System Operation (Conceptual Design)

The system is operated from the main control room.

The standby pump is started automatically in the event the normally operating pump trips or the discharge header pressure drops below a preset limit.

9.2.16.2.4 Safety Evaluation (Interface Requirements)

The COL applicant shall demonstrate that all safety-related components, systems and structures are protected from flooding in the event of a pipeline break in the TSW System.

9.2.16.2.5 Instrumentation and Alarms (Interface Requirements)

TSW System pump status shall be indicated in the main control room.

TSW System trip shall be alarmed and the automatic startup of the standby pump shall be annunciated in the main control room.

High differential pressure across the duplex filters shall be alarmed in the main control room.

9.2.16.2.6 Tests and Inspections (Interface Requirements)

All major components are tested and inspected as separate components prior to installation, and as an integrated system after installation to ensure design performance. The systems are preoperationally tested in accordance with the requirements of Chapter 14.

The components of the TSW System and associated instrumentation are accessible during plant operation for visual examination. Periodic inspections during normal operation are made to ensure operability and integrity of the system. Inspections include measurements of cooling water flows, temperatures, pressures, water quality, corrosion-erosion rate, control positions, and setpoints to verify the system condition.

9.2.17 COL License Information

9.2.17.1 HECW System Refrigerator Requirements

The COL applicant shall provide for the following after refrigerators have been procured:

(1) Means shall be provided for adjusting refrigerator capacity to chilled water outlet temperature.

(2) Means shall be provided for starting and stopping the pump and refrigerator on proper sequence.

(3) Means shall be provided for reacting to a loss of electrical power for periods up to two (2) hours and for automatic restarting of pumps and refrigerators, under the expected environmental conditions during station blackout when electrical power is restored.

(4) Means shall be provided to minimize the potential for coolant leakage or release into system or surrounding equipment environs.

(5) An evaluation of transient effects on starting and stopping or prolonged stoppage of the refrigeration/chiller units. Effects like high restart circuit draw downs on safety buses, coolant-oil interactions, degassing needs, coolant gas leakage or release in equipment areas along with flammability threats, synchronized refrigeration swapping.

Table 9.2-5b Reactor Building Cooling Water System Passive Failure Analysis

Single Active Failure	Analysis
Failure of any RCW System supply or return piping	Essential plant cooling requirements are met by the remaining intact RCW System, which includes their own independent supply and return service water headers. The redundant systems are mechanically and electrically separated to prevent damage to one system from the other systems.
Failure of RCW to RHR heat exchanger	Essential plant cooling requirements are met by the remaining intact redundant RHR System, which includes its own 100% capacity heat exchanger.
Failure of RCW piping to or from the air cooler for an ECCS pump area	Essential plant cooling requirements are met by the redundant ECCS which have their own independently cooled pump areas.
Failure of a single RCW heat exchanger during normal operation	Essential plant cooling requirements are met by the remaining operable, redundant heat exchanger.

Table 9.2-6 HVAC Normal Cooling Water System Component Description

HNCW Chillers	
Type	Centrifugal hermetic
Quantity	5 (including one standby unit)
Cooling Capacity	9.42 GJ/h each
Chilled water flow per unit	450 m^3/r
Supply temperature	7°C
Condenser water flow per unit	420 m^3/r
Supply temperature (max.)	45°C
Control	Inlet guide vane
Condenser	Shell and tube
Evaporator	Shell and tube
HNCW Water Pumps	
Quantity	5 (including one standby unit)
Type	Centrifugal, horizontal
Capacity m^3/hr each	450
Total discharge head	0.49 MPa

Table 9.2-7 HVAC Normal Cooling Water Loads

Name of Area or Unit	During Normal Operation Capacity GJ/h	During Normal Operation Flow m^3/h	During Refueling Shutdown Capacity GJ/h	During Refueling Shutdown Flow m^3/h
Reactor Building				
Drywell Coolers	0.96	69.5	0.80	69.5
RIP Coolers	1.59	20.9	3.06	104
Others (Note 1)	10.05	131	18.84	636
Turbine Building (Note 2)	2.26	43.5	1.13	39
Radwaste Building (Note 4)	5.69	81.2	6.70	232
Service Building	3.64	175	3.64	175
Others (Note 5)	4.61	151	3.56	151
Total	28.89	672	37.68 (Note 6)	1,407

NOTES:

(1) Loads include reactor/turbine building supply units, HVH, FCU and room coolers.

(2) Loads are the offgas cooler condenser (normal operation only) and the electrical equipment supply unit.

(3) Deleted

(4) Loads included are the radwaste building supply unit and the radwaste building electrical equipment room supply unit.

(5) Loads include HVH units not previously included.

(6) The HNCW chillers are 9.38 GJ/heach and the pumps 449m^3/h each. Thus, four HNCW pumps have total capacity in excess of the amount required as shown in the last column of the table.

Table 9.2-8 HECW System Component Description*

HECW Chillers		
Type		Centrifugal hermetic
Quantity		5
Capacity (Refrigerator)	five	2.51 GJ
Chilled water pump flow	five	57 m^3/h
Supply temperature		7°C
Condenser water flow	five	128 m^3/h
Supply temperature (max.)		45°C
Condenser	Shell and tube	
Evaporator	Shell and tube	
HECW Water Pumps		
Quantity		5-57 m^3/h each
Type		Centrifugal, horizontal

* Division A has one pump-refrigerator unit. Divisions B and C both have two pump-refrigerator units.

Table 9.2-9 HVAC Emergency Cooling Water System Heat Loads

		Normal		Emergency	
Division	System	Heat Load (GJ/h)	Chilled Water Flow (m^3/h)	Heat Load (GJ/h)	Chilled Water Flow (m^3/h)
A	Reactor Building Electrical Equipment Room (A)	0.88	14.3	0.88	14.3
	Control Building Electrical Equipment Room (A)	1.26	20.2	1.26	20.2
	Total	2.14	34.5	2.14	34.5
B	Main Control Room	1.42	26	1.30	24
	Reactor Building Electrical Equipment Room (B)	0.92	15	0.92	15
	Control Building Electrical Equipment Room (B)	1.26	20.2	1.26	20.2
	Total	3.60	61.2	3.48	59.2
C	Main Control Room	1.42	26	1.30	24
	Reactor Building Electrical Equipment Room (C)	0.92	15	0.92	15
	Control Building Electrical Equipment Room (C)	1.26	20.2	1.26	20.2
	Total	3.6	61.2	3.48	59.2

Table 9.2-10 HVAC Emergency Cooling Water System Active Failure Analysis

Failure of diesel generator to start or failure of all power to a single Class 1E power system bus.	Loss of one refrigerator and pump in Division B or C would not permit sending chilled water to the Control Room Habitability Area HVAC System from the affected division. The other HECW division would send chilled water to the Control Room Habitability Area HVAC System which would maintain adequate cooling. In Division A, loss of either the refrigerator or the pump would result in loss of cooling water flow to Division A Control Building safety-related Equipment Area HVAC System and Reactor Building safety-related Electrical Equipment HVAC System. Cooling of Control Room Habitability Area HVAC System not affected.
Failure of auto pump or refrigerator signal.	Same analysis as above.
Failure of a single HECW refrigerator.	Same analysis as above.
Failure of a single HECW pump.	Same analysis as above.
Failure of HECW pump and refrigerator room cooling.	Same analysis as above.

Table 9.2-11 Turbine Island Auxiliary Equipment

The TCW System removes heat from the following components:

- HVAC normal cooling water chillers
- Generator stator coolers, hydrogen coolers, seal oil coolers, exciter coolers and breaker coolers
- Turbine lube coolers
- Mechanical vacuum pump coolers
- Isophase bus coolers
- Electro-hydraulic control coolers
- Reactor feed pump and auxiliary coolers
- Standby reactor feed pump motor coolers
- Condensate pump motor coolers
- Heater drain pump motor coolers

Each subsystem consists of an ACU, two 100% capacity supply fans, and two 100% capacity exhaust fans. The ACU contains a medium efficiency bag filter section, and a cooling coil section.

The exhaust fans discharge to the atmosphere.

The C/B SREA HVAC system flow rates are given in Table 9.4-3, and system component descriptions are given in Table 9.4-4.

9.4.1.2.3.1 Safety-Related Subsystem Division A

Subsystem Division A specifically serves:

(1) Safety-related battery Division I

(2) HECW chiller Division A

(3) RCW water pump and heat exchanger Division A

(4) HVAC equipment Division A

(5) Safety-related electrical equipment Division I

(6) Non-safety-related power supplies

(7) Non-safety-related electrical equipment

9.4.1.2.3.2 Safety-Related Subsystem Division B

Subsystem Division B specifically serves:

(1) Safety-related battery Division II and Division IV

(2) HECW chiller Division B

(3) RCW pump and heat exchanger Division B

(4) HVAC equipment Division B

(5) Safety-related electrical equipment Division II and Division IV

Supply and exhaust ducts have fire dampers at firewall penetrations.

9.4.1.2.3.3 Safety-Related Subsystem Division C

Subsystem Division 3 specifically serves:

(1) Safety-related battery Division III

(2) HECW chiller Division C

(3) RCW water pump and heat exchanger Division C

(4) HVAC equipment Division C

(5) Safety-related electrical equipment Division III

(6) Non-safety-related MG sets at El. 7200 in C/B

9.4.1.2.4 Safety Evaluation

The safety-related equipment HVAC System is designed to ensure the operability of the safety-related equipment, and to limit the hydrogen concentration to less than 2% by volume in the battery rooms during system balancing to ensure the rooms exhaust the required air directly to the exhaust fans. All safety-related HVAC equipment and surrounding structures are of Seismic Category I design and are operable during loss of the offsite power supply.

The ductwork which serves these safety functions is termed ESF ductwork, and is of Seismic Category I design. ESF ducting is low-pressure safety grade ductwork designed to withstand the maximum positive and/or negative pressure to which it can be subjected under normal or abnormal conditions. Galvanized steel ASTM A526 or ASTM A527 is used for outdoor air intake and exhaust ducts. All other ducts are welded black steel ASTM A570, Grade A or Grade D. Ductwork and hangers are Seismic Category I. Bolted flange and welded joints are qualified per ERDA 76-21.

Redundant components are provided where necessary to ensure that a single failure will not preclude adequate environmental control.

9.4.1.2.5 Inspection and Testing Requirements

Provisions are made for periodic operational tests of the fans and filters.

The balance of the system is proven operable by its use during normal plant operation. Portions of the system normally closed to flow can be tested to ensure operability and integrity of the system.

9.4.1.2.6 Instrumentation Application

One of the two air conditioning unit supply fans is started manually for normal operation.

On an alarm of exhaust fan or supply fan failure, the standby fan is automatically started, and an alarm is sounded in the main control room, indicating fan failure.

Design inside air temperatures for the secondary containment during normal operation is 40°C maximum in the summer and 10°C minimum in the winter.

9.4.5.1.2 System Description

The Reactor Building secondary containment HVAC System P&ID is shown in Figure 9.4-3. The system flow rates are given in Table 9.4-3, and the system component thermal capacities are given in Table 9.4-4. The HVAC System is a once-through type. Outdoor air is filtered, tempered and delivered to the secondary containment. The supply air system consists of a medium grade bag filter, a heating coil, a cooling coil, and three 50% supply fans located in the Turbine Building. Two are normally operating and the other is on standby. The supply fan delivers conditioned air through ductwork and registers to the secondary containment equipment rooms and passages. The exhaust air system consists of 3 filters and 3-50% capacity fans to be located in the Turbine Building. The exhaust fans pull air from the secondary containment rooms through ductwork, and filters. Monitors measure radioactivity before it is exhausted from the plant stack. HVAC air supply and exhaust used by the ACS for primary containment deinerting is discussed in Subsection 6.2.5.2.1(14) and the shutdown mode of operation in Subsection 6.2.5.2(3). Electric unit heaters are located in the large component entrance building. Supply air is directed into the space when the interior doors are open.

9.4.5.1.3 Safety Evaluation

Operation of the Secondary Containment HVAC System is not a prerequisite to assurance of either of the following:

(1) Integrity of the reactor coolant pressure boundary.

(2) Capability to safely shut down the reactor and to maintain a safe shutdown condition.

However, the system does incorporate features that provide reliability over the full range of normal plant operation. The following signals automatically isolate the Secondary Containment HVAC System:

(1) Secondary containment high radiation signal (LDS)

(2) Refueling floor high radiation signal (LDS)

(3) Drywell pressure high signal (LDS)

(4) Reactor water level low signal (LDS)

(5) Secondary containment HVAC supply/exhaust fans stop

containment, the exhaust filter by-pass dampers are opened, standby exhaust and supply fans are started to provide an increase in airflow through the secondary containment. The divisions that are not on fire shall have their exhaust dampers closed to a partially closed position. This position shall be set during system setup. When the exhaust dampers are partially closed, the non-fire divisions' pressure will be maintained at a positive pressure. The division experiencing the fire will be maintained more negative with respect to the non-fire divisions.

Fire zone dampers can isolate the division with the fire until smoke removal is required. When fire doors are opened between divisions, the air pressure in the non-fire zones will limit smoke intrusion. Fire dampers with fusible links in HVAC ductwork will close under airflow conditions after fusible link melts.

9.4.5.1.4 Inspection and Testing Requirements

The system is designed to permit periodic inspection of important components, such as fans, motors, belts, coils, filters, ductwork, dampers, piping and valves, to assure the integrity and capability of the system. Standby components can be tested periodically to ensure system availability.

All major components are tested and inspected as separate components prior to installation and as integrated systems after installation, to ensure design performance. The system is preoperationally tested in accordance with the requirements of Chapter 14.

9.4.5.1.5 Instrumentation Application

The Secondary Containment HVAC System is started manually. Fan inlet dampers are interlocked to open before the fan is started. A flow switch installed in the operating fans discharge ductwork automatically starts the standby fan on indication of any operating fan failure due to a reduction in air.

The pneumatically-operated secondary containment inboard and outboard isolation dampers fail to the closed position in the event of loss of pneumatic pressure or loss of electrical power to the valve actuating solenoids. Upon receiving a leak detection system signal (Subsection 9.4.5.1.3), the isolation dampers automatically close, supply and exhaust fans stop, and a start signal calls for automatic SGTS operation. The supply fans and exhaust fans are interlocked to prevent operation of the supply fans when the exhaust fans are shut down.

All major components are tested and inspected as separate components prior to installation to ensure design performance. The system is preoperationally tested in accordance with the requirements of Chapter 14.

9.4.5.3.5 Instrumentation Application

The R/B Non-safety-related Equipment HVAC System starts manually.

9.4.5.4 R/B Safety-Related Electrical Equipment HVAC System

9.4.5.4.1 Design Bases

9.4.5.4.1.1 Safety Design Bases

The R/B Safety-Related Electrical Equipment HVAC System is designed to provide a controlled temperature environment to ensure the continued operation of safety-related equipment under accident conditions. The rooms cooled by the R/B Safety-Related Electrical Equipment HVAC System are maintained at positive pressure relative to atmosphere during normal and accident conditions. This is achieved by sizing intake fans larger than exhaust fans.

The power supplies to the HVAC systems for the R/B safety-related electrical equipment rooms allow uninterrupted operation in the event of loss of normal offsite power.

The system and components are located in a Seismic Category I structure that are tornado-missile, and flood protected, including tornado missile barriers on intake and exhaust structures.

For compliance with code standards and regulatory guides, see Sections 3.2 and 1.8.

On a smoke alarm in a division of the Reactor Building Safety-Related Electrical Equipment HVAC System, that division of the HVAC System shall be put into smoke removal mode manually. No other division is affected by this action. For smoke removal, the recirculation damper is closed, the exhaust fan bypass damper opened, and the exhaust fan is stopped. Normal once through ventilation of the day tank rooms also removes smoke from the day tank rooms.

The intake louvers are located at 15.2m above grade. The exhaust louvers are located at 13.3m above grade. (See general arrangement layout, Figures 1.2-10 and 1.2-11.)

9.4.5.4.1.2 Power Generation Design Bases

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

The system design is based on outdoor summer conditions of 46°C and outdoor winter conditions of –40°C. The indoor design temperature in the safety-related electrical equipment areas is 40°C maximum in the summer and a minimum of 10°C in the winter except 45°C in the diesel generator (DG) engine rooms during DG operation. The system along with the DG supply fan maintain DG room temperature below 45°C.

9.4.5.4.2 System Description

Divisions A, B, and C Safety-Related Electrical Equipment HVAC Systems are independent, physically separated, and functionally identical except for their power bus designations and divisional source of cooling water. The HVAC System for each division of safety-related electrical equipment consists of two 100% capacity supply fans, two 100% capacity exhaust fans, and one air conditioning unit. Each air conditioning unit consists of a medium grade filter and a cooling coil. (See Figure 9.4-4 for the system P&ID. See Table 9.4-4 for the component descriptions.) The following divisional rooms are cooled by the Safety-Related Electrical Equipment HVAC System.

(1) Day tank room, Divisions A, B, C

(2) Diesel generator engine room, Divisions A, B, C

(3) Non-safety-related reactor internal pump ASD rooms

(4) Electrical equipment room, Divisions I, II, III, IV

(5) HVAC equipment room, Divisions A, B, C

(6) Remote shutdown panel room, Divisions A, B

(7) Diesel generator MCC area, Divisions A, B, C

(8) Non-Safety-Related FMCRD control panel rooms

HVAC System Division A serves electrical Division I, Division B serves electrical Divisions II and IV, and Division C serves electrical Division III of the electrical equipment rooms. Also, non-safety-related reactor internal pumps ASD rooms are cooled by the Electrical Equipment HVAC System.

9.4.5.4.3 Safety Evaluation

All safety-related equipment is located in a Seismic Category I structure that is tornado-missile, and flood protected. All HVAC equipment is designed to Engineered Safety Feature requirements.

9.4.5.4.4 Inspection and Testing Requirements

The systems are designed to permit periodic inspection of important components, such as fans, motors, coils, filters, ductwork, dampers, piping, and valves to assure the integrity and capability of the system. Standby components can be tested periodically to ensure system availability.

(5) The clean areas served by the clean area HVAC system has an emergency filter train. It is manually operated. In an emergency it supplies filtered air for the TSC, OSC, lunch room, offices, health physics lab, security offices, and other normally clean areas.

9.4.8.2 System Description

(1) The Clean Area HVAC System supplies filtered, heated or cooled air to both the clean and controlled areas through a central fan system consisting of an outside air intake, Air Conditioning Unit consisting of filters, heating coils, cooling coils, two 50% capacity supply air fans and supply air ductwork.

(2) The Clean Area HVAC System has two 50% capacity exhaust air fans. They take air from the clean areas through the exhaust ducts and discharge the air on the Service Building roof.

(3) The Controlled Area HVAC System routes potentially contaminated air to two 50% capacity exhaust air fans to discharge the air to the common plant stack.

(4) The potentially contaminated areas are maintained at a slightly lower pressure than the surrounding clean areas and, therefore, the air flows from the clean areas to these potentially contaminated areas.

(5) Pressure control dampers are employed between clean and potentially contaminated areas and are of the backflow type and fail closed. This minimizes the backflow of contaminated air to clean areas when there is a loss of power and subsequent fan system shutdown.

(6) The clean area HVAC system is provided with an emergency filter train consisting of a heater/demister, prefilter, HEPA filter, 5.1 cm charcoal filter bed, a second HEPA filter, and two fans.

(7) Controls and Instrumentation

(a) Each fan and each exhaust filter package is controlled by hand switches located on local control panels. Pertinent system flow rates and temperatures are also indicated on the local control panels. Trouble on local control panel is annunciated on the main control board.

(b) Controls are pneumatic and electric.

(c) Radiation monitors and provisions for toxic gas monitors at the supply air inlet with alarms to TSC.

(d) On manual initiation, the clean area HVAC system can be put into high radiation mode. On switch over, exhaust fans stop and emergency filter train starts. System pressurizes clean areas of the service building.

(e) Instrumentation is provided for the monitoring system operating variable during normal station operating conditions. The loss of airflow, high and low system temperature, and high differential pressure across various filters are annunciated on the local control panel. Trouble on the local panel is annunciated in the main control room.

(8) All power and water is provided from non-safety-related sources.

(9) The COL applicant will provide a detailed P&ID, system flow rates an equipment list, and compliance with RG 1.140 and toxic gas protection requirements and description of radiation monitors (if any) at the supply air inlet, for the Service Building HVAC system, including the TSC and the OSC, for NRC review. (See Subsection 9.4.10.1 for COL License Information.)

9.4.8.3 Safety Evaluation

(1) The Service Building HVAC System is not safety-related and is not required to assure either the integrity of the reactor coolant pressure boundary or the capability to shut down the reactor and maintain it in a safe shutdown conditions.

(2) Pressure control dampers are employed between clean and potentially contaminated areas and are of the backflow type and fail closed. This minimizes the backflow of contaminated air to clean areas when there is a loss of power and subsequent fan system shutdown.

(3) The system incorporates features to assure its reliable operation over the full range of normal station conditions.

(4) Clean areas are provided with emergency filtration system and a high radiation mode of operation.

(5) There are no sources (except health physics samples and calibration sources) of radioactivity inside the Service Building. However, the radiation levels inside the controlled area of the Service Building can become high due to leakage from the secondary containment or from the Turbine Building. If this happens, the controlled area HVAC system can be manually isolated to prevent releases to the environment via the subject HVAC system exhaust.

9.4.8.4 Testing and Inspection

All equipment is factory inspected and tested in accordance with the applicable equipment specifications and codes. System ductwork and erection of equipment is inspected during various construction stages. Preoperational tests are performed on all mechanical components and the system is balanced for the design air, and water flows and system operating pressures. Controls, interlocks and safety devices on each system are checked, adjusted, and tested to ensure the proper sequence of operation. A final integrated preoperational test is conducted with all equipment and controls operational to verify the system performance.

Maintenance will be performed on a scheduled basis in accordance with the equipment manufacturer's requirements.

The system is in operation during normal plant operation.

9.4.9 Drywell Cooling System

9.4.9.1 Design Bases

The Drywell Cooling System shall have the capability to maintain the drywell temperature, during normal operation, at temperatures specified in Section 3.11.

The Drywell Cooling System shall be capable of controlling the temperature rise of the drywell during normal operational transients so that the average drywell temperature does not exceed 58°C. The local temperature shall not exceed 75°C in the CRD area or 66°C elsewhere in the drywell.

The Drywell Cooling System is designed to provide sufficient air/nitrogen distribution so that proper temperature distribution can be achieved to prevent hot spots from occurring in any area of the drywell.

9.4.9.2 System Description

See Figures 9.4-8 and 9.4-9 for flow diagrams illustrating the drywell cooling system, and Table 9.4-1 for a listing of its components.The Drywell Cooling System is a recirculating system consisting of three fan coil units. Normally, two of the three fan coil units are in operation. Each fan coil unit consists of cooling coils, a drain pan, and a centrifugal fan. Cooling water comes from the RCW and HNCW Systems. Two sets of cooling coils are arranged in series. The return air passes over the first coil, which is cooled by the RCW System. Part of the cooled air is then cooled by the second coil, which is cooled by the HNCW System. This twice-cooled air is mixed with the air that bypasses the second cooling coil. Condensate that drips from the coils is routed to the drain system via the Leak Detection System. Instrumentation is installed in front of the Leak Detection System connection that monitors cooler condensate flow.

The Drywell Cooling System supplies conditioned air to a common distribution header. The air/nitrogen is then ducted to areas within the drywell for equipment cooling. These areas consist of the drywell head area, upper drywell, lower drywell, and reactor shield wall annulus. The Drywell Cooling System heat loads are provided in Table 9.4-2.

Gravity dampers and adjustable balancing dampers control distribution of the air/nitrogen to the various drywell spaces.

High drywell temperatures are alarmed in the main control room, alerting the operator to take appropriate corrective action. During normal plant operation, two fan coil units are operated. During LOPP (when no LOCA signal exists), fan coil units shall restart automatically when power is available from the combustion turbine generator. During a LOPP, chilled water from the HNCW System may or may not be available, but cooling should always be available from the RCW coils. The drywell fan coil units are not operated during a LOCA.

9.4.9.3 Safety Evaluation

Operation of the Drywell Cooling System is not a prerequisite to assurance of either one of the following:

(1) Integrity of the reactor coolant pressure boundary

(2) Capability to safely shut down the reactor and to maintain a safe shutdown condition

However, the system does incorporate features that provide reliability over the full range of normal plant operation. These features include the installation of redundant principal system components such as:

(1) Electric power

(2) Fan coil units

(3) Redundant chillers

(4) Ductwork

(5) Controls

(6) Cross-connection of all fan coil units

9.4.9.4 Inspection and Testing Requirements

Equipment design includes provisions for periodic testing of functional performance and inspection for system reliability. Standby components are fitted with test connections so that system effectiveness, except for airflow or static pressure, can be verified without the units being online. Test connections are provided in the discharge air ducts for verifying calibration of the operating controls.

9.4.9.5 Instrumentation Applications

Drywell cooling unit function is manually controlled from the main control room. The instrumentation which monitors system performance is part of the Atmospheric Control System and the Leak Detection and Isolation Systems.

9.4.10 COL License Information

9.4.10.1 Service Building HVAC System

The COL applicant shall provide a detailed P&ID, system flow rates and an equipment list, compliance with RG 1.140, toxic gas protection requirements, and description of radiation monitors at the supply air inlet (if any), for the Service Building HVAC system, including the TSC and OSC, for NRC review. (Subsection 9.4.8.2)

9.4.10.2 Radwaste Building HVAC System

The COL applicant shall supply detailed equipment lists and system flow rates and compliance with RG 1.140 for the Radwaste Building HVAC System (Subsection 9.4.6.2).

Table 9.4-1 Drywell Cooling System Non-Safety-Related Components

RCW Cooling Coils	
Number	3
Type	Plate Fin
Airflow Rate	1000 m^3/min.
Cooling Capacity	1023.42 MJ/h
Air Temperature (Inlet/Outlet)	57°C/42°C
Water Temperature (Inlet/Outlet)	35°C/40°C
Water Flow Rate	13.5 L/sec
HNCW Cooling Coils	
Number	2
Type	Plate Fin
Air Flow Rate	277 m^3/min.
Cooling Capacity	791.31 MJ/h
Air Temperature (Inlet/Outlet)	44°C/12°C
Water Temperature (Inlet/Outlet)	7°C/12°C
Water Flow Rate	10.5 L/sec
Fans	
Number	3
Type	Centrifugal
Capacity	1000 m^3/min.
Head	1.47E+03 Pa

Table 9.4-2 Drywell Cooling System Non-Safety-Related Heat Loads

Heat Loads		Normal Plant Operation Sensible Heat Load MJ/h
Sensible Heat Loads	Drywell Head Area	146.5
	Upper Drywell	837.4
	Lower Drywell	180
	Shield Wall Annulus	782.9
	Upper Drywell Piping Area	1067.6
Equipment	Fan Motors	33.5
	Heatup Load of Fans	293.1
Sensible Heat Load (Total)		3341*
Latent Heat Load		297.3
Design Heat Load		3638.3

* The sensible heat load during plant maintenance mode is about 460.5 MJ/h.

Table 9.4-3 HVAC Flow Rates (Response to Question 430.243)

Safety-Related HVAC System	Flow Rates (m^3/h)
R/B Electrical HVAC Division A	30,000
R/B Electrical HVAC Division B	30,000
R/B Electrical HVAC Division C	30,000
DG HVAC Division A	160,000
DG HVAC Division B	160,000
DG HVAC Division C	160,000
C/B Electrical HVAC Division A	35,000
C/B Electrical HVAC Division B	35,000
C/B Electrical HVAC Division C	35,000
CRHA HVAC Division B	80,000
CRHA HVAC Division C	80,000
Non-Safety-Related HVAC Systems	**Flow Rates (m^3/h)**
R/B Secondary Containment HVAC	168,500
T/B Ventilation System	341,500
RIP ASD HVAC Division A	50,000
RIP ASD HVAC Division B	50,000
Radwaste Building HVAC*	
Service Building HVAC*	

* The COL applicant shall supply these flow rates. See COL Subsection 9.4.10.1 for the Service Building and 9.4.10.2 for the Radwaste Building.

Table 9.4-4f HVAC System Component Descriptions — Non-Safety-Related Heating/Cooling Coils (Response to Question 430.243)

Heating/Cooling Coils	Quantity	Cooling (MJ/h)	Heating (MJ/h)
R/B Secondary Containment HVAC	(1 bank)	6435.95	9601.17
RIP ASD HVAC Division A	1	2110.15	
RIP ASD HVAC Division B	1	2110.15	

Table 9.4-4g HVAC System Component Descriptions — Non-Safety-Related Fans (Response to Question 430.243)

Fans	Quantity	Capacity (m^3/h)
R/B Secondary Containment Supply Fans	3 (1 on standby)	84,250
R/B Secondary Containment Exhaust Fans	3 (1 on standby)	86,250
R/B Primary Containment Supply Fan	1	22,000
R/B Primary Containment Exhaust Fan	1	22,000
RIP ASD Division A Supply Fans	2 (1 on standby)	50,000
RIP ASD Division B Supply Fans	2 (1 on standby)	50,000

Table 9.4-4h HVAC System Component Descriptions — Non-Safety-Related Filters (Response to Question 430.243)

Filters	Quantity	Capacity (m^3/h)
R/B Secondary Containment HVAC	(1 bank)	172,500
R/B Primary Containment Intake HEPA Filter	1	22,000
R/B Secondary Containment Exhaust Fans	3	57,500 (each)

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

Non–Safety-Related Fan Coil Units	Quantity	Capacity (MJ/h)
Main Steam Tunnel	2	628.02
Refueling Machine Control Room	1	83.74
ISI Room	2	54.43
MG Set Room	2	1047.96
C/B Non-Safety-Related Electric Room	1	211.01
R/B FPC Room	2	28.47
CRD Replacement Room	1	18.42
RIP/CRD/FMCRD Repair Area	2	18.42
PCV L/T Measurement Room	1	0.042
Plant Stack Monitor Room	1	0.046
R/B SPCU Room	1	42.29

* The COL applicant shall supply equipment lists for the Service Building HVAC and the Radwaste Building HVAC System. See Subsection 9.4.10.1 for the Service Building, and 9.4.10.2 for the Radwaste Building.

Table 9.4-5 Turbine Building and Electrical Building HVAC System—Non-Safety-Related Equipment *

Item	Turbine Building Air Supply TRV–F–1A thru C	T/B Clean Area Return/Exhaust TBV–F–2A thru C	T/B Equipment Compartment Exhaust TBV–F–24A &–24B	T/B Lube Oil Exhaust TBV–F–4A & B	Condensate Pump Room Recirc. Unit TBV–F–8A thru C
Type	Builtup unit	Central station air handle	Builtup unit	Fan	Central station air handler
Number of units	1	3	1	2	3-50% each
Flow rate (m^3/h)	341,500	168,000/unit	272,000	12,600	51,000
Fan:					
Type	Centrifugal	Centrifugal	Centrifugal	Centrifugal	Centrifugal
No. of fans per unit	3	1	2	1	1
No. of running fans	2	2	1	1	2
Heating coils:		None		None	None
No. of banks per unit	1	–	1		–
Capacity, each (MJ/h)	11,605.81	–	369.28		–
Cooling coils:		None	None	None	
No. of banks per unit	6	–	–	–	
Capacity, each (MJ/h)	1582.61	–	–	–	949.57
Prefilters:		None	None	None	
Type	Glass, roll	–	–		–
Capacity (m^3/h)	356,800	–	–		–
ASHRAE 52 eff.	35%	–	–		–
Filters:				None	
Type	High eff.	Bag type,	Bag type,	–	Medium eff
Capacity (m^3/h)	341,500	168,000/unit	272,000	–	51,000
ASHRAE 52 eff.	85%	90%	90%	–	85%

* Response to Question 430.242C.

Table 9.4-5a Turbine Building and Electrical Building HVAC Systems—Non-Safety-Related Equipment (Continued)

Item	Heater Drain Pump P1A Room Recirc. Unit TBV–F–9A thru C	Heater Drain Pump P1B Room Recirc. Unit TBV–F–9D thru F	Filter Pump Recirc. and Valve Room Unit TBV–F–10A thru C	Demineralizer Pump and Valve Room Recirc. Unit TBV–F–12A thru C	Reactor Feed Pump Power Supply Room Recirc. Unit TBV–F–13A thru C
Type	Central station air handler	Central station air handler	Central station air handler	Central station air handler	Central station air handler
Number of units	3-50% each	3-50% each	3-50% each	3-50%	3-50%
Flow rate (m^3/h)/unit	11,900	11,900	5,200	8,700	1,825
Fan:					
Type	Centrifugal	Centrifugal	Centrifugal	Centrifugal	Centrifugal
No. of fans per unit	1	1	1	1	1
No. of running fans	2	2	2	2	2
Heating coils:	None:	None:	None:	None:	None
No. of banks per unit	–	–	–	–	–
Capacity, each (MJ/h)	–	–	–	–	–
Cooling coils:					
Capacity, each (MJ/h)	221.57	221.57	97.13	335.36	34.33
Filters:					
Type	Medium eff	Medium eff	Medium eff	Medium eff	Medium eff
Capacity (m^3/h)	11,900	11,900	5,200	8,700	1,825
ASHRAE 52 eff.	85%	85%	85%	85%	85%

Table 9.4-5b Turbine Building and Electrical Building HVAC Systems—Non-Safety-Related Equipment (Continued)

Items	TCW Heat Exchanger Area Recirculation Unit TBV–F–14A thru C	Condenser Compt. Level 2 Recirculation Unit TBV–F–15A thru C	SJAE A Room Recirculation Unit TBV–F–17A thru C	SJAE B Room Recirculation Unit TBV–F–17D thru F	Demineralizer Room Recirculation Unit TBV–F–18A thru C
Type	Central station air handler	Central station air handler	Central station air handler	Central station air handler	Central station air handler
Number of units	3-50% each	3-50% each	3-50% each	3-50%	3-50%
Flow rate (m^3/h)/unit	8,200	24,300	22,100	22,100	2,635
Fan:					
Type	Centrifugal	Centrifugal	Centrifugal	Centrifugal	Centrifugal
No. of fans per unit	1	1	1	1	1
No. of running fans	2	2	2	2	2
Heating coils:	None	None	None	None	None
No. of banks per unit	–	–	–	–	–
Capacity, each (MJ/h)	–	–	–	–	–
Cooling coils:					
Capacity, each (MJ/h)	154.07	454.69	417.84	417.84	48.99
Filters:					
Type	Medium eff	Medium eff	Medium eff	Medium eff	Medium eff
Capacity (m^3/h)	8,200	24,300	22,100	22,100	2,635
ASHRAE 52 eff.	85%	85%	85%	85%	85%

Table 9.4-5c Turbine Building and Electrical Building HVAC System—Non-Safety-Related Equipment (Continued)

Item	Condenser Compt. Level 3 Recir. Unit TRV-F-19A thru C	TCW Pump Area Recirculation Unit TRV-F-20A thru C	Turbine Area Recirculation Unit TRV-F-21A thru C	Steam to Hot Water Heat Exchanger Area TBV-E-01A& 1B
Type	Central station air handler	Central station air handler	Central station air handler	Heat exchanger
Number of units	3-50% each	3-50% each	3-50%	2-100% each
Flow rate (m^3/h)/unit	23,800	11,900	28,900	
Fan:				
Type	Centrifugal	Centrifugal	Centrifugal	–
No. of fans per unit	1	1	1	–
No. of running fans	2	2	2	–
Heating coils:	None	None	None	–
No. of banks per unit	–	–	–	–
Capacity, each (MJ/h)	–	–	–	–
Cooling coils:				
Capacity, each (MJ/h)	444.22	221.48	542.19	–
Filters:				
Type	Medium eff	Medium eff	Medium eff	–
Capacity (m^3/h)	23,800	11,800	28,900	–
ASHRAE 52 eff.	85%	85%	85%	–
Heat Exchanger:				
Type				Shell and Tube
Capacity (MJ/h)				22,156.55

The following figures are located in Chapter 21:

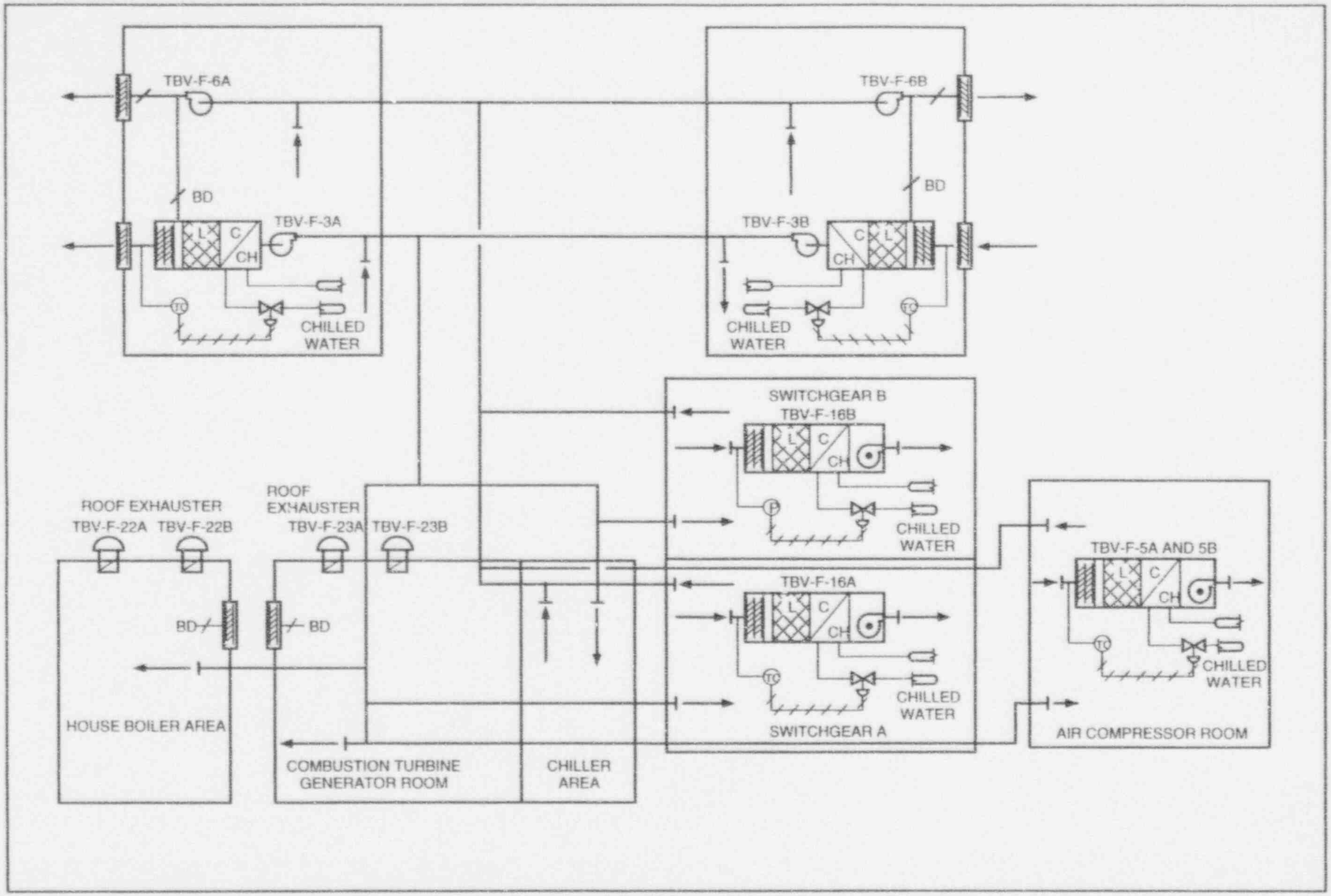


Figure 9.4-2c Electrical Building HVAC System Diagram

meets this requirement, in that it is planned that smoke will be removed by normal HVAC System. In the Reactor Building, the normal supply and exhaust fans are located external to the building. Every room of the Reactor Building secondary containment receives supply air from and exhausts to the building normal HVAC. The emergency ventilation systems for the electrical equipment and diesel generator rooms provide additional smoke removal capability for those rooms.

There is a containment vent and supply system. Neither the supply or exhaust ducts are equipped with fire dampers. The isolation valves on these ducts are normally closed and would remain closed during plant operation so as to maintain the containment in an inerted condition. If a fire occurred in containment during a plant outage, when the valves were open and the containment not inerted, the drywell or wetwell spray would be initiated to protect the containment at a temperature well below the threshold of damage to the ventilation duct. For these reasons, the ABWR design for the containment ventilation is considered proper and adequate. (Further discussed in Subsection 9.5.1.3.12).

The water suppression systems are designed on the basis that, following a safe shutdown earthquake, there will be two manual hose streams available in any area containing equipment required for safe shutdown and that there will be no uncontrolled release of fire suppression water in the areas.

Transformers located within fire areas containing safety-related equipment will be of the drytype only. For those areas utilizing liquid insulated transformers, the COL applicant shall provide features to prevent the insulating liquid from becoming an unacceptable health hazard to employees in the event of release of the material to the building environment.

The quality assurance (QA) program, in accordance with CMEB 9.5.1 for the design of fire protection systems, is presented in Chapter 17.

The consequences of inadvertent operation of a suppression system and of moderate energy line cracks are discussed in Appendix 9A.

Except for fuel and lubricating oil located in the diesel-generator rooms, there are no storage areas in the Reactor or Control Buildings for flammable liquids, oxidizing agents, flammable compressed gases, corrosive material or explosive or highly flammable materials. Nonflammable compressed gases (e.g., air, nitrogen) do not represent a fire hazard.

Small quantities of chemicals may be stored in listed or approved cabinets and containers for immediate use. The CRD maintenance area is an example where such storage is permitted. Identification of the type and location of these materials is a requirement of SRP Section 13.2.2, which is the responsibility of the COL applicant.

9.5.1.3 System Descriptions

9.5.1.3.1 General Description

The Fire Protection System consists of:

(1) Standpipe

(2) Hose stations

(3) Sprinklers

(4) AFFF sprinkler systems

(5) Automatic foam sprinkling systems

(6) Smoke detectors

(7) Alarms

(8) Fire barriers

(9) Fire stops

(10) Portable fire extinguishers

(11) Portable breathing apparatus

(12) Smoke and heat ventilation systems

(13) Associated controls and appurtenances

The suppression systems for the buildings and the plant yard are shown in the following figures:

Area	Figures
Reactor Building	9A.4-1 thru 9A.4-10
Control Building	9A.4-11 thru 9A.4-16
Turbine Building	9A.4-17 thru 9A.4-21
Service Building	9A.4-22 thru 9A.4-27
Radwaste Building	9A.4-28 thru 9A.4-32
Plant Yard	9.5-5

Lighting and other equipment maintenance, in addition to the safety of personnel, plant equipment, and plant operation, is considered in the design. Areas containing flammable materials (e.g., battery rooms, fuel tanks) have explosion proof lighting systems. Areas subject to high moisture have water-proof installations (e.g., drywell, washdown areas). Plant AC lighting systems are generally of the fluorescent type, with mercury lamps (or equivalent) provided for high ceiling, except where breakage could introduce mercury into the reactor coolant system. Incandescent lamps are used for DC lighting systems and above the reactor, fuel pools, and other areas where lamp breakage could introduce mercury into the reactor coolant.

Lighting systems and their distribution panels and cables are identified according to their essentiality and type. Associated and Class 1E lighting systems are located in Seismic Category I structures, and are electrically independent and physically separated in accordance with assigned divisions. Cables are routed in their respective divisional raceways. Normal lighting is separated from standby lighting. DC lighting cables are not routed with any other cables and are distinguished by "DCL" markings superimposed on the color markings at the same intervals.

Plant service buses supply power and heavy duty service outlets to equipment not generally used during normal plant power operation (e.g. Turbine Building and refueling floor cranes, welding equipment). Service outlets have grounded connections and the outlets in wet or moist areas are supplied from breakers with ground current detection.

9.5.3.1 Design Bases

9.5.3.1.1 General Design Bases

The general design bases for the Nuclear Island portion of the lighting systems are as follows:

(1) The lighting guidelines shall be based on Illuminating Engineering Society (IES) recommended intensities. These shall be inservice values as shown on Table 9.5-1, Illumination Levels. Reflected glare will be minimized.

(2) Control room lighting is designed with respect to reduction of glare and shadows on the control boards.

(3) Lighting systems and components are in conformity with the electrical standards of NFPA and OSHA as applicable for safety of personnel, plant and equipment. Light fixtures in safety areas are seismically supported, and are designed with appropriate grids or diffusers, such that broken material will be contained and will not become a hazard to personnel or safety equipment during or following a seismic event.

(4) Each of the normal, standby or emergency lighting systems has the following arrangement criteria:

(a) Areas without doors and hatches (where access is impossible) have no lighting.

(b) Rooms with normal (non-safety-related) lighting shall have on/off switches if the rooms are also used as passageways (e.g., patrol routes).

(c) For high radiation areas, the on/off switches and lamps shall be arranged to facilitate maintenance and to obtain maximum service life from the lamps.

(d) The switches shall be located at the entrance to the rooms, or the side of the passageway.

(e) Normal lighting power for the small rooms with on/off switches shall be supplied from one power bus.

NOTE: A small room means a room with three or less lighting fixtures, except for instrument rack rooms and electrical panel rooms.

(f) DC emergency lighting and associated lighting have no switch on their power supply lines.

(g) Standby lighting shall have no switch on power supply lines, as a rule. However, lighting for conference rooms etc., will have on/off switches.

(h) Power of inner panel lighting and outlets are supplied from one power bus.

(i) Each part of the 120V, 240V and 120/ 240V buses in lighting distribution panels shall have two or three spare circuits.

(j) Installation of fixtures on a high ceiling shall be avoided as far as possible to minimize lamp replacement work.

(k) The fixtures shall be located with due consideration of maintenance and inspection for the equipment in the rooms (such as tank rooms) where a well balanced arrangement is difficult.

(l) For mercury lamps, ballasts can be installed separately for life extension under the defined environment.

(m) The standard installation interval of service power supply boxes should be 45.7 - 61.0 m.

(n) The standard installation interval of outlets should be 15.2 - 30.5 m; however, outlets shall be arranged around instrument racks. The outlet installation level in hazard control areas shall be above the top of dikes.

(o) As a rule, normal lighting power shall be supplied with two power buses. However, a power supply with one power bus can be used for areas with high illumination lighting by standby lighting and in small rooms.

(p) Lighting shall be designed with due consideration of reflection on the CRT screens where CRTs are installed.

(q) Lighting fixtures in rooms with glass windows shall be arranged with due consideration of the mirror effect to keep the window clear.

(r) Power for staircase and passage lighting is from the standby system and shall be supplied from two power buses in the staircases and passages to prevent a total lighting loss. Each bus supplies power to 50% of the standby lighting of the passages and staircases. The two power buses for safety-related area passages and staircases shall consist of the following:

(i) One Class 1E bus (the same division as the safety-related equipment in the area), which is backed by its respective divisional diesel generator. A non-Class 1E bus, which is backed by the combustion turbine generator.

(ii) Under annual inspection of the power supplies, 50% lighting is secured with one lighting power bus. The 50% lighting level shall be sufficient for access and egress of personnel to and from the areas.

(5) Lighting fixtures shall be selected in accordance with the following criteria:

(a) Lighting fixtures inside the plant shall be the following type of fixtures:

(i) **Fluorescent lamps**—As a rule, fluorescent lamps shall be selected as fixtures for the general area.

(ii) **Mercury lamps**—Mercury lamps (or equivalent) shall be selected as fixtures for high ceiling areas (except in reactor building or other areas where lamp breakage could introduce mercury into the reactor coolant).

(iii) **Incandescent lamps**—Incandescent lamps shall be selected as fixtures for DC emergency lighting and as fixtures above the reactors and fuel pool in R/B operating floor.

(b) Standby lighting shall be the rapid start type.

(c) Incandescent lamps shall have waterproof guards inside drywell.

(d) The fixtures can be a general industry type; however, the fixtures for the part of service area in S/B and control rooms shall match the interior finish of the area.

(e) Lighting fixtures above operator consoles, benchboards and RW operator consoles shall be dark green embedded louver lighting to reduce the reflection of fixtures on CRT screens. Illumination levels around the operator console and benchboards shall be adjustable.

(f) Non-Class 1E battery pack lamps shall be self contained units suitable for the environment in which they are located.

(g) The light fixtures for Class 1E battery packs may be located remote from the battery if the environment at the lamp is not within the qualified range of the battery. Alternatively, lamps powered from the station batteries may be provided.

(h) Outlets shall have grounded connections and should be 120V-15A type or 240V-15A type.

(i) Standard service power boxes shall be 3-phase 480V-100A type.

(j) Lighting around the reactor and fuel pool on the R/B operating floor shall be designed with due consideration of the reflection on the water surface to keep from impeding pool work. Lamps located where they may drop in the reactor or fuel pool, shall have guards.

(k) Outdoor lamps shall have automatic on/off switches.

(l) Associated lighting equipment shall be selected for the following areas. Wiring shall be an explosion-proof type.

- (i) Batch oil tank room such as turbine oil tank room and lubrication oil tank room
- (ii) EHC equipments room
- (iii) Battery rooms
- (iv) Diesel generator rooms
- (v) Day tank rooms
- (vi) Hydrogen related panels and seal oil equipment area

(m) Lighting inside the cask cleaning pit shall be an embedded waterproof type fixture

Notes for Figure 9.5-4:

(1) The equivalent of one 100% capacity motor-driven pump and one 100% capacity diesel-driven pump shall be provided. the equivalent capacity of each type may be comprised of multiple pumps of that type.

(2) The motor-driven pumps shall be supplied power from the non-Class 1E busses.

(3) The following specific requirements apply to the components within the phantom box:

  (a) They shall be designed to remain functional following a safe shutdown earthquake.

  (b) The piping and valves as a minimum shall satisfy the requirements of ANSI B31.1.

(4) Each 100% capacity pumping unit and its controls shall be separated from the other pumping unit/units by a fire wall with a minimum rating of 3 hours.

(5) Alarms indicating pump running, driver availability, failure to start and low fire-main pressure shall be provided.

(6) The fire pump installation should conform to NFPA 20, "Standard for the Installation of Centrifugal Fire Pumps."

(7) The water supply shall meet the following requirements:

  (a) Fresh water free of silt and debris shall be used. Filters for makeup supply are acceptable.

  (b) Each supply shall have a minimum storage volume of $1140 m^3$.

  (c) If tanks or other limited volume storage means are used:

    (i) There shall be two storage devices.

    (ii) One storage device shall contain a passively dedicated volume of $456 m^3$ to supply two hose streams for two hours in areas required for safe shutdown.

    (iii) The makeup supply shall be capable of providing $1140 m^3$ for either storage device in 8 hours.

(8) Normally closed valve, opened only to pump from the alternate supply.

(9) Normally closed valve, opened only when motor driven pump is out of service.

(10) Normally closed valve, opened only when a section of the piping connected to normal water supply is valved out for maintenance.

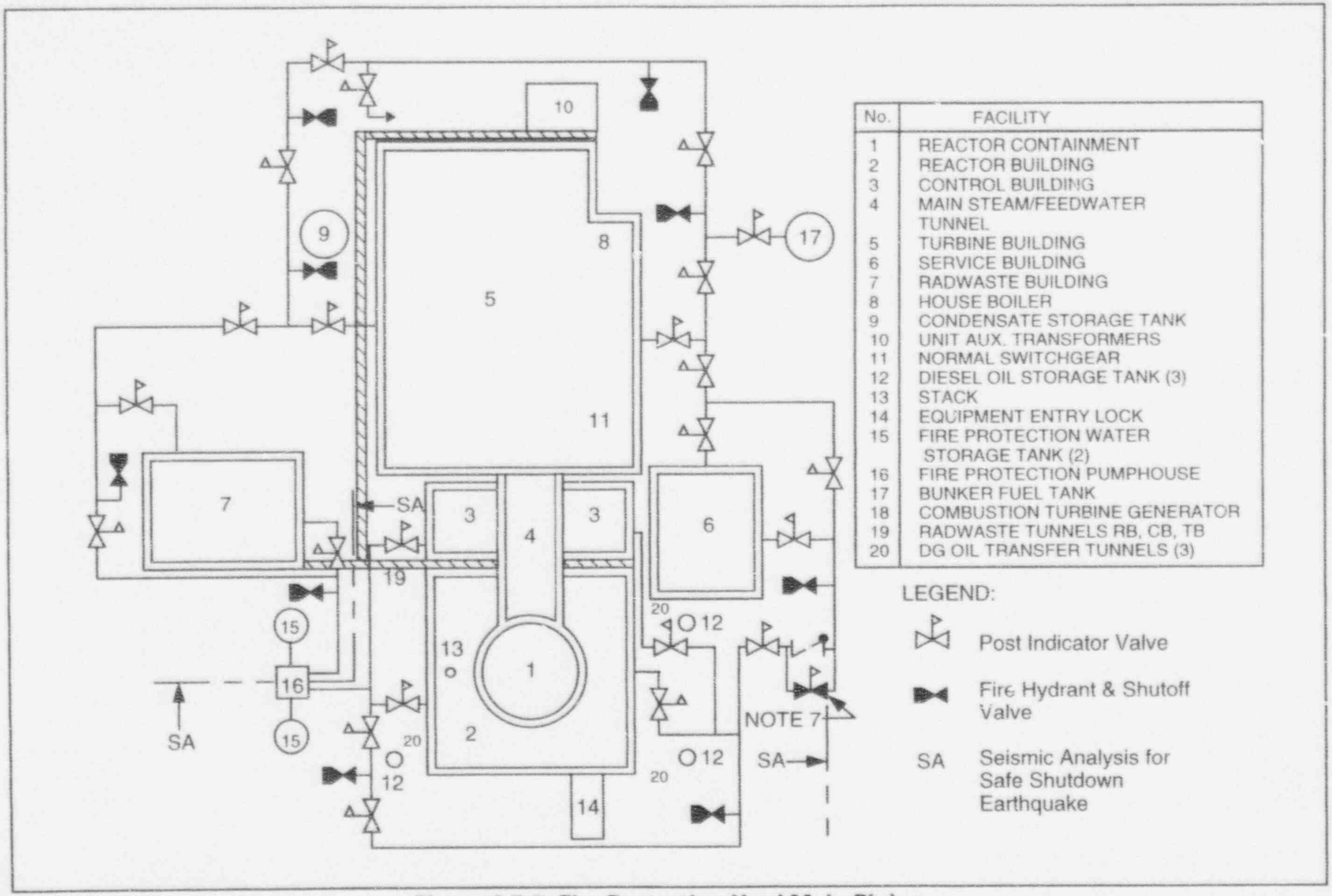


Figure 9.5-5 Fire Protection Yard Main Piping

$$Q_{fs} = 0.45 q_{bs} A \tag{9B-1}$$

where q_{bs} is the bench scale burning rate taken from Table A-7M of the FIVE document (Reference 9B-4). "A" is the burning surface area.

The data estimated from tests at UL was taken from a series of modified IEEE 383 tests conducted in 1976 (Reference 9B-5). Although it was not the purpose of the tests to determine burning rate, it is possible to estimate the burning rate from the reported insulation consumed and cable burning time as determined by time tagged photographs of the tests in progress. Cross-linked polyethylene and Tefzel insulated cables of the constructions discussed earlier in this section (Table 9B-3) were tested with the range of burning rates indicated in Table 9B-4 as the results.

The ventilation limited burning rate was calculated using the FIVE methodology using the Draft FIVE Plant Screening Guide (Reference 9B-4). The equation is:

$$Q_{max}/V = 3600 \text{ kW/m}^3\text{/s} \tag{9B-2}$$

where Q_{max} is the maximum heat release rate, V is the volume flow, and m^3/s is the volume flow rate in cubic meters per second.

(9B-3)

For one m^2 of a room with a ceiling height of 4.57m and a ventilation rate of 3 air changes per hour the ventilation rate is 13.71 m^3/h (1 m^2 x 4.57m x 3 changes per h). Q_{max} is equal to:

$$Q_{max} = 3600 \times 13.71 \tag{9B-4}$$

= 49.37 MJ/hr per m^2

The burning rate for the design normal combustible load limit is the combustible load limit of 727 MJ/m^2 as defined in Subsection 9B.3.2.1, divided by 180 minutes (3 hours) is 4.04 MJ/min per m^2.

Similarly the burning rate of 15.13 MJ/min per m^2 for the fire barrier capability is the capability of 2.72 GJ/m^2 divided by 180 minutes.

The normal combustible load limit of 964 kcal/min per m^2 divided into the burning rate of 6.99 MJ/m^2 to 37.84 MJ/m^2 of open ladder cable tray gives a ratio of 1,581 to 8,742 cm^2 of floor area per 0.093 m^2 of cable tray in a room, depending on the type of insulation used.

The value of the burning rate calculations is that they give an idea of what the localized burning rate might be for a cable fire that is not burning in the ventilation controlled mode. Multiple trays of cables should not be run in rooms such as oil storage tank rooms where there would be an ignition source sufficiently large to ignite the entire amount of cable in the room. Also, areas containing potential ignition sources sufficiently large to ignite large amounts of cables in the areas are sprinkled. For these reasons, the normal combustible loading limit based on the total combustibles per square foot should be used in preference to using the localized burning rate as the basis for setting the limit.

One additional comment is that the low ventilation controlled burning rate of 49.36 MJ/hr per m^2 of floor area as compared to the barrier system capacity of 15.13 MJ/min per m^2 is another indication of the design margin that is provided by the three-hour fire barrier system. The capacity of the barrier system should not be approached by the fire intensity except possibly during the time when the ventilation rate to the area experiencing the fire has been increased to facilitate fire suppression activities.

It is possible that during the detailed design phase certain areas of concentration of cable trays may exceed the normal or electrical combustible loading limit. Multiplexing of signals and the overall plant layout will tend to minimize the number of these areas of concentration of cable trays. There are options available to the detail designer to which will specific concentrations above the general stated combustible loading limits. For example, the designer could use one or more of the following options.

Option 1

Use a cable insulation with a lower required thickness, a low heat of combustion or a low burning rate, such as Tefzel. The number of cable trays could be held constant or the same number of cables could be routed through fewer cable trays.

Option 2

A second option would be to utilize cable trays with solid bottoms and solid covers for the congested areas.

9B.3 References

9B-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

9B-2 Cote, Arthur E., *NFPA Fire Protection Handbook*, National Fire Protection Association, Sixteenth Edition

15.0.4 Event Evaluation

15.0.4.1 Identification of Causes and Frequency Classification

Situations and causes which lead to the initiating event analyzed in the sensitivity study are described within the categories designated in Subsection 15.0.3. The frequency of occurrence of each event is summarized based upon the NSOA and currently available operating plant history for the transient event. Events for which inconclusive data exist are discussed separately within each event section.

Each initiating event within the major groups is assigned to one of three frequency groups defined in Regulatory Guide 1.70.

15.0.4.2 Identified Results

Events analyzed for each plant must meet the criteria in Appendix 4B.

15.0.4.3 Sequence of Events and Systems Operations

Each transient or accident evaluated in the sensitivity study is discussed and evaluated in terms of:

(1) A step-by-step sequence of events from initiation to final stabilized condition

(2) The extent to which normally operating plant instrumentation controls are assumed to function

(3) The extent to which the plant and reactor protection systems are required to function

(4) The credit taken for the functioning of normally operating plant systems

(5) The operation of engineered safety systems that is required

This sequence of events is supported by the NSOA for the transient or accident. The effect of a single equipment failure or malfunction or an operator error on the event is shown in the NSOA.

15.0.4.4 Analysis Basis

The sensitivity study results given in this chapter are based upon the core loading given in Figure 4.3-1. These sensitivities are valid for other fuel designs and core loadings.

15.0.4.4.1 Evaluation Models

The computer codes used in the analysis of the transients and accidents in this chapter are shown in Table 15.0-1-A. These models have been approved by the USNRC.

15.0.4.4.2 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed for the sensitivity analysis documented within this section have values for input parameters and initial conditions as specified in Table 15.0.-1. Analyses which assume data inputs different than these values are designated accordingly in the appropriate event discussion.

The normal maximum allowable reactor operating condition is the 100%-power/100%-flow condition. The maximum power measurement uncertainty is usually ~2%. Therefore, the sensitivity analyses are based on 102% power level. The transient results at this condition are more severe than that at rated condition.

The analytical values for some system characteristics, like SRV delay/stroke time, reactor internal pump coastdown time constant, etc., bound the design specification for that system. These values will be checked during startup tests.

All setpoints for the protection system assumed in the analyses are conservative, which includes instrument uncertainty, calibration error and instrument drift. The nominal and allowable values for these setpoints, (see Technical Specifications) assume that the setpoints will not exceed what are assumed in the analyses.

In conclusion, the input parameters and initial conditions (including uncertainties) used in the sensitivity study are conservative values and bound the operating band.

15.0.4.4.3 Initial Power/Flow Operating Constraints

The power/flow map used for the system response analysis is shown in Figure 15.0-1. The analyses basis for most of the sensitivity analyses is 102% thermal power at rated core flow (100%). Rated core flow can be achieved with either nine or ten pumps in operation. This operating point is the apex of the operating power/flow map which, in response to any classified abnormal operational transients, will yield the minimum pressure and thermal margins. Referring to Figure 15.0.-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (102% rod line A-D), the lower bound is the zero power line H'-J, the right bound is the maximum flow line A'-H', and the left bound is the natural circulation line D-J.

The power/flow map (A-D-J-H'-A') represents the operational region covered by abnormal operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map (e.g., the moisture carryover protection region, the licensed power limit and other restrictions based on pressure and thermal margin criteria) must be observed. See Subsection 4.4.3.3 for power/flow map operating instructions. The upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis and the corresponding constant rod pattern line. This boundary may be truncated by the licensed power and the GETAB operating limit.

15.1.3.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5 for Type 2 events. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.1.4 Inadvertent Safety/Relief Valve Opening

15.1.4.1 Identification of Causes and Frequency Classification

15.1.4.1.1 Identification of Causes

Cause of inadvertent opening is attributed to malfunction of the valve or an operator initiated opening. It is therefore simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

15.1.4.1.2 Frequency Classification

This transient disturbance is categorized as an infrequent incident.

15.1.4.2 Sequence of Events and Systems Operation

15.1.4.2.1 Sequence of Events

Table 15.1-8 lists the sequence of events for this event.

15.1.4.2.1.1 Identification of Operator Actions

The plant operator must reclose the valve as soon as possible and check that reactor and T-G output return to normal. If the valve cannot be closed, plant shutdown should be initiated.

15.1.4.2.2 Systems Operation

This event assumes normal functioning of normal plant instrumentation and controls, specifically the operation of the pressure regulator and level control systems.

15.1.4.3 Core and System Performance

The opening of one SRV allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.

The SB&PCS senses the nuclear system pressure decrease and within a few seconds closes the turbine control valves far enough to stabilize the reactor vessel pressure at a slightly lower value and the reactor settles at nearly the initial power level. Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. MCPR is essentially unchanged and, therefore, the safety limit margin is unaffected and this event does not have to be reanalyzed for specific core configurations.

The discharge of steam to the suppression pool causes the temperature of the suppression pool to increase. When the pool temperature reaches the high temperature setpoint, the suppression pool cooling function of the RHR System is automatically initiated. The pool temperature continues to increase due to the mismatch of cooling capacity and steam discharged into the pool. When the pool temperature reaches the next setpoint of 43.3°C, a reactor scram is automatically initiated. In this analysis a conservative scram set point of 48.9°C was assumed.

15.1.4.4 Barrier Performance

As presented previously, the transient resulting from a stuck open relief valve is a mild depressurization which is within the range of normal load following and therefore has no significant effect on RCPB and containment design pressure limits.

15.1.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Because this activity is contained in the primary containment, there will be no exposures to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with the established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.5 Spectrum of Steam System Piping Failures Inside and Outside Containment in a PWR

This event is not applicable to BWR plants.

15.1.6 Inadvertent RHR Shutdown Cooling Operation

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

At design power conditions, no conc .ble malfunction in the shutdown cooling system could cause a temperature reduction.

In startup or cooldown operation, if the reactor were critical or near critical, a very slow increase in reactor power could result. A shutdown cooling malfunction leading to a moderate temperature decrease could result from misoperation of the cooling water showed that the final peak fuel enthalpy was approximately 146.5 J/g, lower than the RWE criteria for fuel integrity.

15.4.1.2.4 Barrier Performance

An evaluation of the barrier performance is not made for this event, because there is no fuel damage in this event and only with mild change in gross core characteristics.

15.4.1.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel.

15.4.2 Rod Withdrawal Error at Power

15.4.2.1 Features of the ABWR Automatic Thermal Limit Monitoring System (ATLM)

In the ABWR, the Automatic Thermal Limit Monitoring (ATLM) System performs the rod block monitoring function. The ATLM System is a dual channel subsystem of the Rod Control and Information System (RCIS). In each ATLM channel there are two independent thermal limit monitoring devices. One device monitors the MCPR limit and protects the operating limit of the MCPR, and the other device monitors the APLHGR limit and protects the operating limit of the APLHGR. The rod block algorithm and setpoint of the ATLM System are based on actual online core thermal limit information. If any one of the two limits is reached, either due to control rod withdrawal or recirculation flow increase, control rod withdrawal permissive is removed. Detailed description of the ATLM System is presented in Reference 15.4-1 and Chapter 7.

15.4.2.2 Identification of Causes and Frequency Classification

The causes of a potential RWE transient are either a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously. But in either case, the operating thermal limits rod block function will block any further rod withdrawal when the operating thermal limit is reached. That is, the withdrawal of rods will be stopped before the operating thermal limit is reached. Because there is no operating limit violation due to the preventive function of the ATLM, there is no RWE transient event.

15.4.2.3 Sequence of Event and System Operation

Due to an operator error or a malfunction of the automated rod withdrawal sequence control logic, a single control rod or a gang of control rods is withdrawn continuously. The ATLM operating thermal limit protection function of either MCPR or MLHGR protection algorithm stops further control rod withdrawal when either operating limit is reached. There is no basis for occurrence of the continuous control rod withdrawal error event in the power range.

No operator action is required to preclude this event, because the plant design as described above prevents its occurrence.

15.4.2.4 Core and System Performance

The performance of the ATLM System of the RCIS prevents the RWE event from occurring. The core and system performance are not affected by such an operator error or control logic malfunction. There is no need to analyze this event.

15.4.2.5 Barrier Performance

An evaluation of the barrier performance is not made for this event, because there is no postulated set of circumstances for which this event could occur.

15.4.2.6 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel.

15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)

This event is covered with evaluations presented in Subsections 15.4.1 and 15.4.2 and does not have to be reanalyzed for specific core configurations.

15.4.4 Abnormal Startup of Idle Reactor Internal Pump

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from manual action by the operator to initiate pump operation. It is assumed that the remaining nine RIPs are already operating.

The normal restart procedure requires the operator to reduce the pump speeds of running RIPs to , at or near, their minimum speeds (i.e., 30% of rated speed) before the restart of the idle RIP. Plant operating porcedures specifiy the maximum allowable speed for the nine operating RIPs, for a normal restart of one RIP. Therefore, an abnormal restart occurs only when an operator error (i.e., operator ignoring the procedure) occurs. Should an abnormal restart occur, the much higher reverse flow at the idle RIP requires the inverter to provide electrical current much higher than the normal. This overcurrent requirement activates the overcurrent protection logic of the adjustable speed drive (ASD) which supplies the power to the idle RIP. This ASD is tripped by the protection logic. Therefore, an abnormal restart of the idle RIP becomes a trip of one RIP, which is presented in Subsection 15.3.1.

15.4.4.1.1.1 Normal Restart of Reactor Internal Pump

This transient is categorized as an incident of moderate frequency.

15.4.4.1.1.2 Abnormal Startup of Idle Reactor Internal Pump at High Power

This transient should be considered as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

Table 15.4.3 lists the sequence of events for an abnormal startup of an idle RIP.

15.4.4.2.1.1 Operator Actions

The normal sequence of operator actions expected in starting the idle loop is as follows. The operator should:

(1) Adjust rod pattern, as necessary, for new power level following idle RIP start

(2) Reduce the speed of the running RIPs to, at or near, their minimum speeds

(3) Start the idle loop pump and adjust speed to match the running RIPs (monitor reactor power)

(4) Readjust power, as necessary, to satisfy plant requirements per standard procedure

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls. No protection systems action is anticipated. No ESF action occurs as a result of the event.

15.4.4.3 Core and System Performance

An abnormal restart of an idle RIP becomes a trip of one RIP event, which is presented in Subsection 15.3.1.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event because no significant pressure increases are incurred during this transient (Subsection 15.3.1).

15.4.4.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel.

15.4.5 Recirculation Flow Control Failure with Increasing Flow

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

The ABWR Recirculation Flow Control System (RFCS) uses a triplicated, fault-tolerant digital control system. The RFCS controls all ten reactor internal pumps (RIPs) at the same speed. As presented in Subsection 15.1.2.1.1, no credible single failure in the control system results in a maximum demand to all RIPs. A voter or actuator failure may result in an inadvertent runout of one RIP at its maximum drive speed (~40%/s). In this case, the RFCS senses the core flow change and commands the remaining RIPs to decrease speed and thereby automatically mitigate the transient and maintains the core flow.

As presented in Subsection 15.1.2.1.1, multiple failures in the control system might cause the RFCS to erroneously issue a maximum demand to all RIPs. Should this occur, all RIPs could increase speed simultaneously. Each RFCS processing channel has a speed demand limiter which limits the maximum speed change rate to 5%/s. However, the probability of this event occurring is low (less than 7×10^{-5} failures per reactor year), and, hence, the event should be considered as a limiting fault.

15.4.5.1.2 Frequency Classification

15.4.5.1.2.1 Fast Runout of One Reactor Internal Pump

The failure rate of a voter or an actuator is about 0.0088 failure per reactor year. However, it is analyzed as an incident of moderate frequency.

15.4.5.1.2.2 Fast Runout of All Reactor Internal Pumps

This event should be considered as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

15B.3.3.3 Adjustable Speed Drives

The adjustable speed drives (ASD) will be used to supply variable voltage/variable frequency electrical power to the reactor recirculation pumps. The recirculation pumps are single stage, vertical pumps driven by three-phase, four-pole, wet-type, squirrel cage, AC induction motors. Each ASD will supply power to one recirculation pump motor. The ASD receives electrical power from a supply bus at a relatively constant AC voltage and frequency. The ASD converts this constant supply power to a variable frequency/variable voltage output which is supplied to the recirculation pump motor. The output frequency is modulated in response to a demand signal from the system controller in order to vary pump speed.

15B.3.4 RIP Failure Modes Evaluation

The following evaluations and discussions of failure modes which are relevant to the safety of the plant are presented here as summary of detailed analyses.

15B.3.4.1 Missiles Generation

Since the parts of the RIP (impeller) are rotating inside the reactor pressure vessel (RPV), an evaluation has been made to assess the integrity of the RPV should an "impeller missile" occur. Although the rated speed for the RIPs is 157 rad/s, an initial speed of 188.5 rad/s is used for this evaluation. For unidentified reasons, the RIP impeller located approximately 3m below the reactor core bottom is assumed to disintegrate.

The acceptance criterion for a missile striking the RPV cylindrical shell or reactor core shroud is that the kinetic energy (KE) of the missile is less than the critical energy (CE) of the shell and shroud and, therefore, the missile will not degrade the integrity of the core or pressure boundary. The acceptance values are:

(1) RPV shell CE -9.41 MN·m

(2) Core shroud CE -0.24 MN·m

Calculations show that the energy of the impeller missile is:

$$0.09 \text{ MN·m} \tag{15B-1}$$

Comparing the information above, the impeller missile KE is approximately one-half the shroud CE and one-tenth the RPV shell CE.

In conclusion, the integrity of the core and RCPB are maintained in the event of a RIP impeller disintegration.

15B.3.4.2 Pump Seizure

Pump seizure causes rapid reduction of core flow and torsional loads on the RIP casing, RPV RIP nozzle, and RIP motor bottom flange. Several modes of pump seizure have been considered.

The RIP is assumed to be operating at 157 rad/s and for unidentified reasons the following seizures are assumed to occur:

(1) Impeller to diffuser seizure

(2) Rotor winding to stator winding seizure

(3) Thrust bearing seizure

(4) Radial bearing seizure

Any of these seizures will trip off the motor power and transfer the rotating kinetic energy of the impeller and motor rotor shaft into the RPV bottom head RIP nozzle directly or up through the motor housing into the nozzle.

The acceptance criterion for this event is that the torque load resulting from the seizure be less than value specified as the design basis for this event in the reactor vessel loading specification. This value is 42 T-M.

Depending on the location of the seizure in the pump or motor, the impeller-shaft kinetic energy will shear off one set of several bolts and pins in the motor structure. The torque load which shears the bolts and pins is transferred into the bottom flange of the motor housing and up through the housing cylinder into the RPV bottom head RIP nozzle.

In conclusion, any of the calculated torque loads transferred into the RPV RIP nozzles by a RIP or motor seizure are more than a factor of 4 less than the (42 T-M) design torque load specified by the reactor vessel loading specification for this faulted condition. The pump seizure torque will produce stresses in the motor housing and RPV RIP nozzle which are significantly less than Code allowable stresses.

15B.3.4.3 RIP Motor Housing Break

The motor housing and bottom flange are part of the RCPB and therefore are designed not to fail or rupture during normal, upset, emergency, or faulted plant conditions. Regardless of these criteria, and for the purpose of this evaluation, it is assumed that the housing fails creating a temporary small LOCA.

First it is assumed that the RIP impeller and shaft remain intact. The vertical blowout restraint rods prevent the motor and broken housing from being ejected from the RPV

In conclusion, the failure of the purge flow to the RIP will be mitigated by the normal makeup or normal maintenance procedures for secondary seal replacement.

15B.3.4.6 RIP Heat Exchanger Secondary Water Flow Loss

The RIPs are designed to operate normally in the following situations which are the acceptance criteria for these events:

(1) **Failure of Secondary Cooling Water**—The RIP motor shall be capable of continued rated power operation for 5 minutes following failure of the RCW. This time period allows corrective action to prevent an all-pump trip.

(2) **Hot Standby Without RCW**—With the RIP stopped, the motor shall withstand hot standby conditions for one hour with the RCW to the RIP motor heat exchanger (RMHx) shut off. This allows adequate time to take corrective action.

The evaluation of the RCW cooling water failure shows the motor water temperature increase will be as follows:

Time (min.)	Temp. (°C)	Status
0	55	RIP at maximum rated power and cooling water is shut off
2	60	Alarm
4	65	RIP auto runback and trip
65	70	Maximum motor cooling outlet temperature

The entire RIP motor housing, RIP motor heat exchanger, and interconnecting piping is designed for minimum 302°C at 8.62 MPa pressure. Therefore, an indefinite loss of RCW to the RIP motor heat exchanger will not affect the integrity of the RCPB.

The operator will receive a low RCW flow alarm and RMHx primary side inlet and outlet water temperature high alarm. If the RCW cannot be restored to the tripped motor, some damage to the winding insulation and/or secondary shaft seal, may occur. These components can be replaced according to normal RIP maintenance procedures.

15B.3.4.7 RIP Primary Cooling Water Loss

The RIP motor housing, RIP motor heat exchanger, and connecting piping are designed in accordance with the same codes and standards as the RPV. This design precludes the rupture of any of the RCPB components during any plant service condition. Regardless of this design criterion and for the purpose of this evaluation, it is assumed that a rupture of the 65A motor cooling water piping occurs or the RIP motor heat exchanger tubes fail.

Rupture of the motor cooling water piping will result in a small LOCA. This discharge of reactor coolant from the pipe break is restricted by the annulus between the pump shaft and the stretch tube. The acceptance criterion for this event from the viewpoint of nuclear plant safety is that equivalent break size not exceed 20 cm^2, which is the design basis bottom break. The actual flow area of the cooling water piping is restricted by the lower part of the stretch tube flow area is 10 cm^2. This small LOCA is detected by temperature, pressure, and/or level instrumentation for the RPV, drywell and/or RIP motor cooling circuit. The normal makeup systems are designed to mitigate the consequences of this small LOCA.

An RIP motor heat exchanger tube break will result in reactor coolant being discharged into the Reactor Cooling Water (RCW) System. This event will be detected by high motor cooling water temperatures, high RCW temperatures, high RCW surge tank level and/or high RCW radioactivity levels. The radioactivity will be contained in the RCW system and not discharged to the environment. As the reactor is being shut down, the discharge of reactor coolant into the RCW can be terminated by closing the primary containment RCW isolation valves after the RIPs have been stopped.

The heat exchanger tube leak rate will be the same as or less than the leak rate for motor cooling the pipe break. This is due to the fact that the leak rate is controlled by the annulus between the shaft and stretch tube.

It is assumed that any cause of RIP motor primary cooling water due to a rupture in the motor coolant circuit will damage the RIP motor winding insulation by the 278°C RPV water entering the motor. The motor can be replaced according to normal RIP maintenance procedures.

In conclusion, the ABWR RIP motor cooling system and normal ABWR coolant makeup systems are designed to detect and mitigate the consequences of a loss of RIP primary cooling water and consequent loss of reactor coolant.

15B.3.4.8 ABWR RIP Loose Part Prevention and Monitoring

The ABWR RIP is an assembly of many parts, some of which are inside the RPV. The parts in a majority of cases are held together by threaded fasteners such as studs, bolts, nuts, and screws. Although these types of fasteners make disassembly possible, they can become loose due to random vibration of the running pump and lead to gross failure of the other parts. Fragments of broken components can be transferred to the reactor internals and fuel. Due to criticality of loose parts, the RIP fasteners are engineered to be positively locked as described below:

(1) A lock sleeve and pin prevent loosening and disassembly of the impeller.

BASES

BACKGROUND (continued)

operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., 55 to 100% RTP) without having to move control rods and disturb desirable flux patterns.

Each RIP is manually started from the control room. The ASDs provide regulation of individual RIP speed and, therefore, flow. The flow through each RIP can be manually or automatically controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Coolant Recirculation System with 100% core flow is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1) and abnormal operating transients (Ref. 2). Rated core flow can be achieved with either nine or ten RIPs in operation. During a LOCA and an all RIPs trip accident, all operating RIPs are assumed to trip at time zero due to a coincident loss of offsite power. The subsequent core flow coastdown will be immediate and rapid because of the relatively low inertia of the pumps and motors. However, the RIPs are assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the SSAR.

A plant specific LOCA analysis may be performed assuming only [ ] operating RIPs. This analysis shall demonstrate that, in the event of a LOCA, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses of Chapter 15 of the SSAR may also be performed for [ ] RIPs in operation (Ref. 3) to demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During operation with only [ ] RIPs, modification to the Reactor Protection System average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between reactor internal pump flow

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(reverse flow through the pump impellers) and reactor core flow. The APLHGR and MCPR setpoints for [ ] RIPs in operation are to be specified in the COLR. The APRM flow biased simulated thermal power setpoint is in LCO 3.3.1.1, "SSLC Sensor Istrumentation."

RIPs operating satisfies Criterion 2 of the NRC Policy Statement.

LCO

At least nine RIPs are required to be in operation to ensure that during a postulated LOCA or transient the assumptions of the associated analyses are satisfied. With only [ ] RIPs in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and APRM Flow Biased Simulated Thermal Power – High setpoint (LCO 3.3.1.1) may be applied to allow continued operation consistent with the assumptions of Reference 1.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur. In MODES 3, 4, and 5, the consequences of an accident are reduced and the flow and coastdown characteristics of the RIPs are not important.

ACTIONS

A.1

With the requirements of the LCO not met, the unit is required to be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the RIPs are not required to be operating because of the reduced severity of DBAs and minimal dependence on the RIPs flow and coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

B.1

If the suppression pool average temperature cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the power must be reduced to ≤ 1% RTP within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reduce power from full power conditions in an orderly manner and without challenging plant systems.

C.1

Suppression pool average temperature is allowed to be > 35°C when THERMAL POWER is >1% RTP, and when testing that adds heat to the suppression pool is being performed. However, if temperature is > 40.6°C all testing must be immediately suspended to preserve the heat absorption capability of the suppression pool. With the testing suspended, Condition A is entered and the Required Actions and associated Completion Times are applicable.

D.1

When the suppression pool temperature reaches 43.3°C a reactor scram is automatically initiated. Additionally, when suppression pool temperature is > 43.3°C, increased monitoring of pool temperature is required to ensure that it remains ≤ 48.9°C. The once per 30 minute Completion Time is adequate, based on operating experience. Given the high suppression pool average temperature in this Condition, the monitoring Frequency is increased to twice that of Condition A. Furthermore, the 30 minute Completion Time is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

E.1 and E.2

If suppression pool average temperature cannot be maintained at ≤ 48.9°C, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.2

This SR verifies that the required SGT System filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specified test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

This SR requires verification that each SGT train starts upon receipt of an actual or simulated initiation signal. The applicable SRs in LCO 3.3.1.1 and LCO 3.3.1.4 overlap this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.4.3.4

This SR requires verification that the SGT System filter cooler bypass damper can be opened and the fan started. This ensures that the ventilation mode of SGT System operation is available. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.

2. ABWR SSAR, Section 6.2.3.

(continued)

3.7 PLANT SYSTEM

3.7.4 Control Room Habitability Area (CRHA) - Emergency Filtration (EF) System

LCO 3.7.4 Two divisions of the CRHA EF System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One EF division inoperable.	A.1	Restore EF division to OPERABLE status.	7 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	C.1 Place OPERABLE EF division in standby mode.	Immediately
	OR	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two EF divisions inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two EF divisions inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	E.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Operate each EF division for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.4.2	Perform required EF filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.4.3	Verify each EF division actuates on an actual or simulated initiation signal.	18 months

(continued)

3.7 PLANT SYSTEM

3.7.5 Control Room Habitability Area (CRHA) - Air Conditioning (AC) System

LCO 3.7.5 Two divisions of the CRHA AC System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC division inoperable.	A.1 Restore AC division to OPERABLE status.	30 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	C.1 Place OPERABLE AC division in operation.	Immediately
	OR	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two AC divisions inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two AC divisions inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	E.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each CRHA AC division has the capability to remove the assumed heat load.	18 months
SR 3.7.5.2	Verify each CRHA AC division actuates on an actual or simulated initiation signal.	18 months

BASES

APPLICABILITY (continued)

following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Emergency Filtration System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During operations with a potential for draining the reactor vessel (OPDRVs);

b. During CORE ALTERATIONS; and

c. During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one Emergency Filtration division inoperable, the inoperable Emergency Filtration division must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE Emergency Filtration division is adequate to perform MCAE radiation protection. However, the overall reliability is reduced because a single failure in the OPERABLE division could result in loss of Emergency Filtration System function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and that the remaining division can provide the required capabilities.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable Emergency Filtration division cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the

(continued)

ACTIONS (continued)

B.1 and B.2 (continued)

required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, C.2.2, and C.2.3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable Emergency Filtration division cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE Emergency Filtration division may be placed in the filtration mode. This action ensures that the remaining division is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require initiation of the Emergency Filtration System. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

(continued)

BASES

ACTIONS (continued)

D.1

If both Emergency Filtration divisions are inoperable in MODE 1, 2, or 3, the Emergency Filtration System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

E.1, E.2, and E.3

The Required Actions of Condition E are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs with two Emergency Filtration divisions inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require initiation of the Emergency Filtration System. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that a division in standby mode starts on demand and continues to operate. Standby systems should be checked periodically to ensure that they start and function properly. As the environmental and normal operating conditions of this system are not severe, testing each

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The CRHA AC System satisfies Criterion 3 of the NRC Policy Statement.

LCO	Two independent and redundant divisions of the CRHA AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other division. Total system failure could result in the equipment operating temperature exceeding equipment qualification limits. The CRHA AC System is considered OPERABLE when the individual components necessary to maintain the MCAE temperature are OPERABLE in both divisions. These components include the cooling coils, fans, ductwork, dampers, and associated instrumentation and controls.

APPLICABILITY	In MODE 1, 2, or 3, the CRHA AC System must be OPERABLE to ensure that the MCAE temperature will not exceed equipment OPERABILITY limits following control room isolation. In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CRHA AC System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated: a. During operations with a potential for draining the reactor vessel (OPDRVs); b. During CORE ALTERATIONS; and c. During movement of irradiated fuel assemblies in the secondary containment.

(continued)

BASES (continued)

ACTIONS

A.1

With one CRHA AC division inoperable, the inoperable CRHA AC division must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE CRHA AC division is adequate to perform the MCAE air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE division could result in loss of the MCAE air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring MCAE isolation, the consideration that the remaining division can provide the required protection, and the availability of alternate cooling methods.

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable CRHA AC division cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, C.2.2, and C.2.3

The Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE CRHA AC division may be placed immediately in operation.

(continued)

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

This action ensures that the remaining division is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the MCAE. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

D.1

If both CRHA AC divisions are inoperable in MODE 1, 2, or 3, the CRHA AC System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately.

E.1, E.2, and E.3

The Required Actions of Condition E.1 are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs with two CRHA AC divisions inoperable, action must be taken to immediately suspend activities that present

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and handling of irradiated fuel in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR verifies that the heat removal capability of the system is sufficient to remove the MCAE heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 18 month Frequency is appropriate since significant degradation of the CRHA AC System is not expected over this time period.

REFERENCES

1. ABWR SSAR, Section 6.4.

2. ABWR SSAR, Section 9.4.1.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement.

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, such that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow continued operation.

An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure or in the Fast Opening Mode, as applicable. This response is within the assumptions of the applicable analysis (Ref. 2). The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR.

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 40% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, sufficient margin to these limits exists at a power level < 40% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR limits accordingly. The 2 hour Completion Time is

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.8.3.6 (continued) presence of sediment does not necessarily represent a failure of this SR provided that accumulated sediment is removed during performance of the Surveillance.
REFERENCES	1. ABWR SSAR, Section 9.5.4. 2. Regulatory Guide 1.137. 3. ANSI N195, Appendix B, 1976. 4. ABWR SSAR, Chapter 6. 5. ABWR SSAR, Chapter 15. 6. ASTM Standards: D4057-[ ]; D975-[ ]; D4176-[ ]; D975-[ ]; D1552-[ ]; D2622-[ ]; D2276-[ ]. 7. ASME, Boiler and Pressure Vessel Code, Section XI.

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

ABWR EPG Step	BWROG EPG Rev. 4 Step	Differences from BWROG Rev. 4 EPG	Basis for Differences
C5-2	C5-2	• Changed the phrase, "if any MSIV is open, bypass low RPV water level pneumatic system and MSIV isolation interlocks and restore the pneumatic supply to the containment, and," to read as follows: "if any MSL is not isolated, bypass low RPV water level interlocks to maintain the main condenser as a heat sink".	• The pneumatic system supply to the MSIVs in the ABWR has no low RPV water isolation and no high drywell pressure isolation and, therefore, the instruction to bypass isolation interlocks and restore the pneumatic supplies (having been isolated) is not appropriate.
		• Part of Rev. 4 Step C5-2 moved and added as a separate step: "If any MSL is not isolated, bypass low RPV water level interlocks to maintain the main condenser as a heat sink."	• RPV water level is lowered in subsequent steps to minimize both reactor power and core inlet subcooling. Bypassing low RPV water level interlocks is specified to avoid RPV depressurization that could be required as a result of loss of the main condenser. Increased emphasis is placed on maintaining the main condenser as a heat sink to address mitigation of ATWS.
		• Subsequent steps and references to them are renumbered for consistency with the added Step C5-2.	

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

ABWR EPG Step	BWROG EPG Rev. 4 Step	Differences from BWROG Rev. 4 EPG	Basis for Differences
C5-3	C5-2	• Moved the instruction: "If any MSIV is open, bypass RPV low water level ... [to the containment], and " to Step C5-2 with changes noted above.	• This instruction was specified previously as the new Step C5-2 and becomes unnecessary in this step.
C5-4	C5-4	• Entire C5-4 step and the "If while executing..." phrase above the step are deleted.	• This step is only applicable for plants with SLC injecting into the bottom of the RPV. The ABWR SLC injects via HPCF(B) into the core. Therefore, this step is not applicable to the ABWR.
	—	• Step added: "If RPV water level is above [164.9 cm (Maximum Power Control RPV Water Level)] and the reactor is not shutdown: Lower RPV water level to below [164.9 cm (Maximum Power Control RPV Water Level)] by terminating and preventing all injection into the RPV except from boron injection systems and CRD.	• Reducing RPV water level to the Maximum Power Control RPV Water Level assures that water injected through the feedwater sparger will fall a sufficient distance through a steam environment such that its subcooling will be significantly reduced by the time it reaches the RPV water level surface. Raising the enthalpy of the water entering the lower plenum and core inlet is expected to prevent or mitigate large periodic oscillations induced by neutronic /thermal-hydraulic instabilities.

The capacity of a group of ten 45 liter high pressure N_2 gas bottles at 5.96 MPa minimum pressure is about 16 times that needed to open the 8 ADS SRVs, each of which has an actuator piston volume of 16.4 liters (1000 cubic in). Additionally, there are 10 other N_2 bottles that can be valved into service by local manual operation. After the 8 ADS valves are opened there is sufficient N_2 gas to account for at least 7 days leakage from the valve actuators, after which the N_2 bottles must be replaced to hold the ADS valve open. Based on the foregoing, it is concluded that the ADS valves can be operated to depressurize the reactor on loss of normal AC power supplies with the containment at 0.86 MPa. The operator has to manually close and open valves at the valve locations to supply nitrogen from outside the containment to open the 8 SRVs used for the ADS function and to hold them open when the pressure in the RPV drops to near containment pressure.

(3) DC Battery Capacity.

The Division I DC battery will be sized to be capable of operating the RCIC system for approximately 8 hours assuming the expected loading profiles for station blackout with failure of the CTG. These loading profiles will assume acceptable battery area environmental conditions and load shedding, when necessary, and will be defined in detail as the ABWR design progresses.

(4) Water Source Inventory.

The primary water source for the RCIC System is the condensate storage tank (CST) which has been sized to provide sufficient inventory for a minimum of 8 hours in combination with the suppression pool. In the event the CST became depleted, the backup source is the suppression pool. The suction source switches to the suppression pool automatically on high suppression pool level. The RCIC system must be manually overridden to assure that the suction revert to the condensate storage tank to limit heating of the containment.

(5) RCIC Room Temperature.

Failure of the AC power to the room cooling will allow the RCIC room temperature to rise. The ABWR plant will be designed to prevent the room temperature from reaching the equipment design temperature of 340 K (151°F), starting at the normal room temperature of 313 K (104°F), for at least 8 hours.

(6) Control Room Temperatures.

The safety-related equipment required to function during station blackout with failure of the CTG and located in the main, lower and computer control rooms will be designed for a maximum operating temperature of 331 K (136°F). The ABWR plant will be designed to prevent the control room temperature from reaching this equipment design temperature for at least 8 hours, starting at the normal room temperature of 299 K (79°F).

19E.2.1.2.2.3 Primary Containment Vessel (PCV) Integrity

Containment pressure and temperature analyses were performed to determine the containment atmospheric conditions after 8 hours of station blackout conditions with failure of the CTG assuming event initiation at 100% thermal power. An analysis was performed which assumed the RCIC suction was taken from the condensate storage tank for the duration of the event. The drywell and wetwell pressure and temperature were calculated to be less than their design basis of 0.411 MPa and 444 K (340°F) (drywell)/377 K (219°F) (wetwell) after 8 hours. Therefore, PCV integrity is maintained.

19E.2.1.2.2.4 Operator Actions

The loss of normal AC power will lead to indirect turbine trip and reactor scram due to high condenser pressure on loss of circulating water. The subsequent loss of feedwater will cause the RPV to isolate on low water level. Failure of the emergency diesel generators to initiate and failure of the combustion gas turbine will leave the RCIC system as the only source of makeup water to the core. The RCIC system will automatically restore the RPV water level. Operator actions are specified in the EPGs to control the RCIC system and maintain the RPV level between Level 3 and Level 8.

In addition, the operator will be instructed to maintain RPV pressure below the high pressure scram setpoint to avoid SRV cycling by controlling 1 or more SRVs manually. The PCV pressure and temperature will not approach design values for at least 8 hours. Failure of the RCIC (core uncovery) will require the operator to blowdown through the SRVs when the heat capacity temperature limit is exceeded or the water level falls below the top of the active fuel and thereby avoid a high pressure as the core melts.

19E.2.1.2.2.5 Recovery Following Restoration of AC Power

All equipment necessary for restoration of power is located external to the primary and secondary containments in the reactor building. With the exception of the control building, all heat generating sources external to secondary containment are shutdown during station blackout so that the rooms should be at temperatures which allow restart of the support systems under their automatic or manual modes following restoration of AC power. Temperatures in the control building should be such that restart can be accomplished by the operators from the control room. Also, restart could be initiated from the remote shutdown panel or even by local control at the motor control centers and switchgear. Following restoration of power and initiation of the reactor cooling water system, the ECCS areas of secondary containment will be cooled by their safety grade room coolers so normal operation of the safe shutdown systems could be restored. The turbine building electrical systems and the non-safety-related secondary cooling system provide a backup means of restoring cooling to the ECCS equipment areas within secondary containment.

19E.2.1.2.2.6 Conclusions

The ABWR plant is being designed to be capable of maintaining core cooling and containment integrity for at least 8 hours following the loss of offsite and onsite AC electrical power including the combustion turbine generator. This capability assessment follows the general criteria of:

(1) Assuming no additional single failures

(2) Realistic analytical methods and procedures

A summary of the key plant parameters, design basis values and capability assessment is shown in Table 19E.2-2. Note that the response of the ABWR containment to this event would be successful even if the design basis values were exceeded, as long as the ultimate capability were not exceeded.

19E.2.1.2.3 Equipment Survivability

The requirements for equipment survivability are derived from two sources. 10CFR50.34(f) specifies the conditions required for an analysis in which the 100% of the active fuel cladding is oxidized. Additional requirements for demonstrating the survivability of equipment needed to mitigate a severe accident are specified in SECY-90-016. In order to meet these requirements, three categories of events were considered. The first category consists of one event which responds to the requirements of 10CFR50.34(f) paragraphs (2)(ix)(C) and (3)(v). A non-mechanistic scenario is modelled which results in the requisite oxidation but which follows the rules of design basis analysis. The other two categories respond to the requirements of SECY-90-016. The second category consists of events representing the frequency dominant events ending in in-vessel recovery. Similarly, category three is made up of events representing the frequency dominant events ending in ex-vessel recovery. Together the events in categories two and three represent 98% of the core damage frequency.

The list of required instrumentation and equipment was derived from reviews of the safe shutdown equipment list, the EPGs, the PRA, and the severe accident analysis. The list of required equipment varies for the three categories of events described above. The capability of each piece of identified equipment was then compared to the environmental conditions for the appropriate category of events. In reviewing the equipment capability, the environmental qualification standards for assessing compliance to 50.49 were not used as a strict measure. Rather, they were used to provide a measure of confidence that the equipment would survive the expected conditions.

19E.2.1.2.3.1 Definition of Survivability Profiles

For each of the three categories of events, a set of curves representing the bounding environmental conditions for that category were developed for use in evaluating the equipment and instrumentation survivability. These conditions were then compared to the equipment capabilities to provide a measure of confidence that the necessary equipment would survive the expected conditions. It is important to note that the ABWR containment is inerted for all of the events described below. Therefore, there is no containment challenge due to hydrogen burning or detonation.

The basis for each category of events is provided below along with a brief summary of the event progression.

19E.2.1.2.3.1.1 10CFR50.34(f) Category

This category corresponds to an event which could result in the conditions of 10CFR50.34(f)(2) (ix), which specifies that core cooling is degraded sufficiently to result in the generation of 100% oxidation of the active cladding. Core cooling is then recovered before the vessel fails. The PRA has confirmed the results of previous studies which show that the core damage frequency is dominated by accidents initiated from transients. Table 19.3-5 indicates that only 0.4% of all core damage events are initiated by LOCA. Therefore, a transient initiated event is specified for this evaluation.

Best estimate analyses do not result in oxidation of 100% of the active cladding. In order to simulate the hypothetical event, MAAP-ABWR was run using a multiplier to non-mechanistically generate oxidation of the active cladding. Additionally, ECCS was cycled on and off to produce the requisite amount of hydrogen for 100% metal-water reaction. The event progresses as follows:

- An isolation event occurs.
- All core injection is assumed to fail.
- Drywell and wetwell sprays are initiated 30 minutes after the initiation of the accident, water flow is directed through the RHR heat exchanger.
- The core begins to heat up and zirconium begins to oxidize.
- ECCS is recovered.
- Additional hydrogen is generated as the core is quenched.
- Vessel water level is recovered, terminating the event.

Curves representing the environmental conditions during this event are shown in Figures 19E.2-26a through 19E.2-26e. The vessel pressure remains within the range of

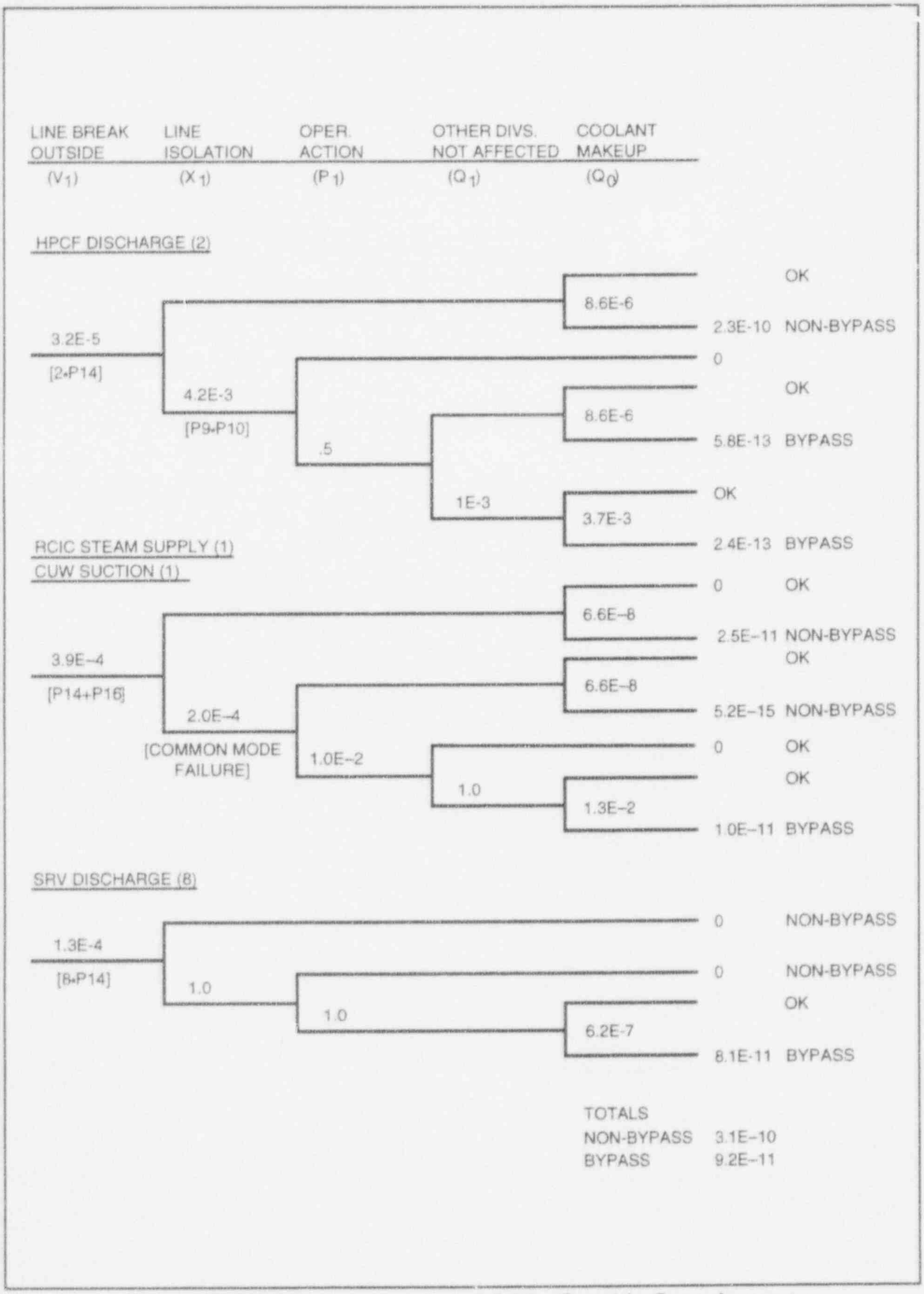


Figure 19E.2-20b Intermediate LOCAs Outside Containment

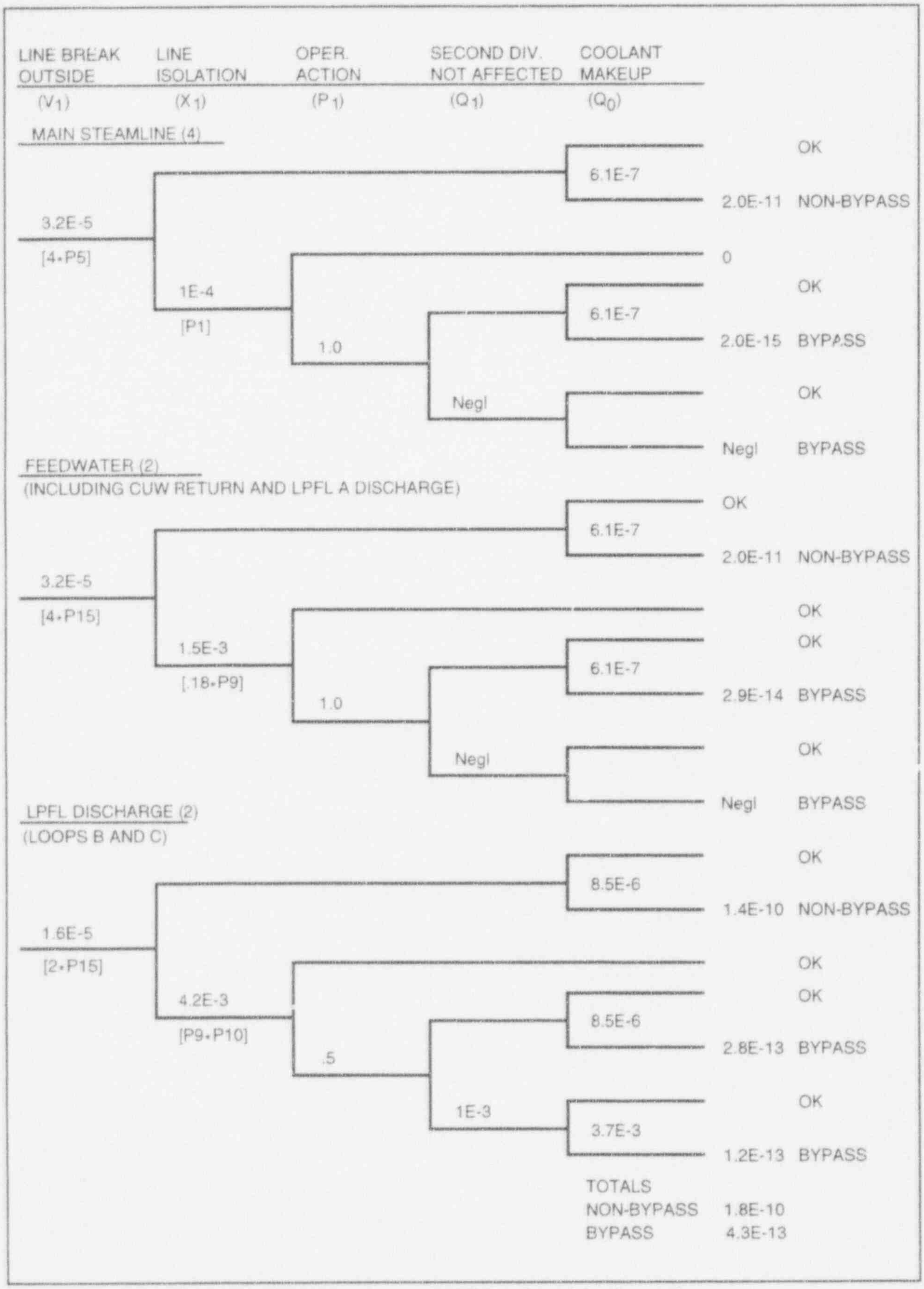


Figure 19E.2-20c Large LOCAs Outside Containment

19H.5 COL License Information

19H.5.1 Seismic Capacity

The COL applicant shall determine the HCLPF values for the plant-specific/as-designed components corresponding to those generic components defined in Subsection 19H.4.3. The values should be compared to their assumed HCLPF values given in Table 19H-1 (or Table 19I-1 on system basis). It should be noted that only the capacities of important contributors (Section 19.8) need to be determined and compared. These important contributions are hereafter referred to as SMA SSCs for systems, structures, and components needed for consideration in the seismic margins assessment.

An explicit evaluation of HCLPF values of only the important contributors (Section 19.8) need to be performed. However, prior to the HCLPF evaluation it is essential to verify that the quality of construction of structures and installation of equipment and systems are in conformance with the commitments in the SSAR and that the as-built structures systems and components meet all the applicable ITACC requirements. These important components are hereafter referred to as SMA SSCs for systems, structures, and components needed for consideration in seismic margins assessment.

The HCLPF calculations can be made using fragility analysis or the conservative deterministic failure margin (CDFM) approach. The location effects should be taken into account in determining the limiting capacity of the same component on different locations.

For structures, equipment and systems other than the important items metioned above, it is only necessary to verify that the site-dependent conditions are within the site envelope parameters in accordance with the procedure described in Subsection 2.3.1.2 or that site-specific SSE responses are bounded by those considered in the standard design, provided that the as-biult structures, systems and components are verified to be designed, constructed, installed and tested in accordance with the SSAR and ITAAC commitments. Otherwise, site-specific HCPLF capacities for these structures and components need to be established.

It is not necessary that in each case the HCLPF equal or exceed the value assumed in the margins analysis of the standardized design, especially since the NRC has judged that HCLPF=0.5 is acceptable. However, depending on the degree of difference and the significance of the component in accident sequences, an evaluation of the site-specific plant level HCLPF capacity may be needed. The level of acceptable seismic margin for the plant should be established in a manner consistent with that used in existing nuclear power plants.

The site should also be investigated for the potential of seismic-induced soil failure (liquefaction, differential settlement, or slope stability) at 1.67 times the site-specific SSE.

In order to increase confidence that the as-designed seismic capacities of the SMA SSCs are realized in the final constructed plant, a seismic walkdown shall be performed by the COL applicant according to the process as follows:

- Step 1—Preparation for Plant Walkdown
- Step 2—Plant Seismic Logic Model Walkdown
- Step 3—Assessment of As-Built SMA SSC HCLPF Values
- Step 4—Seismic Plant Walkdown
- Step 5—Plant Damage State and Plant Level HCLPF Calculations

These steps are discussed in detail in the remainder of this subsection.

Step 1—Preparation for Plant Walkdown

The SMA presented in Appendix 19I contains seismic logic models for the plant. These models include the seismic-induced failures that were considered necessary to be evaluated as part of the SMA. These failures, and the associated HCLPF values of the SMA SSCs shall be reviewed. In preparing for the plant walkdown, all appropriate information regarding these failures should be gathered. These include, but are not necessarily limited to:

- Piping and instrumentation drawings,
- Electrical one-line diagrams,
- Plant arrangement drawings,
- Detailed design drawings,
- Procurement specifications,
- Construction drawings (especially those concentrating on seismic detailing and load paths),
- Quality assurance records,
- Seismic analysis used for defining floor response spectra,
- Floor spectra used as required response spectra by vendors,
- Engineering analyses of seismic performance (especially for representative seismic anchorages), and
- Equipment qualification data/material test data.

Step 2—Plant Seismic Logic Model Walkdown

The walkdown will concentrate on the identification of potential systems interactions that could impact the performance of the front-line and support SSCs included in the models. The original SMA model considered in Appendix 19I included the most significant systems interactions (e.g., collapse of major buildings). However, it is necessary to assure that no other interactions exist in the as-built plant that were not

Table 19P-1 Offsite Accident Cases

Case*	Frequency (per yr)	Person-Sievert Exposure† (per event)	Contribution (per 60 yr)	(%)
Case 1	2.1E-08	138	1.7E–4	6.3
Case 2	7.8E-11	83.28	3.9E–7	0.01
Case 3	0	3714	0	0.00
Case 4	0	2064	0	0.00
Case 5	7.5E-12	933.80	4.2E–7	0.02
Case 6	3.1E-12	24,160	4.5E–6	0.17
Case 7	3.9E-10	27,260	6.4E–4	23.8
Case 8	4.1E-10	32,020	7.9E–4	29.3
Case 9	1.7E-10	33,120	3.4E–4	12.6
NCL	1.3E-07	96	7.5E–4	27.8
Total	1.6E-07		26.9E–4	100.0

* For case descriptions see Table 19E.3-6; frequencies are based on Table 19P-2.

† Average of regional values used; see Subsection 19E.3.

Table 19P-2 Core Damage Frequency Contributors*

Init. Event	1A	1B1	1B2	1B3	1D	II	IIID	IV	Total	% Contrib
	Event Sequence									
Scram	1.1E-08				4.3E-10	9.5E-13			1.1E-08	7.3
Turb Trip	6.8E-09				2.7E-10	3.7E-11			7.1E-09	4.5
Isolation	1.8E-08				7.1E-10	1.1E-11			1.9E-08	11.9
LOOP2	4.1E-09				1.5E-11	4.2E-13			4.1E-09	2.6
LOOP8	2.4E-09				9.6E-12	1.4E-12			2.4E-09	1.5
LOOP8+	5.8E-10				1.1E-09	6.0E-11			1.7E-09	1.1
SBO2	6.6E-12				6.7E-08				6.7E-08	42.9
SBO8		2.6E-08							2.6E-08	16.7
SBO8+			1.5E-08	8.9E-10					1.6E-08	10.3
IORV	1.1E-09				2.0E-10	9.5E-13			1.3E-09	0.8
SBLOCA							2.5E-10		2.5E-10	0.2
ATWS								1.5E-10	1.5E-10	0.1
TOTAL	4.4E-08	2.6E-08	1.5E-08	8.9E-10	7.0E-08	1.1E-10	2.5E-10	1.5E-10	1.57E-07	100

Offsite Release Group**	LCHP	SBRC	LCLP	LHRC	LBLC	ATWS	Total Case
Case 1	3.4E-09	7.9E-10	1.6E-08		5.1E-11		2.0E-08
Case 2			7.8E-11				7.8E-11
Case 3	1.3E-12						1.3E-12
Case 4							0
Case 5					6.3E-12		6.3E-12
Case 6	1.2E-10						1.2E-10
Case 7	1.1E-10		2.6E-10				3.70E-10
Case 8	2.1E-10						2.1E-10
Case 9				1.1E-12		1.5E-10	1.5E-10
NCL (N)	4.0E-08	1.5E-08	8.0E-08		2.0E-10		1.4E-07
Total	**4.4E-08**	**1.6E-08**	**9.6E-08**	**1.1E-12**	**2.5E-10**	**1.5E-10**	**1.57E-07**
Contrib %	28.1	10.3	61.4	0.122	0.2	0.1	100

* Derived from PRA event trees (19D4 and 19D5); some numerical differences exist due to simplification.

** For description see Subsection 19E.2.2

to the second floor level and may start flowing into the other two remaining divisional RCW/RSW rooms. The level sensors in these two divisional rooms will generate a signal to alert the operator about the flood. If the sensors in the first division failed, the sensors in the other divisions are assumed to fail with a probability of 0.1 to account for common cause failures (CCF). Only one division is assumed lost if the operator is successful in isolating the flood, otherwise the loss of all three safety divisions is possible.

Flooding in the RSW pump house was not addressed because the ultimate heat sink including the RSW pump house is outside the scope of GE supply. The COL applicant must complete a plant specific probabilistic analysis of flooding in the RSW pump house.

The event tree in Figure 19R-9 describes the accident progression for a control building RSW flood. A large pipe break in the RSW in the RSW/RCW heat exchanger room is considered to be the worst case flooding in the control building. The description of events shown in Figure 19R-9 follows:

(1) A large RSW pipe break occurs in the RCW/RSW room in the control building (flooding initiator).

(2) Four redundant safety grade water level sensors located at the 0.4 m level detect and alert the control room operator about flooding (detection).

(3) The operator investigates the presence of water and isolates the flooding by tripping the affected pump and/or closing the isolation valve (flooding prevention).

(4) If the first level of detection fails or the operator fails to isolate the flowing water, then water continues rising in the room and the second set of diverse sensors located at 1.5 meters detects the water and trips the affected pump and closes all motor operated valves in the RSW system. Meanwhile the signal alerts the control room operator of the flooding condition (flooding prevention).

(5) If the operator is successful in isolating the flooding, one safety division is assumed lost, otherwise the loss of all three safety divisions may occur (flooding mitigation).

(6) The pump breaker of the affected RSW pump opens to trip the pump and/or the isolation valves close automatically (flooding isolation).

(7) In the unlikely event that the flood is not mitigated by automatic means or operator action, the water rises to the second floor level and starts flowing into the other two remaining RCW/RSW rooms. The first set of level sensors in these two divisional rooms detects the water and alerts the operator the third time (flooding detection). This operator action is considered separately from

the previous high water alarm action because the alarm would occur approximately 45 minutes later and be annunciated as occurring from a different division.

(8) Reactor safe shutdown using available equipment (reactor shutdown).

The core damage probability for an RSW flood is estimated to be approximately 9.7E-9 per reactor year.

19R.5.4.2 Fire Water System Breaks

The ABWR fire water system is a moderate energy system that is designed to withstand a 0.3g seismic event. The system is very rugged and large breaks of the eight inch header piping are not expected. The most probable failure mechanism for the fire water piping would be a crack which would not propagate to a large break because of the low pressure of the system (approximately 0.69 MPa). In keeping with the bounding analysis methodology used in other parts of the flooding PRA, a large break (0.086 m^3/sec) will be assumed. The frequency of this bounding case large break of the fire water piping is 5E-4/year. This value was obtained from a review of the Limerick Generating Station flooding PRA for a fire suppression system pipe double ended shear.

Figure 19R-10 is the event tree for fire water system flooding in the Control Building. The system unavailabilities are taken from the ABWR full power PRA and the operator failure probabilities are based on methods used in Chapter 10 of Swain and Guttman given the many sources of information available indicating a fire water system break and the simple action of stopping the fire water pumps.

Fire water standpipes are located on all floors of the Control Building. A large break on any upper floor will result in a 0.086 m^3/sec gpm flood which will be directed by the floor drain system to the RCW rooms on the first floor.

The ABWR does not contain sills on doors between safety divisions and fire doors can have up to a 1.9 cm gap at the bottom per National Fire Protection Association (NFPA 80) requirements. The floor drain system, although not finalized yet, will be designed so that this break flow can be accommodated taking into account all the drain lines in the three safety divisions. In addition, water may flow under the fire doors and down stairwells and elevator shafts. Due to the available drainage sources, the water level on any upper floor will not exceed 20.32 cm which is the minimum height that all water sensitive equipment must be mounted from the floor (ABWR SSAR Section 3.4). Therefore, no damage to equipment on any of the upper floors will occur due to emersion in the flood water. Spray onto safety-related equipment is not a concern because all fire water system flow will be directed to the three RCW rooms on the first floor.

	Analysis Value	Nominal Value
b. Vessel Dome Pressure - MPaG	7.17	7.07
c. Steam Flow - kg/h - % NBR	$7.84x10^6$ 102.7	$7.64x10^6$ 100

The operating power and steam flow will be limited by the operating license to their nominal values. Therefore, the analysis conditions are conservative.

(2) Equipment Performance Characteristics

In the analysis, the specified limiting equipment performance characteristics are used. These include maximum delay time, maximum response time, fastest/slowest valve spring/closing characteristics. These characteristics will be checked at startup. Therefore, the analysis is conservative.

(3) Safety Setpoints

All safety setpoints used in the analysis will be treated as analytical limits. Then the setpoint methodology approved by the NRC will be used to determine the allowable valves and nominal setpoints. These valves will be included in the technical specifications.

Regarding the nuclear conditions used in the analysis, current ABWR core and fuel design with 8x8N fuel lattice as described in GESTAR II is used. If the core and fuel design changes in the future ABWR application, limiting events will be reanalyzed to determine the operating limits as described in Subsection 15.0.4.5. Since margins have been provided in signing ABWR equipment, it is expected that ABWR can accommodate any new fuel designs.

Question 440.109

Provide an analysis of the loss of instrument air (nitrogen). (15)

Response 440.109

Loss of instrument air systems does not result in any transient more severe than MSIV closure occurs. After the MSIV closure, the reactor pressure is regulated by the safety/relief valves.

Loss of air to SJAE steam valves results in loss of condenser vacuum, which is covered in the SSAR. Availability of safety/relief valve helps reduce severity of any transient that may occur as a result of loss of air.

Primary containment purge isolation valves are not safety-related, and their closure due to loss of air does not result in any NSSS transient more severe than MSIV closure. This is also true for the closure of the ventilation supply and exhaust isolation valves for the reactor building secondary containment, closure of all other air operated isolation valves, and the stoppage of control room exhaust fan.

The most severe results of loss of instrument air, as evidenced from the above discussion, are reactor scram and isolation. This MSIV closure with concurrent scram is a transient which is bounded by the MSIV closure trip scram transient, as analyzed in the SSAR, with respect to thermal and pressure limits. Since the loss of air could, at worst, result in reactor scram and isolation, there is no potential for causing or compounding more severe events.

Question 440.110

In SSAR Table 15.0-2, the following transients are not categorized as moderate frequency event [Category (a)]

(1) Runout of two feedwater pumps (Cat.c)

(2) Opening of all Control and Bypass Valves (Cat.c)

(3) Pressure Regulator Downscale failure (Cat.c)

(4) Generator Load Rejection, Failure of One Bypass Valve (Cat.b)

(5) Generator Load Rejection with Bypass Off (Cat.c)

(6) Turbine Trip with Failure of One Bypass Valve (Cat.b)

(7) Turbine trip, Bypass Off (Cat.c)

(8) Loss of Aux. Power Transformer and one S/up transformer (Cat c)

(9) Trip of all Reactor Internal Pumps (Cat.c)

(10) Fast Runback of all Reactor Internal Pumps (Cat.c)

(11) Inadvertent HPCF pump startup (Cat.b)

Category b refers to Infrequent event and Category c refers to limiting faults.

The above categorization of transients is a significant deviation from the SRP and hence sufficient justification must be submitted to support the change in the categorization. (15)

CDM MODIFICATION PAGES

suction valve, and injection pump is powered from Division I. The power supply to the other motor-operated injection valve, suction valve, and injection pump is powered from Division II. In the SLC system, independence is provided between Class 1E divisions, and also between Class 1E divisions and non-Class 1E equipment.

The SLC System has the following displays, controls and alarms in the main control room:

- Alarms for storage tank temperature and level.
- Parameter displays for the instruments shown on Figure 2.2.4.
- Controls and status indication for the pumps, injection valves, and suction valves.
- A manual system initiation switch for each division.

The motor-operated valves (MOVs) shown on Figure 2.2.4 have an active safety-related function and perform this function under differential pressure, fluid flow and temperature conditions.

The check valves (CVs) shown on Figure 2.2.4 have active safety-related functions to open, close, or both open and close under system pressure, fluid flow, and temperature conditions.

The SLC System is physically separated from and independent of the hydraulic portion of the Control Rod Drive (CRD) System.

The piping and components on the suction side of the pumps up to and including the suction valves and the test loop up to the test tank inlet valve have a design pressure of 2.82 MPaG for intersystem LOCA (ISLOCA) conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.4 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the SLC System.

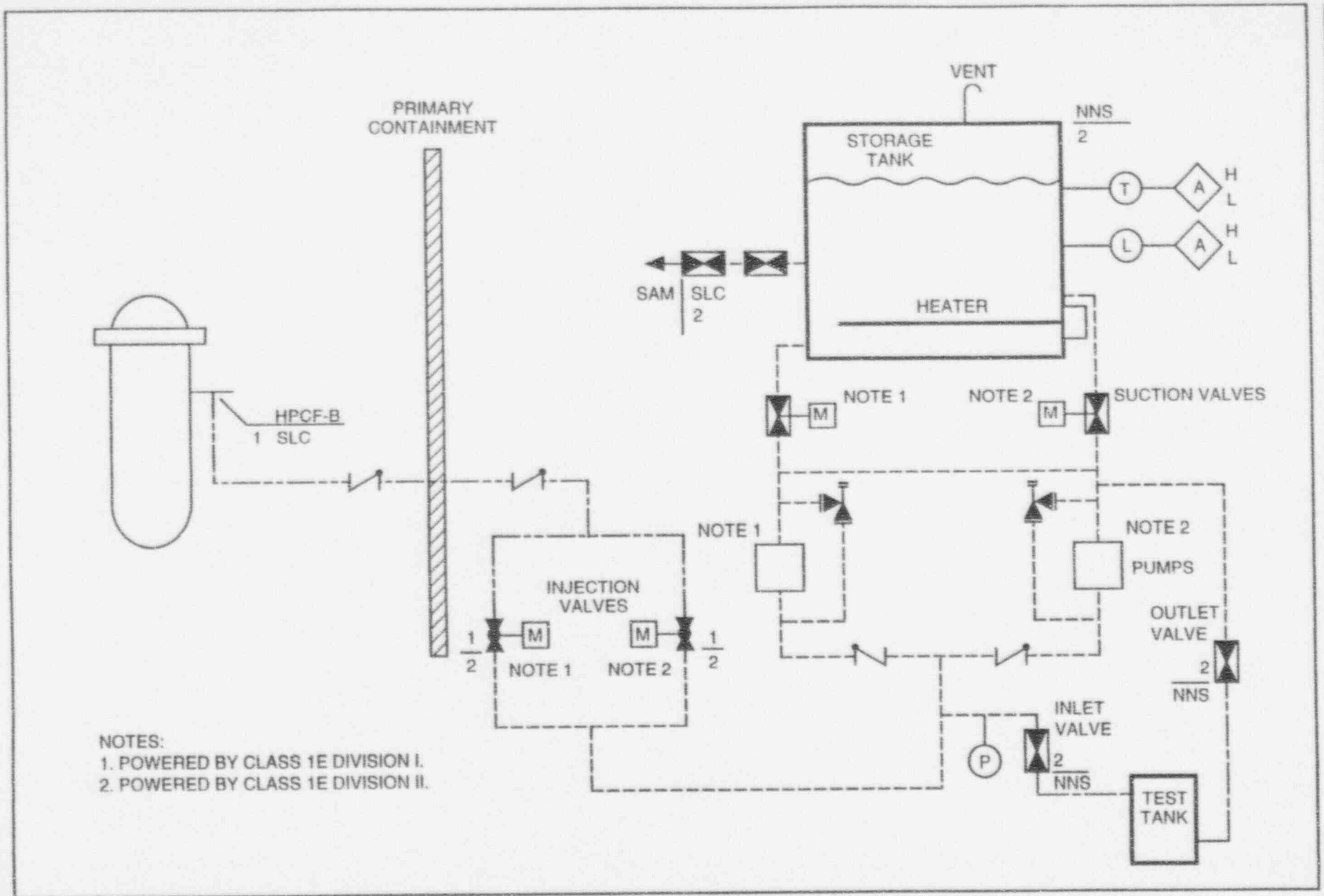


Figure 2.2.4 Standby Liquid Control System

Table 2.2.4 Standby Liquid Control System

Inspections, Tests, Analyses and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the SLC System is shown in Figure 2.2.4.	1. Inspections of the as-built system will be conducted.	1. The as-built SLC System conforms with the basic configuration shown in Figure 2.2.4.
2. The ASME Code components of the SLC System retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. A hydrostatic test will be conducted on those Code components of the SLC System that are required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the SLC System conform with the requirements in the ASME Code, Section III.
3.	3.	3.
a. A test tank and associated piping and valves permit testing of the SLC System during plant operation. The tank is supplied with demineralized water, which is pumped in either a closed loop or is injected into the reactor.	a. Tests will be conducted on each division of the as-built SLC System using installed controls, power supplies and other auxiliaries. The following tests will be conducted:	a.
	(1) Demineralized water will be pumped against a pressure greater than or equal to 8.72 MPaA in a closed loop on the test tank.	(1) Demineralized water is pumped with a flow rate greater than or equal to 189 L/min in the closed loop.
	(2) Demineralized water will be injected from the test tank into the reactor.	(2) Demineralized water is injected from the test tank into the reactor.
b. The SLC System delivers at least 378 L/min of solution with both pumps operating when the reactor pressure is less than or equal to 8.72 MPaA.	b. Tests will be conducted on the as-built SLC System using installed controls, power supplies and other auxiliaries. Demineralized water will be injected from the storage tank into the reactor with both pumps running against a discharge pressure of greater than or equal to 8.72 MPaA.	b. The SLC System injects greater than or equal to 378 L/min into the reactor with both pumps running against a discharge pressure of greater than or equal to 8.72 MPaA.

Table 2.2.4 Standby Liquid Control System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
c. The SLC System delivers at least 189 L/min of solution with either pump operating when the reactor pressure is less than or equal to 8.72 MPaA.	c. Tests will be conducted on the as-built SLC System using installed controls, power supplies and other auxiliaries. Demineralized water will be injected from the storage tank into the reactor with one pump running against a discharge pressure of greater than or equal to 8.72 MPaA.	c. The SLC System injects greater than or equal to 189 L/min into the reactor with either pump running against a discharge pressure greater than or equal to 8.72 MPaA.
d. The SLC System can be manually initiated from the main control room.	d. Tests will be conducted on the as-built SLC System using the manual initiation switch.	d. Each division of the SLC System initiates when the manual initiation switch for that division is actuated.
e. Both divisions of the SLC System are automatically initiated during an ATWS.	e. Tests will be conducted on the as-built SLC System using simulated ATWS signals.	e. Upon receipt of a simulated ATWS signal, both divisions of SLC automatically initiate.
f. Each SLC System pump has an interlock which prevents operation if both the test tank outlet valve and the pump suction valve are closed.	f. Tests will be conducted on each SLC System pump start logic using simulated valve position signals	f. Each SLC System pump is prevented from operating unless signals indicative of one of the following conditions exist: (1) A suction path from the storage tank is available (the pump suction valve is fully open). (2) A suction path from the test tank is available (the test tank outlet valve is fully open).

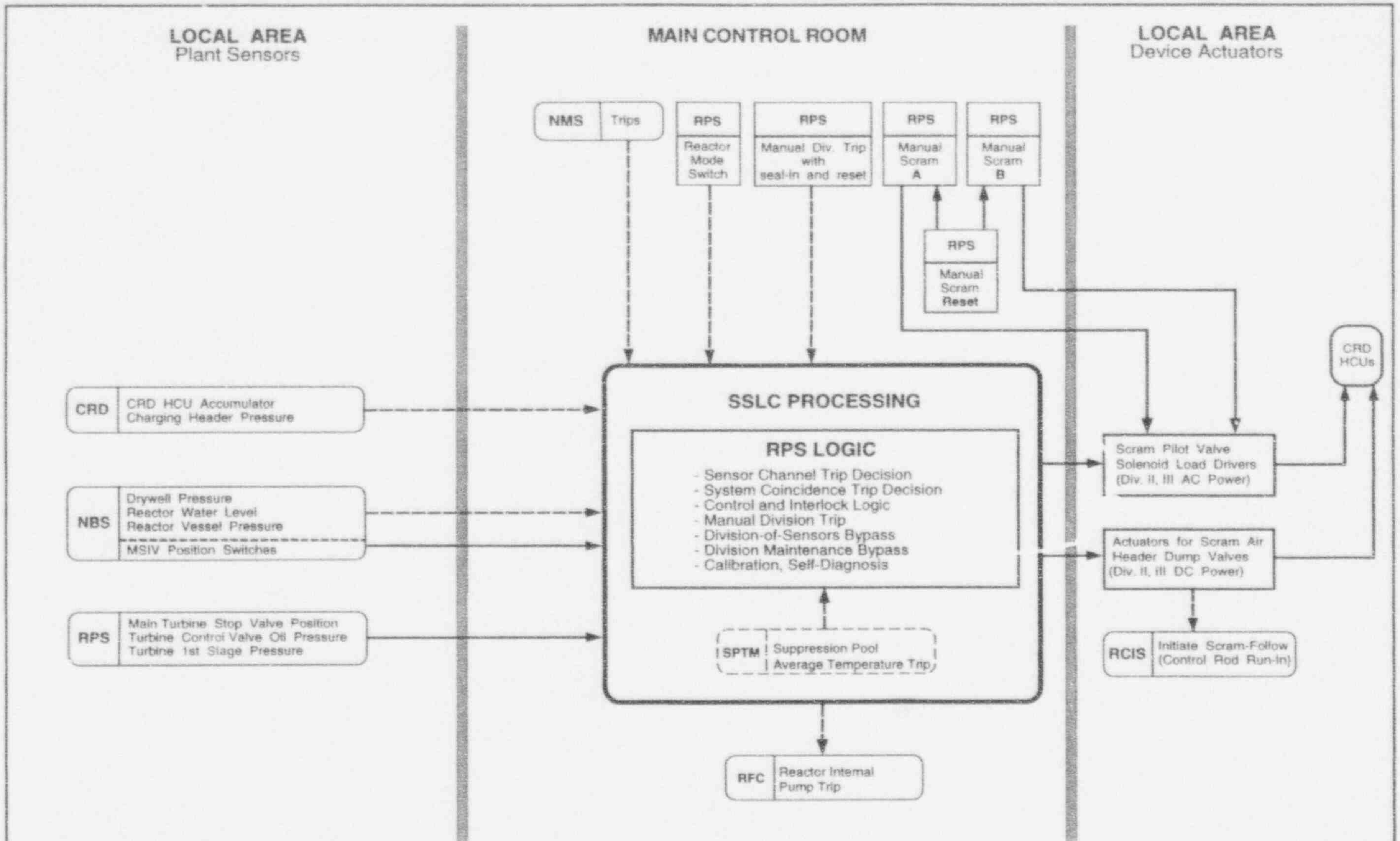
The RPS design is single-failure-proof and redundant. Also, the RPS design is fail-safe in the event of loss of electrical power to one division of RPS logic.

Each of the four RPS divisional logic and associated sensors are powered from their respective divisional Class 1E power supply. In the RPS, independence is provided between Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment.

As shown on Figure 2.2.7a, the RPS has manual divisional trip switches, reactor mode switch, manual scram switches, and scram reset switches for manual controls. Divisional trip displays, and scram solenoids electrical power status lights are also provided. These RPS controls and displays are provided in the main control room. Fail safe RPS sensors are turbine control valve oil pressure switches, turbine stop valve position switches, and turbine first-stage pressure sensors. These sensors are located in the Turbine Building.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.7 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be performed for the RPS.



NOTES:
1. Diagram represents one of four divisions.

Figure 2.2.7a Reactor Protection System Control Interface Diagram

2.6.2 Fuel Pool Cooling and Cleanup System

Design Description

The Fuel Pool Cooling and Cleanup (FPC) System (Figure 2.6.2) removes decay heat generated by the spent fuel assemblies in the spent fuel storage pool. The system also maintains the water quality and monitors and maintains the water level above the spent fuel in the spent fuel storage pool. Figure 2.6.2 shows the basic FPC System configuration and scope.

The FPC System is classified non-safety-related, except for piping connections and valves for safety-related fuel pool makeup and supplemental cooling by the Residual Heat Removal (RHR) System.

The safety-related makeup water source for the spent fuel storage pool is provided by the RHR System, which pumps suppression pool water to the FPC System.

The spent fuel storage pool has no piping connections (inlet, outlet, drains or other piping) located below a point 3m above the top of active fuel located in the spent fuel storage racks.

The FPC System components, with the exception of the filter/demineralizer unit, are classified as Seismic Category I. Figure 2.6.2 shows the ASME Code class for the FPC System piping and components.

The FPC System is located in the Reactor Building.

The FPC System has parameter displays in the main control room for instruments shown on Figure 2.6.2.

The check valves (CVs) shown on Figure 2.6.2 have active safety-related functions to open, close, or both open and close under system pressure, fluid flow, and temperature conditions.

The piping and components of the FPC System at the suction side of the RHR System from the upstream isolation valve have a design pressure of 2.82 MPaG for intersystem LOCA (ISLOCA) conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.2 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria, which will be undertaken for the FPC System.

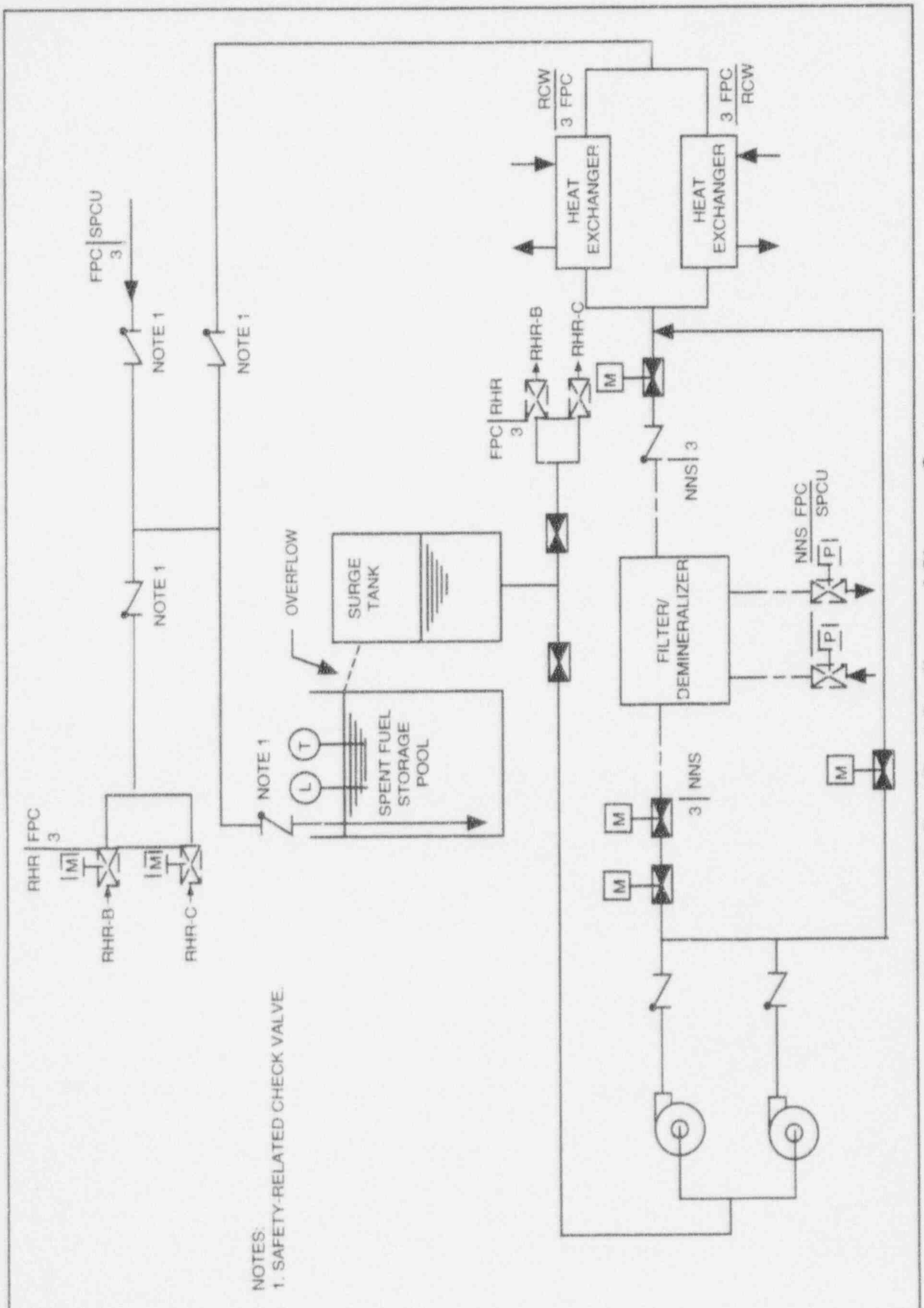


Figure 2.6.2 Fuel Pool Cooling and Cleanup System

Table 2.11.6 HVAC Emergency Cooling Water System

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The basic configuration for the HECW System is shown on Figures 2.11.6a and 2.11.6b.	1. Visual inspections of the as-built system configuration will be conducted.	1. The as-built configuration of the HECW System is in accordance with Figures 2.11.6a and 2.11.6b.
2. The ASME Code components of the HECW System retain their integrity under internal pressures that will be experienced during service.	2. A hydrostatic test will be conducted on those Code components of the HECW System required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the HECW System conform with the requirements in the ASME Code, Section III.
3. Each HEWC System refrigerator unit has a capacity of not less than 2.43 GJ/h.	3. Type tests will be conducted on an as-built HECW System refrigerator units at a test facility.	3. Each HEWC System refrigerator unit has a capacity of not less than 2.43 GJ/h.
4. In Divisions B and C, any refrigerator unit on standby automatically starts if any of the other refrigerator units in Divisions B or C are stopped.	4. Tests will be conducted on each as-built HECW System refrigerator unit in Divisions B and C, using simulated signals indicating another refrigerator unit is stopped.	4. In Divisions B and C, the refrigerator unit on standby automatically starts upon receipt of a simulated signal indicating that the other refrigerator units in Divisions B or C are stopped.
5. Each of the three HECW System divisions is powered from the respective Class 1E divisions as shown on Figures 2.11.6a and 2.11.6b. In the HECW System, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	5. a. Tests will be performed on the HECW System by providing a test signal in only one Class 1E division at a time. b. Inspections of the as-built Class 1E divisions in the HECW System will be performed.	5. a. The test signal exists only in the Class 1E division under test in the HECW System. b. In the HECW System, physical separation or electrical isolation exists between Class 1E divisions. Physical separation or electrical isolation exists between these Class 1E divisions and non-Class 1E equipment.

Table 2.11.6 HVAC Emergency Cooling Water System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
6. Except for the connections to the chemical addition tank, each mechanical division of the HECW System (Divisions A, B, C) is physically separated from the other divisions.	6. Inspections of the as-built HECW System will be conducted.	6. Each mechanical division of the HECW System is physically separated from the other mechanical divisions of the HECW System by structural and/or fire barriers, with the exception connections to the chemical addition tank.
7. Main control room displays and controls provided for the HECW System are as defined in Section 2.11.6.	7. Inspections will be performed on the main control room displays and controls for the HECW System.	7. Displays and controls exist or can be retrieved in the main control room as defined in Section 2.11.6.
8. CVs designated in Section 2.11.6 as having an active safety-related function open, close, or both open and close under system pressure, fluid flow, and temperature conditions.	8. Tests of installed valves for opening, closing, or both opening and closing, will be conducted under system preoperational pressure, fluid flow, and temperature conditions.	8. Based on the direction of the differential pressure across the valve, each CV opens, closes, or both opens and closes, depending upon the valve's safety functions.
9. The pneumatic-operated valves shown in Figures 2.11.6a and 2.11.6b fail as follows in the event that either electric power to the valve actuating solenoid is lost or pneumatic pressure to the valve is lost: the differential pressure control valves fail closed, and the flow control valves to the cooling coils fail open.	9. Tests will be performed on the as-built valves by initiating loss of pneumatic pressure and power to the actuating solenoids.	9. The pneumatic actuated valves listed below fail as specified when either electric power to the valve actuating solenoid is lost or pneumatic pressure to the valve is lost: the differential pressure control valves fail closed, and the flow control valves to the cooling coils fail open.

2.11.9 Reactor Service Water System

Design Description

The Reactor Service Water (RSW) System removes heat from the Reactor Building Cooling Water (RCW) System and rejects this heat to the Ultimate Heat Sink (UHS). The portions of the RSW System that are in the Control Building are within the Certified Design. Those portions of the RSW System that are outside the Control Building are not in the Certified Design. Figure 2.11.9a shows the basic system configuration and scope within the Certified Design. Figure 2.11.9b shows the RSW System control interfaces.

The RSW System provides cooling water flow to either two or three of the RCW System heat exchangers in each division. On a loss-of-coolant accident and/or loss of preferred power (LOCA and/or LOPP) signal, any closed valves for standby heat exchangers are automatically opened and cooling flow is provided to all three heat exchangers in each division.

For each division of the RSW System, the heat exchanger inlet and outlet valves close upon receipt of a signal indicating Control Building flooding in that division.

The RSW System is classified as Seismic Category I and ASME Code Section III, Class 3 and consists of three separate safety-related divisions.

Each of the three RSW divisions is powered by its respective Class 1E division. In the RSW System, independence is provided between Class 1E divisions, and also between the Class 1E divisions and non-Class 1E equipment. Each mechanical division of the RCW system (Divisions A, B, C) is physically separated from the other divisions.

The RSW System has the following main control room (MCR) displays and controls: control and status displays for the valves shown on Figure 2.11.9a. The RSW System components with status displays and control interfaces with the Remote Shutdown System (RSS) are identified in Figure 2.11.9a.

The motor-operated valves (MOVs) shown on Figure 2.11.9a all have active safety-related functions to open and close under differential pressure and fluid flow conditions.

Interface Requirements

Part of the RSW System that are not within the Certified Design shall meet the following requirements:

(1) Design features shall be provided to limit the maximum flood height to 5.0 meters in each RCW heat exchanger room.

(2) The design shall have three divisions which are physically separated. For any structure(s) housing RSW System components, there shall be inter-divisional boundaries (including walls, floors, doors and penetrations) that have three-hour fire rating. In addition, there shall be inter-divisional flood control features which preclude flooding from occuring in more than one division. Each division shall be powered by its respective Class 1E division. Each division shall be capable of removing the design heat capacity (as specified in Section 2.11.3) of the RCW heat exchangers in its division.

(3) Upon receipt of a loss-of-coolant (LOCA) signal, components in standby mode shall start and/or align to the operating mode.

(4) RSW System Divisions A and B shall have control interfaces with the Remote Shutdown System (RSS) as required to support RSW operation during RSS design basis conditions.

(5) If required by the elevation relationships between the UHS and the RSW System components in the Control Building (C/B), the RSW System shall have antisiphon capability to prevent a C/B flood after an RSW System break and after the RSW System pumps have been stopped.

(6) RSW System pumps in any division shall be tripped on receipt of a signal indicating flooding in that division of the C/B basement area.

(7) Any tunnel structures used to route RSW System piping to the Control Building shall be classified as Seismic Category I. Tunnel flooding due to site flood conditions shall be precluded.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.9 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the portions of the RSW System within the Certified Design.

Table 2.14.1 Primary Containment System

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The basic configuration of the PCS is as shown on Figure 2.14.1.	1. Inspections of the as-built system will be conducted.	1. The as-built PCS conforms with the basic configuration shown on Figure 2.14.1.
2. The primary containment pressure boundary defined in Section 2.14.1 is designed to meet ASME Code, Section III requirements.	2. Inspections of ASME Code required documents will be conducted.	2. An ASME Code Certified Stress Report exists for the pressure boundary components.
3. The ASME Code pressure boundary components of the PCS retain their integrity under internal pressures that will be experienced during service.	3. A structural integrity test (SIT) will be conducted on the pressure boundary components of the PCS per ASME Code requirements.	3. The results of the SIT of the pressure boundary components conform with the requirements of the ASME Code.
4. The maximum calculated pressures and temperatures for the design basis accident are less than design conditions.	4. Analyses of the design basis accident will be performed using as-built PCS data.	4. The maximum calculated pressures and temperatures are less than design conditions.
5. The primary containment pressure boundary including penetrations and isolation valves has a leak rate equal to or less than 0.5% per day (excluding MSIV leakage) of containment gas mass at the maximum calculated containment pressure for the design basis accident.	5. An integrated leak rate test of the primary containment will be conducted.	5. The primary containment pressure boundary including penetrations and isolation valves has a leak rate equal to or less than 0.5% per day (excluding MSIV leakage) of containment gas mass at the maximum calculated containment pressure for the design basis accident.
6. The design differential pressure of the diaphragm floor between the drywell and wetwell is 172.6 kPa in the downward direction.	6. An SIT will be conducted of the diaphragm floor with the drywell pressure greater than wetwell pressures by 1.0 times the design differential pressure.	6. An SIT report exists concluding that the diaphragm floor is able to withstand the design differential pressure.
7. The horizontal vent system consists of 30 vents configured as described in Section 2.14.1.	7. Inspection of the installed horizontal vent system will be conducted.	7. Confirmation that horizontal vent system is configured as described in Section 2.14.1.
8. MCR displays and alarms provided for the PCS are as defined in Section 2.14.1.	8. Inspections will be performed on the MCR displays and alarms for the PCS.	8. Displays and alarms exist or can be retrieved in the MCR as defined in Section 2.14.1.

Table 2.14.1 Primary Containment System (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
9. The vacuum breaker position switches have adequate sensitivity to detect the allowable S/P bypass capability of the containment.	9. Analysis of the as-built vacuum breakers will be performed. These analyses will determine the maximum vacuum breaker flow area (drywell-to-wetwell) which could exist undetected by the as-installed position switches. The loss coefficients associated with the flow area will be evaluated on the basis of the drywell-to-wetwell flow path geometric details. The flow area and loss coefficients will be combined into an overall drywell- to-wetwell $A\sqrt{K}$ factor which will be compared to the allowable value.	9. The vacuum breaker position switches have adequate sensitivity to detect the allowable S/P bypass capability of the containment.
10. The water volume in the suppression pool including the vents is equal to or greater than 3580 m^3.	10. Analyses of the as-built PCS will be performed.	10. The water volume in the suppression pool including the vents is equal to or greater than 3580 m^3.
11. The SRVDL quencher arms are located at or below the elevation of the center layer of horizontal vents in the suppression pool. The quenchers are placed in the suppression pool in two radial ring. Eighteen of 10 equally spaced locations in each radial ring have quenchers installed.	11. Inspection of the installed SRVDL quenchers will be conducted.	11. The SRVDL quenchers are located within the suppression pool as described in Section 2.14.1.
12. The corium protection fill contains less than 4% of calcium carbonate material by weight.	12. Tests will be performed on corium protection fill materials to determine the calcium carbonate content in a test facility	12. Corium protection fill contains less than 4% of calcium carbonate material by weight.
13. Lower drywell imbedded sumps are protected by corium shields.	13. Inspections of the lower drywell sump corium protection shields will be performed.	13. Lower drywell imbedded sumps are protected by corium shields.

The pneumatically-operated secondary containment isolation dampers, shown on Figure 2.15.5j, fail to the closed position in the event of loss of pneumatic pressure or loss of electrical power to the valve actuating solenoids.

R/B Primary Containment Supply/Exhaust System

The R/B Primary Containment Supply/Exhaust System removes inert atmosphere and provides air for primary containment prior to personnel entry, and consists of a supply fan, a filter unit, and an exhaust fan as shown on Figure 2.15.5j.

The R/B Primary Containment Supply/Exhaust System is classified as non-safety-related. The R/B Primary Containment Supply/Exhaust System is located in the secondary containment

R/B Main Steam Tunnel HVAC System

The R/B Main Steam Tunnel HVAC System provides cooling to the main steam tunnel and consists of two FCUs. Each FCU has two fans. The FCUs are started manually.

The R/B Main Steam Tunnel HVAC System is classified as non-safety-related. The R/B Main Steam Tunnel HVAC System is located in the Reactor Building.

R/B Non-Safety-Related Equipment HVAC System

The R/B Non-Safety-Related Equipment HVAC System provides cooling to the non-safety-related equipment rooms. There are six fan coil units, and four air handling units in the system, each consisting of cooling coil and fans.

The R/B Non-Safety-Related Equipment HVAC System is classified as non-safety-related, and is located in the Reactor Building.

Reactor Internal Pump ASD HVAC System

The Reactor Internal Pump ASD HVAC System provides cooling to the RIP ASD power panels. The system consists of a two recirculating air conditioning units with cooling coils and four supply fans.

The RIP ASD HVAC System is classified as non-safety-related, and is located in the Reactor Building.

Turbine Island HVAC System

The Turbine Island HVAC System provides heating, cooling, and ventilation for the Turbine Island. The Turbine Island HVAC System consists of the following non-safety-related systems.

(1) Turbine Building (T/B) HVAC System.

(2) Electrical Building (E/B) HVAC System.

Turbine Building (T/B) HVAC System

The T/B HVAC System provides cooling and ventilation for the Turbine Building. The T/B HVAC System consists of:

(1) T/B supply system with an air conditioning unit and three supply fans.

(2) T/B exhaust system with three exhaust fans.

(3) T/B compartment exhaust system with two exhaust fans.

(4) T/B lube oil area exhaust system with two fans.

(5) T/B unit coolers and electric unit heaters.

The T/B HVAC System is classified as non-safety-related. The T/B HVAC System is located in the Turbine Building.

Electrical Building (E/B) HVAC System

The E/B HVAC System provides cooling and ventilation for the electrical equipment rooms. The system consists of two air conditioning units, supply fans, two exhaust fans, unit coolers and electric unit heaters.

The E/B HVAC System is classified as non-safety-related. The E/B HVAC System is located in the Electrical Building of the Turbine Island.

Radwaste Building HVAC System

The Radwaste Building HVAC System provides a controlled environment for personnel comfort and safety for the Radwaste Building areas. The system consists of:

(1) An air conditioning unit and two supply fans for the Radwaste Building control room

(2) An air conditioning unit with, two supply fans, and three exhaust fans for the process areas of the Radwaste Building.

The Radwaste Building HVAC System is classified as non-safety-related, and is located in the Radwaste Building.

Service Building HVAC System

The Service Building (S/B) HVAC System provides controlled environment for personnel comfort in the S/B.

The S/B HVAC System consists of two non-safety-related systems:

(1) Clean Area HVAC System.

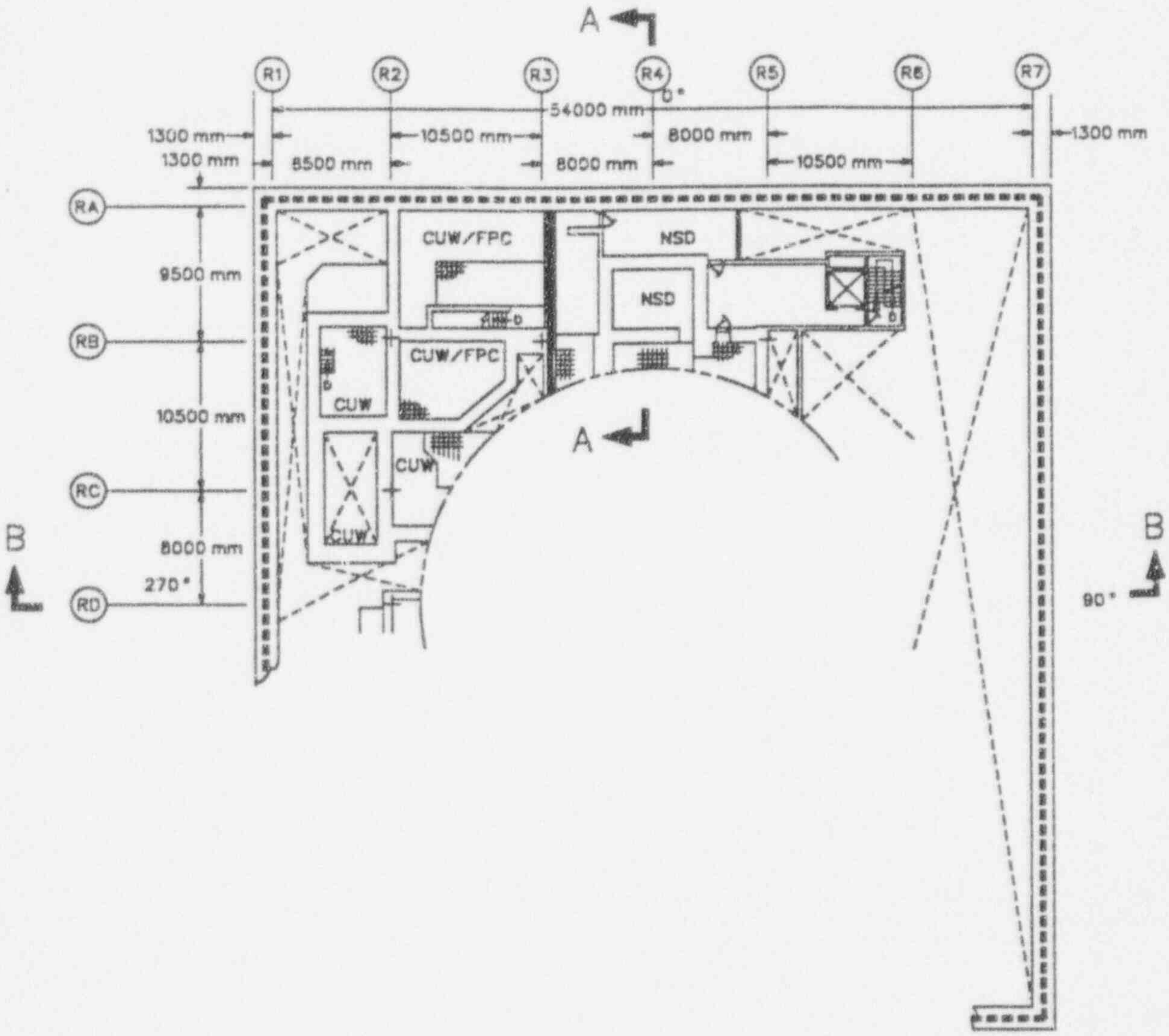
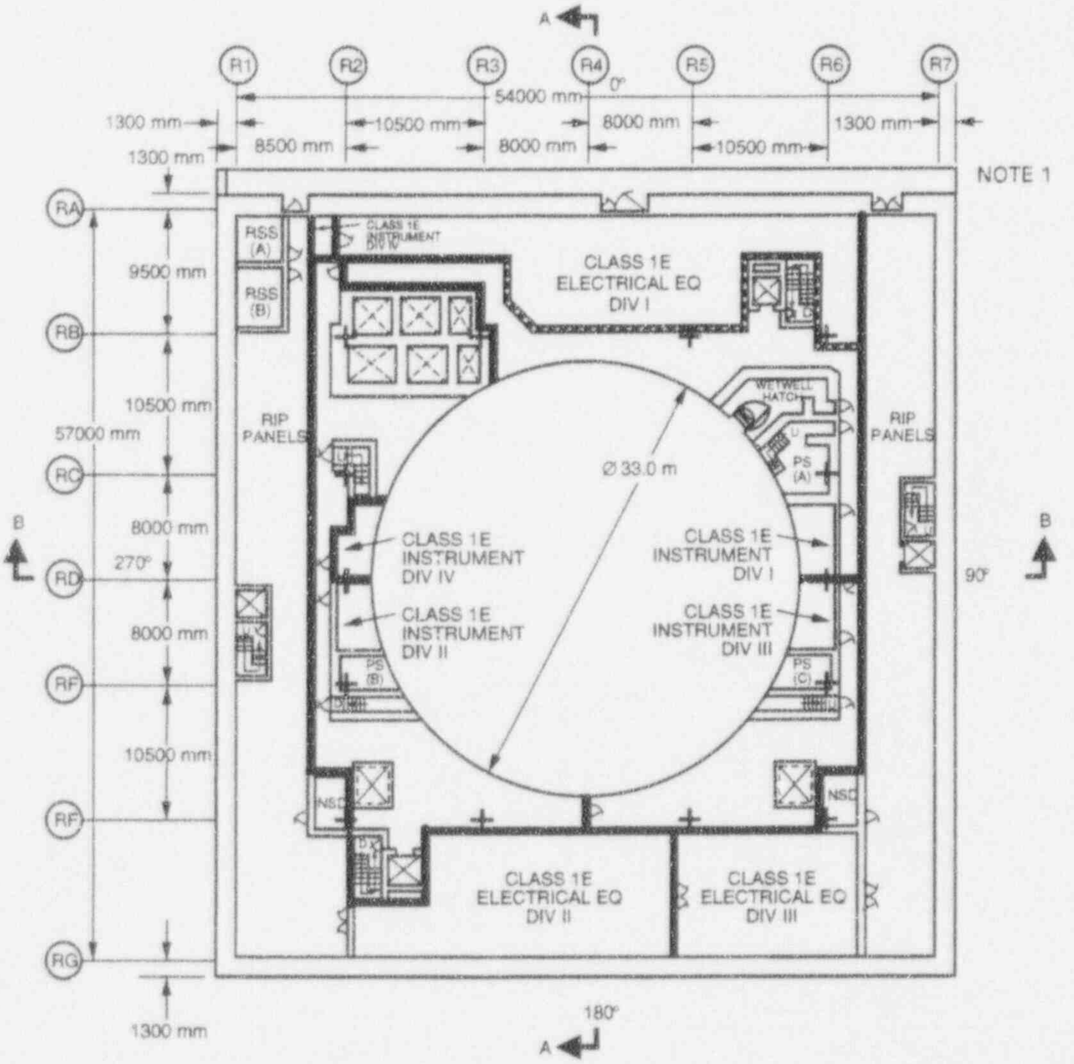


Figure 2.15.10g Reactor Building Arrangement—Elevation 1500 mm



NOTE 1: Watertight doors to be provided in access corridor to prevent flood water external to the Reactor Building from entering the Reactor Building

Figure 2.15.10h Reactor Building Arrangement, Floor B1F—Elevation 4800 mm

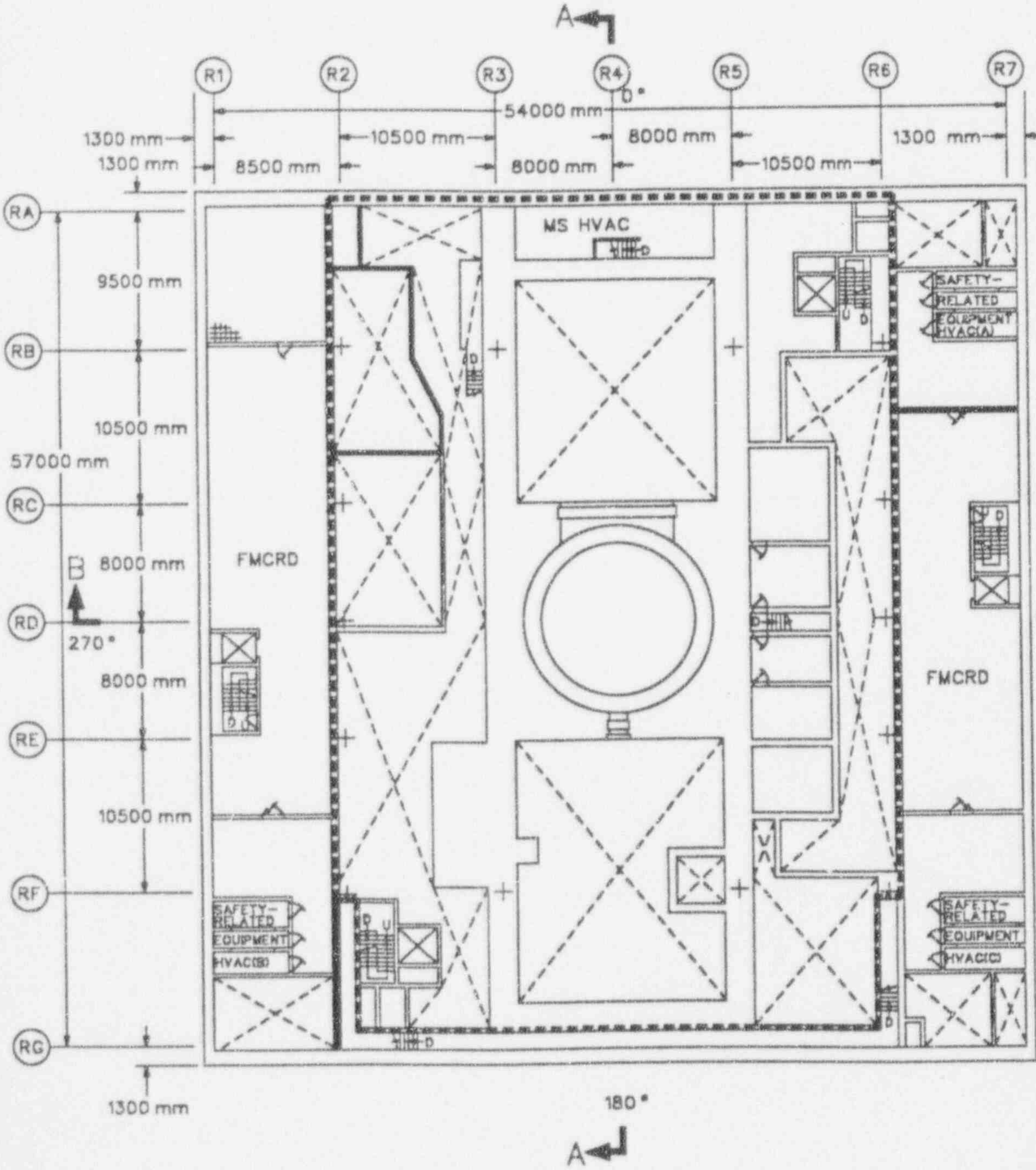


Figure 2.15.10m Reactor Building Arrangement—Elevation 27200 mm

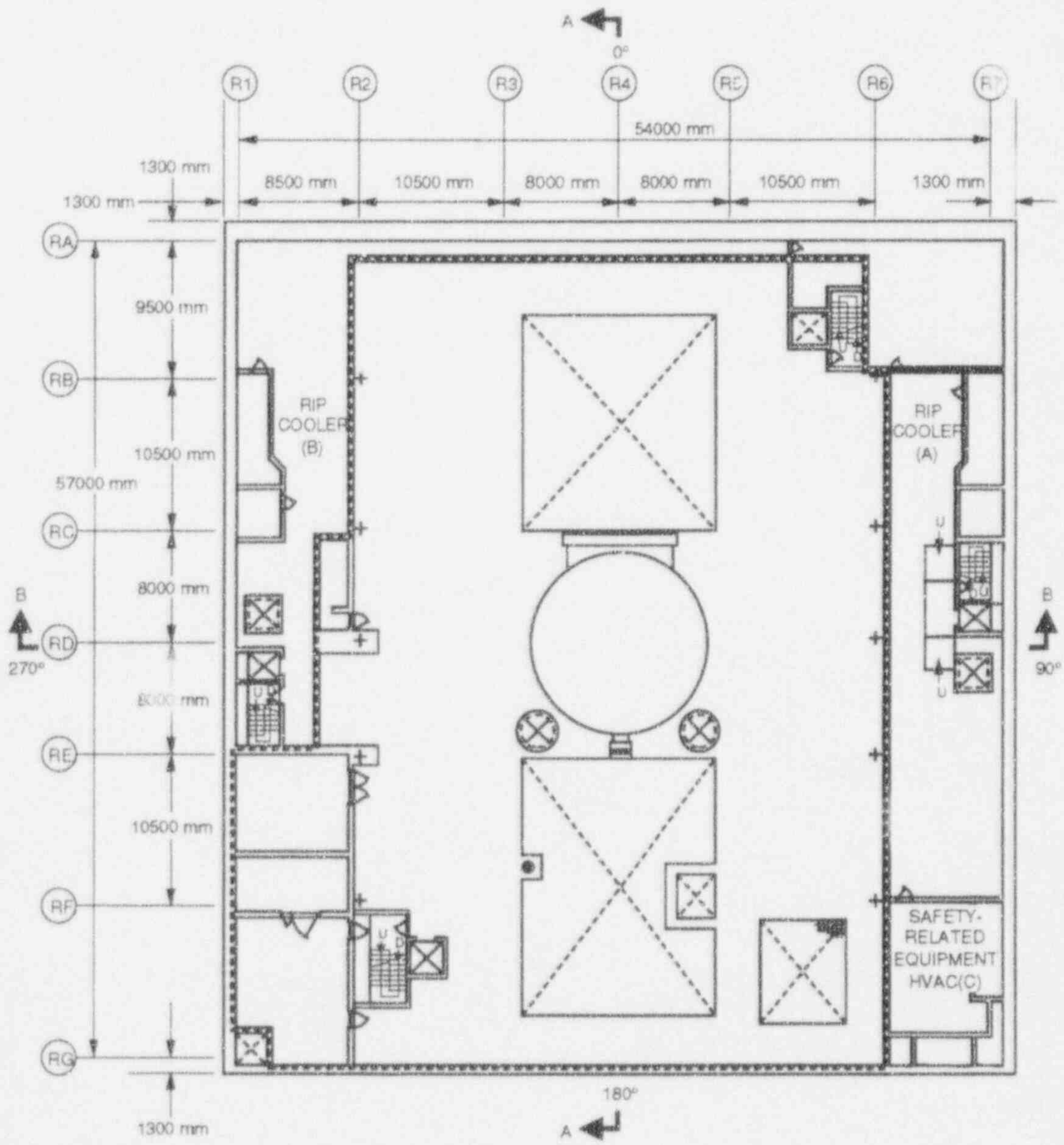


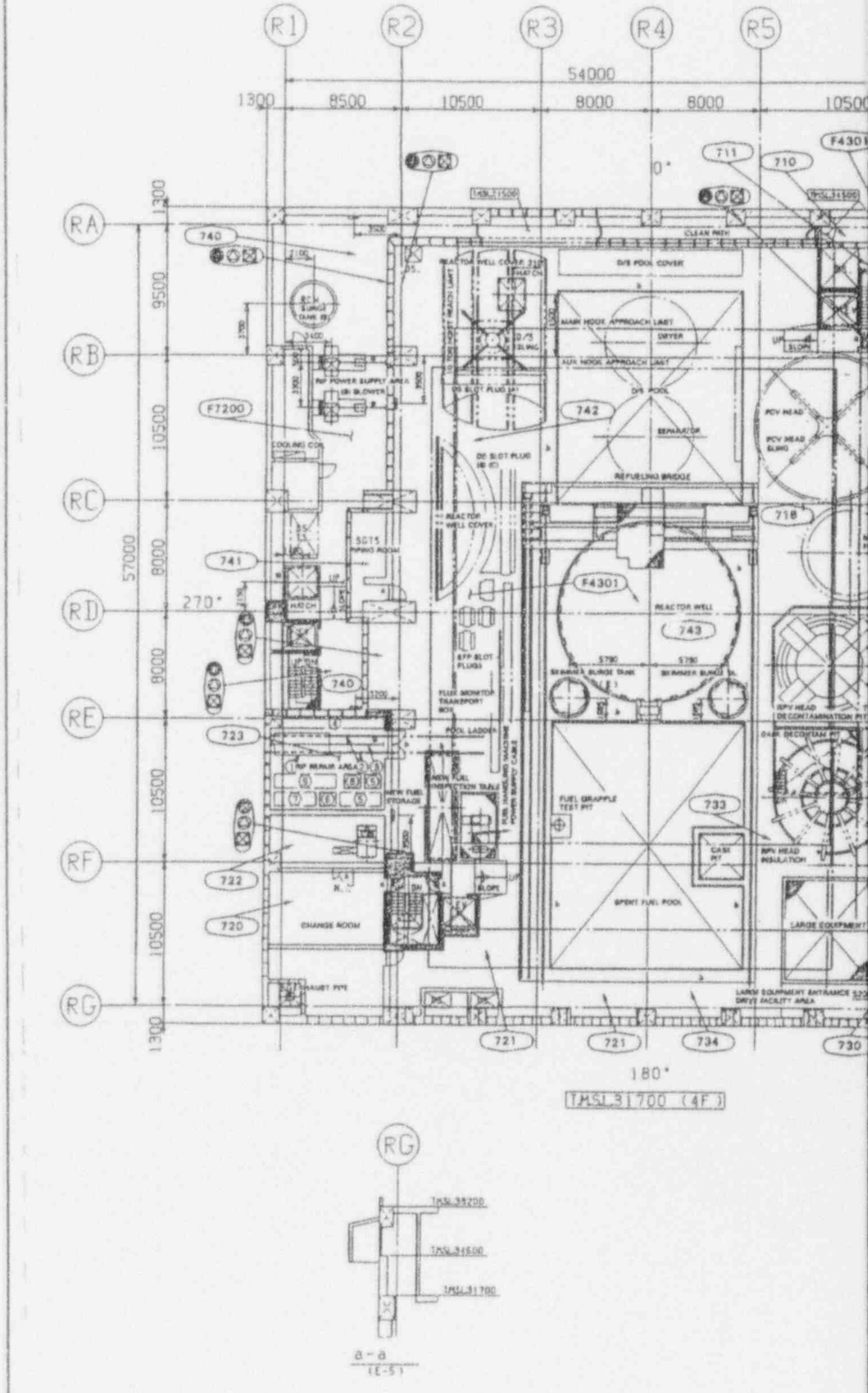
Figure 2.15.10n Reactor Building Arrangement, Floor 4F—Elevation 31700 mm

Appendix C Conversion to ASME Standard Units

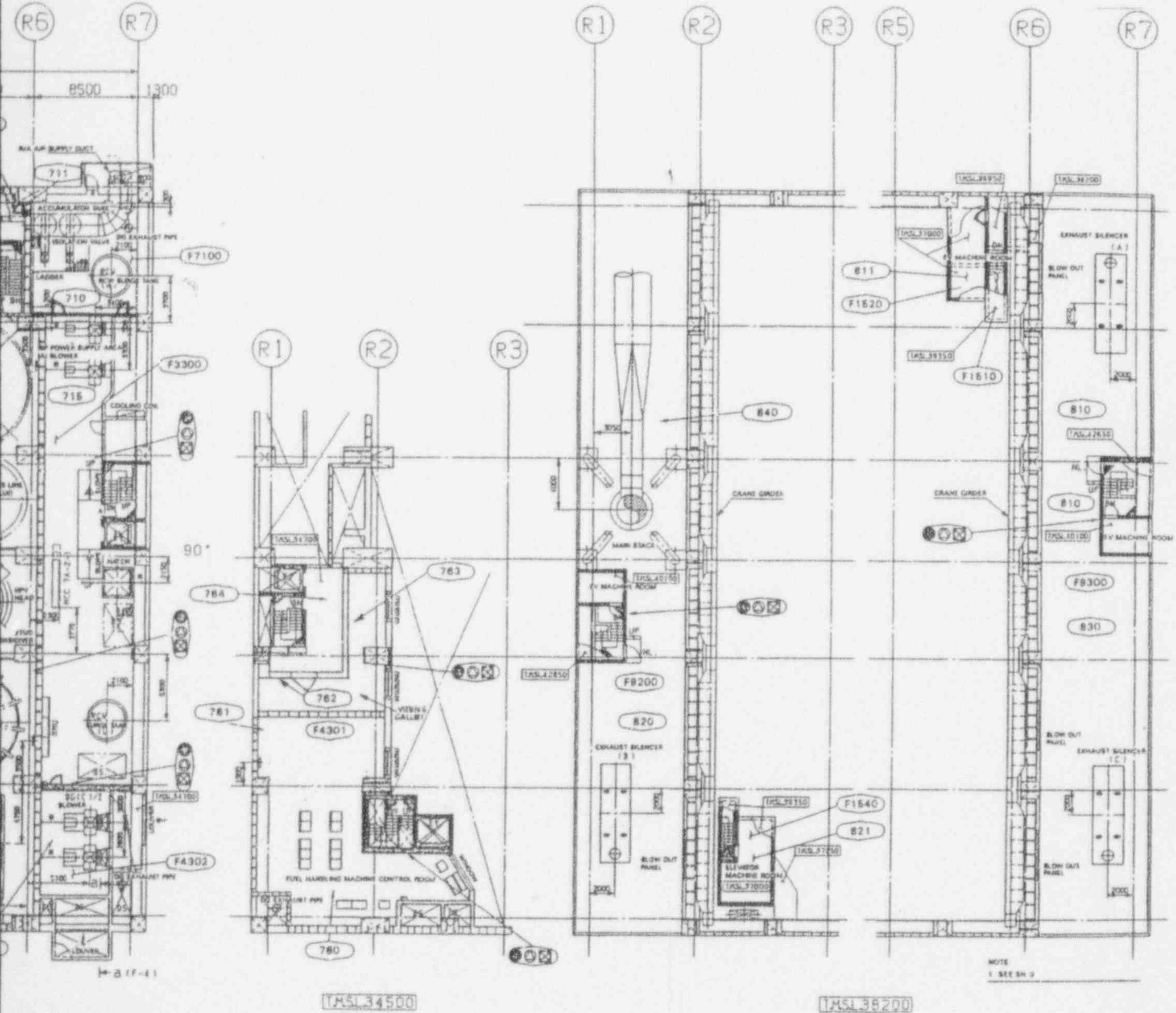
	From	To convert to	Divide by
(1)	**Pressure/Stress**		
	kilopascal	1 Pound/Square Inch	6.894757
	kilopascal	1 Atmosphere (STD)	101.325
	kilopascal	1 Foot of Water (39.2°F)	2.98898
	kilopascal	1 Inch of Water (60°F)	0.24884
	kilopascal	1 Inch of HG (32°F)	3.38638
(2)	**Force/Weight**		
	newton	1 Pound - force	4.448222
	kilogram	1 Ton (Short)	907.1847
	kilogram	1 Tons (Long)	1016.047
(3)	**Heat/Energy/Power**		
	joule	1 Btu	1055.056
	joule	1 Calorie	4.1868
	kilowatt-hour	1 Btu	0.0002930711
	kilowatt	1 Horsepower(U.K.)	0.7456999
	kilowatt-hour	1 Horsepower-Hour	0.7456999
	kilowatt	1 Btu/Min	0.0175725
	joule/gram	1 Btu/Pound	2.326
(4)	**Length**		
	millimeter	1 Inch	25.4
	centimeter	1 Inch	2.54
	meter	1 Inch	0.0254
	meter	1 Foot	0.3048
	centimeter	1 Foot	30.48
	meter	1 Mile	1609.344
	kilometer	1 Mile	1.609344
(5)	**Volume**		
	liter	1 Cubic Inch	0.01638706
	cubic centimeter	1 Cubic Inch	16.38706

	From	To convert to	Divide by
	cubic meter	1 Cubic Foot	0.02831685
	cubic centimeter	1 Cubic Foot	28316.85
	liter	1 Cubic Foot	28.31685
	cubic meter	1 Cubic Yard	0.7645549
	liter	1 Gallon (US)	3.785412
	cubic centimeter	1 Gallon (US)	3785.412
	E-03 cubic centimeter	1 Gallon (US)	3.785412
(6)	**Volume Per Unit Time**		
	cubic centimeter/s	1 Cubic Foot/Min	471.9474
	cubic meter/h	1 Cubic Foot/Min	1.69901
	liter/s	1 Cubic Foot/Min	0.4719474
	cubic meter/s	1 Cubic Foot/Sec	0.02831685
	E-05 cubic meter/s	1 Gallon/Min (US)	6.30902
	cubic meter/h	1 Gallon/Min (US)	0.22712
	liter/s (101.325 kPaA,15.56°C)	1 STD CFM (14.696 psia, 60°F)	0.4474
	cubic meter/h (101.325 kPaA,15.56°C)	1 STD CFM (14.696 psia, 60°F)	1.608
(7)	**Velocity**		
	centimeter/s	1 Foot/Sec	30.48
	centimeter/s	1 Foot/Min	0.508
	meter/s	1 Foot/Min	0.00508
	meter/min	1 Foot/Min	0.3048
	centimeter/s	1 Inches/Sec	2.54
(8)	**Area**		
	square centimeter	1 Square Inch	6.4516
	E-04 square meter	1 Square Inch	6.4516
	square centimeter	1 Square Foot	929.0304
	E-02 square meter	1 Square Foot	9.290304
(9)	**Torque**		
	newton-meter	1 Foot Pound	1.355818
(10)	**Mass Per Unit Time**		
	kilogram/s	1 Pound/Sec	0.4535924

SSAR ENGINEERING DRAWINGS



R1
R2
R3
R4
R5
54000
1300
8500
10500
8000
8000
RA
RB
RC
RD
RE
RF
RG
57000
270°
180°
TMSL31700 (4F)
REACTOR WELL
SPENT FUEL POOL
CHANGE ROOM
REFUELING BRIDGE
FUEL GRAPPLE TEST PIT
NEW FUEL STORAGE
a - a



(NO.) RACK NAME
1. INSPECTION POOL
2. TEMPORARY INSTALLED RAIL
3. MONO-RAIL
4. TEMPORARY INSTALLED RAIL STORAGE AREA
5. IMPELLER SHAFT GRAPPLE STORAGE AREA
5. DIFFUSER WEAR RING GRAPPLE STORAGE AREA
7. DIFFUSER STRETCH TUBE GRAPPLE STORAGE AREA
8. UPPER PLUG STORAGE AREA
9. RIP UPPER PORTION HANDLING CONNECTOR ROD STORAGE AREA

(REMARKS)
EQUIPMENT
D/S POOL
SPENT FUEL STORAGE POOL

9406270337-01

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

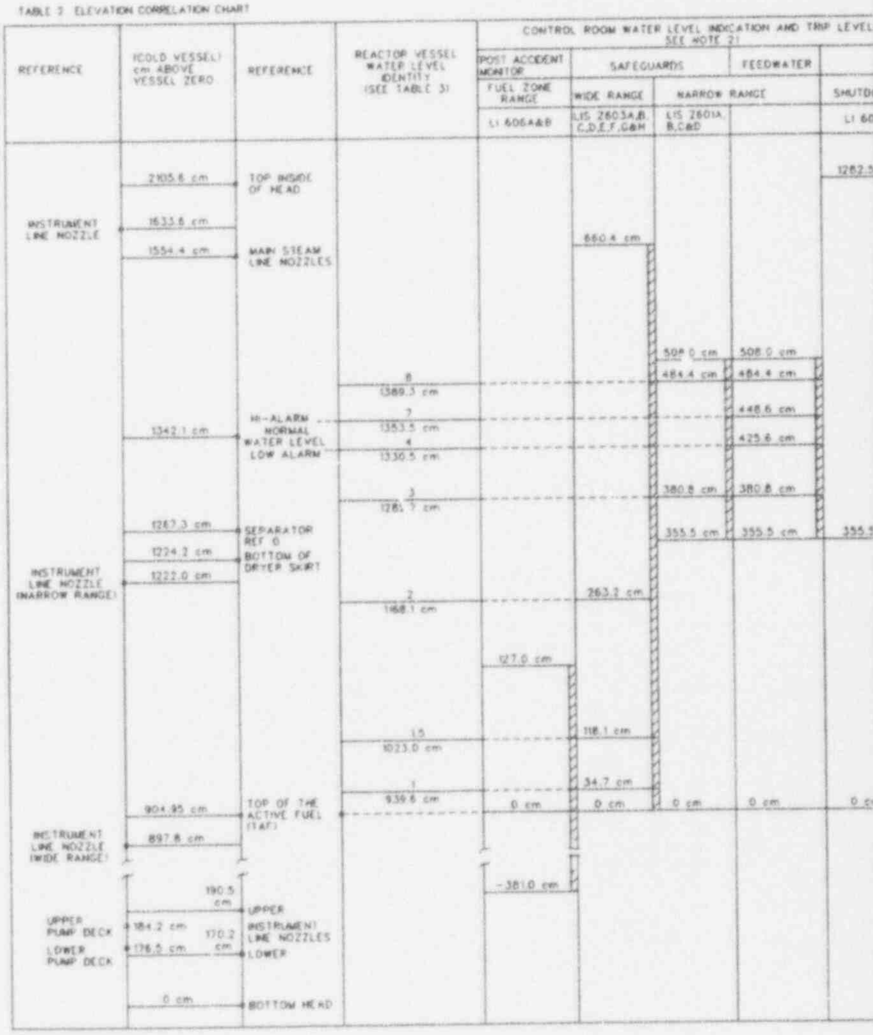
14 13 12 11 10

TABLE 4 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
TEMPERATURE ELEMENT TE029		P	J	M	G	S	B	K	E	U	D	N	H
SOLENOID VALVE	ADS 1											N	H
	ADS 2											N	H
	RV	P	J	M	G	S	B	K	E	U	D	N	H
ACCUMULATOR	AD03											N	H
	AD04	P	J	M	G	S	B	K	E	U	D	N	H
CHECK VALVE	F026											N	H
	F029	P	J	M	G	S	B	K	E	U	D	N	H
SRV OPEN CLOSE MONITORING SYSTEM		P	J	M	G	S	B	K	E	U	D	N	H
	COMPUTER INPUT	NC041	NC042	NC043	NC044	NC045	NC046	NC047	NC048	NC049	NC050	NC051	NC052
* 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
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
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
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
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
	PT007A THRU D/ PS-Z607A-6 THRU D-6	≥2 OF A,B,C,D											
	PT007A THRU D/ PS-Z607A-5 THRU D-5		≥2 OF A,B,C,D										
	PT007A THRU D/ PS-Z607A-4 THRU D-4			≥2 OF A,B,C,D									
	PT007A THRU D/ PS-Z607A-3 THRU D-3							≥2 OF A,B,C,D					
	PT007A THRU D/ PS-Z607A-2 THRU D-2											≥2 OF A,B,C	
	PT007A THRU D/ PS-Z607A-1 THRU D-1												

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

TABLE 2 ELEVATION CORRELATION CHART



	C	L	F	R	A
	C	L	F	R	A
	C	L	F	R	A
	C	L	F	R	A
	C	L	F	R	A
	C	L	F	R	A
	C	L	F	R	A
	C	L	F	R	A
	C	L	F	R	A
	C	L	F	R	A
053	NC054	NC055	NC056	NC057	NC058
1 035	B21 NC036	B21 NC037	B21 NC038	B21 NC039	B21 NC040
.13	8.13	8.20	8.20	8.20	8.20
.56	7.56	7.63	7.63	7.63	7.63
.78	7.79	7.86	7.86	7.86	7.86
.28	7.28	7.35	7.35	7.35	7.35
D					
		R2 OF A,B,C,D			

WN	REACTOR WELL
	LI 604
cm	1800.0 cm
cm	
	0 cm

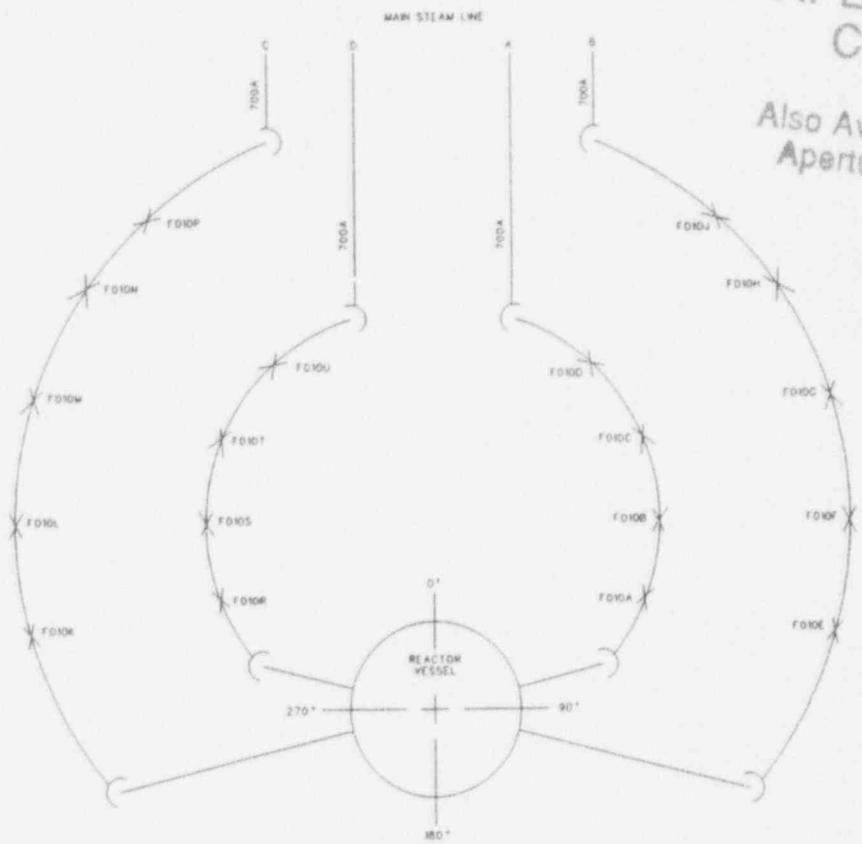


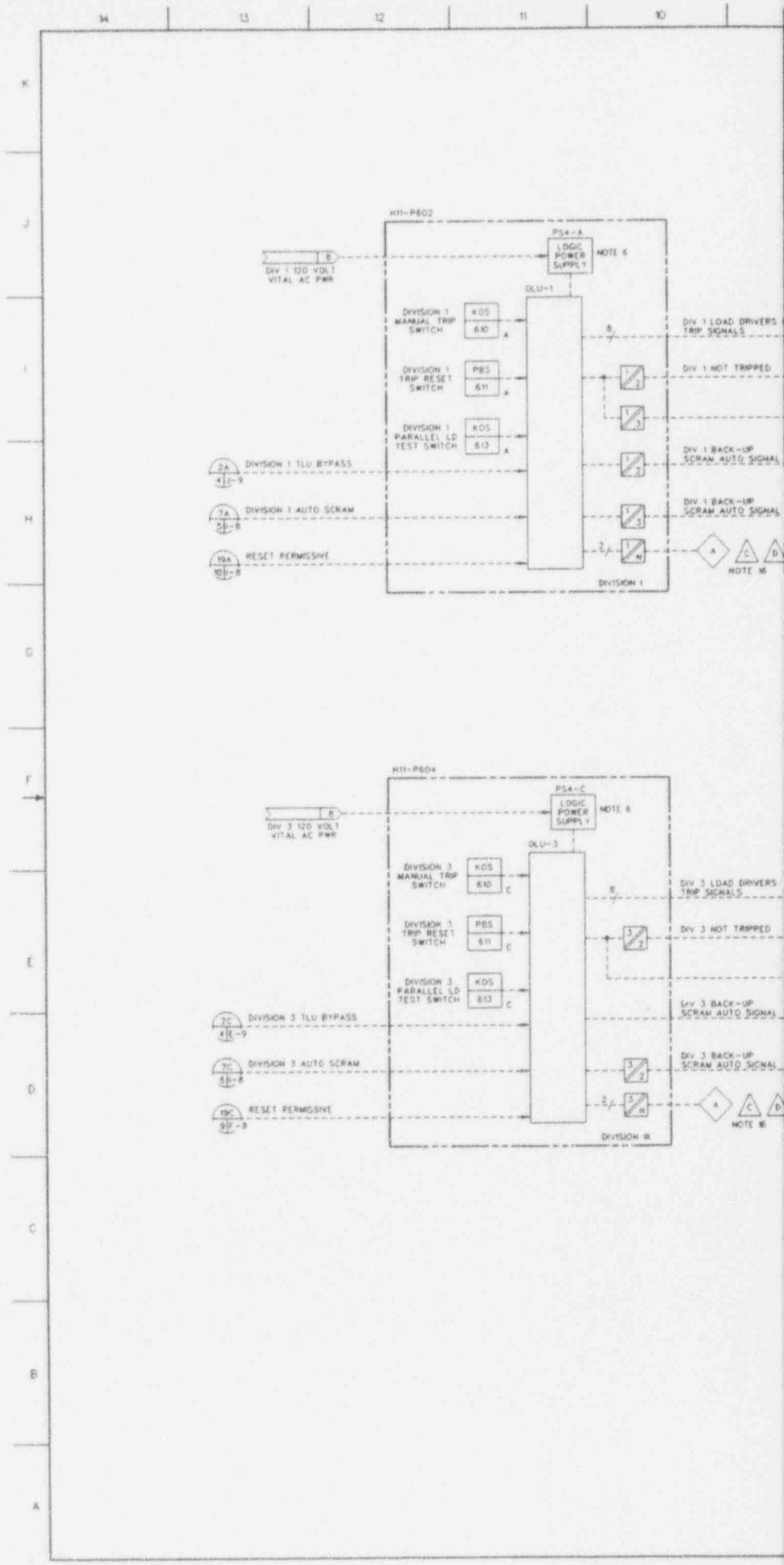
FIG. 3
SAFETY/RELIEF VALVE ORIENTATION AND STEAM PIPING LINE SIZES

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 MSIV'S	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

9406270337-02

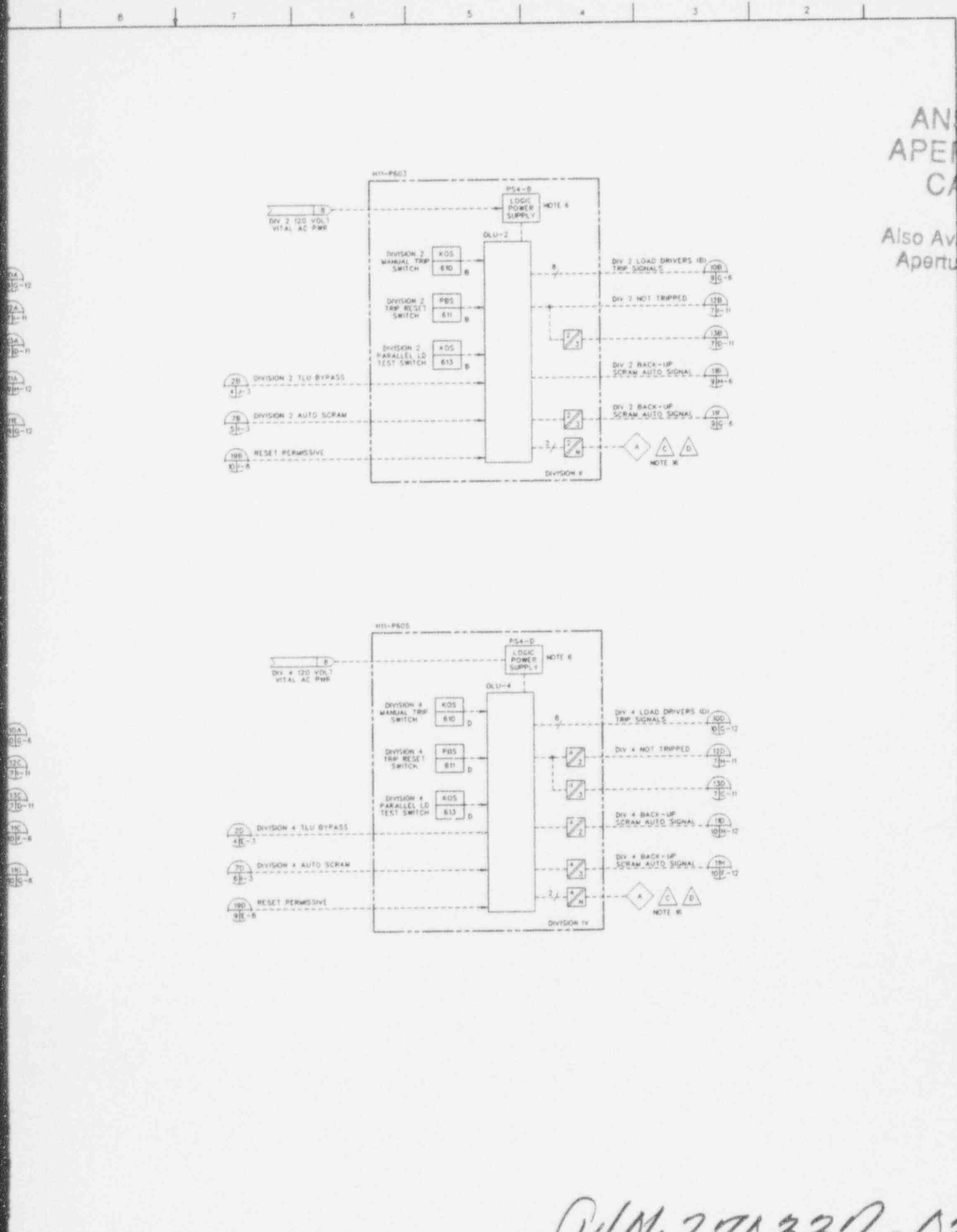
FIGURE 5.1-3 NUCLEAR BOILER SYSTEM P&ID (Sheet 9 of 11)



14
13
12
11
10
K
J
I
H
G
F
E
D
C
B
A
H11-P602
PS4-A
LOGIC POWER SUPPLY
NOTE 6
DIV 1 120 VOLT VITAL AC PWR
DLU-1
DIVISION 1 MANUAL TRIP SWITCH
KDS 610
DIVISION 1 TRIP RESET SWITCH
PBS 611
DIVISION 1 PARALLEL LD TEST SWITCH
KDS 613
DIVISION 1 TLU BYPASS
DIVISION 1 AUTO SCRAM
RESET PERMISSIVE
DIV 1 LOAD DRIVERS TRIP SIGNALS
DIV 1 NOT TRIPPED
DIV 1 BACK-UP SCRAM AUTO SIGNAL
DIV 1 BACK-UP SCRAM AUTO SIGNAL
NOTE 16
DIVISION I
H11-P604
PS4-C
LOGIC POWER SUPPLY
NOTE 6
DIV 3 120 VOLT VITAL AC PWR
DLU-3
DIVISION 3 MANUAL TRIP SWITCH
KDS 610
DIVISION 3 TRIP RESET SWITCH
PBS 611
DIVISION 3 PARALLEL LD TEST SWITCH
KDS 613
DIVISION 3 TLU BYPASS
DIVISION 3 AUTO SCRAM
RESET PERMISSIVE
DIV 3 LOAD DRIVERS TRIP SIGNALS
DIV 3 NOT TRIPPED
DIV 3 BACK-UP SCRAM AUTO SIGNAL
DIV 3 BACK-UP SCRAM AUTO SIGNAL
NOTE 16
DIVISION III

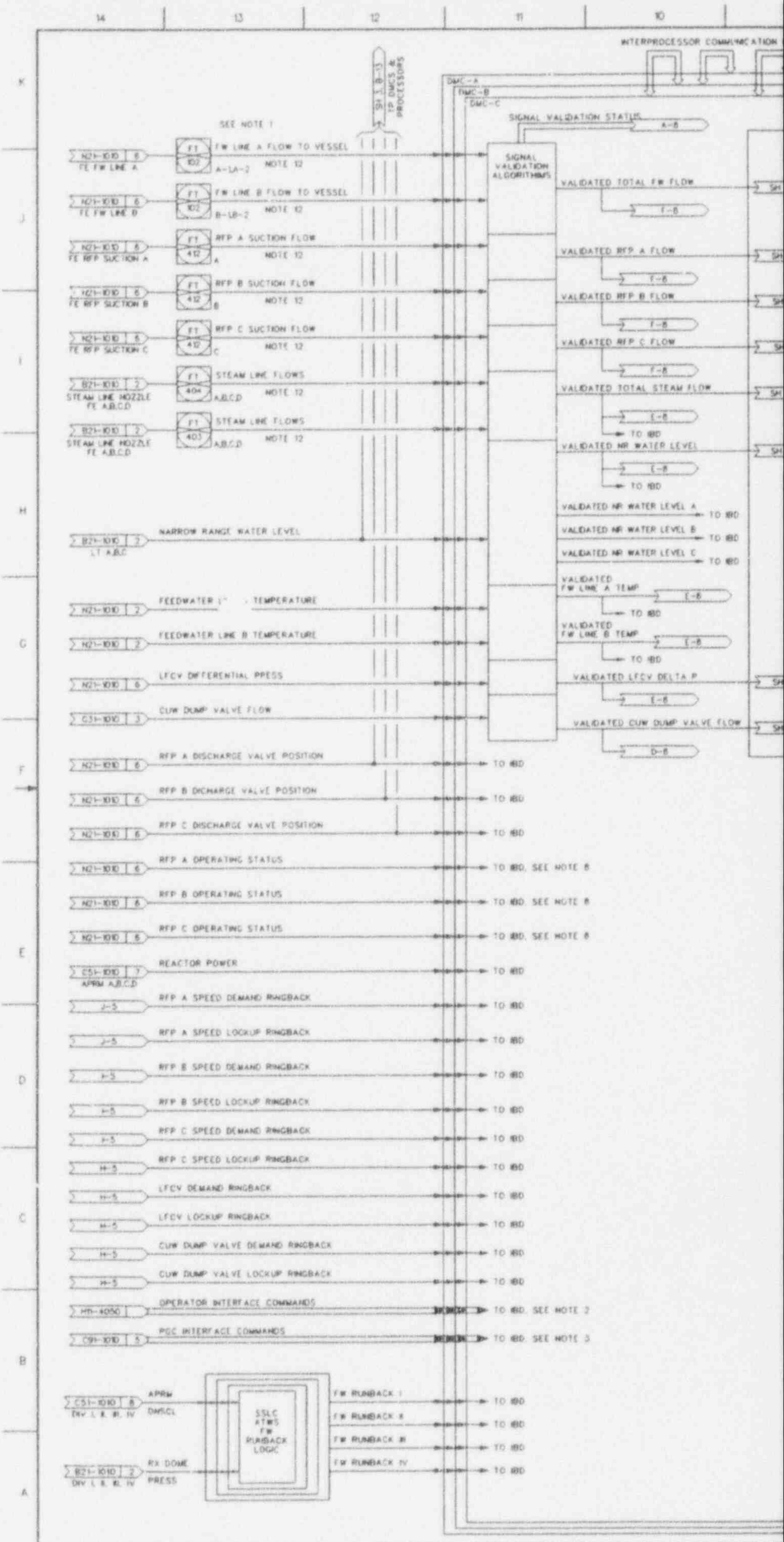
ANSTEC
APERTURE
CARD

Also Available on
Aperture Card

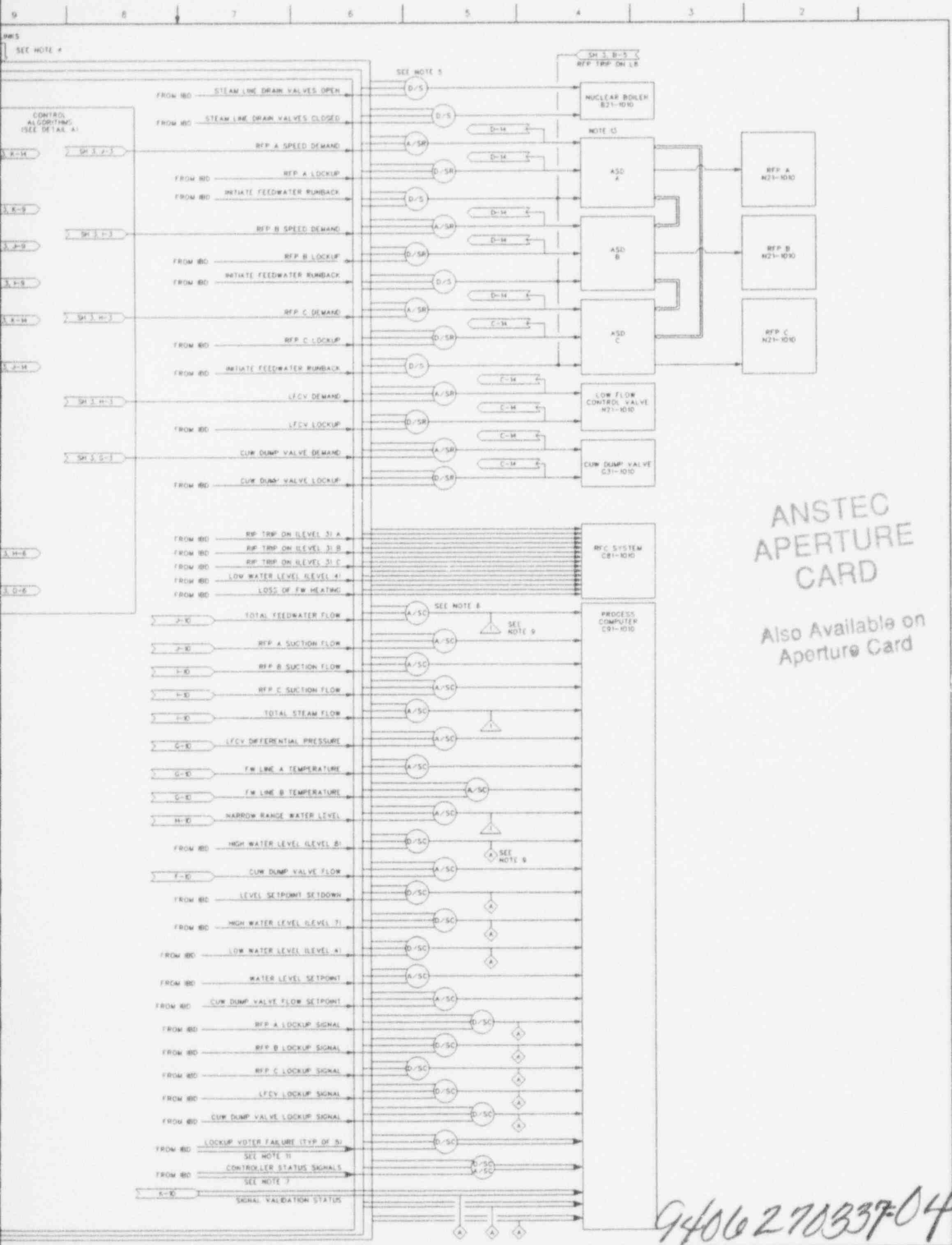


9406270337-03

FIGURE 7.2-9 REACTOR PROTECTION SYSTEM IED (Sheet 8 of 11)



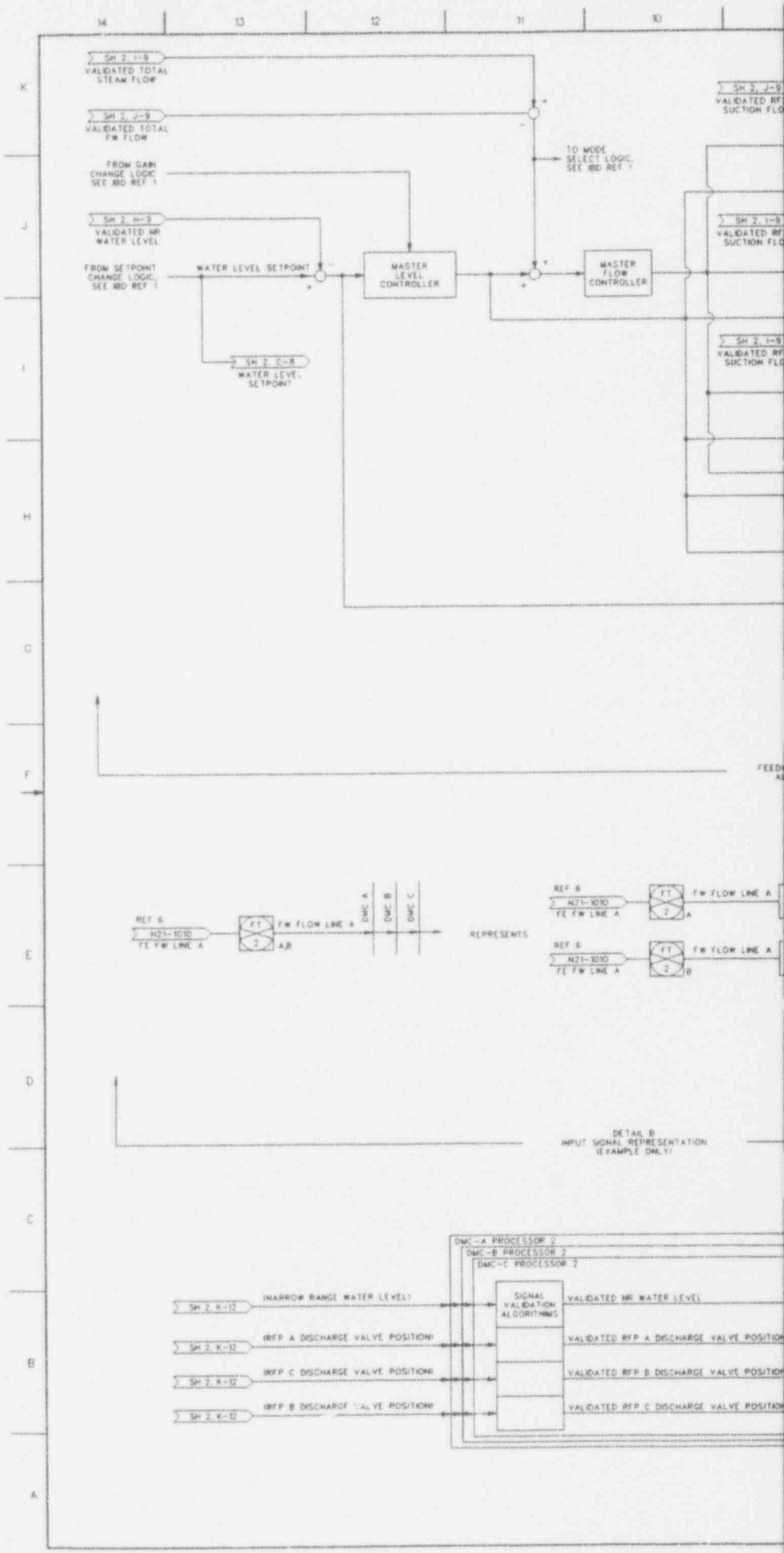
INTERPROCESSOR COMMUNICATION
SH 3, 8-13 TP DMCS & PROCESSORS
DMC-A
DMC-B
DMC-C
SEE NOTE 1
SIGNAL VALIDATION STATUS
SIGNAL VALIDATION ALGORITHMS
N21-1010 6 FE FW LINE A
FT 102 FW LINE A FLOW TO VESSEL A-LA-2 NOTE 12
N21-1010 6 FE FW LINE B
FT 102 FW LINE B FLOW TO VESSEL B-LB-2 NOTE 12
N21-1010 6 FE RFP SUCTION A
FT 412 RFP A SUCTION FLOW A NOTE 12
N21-1010 6 FE RFP SUCTION B
FT 412 RFP B SUCTION FLOW B NOTE 12
N21-1010 6 FE RFP SUCTION C
FT 412 RFP C SUCTION FLOW C NOTE 12
B21-1010 2 STEAM LINE NOZZLE FE A,B,C,D
FT 404 STEAM LINE FLOWS A,B,C,D NOTE 12
B21-1010 2 STEAM LINE NOZZLE FE A,B,C,D
FT 403 STEAM LINE FLOWS A,B,C,D NOTE 12
VALIDATED TOTAL FW FLOW
VALIDATED RFP A FLOW
VALIDATED RFP B FLOW
VALIDATED RFP C FLOW
VALIDATED TOTAL STEAM FLOW
VALIDATED NR WATER LEVEL
VALIDATED NR WATER LEVEL A
VALIDATED NR WATER LEVEL B
VALIDATED NR WATER LEVEL C
VALIDATED FW LINE A TEMP
VALIDATED FW LINE B TEMP
VALIDATED LFCV DELTA P
VALIDATED CUW DUMP VALVE FLOW
B21-1010 2 LT A,B,C NARROW RANGE WATER LEVEL
N21-1010 2 FEEDWATER LINE A TEMPERATURE
N21-1010 2 FEEDWATER LINE B TEMPERATURE
N21-1010 6 LFCV DIFFERENTIAL PRESS
G31-1010 3 CUW DUMP VALVE FLOW
N21-1010 6 RFP A DISCHARGE VALVE POSITION
N21-1010 6 RFP B DISCHARGE VALVE POSITION
N21-1010 6 RFP C DISCHARGE VALVE POSITION
N21-1010 6 RFP A OPERATING STATUS
N21-1010 6 RFP B OPERATING STATUS
N21-1010 6 RFP C OPERATING STATUS
TO IBD, SEE NOTE 8
C51-1010 7 APRM A,B,C,D REACTOR POWER
J-5 RFP A SPEED DEMAND RINGBACK
J-5 RFP A SPEED LOCKUP RINGBACK
I-5 RFP B SPEED DEMAND RINGBACK
I-5 RFP B SPEED LOCKUP RINGBACK
I-5 RFP C SPEED DEMAND RINGBACK
H-5 RFP C SPEED LOCKUP RINGBACK
H-5 LFCV DEMAND RINGBACK
H-5 LFCV LOCKUP RINGBACK
H-5 CUW DUMP VALVE DEMAND RINGBACK
H-5 CUW DUMP VALVE LOCKUP RINGBACK
H11-4050 OPERATOR INTERFACE COMMANDS TO IBD, SEE NOTE 2
C91-1010 5 PGC INTERFACE COMMANDS TO IBD, SEE NOTE 3
C51-1010 8 DIV I, II, III, IV APRM DMSCL
B21-1010 2 DIV I, II, III, IV RX DOME PRESS
SSLC ATWS FW RUNBACK LOGIC
FW RUNBACK I
FW RUNBACK II
FW RUNBACK III
FW RUNBACK IV
TO IBD



9406270337-04

FIGURE 7.7-8 FEEDWATER CONTROL SYSTEM IED (Sheet 2 of 3)

Amendment 35 **ABWR SSAR** *23A6100 Rev 6* *21-455.2*



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D
C
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A
SH 2, I-9
VALIDATED TOTAL STEAM FLOW
SH 2, J-9
VALIDATED TOTAL FW FLOW
FROM GAIN CHANGE LOGIC SEE IBD REF 1
SH 2, H-9
VALIDATED NR WATER LEVEL
FROM SETPOINT CHANGE LOGIC, SEE IBD REF 1
WATER LEVEL SETPOINT
MASTER LEVEL CONTROLLER
TO MODE SELECT LOGIC, SEE IBD REF 1
MASTER FLOW CONTROLLER
SH 2, G-8
WATER LEVEL SETPOINT
REF 6
N21-1010
FE FW LINE A
FT
2
FW FLOW LINE A
DMC A
DMC B
DMC C
REPRESENTS
DETAIL B
INPUT SIGNAL REPRESENTATION (EXAMPLE ONLY)
DMC-A PROCESSOR 2
DMC-B PROCESSOR 2
DMC-C PROCESSOR 2
SH 2, K-12
(NARROW RANGE WATER LEVEL)
(RFP A DISCHARGE VALVE POSITION)
(RFP C DISCHARGE VALVE POSITION)
(RFP B DISCHARGE VALVE POSITION)
SIGNAL VALIDATION ALGORITHMS
VALIDATED NR WATER LEVEL
VALIDATED RFP A DISCHARGE VALVE POSITION
VALIDATED RFP B DISCHARGE VALVE POSITION
VALIDATED RFP C DISCHARGE VALVE POSITION

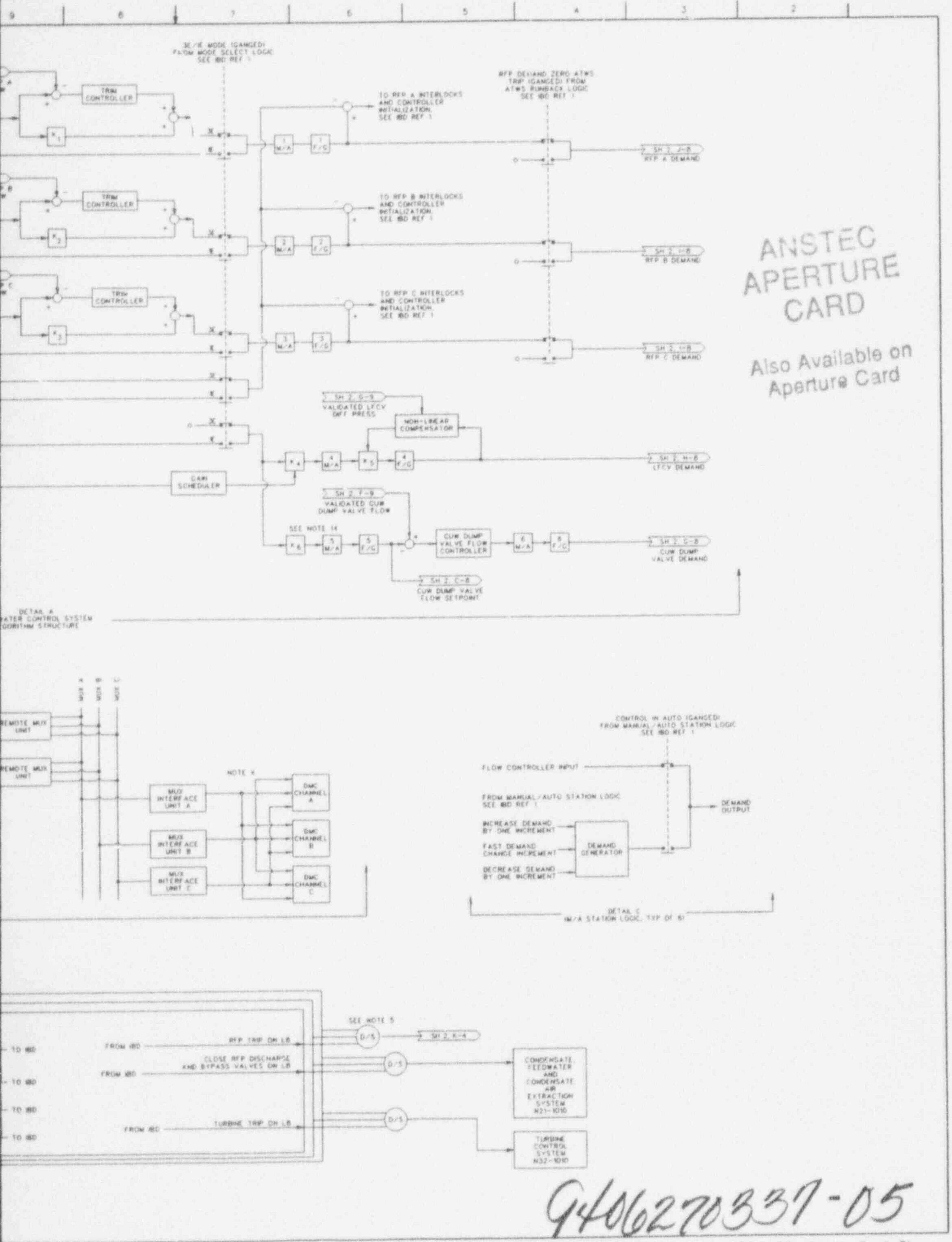

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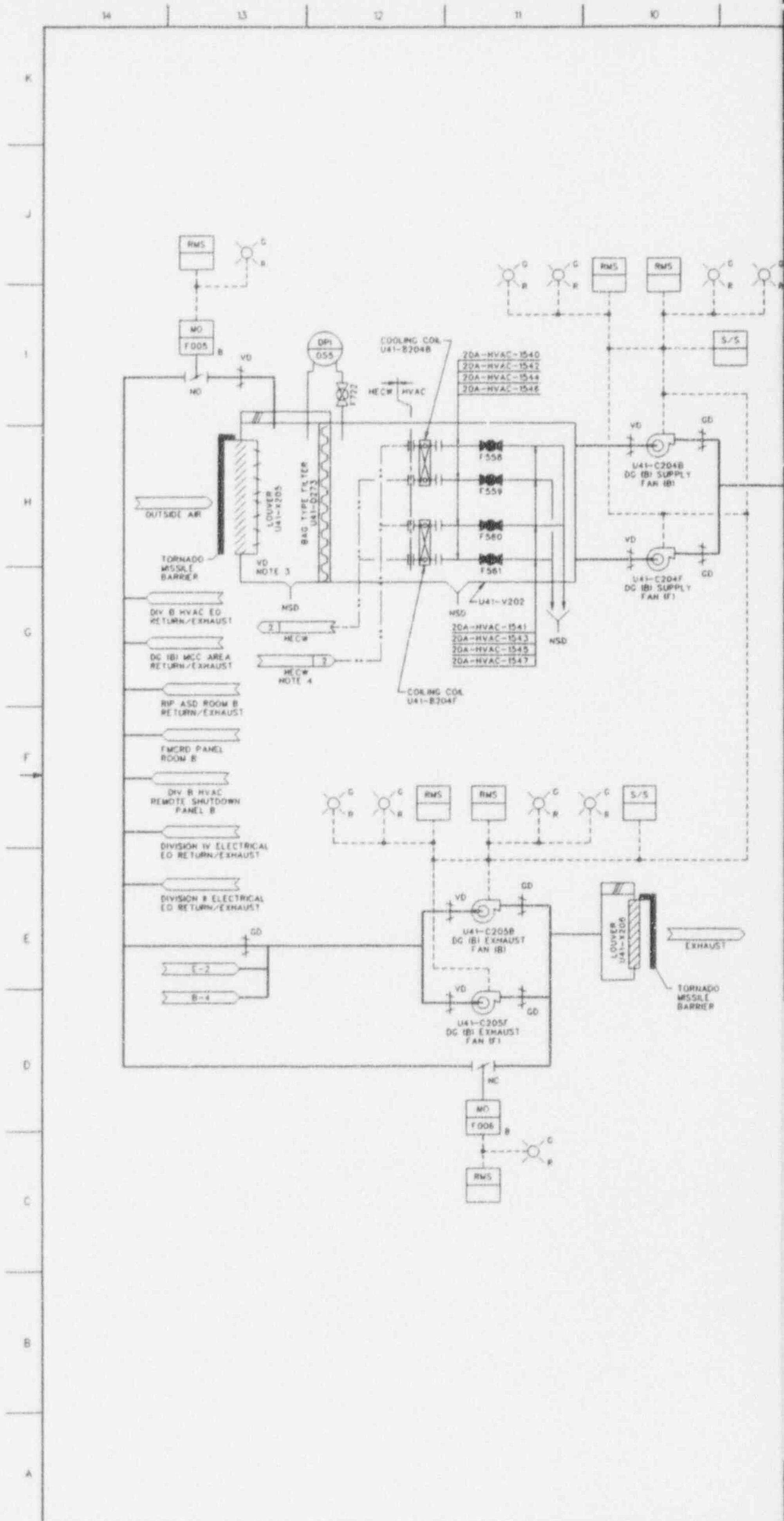
FIGURE 7.7-8 FEEDWATER CONTROL SYSTEM IED (Sheet 3 of 3)



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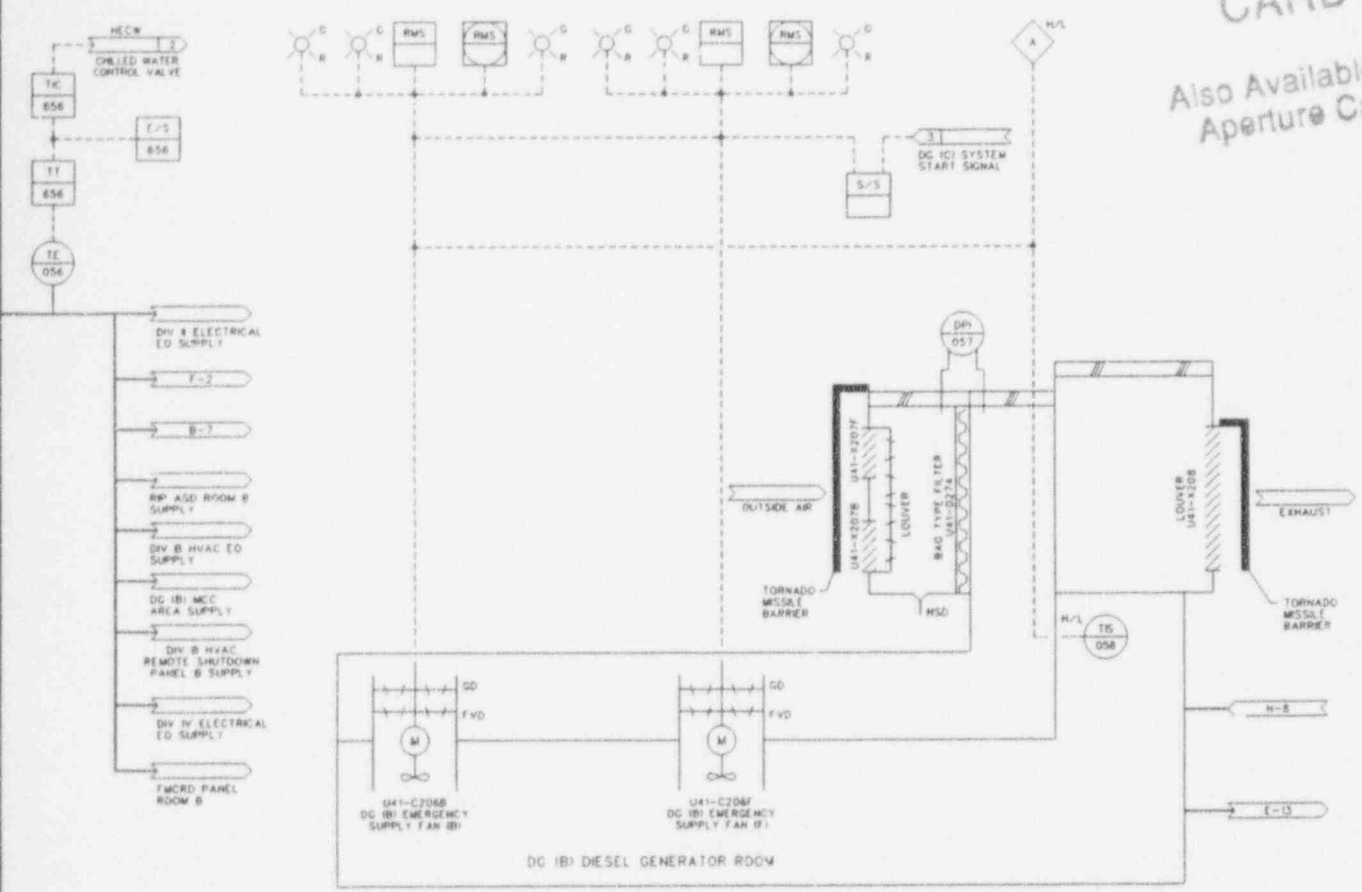
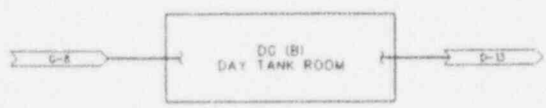
D
C
B
A
RMS
MO
F005
VO
NO
DPI
055
COOLING COIL
U41-B204B
20A-HVAC-1540
20A-HVAC-1542
20A-HVAC-1544
20A-HVAC-1546
HECW
HVAC
F558
F559
F580
F581
OUTSIDE AIR
LOUVER
U41-X205
BAG TYPE FILTER
TORNADO
MISSILE
BARRIER
VD
NOTE 3
NSD
U41-V202
20A-HVAC-1541
20A-HVAC-1543
20A-HVAC-1545
20A-HVAC-1547
COOLING COIL
U41-B204F
HECW
NOTE 4
DIV B HVAC EO
RETURN/EXHAUST
DG (B) MCC AREA
RETURN/EXHAUST
RIP ASD ROOM B
RETURN/EXHAUST
FMCRD PANEL
ROOM B
DIV B HVAC
REMOTE SHUTDOWN
PANEL B
DIVISION IV ELECTRICAL
EO RETURN/EXHAUST
DIVISION II ELECTRICAL
EO RETURN/EXHAUST
S/S
GD
U41-C204B
DG (B) SUPPLY
FAN (B)
U41-C204F
DG (B) SUPPLY
FAN (F)
U41-C205B
DG (B) EXHAUST
FAN (B)
U41-C205F
DG (B) EXHAUST
FAN (F)
E-2
B-4
LOUVER
U41-X206
EXHAUST
TORNADO
MISSILE
BARRIER
NC
MO
F006



NOTES

1. PIPING DESIGN/QUALITY CLASSIFICATION SHALL BE 4C EXCEPT THERE IS NO QUALITY GROUP SINCE THEY ARE NOT ASME CODE COMPONENTS.
2. PIPING MATERIALS FOR VENT AND DRAIN SHALL BE CS.
3. MATERIAL FOR OUTSIDE AIR INTAKE VD SHALL BE SS.
4. ALL SUPPLIED FROM DIVISION B WATER AND DIVISION 2 POWER.
5. ALL EQUIPMENT AND PIPING IS SAFETY RELATED EXCEPT WHERE NOTED.

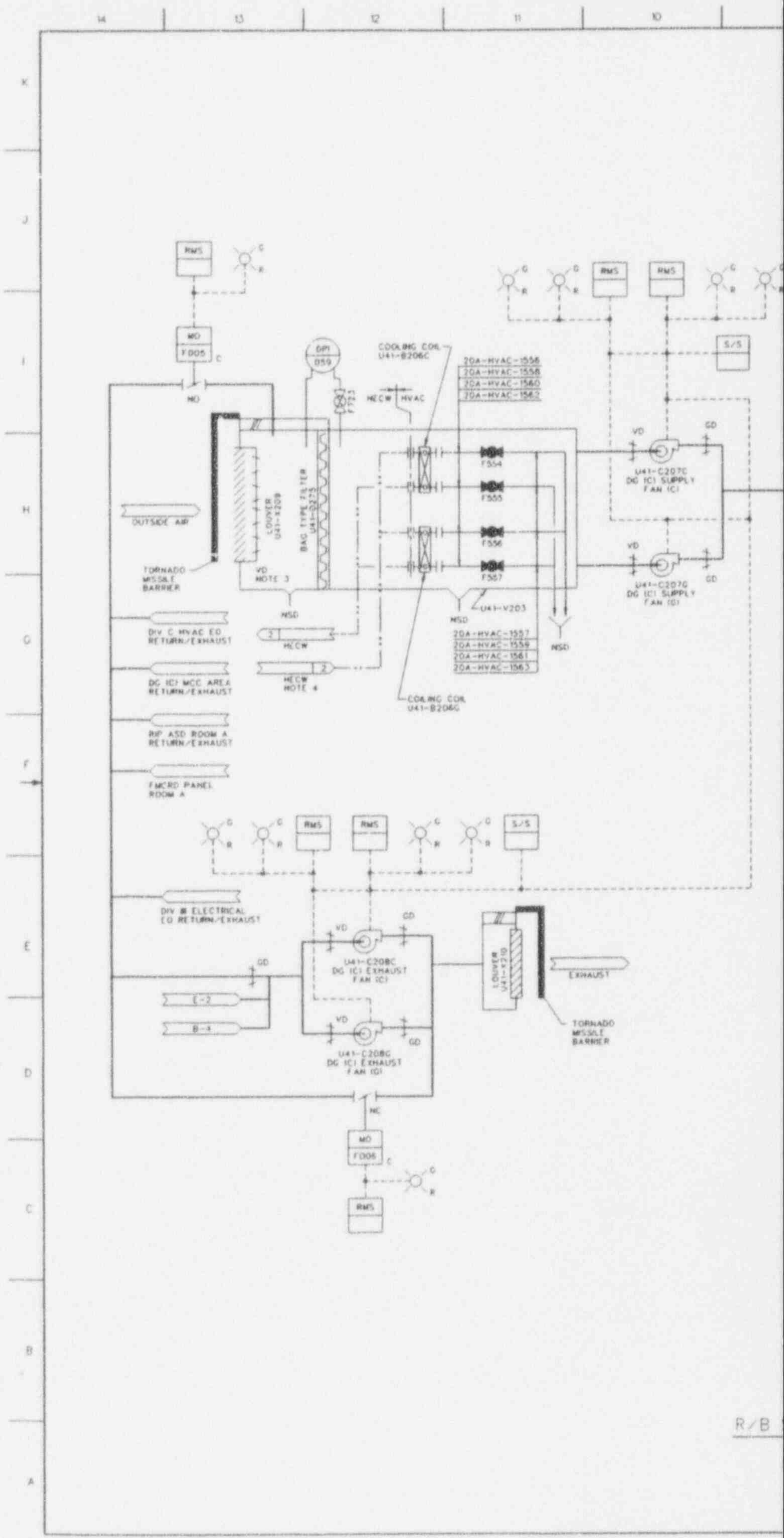
REFERENCE DOCUMENTS	MPL NO.
1. PIPING AND INSTRUMENT SYMBOLS DIAGRAM	A10-3030
2. HVAC EMERGENCY COOLING WATER SYS P&ID	P25-1010
3. DIESEL GENERATOR SYSTEM P&ID	R43-1010



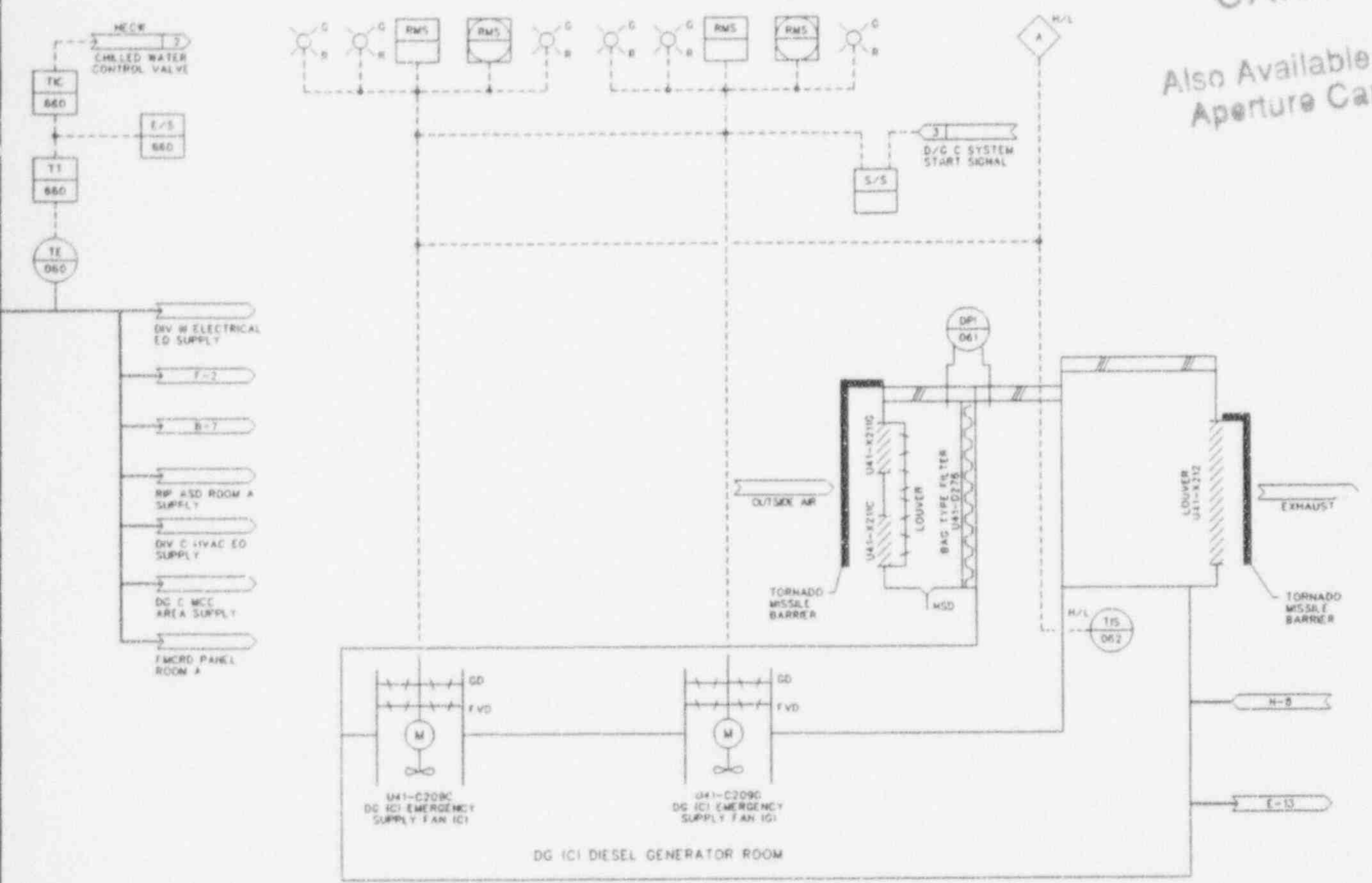
R/B S/R ELECTRICAL EQUIPMENT HVAC (B)

9406270337-06

FIGURE 9.4-4 R/B ESSENTIAL ELECTRICAL EQUIPMENT HVAC SYSTEM (Sheet 2 of 3)



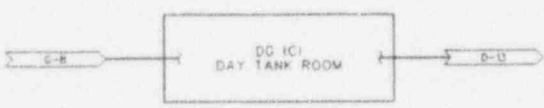
OUTSIDE AIR
TORNADO MISSILE BARRIER
LOUVER U41-X209
BAG TYPE FILTER U41-D275
VD NOTE 3
MO F005
NO
DPI 059
F723
HECW HVAC
COOLING COIL U41-B206C
20A-HVAC-1556
20A-HVAC-1558
20A-HVAC-1560
20A-HVAC-1562
F554
F555
F556
F557
NSD
U41-V203
20A-HVAC-1557
20A-HVAC-1559
20A-HVAC-1561
20A-HVAC-1563
COILING COIL U41-B206G
U41-C207C DG (C) SUPPLY FAN (C)
U41-C207G DG (C) SUPPLY FAN (G)
VD
GD
RMS
S/S
DIV C HVAC EQ RETURN/EXHAUST
HECW
HECW NOTE 4
DG (C) MCC AREA RETURN/EXHAUST
RIP ASD ROOM A RETURN/EXHAUST
FMCRD PANEL ROOM A
DIV III ELECTRICAL EQ RETURN/EXHAUST
E-2
B-4
U41-C208C DG (C) EXHAUST FAN (C)
U41-C208G DG (C) EXHAUST FAN (G)
LOUVER U41-X210
EXHAUST
TORNADO MISSILE BARRIER
NC
MO F006
R/B



NOTES

1. PIPING DESIGN/QUALITY CLASSIFICATION SHALL BE 4C EXCEPT THERE IS NO QUALITY GROUP SINCE THEY ARE NOT ASME CODE COMPONENTS
2. PIPING MATERIALS FOR VENT AND DRAIN SHALL BE CS
3. MATERIAL FOR OUTSIDE AIR INTAKE VD SHALL BE SS
4. ALL SUPPLIED FROM DIVISION C WATER AND DIVISION 3 POWER
5. ALL EQUIPMENT AND PIPING IS SAFETY RELATED EXCEPT WHERE NOTED

REFERENCE DOCUMENTS	MPL NO.
1 PIPING AND INSTRUMENT SYMBOLS DIAGRAM	A10-3030
2 HVAC EMERGENCY COOLING WATER SYS P&ID	P25-1010
3 DIESEL GENERATOR SYSTEM P&ID	R43-1010



ANSTEC APERTURE CARD

Also Available on Aperture Card

S/R ELECTRICAL EQUIPMENT HVAC (C)

9406270337-07

FIGURE 9.4-4 R/B ESSENTIAL ELECTRICAL EQUIPMENT HVAC SYSTEM (Sheet 3 of 3)

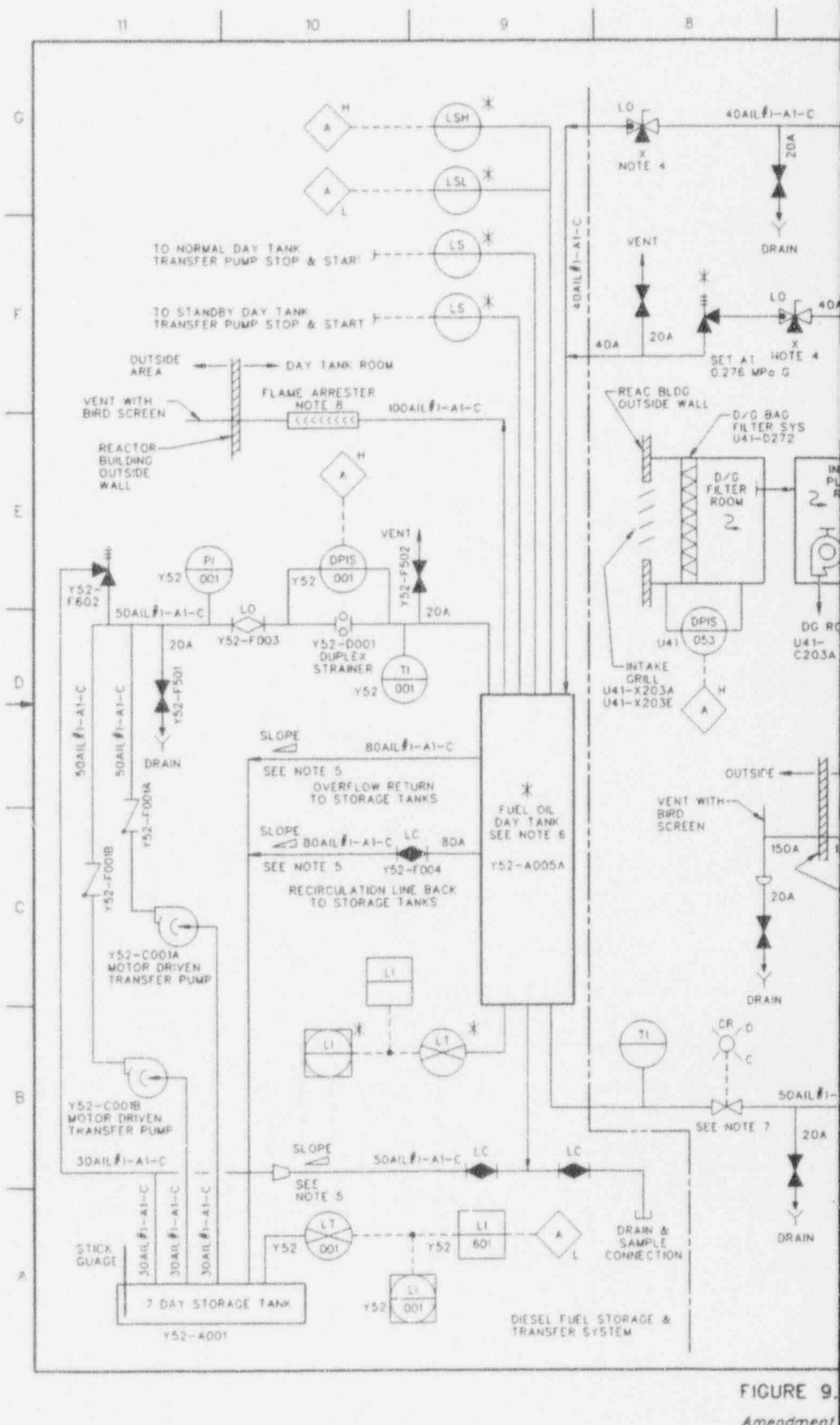
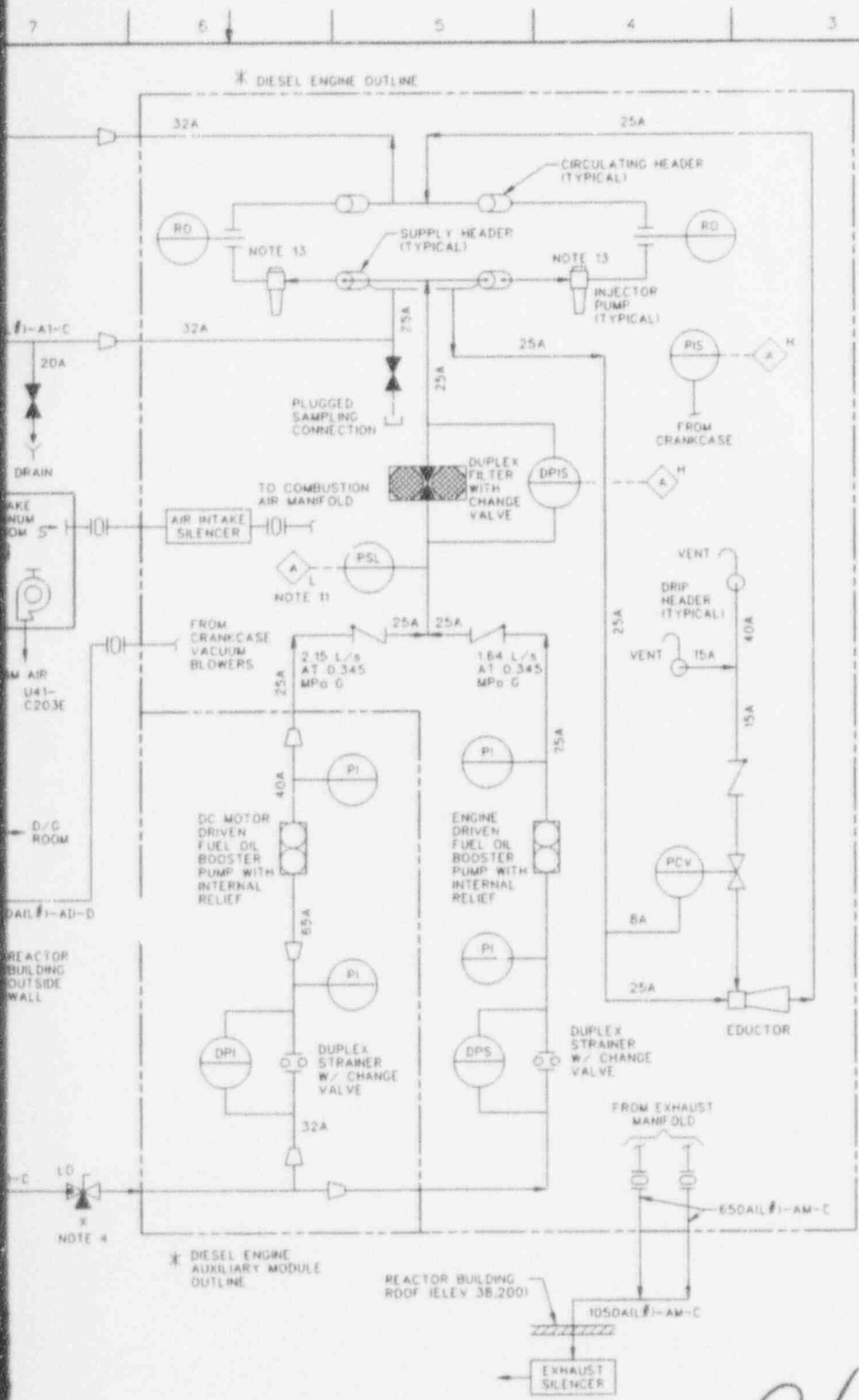


FIGURE 9.

Amendment



NOTES:

1. EQUIPMENT WITHIN DIESEL ENGINE AND DIESEL ENGINE AUXILIARY MODULE OUTLINES ARE FURNISHED ON ENGINE OR MODULE BY SELLER, AND MAY VARY DEPENDING ON UNIQUE DESIGN.
2. ALL VALVES AND INSTRUMENTS ARE PREFIXED BY SYSTEM NUMBER R43 UNLESS OTHERWISE INDICATED.
3. ALARMS ARE LOCATED IN LOCAL DIESEL GENERATOR PANEL WITH GROUP REPEAT TROUBLE ALARM PROVIDED TO CONTROL ROOM PANEL.
4. TWO POSITION, 90° TURN, THREE WAY PLUG VALVE. PLUG PORT MARKED X.
5. MINIMUM SLOPE OF 3.18mm/3.048m REQUIRED.
6. TANK TO BE AT ELEVATION TO PROVIDE FLOODED SUCTION AT BOTH DIESEL ENGINE FUEL OIL PUMPS (ELEV 23,500 mm). ENGINE BASE IS AT ELEV 12,300 mm.
7. LOCATE MANUAL BLOCK VALVE INSIDE DAY TANK ROOM WITH EXTENDED HANDWHEEL IN DIFFERENT FIRE ZONE.
8. FLAME ARRESTER TO BE MOUNTED WITHIN 4.57 m OF VENT OPENING.
9. ALL VENT AND DRAIN VALVES ON FUEL OIL PIPING SHALL BE PROVIDED WITH SCREWED CAPS OR PLUGS.
10. * DESIGNATED EQUIPMENT FURNISHED BY SELLER.
11. THIS ALARM IS DISABLED WHEN THE DIESEL IS NOT RUNNING.
12. FUTURE LINE NUMBERS ARE REPRESENTED BY "(L#)" TO BE SUPPLIED AT FINAL DESIGN STAGE.
13. SUPPLIER SHALL PROVIDE SYSTEM TO PREVENT FLOODING OF CYLINDERS DURING ENGINE SHUTDOWN BY LEAKAGE THROUGH THE FUEL PUMPS DUE TO THE DAY TANK HEAD PRESSURE (NOTE 6).
14. ALL EQUIPMENT AND PIPING EXCEPT THAT WITHIN THE DIESEL ENGINE OUTLINE IS SAFETY CLASS 3, QUALITY GROUP C (ASME, SECTION III).

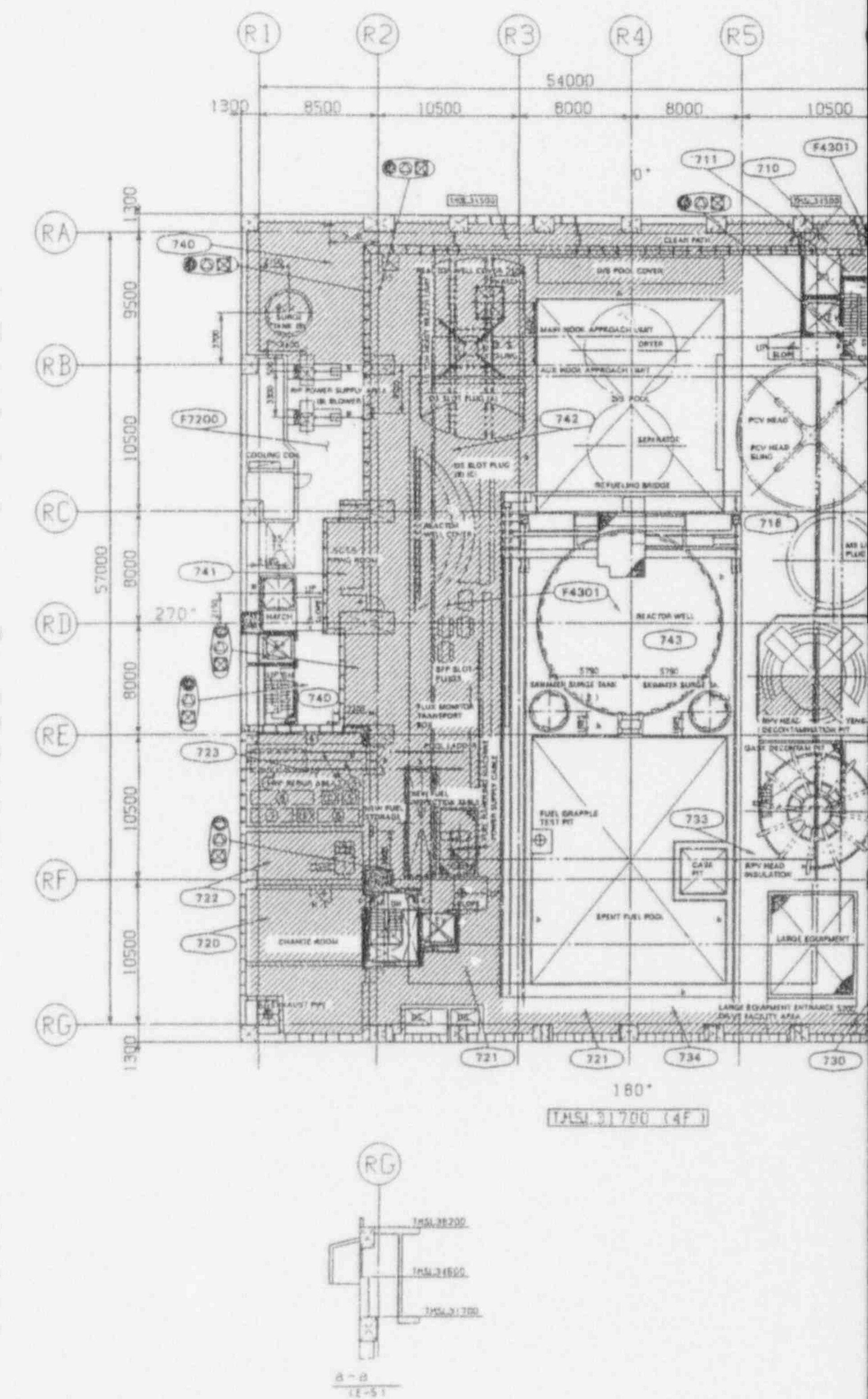
LEGEND:

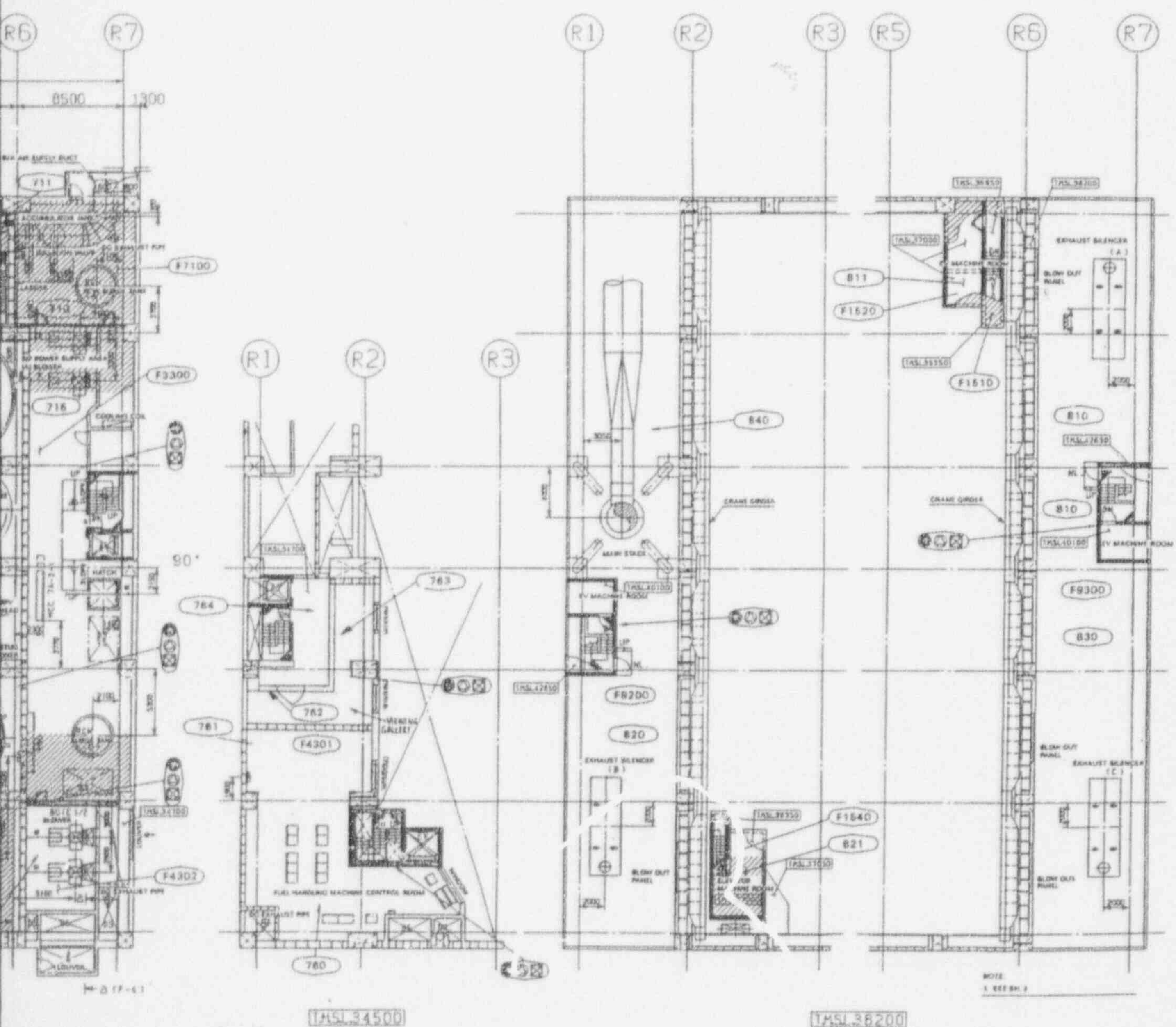
--- DENOTES PIPING SPEC CHANGE

MPL NO. R43-1070

9406270337-08

-6 STANDBY DIESEL GENERATOR FUEL OIL AND COMBUSTION AIR INTAKE AND EXHAUST SYSTEMS (Sheet 1 of 1)


R1
R2
R3
R4
R5
54000
1300
8500
10500
8000
8000
10500
RA
RB
RC
RD
RE
RF
RG
57000
9500
10500
9000
8000
10500
10500
1300
711
710
F4301
740
F7200
742
741
270°
740
743
718
723
733
722
720
721
721
734
730
180°
THSL 31700 (4F)
RG
a - a

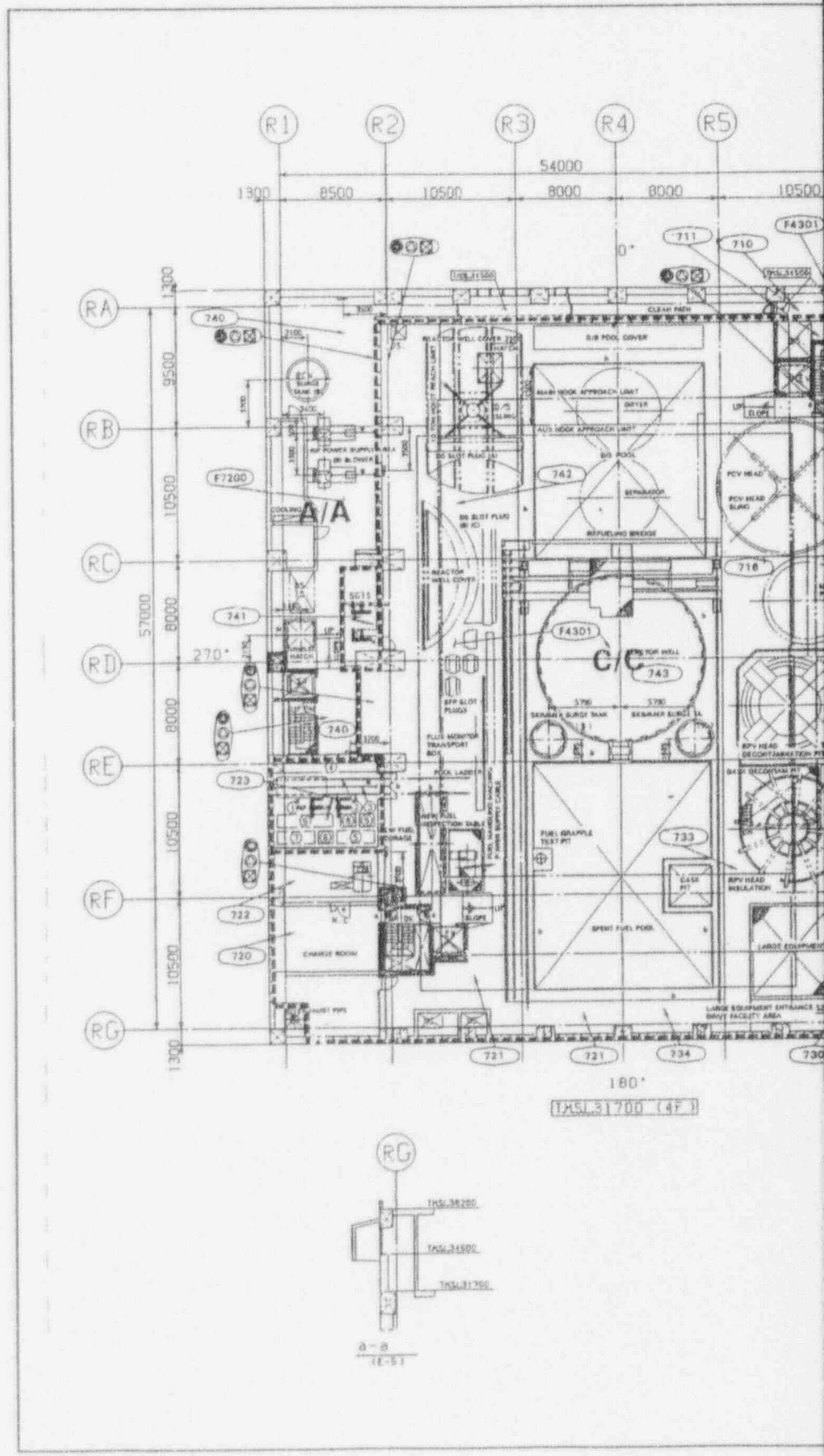


(NO) RACK NAME
1. INSPECTION POOL
2. TEMPORARY INSTALLED RAIL
3. MONO-RAIL
4. TEMPORARY INSTALLED RAIL STORAGE AREA
5. IMPELLER SHAFT GRAPPLE STORAGE AREA
6. DIFFUSER WEAR RING GRAPPLE STORAGE AREA
7. DIFFUSER STRETCH TUBE GRAPPLE STORAGE AREA
8. UPPER PLUG STORAGE AREA
9. RIP UPPER PORTION HANDLING CONNECTOR ROD STORAGE AREA

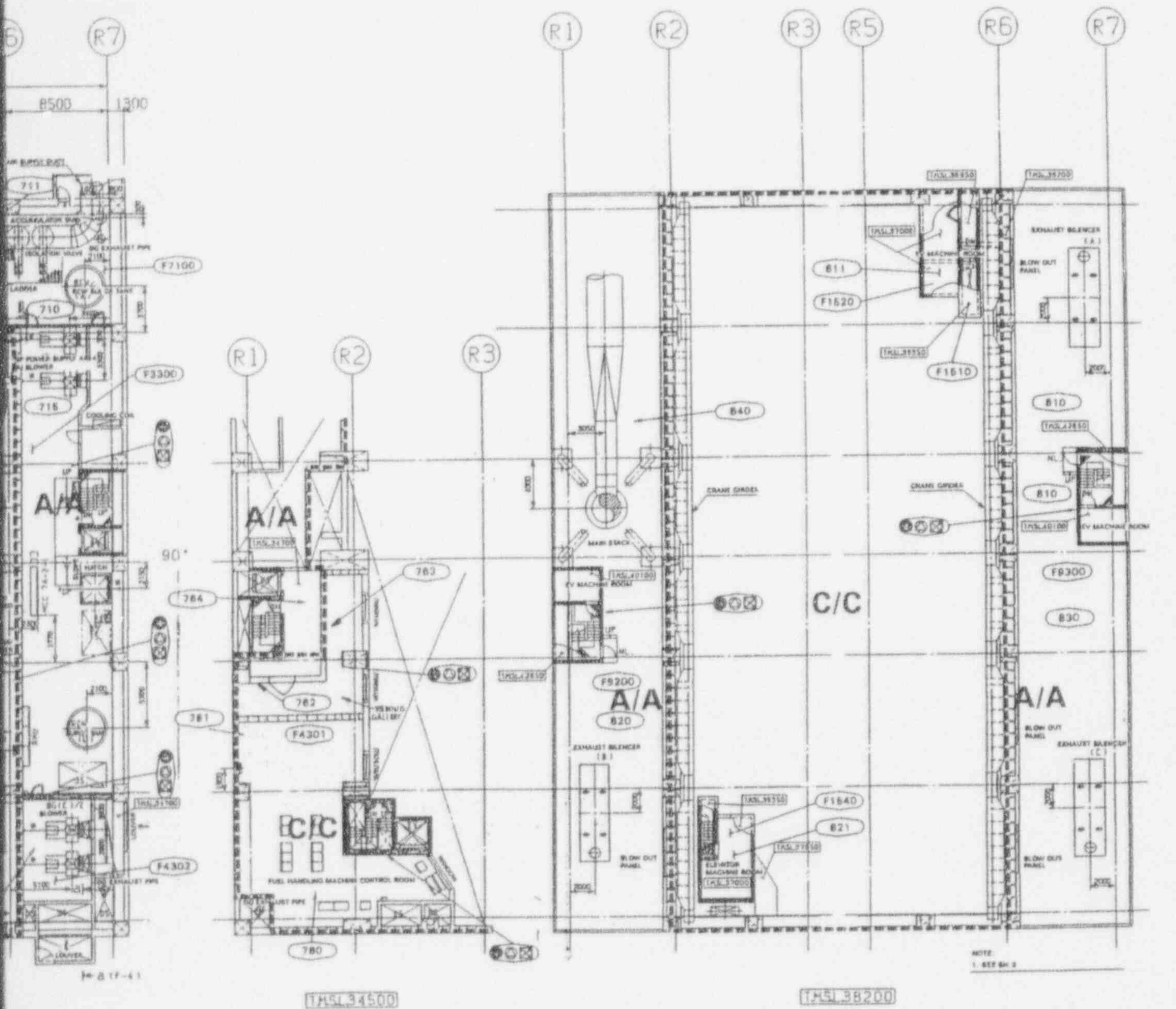
(REMARKS)
EQUIPMENT
D/S POOL
SPENT FUEL STORAGE POOL

9406270337-09

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



R1
R2
R3
R4
R5
54000
1300
8500
10500
8000
8000
10500
RA
RB
RC
RD
RE
RF
RG
57000
9500
10500
8000
8000
10500
10500
270°
180°
0°
A/A
C/C
F/E
740
F7200
741
723
722
720
721
721
734
730
711
710
F4301
762
718
743
733
SPENT FUEL POOL
CHARGE ROOM
D/S POOL COVER
REFUELING BRIDGE
REACTOR WELL COVER
FUEL GRAPPLE TEST PIT
NEW FUEL STORAGE
NEW FUEL INSPECTION TABLE
FLUX MONITOR TRANSPORT BOX
POOL LADDER
SKIMMER SURGE TANK
SKIMMER SURGE TANK
PCV HEAD
PCV HEAD SLING
RPV HEAD DECONTAMINATION PIT
RPV HEAD INSULATION
CASK PIT
SEPARATOR
DRYER
D/S POOL
CLEAN PATH
HATCH
COOLING
SLOPE
CASK PIT
LARGE EQUIPMENT ENTRANCE SHAFT / DRIVE FACILITY AREA
SGTS
TMSL31700 (4F)
TMSL38200
TMSL34500
TMSL31700
a-a
(E-5)



(NO.) RACK NAME

1. INSPECTION POOL
2. TEMPORARY INSTALLED RAIL
3. MONO RAIL
4. TEMPORARY INSTALLED RAIL STORAGE AREA
5. IMPELLER SHAFT GRAPPLE STORAGE AREA
6. DIFFUSER WEAR RING GRAPPLE STORAGE AREA
7. DIFFUSER STRETCH TUBE GRAPPLE STORAGE AREA
8. UPPER PLUG STORAGE AREA
9. RIP UPPER PORTION HANDLING CONNECTOR ROD STORAGE AREA

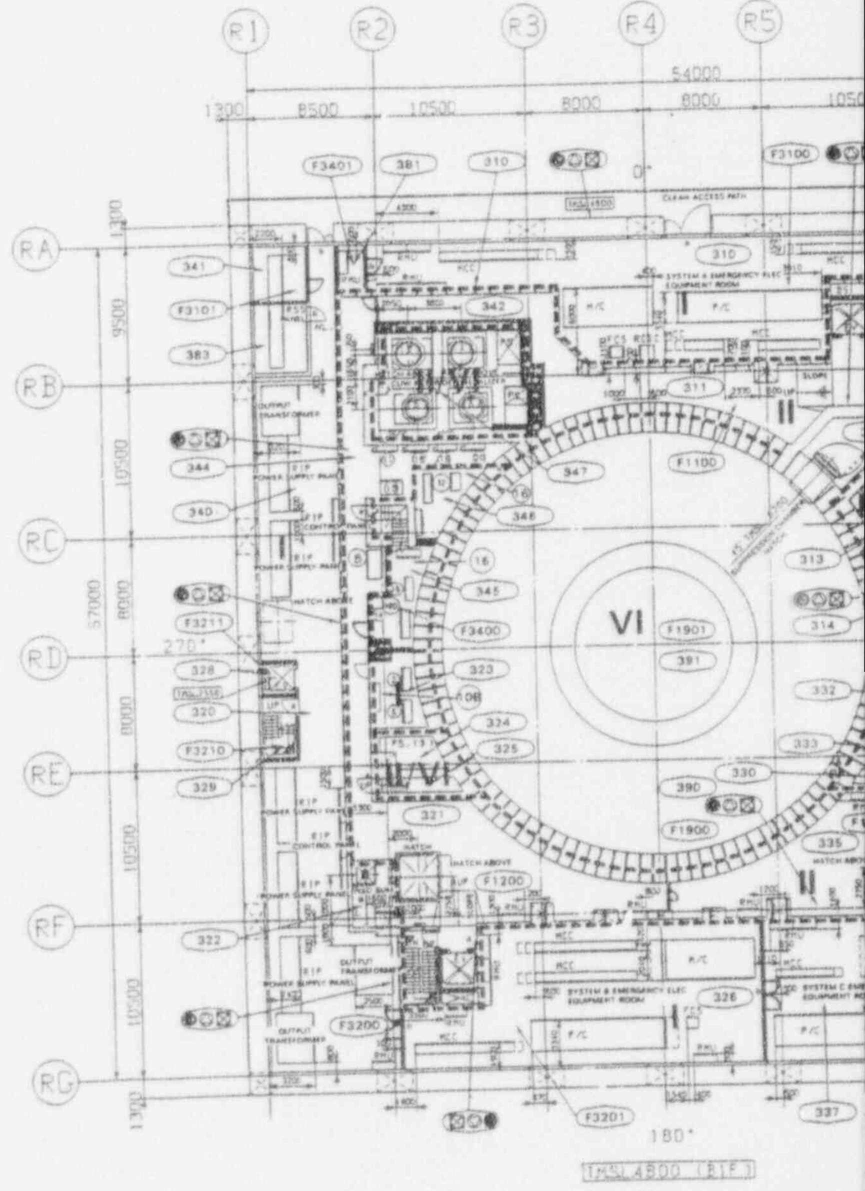
(REMARKS)
EQUIPMENT
D/S POOL
SPENT FUEL STORAGE POOL

FULL POWER/SHUTDOWN
RADIATION LEVEL IN ($\mu Sv/h$)

A $\leq$ 6
B < 10
C < 50
D < 250
E < 1000
F $\geq$ 1000

9406270337-10

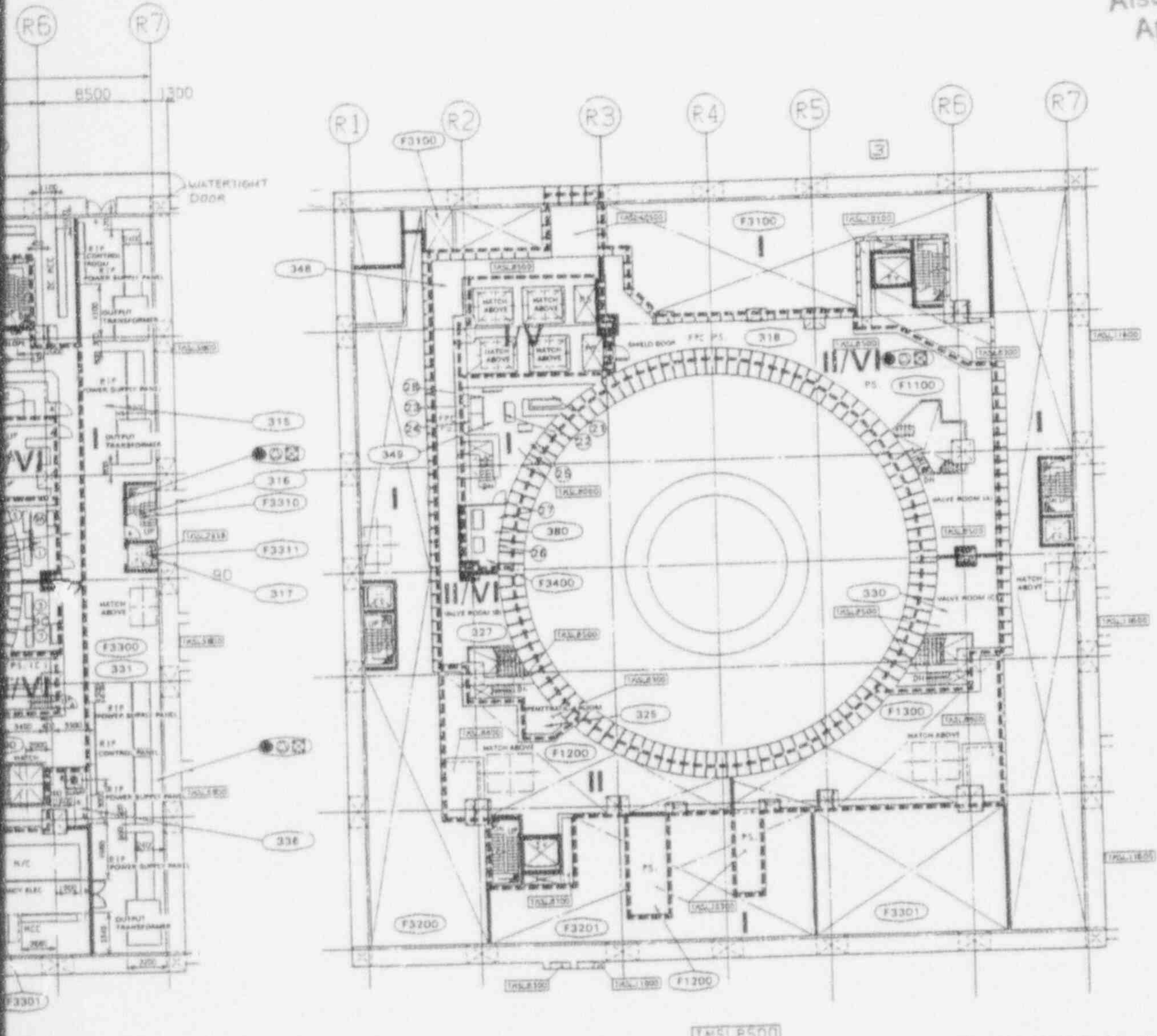
Figure 12.3-9 REACTOR BUILDING RADIATION ZONE MAP FOR FULL POWER AND SHUTDOWN OPERATION AT ELEVATION 31700/38200 mm



TMSL4800 (B1F)

(REMARKS)
EQUIPMENT
EMERGENCY ELECTRIC ROOM (A)
EMERGENCY ELECTRIC ROOM (B)
EMERGENCY ELECTRIC ROOM (C)
RSS PANEL A & B
RIP PANEL

B31	H22-P013A
B31	H22-P013B
B31	H22-P013C
B31	H22-P013D
B31	H22-P014A
B31	H22-P014B
B31	H22-P014C
B31	H22-P014D
E31	H22-P015A
E31	H22-P015C
E31	H22-P015B
E31	H22-P015D
G41	H22-P016



NO	RACK LIST
1.	REACTOR SYSTEM (I) INSTRUMENT RACK
2.	REACTOR SYSTEM (II) INSTRUMENT RACK
3.	REACTOR SYSTEM (III) INSTRUMENT RACK
4.	REACTOR SYSTEM (IV) INSTRUMENT RACK
5.	MAIN STEAM FLOW (I) INSTRUMENT RACK
6.	MAIN STEAM FLOW (II) INSTRUMENT RACK
7.	MAIN STEAM FLOW (III) INSTRUMENT RACK
8.	MAIN STEAM FLOW (IV) INSTRUMENT RACK
9A.	LEAK DETECTION SYSTEM (A) INSTRUMENT RACK
9C.	LEAK DETECTION SYSTEM (C) INSTRUMENT RACK
10B.	LEAK DETECTION SYSTEM (B) INSTRUMENT RACK
10D.	LEAK DETECTION SYSTEM (D) INSTRUMENT RACK
12.	REACTOR WATER SAMPLING TRANSMITTER PANEL
13.	FPC FD SAMPLING TRANSMITTER

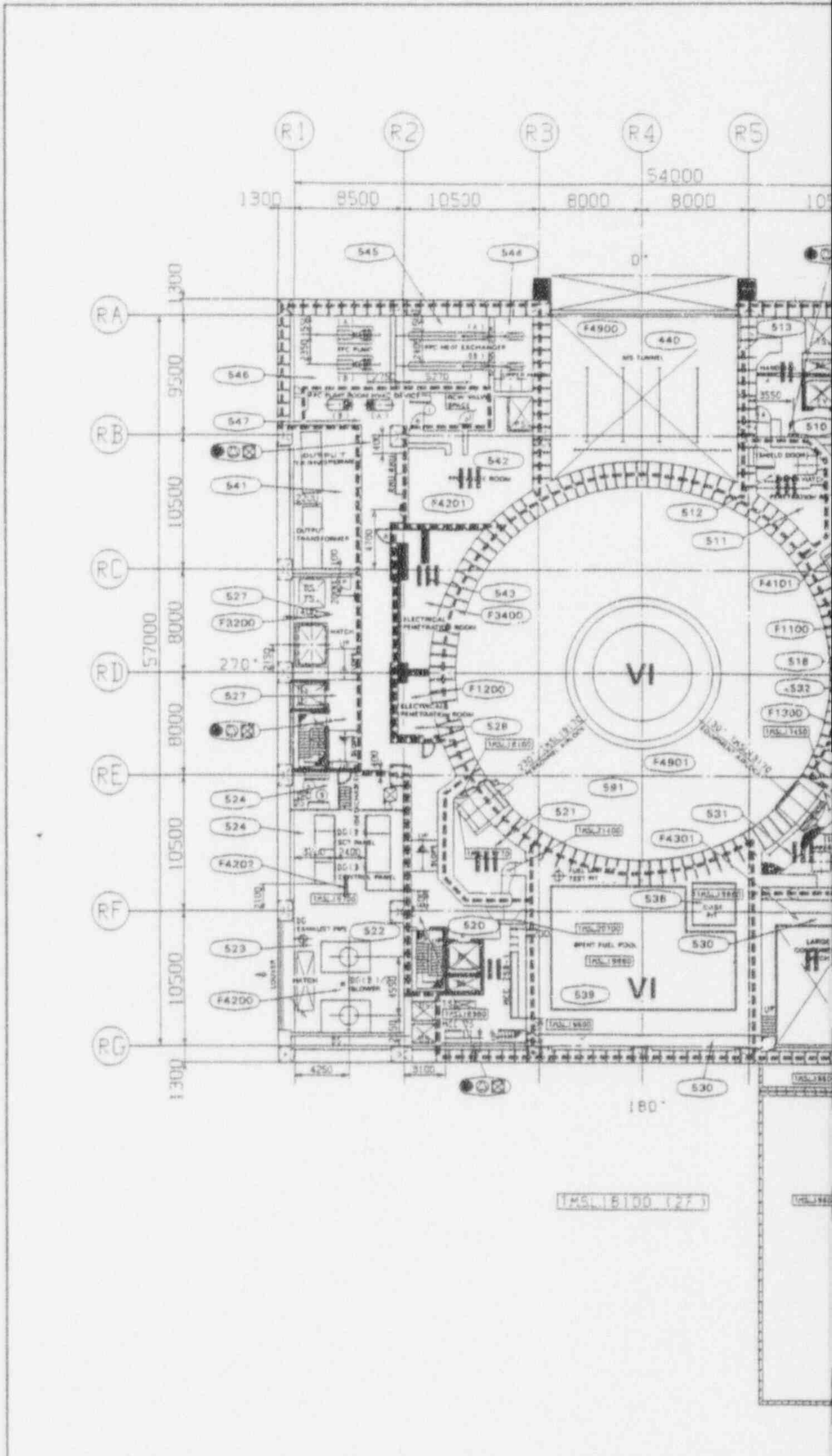
		NO	RACK LIST
G41	H22-P017	14.	FPC FD MAIN VALVE RACK
G41	H22-P018	15.	FPC FD CONDUCTIVITY METER RACK
G41	H22-P019	16.	FPC FD SAMPLING HOOD
G41	H22-P028	17.	FPC F/D INSTRUMENT RACK (B)
G41	H22-P027	18.	FPC F/D INSTRUMENT RACK (A)
G31	H22-P030	19.	CUW F/D INSTRUMENT RACK (B)
G31	H22-P029	20.	CUW F/D INSTRUMENT RACK (A)
P91	H22-P020	21.	REACTOR WATER SAMPLING COOLER RACK
P91	H22-P021	22.	REACTOR WATER SAMPLE DEPRESSURIZATION RACK
P91	H22-P022	23.	REACTOR WATER pH METER RACK
P91	H22-P023	24.	REACTOR WATER DISSOLVED OXYGEN METER
P91	H22-P024	25.	REACTOR WATER CONDUCTIVITY METER RACK
P91	H22-P025A	26.	REACTOR WATER SAMPLING HOOD
P91	H22-P025	27.	REACTOR WATER GRAB SAMPLING RACK
P91	H22-P026	28.	PASS RELATED AO VALVE RACK

ZONE DOSE RATE (Gy/h)

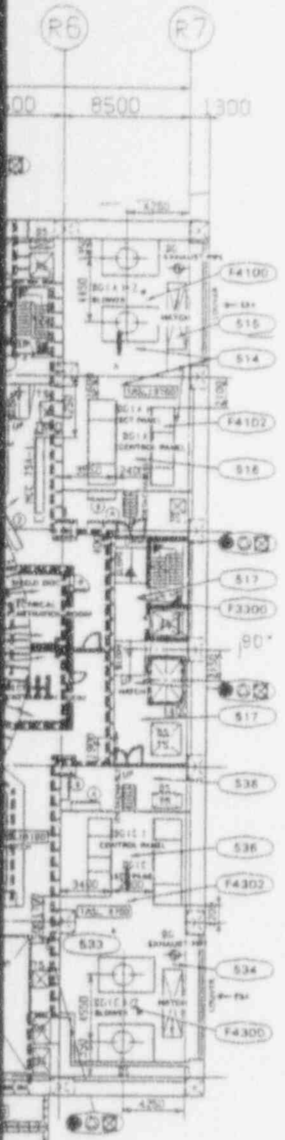
- I ≤ 0.005
- II < 0.05
- III < 0.5
- IV < 5
- V < 50
- VI ≥ 50

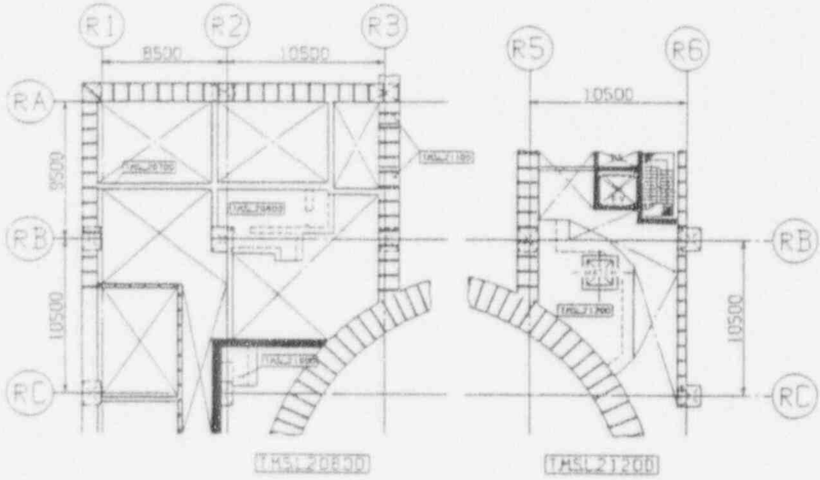
9406230337-11

Figure 12.3-14 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 4800/8500 mm



R1
R2
R3
R4
R5
54000
1300
8500
10500
8000
8000
0°
RA
RB
RC
RD
RE
RF
RG
9500
10500
8000
8000
10500
10500
1300
57000
270°
180°
545
544
546
547
541
527
F3200
527
524
524
F4202
523
F4200
522
520
F4900
440
542
F4201
512
511
543
F3400
F1200
528
F4901
591
521
531
F4301
538
530
539
530
513
510
F4101
F1100
518
532
F1300
OUTPUT TRANSFORMER
ELECTRICAL PENETRATION ROOM
SPENT FUEL POOL
VI
VI
4250
3100
TMSL18100 (2F)





(REMARKS)
EQUIPMENT

FPC HX
FPC PUMP
FUEL HANDLING MACHINE TEST PIT
SPENT FUEL STORAGE POOL
CASK PIT
D/G(A) CONTROL PANEL
D/G(B) CONTROL PANEL
D/G(C) CONTROL PANEL
DG(A)/Z SUPPLY FAN
DG(B)/Z SUPPLY FAN
DG(C)/Z SUPPLY FAN

INSTRUMENT RACK LIST

	NO.	RACK NAME
H22-P042	1	FUEL POOL COOLING CLEANUP SYSTEM INSTRUMENT RACK
H22-P041	2	MSIV LEAK TEST INSTRUMENT RACK

FOAM FIRE EXTINGUISHER LIST

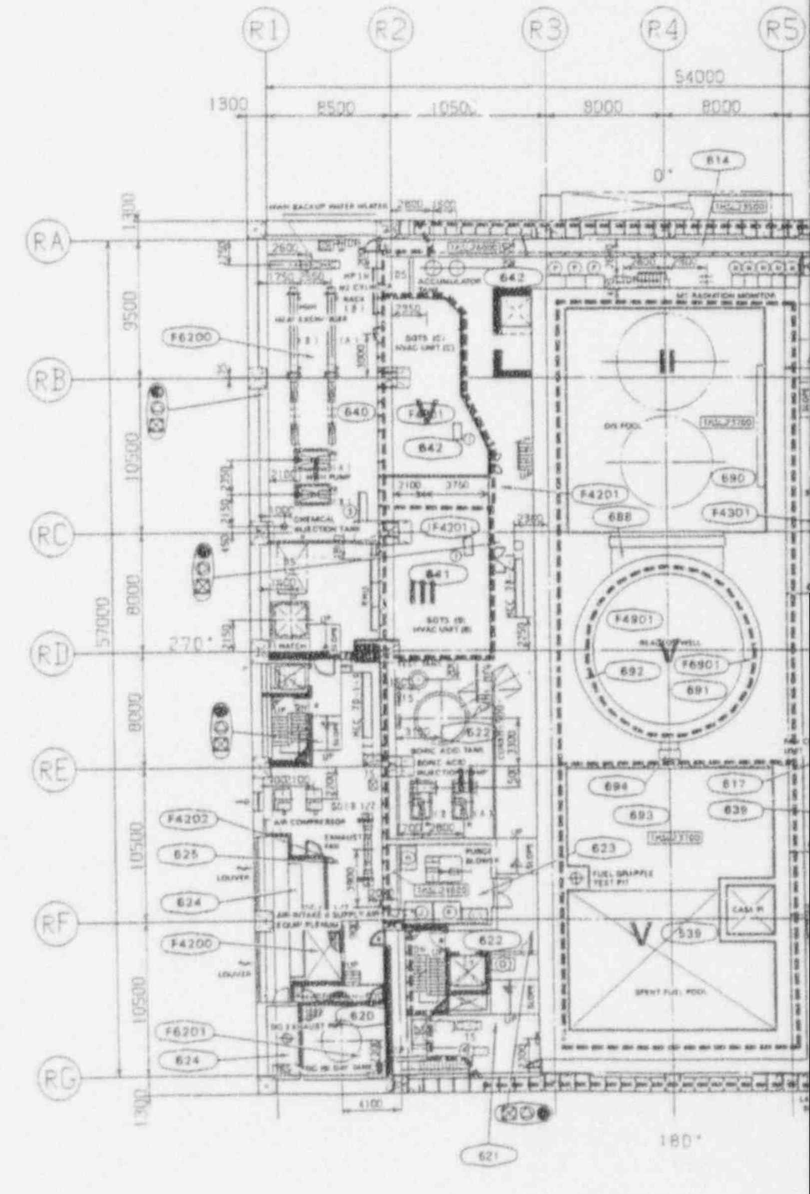
NO	RACK NAME
A	FOAM LIQUID TANK
B	FOAM INJECTION EQUIPMENT

ZONE DOSE RATE (Gy/h)

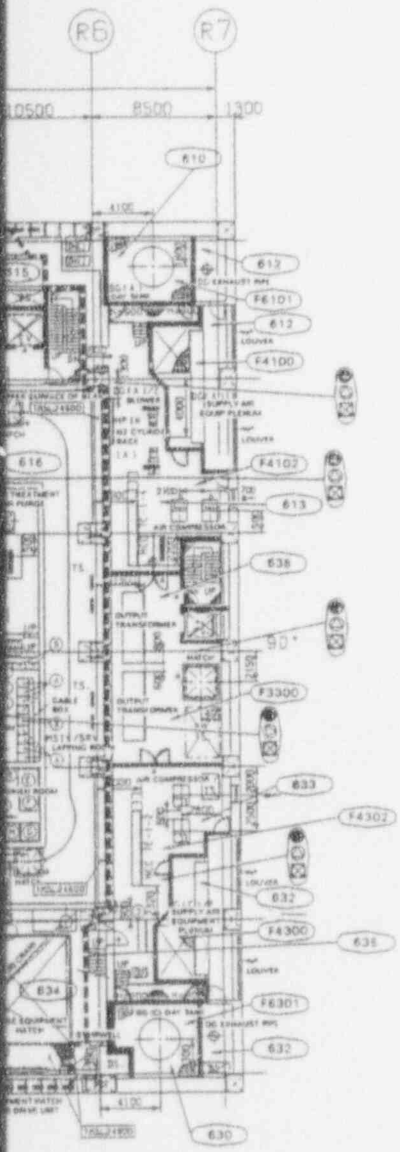
I $\leq$ 0.005
II $<$ 0.05
III $<$ 0.5
IV $<$ 5
V $<$ 50
VI $\geq$ 50

9406230337-12

Figure 12.3-17 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 18100 mm



T.M.S.L. 23500 (3F)



INSTRUMENT RACK

		NO	RACK NAME
C41	H22-PO42	1.	STANDBY GAS TREATMENT SYSTEM INSTRUMENT RACK
D23	H22-PO44A	2.	CONTAINMENT VESSEL ATMOSPHERE MONITOR CALIBRATION GAS CYLINDER RACK A
D23	H22-PO44B	3.	CONTAINMENT VESSEL ATMOSPHERE MONITOR CALIBRATION GAS CYLINDER RACK B
D23	H22--PO54B	4.	CONTAINMENT VESSEL ATMOSPHERE MONITOR CALIBRATION RACK (B)

ISI ROOM AND AUXILIARY FACILITIES

NO	QTY	FACILITY NAME
A	8	CONTROL DATA COLLECTION EQUIPMENT
B	1	STABILIZED POWER SUPPLY SYSTEM
C	3	DESK
D	2	STORAGE
E	1	CALIBRATION TEST PIECE FOR M/S NOZZLE CORNER
F	1	CALIBRATION TEST PIECE FOR NOZZLE CORNER
G	1	CALIBRATION TEST PIECE FOR NOZZLE CORNER
H	1	CALIBRATION TEST PIECE FOR NOZZLF CORNER
J	1	RPV SHELL ADJUST TEST FACILITY
K	1	RPV BOTTOM PLATE ADJUST TEST FACILITY
L	1	RPV NOZZLE ADJUST TEST FACILITY
M	1	PIPING ADJUST TEST FACILITY
N	5	ISI DEVICE STORAGE
P	3	ISI DEVICE STORAGE
Q	1	RPV CALIBRATION TEST PIECE STORAGE
R	2	RPV CONSUMABLE MATERIALS AND CALIBRATION TEST PIECE STORAGE
S	2	PIPING CALIBRATION TEST PIECE STORAGE

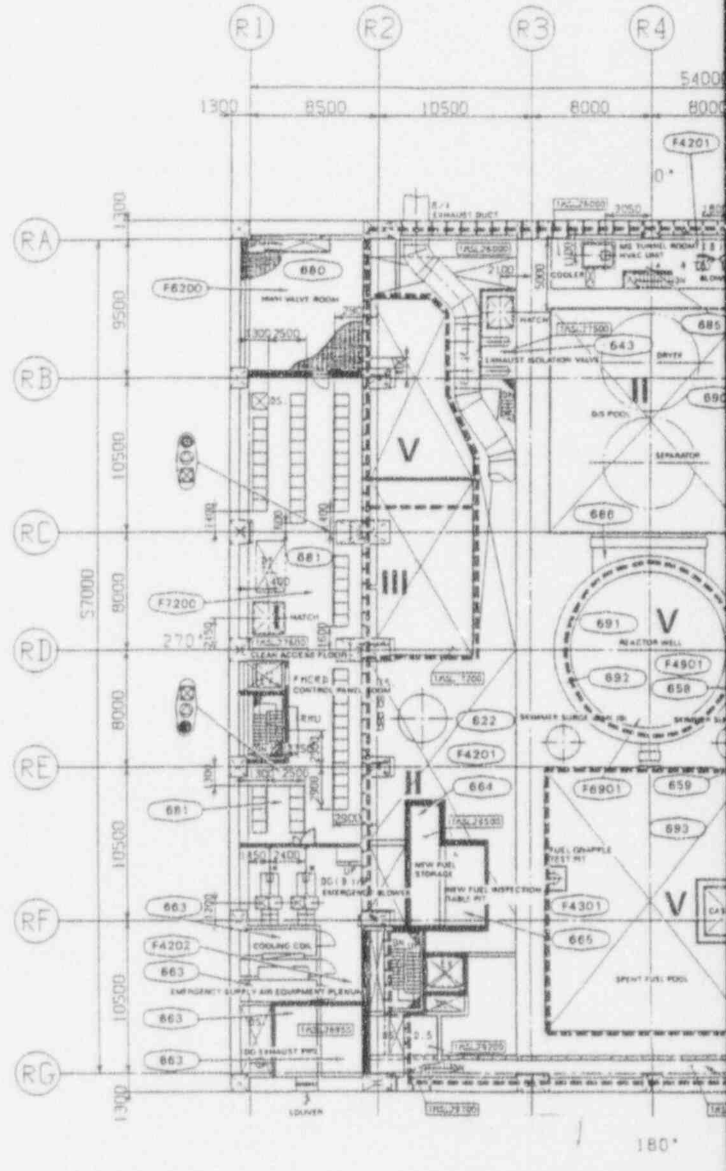
(REMARKS)
EQUIPMENT

D/S POOL
CASK PIT
SPENT FUEL STORAGE POOL
ISI INSPECTION ROOM
SLC PUMP (A)
SLC PUMP (B)
SLC TANK
SLC TEST TANK
DG (A) DAY TANK
DG (B) DAY TANK
DG (C) DAY TANK
HWH PUMP
HWH HX

ZONE DOSE RATE (Gy/h)
I ≤ 0.005
II < 0.05
III < 0.5
IV < 5
V < 50
VI ≥ 50

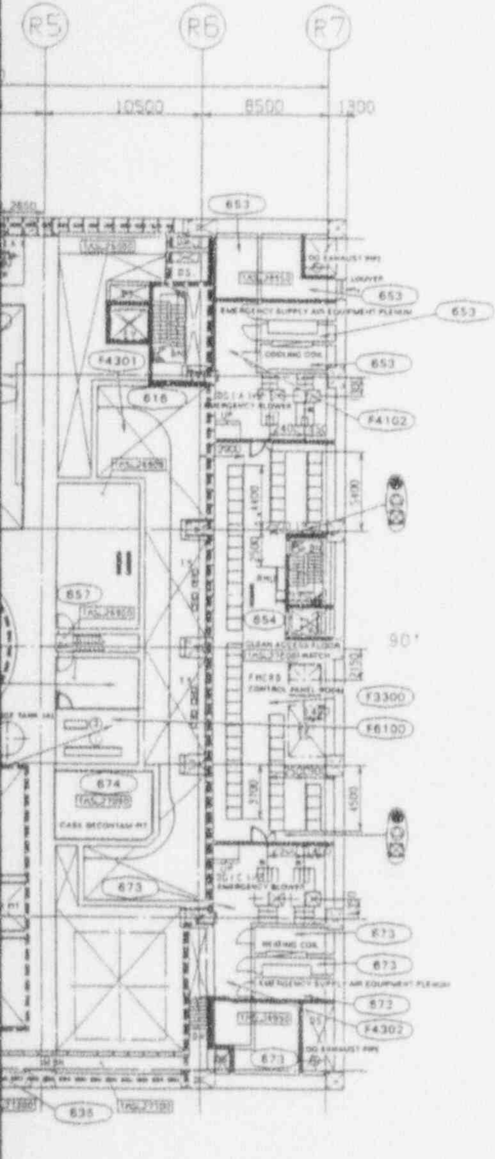
9406230337-13

Figure 12.3-18 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 23500 mm



R1
R2
R3
R4
RA
RB
RC
RD
RE
RF
RG
1300
8500
10500
8000
9500
57000
0°
270°
180°
MAIN VALVE ROOM
HATCH
CLEAN ACCESS FLOOR
FMCRD CONTROL PANEL ROOM
COOLING COIL
EMERGENCY SUPPLY AIR EQUIPMENT PLENUM
LOUVER
D/S POOL
SEPARATOR
DRYER
REACTOR WELL
NEW FUEL STORAGE
NEW FUEL INSPECTION
SPENT FUEL POOL
FUEL GRAPPLE TEST PIT

T.M.S.L. 27200



RACK LIST

	NO	RACK NAME
D23 H22 P053A	1	CONTAINMENT VESSEL ATMOSPHERE MONITOR RACK (A)
D23 H22 P054A	2	CONTAINMENT VESSEL ATMOSPHERE MONITOR CALIBRATION RACK (A)

(REMARKS)
EQUIPMENT

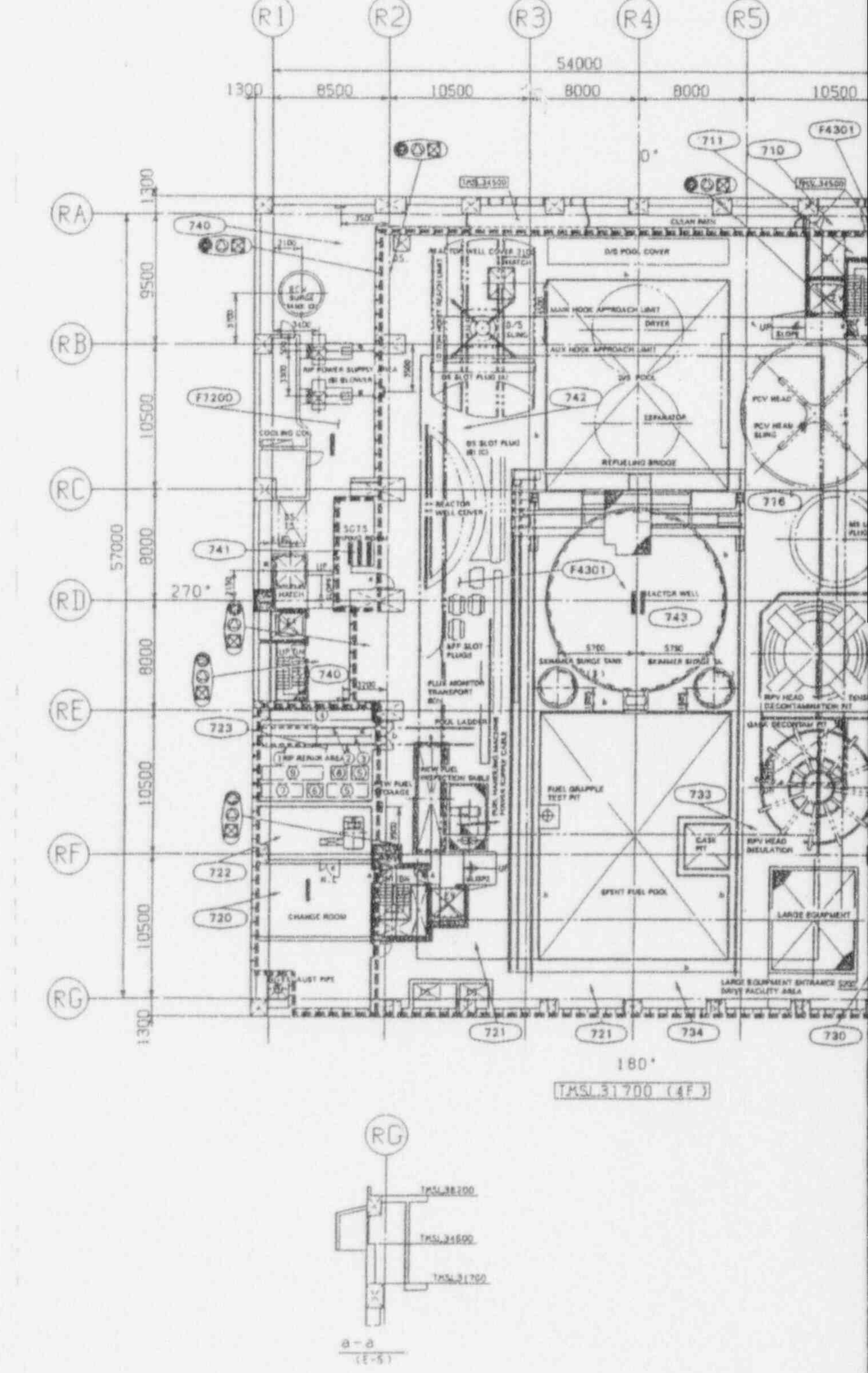
D/S POOL
CASK PIT
SPENT FUEL STORAGE POOL
CASK WASHDOWN PIT
FMCRD PANEL ROOM
NEW FUEL STORAGE PIT
NEW FUEL INSPECTION PIT

ZONE DOSE RATE (Gy/h)

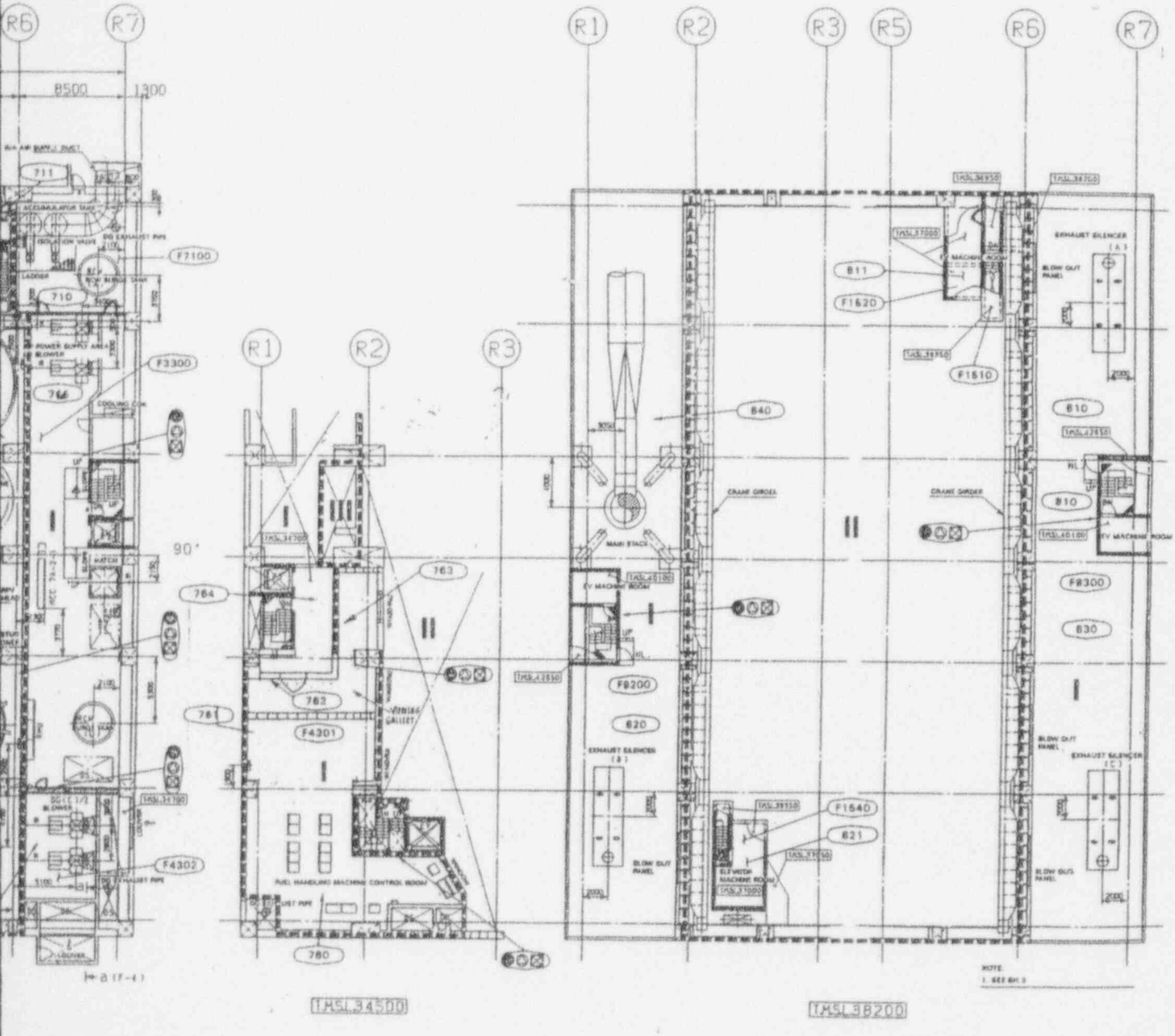
I ≤ 0.005
II < 0.05
III < 0.5
IV < 5
V < 50
VI ≥ 50

G406230337-14

Figure 12.3-19 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 27200 mm



R1
R2
R3
R4
R5
54000
1300
8500
10500
8000
8000
10500
RA
RB
RC
RD
RE
RF
RG
57000
9500
10500
8000
8000
10500
10500
1300
1300
0°
270°
180°
740
741
742
743
F7200
F4301
711
710
716
723
722
720
721
721
734
733
730
REACTOR WELL COVER
D/S POOL COVER
MAIN HOOK APPROACH LIMIT
AUX HOOK APPROACH LIMIT
DRYER
D/S POOL
SEPARATOR
REFUELING BRIDGE
SCTS
HATCH
REACTOR WELL
SKIMMER SURGE TANK
POOL LADDER
FUEL GRAPPLE TEST PIT
SPENT FUEL POOL
CHANGE ROOM
RIP REPAIR AREA
NEW FUEL INSPECTION TABLE
RPV HEAD INSULATION
LARGE EQUIPMENT
LARGE EQUIPMENT ENTRANCE
CLEAN PATH
T.M.S.L.31700 (4F)
RG
a-a
(E-5)



NO	RACK NAME
1.	INSPECTION POOL
2.	TEMPORARY INSTALLED RAIL
3.	MONO-RAIL
4.	TEMPORARY INSTALLED RAIL STORAGE AREA
5.	IMPELLER SHAFT GRAPPLE STORAGE AREA
6.	DIFFUSER WEAR RING GRAPPLE STORAGE AREA
7.	DIFFUSER STRETCH TUBE GRAPPLE STORAGE AREA
8.	UPPER PLUG STORAGE AREA
9.	RIP UPPER PORTION HANDLING CONNECTOR ROD STORAGE AREA

(REMARKS)
EQUIPMENT
D/S POOL
SPENT FUEL STORAGE POOL

ZONE DOSE RATE (Gy/h)

I ≤ 0.005
II < 0.05
III < 0.5
IV < 5
V < 50
VI ≥ 50

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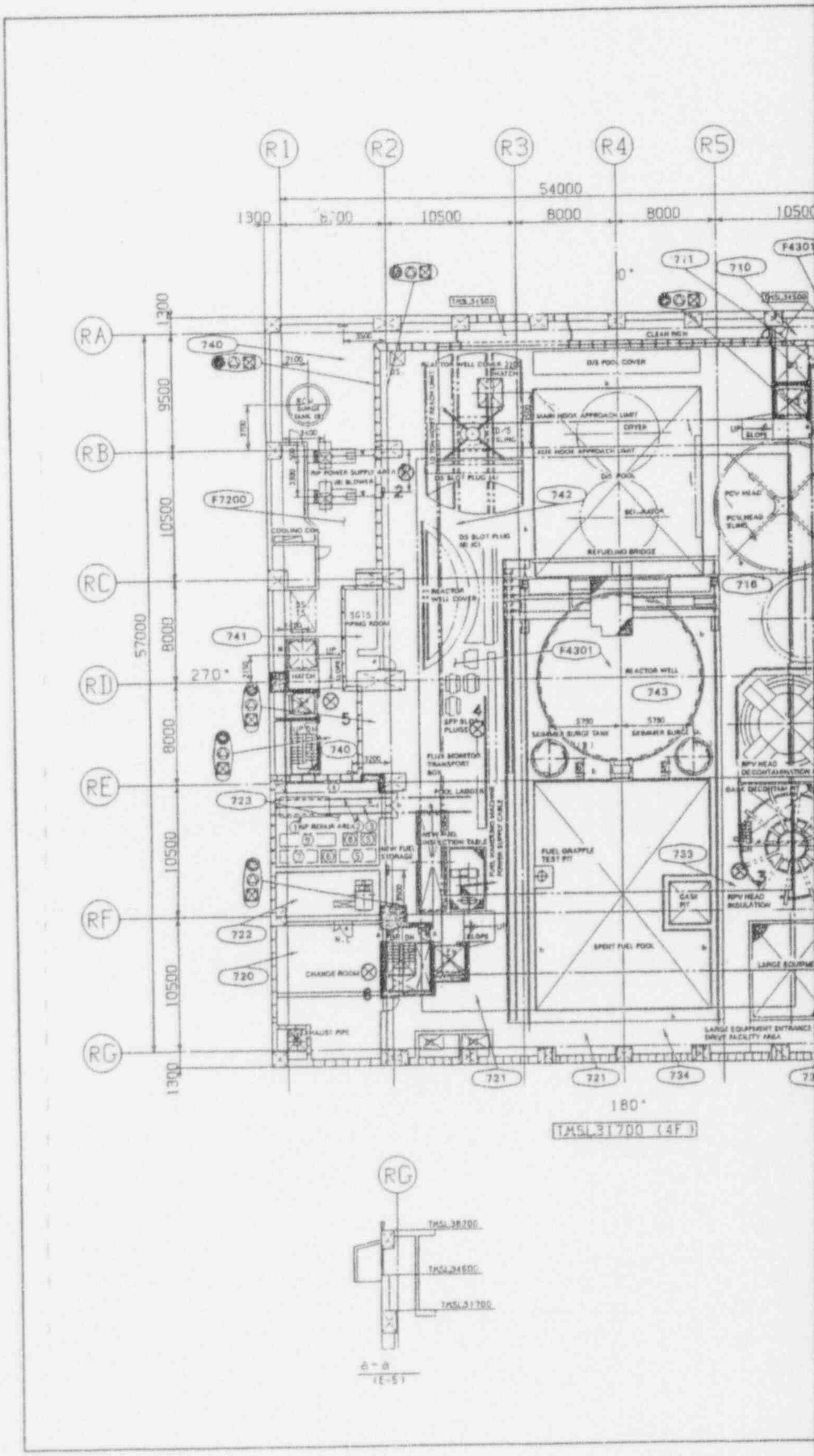
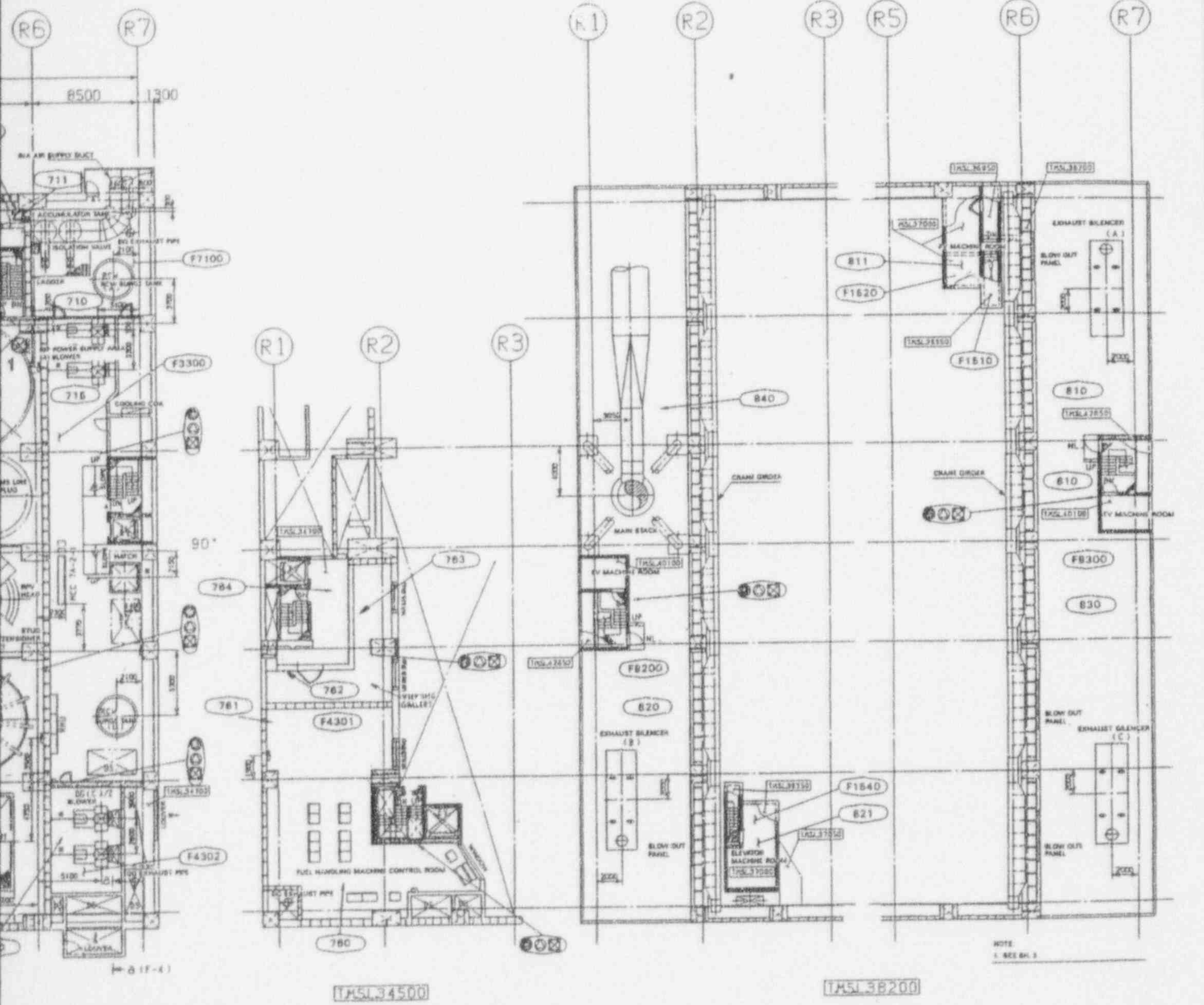
Figure 12.3-20 REACTOR BUILDING RADIATION ZONE MAP POST LOCA AT ELEVATION 31700/38200 mm


R1
R2
R3
R4
R5
54000
1300
8700
10500
8000
8000
10500
RA
RB
RC
RD
RE
RF
RG
1300
9500
10500
8000
8000
10500
10500
1300
57000
270°
0°
180°
TMSL31700 (4F)
740
F7200
741
740
723
722
720
721
721
734
742
743
F4301
F4301
711
710
716
733
REACTOR WELL
SPENT FUEL POOL
REFUELING BRIDGE
FUEL GRAPPLE TEST PIT
CHANGE ROOM
RG
a-a
(E-5)



(NO) RACK NAME

1. INSPECTION POOL
2. TEMPORARY INSTALLED RAIL
3. MONO-RAIL
4. TEMPORARY INSTALLED RAIL STORAGE AREA
5. IMPELLER SHAFT GRAPPLE STORAGE AREA
6. DIFFUSER WEAR RING GRAPPLE STORAGE AREA
7. DIFFUSER STRETCH TUBE GRAPPLE STORAGE AREA
8. UPPER PLUG STORAGE AREA
9. RIP UPPER PORTION HANDLING CONNECTOR ROD STORAGE AREA

(REMARKS)
EQUIPMENT
D/S POOL
SPENT FUEL STORAGE POOL

AREA RADIATION MONITOR SYMBOL

$\otimes_n$ n = MONITOR ASSIGNED NUMBER

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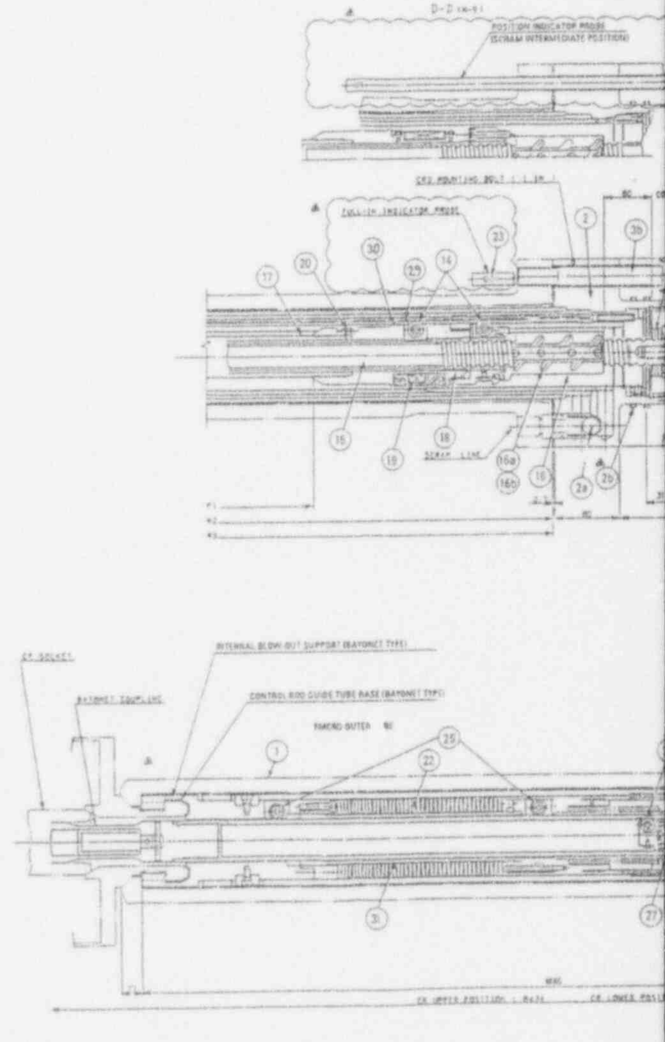
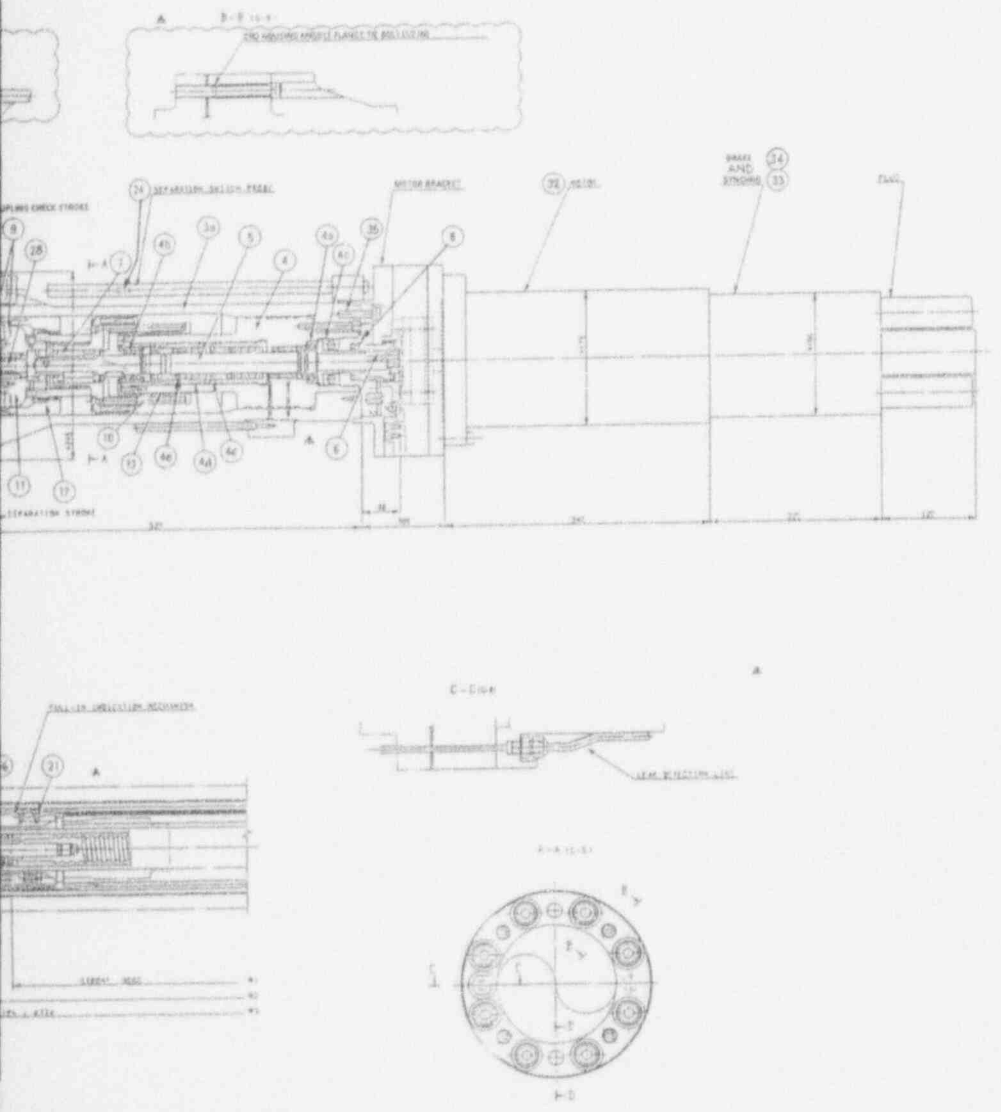
Figure 12.3-62 REACTOR BUILDING AREA RADIATION MONITORS AT ELEVATION 31700mm



POSITION INDICATOR PROBE
(SCRAM INTERMEDIATE POSITION)
CRD MOUNTING BOLT
FULL-IN INDICATOR PROBE
SCRAM LINE
INTERNAL BLOW-OUT SUPPORT (BAYONET TYPE)
CONTROL ROD GUIDE TUBE BASE (BAYONET TYPE)
CR SOCKET
BAYONET COUPLING
17
20
30
29
14
23
2
3b
15
19
18
16a
16b
16
2a
2b
1
25
22
31
27



ANSTEC APERTURE CARD

Also Available on Aperture Card

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Figure 15B-3 FINE MOTION CONTROL ROD DRIVE