GE Nuclear Energy

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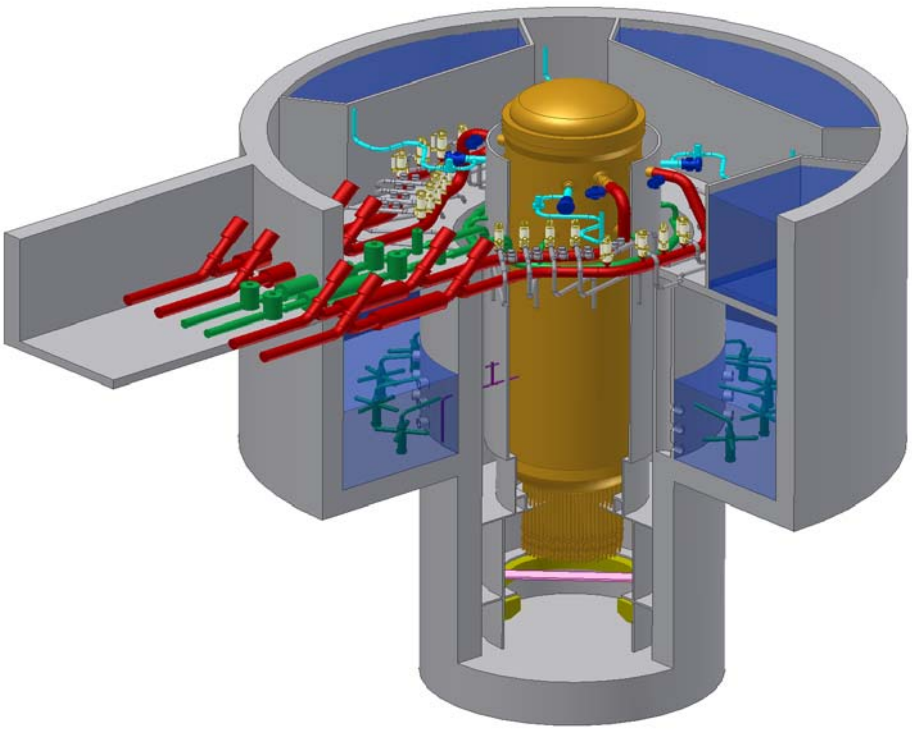

ESBWR Design Control Document

Tier 1

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Global Abbreviations And Acronyms List

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
A/D	Analog-to-Digital
AASHTO	American Association of Highway and Transportation Officials
AB	Auxiliary Boiler
ABMA	Anti-Friction Bearing Manufacturers Association
ABS	Auxiliary Boiler System
ABWR	Advanced Boiling Water Reactor
ac / AC	Alternating Current
AC	Air Conditioning
ACF	Automatic Control Function
ACI	American Concrete Institute
ACS	Atmospheric Control System
AD	Administration Building
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
AFIP	Automated Fixed In-Core Probe
AGMA	American Gear Manufacturer's Association
AHS	Auxiliary Heat Sink
AHU	Air Handling Units
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
AL	Analytical Limit
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
AMCA	Air Movement and Control Association
ANI	American Nuclear Insurers
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
API	American Petroleum Institute
APRM	Average Power Range Monitor
APR	Automatic Power Regulator
APRS	Automatic Power Regulator System
ARI	Alternate Rod Insertion
ARI	Air-Conditioning and Refrigeration Institute
ARMS	Area Radiation Monitoring System
ASA	American Standards Association

Global Abbreviations And Acronyms List

Term	Definition
ASA	Acoustical Society of America
ASCE	American Society of Civil Engineers
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASQ	American Society for Quality
AST	Alternate Source Term
ASTM	American Society of Testing Methods
ASTM	American Society for Testing and Materials
AT	Unit Auxiliary Transformer
ATLM	Automated Thermal Limit Monitor
ATWS	Anticipated Transients Without Scram
AV	Allowable Value
AWS	American Welding Society
AWWA	American Water Works Association
B&PV	Boiler and Pressure Vessel
BAF	Bottom of Active Fuel
BHP	Brake Horse Power
BiMAC	Basemat-Internal Melt Arrest Coolability
BOC	Beginning of Cycle
BOP	Balance of Plant
BPU	Bypass Unit
BPV	Bypass Valve
BPWS	Banked Position Withdrawal Sequence
BRE	Battery Room Exhaust
BRL	Background Radiation Level
BTP	NRC Branch Technical Position
BTU	British Thermal Unit
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAV	Cumulative Absolute Velocity
C&FS	Condensate and Feedwater System
C&I	Control and Instrumentation
C/C	Cooling and Cleanup
CB	Control Building
CBGAHVS	Control Building General Area
CBHVAC	Control Building HVAC
CBHVS	Control Building Heating, Ventilation and Air Conditioning System
CCI	Core-Concrete Interaction

Global Abbreviations And Acronyms List

Term	**Definition**
CDF	Core Damage Frequency
CDU	Condensing Unit
CEA	Consumer Electronics Association
CFR	Code of Federal Regulations
CH	Chugging
CIRC	Circulating Water System
CIS	Containment Inerting System
CIV	Combined Intermediate Valve
CLAVS	Clean Area Ventilation Subsystem of Reactor Building HVAC
CM	Cold Machine Shop
CMS	Containment Monitoring System
CMU	Control Room Multiplexing Unit
CO	Condensate Oscillation
COL	Combined Operating License
COLR	Core Operating Limits Report
CONAVS	Controlled Area Ventilation Subsystem of Reactor Building HVAC
CPR	Critical Power Ratio
CPS	Condensate Purification System
CPU	Central Processing Unit
CR	Control Rod
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDH	Control Rod Drive Housing
CRDHS	Control Rod Drive Hydraulic System
CRDS	Control Rod Drive System
CRGT	Control Rod Guide Tube
CRHA	Control Room Habitability Area
CRHAHVS	Control Room Habitability Area HVAC Subsystem
CRT	Cathode Ray Tube
CS&TS	Condensate Storage and Transfer System
CSDM	Cold Shutdown Margin
CS / CST	Condensate Storage Tank
CT	Main Cooling Tower
CTI	Cooling Technology Institute
CTSS	Communications Continuous Tone-Controlled Squelch System
CTVCF	Constant Voltage Constant Frequency
CUF	Cumulative usage factor
CWS	Chilled Water System
D-RAP	Design Reliability Assurance Program

Global Abbreviations And Acronyms List

Term	Definition
DAC	Design Acceptance Criteria
DAW	Dry Active Waste
DBA	Design Basis Accident
DBE	Design Basis Event
DB%	Dry-Basis-Percent
dc / DC	Direct Current
DCD	Design Control Document
DCS	Drywell Cooling System
DCIS	Distributed Control and Information System
DEPSS	Drywell Equipment and Pipe Support Structure
DF	Decontamination Factor
D/F	Diaphragm Floor
DG	Diesel-Generator
DHR	Decay Heat Removal
DPS	Diverse Protection System
DM&C	Digital Measurement and Control
DOF	Degree of Freedom
DOI	Dedicated Operators Interface
DORT	Discrete Ordinates Techniques
DOT	Department of Transportation
dPT	Differential Pressure Transmitter
DPS	Diverse Protection System
DPV	Depressurization Valve
DR&T	Design Review and Testing
DTM	Digital Trip Module
DW	Drywell
EAB	Exclusion Area Boundary
EB	Electrical Building
EBAS	Emergency Breathing Air System
EBHV	Electrical Building HVAC
ECA	Electronic Components Assemblies Materials Association
ECCS	Emergency Core Cooling System
E-DCIS	Essential DCIS (Distributed Control and Information System)
EDO	Environmental Qualification Document
EFDS	Equipment and Floor Drainage System
EFPY	Effective Full Power Years
EFU	Emergency Filter Unit
EHC	Electro-Hydraulic Control (Pressure Regulator)
EIA	Electronic Industries Alliance

Global Abbreviations And Acronyms List

Term	**Definition**
ENS	Emergency Notification System
EOC	Emergency Operations Center
EOC	End of Cycle
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EPDS	Electric Power Distribution System
EPG	Emergency Procedure Guidelines
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ERICP	Emergency Rod Insertion Control Panel
ERIP	Emergency Rod Insertion Panel
ESF	Engineered Safety Feature
ESP	Early Site Permit
ETS	Emergency Trip System
FAC	Flow-Accelerated Corrosion
FAPCS	Fuel and Auxiliary Pools Cooling System
FATT	Fracture Appearance Transition Temperature
FB	Fuel Building
FBHV	Fuel Building HVAC
FCI	Fuel-Coolant Interaction
FCI	Fluid Controls Institute Inc.
FCISL	Fuel Cladding Integrity Safety Limit
FCM	File Control Module
FCS	Flammability Control System
FCU	Fan Cooling Unit
FDA	Final Design Approval
FDDI	Fiber Distributed Data Interface
FEBAVS	Fuel Building Ventilation System
FFT	Fast Fourier Transform
FFWTR	Final Feedwater Temperature Reduction
FHA	Fire Hazards Analysis
FHA	Fuel Handling Accident
FIV	Flow-Induced Vibration
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analysis
FPS	Fire Protection System
FO	Diesel Fuel Oil Storage Tank
FOAKE	First-of-a-Kind Engineering
FPC	Fuel Pool Cleanup

Global Abbreviations And Acronyms List

Term	Definition
FPE	Fire Pump Enclosure
FS	Partial Full Scale
FSI	Fluid Structure Interaction
FTDC	Fault-Tolerant Digital Controller
FW	Feedwater
FWCS	Feedwater Control System
FWLB	Feedwater Line Break
FWS	Fire Water Storage Tank
GCS	Generator Cooling System
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GEN	Main Generator System
GENE	General Electric Nuclear Energy
GETAB	General Electric Thermal Analysis Basis
GL	Generic Letter
GM	Geiger-Mueller Counter
GM-B	Beta-Sensitive GM (Geiger-Mueller Counter) Detector
GNF	Global Nuclear Fuel
GSIC	Gamma-Sensitive Ion Chamber
GSOS	Generator Sealing Oil System
GWSR	Ganged Withdrawal Sequence Restriction
HAZ	Heat-Affected Zone
HCU	Hydraulic Control Unit
HCW	High Conductivity Waste
HDVS	Heater Drain and Vent System
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HELSA	High Energy Line Separation Analysis
HEP	Human Error Probability
HEPA	High Efficiency Particulate Air/Absolute
HFE	Human Factors Engineering
HFF	Hollow Fiber Filter
HGCS	Hydrogen Gas Cooling System
HI	Hydraulic Institute
HIC	High Integrity Container
HID	High Intensity Discharge
HIS	Hydraulic Institute Standards
HM	Hot Machine Shop & Storage

Global Abbreviations And Acronyms List

Term	Definition
HP	High Pressure
HPNSS	High Pressure Nitrogen Supply System
HPT	High-Pressure Turbine
HRA	Human Reliability Assessment
HSI	Human-System Interface
HSSS	Hardware/Software System Specification
HVAC	Heating, Ventilation and Air Conditioning
HVS	High Velocity Separator
HVT	Horizontal Vent Test
HWC	Hydrogen Water Chemistry
HWCS	Hydrogen Water Chemistry System
HWS	Hot Water System
HX	Heat Exchanger
I&C	Instrumentation and Control
I/O	Input/Output
IAS	Instrument Air System
IASCC	Irradiation Assisted Stress Corrosion Cracking
IBA	Intermediate Break Accident
IBC	International Building Code
IC	Ion Chamber
IC	Isolation Condenser
ICC	International Code Council
ICD	Interface Control Diagram
ICP	Instrument and Control Power
ICPR	Initial Critical Power Ratio
ICS	Isolation Condenser System
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IEC	International Electrotechnical Commission
IED	Instrument and Electrical Diagram
IEEE	Institute of Electrical and Electronic Engineers
IESNA	Illuminating Engineering Society of North America
IFTS	Inclined Fuel Transfer System
IGSCC	Intergranular Stress Corrosion Cracking
IIS	Iron Injection System
ILRT	Integrated Leak Rate Test
IOP	Integrated Operating Procedure
IMC	Induction Motor Controller
IMCC	Induction Motor Controller Cabinet

Global Abbreviations And Acronyms List

Term	Definition
IRM	Intermediate Range Monitor
ISA	Instrument Society of America
ISI	In-Service Inspection
ISLT	In-Service Leak Test
ISM	Independent Support Motion
ISMA	Independent Support Motion Response Spectrum Analysis
ISO	International Standards Organization
ITA	Inspections, Tests or Analyses
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
ITA	Initial Test Program
LAPP	Loss of Alternate Preferred Power
LBB	Leak Before Break
LCO	Limiting Conditions for Operation
LCS	Leakage Control System
LCW	Low Conductivity Waste
LD	Logic Diagram
LDA	Lay down Area
LDW	Lower Drywell
LD&IS	Leak Detection and Isolation System
LED	Light Emitting Diode
LERF	Large Early Release Frequency
LFCV	Low Flow Control Valve
LHGR	Linear Heat Generation Rate
LLRT	Local Leak Rate Test
LMU	Local Multiplexer Unit
LO	Dirty/Clean Lube Oil Storage Tank
LOCA	Loss-of-Coolant-Accident
LOFW	Loss-of-feedwater
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPCRD	Locking Piston Control Rod Drive
LPMS	Loose Parts Monitoring System
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LUA	Lead Use Assembly
LWMS	Liquid Waste Management System
MAAP	Modular Accident Analysis Program

Global Abbreviations And Acronyms List

Term	**Definition**
MAPLHGR	Maximum Average Planar Linear Head Generation Rate
MAPRAT	Maximum Average Planar Ratio
MBB	Motor Built-In Brake
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MCOP	Manual containment overpressure protection (function)
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MCRP	Main Control Room Panel
MELB	Moderate Energy Line Break
MLHGR	Maximum Linear Heat Generation Rate
MMI	Man-Machine Interface
MMIS	Man-Machine Interface Systems
MOV	Motor-Operated Valve
MPC	Maximum Permissible Concentration
MPL	Master Parts List
MRBM	Multi-Channel Rod Block Monitor
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSLB	Main Steamline Break
MSLBA	Main Steamline Break Accident
MSR	Moisture Separator Reheater
MSS	Manufacturers Standardization Society
MSV	Mean Square Voltage
MT	Main Transformer
MTTR	Mean Time To Repair
MWS	Makeup Water System
NBR	Nuclear Boiler Rated
NBS	Nuclear Boiler System
NCIG	Nuclear Construction Issues Group
NDE	Nondestructive Examination
NE-DCIS	Non-Essential Distributed Control and Information System
NDRC	National Defense Research Committee
NDT	Nil Ductility Temperature
NEMA	National Electrical Manufacturers Association
NFPA	National Fire Protection Association
NIST	National Institute of Standard Technology
NICWS	Nuclear Island Chilled Water Subsystem

Global Abbreviations And Acronyms List

Term	Definition
NMS	Neutron Monitoring System
NOV	Nitrogen Operated Valve
NPHS	Normal Power Heat Sink
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NS	Non-seismic
NSSFC	National Severe Storms Forecast Center
NSSS	Nuclear Steam Supply System
NT	Nitrogen Storage Tank
NTSP	Nominal Trip Setpoint
O&M	Operation and Maintenance
O-RAP	Operational Reliability Assurance Program
OBCV	Overboard Control Valve
OBE	Operating Basis Earthquake
OGS	Offgas System
OHLHS	Overhead Heavy Load Handling System
OIS	Oxygen Injection System
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLU	Output Logic Unit
OOS	Out-of-Service
OPRM	Oscillation Power Range Monitor
ORNL	Oak Ridge National Laboratory
OSC	Operational Support Center
OSHA	Occupational Safety and Health Administration
OSI	Open Systems Interconnect
P&ID	Piping and Instrumentation Diagram
PA/PL	Page/Party-Line
PABX	Private Automatic Branch (Telephone) Exchange
PAM	Post Accident Monitoring
PAR	Passive Autocatalytic Recombiner
PAS	Plant Automation System
PASS	Post Accident Sampling Subsystem of Containment Monitoring System
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PCV	Primary Containment Vessel
PDA	Piping Design Analysis
PFD	Process Flow Diagram

Global Abbreviations And Acronyms List

Term	**Definition**
PGA	Peak Ground Acceleration
PGCS	Power Generation and Control Subsystem of Plant Automation System
PH	Pump House
PL	Parking Lot
PM	Preventive Maintenance
PMCS	Performance Monitoring and Control Subsystem of NE-DCIS
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PQCL	Product Quality Check List
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitoring
PS	Plant Stack or Pool Swell
PSD	Power Spectral Density
PSS	Process Sampling System
PSTF	Pressure Suppression Test Facility
PSWS	Plant Service Water System
PT	Pressure Transmitter
PWR	Pressurized Water Reactor
QA	Quality Assurance
RACS	Rod Action Control Subsystem
RAM	Reliability, Availability and Maintainability
RAPI	Rod Action and Position Information
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBC	Rod Brake Controller
RBCC	Rod Brake Controller Cabinet
RBCWS	Reactor Building Chilled Water Subsystem
RBHV	Reactor Building HVAC (Heating, Ventilation and Air Conditioning)
RBS	Rod Block Setpoint
RBV	Reactor Building Vibration
RC&IS	Rod Control and Information System
RCC	Remote Communication Cabinet
RCCV	Reinforced Concrete Containment Vessel
RCCWS	Reactor Component Cooling Water System
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RDA	Rod Drop Accident
RDC	Resolver-to-Digital Converter

Global Abbreviations And Acronyms List

Term	Definition
REPAVS	Refueling and Pool Area Ventilation Subsystem of Fuel Building HVAC (Heating, Ventilation and Air Conditioning)
RFP	Reactor Feed Pump
RG	Regulatory Guide
RHR	Residual Heat Removal (function)
RHX	Regenerative Heat Exchanger
RMS	Root Mean Square
RMS	Radiation Monitoring Subsystem
RLP	Reference Loading Pattern
RMU	Remote Multiplexer Unit
RO	Reverse Osmosis
ROM	Read-only Memory
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRPS	Reference Rod Pull Sequence
RSM	Rod Server Module
RSPC	Rod Server Processing Channel
RSS	Remote Shutdown System
RSSM	Reed Switch Sensor Module
RSW	Reactor Shield Wall
RTD	Resistance Temperature Detector
RTIF	Reactor Trip and Isolation Function(s)
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RTP	Reactor Thermal Power
RW	Radwaste Building
RWBCR	Radwaste Building Control Room
RWBGA	Radwaste Building General Area
RWBHVAC	Radwaste Building HVAC (Heating, Ventilation and Air Conditioning)
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
SA	Severe Accident
SAM	Severe Accident Management
SAR	Safety Analysis Report
SB	Service Building
SBA	Small Break Accident
S/C	Digital Gamma-Sensitive GM (Geiger-Mueller Counter) Detector
SC	Suppression Chamber
S/D	Scintillation Detector

Global Abbreviations And Acronyms List

Term	Definition
S/DRSRO	Single/Dual Rod Sequence Restriction Override
S/N	Signal-to-Noise
S/P	Suppression Pool
SAS	Service Air System
SB&PC	Steam Bypass and Pressure Control System
SBO	Station Blackout
SBWR	Simplified Boiling Water Reactor
SCEW	System Component Evaluation Work
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling
SDM	Shutdown Margin
SDS	System Design Specification
SEOA	Sealed Emergency Operating Area
SER	Safety Evaluation Report
SF	Service Water Building
SFA	Spent Fuel Assembly
SFP	Spent fuel pool
SIL	Service Information Letter
SIT	Structural Integrity Test
SIU	Signal Interface Unit
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SM	
SMU	SSLC (Safety System Logic and Control) Multiplexing Unit
SOV	Solenoid Operated Valve
SP	Setpoint
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SPTMS	Suppression Pool Temperature Monitoring Subsystem of Containment Monitoring System
SR	Surveillance Requirement
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRS	Software Requirements Specification
SRSRO	Single Rod Sequence Restriction Override
SRSS	Square Root Sum of Squares

Global Abbreviations And Acronyms List

Term	**Definition**
SRV	Safety Relief Valve
SRVDL	Safety Relief Valve Discharge Line
SSAR	Standard Safety Analysis Report
SS	Sub-scale
SST	Sub-scale Test
SSC(s)	Structure, System and Component(s)
SSE	Safe Shutdown Earthquake
SSI	Soil Structure Interaction
SSLC	Safety System Logic and Control
SSPC	Steel Structures Painting Council
ST	Spare Transformer
STI	Startup Test Instruction
STP	Sewage Treatment Plant
STRAP	Scram Time Recording and Analysis Panel
STRP	Scram Time Recording Panel
SV	Safety Valve
SWH	Static Water Head
SWMS	Solid Waste Management System
SY	Switch Yard
TAF	Top of Active Fuel
TASS	Turbine Auxiliary Steam System
TB	Turbine Building
TBCE	Turbine Building Compartment Exhaust
TBAS	Turbine Building Air Supply
TBE	Turbine Building Exhaust
TBLOE	Turbine Building Lube Oil Area Exhaust
TBS	Turbine Bypass System
TBHV	Turbine Building HVAC (Heating, Ventilation and Air Conditioning)
TBV	Turbine Bypass Valve
TC	Training Center
TCCWS	Turbine Component Cooling Water System
TCS	Turbine Control System
TCV	Turbine Control Valve
TDH	Total Developed Head
TEDE	Total Effective Dose Equivalent
TEMA	Tubular Exchanger Manufacturers' Association
TFSP	Turbine First Stage Pressure
TG	Turbine Generator
TGSS	Turbine Gland Seal System

Global Abbreviations And Acronyms List

Term	**Definition**
THA	Time-History Accelerograph
TIP	Traversing In-core Probe
TLOS	Turbine Lubricating Oil System
TLU	Trip Logic Unit
TMI	Three Mile Island
TMSS	Turbine Main Steam System
TRAC	Transient Reactor Analysis Code
TRM	Technical Requirements Manual
TS	Technical Specification(s)
TSC	Technical Support Center
TSI	Turbine Supervisory Instrument
TSV	Turbine Stop Valve
TTWFATBV	Turbine trip with failure of all bypass valves
UBC	Uniform Building Code
UHS	Ultimate Heat Sink
UL	Underwriter's Laboratories Inc.
UPS	Uninterruptible Power Supply
URD	Utilities Requirements Document
USE	Upper Shelf Energy
USM	Uniform Support Motion
USMA	Uniform Support Motion Response Spectrum Analysis
USNRC	United States Nuclear Regulatory Commission
USS	United States Standard
UV	Ultraviolet
V&V	Verification and Validation
Vac / VAC	Volts Alternating Current
Vdc / VDC	Volts Direct Current
VDU	Video Display Unit
VW	Vent Wall
VWO	Valves Wide Open
WD	Wash Down Bays
WH	Warehouse
WS	Water Storage
WT	Water Treatment
WW	Wetwell
XMFR	Transformer
ZPA	Zero Period Acceleration

1. INTRODUCTION

This document provides the Tier 1 material of the ESBWR Design Control Document (DCD).

Tier 1 is the portion of the design-related information contained in the ESBWR DCD that is to be NRC approved and certified. The Tier 1 design descriptions, interface requirements, and site parameters are derived from Tier 2 information. Tier 1 information includes:

(1) Definitions and general provisions;

(2) Design descriptions;

(3) Inspections, tests, analyses, and acceptance criteria (ITAAC);

(4) Significant site parameters; and

(5) Significant interface requirements.

Tier 2 means the portion of the design-related information contained in a generic DCD that is approved but not certified. Compliance with Tier 2 is required, but generic changes to, and plant-specific departures from, Tier 2 are governed a change process defined in the design certifcation rule (typically Section VIII) that become an appendix t Part 52. A Tier 2 change that does not require a Tier 1 or Technical Specifications change may be implemented without prior NRC approval, if the change would be allowable per a 10 CFR 50.59-like evaluation process, also specified in the design certification rule. Compliance with Tier 2 provides a sufficient, but not the only acceptable, method for complying with Tier 1.

1.1 TIER 1 LEVEL OF DETAIL AND SCOPE

The information in Tier 1 cannot be changed without NRC rulemaking. As a result, the Tier 1 topics and their level of detail should ideally be limited to information from Tier 2 that is unlikely to be changed. However, in many cases, descriptions must be expanded for better understanding. Within the scope of this DCD, the structures and systems important to safety (defined in Subsection 1.2.1) and their functions with respect to nuclear safety are described within this DCD Tier 1. For completeness, the major structures and systems that are not important to safety or are within a Combined Operating License (COL) applicant scope are included, but only by name.

1.1.1 General Plant Description

The following summarizes the ESBWR Standard Plant principal design features and criteria.

Standard Plant Scope

The ESBWR Standard Plant includes buildings dedicated exclusively or primarily to housing systems and equipment related to the nuclear system or controlled access to these systems and equipment. Six such main buildings (see Figure 1.1-1) are within the scope for the ESBWR. These are:

(1) Reactor Building – houses safety-related structures, systems and components (SSC), except for the main control room, safety-related Distributed Control and Information System equipment rooms in the Control Building and spent fuel storage pool and

associated auxiliary equipment in the Fuel Building. The Reactor Building includes the reactor, containment, refueling area and auxiliary equipment.

(2) Control Building – houses the main control room and safety-related controls outside the reactor building.

(3) Fuel Building – houses the spent fuel storage pool and its associated auxiliary equipment.

(4) Turbine Building – houses equipment associated with the main turbine and generator, and their auxiliary systems and equipment, including the condensate purification system and the process offgas treatment system.

(5) Radwaste Building – houses equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

(6) Electrical Building – houses the two nonsafety-related standby diesel generators and their associated auxiliary equipment.

Buildings and structures not in the ESBWR Standard Plant scope include the main transformer; switchyard; heat sinks for the main condenser, decay heat, and system waste heat; sewage and water treatment building; and storage tanks for fuel oil, nitrogen and demineralized water. These building and structures are site-specific.

Number of Plant Units

For the purpose of this design certification, a single unit standard plant is described. All changes with regard to a multiple unit plant are COL scope.

Type of Nuclear Steam Supply

This plant will have a boiling water reactor (BWR) nuclear steam supply system (NSSS), designed by GE, and designated as the ESBWR.

Type of Containment

This plant will have a containment vessel comprised of a drywell and wetwell. The containment structure is a reinforced right circular cylindrical concrete vessel integrated with the Reactor Building.

Core Thermal Power Levels and Principal Design Parameters

The plant uses a single-cycle, natural circulation BWR with the design parameters shown in Table 1.1-1.

Table 1.1-1

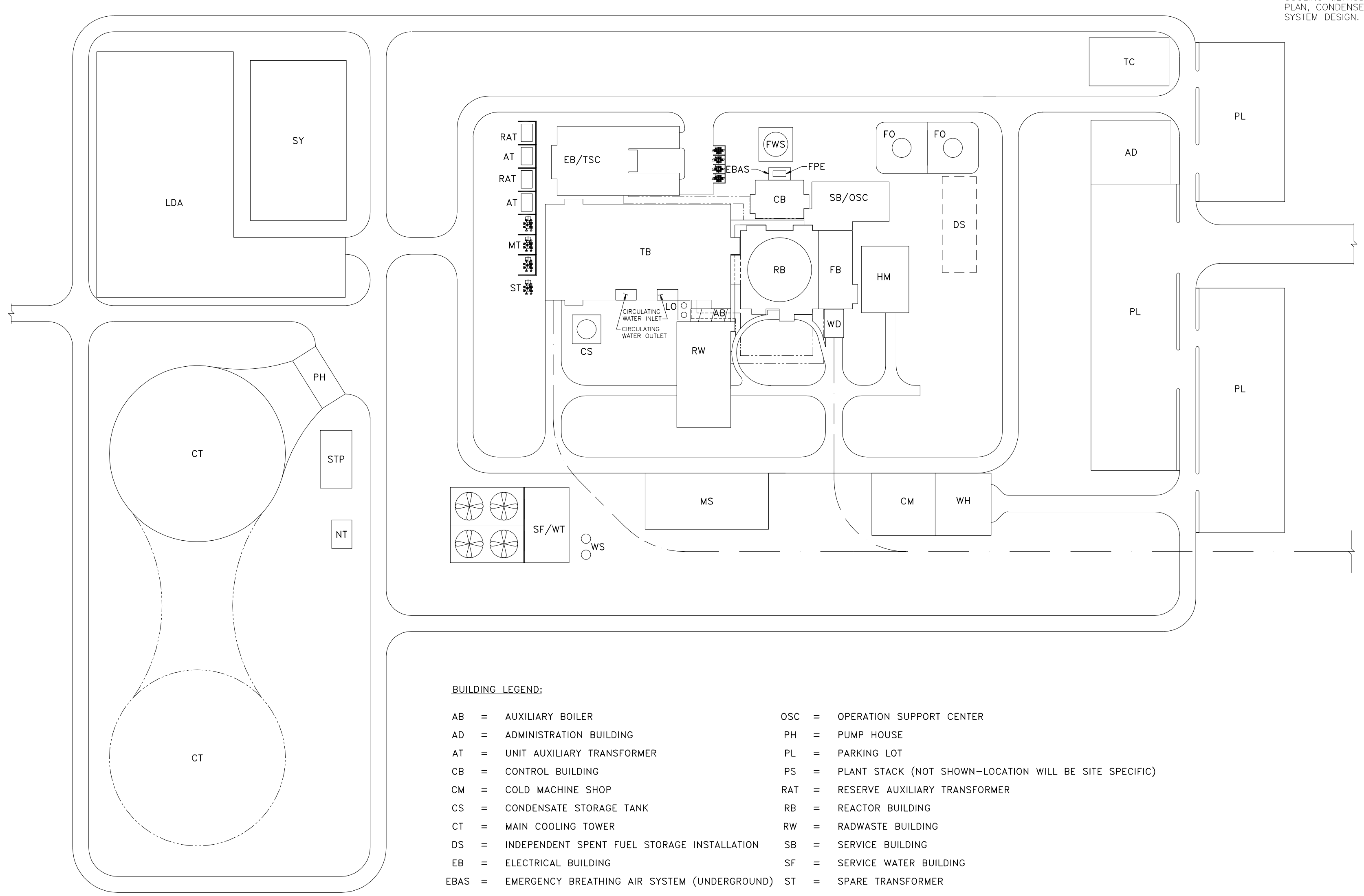
Principal Design Parameters

Parameter	Value
Rated thermal power	4500 MWt
Power-dependent safety analysis design thermal power	102% of rated
Reactor dome pressure at rated thermal power	7.17 MPa absolute (1040 psia)
Feedwater temperature at rated thermal power	210 to 220°C (410 to 428°F)
Steam flow rate at rated thermal power with 215.6°C (420.0°F) feedwater temperature	2432 kg/s (1.93E7 lbm/hr)
Nominal core coolant flow rate at rated thermal power *	10000 kg/s (7.94E7 lbm/hr)
Reactor Pressure Vessel (RPV) and Reactor Coolant Pressure Boundary (RCPB) design pressure	8.62 MPa gauge (1250 psig)
RPV and RCPB design temperature	302°C (575°F)
Containment internal design pressure	310 kPa gauge (45 psig)
Containment design temperature	171°C (340°F)
Containment vessel leak rate (excluding MSLs)	0.5%/day
Number of fuel assemblies (bundles)	1132
Number of control rods	269
Approximate net electrical power output *	1535 ± 50 MWe

* Nominal values, only presented for information.

NOTES:

1. THIS PLOT PLAN REPRESENTS THE STANDARD ESBWR CONFIGURATION. THIS CONFIGURATION WILL BE MODIFIED FOR SITE SPECIFIC REQUIREMENTS DURING COMBINED OPERATING LICENSE APPLICATIONS.
2. THE REFERENCE NORMAL HEAT SINK IS SHOWN AS NATURAL DRAFT COOLING TOWERS; HOWEVER, SITE SPECIFIC AVAILABLE WET BULB AND COOLING WATER TEMPERATURES, ENVIRONMENTAL LIMITATIONS AND SPECIFIC TURBINE CONFIGURATION MAY DICTATE EITHER ONCE THROUGH OR MECHANICAL DRAFT TOWER COOLING. THESE SITE SPECIFIC ALTERNATE COOLING METHODS MAY ALSO CHANGE THE PLOT PLAN, CONDENSER AND CIRCULATING WATER SYSTEM DESIGN.



BUILDING LEGEND:

AB	=	AUXILIARY BOILER
AD	=	ADMINISTRATION BUILDING
AT	=	UNIT AUXILIARY TRANSFORMER
CB	=	CONTROL BUILDING
CM	=	COLD MACHINE SHOP
CS	=	CONDENSATE STORAGE TANK
CT	=	MAIN COOLING TOWER
DS	=	INDEPENDENT SPENT FUEL STORAGE INSTALLATION
EB	=	ELECTRICAL BUILDING
EBAS	=	EMERGENCY BREATHING AIR SYSTEM (UNDERGROUND)
FB	=	FUEL BUILDING
FO	=	DIESEL FUEL OIL STORAGE TANK
FPE	=	FIRE PUMP ENCLOSURE
FWS	=	FIRE WATER STORAGE TANK
HM	=	HOT MACHINE SHOP & STORAGE
LDA	=	LAY DOWN AREA
LO	=	DIRTY/CLEAN LUBE OIL STORAGE TANK
MS	=	MISCELLANEOUS SERVICE AREA
MT	=	MAIN TRANSFORMER
NT	=	NITROGEN STORAGE TANK
OSC	=	OPERATION SUPPORT CENTER
PH	=	PUMP HOUSE
PL	=	PARKING LOT
PS	=	PLANT STACK (NOT SHOWN–LOCATION WILL BE SITE SPECIFIC)
RAT	=	RESERVE AUXILIARY TRANSFORMER
RB	=	REACTOR BUILDING
RW	=	RADWASTE BUILDING
SB	=	SERVICE BUILDING
SF	=	SERVICE WATER BUILDING
ST	=	SPARE TRANSFORMER
STP	=	SEWAGE TREATMENT PLANT
SY	=	SWITCH YARD
TB	=	TURBINE BUILDING
TC	=	TRAINING CENTER
TSC	=	TECHNICAL SUPPORT CENTER
WD	=	WASH DOWN BAYS (EQUIPMENT ENTRY)
WH	=	WAREHOUSE
WS	=	WATER STORAGE
WT	=	WATER TREATMENT

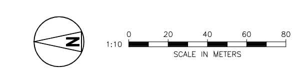


Figure 1.1-1. ESBWR Standard Plant General Site Plan

1.2 DEFINITIONS AND GENERAL PROVISIONS

1.2.1 Definitions

The following definitions apply to terms used in the Design Descriptions and associated Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC):

Acceptance Criteria means the performance, physical condition, or analysis results for a structure, system, or component that demonstrates a design commitment is met.

An **accident** is defined as a postulated design basis event that is not expected to occur during the lifetime of a plant, which equates to either an ASME Code Service Level C or D incident, and results in radioactive material releases with calculated doses comparable to (but not to exceed) the 10 CFR 50.34(a) exposures.

Analysis means the calculation, mathematical computation, or engineering or technical evaluation. Engineering or technical evaluations could include, but are not limited to, comparisons with operating experience or design of similar structures, systems, or components.

Anticipated operational occurrences (AOOs), from 10 CFR 50 App. A, "*mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power*." For the ESBWR, an *AOO* is defined as any abnormal event that has a probability of occurrence of ≥ 1/100 per year.

As-built means the physical properties of the structure, system or component, following the completion of its installation or construction activities at its final location at the plant site. In this context as-built can mean confirmation that the installed system conforms with the design within the allowed tolerances.

Basic Configuration (for a Building) means the arrangement of the building features (e.g., floors, ceilings, walls, basemat and doorways) and of the structures, systems, or components within, as specified in the building Design Description.

Basic Configuration (for a System) means the functional arrangement of structures, systems, and components specified in the Design Description, as specified in Section 1.2.

Cold shutdown means a *safe shutdown* with the average reactor coolant temperature ≤ 93.3°C (200°F).

Containment means the Primary Containment System, unless explicitly stated otherwise.

Design basis accident: Section B of RG 1.183 states "The design basis accidents (DBAs) were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the facility's engineered safety features." Therefore, a *design basis accident* is an accident postulated and analyzed to confirm the adequacy of a plant engineered safety feature.

Design basis events: Per 10 CFR 50.49(b), *design basis events* are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external

events, and natural phenomena for which the plant must be designed to ensure the safety-related functions.

Design Commitment means that portion of the Design Description that is verified by ITAAC.

Design Description means that portion of the design that is certified.

Division refers to safety-related electrical and/or instrumentation and control (I&C) equipment connected to a common electrical power source. ("Division" does not apply to mechanical equipment or any nonsafety-related electrical and/or I&C equipment.)

Engineered safety feature: Based on Regulatory Guide (RG) 1.70, an engineered safety feature (ESF) directly mitigates the consequence of a postulated accident. Consistent with RG 1.70, NUREG-0800, Standard Review Plan 6.1.1, subsection I states "Engineered safety features (ESF) are provided in nuclear plants to mitigate the consequences of design basis or loss-of-coolant accidents."

Hot shutdown means a *safe shutdown* with the average reactor coolant temperature > 215.6°C (420°F).

Hot standby means a subcritical or critical condition (1) with thermal power (including decay heat) ≤ 5% of rated, (2) in which reactor temperatures and pressures are near normal operating conditions, and (3) from which normal power operation can readily be achieved.

Important to safety: As defined in Appendix A of 10 CFR 50, structures, systems and components *important to safety* are those items that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. For the ESBWR, equipment/functions/conditions *important to safety* means:

(1) Safety-related SSCs (including supporting auxiliaries) as defined in 10 CFR 50.2 and their associated safety-related functions;

(2) Equipment/function(s) assumed or used to mitigate the AOOs evaluated in DCD Tier 2;

(3) Equipment/function(s) assumed or used to prevent or mitigate the special events (e.g., ATWS and Station Blackout), as described in DCD Tier 2;

(4) Equipment/function(s) whose failure or malfunction could lead to an accident, or impair the ability of other equipment to perform a safety-related function;

(5) Equipment/function(s) requiring (for ensuring nuclear safety) elevated quality assurance or design requirements (i.e., special treatment), but not to full safety-related standards;

(6) Nonsafety-related readiness functions and their associated plant condition(s) assumed, prior to the initiation of an accident, in any accident safety analysis described in DCD Tier 2;

(7) As described in DCD Tier 2, nonsafety-related SSCs used to control the release of radioactive wastes; and

(8) As defined in DCD Tier 2, the nonsafety-related equipment and their associated supporting auxiliary system(s) that are essential in performing Regulatory Treatment of Non-Safety Systems (RTNSS) functions.

Inspect or Inspection means visual observations, physical examinations, or review of records based on visual observation or physical examination that compare the structure, system, or

component condition to one or more Design Commitments. Examples include walk-downs, configuration checks, measurements of dimensions, and non-destructive examinations.

Loop means a train that forms a closed loop.

Safe shutdown (generic definition from ANSI/ANS-58.14-1993) is a shutdown with:

(1) The reactivity of the reactor kept to a margin below criticality consistent with Technical Specifications;

(2) The core decay heat being removed at a controlled rate sufficient to prevent core or reactor coolant system thermal design limits from being exceeded;

(3) Components and systems necessary to maintain these conditions operating within their design limits; and

(4) Components and systems, necessary to keep doses within prescribed limits, operating properly.

Safe shutdown (non-design basis accident (non-DBA)) **for station blackout** (from 10 CFR 50.2) means bringing the plant to those shutdown conditions specified in plant Technical Specifications as Hot Standby or Hot Shutdown, as appropriate (plants have the option of maintaining the RCS at normal operating temperatures or at reduced temperatures).

Stable shutdown (*safe stable condition* from SECY-94-084) means a *safe shutdown* with the average reactor coolant temperature ≤ 215.6°C (420°F).

Special events * are not included as *design basis events* defined by 10 CFR 50.49, and

i. Are postulated in the 10 CFR regulations to demonstrate some specified prevention, coping or mitigation capabilities, without specifically requiring a radiological evaluation, and/or

ii. Include a common mode equipment failure or additional failure(s) beyond the SFC.

* *Special events* do not include severe accidents and other events that are only evaluated as part of the plant PRA.

Test means the actuation, operation, or establishment of specified conditions, to evaluate the performance or integrity of as-built structures systems, or components, unless explicitly stated otherwise.

Train means a redundant, identical mechanical function within a system. (It is the mechanical equivalent of an electrical division.)

Type Test means a test on one or more sample components of the same type and manufacturer to qualify other components of that same type and manufacturer. A type test is not necessarily a test of the as-built structures, systems, or components.

1.2.2 General Provisions

The following general provisions are applicable to the design descriptions and associated ITAAC.

1.2.2.1 Verifications for Basic Configuration for Systems

Verifications for basic configuration of systems include and are limited to inspection of the system functional arrangement and the following inspections, tests, and analyses:

(1) Inspections, including non-destructive examination (NDE), of the as-built, pressure boundary welds for ASME Code Class 1, 2 or 3 components identified in the Design Description to demonstrate that the requirements of ASME Code Section III for the quality of pressure boundary welds are met.

(2) Type tests, analyses, or a combination of type tests and analyses of the Seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) identified in the Design Description to demonstrate that the as-built equipment, including associated anchorage, is qualified to withstand design basis dynamic loads without loss of its safety-related function.

(3) Type tests, or type tests and analyses, of the Class 1E electrical equipment identified in the Design Description (or on accompanying figures) to demonstrate that it is qualified to withstand the environmental conditions that would exist during and following a design basis accident without loss of its safety-related function for the time needed to be functional. These environmental conditions, as applicable to the bounding design basis accident(s), are as follows: expected time-dependent temperature and pressure profiles, humidity, chemical effects, radiation, aging, submergence, and their synergistic effects which have a significant effect on equipment performance. As used in this paragraph, the term “Class 1E electrical equipment” constitutes the equipment itself, connected instrumentation and controls, connected electrical components (such as cabling, wiring, and terminations), and the lubricants necessary to support performance of the safety-related functions of the Class 1E electrical components identified in the Design Description, to the extent such equipment is not located in a mild environment during or following a design basis accident.

    Electrical equipment environmental qualification shall be demonstrated through analysis of the environmental conditions that would exist in the location of the equipment during and following a design basis accident and through a determination that the equipment is qualified to withstand those conditions for the time needed to be functional. This determination may be demonstrated by:

    a. Type testing of an identical item of equipment under identical or similar conditions with a supporting analysis to show that the equipment is qualified; or

    b. Type testing of a similar item of equipment under identical or similar conditions with a supporting analysis to show that the equipment is qualified; or

    c. Experience with identical or similar equipment under identical or similar conditions with supporting analysis to show that the equipment is qualified; or

d. Analysis in combination with partial type test data that supports the analytical assumptions and conclusions to show that the equipment is qualified.

(4) Tests or type tests of active safety-related motor-operated valves (MOVs) identified in the Design Description to demonstrate that the MOVs are qualified to perform their safety-related functions under design basis differential pressure, system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and/or maximum stroke times.

1.2.2.2 Treatment of Individual Items

The absence of any discussion or depiction of an item in the Design Description or accompanying figures shall not be construed as prohibiting a licensee from utilizing such an item, unless it would prevent an item from performing its safety functions as discussed or depicted in the Design Description or accompanying figures.

When the term "operate," "operates," or "operation" is used with respect to an item discussed in the Acceptance Criteria, it refers to the actuation and running of the item. When the term "exist", "exists", or "existence" is used with respect to an item discussed in the Acceptance Criteria, it means that the item is present and meets the Design Description.

1.2.2.3 Implementation of ITAAC

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) are provided in tables with the following three-column format:

Design Commitment **Inspections, Tests, Analyses** **Acceptance Criteria**

Each Design Commitment in the left-hand column of the ITAAC tables has an associated requirement for Inspections, Tests or Analyses (ITA) specified in the middle column of the tables. The identification of a separate ITA entry for each Design Commitment shall not be construed to require that separate inspections, tests, or analyses must be performed for each Design Commitment. Instead, the activities associated with more than one ITA entry may be combined, and a single inspection, test, or analysis may be sufficient to implement more than one ITA entry.

An ITA may be performed by the licensee of the plant, or by its authorized vendors, contractors, or consultants. Furthermore, an ITA may be performed by more than a single individual or group, may be implemented through discrete activities separated by time, and may be performed at any time prior to fuel load (including before issuance of the Combined Operating License for those ITAAC that do not necessarily pertain to as-installed equipment). Additionally, ITA may be performed as part of the activities that are required to be performed under 10 CFR Part 50 (including, for example, the Quality Assurance (QA) program required under Appendix B to Part 50). Therefore, an ITA need not be performed as a separate or discrete activity.

1.2.2.4 Discussion of Matters Related to Operations

In some cases, the Design Descriptions in this document refer to matters that relate to operation, such as normal valve or breaker alignment during normal operation modes. Such discussions are provided solely to place the Design Description provisions in context (e.g., to explain automatic features for opening or closing valves or breakers upon off-normal conditions). Such

discussions shall not be construed as requiring operators during operation to take any particular action (e.g., to maintain valves or breakers in a particular position during normal operation).

1.2.2.5 Interpretation of Figures

In many but not all cases, the Design Descriptions in Section 2 include one or more figures, which may represent a functional diagram, general structural representation, or other general illustration. For I&C systems, the figures also represent aspects of the relevant logic of the system or part of the system. Unless specified explicitly, these figures are not indicative of the scale, location, dimensions, shape, or spatial relationships of as-built structures, systems, or components. In particular, the as-built attributes of structures, systems, and components may vary from the attributes depicted on these figures, provided that those safety functions discussed in the Design Description pertaining to the figure are not adversely affected.

1.2.2.6 Rated Reactor Core Thermal Power

The rated reactor core thermal power for the ESBWR is provided in Table 1.1-1.

2. DESIGN DESCRIPTIONS AND ITAAC

This section provides the certified design material for each of the ESBWR systems that is either fully or partially within the scope of the Certified Design.

Note: Values with “[]” are estimates and are subject to change during final design.

2.1 NUCLEAR STEAM SUPPLY

The following subsections describe the major Nuclear Steam Supply Systems (NSSS) and the natural circulation process for the ESBWR.

2.1.1 Reactor Pressure Vessel System

Design Description

The reactor pressure vessel (RPV) assembly consists of the pressure vessel and its appurtenances, supports and insulation, and the reactor internals enclosed by the vessel (excluding the core, in-core nuclear instrumentation, neutron sources, control rods, and control rod drives).

The reactor coolant pressure boundary (RCPB) of the RPV retains integrity as a radioactive material barrier during normal operation and following anticipated operational occurrences. The RPV retains integrity to contain coolant during design basis accidents (DBAs).

Certain RPV internals support the core and instrumentation used during a DBA. Other RPV internals direct coolant flow, separate steam from the steam/water mixture leaving the core, hold material surveillance specimens, and support instrumentation used for normal operation.

The RPV, together with its internals, provides guidance and support for the fine-motion control rod drives (FMCRDs). Reactor internals associated with the SLC system are used to distribute sodium pentaborate solution, when necessary, to achieve core subcriticality via means other than inserting control rods.

The RPV restrains the FMCRDs to prevent ejection of a control rod connected with a drive in the event of a postulated failure of a CRD housing.

Reactor Pressure Vessel

The RPV consists of a vertical, cylindrical pressure vessel of welded construction, with a removable top head, and head flanges, seals and bolting. The vessel also includes penetrations, nozzles, shroud support, and venturi shaped flow restrictors in the steam outlet nozzles. The shroud support carries the weight of peripheral fuel assemblies, neutron sources, core plate, top guide, shroud, chimney and chimney head with steam separators, and it laterally supports the fuel assemblies. Sliding block type supports near the bottom of the vessel support and anchor the vessel on the RPV support structure in the containment.

The RPV dimensions are shown in Table 1.1-1, and its key features are shown in Figures 2.1.1-1 and 2.1.1-2.

The overall RPV height permits natural circulation driving forces to produce abundant core coolant flow. An increased flow-path length relative to most prior BWRs is provided by a long “chimney” in the space, which extends from the top of the core to the entrance to the steam

separator assembly. This chimney feature existed in Humboldt Bay and Dodewaard natural circulation BWRs. The chimney and steam separator assembly are supported by a shroud assembly, which extends to the top of the core. The large RPV volume provides a large reserve of water above the core, which translates directly into a much longer period of time, compared to prior BWRs, before core uncovery can occur as a result of feedwater flow interruption or a LOCA. This gives an extended period during which automatic systems or plant operators can reestablish reactor inventory control using any of several normal, nonsafety-related systems capable of injecting water into the reactor. Timely initiation of these systems precludes the need for activation of emergency safety equipment. The large RPV volume also reduces the reactor pressurization rates that develop which can eventually lead to actuation of the safety-relief valves, when the reactor is suddenly isolated from the normal heat sink.

The FMCRDs are mounted into permanently attached CRD housings. The CRD housings extend through, and are welded to CRD penetrations (stub tubes) that are welded into the RPV bottom head.

A flanged nozzle is provided in the top head for bolting on of the flange associated with the instrumentation for the initial vibration test of internals.

Sliding block type supports carry the vessel. The sliding supports are provided at a number of positions around the periphery of the vessel. One end of each sliding support is attached to a circumferential RPV flange and the other end is captured into sets of guide blocks that are anchored to the pedestal support brackets. Stabilizers help the upper portion of the RPV resist horizontal loads. Lateral support among the CRD housings and in-core housings are provided by restraints that, at the periphery, are supported by the CRD housing restraint beams.

The RPV insulation is supported from the shield wall surrounding the vessel. A steel frame, that is independent of the vessel and piping, supports insulation for the upper head and flange. Insulation access panels and insulation around penetrations are designed for ease of installation and removal for vessel inservice inspection and maintenance operations.

Access for examinations of the installed RPV is incorporated into the design of the vessel, reactor shield wall, and vessel insulation.

The RCPB portions of the RPV and appurtenances are classified as Quality Group A, Seismic Category I. The following ASME materials (or their equivalents) are used in the RPV pressure boundary: SA-533, Type B Class 1 (plate); SA-508, Grade 3, Class 1 (forging); SA-182 or SA-336, Type / Class F304/F304L/F316/F316L; Ni-Cr-Fe ASME Code Case N-580-1; and SA-540, Grade B23 or B24 (bolting).

A stainless steel weld overlay is applied to the interior of the RPV cylindrical shell and the main steam outlet and bottom head drain nozzles. The bottom head is clad with Ni-Cr-Fe alloy.

The materials of the low alloy steel plates and forgings used in construction of the RPV pressure boundary are melted using vacuum degassing and manufactured to fine grain practice and are supplied in the quenched and tempered condition.

Electroslag welding is not applied for the RPV pressure boundary welds. Preheat and interpass temperatures employed for welding of the RPV pressure boundary low alloy steel meet or exceed the values given in ASME Code Section III, Appendix D. Post-weld heat treatment at 593°C minimum is applied to these low-alloy steel welds.

Volumetric examination and surface examination are performed on all pressure-retaining welds as required by ASME Code Section III, Subsection NB-5320. In addition, all pressure-retaining welds are given a supplemental ultrasonic pre-service examination in accordance with ASME Code Section XI.

Fracture toughness properties of the RPV pressure boundary ferritic materials are measured and controlled in accordance with the requirements of ASME Code Section III Division 1. Transverse specimens are used to determine the required minimum upper shelf energy level of the core beltline materials. The minimum initial upper shelf energy level for base material and weld metal in the beltline region meets or exceeds 102 J. Separate, unirradiated baseline specimens are used to determine the transition temperature curve of the core beltline base material, heat affected zone and weld metal.

For the RPV material surveillance program, specimens are provided from a forging actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld material, and the weld HAZ material. The base metal and weld are heat treated in a manner, which simulates the actual heat treatment performed on the beltline region of the completed vessel. The specimen capsules contain the specimens and temperature monitors. The surveillance specimen holders having brackets welded to the vessel cladding in the core beltline region are provided to hold the specimen capsules and a neutron dosimeter.

Reactor Pressure Vessel Internals

The reactor pressure vessel internals consist of core support structures and other equipment.

The core support structures locate and support the fuel assemblies, form partitions within the reactor vessel to sustain pressure differentials across the partitions, and direct the flow of coolant water. The structures consist of a shroud, shroud support, core plate, top guide, orificed fuel supports and control rod guide tubes (CRGTs).

The other reactor internals consist of control rods, feedwater spargers, SLC system distribution headers, in-core guide tubes, surveillance specimen holders, chimney, chimney partitions, chimney head and steam separator assembly, and the steam dryer assembly.

The shroud and chimney make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow outside the core. This partition separates the core region from the downcomer annulus.

The core plate consists of a circular stainless steel plate with round openings and is stiffened with a beam structure. The core plate provides lateral support and guidance for the CRGTs, in-core flux monitor guide tubes, peripheral fuel supports and startup neutron sources. The core plate also supports the last two items vertically.

The top guide consists of a circular plate with square openings for fuel assemblies. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, less than four fuel assemblies. Holes are provided in the bottom surface of the top guide where the sides of the openings intersect, to anchor the in-core instrumentation detectors and start-up neutron sources.

The fuel assemblies are vertically supported in two ways depending upon whether they are located next to a control rod or not. The peripheral fuel assemblies, which are located at the outer edge of the active core, not adjacent to a control rod, are supported by the peripheral fuel

supports. The peripheral fuel supports are welded to the core plate and each support one assembly. The peripheral fuel supports contain flow-restricting sections to provide the appropriate coolant flow rate to the peripheral fuel assemblies. The remaining fuel assemblies, which are adjacent to the control rods, are supported by the orificed fuel supports and CRGTs. Each orificed fuel support and CRGT supports four fuel assemblies vertically upward and provides lateral support to the bottom of the fuel. The orificed fuel support is supported in the CRGT that is supported laterally by the core plate.

The control rod passes through a cruciform opening in the center of the orificed fuel support. Each guide tube is designed as a guide for the lower end of the control rod. The lower end of the CRGT is supported by the control rod drive (CRD) housing, which in turn transmits the weight of the orificed fuel support and CRGT, and the four fuel assemblies to the reactor vessel bottom head. The upper end of the CRD housing is welded to a stub tube that is directly welded to the bottom of the vessel. Coolant flow, which has entered the lower plenum of the vessel, travels upward, adjacent to the guide tubes and enters the orificed fuel supports just below the core plate. The orificed fuel supports contain four flow restricting openings that control coolant flow to the fuel assemblies.

The base of the CRGT is provided with a device for coupling to the FMCRD. The CRD is restrained from ejection, in the case of a stub tube to CRD housing weld failure, by the coupling of the drive with the guide tube base. In this event, the guide tube flange contacts the core plate and thus restrain the ejection. The coupling also prevents ejection if the CRD housing fails below the stub tube weld. In this event, the guide tube and fuel support remain supported by the CRD housing left intact above the stub tube weld.

The control rods are cruciform-shaped neutron absorbing members that can be inserted or withdrawn from the core by the FMCRD to control reactivity and reactor power.

Each of the feedwater lines is connected to a sparger via a RPV nozzle. The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. Each sparger, in two halves, with a tee connection at the middle, is fitted to the corresponding RPV feedwater nozzle. The sparger tee inlet is connected to the RPV nozzle safe end by a double thermal sleeve arrangement. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryers.

In-core guide tubes (ICGTs) protect the in-core flux monitoring instrumentation from flow of water in the bottom head plenum. The ICGTs extend from the top of the in-core housing to the top of the core plate. The local power range monitoring (LPRM) detectors for the Power Range Neutron Monitoring (PRNM) subsystem and the detectors for the Startup Range Neutron Monitoring (SRNM) subsystem are inserted through the guide tubes.

A latticework of clamps, tie bars, and spacers give lateral support and rigidity to the ICGTs.

Surveillance specimen capsules, which are held in capsule holders mentioned earlier, are located at a common elevation in the core beltline region. The capsule holders are nonsafety-related internals. Capsule holder brackets welded to the vessel cladding mechanically retain the capsule holders, which allow for capsule removal and re-installation.

As a natural circulation reactor, the ESBWR requires additional elevation head created by the density difference between the saturated water-steam mixture exiting the core and the subcooled water exiting the region just below the separators and the feedwater inlet. The chimney provides this elevation head or driving head necessary to sustain the natural circulation flow. The chimney is a long cylinder mounted to the top guide and which supports the steam separator assembly. The chimney forms the annulus separating the subcooled recirculation flow returning downward from the steam separators and feedwater, from the upward steam-water mixture flow exiting the core. Inside the chimney are partitions that separate groups of 16 fuel assemblies and thereby form smaller chimney sections limiting cross flow and flow instabilities.

The BWR direct cycle requires separation of steam from the steam-water mixture leaving the core. This is accomplished inside the RPV by passing the mixture sequentially first through an array of steam separators attached to a removable cover on the top of the chimney assembly, and then through standard BWR steam dryers. The steam dryer and the separator assembly are designed to provide outlet dry steam with a moisture content $\leq 0.1\%$

The core support structures are classified as ASME Code Class CS, Seismic Category I. The design, materials, manufacturing, fabrication, examination, and inspection used in the construction of the core support structures meet the requirements of ASME Code Section III, subsection NG, Core Support Structures.

These structures are code-stamped accordingly. Other reactor internals are designed per the guidelines of ASME Code NG-3000 and are constructed so as not to adversely affect the integrity of the core support structures as required by NG-1122.

Special controls on material fabrication processes are exercised when austenitic stainless steel is used for construction of RPV internals in order to avoid stress corrosion cracking during service.

Design and construction of the RPV internals ensure that the internals can withstand the effects of flow-induced vibration (FIV).

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.1-2 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Reactor Pressure Vessel System.

Table 2.1.1-1

Key Dimensions of RPV Components and Acceptable Variations

Description	**Dimension/ Elevation (Figure 2.1.1-1)**	**Nominal Value** (mm)	**Acceptable Variation(s)** (mm)
RPV bottom head inside invert elevation	A	0	Reference 0
Top of core plate elevation	B	4178	[±16]
Bottom of top guide elevation	C	7718	[±16]
RPV top head inside invert elevation	D	27560	[±100]
RPV inside diameter (inside cladding)	E	7112	[±51]
RPV wall thickness in beltline (including cladding)	F	182	[190.5 max]
RPV top and bottom head inside radius	G	4866	[±25]

Table 2.1.1-2
ITAAC For Reactor Pressure Vessel System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RPV system is as defined as Subsection 2.1.1, Table 2.1.1-1 and Figure 2.1.1-1.	1. Inspections of the as-built RPV System will be conducted.	1. The RPV system conforms with the Certified Design Specification and the basic configuration defined in Subsection 2.1.1, Table 2.1.1-1 and Figure 2.1.1-1.
2. The RPV pressure boundary defined in Subsection 2.1.1 is designed to meet the ASME Code Class 1 vessel requirements.	2. Inspections of the ASME Code required documents will be conducted.	2. An ASME Code Certified Stress Report exists for the RPV pressure boundary components.
3. The ASME Code components of the RPV system retain their pressure boundary integrity under internal pressure that will be experienced during service.	3. A hydrostatic test will be conducted on those code components of the RPV system required to be hydrostatically tested by the ASME Code.	3. The results of the hydrostatic test of the ASME Code components of the RPV system conform with the requirements in the ASME Code, Section III.
4. The materials selection and materials testing requirements for the RPV system are as defined in Subsection 2.1.1.	4. Inspections of the RPV system fabrication records will be conducted.	4. The RPV system conforms with the materials selection and materials testing requirements defined in Subsection 2.1.1.
5. The fabrication process and examination process requirements for the RPV system are as defined in Subsection 2.1.1.	5. Inspections of the RPV system fabrication records will be conducted.	5. The RPV system conforms with the fabrication process and examination process requirements defined in Subsection 2.1.1.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The material surveillance commitments for the reactor pressure vessel core beltline materials are as defined in Subsection 2.1.1.	6. Inspections of the as-built RPV system will be conducted for implementation of the material surveillance commitments.	6. The material surveillance program for the reactor pressure vessel core beltline materials conforms with the commitments defined in Subsection 2.1.1.
7. The RPV internals withstand the effects of FIV.	7. A vibration test program will be developed for the RPV internals of the lead ESBWR.	7. The required instrumentation for the vibration test program for the lead ESBWR has been installed on the necessary components of the RPV and Internals.

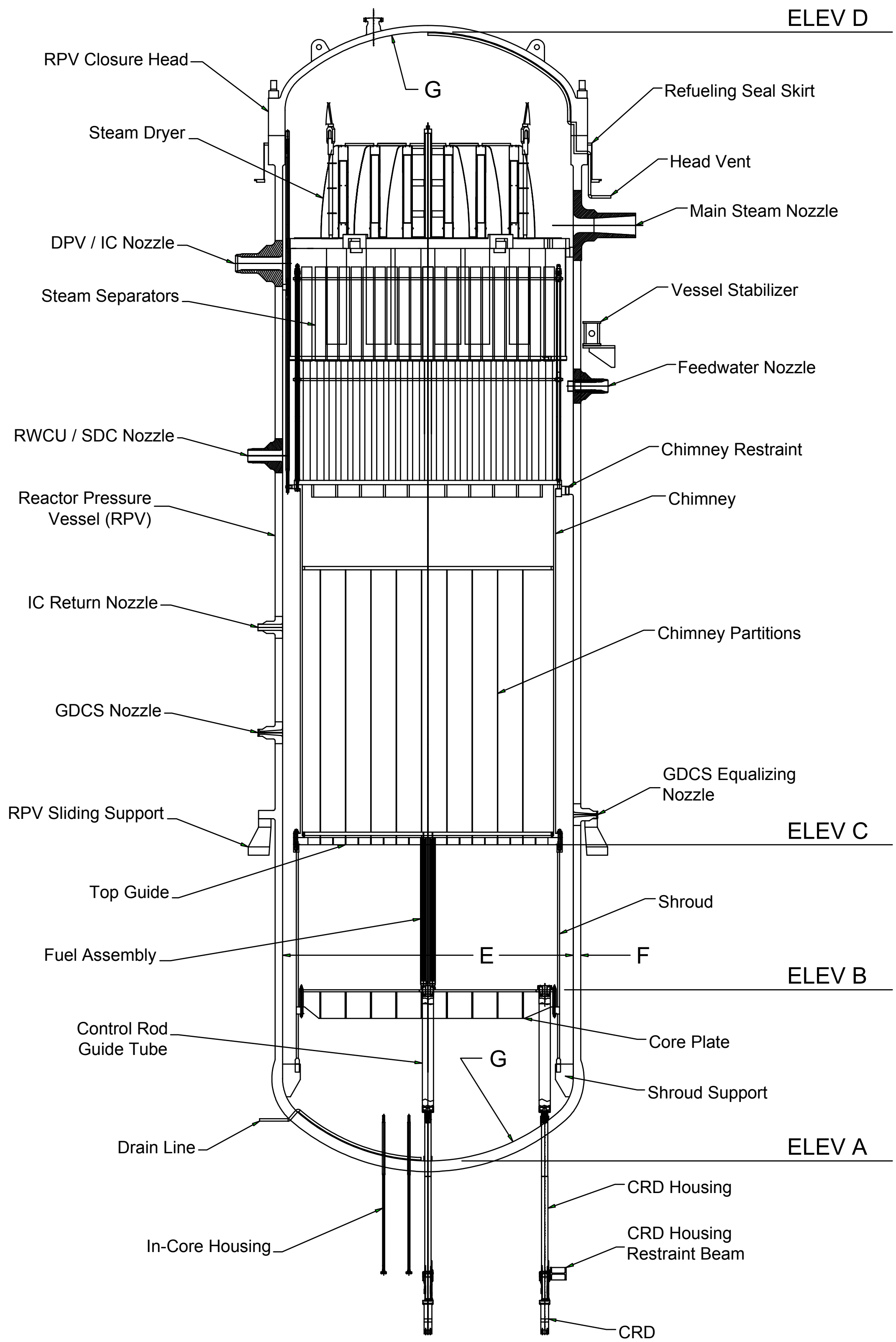
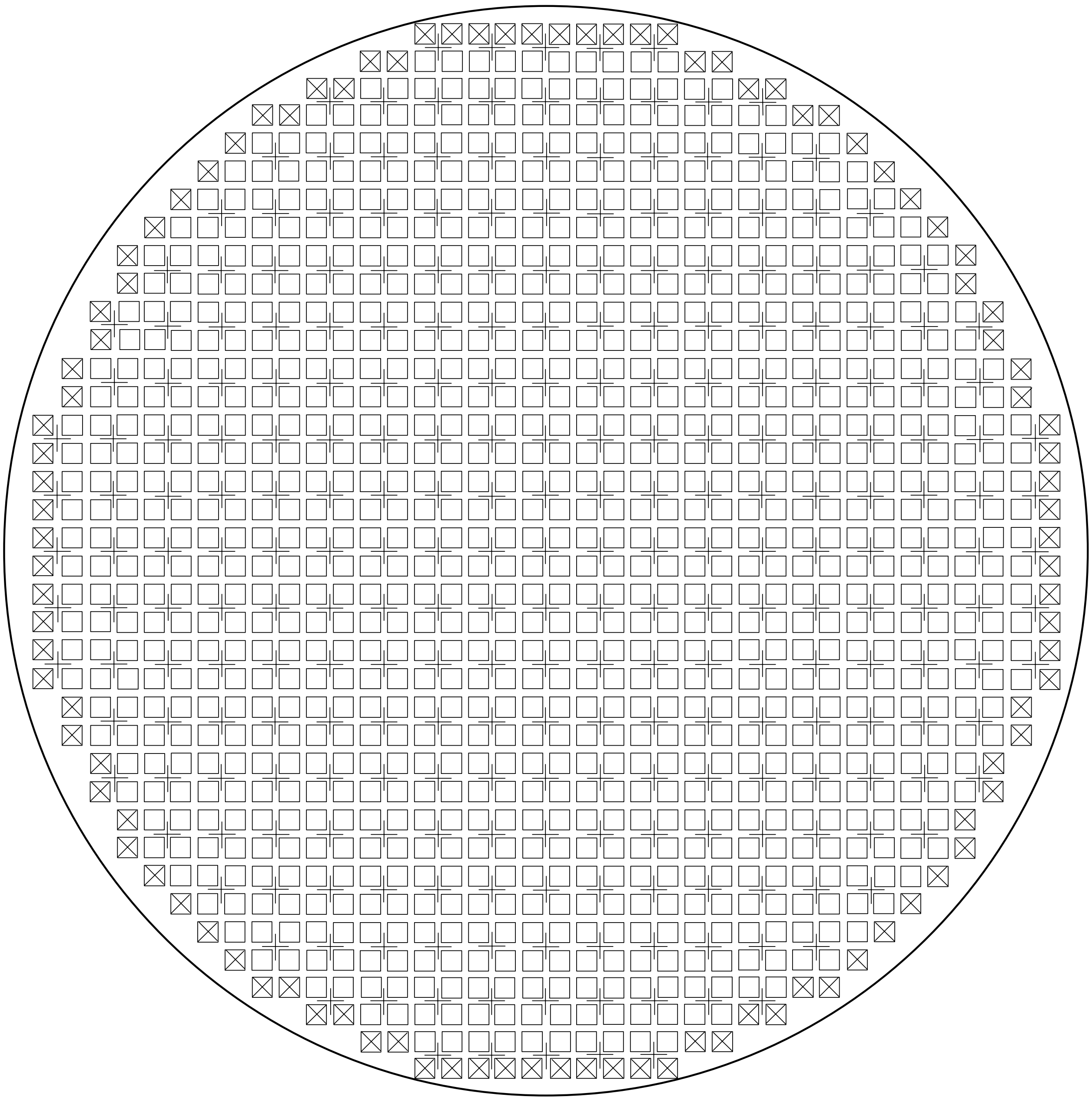


Figure 2.1.1-1. Reactor Pressure Vessel System Key Features



Central Region Bundle	1028	Control Rod	269
Peripheral Region Bundle	104		
Total	1132		

ESBWR Core Map

Figure 2.1.1-2. Reactor Core Arrangement

2.1.2 Nuclear Boiler System

Design Description

The NBS consists of Main Steam Lines (MSLs), Main Steam flow restrictors, a Steam Line Drain/Bypass Subsystem, Feedwater (FW) lines, Safety Relief Valves (SRVs), Main Steam Isolation Valves (MSIVs), Depressurization Valves (DPVs), an RPV head vent subsystem, and system instrumentation.

There are four Main Steam Lines (MSLs) that transport steam from the RPV to the main turbine. Each MSL contains two MSIVs in series and is connected to an outlet nozzle in the RPV. The inside of the main steam outlet nozzle, which is part of the RPV, has the shape of a venturi type flow limiter. A MSL flow restrictor limits the coolant blowdown rate from the RPV in the event a MSL break occurs downstream of the nozzle. The flow restrictors also contain instrument line taps used for detecting and monitoring steam flow.

The Main Steam Line Bypass/Drain subsystem drains condensate from the main steam lines to the main condenser during low power operation, startup, shutdown and when a steam line is isolated during operation.

The Condensate and Feedwater System supplies feedwater to the RPV at the required flow, pressure and temperature during startup, shutdown, at power levels up to and including rated load, and during abnormal events. There are two main FW lines inside the primary containment. The FW lines provide a path for return flow from the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system, Control Rod Drive (CRD) system and Fuel and Auxiliary Pools Cooling System (FAPCS) to the RPV.

The SRVs provide overpressure protection relief for the RPV. Ten of the SRVs, which operate in the overpressure safety mode and the Automatic Depressurization mode, transport steam from the main steam lines to quenchers located below the surface of the water in the Suppression Pool (SP). The remaining eight SRVs, which operate in the overpressure safety mode, are arranged into two groups of four valves. Each group is connected to a horizontal header that has a rupture disc at each end. Each header has a discharge line that terminates in a quencher in the SP. These valves either discharge directly into the drywell or through the discharge line to the SP. The SRVs, in conjunction with a reactor trip, assist in limiting peak pressure in the RPV during plant transients of a severity beyond those transients for which the ESBWR Isolation Condensers provide pressure-limiting action. Additionally, the ten ADS-SRVs enhance the depressurization rate following a LOCA.

Two vacuum breakers are connected in parallel on each SRV discharge pipeline that connects to a quencher in the SP. The vacuum breakers prevent drawing an excessive amount of water into the line as a result of steam condensation following termination of SRV operation.

There is an RPV head vent subsystem, which permits air to be released from the RPV to the Equipment and Floor Drain System so that the vessel can be filled with water for hydrostatic testing. The RPV head vent line is capable of being cross connected within the drywell to one of the main steamlines to permit venting non-condensable gases from the RPV during reactor operation.

The DPVs provide rapid depressurization of the RPV in the event of an accident so that an emergency source of water can be supplied to the RPV.

The NBS instrumentation consists of sensors to measure and monitor RPV pressure, temperature and water level. Additionally, there are sensors to measure and monitor steam line pressure, steam line flow, main condenser vacuum and RPV metal temperatures.

Safety Requirements:

The NBS shall perform the following safety-related functions:

- Provide containment isolation of the MSLs using MSIVs, to limit release of reactor coolant to the environment following an accident.
- Limit the reactor coolant release rate following a MSL break outside the containment.
- Prevent backflow in the feedwater lines and provide containment isolation using FW isolation valves.
- Maintain reactor coolant pressure boundary (RCPB).
- Provide overpressure protection for the RCPB in conjunction with the Reactor Protection System (RPS) scram function.
- Provide the capability of depressurizing the Reactor Pressure Vessel (RPV) automatically by the ADS (ADS-SRVs and DPVs) in the event of a Loss-Of-Coolant Accident (LOCA).
- Provide instrumentation to monitor the reactor coolant system pressure, RPV water level, MSIV position, SRV position, and DPV position during normal operations and accident conditions.

All NBS piping connected to the RPV up to and including the outboard containment isolation valves shall be ASME Section III Class 1and classified as:

- Safety Class 1
- Quality Group A
- Seismic Category I

The main steam piping beyond the second MSIV up to the main turbine stop valve shall be ASME Section III Class 2 and Quality Group B. The piping is Seismic Category I to the seismic restraint.

The feedwater lines between the testable check valve and the motor operated isolation valve shall be ASME Section III Class 2 and Quality Group B. Piping between the motor operated isolation valve and the seismic restraint shall be ANSI B31.1 and Quality Group D. Piping from the RPV to the seismic restraint upstream of the motor operated valve shall be Seismic Category I and piping upstream of the seismic restraint shall be non-seismic and Quality Group D.

The MSIVs, in conjunction with the flow restrictors built into the RPV nozzles, shall prevent excessive release of radioactivity to the environs under assumed condition of an MSL break outside the containment. In the worst postulated case, if the main steam line should rupture downstream of the outboard MSIV, steam flow quickly increases. The venturi type flow restrictor

shall prevent the steam flow from exceeding 200% of rated flow at normal reactor operating pressure. The flow restrictor shall have a maximum throat diameter of [355 mm] to meet the choke flow requirements.

The ten ADS-SRVs and DPVs together with instrumentation and control system logic constitute the Automatic Depressurization System (ADS) of the NBS.

The safety function of the SRVs shall limit the reactor pressure to less than 20% over the design pressure upon reactor isolation with a failure to scram. This is defined as an Anticipated Transient Without Scram (ATWS) event. Neutron flux and reactor pressure or water level signals are used to confirm this condition. Also, the SRVs open to provide overpressure protection of the reactor coolant pressure boundary in accordance with the ASME Code.

While in the Run mode, the ADS shall automatically initiate upon detection of low water level with or without the presence of high drywell pressure. The depressurization allows re-supply of water to the RPV at low pressure via the Gravity-Driven Cooling System (GDCS). The depressurization must be completed in time to allow GDCS injection flow to replenish core coolant in order to prevent core uncovery assuming a failure of any single active component.

The NBS shall be designed to meet the single failure criterion of 10 CFR 50, Appendix A.

NBS shall be designed to maintain all safety-related functional capability following a design basis LOCA and during a safe shutdown earthquake (SSE), which is postulated to occur simultaneously with a LOCA event for structural analyses. The NBS piping layout and support arrangement shall be designed to minimize jet impingement impact on the surrounding safety-related components.

Class 1E components in NBS shall be powered from their respective Class 1E division. In NBS, independence is provided between Class 1E divisions, and between Class 1E and non-Class 1E equipment.

The motor-operated valves (MOVs) and the pneumatic-operated valves that have active safety-related functions to open, close, or both open and close, shall be designed to maintain containment integrity by providing containment isolation functions under the most severe differential pressure, fluid flow, and temperature conditions they may experience.

The feedwater positive acting outboard isolation check valves shall perform a containment isolation safety-related function by closing to maintain containment integrity.

Instruments:

The NBS shall contain instrumentation to:

- Detect and monitor position of MSIVs, and provide open, closed and intermediate indication on display units in the main control room;
- Detect and monitor position of DPVs, and provide open and closed indication on display units in the main control room;
- Detect and monitor position of SRVs, and provide open and closed indication on display units in the main control room;
- Detect and monitor RPV pressure, temperature and water level, and provide indication on display units in the main control room;

- Detect and monitor main condenser vacuum, and provide indication on display units in the main control room;
- Detect and monitor differential pressure between the two feedwater lines, and provide indication on display units in the main control room; and
- Detect and monitor continuity circuit for each DPV squib device and provide indication on display units in the main control room.

Controls:

The following controls are available to the operator in the main control room:

- Manual control to enable the operator to open and close each MSIV;
- Manual control to enable the operator to initiate the ADS;
- Manual control to enable the operator to open and close each of the ADS-SRVs ; and
- Manual control to enable the operator to inhibit the automatic initiation of the ADS.

The NBS is shown in Figures 2.1.2-1 through 2.1.2-4.

The flow capacities of the SRVs and DPVs are provided in Table 2.1.2-1

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.2-2 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria, which will be undertaken for the NBS.

Table 2.1.2-1

SRV Capacities

Valves	**Number of Valves**	**ASME Rated Capacity at 103% Spring Set Pressure** [1] kg/s (Mlb/hr) each	**Used For ADS**
Non-ADS SRV	8	126 (1.000)	0
ADS-SRV	10	124 (0.984)	10

DPV Capacities

Valves	**Number of Valves**	**Capacity** [2] kg/s (Mlb/hr) each	**Used For ADS**
DPV	8	239 (1.897)	8

(1) Minimum capacity per the ASME Boiler and Pressure Vessel Code, Section III.

(2) Minimum capacity in ADS mode.

Table 2.1.2-2

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the NBS is defined in Subsection 2.1.2.	1. Inspections of the as-built system will be conducted.	1. The as-built NBS conforms with the basic configuration as defined in Subsection 2.1.2.
2. Portions of the NBS are classified as ASME Code class as indicated in Subsection 2.1.2. They are designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III.	2. ASME Code Data Reports will be reviewed and inspections of Code stamps will be conducted for ASME components in the NBS.	2. Those portions of the NBS identified as ASME Code Class in Subsection 2.1.2 have ASME Code Section III, Code Data Reports and Code stamps (or alternative markings permitted by the Code).
3. The throat diameter of each MSL flow limiter is less than or equal to 355 mm.	3. Inspections of the as-built MSL flow limiters will be taken.	3. The throat diameter of each MSL flow limiter is less than or equal to 355 mm.
4. Each MSL flow limiter has taps for two instrument lines. These instrument lines are used for monitoring the flow through each MSL.	4. Inspections will be conducted of the MSL instrument lines.	4. The MSL flow measurement instrument lines are installed.
5. The ASME Code portions of the NBS retain their integrity under internal pressures that will be experienced during service.	5. A hydrostatic test will be conducted on those Code components of the NBS required to be hydrostatically tested by the ASME Code.	5. The results of the hydrostatic test of the ASME Code components of the NBS conform with the requirements in the ASME Code, Section III.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The combined steamline volume from the RPV to the main steam turbine stop valves and steam bypass valves is greater than or equal to 135 m^3.	6. Calculations will be performed using the as-built dimensions of the steamlines to determine the combined steam line volume.	6. The combined steamline volume is greater than or equal to 135 m^3.
7. There are indications in the main control room for NBS parameters as defined in Subsection 2.1.2.	7. Inspections will be performed in the main control room of the NBS indications defined in Subsection 2.1.2.	7. The NBS indications defined in Subsection 2.1.2 are displayed in the main control room.
8. MSIV closing time is equal to or greater than 3 seconds and less than or equal to 5 seconds when N_2 or air is admitted into the valve pneumatic actuator. The MSIVs are capable of closing within 3 to 5 seconds under differential pressure, fluid flow and temperature conditions	8. Tests of the as-built MSIV will be conducted under preoperational test conditions. Tests or type tests, of an MSIV will be conducted under design basis differential pressure, flow and temperature conditions.	8. MSIV closing time is equal to or greater than [3 seconds] and less than or equal to [5 seconds].
9. When all MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters per minute at standard temperature of 20°C (68°F) and pressure (one atmosphere absolute pressure) with the differential pressure across the MSIV equal to or greater than 0.269 MPaD (40 psid).	9. Tests and analysis will be performed on the as-built MSIVs to determine the leakage.	9. When all MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters per minute at standard temperature of 20°C (68°F) and pressure (one atmosphere absolute pressure) with the differential pressure across the MSIV equal to or greater than 0.269 MPaD (40 psid).

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The SRV flow capacities are given in Table 2.1.2-1. The opening time for the SRVs from when the pressure exceeds the valve set pressure to when the valve is fully open shall be less than or equal to 1.7 second.	10. Analysis and tests (at a test facility) will be conducted in accordance with the ASME Code.	10. Test reports and analyses exist and conclude that the SRVs have the capacities shown in Table 2.1.2-1. The opening time for the SRVs from when the pressure exceeds the valve set pressure to when the valve is fully open is less than or equal to 1.7 second.
11. The SRVs and DPVs are provided with instrumentation that will provide indication (i.e. by direct measurement) of valve position.	11. Inspection will be performed on the SRV and DPV position indication instrumentation.	11. The SRV and DPV position indicators provide open and close indication.
12. Upon receipt of an ADS initiation signal, the ADS logic generates signals to the SRVs and the DPVs.	12. Tests will be conducted using simulated input signals for each NBS process variable to cause trip conditions in the instrument channels of the same process variable associated with each of the ADS logic divisions.	12. Upon receipt of an ADS initiation signal, the ADS logic generates signals to the SRVs and the DPVs.
13. The 10 SRV discharge lines associated with the ADS function are piped directly to quenchers located below the surface of the suppression pool.	13. Inspections will be performed to review the configuration of the SRV discharge line quenchers.	13. The 10 SRV discharge lines associated with the ADS function have been installed and are piped directly to the quenchers located below the surface of the suppression pool.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
14. When actuated by either of two initiators, the booster assembly opens the DPV in less than or equal to 0.45 seconds with an inlet pressure of 6.89 Mpa gauge (1000 psig.) or greater.	14. Tests will be performed on the booster assemblies during factory tests to confirm that they are capable of opening the valve. Tests and analyses will be performed to demonstrate that the booster opens the DPV.	14. Test reports and analyses exist and conclude that the DPV opens when actuated by the booster assembly in less than or equal to 0.45 seconds with an inlet pressure of 6.89 Mpa gauge (1000 psig.) or greater.
15. There are four DPVs attached to stub tubes off of the RPV and four DPVs attached to the main steam lines.	15. Inspections will be performed to review the configuration of the DPVs.	15. Four DPVs are attached to stub tubes off of the RPV and four DPVs are attached to the main steam lines.
16. The DPV minimum flow capacity is 239 kg/s (1.897 Mlb/hr).	16. Analyses and tests (at a test facility) will be performed.	16. Test reports and analyses exist and conclude that the DPV flow capacity is greater than or equal to 239 kg/s (1.897 Mlb/hr).
17. Vacuum breakers are provided on SRV discharge lines to reduce the post-discharge reflood height of water.	17. An inspection will be performed to confirm that the vacuum breakers are installed.	17. Vacuum breakers are installed on the SRV discharge lines. An analysis exists that demonstrates that the vacuum breaker capacity and setpoint limit the water column in the discharge line.
18. The MSIVs close upon any of the following conditions: (a) Low RPV water level, (b) Low turbine inlet pressure (RUN mode) and (c) Low main condenser vacuum.	18. Valve closure tests will be performed on the MSIVs using simulated signals.	18. The MSIVs close upon generation of any of the following simulated signals: (a) Low RPV water level, (b) Low turbine inlet pressure (RUN mode) and (c) Low main condenser vacuum.
19. The ADS has an Automatic Inhibit of the automatic ADS initiation.	19. A test of the ADS will be conducted with a simulated APRM ATWS permissive signal present.	19. ADS actuation does not occur.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
20. The ADS has a Manual Inhibit of the automatic ADS initiation.	20. A test of the ADS will be conducted with a generated signal of the ADS Manual Inhibit set to inhibit.	20. ADS actuation does not occur.

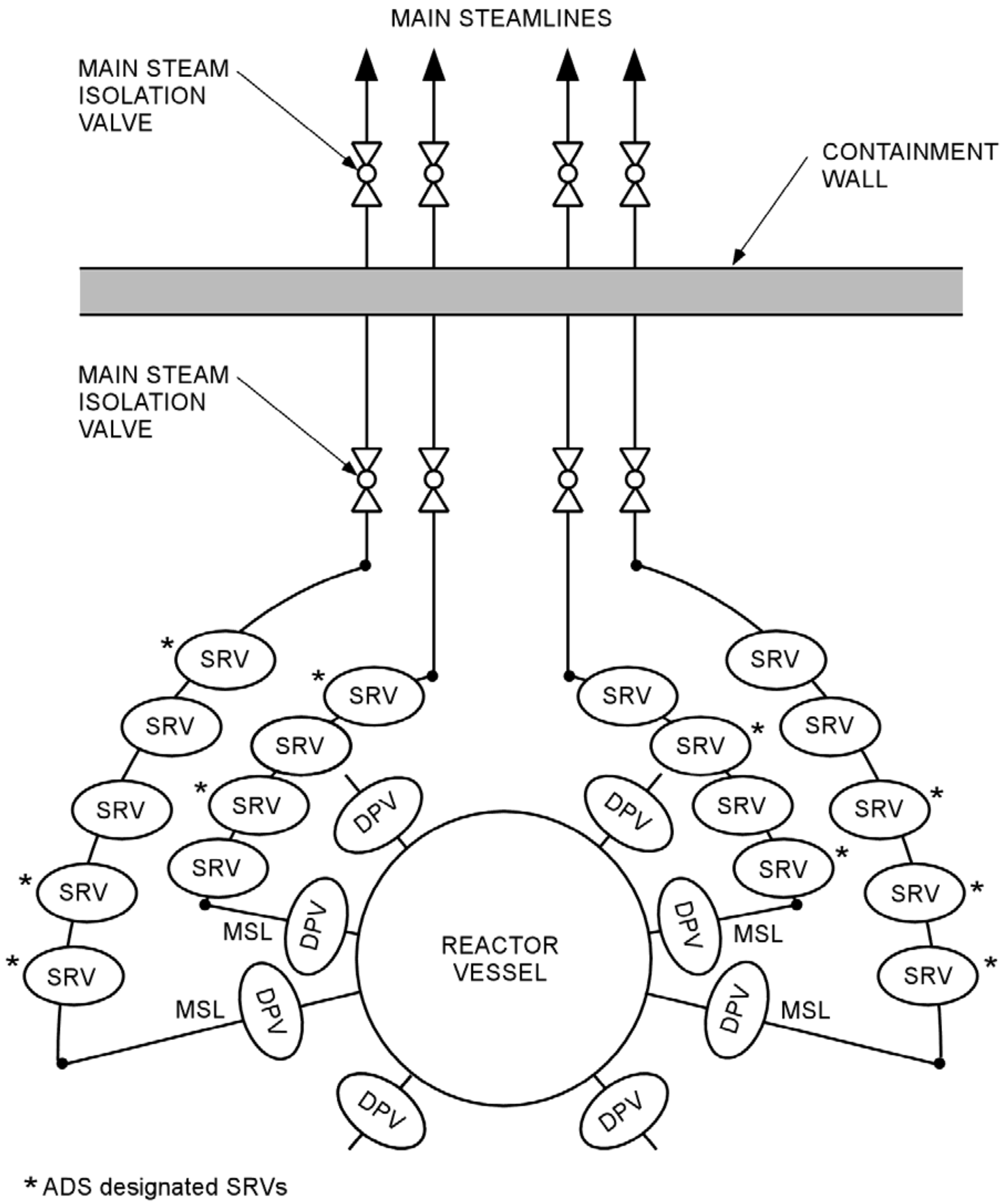


Figure 2.1.2-1. Safety-Relief Valves, Depressurization Valves and Steamline Diagram

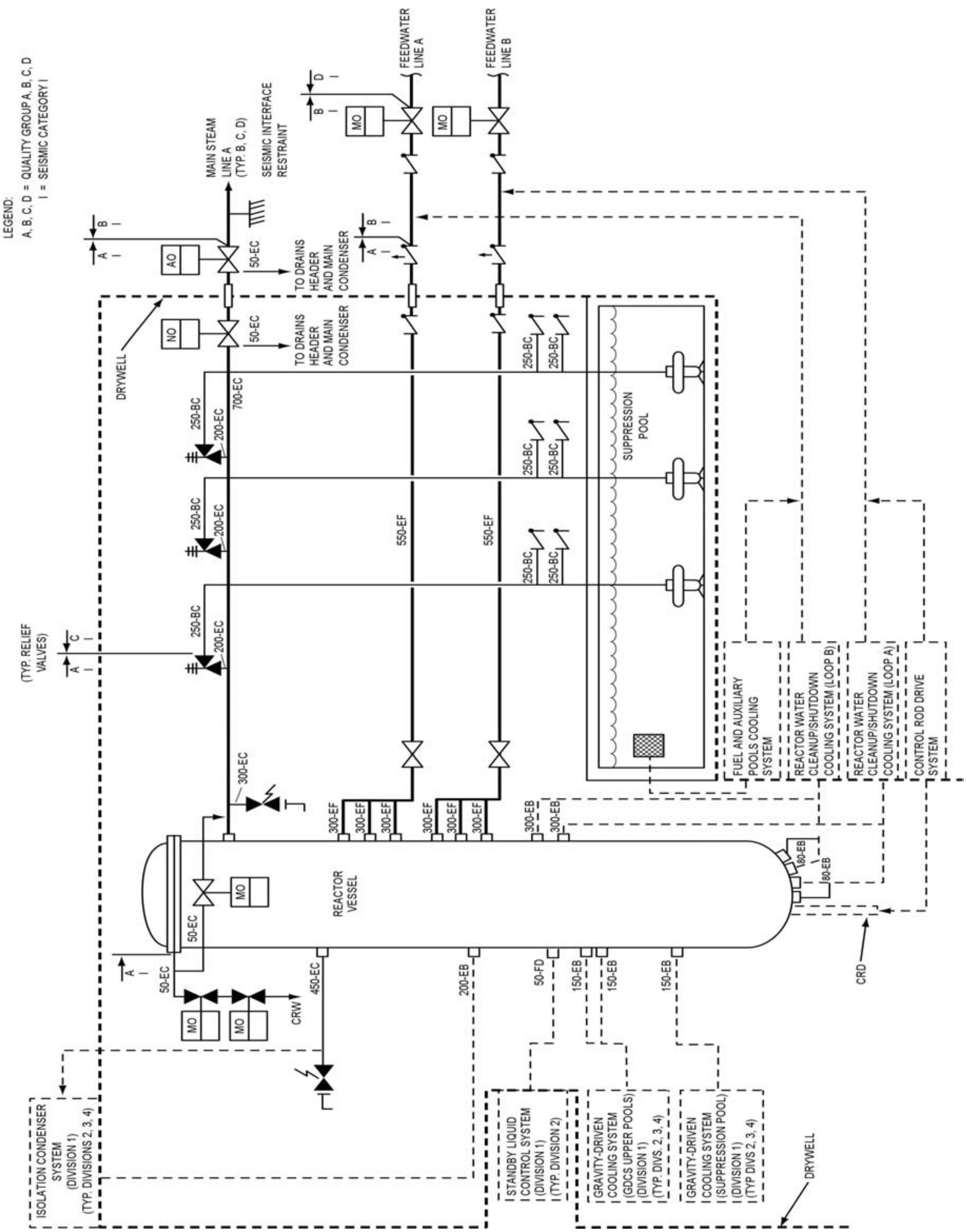


Figure 2.1.2-2. NBS Steamlines and Feedwater Lines

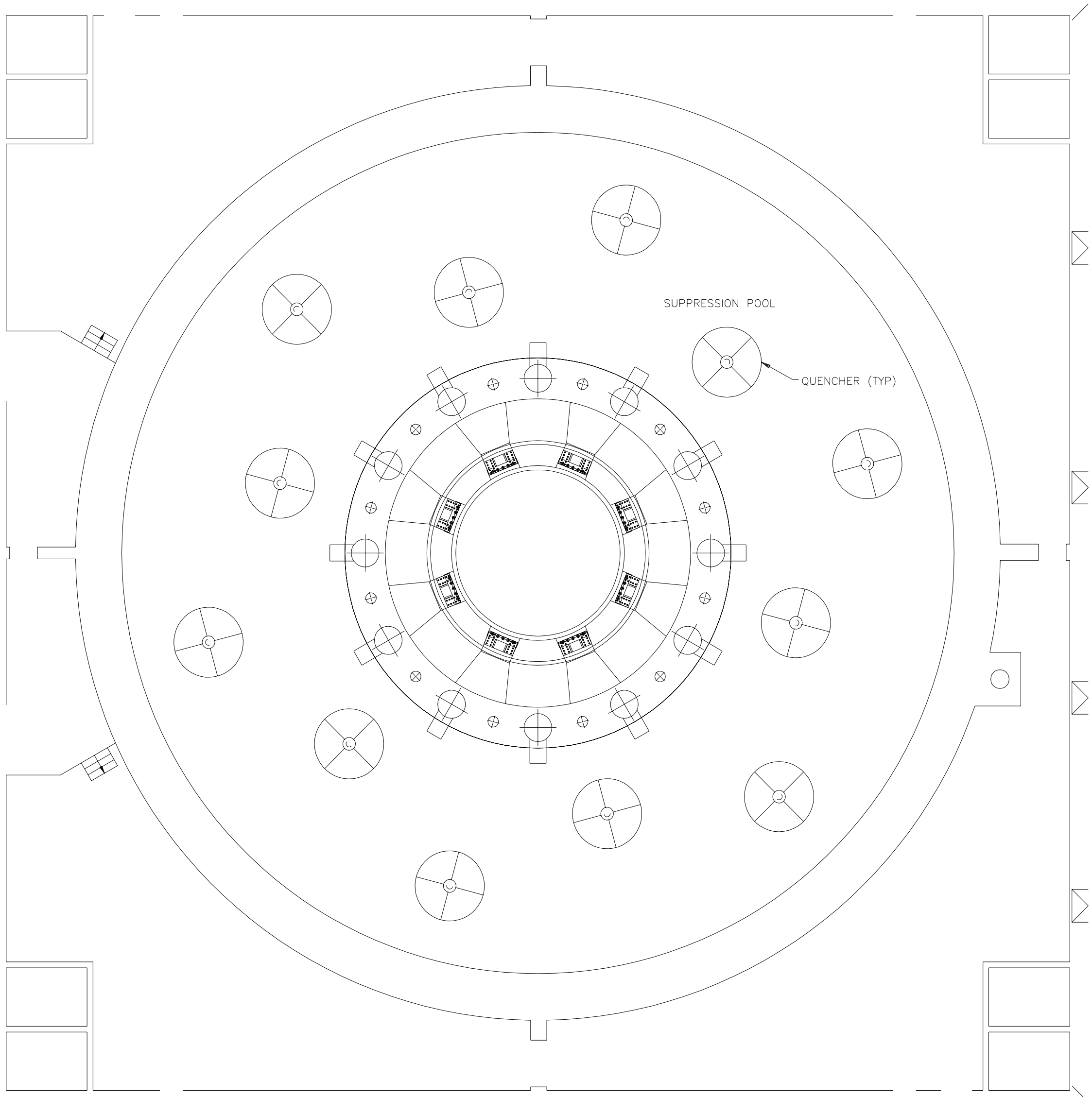


Figure 2.1.2-3. Safety-Relief Valve Discharge Line Quencher Arrangement

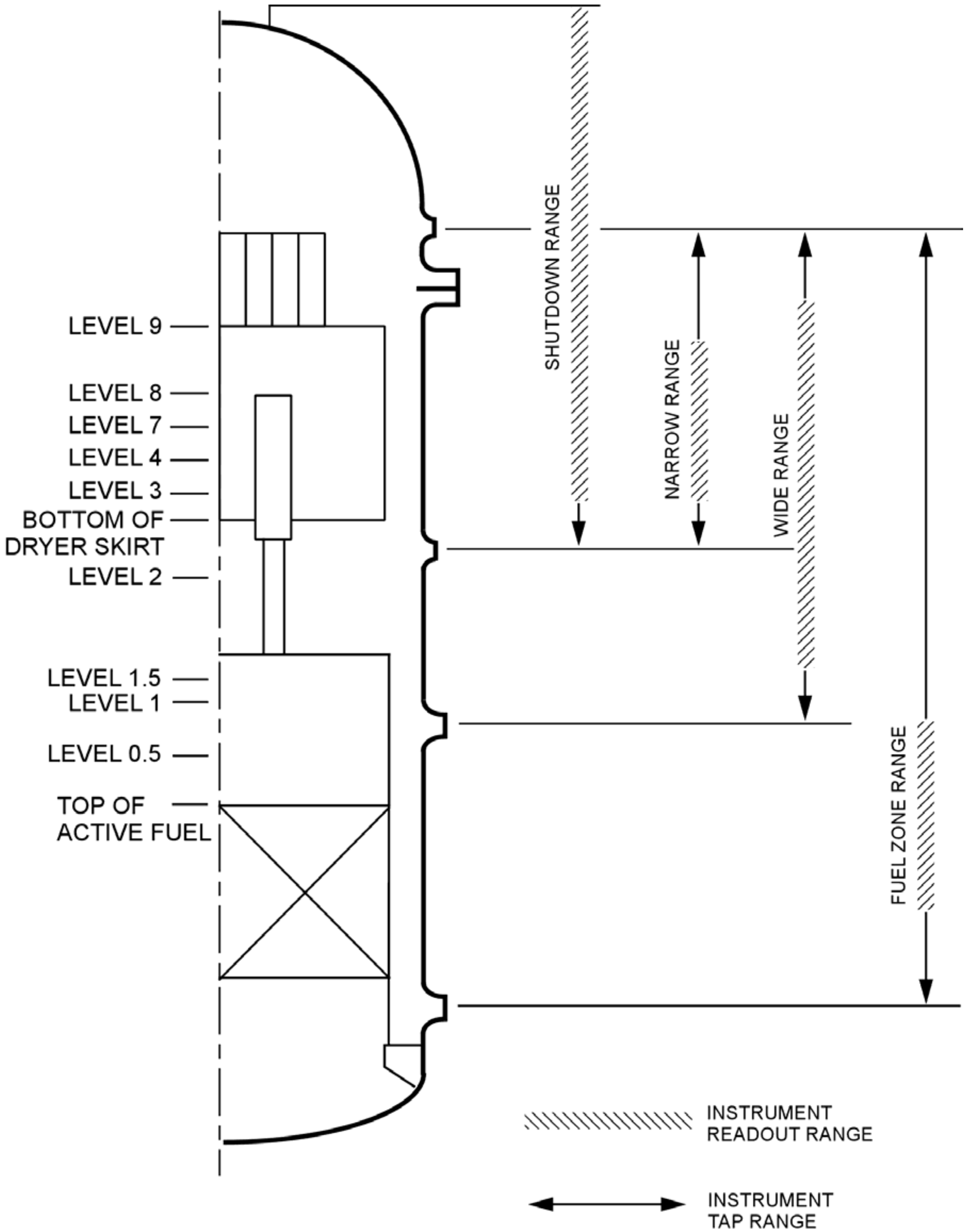


Figure 2.1.2-4. NBS Water Level Instrumentation

2.1.3 RPV Natural Circulation Process

Design Description

The ESBWR uses natural circulation to provide core flow. Natural circulation in the ESBWR is established due to the density differences between the water in the vessel annulus (outside the shroud and chimney) and the steam/water mixture inside the shroud and chimney. The colder higher density water in the annulus creates a higher pressure or a driving head when compared to the hotter lower density fluid (steam/water) in the core and chimney. It is the energy produced in the reactor core, which heats and begins to convert the water entering at the bottom of the core, to a steam/water mixture. In the core the subcooled water is first heated to the saturation temperature and then additional heat is added, starting the boiling process of the core coolant. As the coolant travels upward through the core the percent of saturated steam increases until it exits the core. This steam/water mixture travels upward through the chimney to the steam separators where centrifugal force separates the steam from the water. The separated, saturated water returns to the volume around the separators while the slightly "wet" steam travels upward through the steam dryers and eventually out the main steam nozzle and piping to the turbine.

Cooler feedwater re-enters the vessel at the top of the annulus and mixes with the saturated water around the separators and subcools this water. The resulting mixture is subcooled below the saturation temperature. The cooler mixture then travels downward through the annulus to re-enter the core. The water therefore forms a recirculation loop within the vessel. The mass of steam leaving the vessel is matched by the mass of feedwater entering.

The chimney adds height to this density difference, in effect providing additional driving head to the circulation process.

Figure 2.1.3-1 illustrates the natural circulation process for the ESBWR.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.3-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the natural circulation process.

Table 2.1.3-1

ITAAC For RPV Natural Recirculation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The pressure loss coefficients of the following components are less than what was used in the natural circulation flow analysis: a. steam separator b. fuel bundle c. fuel support piece orifice d. control rod guide tubes e. control rod drive housings f. shroud support bracket geometry	1. Test records will be reviewed and/or analyses will be performed to confirm the pressure loss coefficients.	1. Test reports and analyses exist and conclude that the pressure loss coefficients of the components are no greater than what was used in the natural circulation flow analysis.

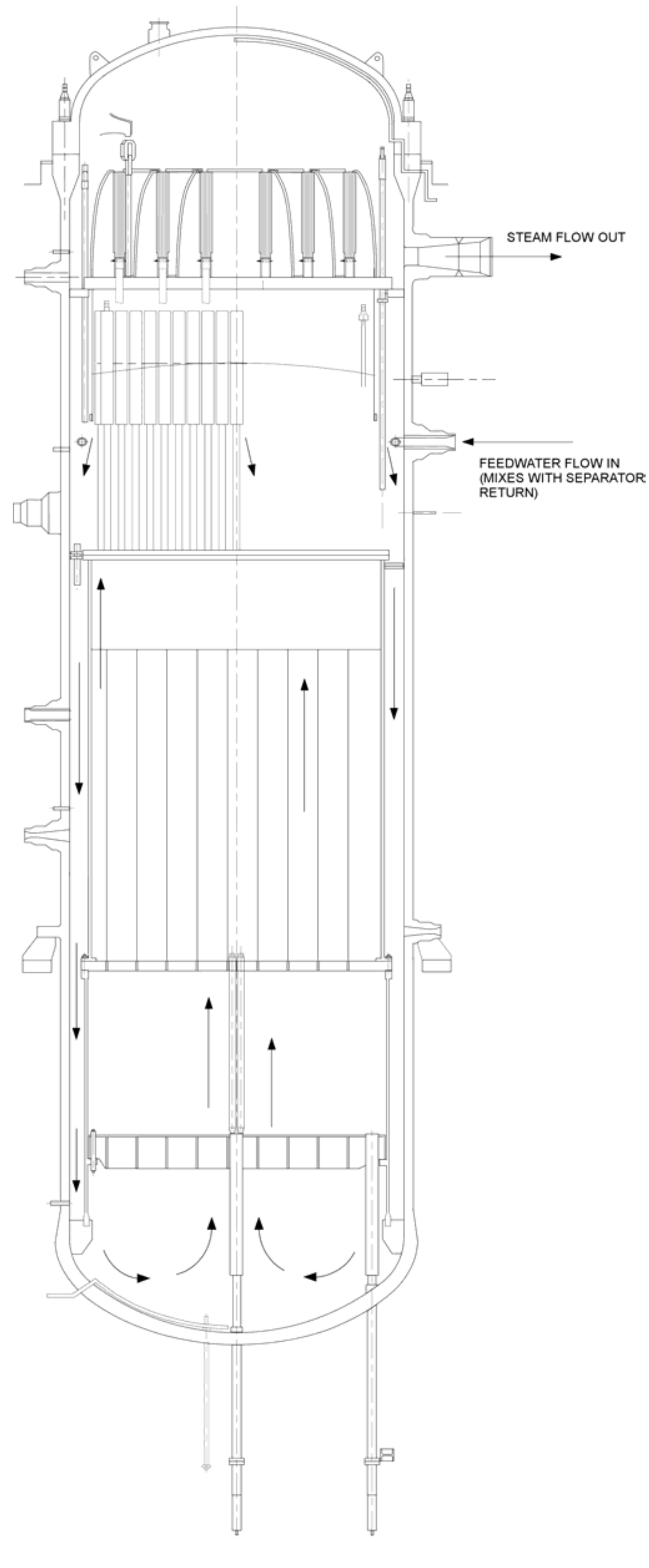


Figure 2.1.3-1. RPV Natural Circulation Process

2.2 INSTRUMENTATION AND CONTROL SYSTEMS

The following subsections describe the major instrumentation and control (I&C) systems for the ESBWR.

2.2.1 Rod Control and Information System

Design Description

The Rod Control and Information System (RC&IS) controls and monitors positioning of the control rods in the reactor by the Fine Motion Control Rod Drive (FMCRD) units of the Control Rod Drive (CRD) System. The RC&IS controls rod position to permit changes in core reactivity so that reactor power level and power distribution can be controlled.

The RC&IS utilizes a dual-redundant architecture for normal monitoring of control rod positions and executing normal control rod movement commands. The major components of the RC&IS and their interconnections and interfaces with other plant systems are shown on Figure 2.2.1-1.

The RC&IS does not perform or ensure any safety-related function, and thus, is a non-seismic , nonsafety-related system.

The RC&IS provides the following:

- The capability to control reactor power level by means of movement control of control rods in reactor core in manual, semiautomatic, and automatic modes of operation.
- Controls for RC&IS bypass and surveillance test functions, and summary information of control rods position and status on the RC&IS operator interface in the main control room.
- Transmission of control rods position and status data to other plant systems (e.g., the Non-Essential DCIS).
- Automatic control rod run-in of operable control rods following a scram (scram follow function).
- Automatic enforcement of rod movement blocks to prevent potentially undesirable rod movements (these blocks do not have an effect on scram function).
- The capability to control insertion of control rods by an alternate and diverse method [Alternate Rod Insertion (ARI) motor run-in function], which is electro-mechanical.
- The capability to enforce pre-established control rod pattern restrictions when reactor power is below the low power setpoint.
- The capability to enforce fuel operating and safety thermal limits when reactor power is above the low power setpoint.
- The capability to insert a selected group of control rods to their target position upon receipt of Selected Control Rod Run In (SCRRI) signals from Non-Essential DCIS (NE-DCIS).

The RC&IS equipment is located in the Reactor Building and Control Building.

The RC&IS dual channel scope equipment is powered by two separate, non-divisionalAC power sources with at least one power source being a non-Class 1E uninterruptible power supply. The Induction Motor Controller Cabinets (IMCCs), Rod Brake Controller Cabinets (RBCCs) and Emergency Ros Insertion Panels (ERIPs) are powered by the Low Voltage Distribution System.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.1-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the RCIS.

Table 2.2.1-1

ITAAC For Rod Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The two channels of RC&IS are independent of each other, such that each channel can independently cause a rod block; and for the normal RC&IS functions of control rod movements and control rod position monitoring, the two channels must be in agreement.	1. Tests will be performed to confirm channel redundancy, channel protective function independence, and two channel agreement for normal RC&IS operation.	1. When one RC&IS channel is disabled, the other channel causes a rod block. It takes the agreement of the two channels to allow movement of control rods during normal RC&IS operation.
2. RC&IS is designed to be capable of continued operation when different subsystems of RC&IS are bypassed. RC&IS bypass interlock logic precludes a bypass state that would render RC&IS inoperable.	2. Tests will be conducted to confirm RC&IS bypass capabilities and to confirm the function of the bypass interlock logic.	2. When different RC&IS subsystems are bypassed, as allowed by RC&IS bypass interlock logic, RC&IS is capable of continued operation. RC&IS bypass interlock logic prevents a bypass state that would render RC&IS inoperable.
3. When reactor power level is below low power setpoint, the Rod Worth Minimizer (RWM) of RC&IS enforces control rod withdrawal and insertion sequence to comply with pre-established sequence restrictions, by issuing a rod movement block signal whenever an out of sequence rod pattern is detected or whenever an out of sequence individual control rod or gang of control rods is selected.	3. Tests of RC&IS RWM will be conducted to withdraw/insert control rods that are both in-compliance and not-in-compliance with the pre-established sequence restrictions, using simulated signals for reactor power below the low power setpoint.	3. A rod block signal by RWM is initiated when an out of sequence rod withdraw/insert is performed or attempted, and when the simulated reactor power signals are below the low power setpoint.

Table 2.2.1-1

ITAAC For Rod Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. When reactor power is above low power setpoint, the Automated Thermal Limit Monitor (ATLM) of RC&IS enforces fuel operating and safety thermal limits (both MCPR and MLHGR) by issuing a rod withdrawal block signal whenever local fuel operating thermal limits are approached.	4. Tests of RC&IS ATLM will be conducted using simulated signals for the LPRMs, APRMs and control rod position data inputs to ATLM.	4. Initiation of rod block signal by ATLM upon inputting a simulated condition of approaching fuel operating thermal limits to ATLM, when reactor power is above the low power setpoint.

Table 2.2.1-1

ITAAC For Rod Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. When RC&IS is in "Automatic Mode" of operation, the Rod Action and Position Information (RAPI) of RC&IS, automatically withdraws and inserts control rods in compliance with a pre-established rod withdraw/insert sequence called the Reference Rod Pull Sequence (RRPS), under the command of the Plant Automation System (PAS). When RC&IS is in "Semiautomatic Mode," RAPI automatically selects and withdraws/inserts control rods based on RRPS when the operator activates withdrawal or insertion movements. When RC&IS is in "Manual Mode" the operator can withdraw/insert rods manually. When RC&IS is in "Manual Mode", when manual withdrawal or insertion of control rods results in a rod pattern not in compliance with RRPS, RC&IS generates an alarm.	5. Tests of RC&IS will include tests to verify that RAPI of RC&IS, in compliance with RRPS, executes rod withdraw/insert commands based on simulated signals of the PAS in the Automatic Mode; that in Semiautomatic Mode, RC&IS in compliance with RRPS, automatically performs the selection and movement of control rods when the operator activates insertion and withdrawal movements; that when RC&IS is in Manual Mode, control rods can be withdrawn/inserted manually; and that when in Manual Mode, RC&IS generates an alarm when a rod pattern that does not comply with the applicable RRPS is detected.	5. The certified design commitment is met.
6. On receipt of Selected Control Rod Run In (SCRRI) signals from the NE-DCIS, RC&IS automatically inserts a predetermined number of control rods to a predetermined position for each control rod.	6. Tests of RC&IS will be conducted using simulated SCRRI signals	6. The certified design commitment is met.

Table 2.2.1-1

ITAAC For Rod Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. On receipt of Scram Follow signals from Reactor Protection System (RPS), RC&IS automatically initiates motor-driven run-in of the FMCRDs to their full-in position.	7. Tests of RC&IS will be conducted using simulated Scram Follow signals from RPS.	7. The certified design commitment is met.
8. RC&IS, on receipt of an Alternate Rod Insertion (ARI) signals from the NE-DCIS, automatically initiates motor-driven run-in of the FMCRDs to their full-in position.	8. Tests of RC&IS will be conducted using simulated ARI signals from NE-DCIS.	8. The certified design commitment is met.
9. RC&IS transmits control rod position and status data to the NE-DCIS and Neutron Monitoring System	9. Tests of RC&IS will be conducted to output control rods coordinates, positions, and status.	9. The certified design commitment is met.
10. RC&IS enforces control rod withdrawal blocks as required by RPS, CRDS, and NMS.	10. Tests of RC&IS will be conducted using simulated inputs from RPS, CRDS, and NMS.	10. The certified design commitment is met.
11. The Induction Motor Controller Cabinets (IMCCs), Rod Brake Controller Cabinets (RBCCs) and Emergency Rod Insertion Panels (ERIPs) of RC&IS are powered from the Low Voltage Distribution System.	11. A test of the Low Voltage Distribution System availability to the IMCCs, RBCCs and ERIPs of RC&IS will be conducted.	11. The certified design commitment is met.

Table 2.2.1-1

ITAAC For Rod Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. Rod Action Control Cabinets (RACCs), Remote Communication Cabinets (RCCs) and the DOI of RC&IS are powered from two independent non-Class 1E power sources, with at least one of the independent sources being a non-Class 1E uninterruptible power source.	12. A test of the non-Class 1E redundant power source availability to the RACCs, RCCs and DOI of RC&IS will be conducted.	12. The certified design commitment is met.
13. The equipment comprising the RC&IS is defined in Subsection 2.2.1.	13. Inspections of the as-built system will be performed.	13. The as-built RC&IS conforms with the description in Subsection 2.2.1.

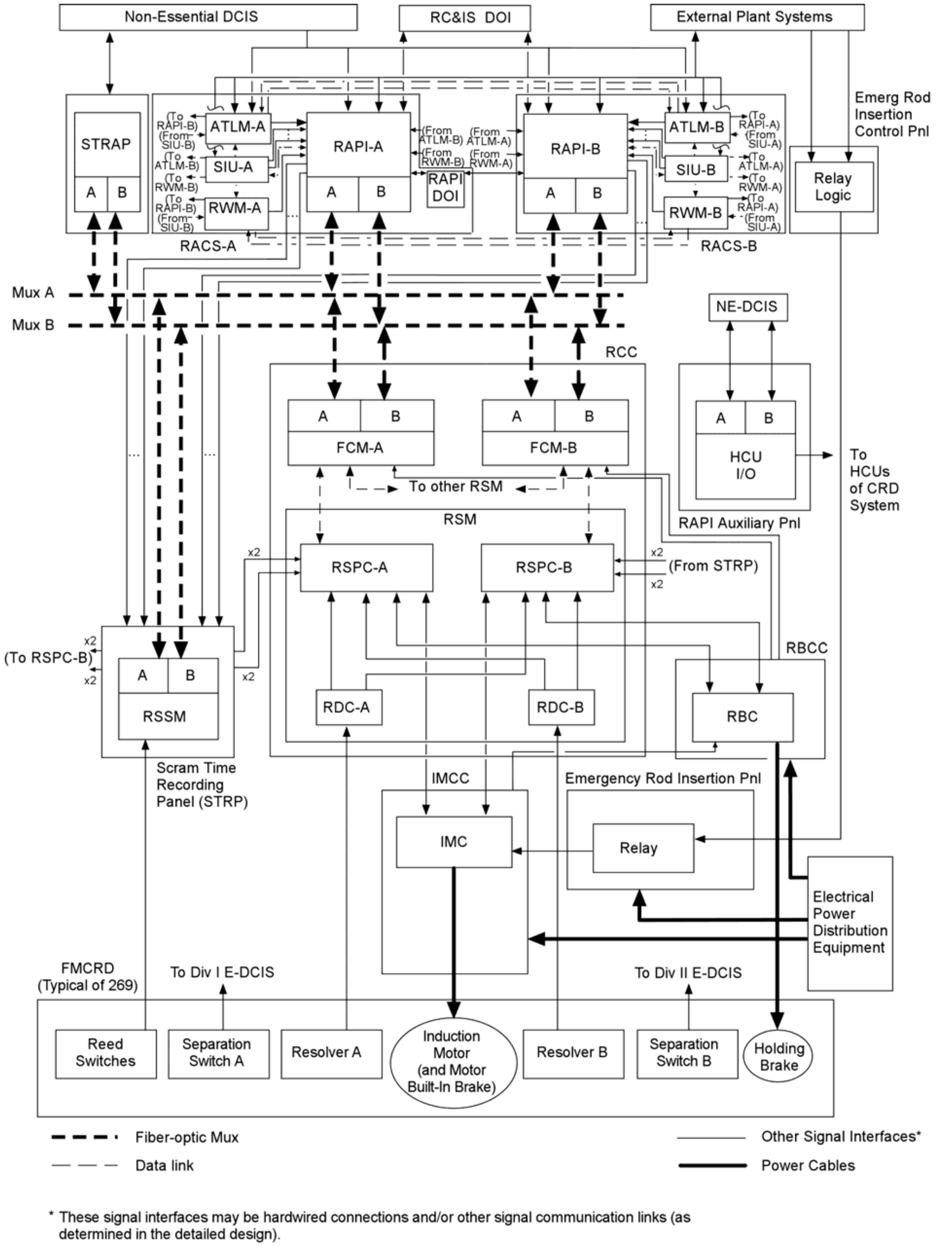


Figure 2.2.1-1. Rod Control and Information System Control Logic Block Diagram

2.2.2 Control Rod Drive System

Design Description

The Control Rod Drive (CRD) system controls changes in core reactivity during power operation by movement and positioning of the neutron absorbing control rods within the core in fine increments in response to control signals from the Rod Control and Information System (RC&IS). The CRD system provides rapid control rod insertion (scram) in response to manual or automatic signals from the Reactor Protection System (RPS). Figure 2.2.2-1 shows the basic system configuration and scope.

The CRD system consists of three major elements:

(1) The electro-hydraulic fine motion control rod drive (FMCRD) mechanisms;

(2) The hydraulic control unit (HCU) assemblies; and

(3) The control rod drive hydraulic subsystem (CRDHS).

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) for abnormal operating conditions. Simultaneous with scram, the FMCRDs also provide electric-motor driven run-in of control rods as a path to rod insertion that is diverse from the hydraulic-powered scram. The hydraulic power required for scram is provided by high pressure water stored in the individual HCUs. Each HCU can scram two FMCRDs. It also provides the flow path for purge water to the associated drives during normal operation. The CRDHS supplies pressurized water for charging the HCU scram accumulators and purging to the FMCRDs. The CRDHS also provides high pressure makeup water to the reactor during events in which the feedwater system is unable to maintain reactor water level.

The FMCRDs are mounted in housings, welded into the reactor vessel bottom head. The FMCRD has a movable hollow piston tube that is coupled at its upper end, inside the reactor vessel, to the bottom of a control rod. The FMCRD can move the control rod up or down over its entire range, by a ball nut and ball screw driven by the electric motor. In response to a scram signal, the piston inserts the control rod into the core hydraulically using stored energy in the HCU scram accumulator. The scram water is introduced into the drive through a scram inlet connection on the FMCRD housing, and is then discharged directly into the reactor vessel via clearances between FMCRD parts. The average scram times of all FMCRDs are:

Percent Insertion	Time (sec)
10	≤ 0.34
40	≤ 0.80
60	≤ 1.15
100	≤ 2.23

These times are measured starting from loss of signal to the scram solenoid pilot valves in the HCUs.

The FMCRD has an electro-mechanical brake with a minimum holding torque of 49 N·m on the motor drive shaft and a ball check valve at the point of connection with the scram inlet line.

Two redundant and separate switches in the FMCRD detect separation of the hollow piston from the ball nut.

Each HCU provides water stored in a pre-charged accumulator for scramming two FMCRDs. Figure 2.2.2-1 shows the major HCU components. The accumulator is connected to its associated FMCRDs by a hydraulic line that includes a scram valve held closed by pressurized control air. To cause a scram, the RPS provides a signal to de-energize the scram solenoid pilot valve (SSPV) that vents the control air from the scram valve, which then opens by spring action. Loss of either electrical power to the SSPV or loss of control air pressure causes scram. A pressure switch detects low accumulator gas pressure and actuates an alarm in the main control room.

The CRD system also provides alternate rod insertion (ARI) as a means of actuating hydraulic scram when an anticipated transient without scram (ATWS) condition exists. Following receipt of an ARI signal, solenoid valves on the scram air header open to reduce pressure in the header, allowing the HCU scram valves to open. The control rod drives then insert the control rods hydraulically.

The CRDHS has pumps, valves, filters, instrumentation, and piping to supply pressurized water for charging the HCUs and purging the FMCRDs.

The CRD system components classified as safety-related are: the HCU components required for scram; the FMCRD components required for scram; the scram inlet piping; the FMCRD reactor coolant primary pressure boundary components; the FMCRD brake and ball check valve; the internal drive housing support; the FMCRD separation switches; and the HCU charging water header pressure instrumentation.

The CRD system components classified as Seismic Category I are: the HCU components required for scram; the FMCRD components required for scram; the scram inlet piping; the FMCRD reactor coolant primary pressure boundary components; the FMCRD brake and ball check valve; the internal drive housing support; the FMCRD separation switches; and the HCU charging water header pressure instrumentation.

Figure 2.2.2-1 shows the ASME Code class for the CRD system piping and components.

The FMCRDs are mounted to the reactor vessel bottom head inside primary containment. The HCUs and CRDHS equipment are located in the Reactor Building at the basemat elevation. The HCUs are housed in the Reactor Building at the basemat elevation.

Each of the four divisional HCU charging header pressure sensors is powered from their respective divisional Class 1E power supply. Independence is provided between the Class 1E divisions for these sensors, and between the Class 1E divisions and non-Class 1E equipment.

For the FMCRD separation switches, independence is provided between the Class 1E divisions, and between the Class 1E divisions and non-Class 1E equipment.

The CRD system has the following alarms, displays, and controls in the main control room:

(1) Alarms for separation of the hollow piston from the ball-nut and low HCU accumulator gas pressure.

(2) Parameter displays for the instruments shown in Figure 2.2.2-1.

(3) Controls and status indication for the CRD pumps and flow control valves shown on Figure 2.2.2-1.

(4) Status indication for the scram valve position .

The following CRD system safety-related electrical equipment are located in either the Reactor Building or primary containment and are qualified for a harsh environment: the HCU charging header pressure instrumentation, the scram solenoid pilot valves, and FMCRD separation switches.

The FMCRD ball check valve has a safety related function to actuate to close the scram inlet port by reverse flow under system pressure, fluid flow and temperature conditions caused by a break in the scram line.

The check valves (CVs) shown inside the HCU boundary on Figure 2.2.2-1 have active safety-related functions to close under system pressure, fluid flow, and temperature conditions of scram.

The minimum flow supplied to the reactor in the high pressure makeup mode of operation is 3920 l/m with both CRD pumps operating and 1960 l/m with one pump operating and reactor pressure less than or equal to 8.62 MpaG.

The piping and components of the CRD pump suction supply, which extends from the CRD system interfaces with the Condensate and Feedwater System (C&FS) and Condensate Storage and Transfer System (CS&TS) to the inlet connections of the CRD pumps, are designed for 2.82 MPaG for intersystem loss-of-coolant accident (ISLOCA) conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.2-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the CRD system.

Table 2.2.2-1

ITAAC For Control Rod Drive System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the CRD system is as shown on Figure 2.2.2-1.	1. Inspections of the as-built system will be conducted.	1. The as-built CRD system conforms with the basic configuration shown on Figure 2.2.2-1.
2. The ASME Code components of the CRD 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 CRD system required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the CRD system conform with the requirements in the ASME Code, Section III.
3. The FMCRD can move the control rod up or down over its entire range by a ball nut and ball screw driven by the electric motor.	3. Tests will be conducted on each installed FMCRD.	3. Each control rod moves up and down over its entire range.
4. The average scram times of all FMCRDs with the reactor pressure as measured at the vessel bottom below 7.481 MPaG (1085 psig) are: Percent Insertion — Time (s) 10 — ≤ 0.34 40 — ≤ 0.80 60 — ≤ 1.15 100 — ≤ 2.23 These times are measured starting from loss of signal to the scram solenoid pilot valves in the HCU.	4. Tests will be conducted on each installed HCU and its associated FMCRD. The results of the tests performed at low reactor pressure will be extrapolated to the Design Commitment pressure.	4. The average scram times of all FMCRDs with the reactor pressure as measured at the vessel bottom below 7.481 MPaG (1085 psig) are: Percent Insertion — Time (s) 10 — ≤ 0.34 40 — ≤ 0.80 60 — ≤ 1.15 100 — ≤ 2.23 These times are measured starting from loss of signal to the scram solenoid pilot valves in the HCU.

Table 2.2.2-1

ITAAC For Control Rod Drive System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The FMCRD has an electro-mechanical brake with a minimum holding torque of 49 N m on the motor drive shaft.	5. Tests of each FMCRD brake will be conducted in a test facility.	5. The FMCRD electro-mechanical brake has a minimum holding torque of 49 N m on the motor drive shaft.
6. Two redundant and separate switches in the FMCRD detect separation of the hollow piston from the ball nut.	6. Tests of each as-built FMCRD will be conducted.	6. Both switches in each FMCRD detect separation of the hollow piston from the ball nut.
7. Following receipt of an ARI signal, solenoid valves on the scram air header actuation open to reduce pressure in the header, allowing the HCU scram valves to open.	7. Tests will be conducted on the as-built ARI valves using a simulated signal.	7. Following receipt of a simulated ARI signal, solenoid valves on the scram air header open to reduce pressure in the header, allowing the HCU scram valves to open.
8. Each of the four divisional HCU charging header pressure sensors are powered from their respective divisional Class 1E power supply. For the four HCU charging water header pressure sensors, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	8. a. Tests will be conducted on the as-built charging water header sensors by providing a test signal in only one Class 1E division at a time. b. Inspections of the as-installed charging water header sensor Class 1E divisions will be conducted.	8. a. The test signal exists only in the Class 1E Division under test. b. 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 2.2.2-1

ITAAC For Control Rod Drive System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. For the FMCRD separation switches, independence is provided between the Class 1E divisions and also between the Class 1E divisions and non	9. Inspections of the as-installed Class 1E divisions in the CRD system will be performed.	9. In the CRD system, physical separation or electrical isolation exists between Class 1E divisions. Physical separation or electrical isolation exists between Class 1E divisions and non-Class 1E equipment.
10. Main control room alarms, displays and controls provided for the CRD system are defined in Subsection 2.2.2.	10. Inspections will be performed on the main control room alarms, displays and controls for the CRD system.	10. Alarms, displays and controls exist or can be retrieved in the main control room as defined in Subsection 2.2.2.
11. CVs designated in Subsection 2.2.2 as having an active safety-related function close under system pressure, fluid flow, and temperature conditions.	11. Tests of installed valves for closing will be conducted under system preoperational pressure, fluid flow, and temperature conditions.	11. Each CV closes.
12. For the high pressure makeup mode of operation, the minimum flow supplied to the reactor is 3920 l/m with both CRD pumps operating and 1960 l/m with one pump operating with reactor pressure less than or equal to 8.62 MpaG.	12. Tests of the high pressure makeup flow capacity of the as-built system will be conducted.	12. The CRD system delivers a minimum flow to the reactor of 3920 l/m with both CRD pumps operating and 1960 l/m with one pump operating with reactor pressure less than or equal to 8.62 MpaG.

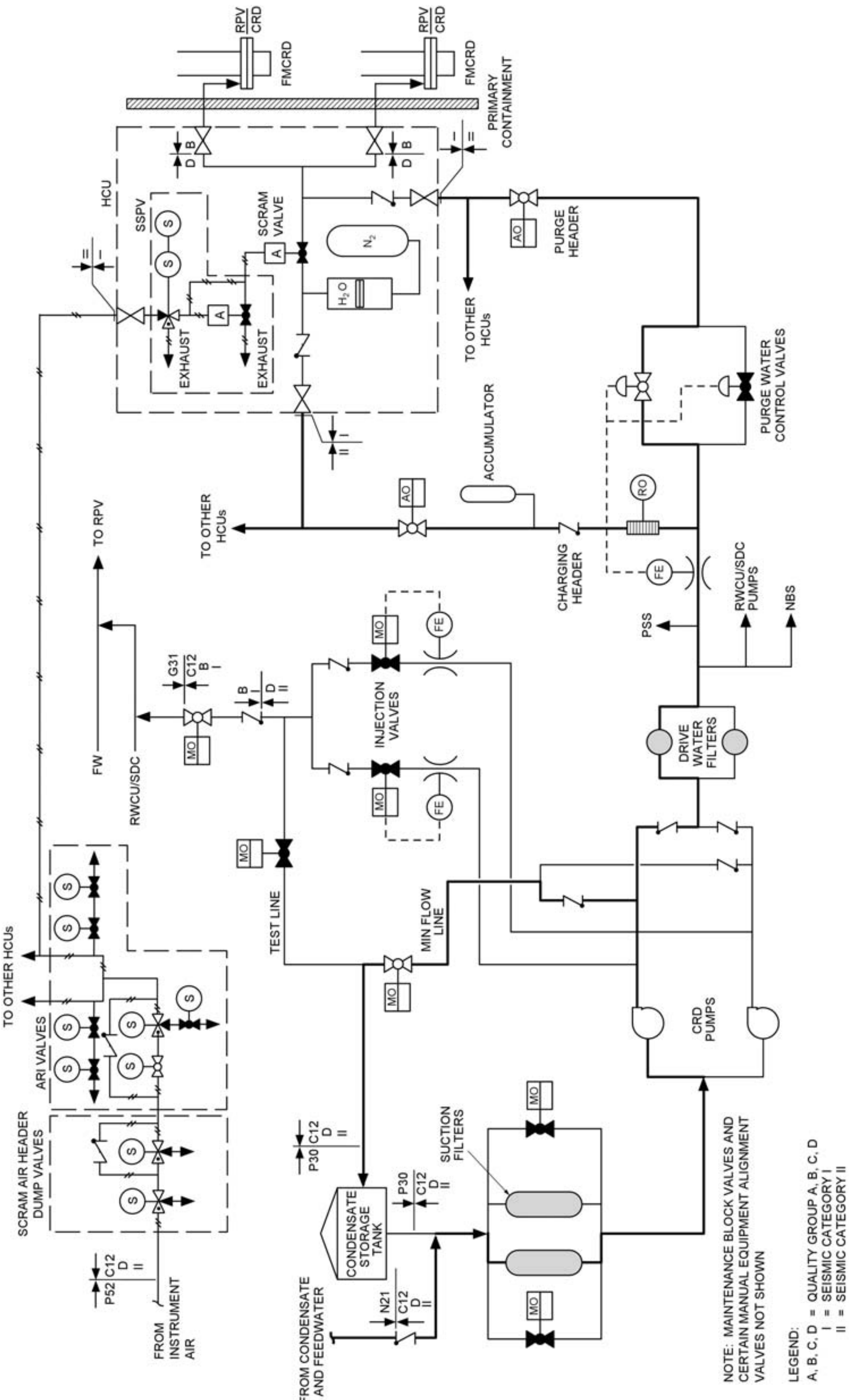


Figure 2.2.2-1. Control Rod Drive System

2.2.3 Feedwater Control System

Design Description

The Feedwater Control System (FWCS) provides logic for controlling the supply of feedwater flow to the reactor vessel in response to automatic or operator manual control signals. This control maintains reactor water level within predetermined limits for all operating conditions including startup. A fault-tolerant, triplicated, digital controller uses water level, steam flow and feedwater flow signals to form a three-element control strategy to accomplish this function. Single-element control based only on reactor water level is used when steam flow or feedwater flow signals are not available. At very low steam flow conditions during plant startup, the FWCS regulates the Reactor Water Cleanup System/Shutdown Cooling (RWCU/SDC) overboard flow to maintain reactor water level and to minimize feedwater temperature oscillations. FWCS receives input from and provides output to other systems through the Non-Essential Distributed Control & Information System (NE-DCIS) to accomplish its control function, as shown in the control interface diagram in Figure 2.2.3-1.

FWCS equipment consists of a Fault-Tolerant Digital Controller (FTDC), which is a triplicated, microprocessor-based, controller that executes the control software and logic required for reactor level control and other FWCS functions. There are three identical processing channels (operating in parallel) that receive inputs from other systems and issue actuator and speed demands, process measurement data, interlock and trip signals. The FTDC issues actuator demand signals to the Low Flow Control Valve (LFCV) and the RWCU/SDC overboard flow control valve and a speed demand signal to the Feedwater Pump variable speed controllers, which are all components of other systems.

The FWCS does not perform or ensure any safety-related function, and thus, is classified as nonsafety-related.

The normal range of reactor water level is between Level 4 and Level 7. If either of these limits is reached during normal operation, an alarm occurs in the control room to alert the operator.

A loss of feedwater heating, resulting in a significant decrease in feedwater temperature, generates a signal that initiates a Selected Control Rod Run-In (SCRRI). This interlock limits the consequences of a reactor power increase due to cold feedwater. In addition, the temperature difference between feedwater lines A and B aredifference between feedwater lines A and B is monitored and alarmed if found to be excessive.

If high water Level 8 is reached, a signal is generated to initiate runback of the feedwater demand to zero and trip the main turbine. This protects the turbine from excessive moisture carryover in the main steam. This interlock is implemented in a physically separate controller to ensure a trip function is available upon a common-mode failure of the FWCS FTDCs.

Upon receipt of an Anticipated Transient Without Scram (ATWS) trip signal from the ATWS logic cards of Safety System Logic and Control (SSLC) System, FWCS initiates a runback of feedwater pump feedwater demand to zero and closes the LFCV and the RWCU/SDC Overboard flow control valve. This reduces power and prevents dilution of the boron that would be injected to shut the reactor.

The total feedwater flow is displayed on the main control panel . The FWCS operating mode is selectable from the main control room.

Digital controllers used for the FWCS are redundant, with diagnostic capabilities that identify and isolate failure of level input signals.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.3-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Feedwater Control System.

Table 2.2.3-1

Feedwater Control Modes

Function	Modes
RPV water level control	Single Element (level only) Three Element (level, main steam flow, feedwater flow)
Variable speed feedwater pump speed demand	Manual Auto (speed control)
LFCV position demand	Manual Auto (level control) Auto-standby
RWCU/SDC Overboard Flow Control valve position demand	Manual Auto-level control Auto-flow control
Automation	Power Generation and Control Subsystem (PGCS), of Plant Automation System, mode Not in PGCS mode

Table 2.2.3-2

ITAAC For Feedwater Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The FWCS incorporates redundant, fault tolerant digital controllers (FTDC).	1. A test will be performed by simulating failure of an operating FWCS FTDC.	1. There is no loss of FWCS output upon loss of any one FTDC.
2. The FWCS FTDCs identify and isolate failure of process input signals.	2. Tests will be performed by simulating input signal failures to the FWCS FTDCs.	2. The FWCS FTDCs output signal is based upon the remaining valid input signals.
3. The FWCS is powered by redundant, uninterruptible power supplies.	3. A test shall be performed by simulating failure of a power supply to the FWCS.	3. There is no loss of FWCS output upon loss of any one power supply.
4. The FWCS configuration, monitored variables, trip functions and interfaces are as described in Subsection 2.2.3, Table 2.2.3-1 and Figure 2.2.3-1.	4. Inspections will be performed on the FWCS components and installed configuration. Using simulated signals, testing will be performed on the FWCS.	4. The certified design commitment is met.
5. Control Room indications and controls provided for the FWCS are as defined in Subsection 2.2.3.	5. Inspections will be performed on the control room indications and controls for the FWCS.	5. Indications and controls exist or can be retrieved in the Control Room as defined in Subsection 2.2.3.

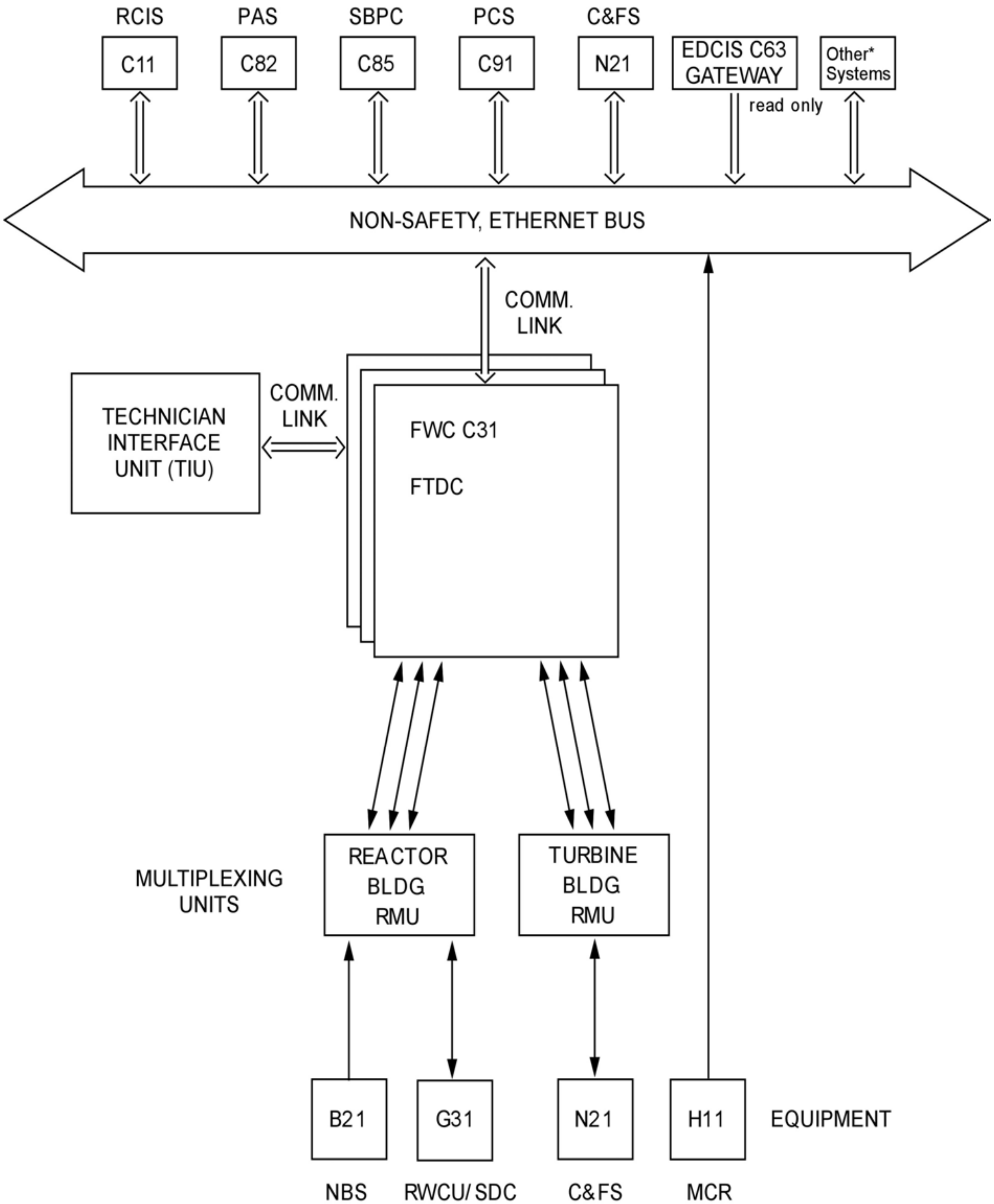


Figure 2.2.3-1. Feedwater Control System Logic Functional Diagram

2.2.4 Standby Liquid Control System

Design Description

The Standby Liquid Control (SLC) system injects neutron absorbing poison into the reactor using a boron solution, thus providing an alternate method of reactor shutdown, i.e., without control rods from full power to cold subcritical condition at its most reactive state.

The SLC system has two independent 50% capacity trains, which include piping, valves, accumulators and instrumentation that can inject a neutron absorber solution into the reactor. The SLC system is designed to operate over the range of reactor pressure conditions up to the elevated pressures of an anticipated transient without scram (ATWS) event, and to inject sufficient neutron absorber solution to reach subcritical conditions. The SLC system safety-related design parameters are presented in Table 2.2.4-1.

The SLC system interfaces with Class 1E divisional power for the squib-type injection valves; for the valve which isolates the accumulator after injection; for accumulator solution level measurement, trip, and alarm functions; and for the particular nuclear boiler system (NBS) instrumentation and safety system logic and control (SSLC) control logic which generates the ATWS signal for automatic SLC system initiation.

The SLC system is also designed to provide makeup water to the RPV to mitigate the consequences of loss of coolant accident (LOCA). The emergency core cooling system (ECCS) and the SLC are designed to flood the core during LOCA to provide required core cooling. By providing core cooling following a LOCA, the ECCS and SLC, in conjunction with the containment, limits the release of radioactive materials to the environment following a LOCA.

<u>Instrumentation</u>

All critical and essential instruments and control are displayed in Main Control Room (MCR) and available to operator for monitoring the status of the SLC system. The following key parameters for SLC system are displayed in MCR:

Accumulator level monitoring and alarms.

Accumulator pressure monitoring and alarms.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.4-2 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the SLC system.

Table 2.2.4-1

Standby Liquid Control System Parameters

Parameter	Value
SLC ATWS mitigation - Reactor pressure	8.61 MPa (1250 psia)
Approximate injection flow rate at the above reactor pressure	18.4 l/s (292 gpm)
Approximate average injection velocity for the first 5.4 m^3 of the flow	30.5 m/s (100 ft/s)
Approximate average injection velocity for the second 5.4 m^3 of the flow	18.4 m/s (60 ft/s)
Total boron solution injection at the above reactor pressure per each train	5.4 m^3 (1427 gal)
Equivalent natural boron concentration for the total solution injection volume, based on a hot shutdown liquid inventory. *	≥ 1600 ppm
With the water level at the main steam line, the total injection solution inventory (per each train), and equivalent natural boron concentration at cold shutdown conditions	≥ 7.8 m^3 (2061 gal) > 1100 ppm **

* Liquid inventory in RPV calculated based on reactor coolant level extending up to main steam line nozzle.

** This concentration ensures maintaining shutdown reactivity even after initiation and operation of the reactor shutdown cooling system.

Table 2.2.4-2

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the SLC system is shown in Figure 2.2.4-1.	1. Inspections of the as built system will be conducted.	1. The as built SLC system conforms to the basic configuration shown in Figure 2.2.4-1.
2. The performance of the SLC system is based on the following plant parameters.	2.	2.
a. Accumulator tank injectable boron solution volume equal to or greater than 7.8 m^3 (2061 gal) for each train.	a. The as-built dimensions will be used in a volumetric analysis to calculate the minimum injectable boron solution volume from each accumulator tank.	a. Accumulator injectable boron solution volume is equal to or greater than 7.8 m^3 (2061 gal) for each train.
b. The equivalent natural boron concentration for the total solution injection volume is ≥ 1600 ppm, based on the reactor in a hot shutdown condition with the liquid inventory in the RPV at the main steam line nozzle elevation.	b. An analysis will be performed to determine the equivalent natural boron concentration for the total solution injection volume based on the reactor in the hot shutdawn condition with the liquid inventory in the RPV at the main steam line nozzle elevation.	b. The equivalent natural boron concentration for the total solution injection volume is ≥ 1600 ppm, based on the reactor in a hot shutdown condition with the liquid inventory in the RPV at the main steam line nozzle elevation.
c. The equivalent natural boron concentration at cold shutdown conditions for the total solution injection volume is > 1100 ppm, based on the liquid inventory in the RPV at the main steam line nozzle elevation plus the liquid inventory in the reactor shutdown cooling piping and equipment of the RWCU/SDC System.	c. An analysis will be performed to determine the equivalent natural boron concentration at cold shutdown conditions for the total solution injection volume based on the liquid inventory in the RPV at the main steam line nozzle elevation plus the liquid inventory in the reactor shutdown cooling piping and equipment of the RWCU/SDC System.	c. The equivalent natural boron concentration at cold shutdown conditions for the total solution injection volume is > 1100 ppm, based on the liquid inventory in the RPV at the main steam line nozzle elevation plus the liquid inventory in the reactor shutdown cooling piping and equipment of the RWCU/SDC System.
d. An accumulator tank with a 12.5%		d. The solution equal to or greater than

Table 2.2.4-2

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
solution of sodium pentaborate with boron content enriched to 94% of the B_{10} isotope.	d. The solution will be tested for concentration and B_{10} enrichment.	12.5% sodium pentaborate and the B_{10} enrichment is equal to or greater than 94%.
3. The system shall be capable of delivering 10.8 m^3 of the injectable boron solution volume at the average velocities given in Table 2.2.4-1 in the ATWS event with accumulator at 14.82 MPa and reactor at 8.61 MPa..	3. Tests will be conducted with water to demonstrate acceptable system performance. Test of the system, while simulating high RPV pressure to measure the injection nozzle velocity	3. SLC system can inject 5.4 m^3 of the injectable boron solution volume per each train. The injection nozzle velocity meets the time average velocity design requirement for first 5.4 m^3 of the system flow and the next 5.4 m^3 of the total system flow.
4. The SLC system shall be capable of delivering a total of 7.8 m^3 per accumular of the injectable boron solution volume to provide makeup water to the RPV in response to a loss of coolant accident (LOCA).	4. Tests will be conducted with water to demonstrate acceptable system performance.	4. SLC system can inject a total volume of 7.8 m^3 per accumular of the injectable boron solution, in response to a LOCA.
5. Injection of boron into the reactor core begins within 5 seconds of reaching a system initiation parameter setpoint.	5. Tests of the system with unborated water will be conducted by simulating a system initiation parameter signal.	5. SLC injection into the reactor core begins within 5 seconds of reaching a system initiation parameter setpoint.
6. All power for the safety functions of SLC system are derived from the Class 1E 250 VDC and 120VAC electrical systems. Divisional assignments are made to ensure independence of redundant components.	6. Tests will be conducted after installation to confirm that the electrical power supply configurations are in compliance with design commitments.	6. The safety functions of SLC system are dependent only on Class 1E power supply 250 VDC and 120VAC and redundant components are located on separate divisions.

Table 2.2.4-2

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The ASME portions of the SLC system retain their integrity under internal pressures that will be experienced during service.	7. A hydrostatic test will be conducted on those portions of the SLC system that are required to be hydrostatically tested by the ASME Code.	7. The results of the hydrostatic test of the ASME portions of the SLC system conform to the requirements in the ASME Code, Subsection III.
8. Control Room alarms, indications and controls provided for the SLC system are as defined in Subsection 2.2.4.	8. Inspections will be performed on the control room alarms, indications and controls for the SLC system.	8. Alarms, indications and controls exist or can be retrieved in the Control Room as defined in Subsection 2.2.4.

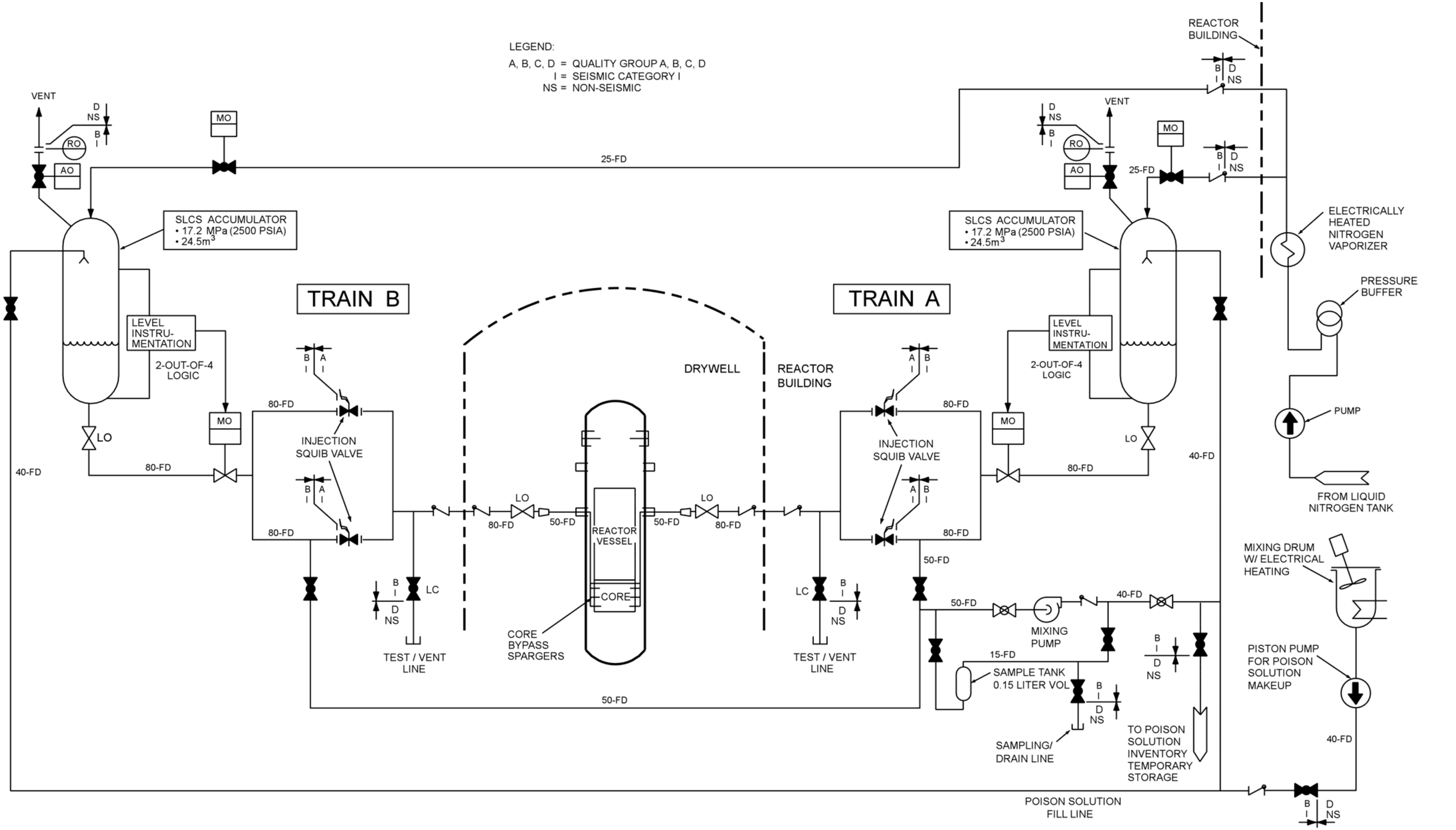


Figure 2.2.4-1. Standby Liquid Control System

2.2.5 Neutron Monitoring System

Design Description

The Neutron Monitoring System (NMS) is a neutron flux monitoring and supports the Reactor Protection System. The functions of the system are to:

(1) Monitor the thermal neutron flux in the reactor core;

(2) Provide trip signals to the Reactor Protection System (RPS);

(3) Provide plant power and power distribution information to the operator and plant control systems

(4) Provide permissives to ATWS, SSLC, and SLC

(5) Provide permissive inhibit to ADS

The startup range neutron monitor (SRNM), the local power range monitor (LPRM), and the average power range monitor (APRM) are classified as (Class 1E) safety-related. The automated incore instrument calibration system (automated fixed in-core probe) and the rod block monitoring function (multi-channel rod block monitor, [MRBM]) are classified as nonsafety-related.

The SRNM monitors neutron flux from the source range to 15% of the rated power. The SRNM has multiple channels, each with one detector, with the multiple channels distributed throughout the reactor core and assigned to four divisions. The SRNM detector is a fixed in-core sensor. Detector cables are separated according to different divisional assignment, connected to their designated preamplifiers located in the Reactor Building. The detector signals are then transmitted to signal processing electronic units in the Control Building.

The LPRM monitors local neutron flux in the power range up to 125% of the rated power, and overlaps part of the SRNM range. LPRM detector assemblies are distributed in the core, with four sensors per each LPRM assembly, to monitor local neutron flux level throughout the core. The LPRM assembly also contains space for the automated fixed in-core calibration detector. The LPRM detector outputs are connected to the APRM signal conditioning units, where the signals are processed and amplified. LPRM detector signals are divided and assigned to four APRM channels corresponding to four divisions. LPRM signals in each APRM channel are averaged and normalized to form an APRM signal, which represents the core average power.

The oscillation power range monitor (OPRM) is part of the APRM. Each OPRM receives the identical LPRM signals from the corresponding APRM channel as inputs, and forms many OPRM cells to monitor the neutron flux behavior of all regions of the core. The LPRMs signals assigned to each cell are averaged to provide an OPRM signal for this cell. The OPRM trip protection algorithm detects thermal hydraulic instability and provides trip output to the RPS if the trip setpoint is exceeded. The OPRM bypass is controlled by the bypass of its associated APRM channel.

The automated fixed in-core instrument calibration system provides local power information at various core locations that correspond to LPRM locations. The automated fixed in-core instrument calibration system uses its own set of in-core detectors for local power measurement and provides local power information for three-dimension core power determination and for the

calibration of the LPRMs. The measured data are sent to the plant computer for such calculation and LPRM calibration.

The Rod Control and Information System (RC&IS) uses LPRM signals to detect local power change during the rod withdrawal. If the averaged LPRMs signal exceeds a preset rod block setpoint, a control rod block demand is issued.

Figures 2.2.5-1 and 2.2.5-2 show the configuration of each SRNM division and APRM division.

Each of the four divisions of the SRNM, LPRMs and APRMs instruments is powered by its respective divisional Class 1E power supplies. In the NMS outside the primary containment, independence is provided between Class 1E divisions, and between the Class 1E divisions and non-Class 1E equipment.

The SRNM and APRM trip signal outputs are in four divisions. The SRNM trip and the APRM trip logic are independent from each other. The SRNM generates a high neutron flux trip or a short period trip signal. Any single SRNM channel trip causes a trip in its division. The APRM generates a high neutron flux trip, a simulated thermal power trip signal, or a core power oscillation trip signal. The NMS provides these trip signals to the RPS.

The SRNM and APRMs are fail-safe in the event of loss of electrical power to any division of their logic equipment.

The NMS bypass function is performed within the NMS. Within the NMS, the bypass functions of the SRNM and the APRMs are separate and independent from each other. The SRNM channels are grouped into four bypass groups. Individual SRNM channels can be bypassed, with one channel being able to be bypassed at any time within each bypass group. At any one time, up to four SRNM channels can be bypassed. At any one time, only one APRM channel can be bypassed. A bypassed SRNM channel or a bypassed APRM channel does not cause a trip output sent to the RPS.

The NMS provides SRNM flux permissive signal to the Standby Liquid Control (SLC) system and feedwater runback logics within the Safety System Logic and Control (SSLC), and an APRM flux permissive signal to the Nuclear Boiler System (NBS) logic within SSLC as part of the anticipated transient without scram (ATWS) logics. The SRNM and APRM flux permissive signals from the NMS indicate when the reactor power level is above or below the setpoint in order to allow or disallow the initiation of ATWS mitigation features, such as SLC initiation and Automatic Depressurization System (ADS) inhibit (in NBS).

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

- SRNM, LPRM, and APRM neutron flux and period displays;
- Trip and bypass status displays; and
- Bypass control devices including SRNM bypass switches (one per bypass group) and APRM bypass switch.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.5-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the NMS.

Table 2.2.5-1

ITAAC For The Neutron Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the NMS is as shown in Figures 2.2.5-1 and 2.2.5-2.	1. Inspections of the as-built NMS will be conducted.	1. The as-built NMS conforms with the basic configuration shown in Figure 2.2.5-1 and 2.2.5-2.
2. The trip functions of the SRNM and APRMS are implemented as described in Subsection 2.2.5.	2. SRNM and APRMS trip functions will be tested by using simulated signals.	2. The NMS system issues trip signals following receipt of simulated signals for the following trip functions: a. SRNM upscale and period trip; b. APRMS upscale trip; c. APRMS thermal power upscale trip; d. SRNM and APRMS inoperative trip.
3. The SRNM and PRNM power supplies are provided by the four 120VAC UPS busses.	3. Tests will be conducted after installation.	3. The installed safety-related equipment is powered from the four divisional Class 1E UPS.
4. The bypass logics of the SRNM subsystem and the APRMS subsystem are as described in Subsection 2.2.5 and are separate and independent of each other.	4. SRNM and APRMS Bypass functions will be tested by using simulated signals.	4. The as-built SRNM and APRMS bypass logics are in accordance with Subsection 2.2.5.

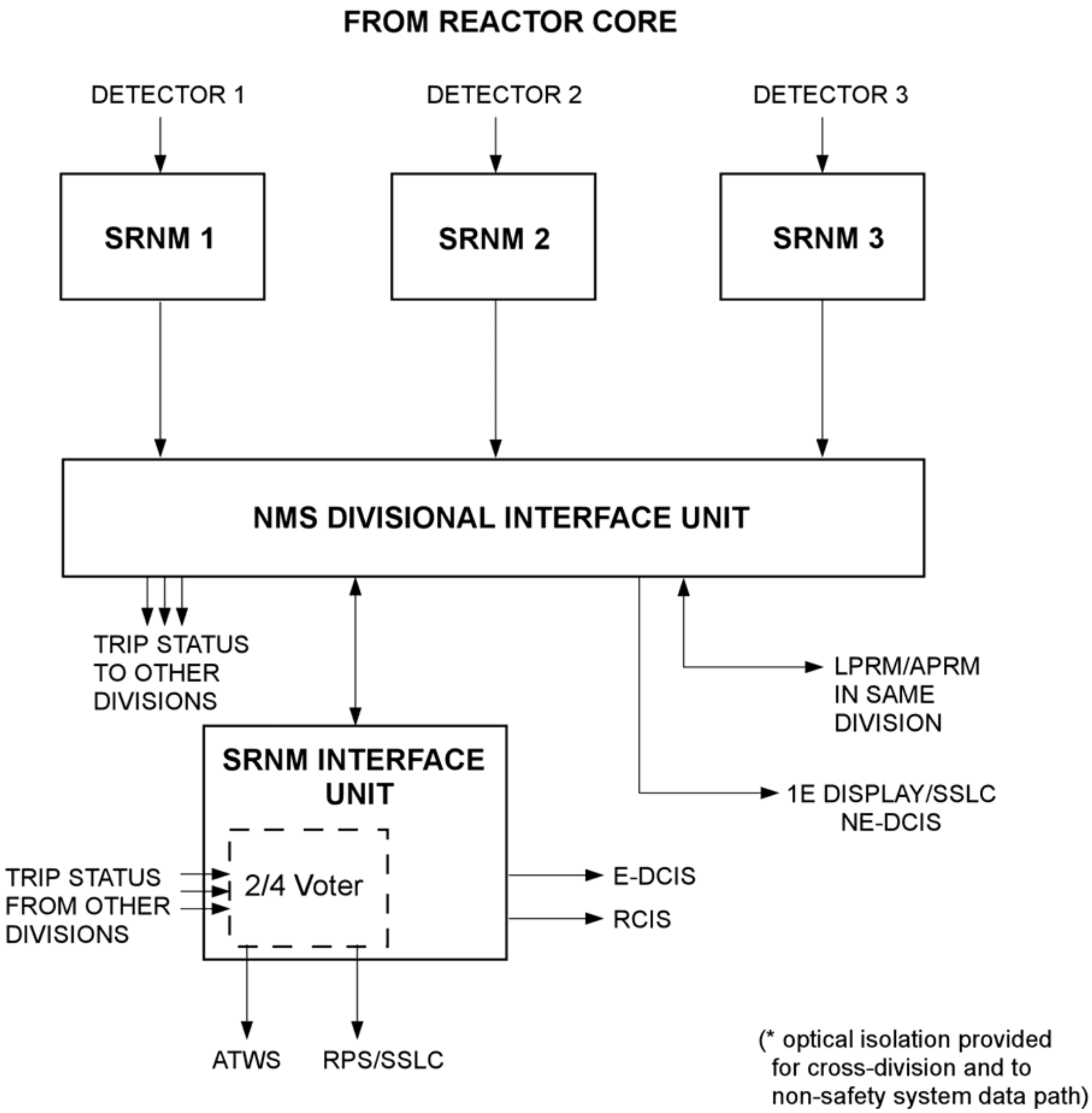


Figure 2.2.5-1. Basic Configuration of a Typical SRNM Division (Subsystem)

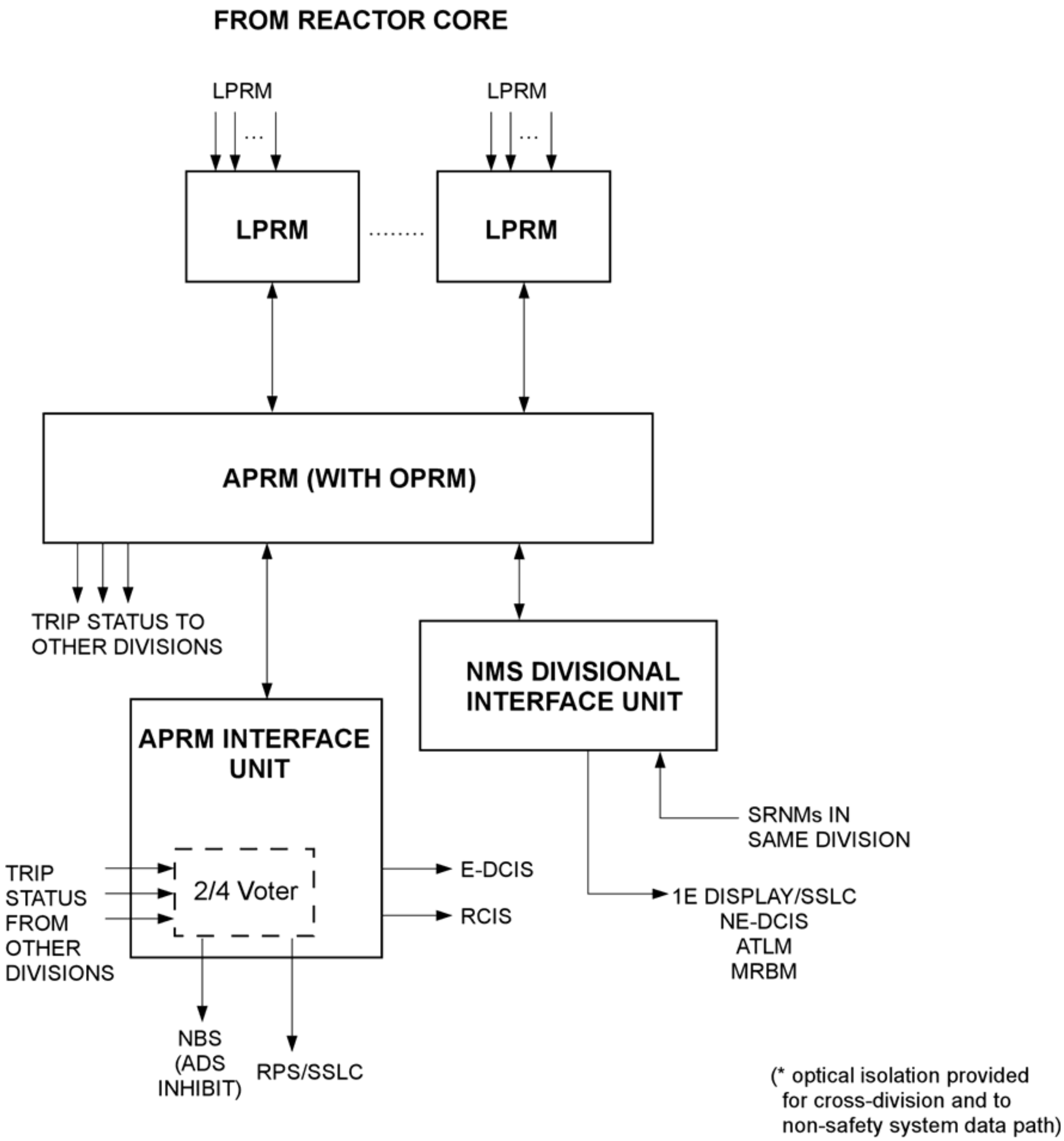


Figure 2.2.5-2. Basic Configuration of a Typical PRNM Division (Subsystem)

2.2.6 Remote Shutdown System

Design Description

The Remote Shutdown System (RSS) provides the means to safely shut down the reactor from outside the main control room (MCR). The RSS provides remote manual control of the systems necessary to: (a) achieve safe (hot) shutdown of the reactor after a scram, (b) achieve subsequent cold shutdown of the reactor, and (c) maintain safe conditions during shutdown. Figure 2.2.6-1 shows the basic system configuration and scope.

The RSS is classified as a safety-related system. The RSS includes control interfaces with both nuclear safety-related and nonsafety-related equipment.

To achieve a safe and orderly plant shutdown from outside the MCR, controls and indicators necessary for operation of the following system and equipment are provided on the remote shutdown panel through the use of safety-related and nonsafety-related touch screen Video Display Units (VDUs).

- Isolation Condenser System (ICS)
- Gravity Driven Cooling System (GDCS)
- Automatic Depressurization System (ADS)
- Standby Liquid Control system
- RWCU/SDC system
- CRD system (makeup function)
- Reactor Component Cooling Water System (RCCWS)
- Plant Service Water System (PSWS)
- Electrical Power Distribution system
- NBS instrumentation
- Reactor Building HVAC

When evacuation of the main control room is necessary, the reactor is manually scrammed by the operators prior to evacuation.

Each of the two RSS divisions is powered from separate independent divisions.

The RSS panels are located in two separate rooms in separate divisional quadrants of the Reactor Building.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.6-1 provides a definition of the visual inspections, tests and/or analyses, together with associated acceptance criteria, which will be undertaken for the RSS.

Table 2.2.6-1

ITAAC For The Remote Shutdown System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The RSS has two divisional panels for monitoring and controlling of the interfacing systems. The panels are physically separated and are located in different divisional areas of the Reactor Building	1. Inspections will be conducted to confirm the appropriate location, isolation, and seismic capabilities of the panels.	1. The panels conform to their requirements for divisional separation and seismic criteria. They are located in separate divisional areas of the Reactor Building.
2. Each RSS panel consists of one safety-related touch screen VDU and one nonsafety-related touch screen VDU.	2. Inspections will be conducted to confirm that each RSS panel has the required operator interface devices.	2. Each RSS panel has at least one safety-related and one nonsafety-related touch screen VDU.

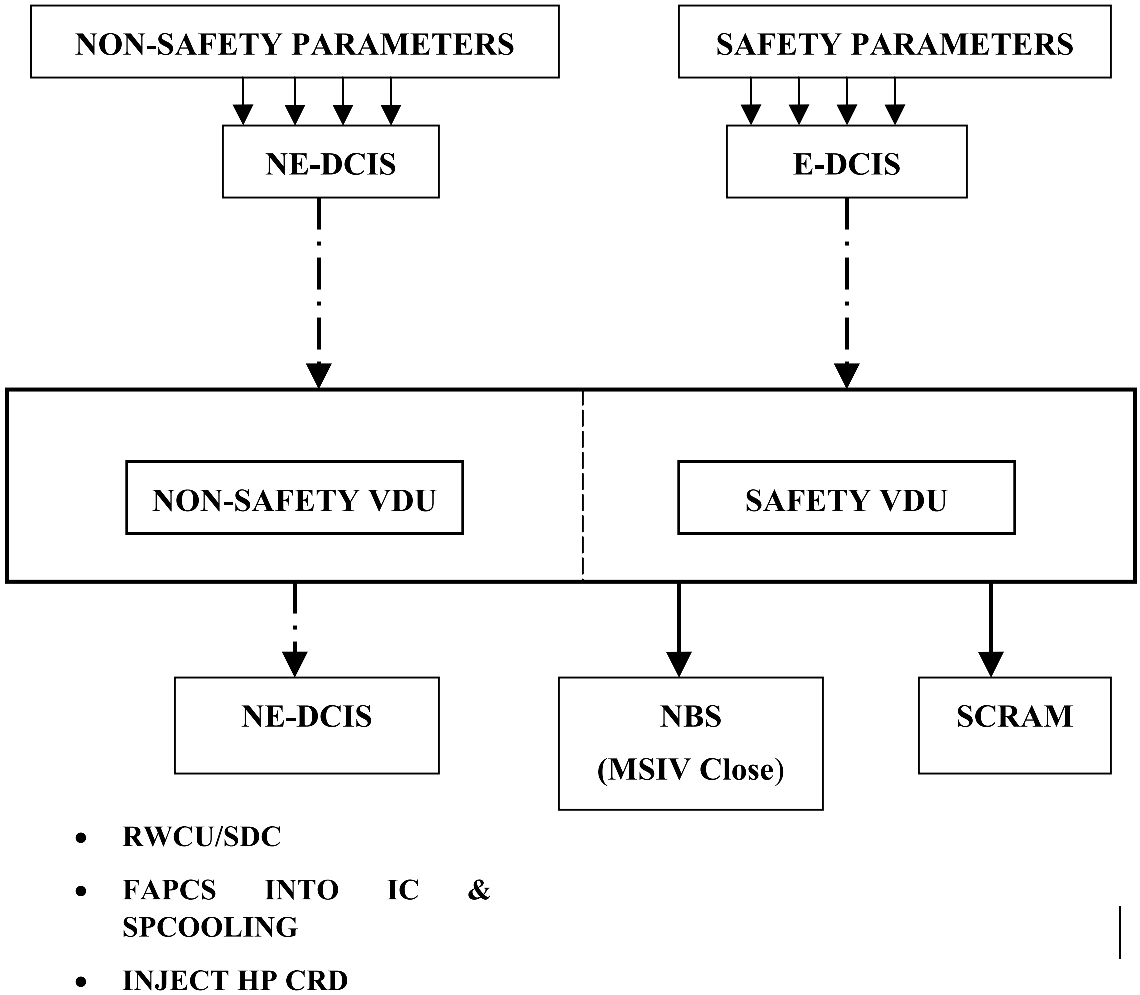


Figure 2.2.6-1. Remote Shutdown System

2.2.7 Reactor Protection System

Design Description

The Reactor Protection System (RPS) initiates an automatic and prompt reactor trip (scram) by means of rapid hydraulic insertion of all control rods whenever selected plant variables exceed preset limits. The primary function is to achieve a reactor shutdown before fuel damage occurs. The RPS also provides reactor status information to other systems, and causes an alarm annunciation in the main control room (MCR) whenever selected plant variables exceed the preset limits.

The RPS is a safety protection system, differing from a reactor control system or a power generation system. The RPS and its components are safety-related. The RPS and the system electrical equipment are classified as safety-related, Seismic Category I and IEEE Class 1E.

The RPS initiates reactor trip signals within individual sensor channels when any one or more of the conditions listed below exists during reactor operation. Reactor scram results if system logic is satisfied.

- High drywell (DW) pressure;
- Turbine Stop Valve (TSV) closure with insufficient turbine bypass capacity being available;
- Turbine Control Valve (TCV) fast closure with insufficient turbine bypass capacity being available;
- Reactor power (neutron flux or simulated thermal power) exceeds limit for operating mode;
- Reactor power rapid increase (short period);
- Reactor vessel pressure high;
- Reactor water level low (Level 3);
- Reactor water level high (Level 8);
- Main steam isolation valves closed (Run mode only);
- CRD HCU accumulator charging header pressure low;
- Suppression pool temperature high;
- Main condenser vacuum low;
- Loss of feedwater flow;
- Operator-initiated manual scram; or
- Reactor mode switch in “Shutdown” position.

The RPS is divided into four redundant divisions of sensor channels, trip logics, and trip actuators, and two divisions of manual scram controls and logic circuitry that initiates the rapid insertion of control rods by hydraulic force to scram the reactor when potentially unsafe conditions are detected. Figure 2.2.7-1 shows the RPS functional block diagram and the signal

flow paths from sensors to scram pilot valve solenoids. Each division has a separate IEEE Class 1E power supply taken from the safety-related Uninterruptible Power Supply (UPS) 120 Vital AC (VAC) power supply. The automatic and manual scram initiation logic systems are independent of each other. The manual scram uses two independent manual trip channels to initiate a reactor scram. The RPS design is such that, once a full reactor scram has been initiated automatically or manually, this scram condition seals-in such that the intended fast insertion of control rods into the reactor core continues to completion. After a time delay, the design requires operator action to reset the scram logic to the untripped state.

The RPS scram logic circuits are arranged so that coincident trips in two of the four divisions (2-out-of-4 logic) of sensor channels and in two of the four trip system outputs to the actuating devices are required to initiate a scram. This arrangement permits a single failure in one division to occur without either causing a scram or preventing the other three divisions from causing a scram. For example, the single failure may be in either system logic or the individual power supply for that division.

Each logic division and its associated power supply is separated both physically and electrically from the other divisions. This arrangement permits one division at a time to be taken out of service (bypassed) for testing or repair during reactor operation. The other divisions then perform the RPS function with system logic in a 2-out-of-3 arrangement.

The RPS has the following basic display and control functions (displays in microprocessor-based display units).

Process parameters displays:

Reactor vessel pressure;

Reactor water levels;

Primary containment drywell pressures;

CRD Hydraulic Contorl Unit (HCU) accumulator charging header pressures;

Suppression pool (local or bulk) temperatures; and

Power Generation Bus voltages.

Status alarms:

RPS NMS trip (generated in NMS);

Reactor vessel pressure high;

Reactor water level low (<= Level 3);

Reactor water level high (>= Level 8);

Containment (drywell) pressure high;

MSIV closure trip;

TSV closure;

TCV fast closure;

Main condenser vacuum pressure high

Loss of Power Generation Bus (Loss of Feedwater Flow)

CRD HCU accumulator-charging-header-pressure low;

Suppression pool temperature high;

RPS divisional automatic trip (auto-scram) (each of the four, i.e., Div. I, II, III, IV automatic trip);

RPS divisional manual trip (each of the four, i.e., Div. I, II, III, IV manual trip);

Manual scram trip (two: both Manual A and/or Manual B);

Mode switch in SHUTDOWN;

SHUTDOWN mode trip bypassed;

NON-COINCIDENT NMS trip mode in effect (in NMS);

NMS trip mode selection switch still in NON-COINCIDENT position with plant in RUN mode (in NMS);

Division of channel A (or B, C, D) sensors bypassed (four);

Tripped conditions in Division I (or II, III, IV) and Division I (or II, III, IV) sensors bypassed (four);

Division I (or II, III, IV) Trip Logic Unit (TLU) out-of-service bypass (four);

Bypass of CRD accumulator-charging-header-pressure low trip;

Any CRD accumulator-charging-header trip, bypass switch in BYPASS (STARTUP or RUN); and

Auto-scram test switch in TEST mode (manual trip of automatic logic) (four).

Bypasses:

The turbine stop valve closure trip bypass and control valve fast closure trip bypass;

Bypass of scram trip for CRD-accumulator-charging-header low pressure after scram;

Bypass of scram trip for main steam isolation valve closure;

Bypass of scram trip for Loss of Power Generation Bus;

Bypass of scram trip on account of mode switch in SHUTDOWN position;

Bypass of NMS SRNM trip functions in RUN mode;

Bypass of non-coincident NMS trips in RUN mode

Maintenance bypass of detector inputs (division-of-channel-sensors bypass); and

RPS trip system output logic maintenance bypass TLU output bypass (Division-out-of-service bypass).

Manual Controls:

Initiation of scram by manual scram switches;

Mode switch operation (results in scram if placed in the SHUTDOWN position);

Reset of automatic trip systems after trip input signals clear;

Reset of manual trip systems (preferably after reset of the automatic trip systems);

Manual bypasses for conditions that are specifically permitted; and

Manual initiation of selected trip systems or trip actuators using trip logic test switches.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.7-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be performed for the RPS.

Table 2.2.7-1

ITAAC For The Reactor Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. RPS logic design is fail-safe such that, loss of electrical power to a division of RPS logic, results in the load drivers (LDs) of that division to change their states from "close" to "open.”	1. Tests will be conducted by disconnecting electrical power to a division of RPS at a time.	1. Design commitment is met.
2. RPS logic design is single-failure-proof such that the failure of one division of RPS only causes the LDs of that division to change their states from "close" to "open" without causing the LDs of other divisions to change their states.	2. Tests will be conducted by disconnecting electrical power to a division of RPS and checking the "open" "close" states of the failed division LDs and other divisions LDs.	2. Design commitment is met.
3. RPS automatic scram logic is designed to avoid inadvertent scrams by requiring coincident trip of at least two divisions of RPS to cause the change of the states of all RPS LDs from "close" to "open.”	3. Tests will be conducted using simulated inputs to cause a trip condition in only one division of RPS at a time and checking the status of all RPS LDs.	3. Only the LDs of the RPS division in trip condition are in "open" state and the LDs of other divisions are in "close" state.
4. RPS logic uses four redundant sensor monitoring and trip channels for its automatic scram function. All RPS LDs change their state from "close" to "open" when any two out of four automatic scram channels have tripped.	4. Tests will be conducted using simulated inputs to cause trip conditions in two, three, and four channels of RPS.	4. In all test conditions, all RPS LDs change their states from "close" to "open.”

Table 2.2.7-1

ITAAC For The Reactor Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. RPS manual scram function is designed to require both manual scram push-buttons A & B to be depressed to interrupt Class 1E ac power to all A & B scram solenoids.	5. Tests will be conducted by depressing the A scram push-button and the B scram push-button.	5. When manual scram push-button A is depressed Div I Class 1E ac power is interrupted. When scram push-button B is depressed Div II Class 1E ac power is interrupted.
6. The RPS logic design incorporates channel bypass provisions for on-line test and repair during plant operation. When a channel of RPS is placed in by-pass condition, the RPS logic goes to two-out- of-three channel trip for changing the states of all RPS LDs from "close" to "open.”	6. While a channel is placed in bypass, simulated input signals, that cause trip conditions, are provided to un-bypassed channels of RPS.	6. All RPS LDs change state from "close" to "open" when at least two out of three un-bypassed channels have tripped.
7. RPS logic is designed to ensure scram completion by requiring deliberate operator action and inhibiting the reset of scram circuitry for a time period of 10 seconds after scram initiation.	7. Tests will be conducted by attempting to reset RPS scram circuitry during the10 seconds time period after scram initiation.	7. Attempts to reset RPS scram circuitry during the 10 seconds time period after scram initiation do not result in reset.
8. RPS logic is designed to supply dc electrical power to the solenoids of scram air header dump valves (back-up scram valves), and provide scram follow signals to RC&IS following automatic or manual scram.	8. Test will be conducted by both inputting simulated scram causing signals to RPS automatic scram channels and depressing manual scram push-buttons and then checking the status of RPS back- up scram LDs (or relay contacts) and RC&IS relay contacts (scram follow signals).	8. RPS back-up scram LDs states are changed from "open" to "close" (if relays are used the back-up scram relays are in "energized" state), and scram-follow relays are also in the "energized" state after automatic or manual scram.

Table 2.2.7-1

ITAAC For The Reactor Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. Control room indications, alarms, and controls are provided for RPS defined in Subsection 2.2.7.	9. Inspection will be performed on the control room indications, alarms, and controls for the RPS.	9. Indications, alarms, and controls exist or can be retrieved in control room as defined in Subsection 2.2.7.

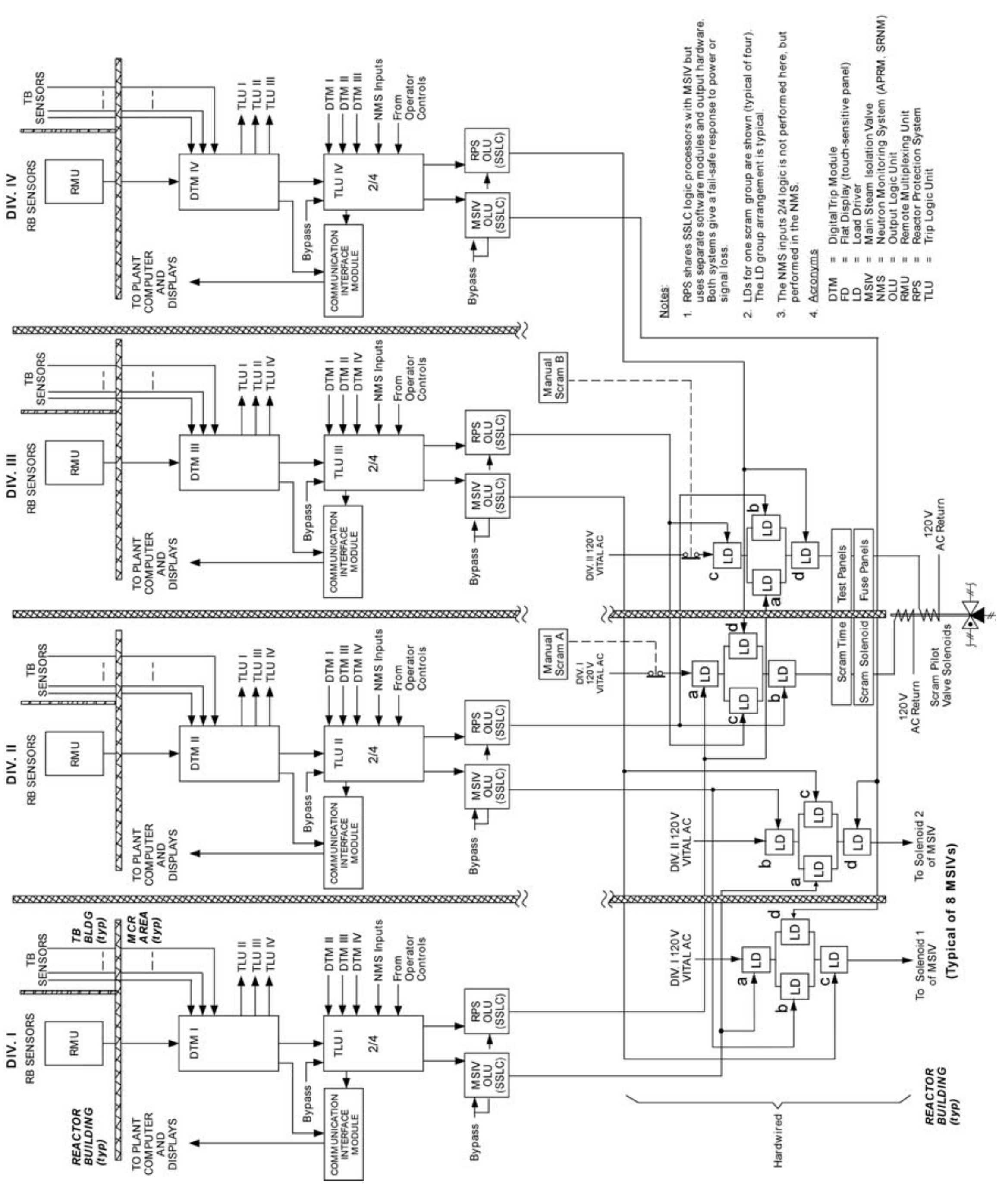


Figure 2.2.7-1. Reactor Protection System Simplified Functional Block Diagram

2.2.8 Plant Automation System

Design Description

The Plant Automation System (PAS) is classified as a power generation system, is not required for safety, and thus, is classified as nonsafety-related. Events requiring control rod scram are sensed and controlled by the safety-related RPS, which is completely independent of the Plant Automation System.

The PAS system provides the capability for supervisory control of the entire plant by supplying set-point commands to independent nonsafety-related automatic control systems as changing load demands and plant conditions dictate.

PAS controls the overall plant startup, power operation, and shutdown functions under operator break-point control. PAS receives input from the Neutron Monitoring System, the NE-DCIS, the Steam Bypass and Pressure Control system, and the operator's control console. The output demand signals from PAS are sent to the RC&IS to position the control rods, and to the Steam Bypass and Pressure Control system for automatic load following operations.

The PAS operates in either manual or automatic control mode. The system control logic is performed by redundant, digital controllers. The digital controller receives inputs from interfacing system via the Non-Essential Distributed Control & Information System (NE-DCIS). It performs power control calculations and provides system outputs to the NE-DCIS.

The PAS digital controllers are located in the Control Building. A simplified functional block diagram of the PAS is provided in Figure 2.2.8-1.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

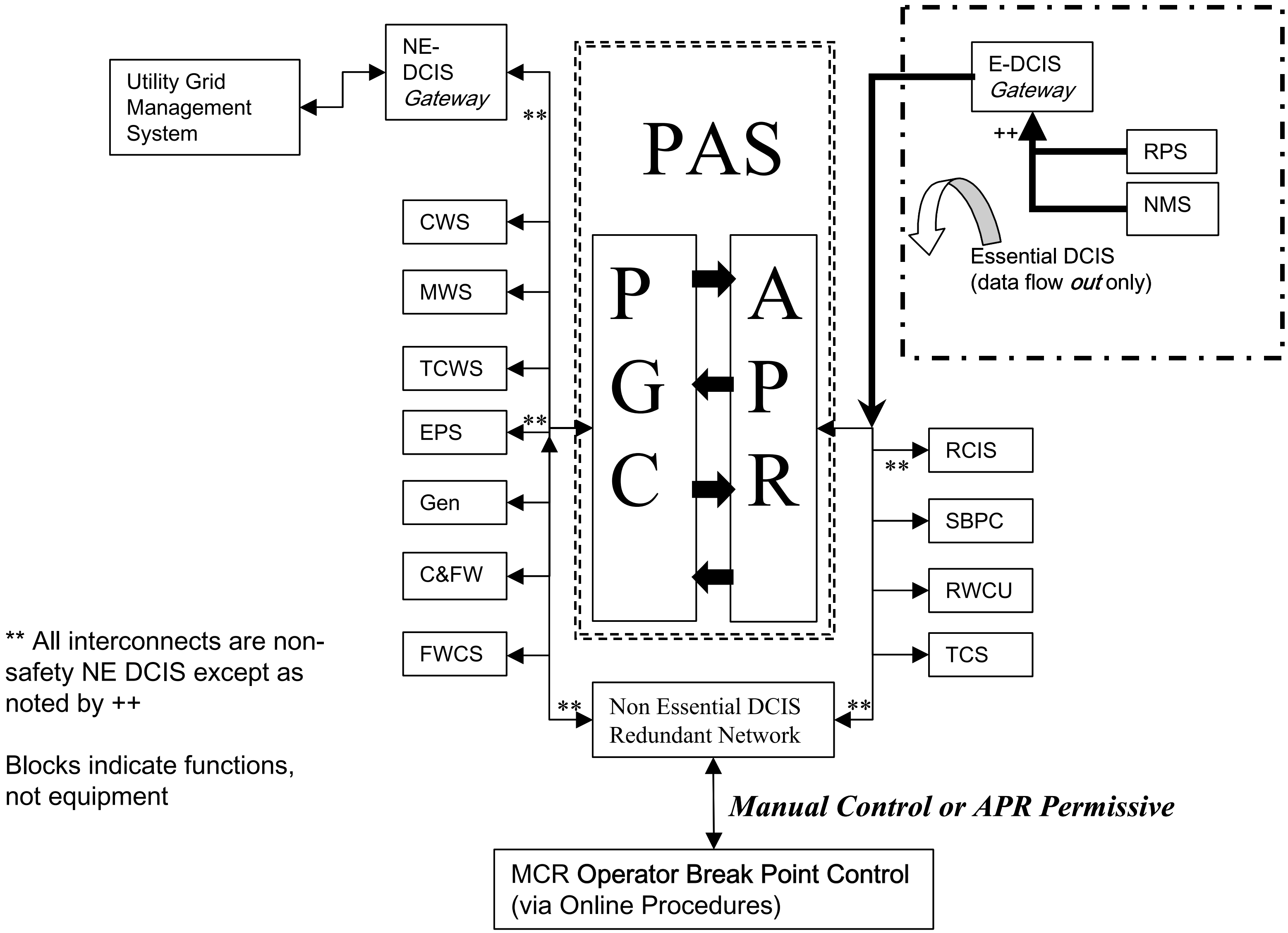


Figure 2.2.8-1. Plant Automation System

2.2.9 Steam Bypass and Pressure Control System

Design Description

The Steam Bypass and Pressure Control (SB&PC) system controls the reactor pressure during reactor startup, power generation, and reactor shutdown by control of the turbine bypass valves and signals to the Turbine Control System, which controls the turbine control valves. The SB&PC system consists of redundant digital controllers and has the interfaces shown in the SB&PC Control Interface Simplified Block Diagram on Figure 2.2.9-1.

The SB&PC system does not perform or ensure any safety-related function, and thus, is classified as nonsafety-related. Basic functions are shown on Figure 2.2.9-2, SB&PC Simplified Functional Block Diagram.

The SB&PC system operates in either manual or automatic control modes. The system control calculations and logic are performed by redundant digital controllers.

The SB&PC system digital controllers are located in the Control Building.

Controls and Instruments

Essential controls and instruments are available on the displays in the Main Control Room (MCR) for the operator.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.9-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the SB&PC system.

Table 2.2.9-1

ITAAC For The Steam Bypass and Pressure Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Each Fault Tolerant Digital Controller (FTDC) is equipped with on-line diagnostic capabilities for identifying failure of input/output signals, busses, power supplies, processors and interprocessor communications. These on-line diagnostics can be performed without interrupting the control operation of the SB&PC System.	1. Perform an on-line diagnostics based on the parameters shown in the design description (Subsection 2.2.9).	1. The on-line diagnostics identifies failure of input/output signals, busses, power supplies, processors and interprocessor communications without interrupting the SB&PC system operations.
2. The system incorporates redundant control channels.	2. The system will be tested by simulating failure of any one operating controller.	2. The system continues to function during loss of any one operating controller.
3. The system is powered by redundant uninterruptible power supplies.	3. Loss of one power supply will demonstrate no loss of functions of SB&PC system.	3. There is no loss of SB&PC functions upon loss of any one power supply.

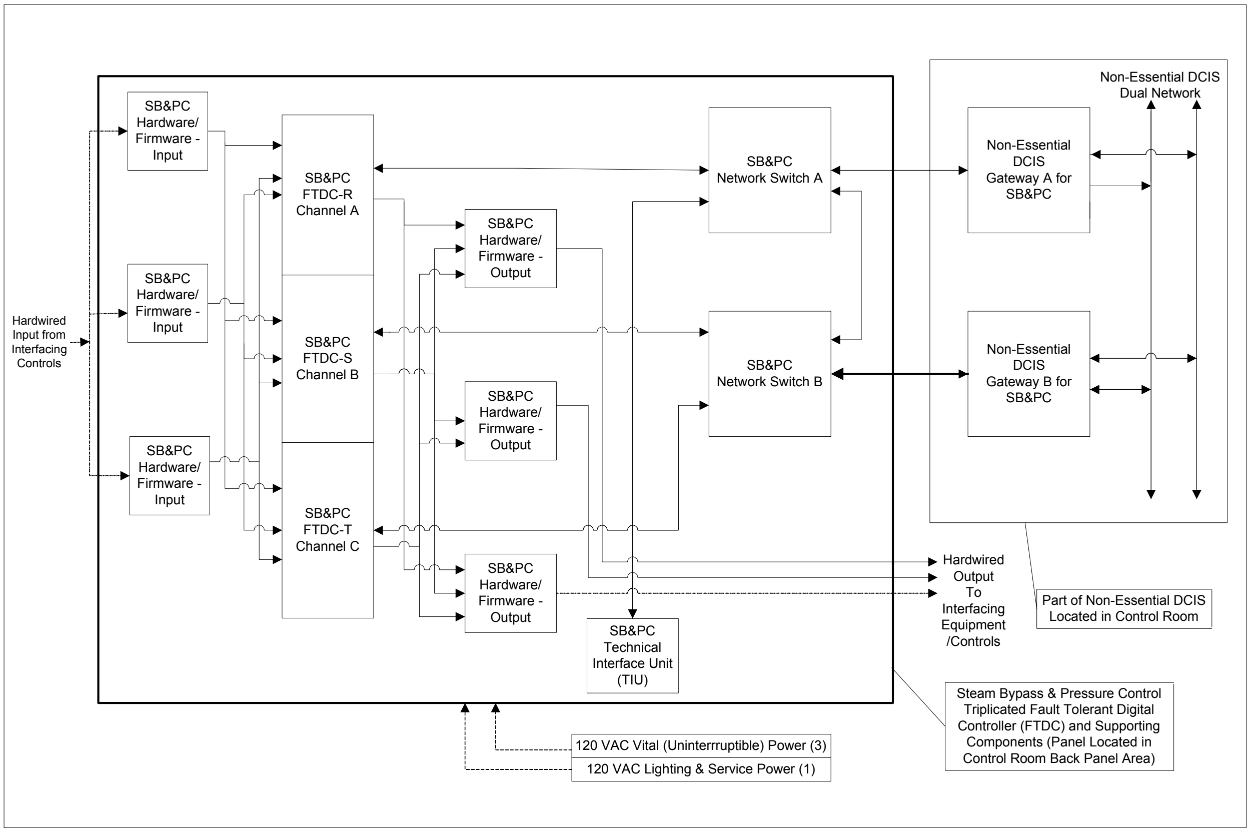


Figure 2.2.9-1. SB&PC Control Interface Simplified Block Diagram

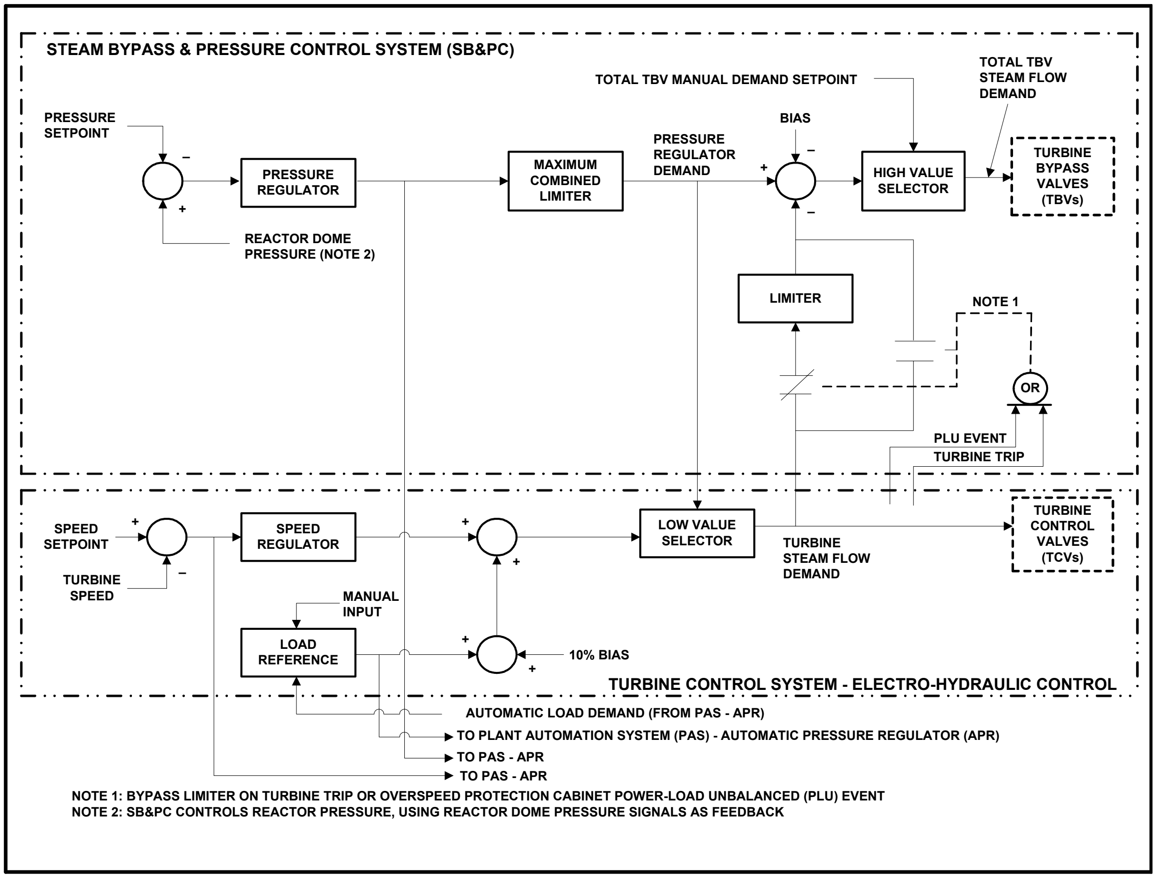


Figure 2.2.9-2. SB&PC Simplified Functional Block Diagram

2.2.10 Essential Distributed Control and Information System

Design Description

The design basis functions of the Essential Distributed Control and Information System (E-DCIS) are to:

- Read signals from the safety-related instrumentation via remote multiplexing units (RMUs);
- Perform the required signal conditioning if this function is required; digitize and format the input signals into messages for transmission on the E-DCIS data network or data path; and
- Transmit the data signals and commands onto the E-DCIS network or data path for interface with other safety related systems and main control room Video Display Units (VDUs). Also transmit the actuation signals to safety-related equipment as output from the RMUs.

It also transmits plant parameters and other safety-related system data through isolation devices to the gateways that lead to the NE-DCIS that provide interfaces to nonsafety-related system logic and displays for power generation.

E-DCIS provides redundant and distributed I&C communications network that includes electrical devices and circuitry that connect to field sensors, power supplies, and actuators, which are part of safety-related systems. It replaces most of conventional, long-length, copper-conductor cables with a dual-redundant, fiber optic, data network. The interconnections among divisions of the E-DCIS are provided by isolated digital interfaces. For the E-DCIS data communication, the system timing for each division is asynchronous with respect to other divisions.

The local E-DCIS RMU performs signal conditioning and A/D signal conversion for continuous process signals, and performs signal conditioning. The RMU function can be applied for performing both input and output signal processing functions. The RMU formats the acquired signals into data messages and transmits the data via the dual redundant data path network to the various SSLC and safety-related systems components for logic processing. The RMU designated as logic output processor then receives such trip command and control signals from SSLC and safety-related system components via the network. It then provides terminal points for distributing the signals to the final actuating devices of the safety systems.

Operator interfaces for control and display is realized through the visual display unit, which is also connected to the network. An illustration of a typical division of E-DCIS that includes network and data paths interfacing with the associated SSLC and safety-related system components is shown in Figure 2.2.10-1.

E-DCIS contains continuous online-diagnostic functions that monitor transmission path quality and integrity. The dual redundant data communication channels are repairable on-line if one channel fails. E-DCIS failures are alarmed in the MCR. Periodic surveillance, using off-line tests with simulated input signals, is used to verify the overall system integrity. Any E-DCIS control equipment software, such as that in the RMU, is implemented based on read only memory that cannot be modified by plant personnel. E-DCIS power is supplied by the divisional

120 VAC safety-related power supply systems. Loss of power causes a controlled transition to a safe-state without transients occurring that could cause inadvertent initiation or shutdown of driven equipment.

E-DCIS equipment is classified as safety-related, Class 1E, and Seismic Category I.

The E-DCIS includes test facilities in the MCR that monitor data transmission to ensure that data transport, routing, and timing specifications are accurate.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.10-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the E-DCIS.

Table 2.2.10-1

ITAAC For The Essential Distributed Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Four divisions of independent and redundant E-DCIS acquire and transmit the safety-related sensor inputs and data to the safety-related I&C systems.	1. Inspection of the installed equipment will confirm the identity and location of E-DCIS instrumentation, equipment panels, and their interconnections.	1. E-DCIS configuration is in accordance with equipment arrangement shown in the SSLC ITAAC and in Figure 2.2.10-1. The figures indicate the required relationship of E-DCIS to other safety-related system processing equipment.
2. The four divisions of redundant E-DCIS are physically and electrically separated from each other. There are no interconnections among divisions of E-DCIS except through isolation devices. Data communications to the plant computer or display controllers uses an isolation device such as fiber optic cables.	2. Inspections of fabrication and installation records and construction drawings or visual field inspections of the installed E-DCIS equipment will be used to confirm electrical and physical separation.	2. The installed E-DCIS equipment conforms to certified commitment.
3. The RMUs and other controllers in each E-DCIS division are powered independently from the divisional 120 VAC UPS power sources.	3. System tests will be conducted after installation to confirm the electrical power supply configurations.	3. The installed instrument channels are operational with the power sources specified in the certified commitment.

Table 2.2.10-1

ITAAC For The Essential Distributed Control and Information System

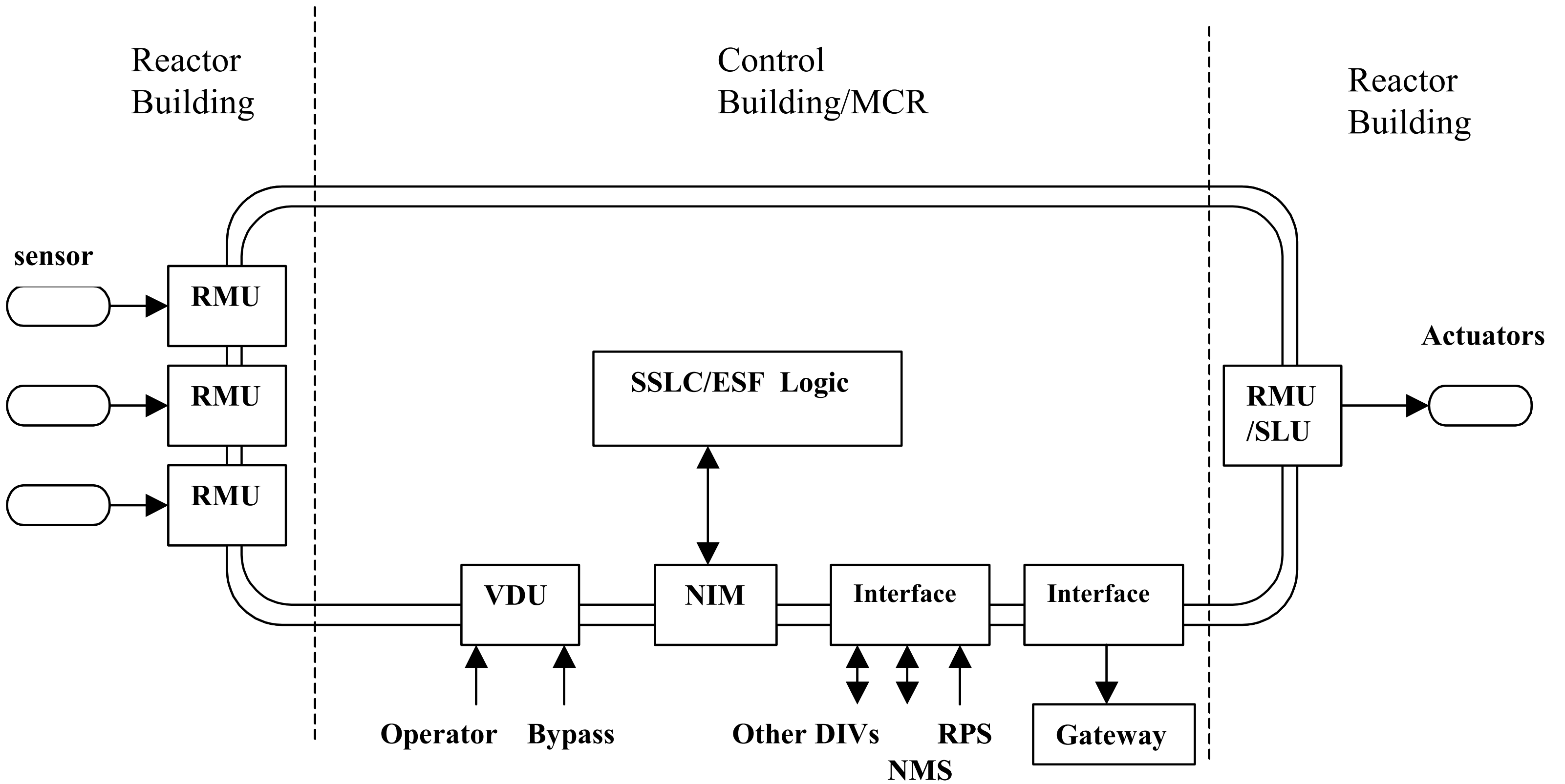
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. E-DCIS includes test facilities that will monitor data transmission to ensure that data transport, routing, and timing are accurate.	4. Tests will be conducted on the installed E-DCIS equipment. These tests will confirm the basic functionality of each component. The tests will include simulation of typical input parameters and monitoring of the received transmitted parameters.	4. Operability of the installed E-DCIS equipment satisfies the following conditions (for each division): a. Monitored output signals match simulated input signals for accuracy of signal conversion and transmission time. b. Simulated data errors are detected and annunciated to operator.
5. Loss of power causes a controlled transition to a safe state without transients occurring that could cause inadvertent initiation or shutdown of driven equipment.	5. Tests will be conducted to monitor the degradation of E-DCIS system outputs upon momentary or long-term loss of divisional power or power to individual E-DCIS components.	5. a. Loss of one division of power does not cause false output trip or inadvertent initiation of final system actuators. Loss of power and loss of divisional trip signals are annunciated. b. Loss of power to individual component produces a safe-state output condition without false outputs {normally energized outputs de- energize, normally de-energized outputs remain de-energized}. c. Restart (initialization) of component or system upon recovery of power does not cause inadvertent output action (outputs remain in safe-state condition until sensed inputs are evaluated in processing circuitry). d. Transient power loss (e.g., < 1 second) causes no false output trip or inadvertent initiation of final system actuators or removal of previous tripped state.

Table 2.2.10-1

ITAAC For The Essential Distributed Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. E-DCIS is fault-tolerant in each division and provides capability for automatically reconfiguring after failure of an RMU or interconnecting cable.	6. Faults will be simulated and the response monitored. These tests will verify that a single failure of a multiplexing component will not impair total system function.	6. a. A single cable break does not affect network operation. b. Loss of one RMU removes that unit from service; network continues normal operation. c. Fault occurrence and notice of reconfiguration is displayed to operator.

(Typical Division)



RMU: Remote Multiplexing Unit (multiple units for sensor inputs)

NIM: Network Interface Module Function

SLU: System Logic Unit (Output to actuators)

VDU: Visual Display Unit (operator display/control interface)

E-DCIS dedicated communication paths in RPS, NMS and other safety-related systems not shown.

Figure 2.2.10-1. E-DCIS with SSLC (ESF) Components

2.2.11 Non-Essential Distributed Control and Information System

Design Description

The nonsafety-related Non-Essential Distributed Control and Information System (NE-DCIS) is the data communication method for all control systems, and certain individual control functions, that are not part of safety-related control systems. The NE-DCIS equipment is based upon fiber optics communications technology and computer controls. The system transfers data between control system equipment and the main control room. The NE-DCIS also includes network gateways, which allow the transfer of data between discrete data highway systems. All interconnections use fiber optic data links. An illustration of a typical NE-DCIS network and associated components and links are shown in Figure 2.2.11-1, Instrumentation and Control Simplified Block Diagram.

NE-DCIS has no safety-related function.

The Plant Computer Function (PCF), which is part of NE-DCIS, provides nuclear steam supply (NSS) performance and prediction calculations, video display control, point log and alarm processing and balance-of-plant (BOP) performance calculations.

The calculations performed by PCF include process validation and conversion, combination of points, nuclear system supply performance calculations, and balance-of-plant performance calculations. PCF also performs the functions and calculations defined as being necessary for the effective evaluation of nuclear power plant operation and provides a permanent record of plant operating activities.

In addition, the plant computer also has access to all parameters monitored by the safety-related systems and can be used to re-create the plant response to transients and accidents.

Controls and Instruments

PCF performance monitoring data is available in the Main Control Room (MCR), Technical Support Center (TSC) and Emergency Operations Facility (EOF).

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.11-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the NE-DCIS system.

Table 2.2.11-1

ITAAC For The Non-Essential Distributed Control and Information System (NE-DCIS)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. On-line diagnostics can be performed on any one channel of the redundant network without interrupting the control operation of plant systems.	1. Perform on-line diagnostics on any one channel of the redundant network to show that diagnostics can be preformed without interrupting the control operation of plant systems.	1. On-line diagnostics of any one channel of redundant network does not interrupt control operation of plant systems.
2. The system is powered by redundant uninterruptible power supplies.	2. Loss of one power supply will demonstrate no loss of functions of NE-DCIS.	2. There is no loss of NE-DCIS functions upon loss of any one power supply.

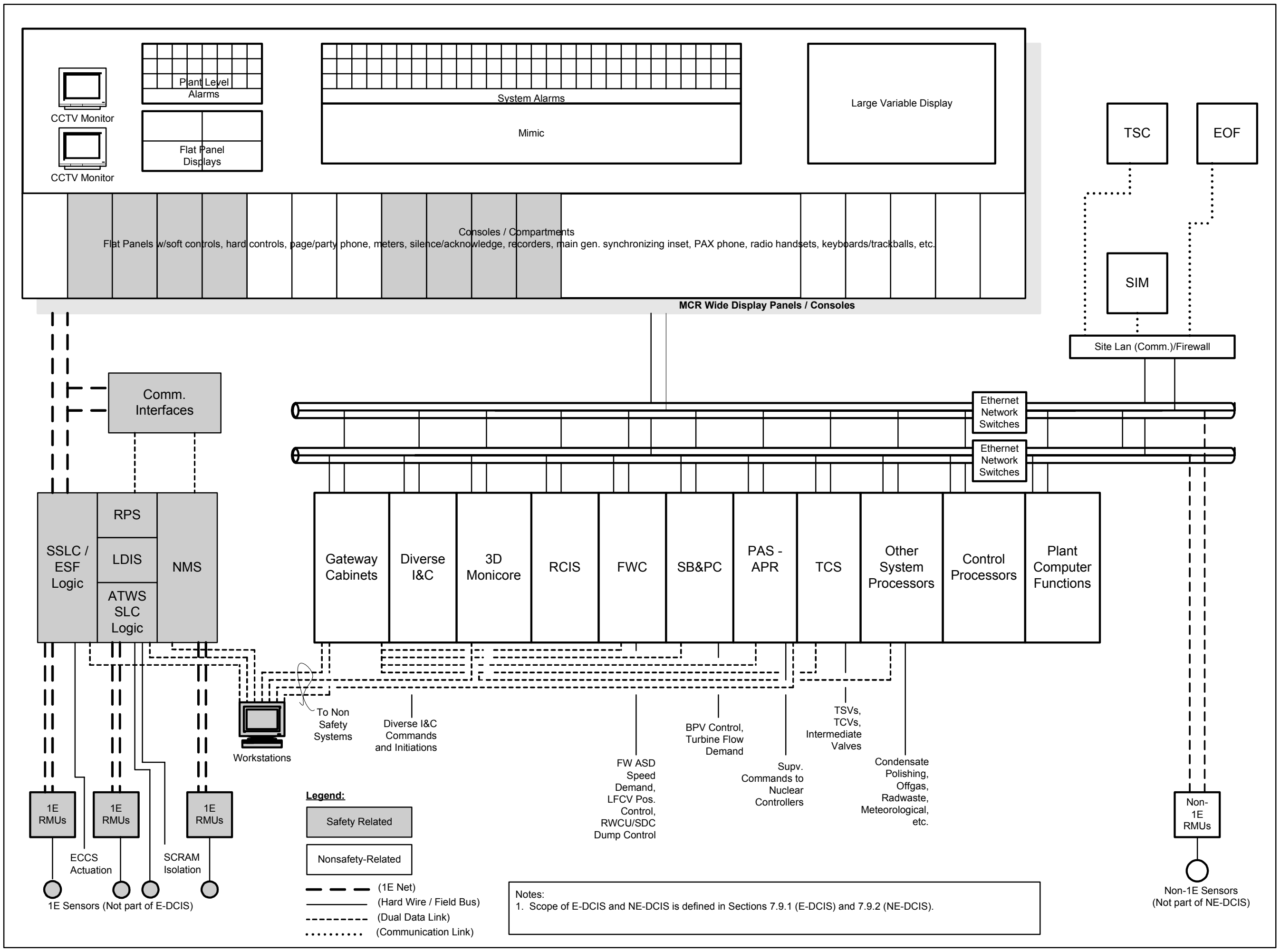


Figure 2.2.11-1. Instrumentation & Control Simplified Block Diagram

2.2.12 Leak Detection and Isolation System

Design Description

The Leak Detection and Isolation System (LD&IS) detects and monitors leakage from the containment, preventing the release of radiological leakage from the reactor coolant boundary to the environment. The system initiates safety isolation functions by closure of inboard and outboard containment isolation valves. The LD&IS interfaces are shown in Figure 2.2.11-1.

The following functions are supported by the LD&IS:

- Containment isolation following a LOCA event;
- Main steamline isolation;
- Isolation Condenser System process lines isolation;
- RWCU/SDC system process lines isolation;
- Fuel and Auxiliary Pools Cooling system process lines isolation;
- Chilled Water System lines to DW coolers isolation;
- Drywell sumps liquid drain lines isolation;
- Containment purge and vent lines isolation;
- Reactor Building HVAC air exhaust ducts isolation;
- Fission products sampling line isolation;
- Monitoring of identified and unidentified leakages in the drywell;
- Monitoring of condensate flow from the drywell air coolers; and
- Monitoring of the vessel head flange seal leakage.

The LD&IS monitors plant parameters such as flow, temperature, pressure, water level, etc., which are used to alarm and initiate the isolation functions. The LD&IS transfers the Table 2.2.12-1 signals to electronic processors for use in isolation logic, alarms and indication.

At least two parameters are monitored for an isolation function. The signal parameters are processed by the Safety System and Logic Control (SSLC) system, which generates the trip signals for initiation of isolation functions.

The LD&IS safety-related functions have four divisional channels of sensors for each parameter. Two-out-of-four coincidence voting within a channel is required for initiation of the isolation function. The control and decision logic are of fail-safe design, which ensures isolation on loss of power. The logic is energized at all times and de-energizes to trip for isolation functions.

Loss of one divisional power or one monitoring channel does not cause inadvertent isolation of the containment. Different divisional isolation signals are provided to the inboard and outboard isolation valves.

The LD&IS allows periodic testing of each channel to verify it is capable to perform the intended function.

The safety-related portions of the LD&IS are Seismic Category I.

The LD&IS initiates isolation functions automatically. All isolation valves have individual manual control switches and valve position indication in the MCR. However, the isolation signal overrides any manual control to close the isolation valves.

Manual control switches in the control logic provide a backup to automatic initiation of isolation as well as capability for reset, bypass and test of functions.

The monitored plant parameters are measured and recorded by the NE-DCIS, and are displayed on demand. The abnormal indications and initiated isolation functions are alarmed in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.12-2 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the LD&IS.

Table 2.2.12-1

LD&IS Interfacing Sensor Parameters

Temperatures:

- MSL Tunnel Area Temperature
- Drywell Temperature
- RWCU/SDC Valve Room Temperature
- MSL Temperature in Turbine Building
- Isolation Condenser Area Room Temperature
- RWCU/SDC System Temperature

Pressures:

- MSL Turbine Inlet Low Pressure
- Main Condenser Low Vacuum
- Reactor Vessel Head Flange Seal Pressure Leakage
- Drywell Pressure

Radiation Levels:

- RCCWS Intersystem Leakage
- Drywell Fission Product
- Reactor Building HVAC Exhaust
- Refueling Handling Area Air Exhaust
- Drywell Sump Low Conductivity Waste (LCW) Drain Line to Radwaste
- Drywell Sump High Conductivity Waste (HCW) Drain Line to Radwaste
- Isolation Condensers Vent Discharge

Flows:

- MSL High Flow
- RWCU/SDC High Flow (Temperature Compensated)
- Drywell Air Cooler Condensate Flow
- Isolation Condenser Steam Line Flow
- Isolation Condenser Condensate Return Line Flow

Levels:

- Various RPV Water Levels
- Drywell and Containment Sump Levels

Table 2.2.12-2

ITAAC For Leak Detection and Isolation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the LD&IS is defined in Subsection 2.2.12.	1 Inspection of the as built system will be conducted.	1. The as built LD&IS system conforms with the description in Subsection 2.2.12.
2. LD&IS monitors and detects leakages from the RCPB, and initiates closure of primary and secondary containment isolation valves.	2. Each LD&IS instrument channel shall be tested using simulated signal inputs to test the trip conditions.	2. Each channel trips.
3. The LD&IS isolation logic uses four redundant instrument channels to monitor each RCPB leakage parameter. The isolation signal is initiated when any two-out-of-four channels have tripped.	3. The instrument channels of each LD&IS isolation function shall be tested using simulated signal inputs.	3. Isolation signal is initiated when at least any two-out-of-four channels have tripped.
4. The LD&IS isolation logic incorporates channel bypass provisions for on line testing and repair. In this mode, the isolation signal is initiated when any two out of three channels have tripped.	4. In channel bypass mode, each LD&IS logic isolation function shall be tested using simulated signal inputs.	4. Isolation signal is initiated when at least any two out of three channels have tripped.
5. Each MSIV can be subjected to a partial closure test from the control room.	5. Actuate each MSIV test switch to check partial closure of the valve.	5. Each MSIV partially closes and then reopens automatically when its test switch is actuated.

Table 2.2.12-2

ITAAC For Leak Detection and Isolation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. LD&IS provides separate manual controls in the control room for MSIV closure, and for isolation of the containment and each RWCU/SDC division.	6. a Simultaneously actuate two of the four MSIV isolation switches (Div. 1 and 4 or Div. 2 and 3) to close all the MSIVs. Repeat the same test by actuating the other two MSIV isolation switches. b. Actuate each containment isolation switch (Div. 1 and 2) to isolate the containment. c. Actuate each RWCU/SDC isolation switch.	6. a. Closure of all the MSIVs occurs only when Divisions 1 and 4 or 2and 3 switches are actuated. b. Each divisional containment isolation switch closes only its respective containment isolation valves. c. Each RWCU/SDC isolation switch closes its respective isolation valves.
7. Manual reset controls are provided to perform reset functions as described in Subsection 2.2.12.	7. Tests will be performed using the LD&IS reset functions in the control room.	7. The logic circuitry resets for normal operation.
8. Control room indications and controls for this system are defined in Subsection 2.2.12.	8. Inspections will be performed on the control room indications and controls for this system.	8. Controls exist and indications exist or can be retrieved in the control room as defined in Subsection 2.2.12.
9. LD&IS logic design is fail-safe, such that loss of electrical power to one LD&IS divisional logic channel initiates a channel trip.	9. Tests will be conducted to simulate electrical power failure to each divisional LD&IS channel.	9. The faulted channel trips.
10. The divisional LD&IS logic channels and associated sensors are powered from Class IE divisional power.	10. Tests will be performed on the LD&IS system by providing a test signal in only one Class IE division at a time.	10. The test signal exists only in the Class IE division of the LD&IS system under test.

Table 2.2.12-2

ITAAC For Leak Detection and Isolation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. Independence is provided in the system between Class IE divisions, and between Class IE divisions and non-Class IE equipment.	11. Inspection of the installed LD&IS Class IE divisions will be performed.	11. Physical separation exists in LD&IS between Class IE divisions, and between the Class IE divisions and the non-Class IE equipment.

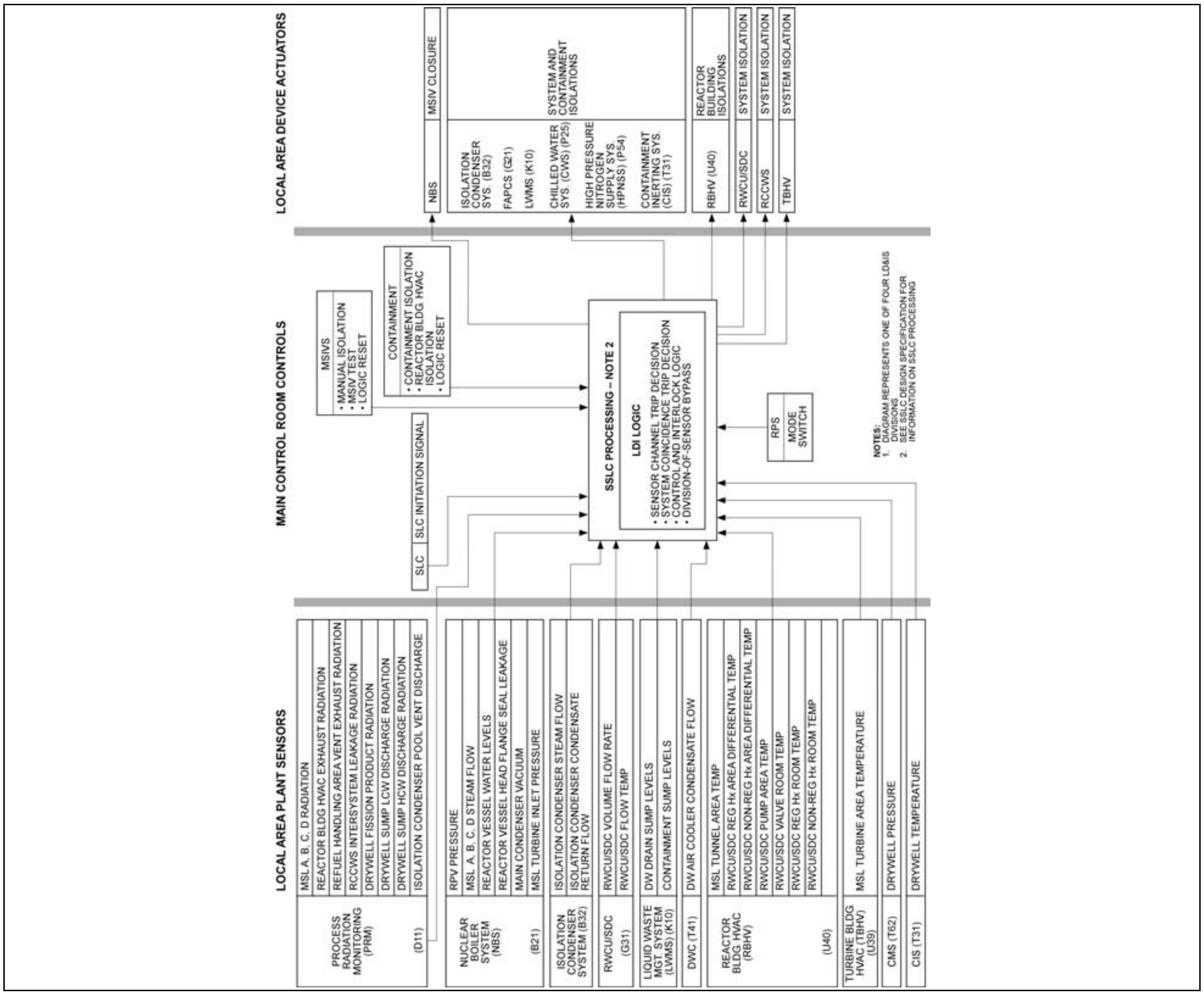


Figure 2.2.12-1. Leak Detection and Isolation System Diagram

2.2.13 Safety System Logic and Control System

Design Description

The Safety System Logic and Control (SSLC) system is the decision-making control logic segment of the automatic reactor protection and engineered safety features systems. SSLC processes automatic and manual demands for reactor trip (scram), nuclear system isolation, and engineered safety features actuation based upon sensed plant process parameters or operator request.

SSLC permits the above safety-related systems to provide protective action by implementing the protection logic functions of these safety-related systems. SSLC runs without interruption in all modes of plant operation to support the required safety-related functions.

The SSLC system includes the logic of the reactor protection system (RPS), main steam isolation valve closure logic, isolation function logic for the leak detection and isolation system (LD&IS), and the injection of boron by the Standby Liquid Control (SLC) system associated with the anticipated transient without scram (ATWS) mitigation function. (The ATWS mitigation logic includes signals for feedwater runback and automatic depressurization system inhibit.) The SSLC also includes the safety-related logic of engineering safety feature (ESF) functions and a portion of the safe shutdown systems. SSLC logic for ESF does not require operator intervention during normal operation.

The SSLC system is configured as a four-division data acquisition and control system, with each division containing an independent set of microprocessor-based, software-controlled logic processors. (The ATWS mitigation function within SSLC is implemented using non-microprocessor based discrete logic to provide diversity from the microprocessor based SSLC logic.) The four divisions exchange data via fiber optic data links to implement cross-channel data comparison. The fiber optic links provide electrical isolation between divisions.

- The SSLC system acquires data from redundant sets of sensors of the interfacing safety-related systems and provides control outputs to the final component actuators. Data is received from the E-DCIS or directly hardwired from transmitters or sensors. Trip signals from the four redundant sensor divisions are processed in a 2-out-of-4 coincident voting logic to generate an output trip signal to actuation devices. Provisions are made to allow a single division of sensor bypass for on-line maintenance, testing and repair without losing reliable trip capability. In this bypass condition, the system automatically defaults to 2-out-of-3 coincident voting. (For SSLC/ESF) safety critical automatic operations are provided with manual switches in each division for equipment initiation.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.13-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the SSLC system.

Table 2.2.13-1

ITAAC For Safety System Logic and Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the SSLC is defined in Subsection 2.2.13.	1. Inspection of the as built system will be conducted.	1. The as built SSLC system conforms with the description in Subsection 2.2.13.
2. The SSLC logic uses four independent and redundant instrument channels to monitor each safety-related parameter. A trip signal is initiated when any two-out-of-four channels have tripped.	2. The instrument channels of each SSLC safety function shall be tested using simulated signal inputs.	2. Each trip signal is initiated when at least any two-out-of-four channels have tripped.
3. The SSLC bypass provisions for on line testing and repair. In this mode, the isolation signal is initiated when any two out of three channels have tripped.	3. In channel bypass mode, each SSLC logic function shall be tested using simulated signal inputs, with the one channel under bypass condition. No trip signal shall be resulted from the bypassed channel.	3. Each signal is initiated when at least any two out of three channels have tripped, with one channel under bypass condition. No trip signal is resulted from the bypassed channel.
4. SSLC provides separate manual initiation controls in the control room for each safety-related ESF function. (Reactor scram function provided and tested under RPS.)	4. Tests will be performed using the SSLC functions in the control room.	4. Each trip signal is initiated when manual initiation controls have been initiated.
5. Separation a. The four divisions are physically and electrically separated from each other. b. Interconnections between divisions use an isolation device. Outputs to nonsafety-related systems also use an isolation device.	5. Separation Inspections of the as-built SSLC equipment will be performed.	5. Separation a. Electrical and physical separation is provided. b. Interconnections between divisions and outputs to nonsafety-related systems use isolation devices.

Table 2.2.13-1

ITAAC For Safety System Logic and Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Independence SSLC equipment in each division is powered from the divisional, Class 1E power sources.	6. Independence Apply divisional power to the assigned equipment division, perform self-test on each SSLC controller in the division.	6. Independence Applied power in a division only energizes controllers in that division.
7. Bypass Implementation SSLC provides the following bypass functions: a. Division-of-sensors bypass b. Output trip logic bypass	7. Bypass Implementation Preoperational tests will exercise the SSLC bypass functions.	7. Bypass Implementation a. Division-of-sensors bypass: Bypass Unit in a division blocks trip signals from the Digital Trip Module in that division from being processed in the trip logic of any division. Bypass status is indicated at main control panel. Only one division is allowed to be bypassed at a time. b. Output trip logic bypass: Division out of service bypass blocks trip signals from the Output Trip Logic Unit in that division from de-energizing RPS, MSIV load drivers associated with that division (RTIF). Division out of service bypass also blocks trip signals from the RMU output in that division from initiating the ESF actuation (ESF). For the squib valves, keylock switches provide effective bypass at the actuator level. Bypass status is indicated at main control panel.

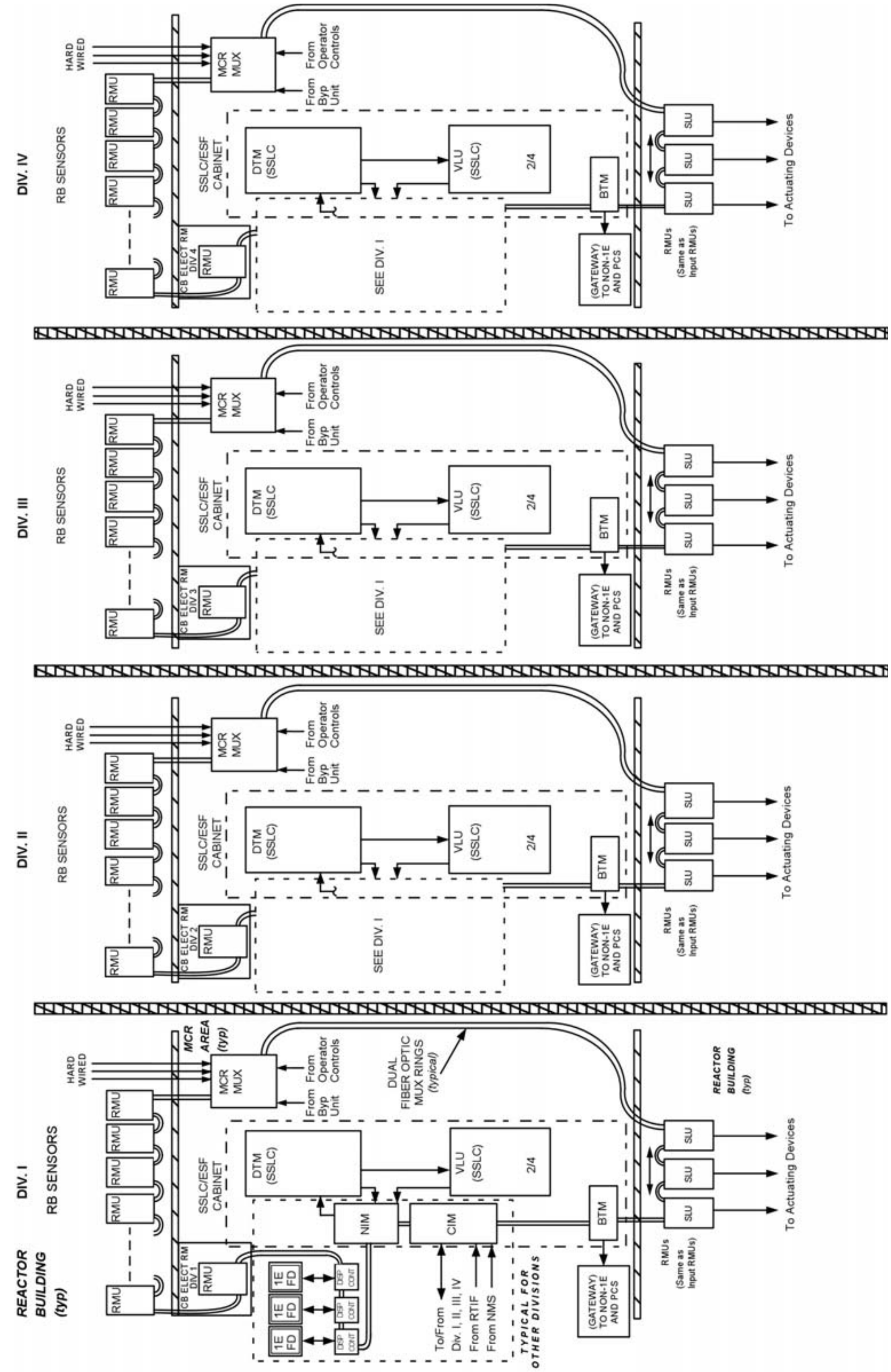
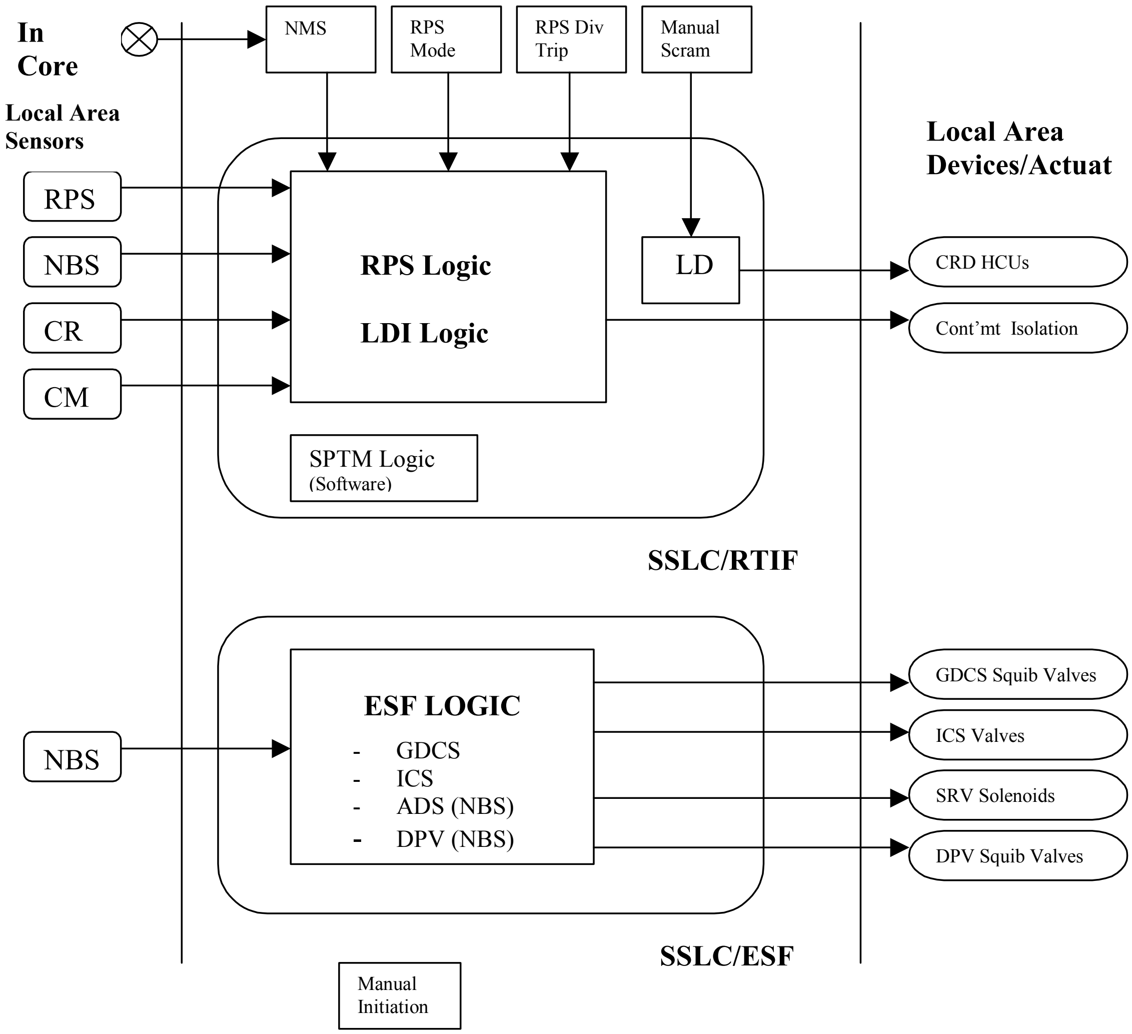


Figure 2.2.13-1. Safety System Logic and Control System Block Diagram – ESF Portion



Note: Local area sensors include:
RPS: turbine stop valve position, turbine CV oil pressure, turbine bypass valve position
NBS: MSIV position (for RTIF only), RPV pressure, water level
CRD: HCU accumulator charging water header pressure
CMS: drywell pressure

Figure 2.2.13-2. Safety System Logic and Control Interface Diagram

2.2.14 Diverse Instrumentation and Controls

Design Description

The diverse instrumentation and controls address concerns about common cause failures in software-based Reactor Protection System (RPS) and engineered safety features (ESF) systems. The BTP requires a diverse system to ensure proper operation of RPS and ESF functions in the event of a common cause type failure of the primary protection systems.

The diverse instrumentation and controls consist of three components, which address the diverse back-up functions, as follows:

- A set of protection logics that provide diverse means to scram the reactor via control rod insertion using separate and independent hardware and software from the primary RPS.
- A set of ESF initiation logics that provide diverse means to initiate the ESF functions using separate and independent hardware and software from the primary ESF systems.
- Alternate rod insertion (ARI) and associated logics (e.g., control rod run in) to scram the plant. The ARI logic is part of the Anticipated Transient Without Scram (ATWS) mitigation function. (An ATWS mitigation logic, using liquid boron injection for emergency shutdown, is included in the Safety System Logic and Control (SSLC) system.)

A simplified block diagram of the diverse instrumentation and controls is shown in Figure 2.2.14-1.

Backup of Reactor Protection System Functions:

A set of diverse logics, using separate and independent hardware and software to scram the reactor via control rod insertion, is included in the diverse instrumentation and controls. For the ESBWR, it is sufficient to include a subset of the existing RPS scram logic functions in the diverse instrumentation and controls to ensure acceptable diverse protection results. This set of diverse protection logics for reactor scram, combined with other diverse backup scram protection and diverse ESF functions, provide the necessary diverse functions to meet the required design position called out in the BTP HICB 19. The following scram signals are included in the diverse instrumentation and controls:

- High Reactor Pressure;
- High Reactor Water Level (L8);
- Low Reactor Water Level (L3);
- High Drywell Pressure; and
- High Suppression Pool Temperature.

This diverse set of RPS scram logics resides in independent and separate hardware and software equipment from the primary RPS. The process variables sensors that provide input to this diverse set of logics use different sets of sensors from that used in the primary RPS. The diverse logic equipment is nonsafety-related with triplicate redundant channels. The power sources of

this diverse equipment are from the nonsafety-related load groups. The scram initiation logic is "energize to actuate." The trip logic is based on 2-out-of-3 voting.

Backup of ESF Functions:

The ESBWR has several ESF functions including Gravity-Driven Cooling System (GDCS), Isolation Condenser System (ICS), Standby Liquid Control (SLC) system, and Automatic Depressurization System (ADS) function using safety relief valves (SRVs) and depressurization valves (DPVs). To provide adequate diverse vessel depressurization and core cooling functions, the diverse instrumentation and controls include initiation logic for GDCS, SRVs and DPVs that is diverse from the primary ESF function logic. This set of diverse logic for ESF function initiation, combined with other diverse backup scram protection and selected diverse RPS logic, provides the necessary diverse functions to meet the required design position called out in the BTP HICB 19.

This set of diverse ESF logics resides in separate and independent hardware and software equipment from the primary ESF systems. The process variables sensors that provide inputs to this diverse set of logics use different sets of sensors from that used in the primary ESF systems. The diverse logic equipment is nonsafety-related with triplicate redundant channels. The diverse equipment power source is nonsafety-related. The initiation logic is "energize to actuate" similar to the primary ESF. The trip logic is based on 2-out-of-3 voting.

Backup of ATWS mitigation functions (ARI and associated functions):

The diverse instrumentation and controls includes the nonsafety-related alternate rod insertion (ARI) logic for reactor scram, which is also considered as part of ATWS Mitigation Logic. This logic generates the following signals to support the mitigation of an ATWS event:

- A signal to open the ARI air header dump valves in the Control Rod Drive (CRD) system on a high reactor vessel pressure signal, a low reactor water level signal, or a manual ATWS initiation signal.
- A signal to the Rod Control and Information System (RCIS) to initiate electrical insertion of all operable Fine Motion Control Rod Drive (FMCRD) control rods on a high reactor vessel pressure signal, a low reactor water level signal, or a manual ATWS initiation signal.

ARI/FMCRD Run-In logic resides in the nonsafety-related diverse instrumentation and controls as a triplicate channel system, and is powered by nonsafety-related load group power sources. The logic is totally separate and independent from the RPS scram logic and uses diverse hardware and software. The input sensors for the ARI logic are independent and separate from the sensors used in the RPS scram logic.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.14-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the diverse instrumentation and controls.

Table 2.2.14-1

ITAAC For Diverse Instrumentation and Controls

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the diverse instrumentation and controls is defined in Subsection 2.2.14. The equipment includes: a set of protection logics that provide diverse means to scram the reactor via control rod insertion using separate and independent hardware and software from the primary RPS; a set of ESF initiation logics that provide diverse means to initiate the ESF functions using separate and independent hardware and software from the primary ESF systems; and a set of alternate rod insertion (ARI) and associated logics (e.g., control rod run in) via control rod insertion through alternate means by opening the air header dump valves of the control rod drive system.	1. Inspection of the as built system will be conducted.	1. The as built diverse instrumentation and controls system conforms with the description in Subsection 2.2.14.
2. The diverse instrumentation and controls logic uses 3 redundant instrument channels to monitor each parameter. A RPS trip signal is initiated when any 2-out-of-3 channels have tripped. An ESF trip signal is initiated when any 2-out-of-3 channels have tripped. An ARI/FMCRD Run-In signal is initiated when any 2-out-	2. The instrument channels of each diverse instrumentation and control function shall be tested using simulated signal inputs.	2. Each RPS trip signal is initiated when at least any 2-out-of-3 channels have tripped. Each ESF trip signal is initiated when at least any 2-out-of-3 channels have tripped. Each ARI/FMCRD Run-In signal is initiated when at least any 2-out-of-3 channels have tripped.

Table 2.2.14-1

ITAAC For Diverse Instrumentation and Controls

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
of-3 channels have tripped.		
3. The diverse instrumentation and controls provide separate manual controls in the control room for each function.	3. Tests will be performed using the diverse instrumentation and controls manual functions in the control room.	3. Each trip signal is initiated when manual controls have been initiated.
4. Separation a. The divisions of redundant instrumentation are physically and electrically separated from each other. b. Interconnections among divisions, such as data communications for coincident trip logic decisions, use an isolating transmission medium. Outputs to nonsafety-related systems also use an isolating transmission medium.	4. Separation Inspections of the as-built diverse instrumentation and controls equipment will be performed.	4. Separation a. Electrical separation is provided. b. Interconnections among divisions and outputs to nonsafety-related systems use an isolating transmission medium.
5. Independence Diverse instrumentation and controls equipment in each division is powered from the non-Class 1E, multi load group plant power sources.	5. Independence Apply power to an equipment division and perform self-test on each diverse instrumentation and control controller.	5. Independence Applied power in a division only energizes controllers in that division.

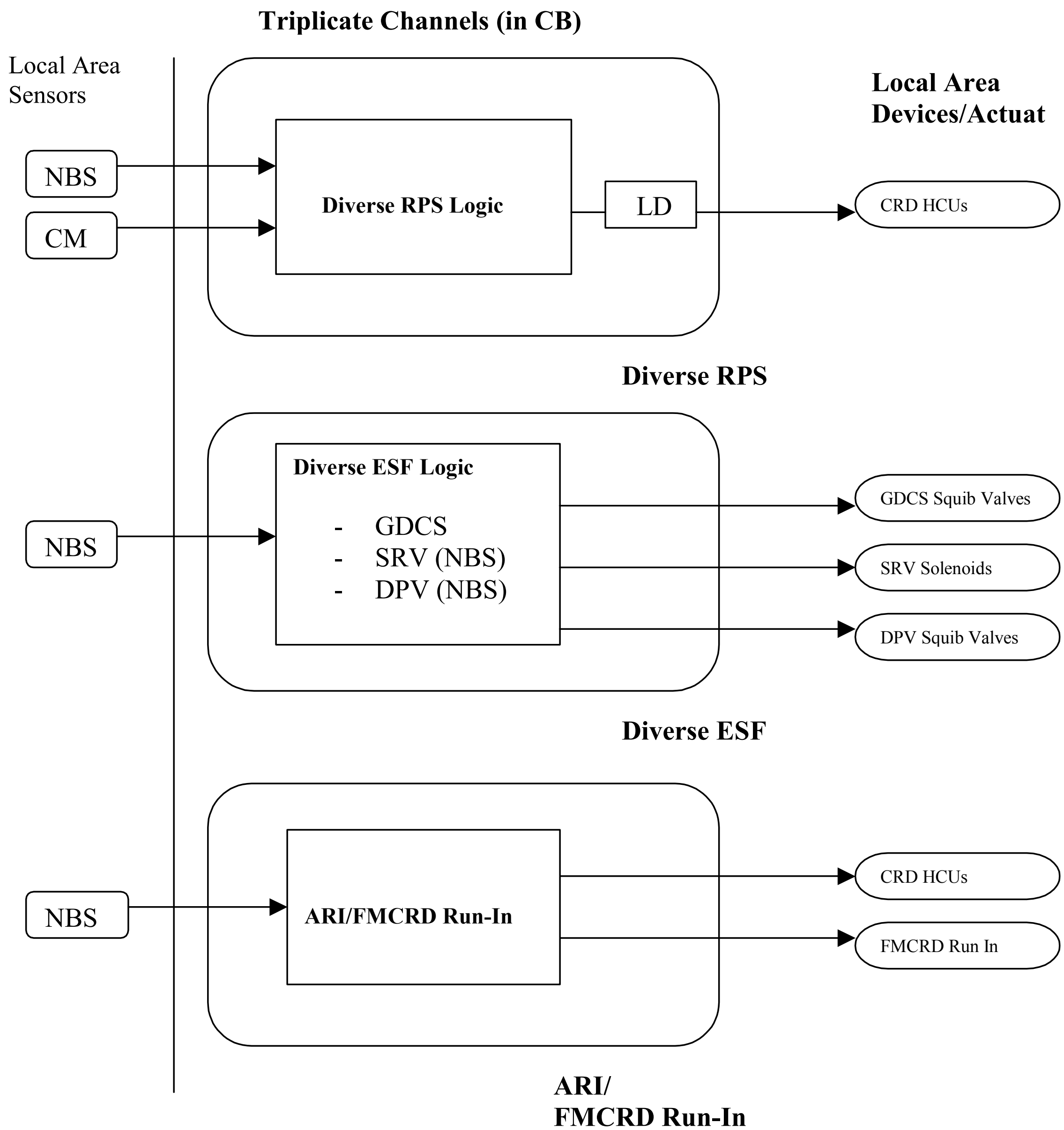


Figure 2.2.14-1. Simplified Diverse Logic and Controls Block Diagram

2.3 RADIATION MONITORING SYSTEMS

The following subsections describe the major radiation monitoring systems for the ESBWR.

2.3.1 Process Radiation Monitoring System

Design Description

The Process Radiation Monitoring System (PRMS) measures and provides for display of radioactivity levels in process and effluent gaseous and liquid streams, initiates protective actions, and activates alarms in the Main Control Room (MCR) on high radiation signals. The PRMS provides radiological monitoring during plant operation and following an accident. Subsystems of the PRMS consist primarily of Radiation Detection Assemblies, off-line liquid and gaseous sampling panels/skids, in-line sample chambers and Signal Conditioning Units. The PRMS consists of independent subsystems, each of which contains between one and eight monitoring channels. The PRMS safety-related channel trip signals are provided as inputs to the Safety System Logic and Control (SSLC) for generation of protective action signals.

The primary functions of the PRMS are to:

- Monitor the various gaseous and liquid process streams and effluent releases and provide main control room display, recording and alarm capability;
- Initiate alarms in the main control room to warn operating personnel of high radiation activity; and
- Initiate the appropriate actions and controls to prevent further radioactivity releases to the environment.

This PRMS provides instrumentation for radiological monitoring, sampling and analysis of identified process and effluents streams throughout the plant. The process and effluent paths and/or areas listed below are monitored for potential high radioactivity releases. The radiation monitors of the first seven items, which are safety-related, provide safety related functions. The remaining PRMS subsystems provide nonsafety-related functions.

- The Reactor Building HVAC Exhaust Vent RMS continuously monitors the gross gamma quantity of radioactivity being exhausted via this Exhaust duct and the Refueling Area Air Exhaust duct. The discharge point from the duct is monitored with four physically and electrically independent and redundant divisions. In the event of radioactive releases due to system failures in the Reactor Building, or due to a fuel handling accident, the Reactor Building HVAC exhaust fans are stopped.
- The Control Room Air Intake RMS consists of eight channels. Four divisonalized Radiation Detection Assemblies are mounted external to each ventilation intake duct for the Control Room HVAC. The Radiation Detection Assemblies continuously monitor the gamma radiation levels from each air intake plenum for the building or area containing the MCR and auxiliary rooms. The Control Room outside air intake is secured in the event of a high radiation levels in order to protect the operating staff.
- The Isolation Condenser Vent Discharge RMS continuously monitors the four Isolation Condenser Discharge Vents for gross gamma radiation by sixteen local detectors (four

per isolation condenser vent). High radiation in the exhaust of a vent results in isolation of the affected Isolation Condenser loop.

- The Refuel Handling Area Air Exhaust RMS continuously monitors gamma radiation levels in the exhaust plenum of the HVAC exhaust ducts in the Refuel Handling Area of the Reactor Building with four divisions of Radiation Detection Assemblies and channels. In the event of a radioactive release due to an accident while handling spent fuel, the Reactor Building HVAC exhaust fans are tripped off.
- The Fuel Building Main Area HVAC RMS consists of four channels that monitor the gamma radiation level of the air exiting the spent fuel pool and associated fuel handling areas as well as the rooms with the fuel pool cooling and cleanup equipment. In the event of radioactive releases due to an accident while handling spent fuel, Fuel Building HVAC exhaust fans are stopped.
- The Drywell Sump LCW/HCW Discharge RMS continuously monitors gamma radiation levels in the transfer pipes from the Drywell Low Conductivity Waste (LCW) and High Conductivity Waste (HCW) sumps to the Radwaste System. The two locations monitored are downstream of the Drywell LCW sump discharge pipe isolation valve and downstream of the Drywell HCW sump discharge isolation valve. Automatic isolation of the two sump discharge pipes occurs if high radiation levels are detected during liquid waste transfers.
- The Containment Purge Exhaust RMS consists of four channels that monitor the gross radiation level in the exhaust duct leading from the primary containment. In the event of radioactive releases, the monitors initiate closure of the ventilation isolation dampers prior to exceeding radioactive effluent limits. In addition to the closure of the Reactor Building HVAC isolation dampers, the RB HVAC exhaust fans are stopped.
- The Main Steamline (MSL) RMS continuously monitors the gamma radiation level of the main steamlines in the MSL tunnel area for high gross gamma radioactivity in the steam flow to the turbine. The subsystem provides input to logic that results in shutdown of the main turbine condenser mechanical vacuum pump (MVP) and MVP valve closure. Although the MSL RMS is safety related, its functions are non-safety.
- The Offgas Pre-Treatment sampling RMS has a single channel. The subsystem samples the Offgas stream at the discharge from the Offgas cooler and condenser. Typically, the first indication of a fuel failure is detected by this subsystem.
- The Offgas Post-Treatment RMS monitors the release of radiation at the discharge from the Offgas System, after the process stream has passed through the charcoal hold-up system. The subsystem consists of two independent skids and a gas sampler. The subsystem is equipped with a flow controller capable of continuously measuring the mass flows of both the main process and the sample and automatically maintaining the sample flow proportional to the process flow.
- The Charcoal Vault Ventilation Exhaust RMS, consisting of one channel, monitors the radioactivity exhausting in the ventilation air from the charcoal vault.

- The Turbine Building Normal Ventilation Exhaust consists of two non-divisional channels that continuously monitor the normal ventilation air HVAC from the clean area in the Turbine Building for gross radiation levels.
- The Turbine Building Compartment Area Air HVAC RMS consists of two non-divisional channels that continously monitor the air in the compartment area HVAC in the Turbine Building for gross radiation levels.
- The Turbine Building Combined Ventilation Exhaust RMS monitors the Turbine Building Combined Ventilation exhaust for halogens, particulates and noble gas releases during normal and accident conditions. The subsystem has provision for monitoring tritium.
- The Main Turbine Gland Seal Steam Condenser Exhaust RMS continuously monitors the gland seal steam offgas, discharged into the Turbine Building Ventilation System, for radioactive noble gases. A sampler, similar to the offgas post-treatment radiation monitor sampler, is capable of grabbing gaseous samples.
- The Radwaste Building Ventilation Exhaust RMS continuously monitors halogens, particulates and noble gas releases from the Radwaste Building vent to the atmosphere for both normal and accident conditions. The subsystem has provision for monitoring tritium.
- The Liquid Radwaste Discharge RMS, consisting of a single channel, continuously monitors the gross gamma radiation level in the liquid effluent stream. The Liquid Radwaste Discharge RMS initiates the closure of the Radwaste Discharge system isolation valves on high radiation level. A sampling skid is provided.
- The Drywell Fission Product RMS consists of two channels that monitor the drywell air space radiation levels for leakage detection. The Drywell Fission Product RMS monitors a continuous sample, extracted from the drywell, for the presence of radioactive particulates and noble gases. The subsystem shall be utilized to aid in meeting the detection requirements for reactor coolant leakage. The subsystem includes local sampling panels and a Signal Conditioner connected to each radiation detector assembly.
- The Reactor Component Cooling Water (RCCW) Intersystem Leakage RMS consists of two channels. These channels monitor for gross radiation levels that are indicative of leakage through the heat exchangers in the RCCW system.
- A single channel radiation monitor continuously monitors the Technical Support Center Ventilation intake duct. Upon detection of radioactivity at the outside air intake, the Air Handling Unit (AHU) outdoor air damper is closed and a filter train fan is started.
- The Fuel Building Ventilation Exhaust AHU RMS consists of four channels that monitor the radiation level of the air entering the Fuel Building Ventilation unit area exhaust AHUs.
- The Fuel Building Combined Ventilation Exhaust RMS continuously monitors halogens, particulates and noble gases releases being exhausted from the Fuel Building to the plant stack under both normal and accident conditions. The subsystem has provision for monitoring tritium.

- The Plant Stack RMS monitors particulate, iodine and gaseous concentrations in the main stack effluent for both normal and accident plant conditions. It is composed of three sampling channels that are designed to meet the requirements of both 10 CFR 20 for low level effluent releases and Regulatory Guide 1.97 for accident effluent releases. Provisions for monitoring tritium are also provided.

Figure 2.3.1-1 shows the PRMS control interfaces.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.3.1-1 provides a definition of the inspections, tests and/or analyses, together with the associated acceptance criteria, which will be undertaken for the Process Radiation Monitoring System.

Table 2.3.1-1

ITAAC For The Process Radiation Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The PRMS samples and/or monitors radiation levels in process and effluent streams and/or paths in the plant, and initiates protective actions signals and alarms. The PRMS consists of independent subsystems as described in Subsection 2.3.1.	1. Tests shall be performed on each subsystem described in Subsection 2.3.1.	1. Initiation of the PRMS protective action signals and/or alarm signals occurs when a test setpoint has been exceeded. This acceptance criteria applies to each PRMS subsystem described in Subsection 2.3.1.
2. Control Room indications and controls provided for each PRMS subsystem are as defined in Subsection 2.3.1.	2. Inspections will be performed on the control room PRMS subsystem indications and controls.	2. Indications and controls exist or can be retrieved in the control room as defined in Subsection 2.3.1.
3. The safety-related PRMS subsystems as identified in Subsection 2.3.1 are powered from uninterruptible safety-related power sources.	3. Tests will be conducted to determine the power sources to the PRMS safety-related subsystems as described in Subsection 2.3.1.	3. The safety-related PRMS subsystems described in Subsection 2.3.1 receive electrical power from uninterruptible safety-related busses.
4. The nonsafety-related PRMS subsystems as described in Subsection 2.3.1 are powered from nonsafety-related power sources.	4. Tests will be conducted to verify the availability of nonsafety-related power to the PRMS nonsafety-related subsystems as described in Subsection 2.3.1.	4. The nonsafety-related PRMS subsystems as described in Subsection 2.3.1 receive electrical power from nonsafety-related busses.

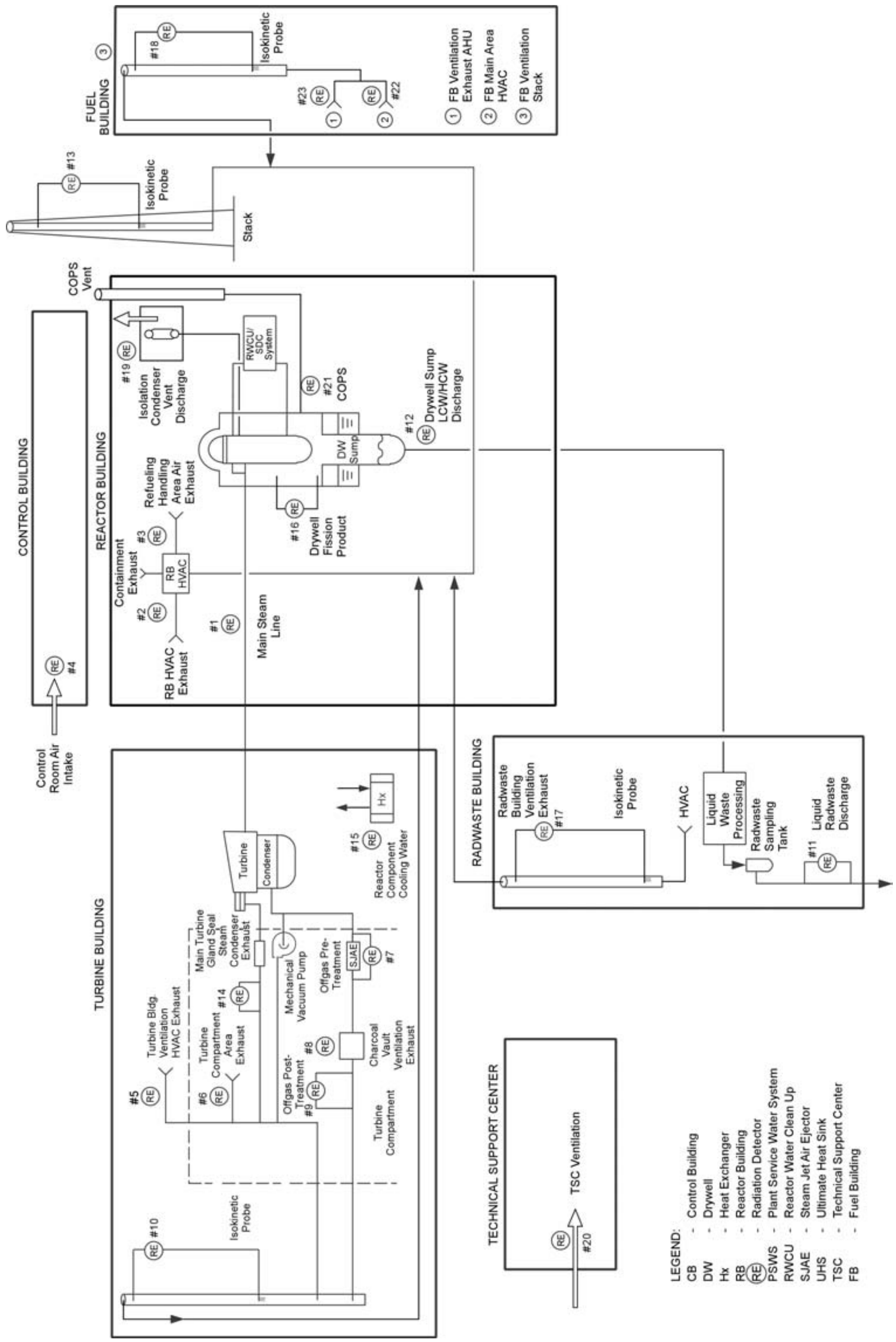


Figure 2.3.1-1. Process Radiation Monitoring System Diagram

2.3.2 Area Radiation Monitoring System

Design Description

The Area Radiation Monitoring System (ARMS) continuously monitors the gamma radiation levels within the various areas of the plant and provides an early warning to operating personnel when high radiation levels are detected so the appropriate action can be taken to minimize occupational exposure.

The ARMS consists of a number of channels, each consisting of a Radiation Detection Assembly and a Signal Conditioning Unit. When required, a local Auxiliary Unit with a display and audible alarm is also provided. Each ARMS radiation channel has two independently adjustable trip alarm circuits. One circuit is set to trip on High radiation and the other is set to trip on downscale indication (loss of sensor input). ARMS alarms in both the MCR and at plant local areas. Each ARM Signal Conditioning Unit is equipped with a test feature that monitors for gross failures and activates an alarm on loss of power or when a failure is detected.

This system is nonsafety-related. The radiation monitors are powered from the non-Class 1E 120 VAC sources.

The trip alarm setpoints are established in the field following equipment installation at the site. The exact settings are based on sensor location, background radiation levels, expected radiation levels, and low occupational radiation exposures.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.3.2-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Area Radiation Monitoring system.

Table 2.3.2-1

ITAAC For The Area Radiation Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the ARM system is defined in Subsection 2.3.2.	1. Inspection of the as-built system will be conducted.	1. The as-built ARM system conforms with the description in Subsection 2.3.2.
2. Each ARM channel monitors radiation level in its assigned area, and initiates a MCR alarm and a local audible alarm (if provided) when the radiation level exceeds a preset limit.	2. Tests will be conducted using simulated	2. The MCR alarm and local audible alarm (if provided) are initiated when the simulated radiation level exceeds a preset limit.
3. MCR alarms and displays provided for the ARM system are as defined in Subsection 2.3.2.	3. Inspections will be performed on the MCR alarms and displays for the ARM system.	3. Alarms and displays exist or can be retrieved in the MCR as defined in Subsection 2.3.2.

2.4 CORE COOLING SYSTEMS USED FOR ABNORMAL EVENTS

The following subsections describe the core cooling systems in response to AOOs and accidents.

2.4.1 Isolation Condenser System

Design Description

Figure 2.4.1-1 shows the Isolation Condenser System (ICS), which removes decay heat after any reactor isolation during power operations. Decay heat removal limits further pressure rise and keeps the RPV pressure below the SRV pressure setpoint. It consists of 4 independent trains, each containing a heat exchanger that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC/PCC pool, which is vented to the atmosphere.

The ICS is initiated automatically on a high reactor pressure, MSIV closure, loss of power generation busses, and low RPV water level. To start an IC into operation, a condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line. The operator from the MCR can also initiate the ICS manually. A fail-open nitrogen piston-operated condensate return bypass valve opens if the DC power is lost.

The ICS is isolated automatically when either a high radiation level in the IC pool area is detected or excess flow is detected in the steam supply line and condensate return line.

The IC/PCC pool is divided into sub compartments that are interconnected at their lower ends to provide full use of the water inventory for heat removal by any IC. The Fuel and Auxiliary Pools Cooling System (FAPCS) perform Cooling and cleanup of IC/PCC pool water. During IC operation, IC/PCC pool water can boil, and the steam produced is vented to the atmosphere. This boil-off action of non-radioactive water is a safe means for removing and rejecting all reactor decay heat.

The IC/PCC pools have an installed capacity that provides at least 72 hours of reactor decay heat removal capability. The heat rejection process can be continued indefinitely by replenishing the IC/PCC pool inventory. A safety-related independent FAPCS makeup line is provided to convey emergency makeup water into the IC/PCC pool, from either the site Fire Protection System or from piping connections located at grade level in the reactor yard external to the Reactor Building. This makeup can be accomplished without any valving changes in the Reactor Building no matter what the prior operating mode of the FAPCS might have been.

The ICS passively removes sensible and core decay heat from the reactor with minimal loss of coolant inventory from the reactor, when the normal heat removal system is unavailable following any of the following events:

- Sudden reactor isolation at power operating conditions;
- During station blackout (i.e., unavailability of all AC power); and
- Anticipated Transient Without Scram (ATWS).
- Loss of Coolant Accident (LOCA)

The ICs are sized to remove post-reactor isolation decay heat with 3 of 4 ICs operating and to reduce reactor pressure and temperature to safe shutdown conditions, with occasional venting of

radiolytically generated noncondensable gases to the suppression pool. Because the heat exchangers (ICs) are independent of plant AC power, they function whenever normal heat removal systems are unavailable, to maintain reactor pressure and temperature below limits.

Periodic surveillance testing of the ICS valves can be performed by the control room operator via remote manual controls that actuate the isolation valves and the condensate return valves. The opening and closure of the valves is verified by their status lights.

The portions of the ICS (including isolation valves), which are located inside the containment and out to the IC flow restrictors, are designed to ASME Code Section III, Class 1, Quality Class A. Other portions of the ICS are ASME Code Section III, Class 2, Quality Class B. The IC/PCC pools are safety-related and Seismic Category I.

Safety Requirements:

The ICS performs the following safety-related functions:

- Automatically limit the reactor pressure and prevent SRV operation when the reactor becomes isolated following reactor scram during power operations.
- In event of a LOCA, ICS provides additional liquid inventory upon opening of the condensate return valves to initiate the system. The ICS also provides an initial depressurization of the reactor on loss of feedwater flow, such that the ADS can take place from a lower water level.
- The ICS shall, in conjunction with the water stored in the RPV, conserve sufficient reactor coolant volume to avoid automatic depressurization caused by low reactor water level.
- Remove reactor decay heat produced during and following an abnormal event, which involve reactor scram and containment isolation. The abnormal events include Station Blackout, Anticipated Transient Without Scram (ATWS) and LOCA.
- Maintain reactor coolant pressure boundary (RCPB) integrity.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.4.1-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Isolation Condenser System.

Table 2.4.1-1

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the ICS is as shown in Figure 2.4.1-1.	1. Inspections of the as-built system will be conducted.	1. The as-built IC System conforms with the basic configuration shown in Figure 2.4.1-1.
2. Each ICS Class 2 branch line from the steam supply line outside the containment has a flow limiter located in the Class 1 line upstream of the Class 2 branch as indicated on Figure 2.4.1-1.	2. Inspection will be performed to confirm that a flow limiter is included in the branch line.	2. Each branch line contains a flow limiter which is one-half the inside diameter (or less) of the downstream branch line.
3. The ASME code components of the ICS retain their pressure boundary integrity under internal pressures that will be experienced in service.	3. A hydrostatic test will be conducted on those code components of the IC System required to be hydrostatically tested by the ASME Code.	3. The results of the hydrostatic test of the ASME code components of the ICS conform with the requirements of the ASME Code, Section III.
4. Each isolation valve (which are safety-related) shown on Figure 2.4.1-1 close to limit offsite doses below limit values in case of an IC line break.	4. Opening and/or closing tests of valves will be conducted under pre-operational fluid flow	4. Each isolation valve (which are safety-related) shown on Figure 2.4.1-1 close to limit offsite doses below limit values in case of an IC line break. Closure time will be less that or equal to [70 sec].
5. Each condensate return valve (which are safety-related) shown on Figure 2.4.1-1 will open to initiate the IC system.	5. Opening and/or closing tests of valves will be conducted under pre-operational differential pressure, fluid flow and temperature conditions.	5. Each condensate return valve opens to initiate the IC system with an open-stroke time less than or equal to [30 sec].

Table 2.4.1-1

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The normally open ICS isolation valves in the steam supply and condensate return lines close automatically on receipt of high vent line radiation from the Process Radiation Monitoring System (PRMS).	6. An isolation valve closure test will be performed using simulated signals.	6. Isolation valves close upon insertion of signals from the PRMS.
7. The normally open ICS isolation valves in the steam supply and condensate return lines close automatically on receipt of signals from the LD&IS.	7. An isolation valve closure test will be performed using simulated signals.	7. Isolation valves close upon insertion of signals from the LD&IS.
8. Each ICS train normally closed condensate return valve opens upon receipt of an automatic actuation signal for MSIV closure or loss of power busses with the reactor mode switch in RUN, or on high RPV pressure or RPV low water level with the reactor mode switch in STARTUP or RUN.	8. Valve opening tests will be performed using simulated MSIV closure signals with the reactor mode switch in RUN, and on RPV high-pressure signal. Valve opening test will be performed using simulated low reactor water level signals with the reactor mode switch in STARTUP or RUN, and on the loss of feedwater pumps.	8. The condensate return valves open upon insertion of all simulated signals

Table 2.4.1-1

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. Each ICS train normally closed condensate return bypass valve opens upon receipt of an automatic actuation signal for MSIV closure or loss of power busses with the reactor mode switch in RUN, or on high RPV pressure or RPV low water level with the reactor mode switch in STARTUP or RUN.	9. Valve opening tests will be performed using simulated MSIV closure signals with the reactor mode switch in RUN, and on RPV high-pressure signal. Valve opening test will be performed using simulated low reactor water level signals with the reactor mode switch in STARTUP or RUN, and on the loss of feedwater pumps	9. The condensate return valves open upon insertion of all simulated signals
10. The two-series, solenoid-operated bottom vent line valves open on high RPV pressure after time delay following condensate return or condensate bypass valve opening signals.	10. A valve opening test will be performed using simulated high reactor pressure after a time delay following condensate return or condensate bypass valve opening signals.	10. The two-series, solenoid-operated vent line valves open on a simulated high RPV pressure signal of equal to or greater than [7.516] MPa gauge after a time delay following condensate return or condensate bypass valve opening signals.
11. The three vent lines with two-series, solenoid-operated top and bottom vent line valves open on manual initiation only if condensate return or condensate bypass valve is not closed.	11. A vent valve opening test will be performed using simulated during pre-operational testing following condensate return or condensate bypass valve opening signals.	11. The three vent lines with two-series, solenoid-operated vent line valves each, opens on a manual initiation following condensate return or condensate bypass valve opening signals.

Table 2.4.1-1

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. The accumulator for the pneumatic motor (PM) isolation valves in the ICS steam supply and condensate return valves have the capacity to close the valves three times with the drywell at the drywell design pressure.	12. An analysis and/or test will be performed to demonstrate the capacity of the PM isolation valve accumulators.	12. Either: a. The PM isolation valve accumulators have the capacity to close the valves three times with the drywell pressure at, or above the design pressure, or b. The PM isolation valve accumulators have the capacity to close the valves 2-3 times with the drywell at atmospheric pressure, and an analysis or test that shows the 2-3 closures with the drywell at the drywell design pressure is achievable.
13. Upon loss of pneumatic pressure to the condensate bypass valve (fail open), the valve strokes to the fully open position.	13. Tests will be performed to demonstrate that the condensate bypass valve will stroke to the full open position upon the loss of pneumatic pressure to the condensate bypass valve accumulator.	13. The condensate bypass valve opens when pneumatic pressure is removed from the condensate bypass valve.
14. Class 1E loads for the ICS are powered from the correct Class 1E Divisions.	14. Tests will be performed on the IC System by providing a test signal in only one Class 1E Division at a time.	14. The test signal exists only in the Class 1E Division under test in the ICS.
15. Control Room indications and controls provided for the ICS are operable.	15. Inspections will be performed on the Control Room indications and controls for the ICS.	15. Indications and controls exist or can be retrieved in the Control Room.

Table 2.4.1-1

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
16. Each ICS train minimum heat removal capacity is 33.75 MWt with reactor above rated pressure.	16. A full scale, one-half capacity prototype IC unit will be performance tested at a steam flow and pressure which is dependent upon the available capacity of the test supply boiler and the actual condensing capacity of the IC unit.	16. ICS loop unit heat removal capacity is ≥ 33.75 MWt as determined by analysis and/or by full scale, one-half capacity IC unit prototype tests.
17. The Isolation Condenser System provides a minimum drainable liquid volume available for return to the RPV.	17. An analysis will be performed for the as-built isolation condenser system.	17. An analysis exists and demonstrates that the as-built isolation condenser system provides at least [4.98 m3] of liquid available for return to the RPV.

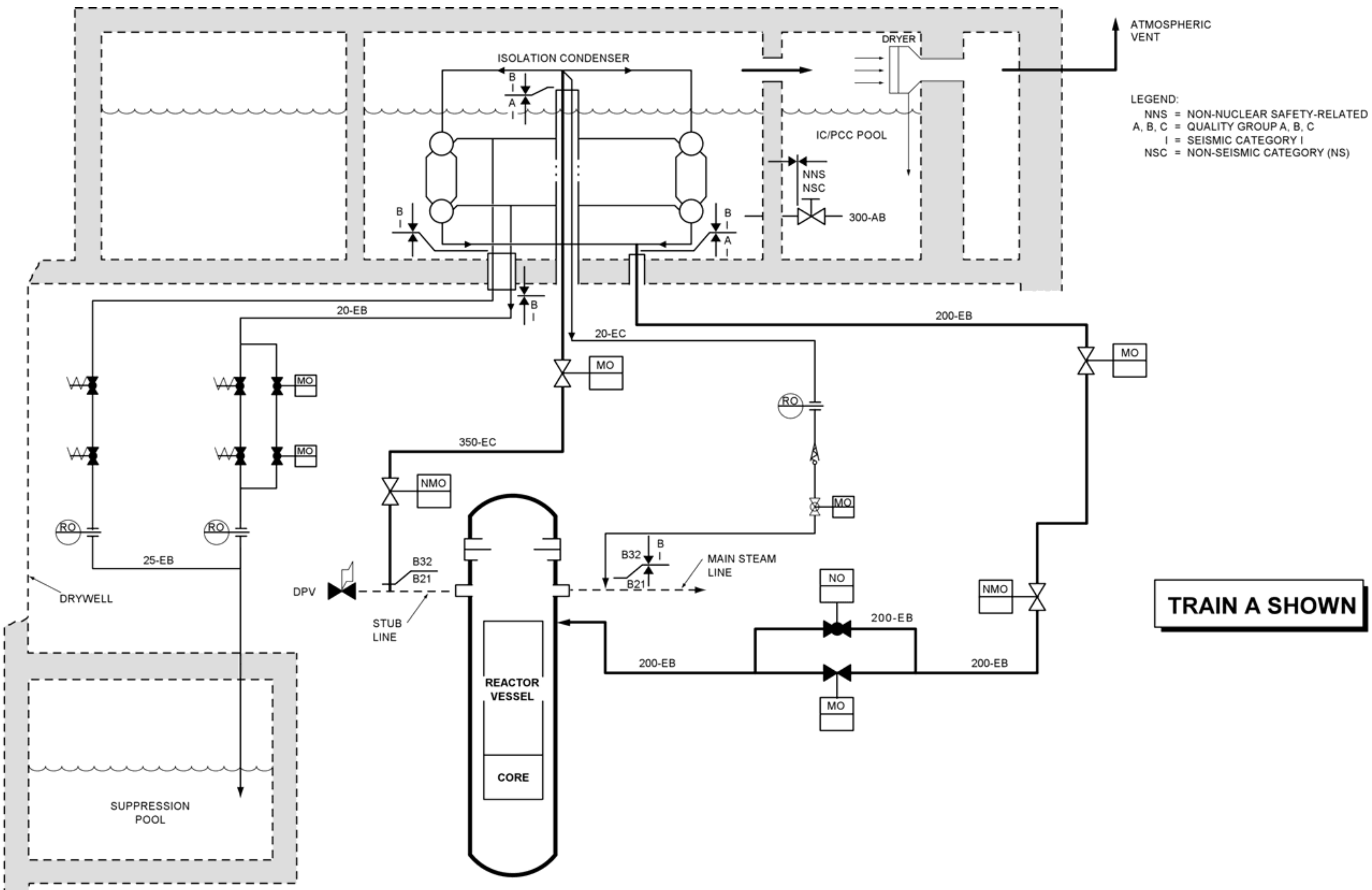


Figure 2.4.1-1. Isolation Condenser System Schematic

2.4.2 Emergency Core Cooling System - Gravity-Driven Cooling System

Design Description

Emergency core cooling is provided by the Gravity-Driven Cooling System (GDCS) in conjunction with the ADS in case of a LOCA. When an ECCS LOCA signal (low RPV water level) is received, the ADS depressurizes the reactor vessel and the GDCS injects sufficient cooling water to maintain the fuel cladding temperatures below temperature limits defined in 10 CFR 50.46. The GDCS is shown in Figure 2.4.2-1

In the event of a severe accident that results in a core melt with the molten core in the lower drywell region, GDCS floods the lower drywell cavity region with the water inventory of the three GDCS pools and the suppression pool (SP).

The GDCS is an engineered safety feature (ESF) system. It is classified as safety-related and Seismic Category I. GDCS instrumentation and DC power supply are IEEE Class 1E.

Basic system parameters are:

- Three independent subsystems
  - Short-term cooling (injection)
  - Long-term cooling (equalization)
  - Deluge (drywell flooding)
- Initiation signal: confirmed ECCS initiation signal from NBS
  - Sealed-in divisional ECCS initiation signal
  - Four channels
- A time delay between initiation and actuation for short-term water injection
- A time delay between initiation and actuation for long-term water injection
  - With low RPV water level permissive
- Squib valve firing logic is 2-out-of-4
- Manual actuation:
  - Two channels
  - Permissive: Interlocked to RPV low pressure signal for short- and long-term cooling subsystems and interlocked to RPV high-high drywell pressure
  - Logic is simultaneous operation of two operator inputs of the same division
- Monitored parameters:
  - GDCS Pool water level
  - GDCS valve positions
  - Squib valve continuity

The GDCS injects water into the downcomer annulus region of the reactor after a LOCA and reactor vessel depressurization. It provides short-term (injection line) gravity-driven water makeup from three separate water pools located within the upper drywell at an elevation above the active core region. The system also provides long-term (equalization line) post-LOCA makeup from the suppression pool to meet long-term core decay heat boil-off requirements. During severe accidents the system floods the lower drywell region with water through deluge lines if the core melts through the RPV.

The GDCS is completely automatic in actuation and operation. A backup to automatic actuation is the ability to actuate by operator action.

The GDCS consists of four identical trains completely independent of each other both electrically and mechanically, with the exception of two trains sharing one of the GDCS pools. A confirmed ECCS LOCA signal actuates the ADS to reduce RPV pressure. When a coincident high drywell pressure signal is present, ADS initiates earlier and at a higher RPV water level. Simultaneously, short-term (injection) system timers, and long-term (equalization) system timers in the GDCS logic are started, which, after time-out, actuate squib valves providing an open flow path from the respective water sources to the vessel.

The short-term system supplies gravity-driven flow to eight separate injection nozzles on the vessel with flow from the three separate GDCS pools. The long-term system supplies gravity-driven flow to four other nozzles with flow from the suppression pool through equalizing lines.

Both the short-term and long-term systems are designed to ensure that adequate reactor vessel inventory is provided assuming a LOCA in one GDCS line and failure of one GDCS injection (squib) valve to actuate in a separate GDCS train.

GDCS deluge lines, each having one squib actuated valve, provide a means of flooding the lower drywell cavity in the event of a core melt sequence which causes failure of the lower vessel head and allows molten fuel to reach the lower drywell cavity floor. These squib-activated valves are driven by logics receiving input signals from an array of temperature sensors located in the lower drywell.

GDCS pool level is the only essential system parameter that must be monitored in the main control room to verify system readiness and its proper function following initiation. Low level alarm instrumentation is included as part of GDCS.

All piping and valves connected with the RPV, including squib valves, and up to and including the biased-open check valve shall be classified as follows:

- Safety-Related
- Quality Group: A
- Seismic Category: I

All piping and valves connecting the GDCS pools and S/P to the biased-open check valve, and all piping and valves (including supports) connecting GDCS pool to lower Drywell shall be classified as follows:

- Safety-Related
- Quality Group: C

- Seismic Category: I

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.4.2-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Gravity-Driven Cooling System.

Table 2.4.2-1

ITAAC For The Gravity-Driven Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the GDCS is as described in Subsection 2.4.2 and shown on Figure 2.4.2-1.	1. Inspections of the as-built system will be conducted.	1. The as-built GDCS conforms with the basic configuration described in Subsection 2.4.2 and shown in Figure 2.4.2-1.
2. The GDCS provides sufficient flow to maintain water coverage one meter above the Top of Active Fuel (TAF) for 72 hours following the design basis LOCA.	2. a. For each loop of the GDCS, an open reactor vessel test will be performed utilizing two test valves in place of the parallel squib valves in the GDCS gravity drain line and connected to the GDCS actuation logic. Flow measurements will be taken on flow into the RPV. b. For each loop of the GDCS, open reactor testing will be performed utilizing one test valve in place of the squib valve in the equalizing line and connected to the GDCS actuation logic. Flow measurements will be taken on flow into the RPV.	2. a. An analysis exists that demonstrates that the observed flow rate, in conjunction with vessel depressurization and other modes of GDCS operation, will maintain water coverage one meter above TAF for 72 hours following the design basis LOCA. b. An analysis exists that demonstrates that the observed flow rate, in conjunction with vessel depressurization and other modes of GDCS operation, will maintain water coverage one meter above TAF for 72 hours following the design basis LOCA.
3. The GDCS squib valve used in the injection and equalization applications has a flow coefficient (Cv) that will permit development of full GDCS flow.	3. A type test will be performed on a squib valve that has been previously activated to the open position.	3. The GDCS squib valve used in the injection and equalization applications has a Cv that will permit development of full GDCS flow.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The GDCS check valves will meet the minimum flow requirements for a valve stuck in the "valve biased" open position.	4. An analysis and test (at test facility) will be performed on a GDCS check valve held in the "valve biased" position.	4. The GDCS check valves will meet the minimum flow requirements for a valve stuck in the "valve biased" open position
5. The GDCS squib valve used in the deluge valve applications has a C_v that will permit development of full GDCS flow.	5. A type test will be performed on a squib valve that has been previously activated to the open position.	5. The GDCS squib valve used in the deluge valve applications has a Cvthat will permit development of full GDCS flow.
6. The GDCS deluge valves open upon receipt of a signal indicating high temperatures on the lower drywell floor.	6. A test will be performed by simulating an actuation signal that would be transmitted from the thermocouple grid in the lower drywell floor. Explosive charges will not be detonated by this test	6. A signal is received in each loop of the GDCS at the deluge valves.
7. Control Room indications and controls provided for the GDCS are operable.	7. Inspections will be performed on the Control Room indications and controls for the GDCS.	7. Indications and controls exist or can be retrieved in the control room as defined in Subsection 2.4.2.
8. GDCS squib valves will maintain RPV backflow leaktightness and maintain reactor coolant pressure boundary integrity during normal plant operation.	8. A test will be performed to demonstrate the squib valves are leaktight during normal plant conditions.	8. GDCS squib valves will have zero leakage at normal plant operation pressure
9. Each GDCS injection nozzle flow limiter is less than or equal to 4.562E-3 m^2 (0.0491 ft^2).	9. Inspections of the as-built GDCS injection flow limiters will be taken	9. Each GDCS injection nozzle flow limiter is less than or equal to 4.562E-3 m^2 (0.0491 ft^2).
10. Each GDCS equalizing line nozzle flow limiter is less than or equal to 2.027E-3 m^2 (0.0218 ft2).	10. Inspections of the as-built GDCS equalizing flow limiters will be taken	10. Each GDCS equalizing line nozzle flow limiter is less than or equal to 2.027E-3 m^2 (0.0218 ft^2).

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. The ASME Code portions of the GDCS retain their integrity under internal pressures that will be experienced during service.	11. A hydrostatic test will be conducted on those Code components of the GDCS required to be hydrostatically tested by the ASME Code.	11. The results of the hydrostatic test of the ASME Code components of the GDCS conform with the requirements in the ASME Code, Section III.
12. Portions of the GDCS are classified as ASME Code class as indicated in Subsection 2.1.2. They are designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III.	12. ASME Code Data Reports will be reviewed and inspections of Code stamps will be conducted for ASME components in the GDCS.	12. Those portions of the GDCS identified as ASME Code Class in Subsection 2.1.2 have ASME Code Section III, Code Data Reports and Code stamps (or alternative markings permitted by the Code).

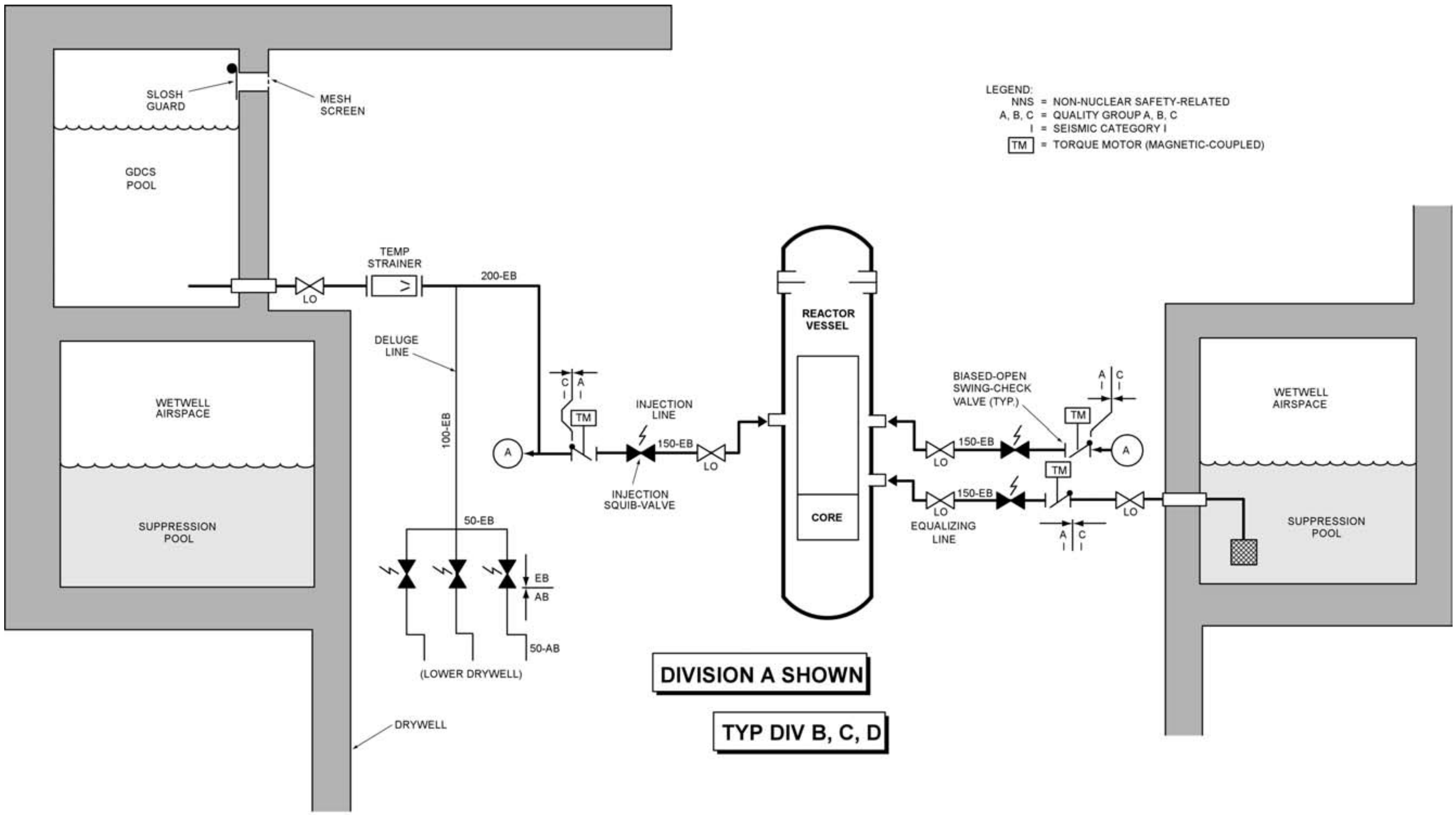


Figure 2.4.2-1. Gravity-Driven Cooling System

2.5 REACTOR SERVICING EQUIPMENT

The following subsections describe the major reactor servicing equipment for the ESBWR.

2.5.1 Fuel Servicing Equipment

No entry for this system.

2.5.2 Miscellaneous Servicing Equipment

No entry for this system.

2.5.3 Reactor Pressure Vessel Servicing Equipment

No entry for this system.

2.5.4 RPV Internals Servicing Equipment

No entry for this system.

2.5.5 Refueling Equipment

Design Description

The Reactor Building (RB) is supplied with a refueling machine for fuel movement.

The RB refueling machine is a gantry-type crane that spans the reactor vessel cavity and fuel and storage pools to handle fuel and perform other ancillary tasks. It is equipped with a traversing trolley on which is mounted a telescoping mast and integral fuel grapple. An auxiliary hoist is also provided. The machine is a rigid structure built to ensure accurate and repeatable positioning during the refueling process.

The refueling machine is classified as nonsafety-related, but designed as Seismic Category I.

A position indicating system and travel limit computer are provided to locate the grapple over the vessel core and prevent collision with pool obstacles. The mast grapple has a redundant load path so that no single component failure results in a fuel bundle drop. Interlocks on the machine:

(1) Prevent hoisting a fuel bundle over the vessel unless an all-control-rod-in permissive is present;

(2) Limit vertical travel of the fuel grapple to provide shielding over the grappled fuel during transit; and

(3) Prevent lifting of fuel without grapple hook engagement and load engagement.

The fuel handling machine is only used for fuel servicing and transporting tasks in the Fuel Building. It is equipped with a traversing trolley on which is mounted a telescoping mast and integral fuel grapple. An auxiliary hoist is also provided. The machine is a rigid structure built to ensure accurate and repeatable positioning while handling fuel.

The fuel handling machine is classified as nonsafety-related, but designed as Seismic Category I.

A position indicating system and travel limit computer are provided to locate the grapple over the spent fuel storage racks and prevent collision with pool obstacles. The mast grapple has a redundant load path (i.e., two independent 100% load support mechanisms) so that no single component failure results in a fuel bundle drop. Interlocks on the machine:

- Limit vertical travel of the fuel grapple to provide shielding over the grappled fuel during transit; and
- Prevent lifting of fuel without grapple hook engagement and load engagement.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.5.5-1 provides a definition of the inspection, test, and/or analyses, together with associated acceptance criteria, which will be undertaken for the refueling machine.

Table 2.5.5-1

ITAAC For The Refueling Machine

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The RB refueling machine has two auxiliary hoists.	1. Load tests on both auxiliary hoists will be conducted at 125% of rated load.	1. A successful load test of each auxiliary hoist has been performed.
2. The RB refueling machine is provided with controls interlocks which: a. Maintain water shielding over fuel when grappled on mast. b. Allow no fuel movement over vessel when control rod is removed. c. Provide fuel grapple travel limit. d. Prevent collision with fuel pool walls and other structures. e. Interlock grapple hook engagement with hoist load and hoist up power. f. Provides automatic sequencing control for transfer operation.	2. Test will be performed with actual or simulated signals to demonstrate that the interlocks function as required. Tests may utilize a combination of in-situ and off-site tests.	2. The tests have been completed and the results demonstrate that the required interlocks function as required. .
3. The FB refueling machine has two auxiliary hoists.	3. Load tests on both auxiliary hoists will be conducted at 125% of rated load.	3. A successful load test of each auxiliary hoist has been performed.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The FB refueling machine is provided with controls interlocks which: a. Maintain water shielding over fuel when grappled on mast. b. Provide fuel grapple travel limit. c. Prevent collision with fuel pool walls and other structures. d. Interlock grapple hook engagement with hoist load and hoist up power. e. Provides automatic sequencing control for transfer operation.	4. Test will be performed with actual or simulated signals to demonstrate that the interlocks function as required. Tests may utilize a combination of in-situ and off-site tests.	4. The tests have been completed and the results demonstrate that the required interlocks function as required. .

2.5.6 Fuel Storage Facility

New and spent fuel storage facilities are required for fuel and associated equipment.

New Fuel Storage Design Description

New fuel is initially stored in racks of stainless steel construction with neutron absorbing material in the spent fuel pool prior to relocation to the reactor building buffer pool. Fully loaded fuel storage racks shall remain subcritical by 5% Δk, under all conditions.

Spent Fuel Storage Design Description

Spent fuel is stored in spent fuel storage racks in the spent fuel pool and the reactor spent fuel portion of the buffer pool, and are of stainless steel laminate construction with neutron absorbing material. This ensures that a full array or loaded spent fuel remain subcritical by 5% Δk, under all conditions.

Adequate water shielding is always maintained in storage pools by the use of level sensors and design features. All storage pools are constructed with stainless steel liners to form a leak-tight barrier. A leak detection system monitors liner integrity.

The thermal-hydraulic design of the rack provides sufficient natural convection cooling flow to remove decay heat without exceeding 100°C (212°F).

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.5.6-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the new and spent fuel storage racks.

Table 2.5.6-1

ITAAC For The Fuel Storage Racks

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. A full spent fuel rack is subcritical by at least 5% Δk, i.e. $k_{eff} \leq 0.95$.	1. Calculations will be performed to determine k_{eff} for full spent fuel pool storage racks.	1. Calculations have been performed and demonstrate that the maximum calculated $k_{eff} \leq 0.95$.
2. The maximum rack water coolant flow temperature through the rack shall be [≤ 100°C].	2. Calculations will be performed to determine the maximum temperature of the spent fuel racks.	2. Calculations have been performed and demonstrate that the maximum temperature in the spent fuel racks is [< 100°C].

2.5.7 Under-Vessel Servicing Equipment

No entry for this system.

2.5.8 FMCRD Maintenance Area

No entry for this system.

2.5.9 Fuel Cask Cleaning

No entry for this system.

2.5.10 Fuel Transfer System

Design Description

The ESBWR is equipped with an Inclined Fuel Transfer System (IFTS). In general the arrangement of the IFTS consists of a terminus at the upper end in the Reactor Building refueling pool that allows the fuel to be tilted from a vertical position to an inclined position prior to transport to the spent fuel pool in the Fuel Building. There is means to lower the transport device (i.e., a carriage), means to seal off the top end of the transfer tube, and a control system to affect transfer. It has lower terminus in the fuel building storage pool, and a means to tilt the fuel to be removed from the transport cart. There are controls contained in local control panels to affect transfer. There is a means to seal off the upper and lower end of the tube while allowing filling and venting of the tube.

There is sufficient redundancy and diversity in equipment and controls to prevent loss of load (carriage with fuel is released in an uncontrolled manner) and that there are no modes of operation that allow simultaneous opening of any set of valves that could cause draining of water from the upper pool in an uncontrolled manner.

The IFTS has sufficient cooling such that a freshly removed pair of fuel assemblies can remain in the IFTS until they can be removed without damage to the fuel or excessive overheating.

No IFTS component is required to remain operable over the anticipated range of the abnormal events, accidents, or harsh plant environment. However, the IFTS tubes and supporting structure can withstand an SSE without failure of the basic structure or compromising the integrity of adjacent equipment and structures. Therefore, the portion of the IFTS transfer tube assembly from where it interfaces with the upper fuel pool, the portion of the tube assembly extending through the building, the drain line connection, and the lower spent fuel pool terminus equipment [tube, valve, support structure, and bellows] are designated as nonsafety-related and Seismic Category I. The remaining equipment is designated as nonsafety-related and Seismic Category NS.

The IFTS is anchored to the bottom of the refueling pool floor in the Reactor Building. The IFTS penetrates the Reactor Building at an angle down to the fuel storage pool in the Fuel Building.

The IFTS terminates in the fuel storage pool. The lower terminus of the IFTS allows for thermal expansion [axial movement relative to the anchor point in the Reactor Building]. The lower terminus allows for differential movement between the anchor point in the Reactor Building and the fuel pool terminus, and also allows it to have rotational movement at the end of the tube relative to the anchor point in the Reactor Building. The lower end interfaces with the fuel storage pool with a bellows to seal between the transfer tube and the spent fuel pool wall.

The IFTS carriage primarily handles nuclear fuel using a removable insert, and is capable of handling control blades with a separate insert in the transfer cart.

For radiation protection, personnel access into areas of high radiation or areas immediately adjacent to the IFTS is controlled. Access to any area adjacent to the transfer tube is controlled through a system of physical barriers, interlocks and alarms. Specifically,

- Controls prevent personnel from inadvertently or unintentionally being left in those areas at the time the access doors are closed;
- During normal operation or shutdown, personnel are prevented from (a) either reactivating the IFTS while personnel are in a controlled maintenance area, or (b) entering a controlled IFTS maintenance area while irradiated fuel or component are in any part of the IFTS;
- Both an audible alarm and flashing red lights are provided inside and outside any maintenance area immediately adjacent to IFTS for the indicating operation;
- Radiation monitors with alarms are provided both inside and outside any maintenance area; and
- A system of key-locks in both the IFTS main operation panel and in the control room is provided to prevent unauthorized access to any IFTS maintenance area.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.5.10-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Inclined Fuel Transfer System.

Table 2.5.10-1

ITAAC For The Inclined Fuel Transfer System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The IFTS permits functional testing and required maintenance during plant operation.	1. Using installed controls and power supplies, a functional test will be conducted utilizing dummy fuel bundles for demonstrating fuel movement from the refuel pool to the spent fuel pool and return.	1. The as-built IFTS permits functional testing and required maintenance during plant operation.
2. The physical IFTS radiation protection features (described in Subsection 2.5.10) permit functional testing and required maintenance during plant operation.	2. Each feature shall be individually tested using simulated or actual signals.	2. The as-built IFTS permits functional testing and required maintenance during plant operation.
3. The IFTS is designed such that no single malfunction in combination with any single active component failure, or single operator error shall cause the transfer tube to establish an uncontrolled drain path.	3. Inspections will confirm that the IFTS is equipped with a combination of physical controls and interlocks of the water tight barriers that prevent all barriers from being open at any one time.	3. The as-built IFTS has no system malfunction in combination with any single active component failure, or single operator error which would allow the transfer tube to establish an uncontrolled drain path.

2.5.11 Loose Parts Monitoring System

Design Description

The Loose Parts Monitoring System (LPMS) detects loose metallic parts within the RPV. The LPMS detects structure-borne sound that can indicate the presence of loose parts impacting against the RPV internals. The system alarms when the signal amplitude exceeds a preset limit. The LPMS detection system can evaluate some aspects of selected signals. However, the system by itself does not diagnose the presence and location of a loose part.

The LPMS continuously monitors the RPV and appurtenances for indications of loose parts. The LPMS consists of sensors, cables, signal conditioning equipment, alarming monitor, signal analysis and data acquisition equipment, and calibration equipment. The alarm setting is set to meet the sensitivity requirements, and is designed to reduce the effect of background noise and eliminate spurious alarms.

The array of LPMS sensors consists of a set of sensor channels that are strategically mounted on the external surface of the primary pressure boundary at various elevations and azimuths at natural collection regions for potential loose parts. General mounting locations are at the (a) main steam outlet nozzle, (b) feedwater inlet nozzle, (c) control rod drive housings, and (d) standby liquid control nozzle.

The LPMS includes provisions for both automatic and manual start-up of data acquisition equipment with automatic activation in the event the preset alert level is reached or exceeded. The system also initiates an alarm to the control room personnel when an alert condition is reached.

Inspections, Tests, Analyses and Acceptance Criteria

No entries for this system.

2.5.12 Inservice Inspection Equipment

No entry for this system.

2.6 REACTOR AND CONTAINMENT AUXILIARY SYSTEMS

The following subsections describe the auxiliary systems for the ESBWR.

2.6.1 Reactor Water Cleanup/Shutdown Cooling System

Design Description

The Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system has the following primary functions:

- Purifies reactor coolant during normal operation and shutdown.
- Transfers sensible and core decay heat produced when the reactor is in the shutdown condition.
- Provides decay heat removal and high pressure cooling of the primary coolant during periods of reactor isolation (hot standby).
- Implements the overboarding of excess reactor coolant during startup and hot standby.
- Maintains coolant flow from the reactor vessel bottom head to reduce thermal stratification.
- Warms the reactor coolant prior to startup and hydro-testing.

The system consists of two independent trains. Each train includes:

- One non-regenerative heat exchanger (NRHX):
- One regenerative heat exchanger (RHX);
- One low capacity pump;
- One high capacity pump;
- One demineralizer; and
- Associated valves and pipes.

The RWCU/SDC system is classified as a nonsafety-related system; however, its Reactor Coolant Pressure Boundary (RCPB), containment isolation, and detection of system pipe break outside containment functions are safety-related, and thus, those functions are Seismic Category I. The safety-related electrical components are Class 1E and are powered from Class 1E busses.. The electrical power supplies to the two trains are from separate diesel-generator backed electrical sources.

During normal plant operation, the system operates at reduced flow in the cleanup mode continuously withdrawing water from RPV. The water is cooled through the heat exchangers and is circulated by the pump to the demineralizer for removal of impurities. Purified water returns to the RHX where it is reheated, and then flows into the feedwater lines and is returned to the RPV. One train is in operation while the other is in standby.

During shutdown cooling, the RPV water is cooled through the NRHX and returned to the reactor through the feedwater lines. Redundant trains permit shutdown cooling if only one train is available. The cooldown time is extended when using only one train. In the event of loss of

preferred power and the most limiting single active failure, the RWCU/SDC systems brings the RPV to a ≤ 93.3°C (≤ 200°F) cold shutdown condition in conjunction with operation of the Isolation Condensers.

During hot standby and startup, excess water resulting from CRD system purge water injection and expansion during plant heatup is dumped, or overboarded, to the main condenser or the radwaste system to control reactor water level.

The RWCU/SDC system maintains the temperature difference between the reactor dome and the bottom head drain to preclude excessive thermal stratification.

Flow rate, pressure, temperature and conductivity are measured, recorded or indicated, and alarmed if appropriate, in the MCR.

Pumps are provided with interlocks for the automatic operation and with switch and status indication for manual operation from the MCR. Pneumatically-operated and motor-operated isolation valves are automatically and manually actuated.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.1-1 provides the inspections, tests, and/or analyses which will be undertaken for the RWCU/SDC system.

Table 2.6.1-1

ITAAC For The Reactor Water Cleanup/Shutdown Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the system is as shown on Figure 2.6.1-1.	1. Inspections of the as-built system will be conducted.	1. The as-built system conforms with the basic configuration shown in Figure 2.6.1-1.
2. The containment isolation valves close within the specified time upon receipt of actuation signal against design differential pressure.	2. Test and/or anaylses of containment isolation valves operation will be performed.	2. Containment isolation valves will close against its differential pressure within the specified time upon receipt of actuation signal.
3. The ASME portions of the system retain their integrity under internal pressures that will be experienced during service.	3. A hydrostatic test will be conducted on those portions of the system required to be hydrostatically tested by the ASME Code.	3. The results of the hydrostatic test of the ASME portions of the system conform with the requirements in the ASME Code, Subsection III.
4. Control room features provided for system parameters are defined in Subsection 2.6.1.	4. Inspections will be performed on the control room features for the system.	4. Features are available in control room as defined in Subsection 2.6.1.
5. Manual closure of the RPV bottom head isolation valve can be accomplished remotely.	5. Remote manual closure testing of the RPV bottom head isolation valve will be performed by closing the inboard containment isolation valve in the RWCU/SDC system suction line from the RPV bottom head.	5. The RPV bottom head isolation valve can be manually closed remotely.
6. Safety-related components are powered from Class 1E busses.	6. A test of the power availability to safety-related components will be conducted with power supplied from the permanently installed electric power busses.	6. Safety-related components system receive electrical power from Class 1E busses only.

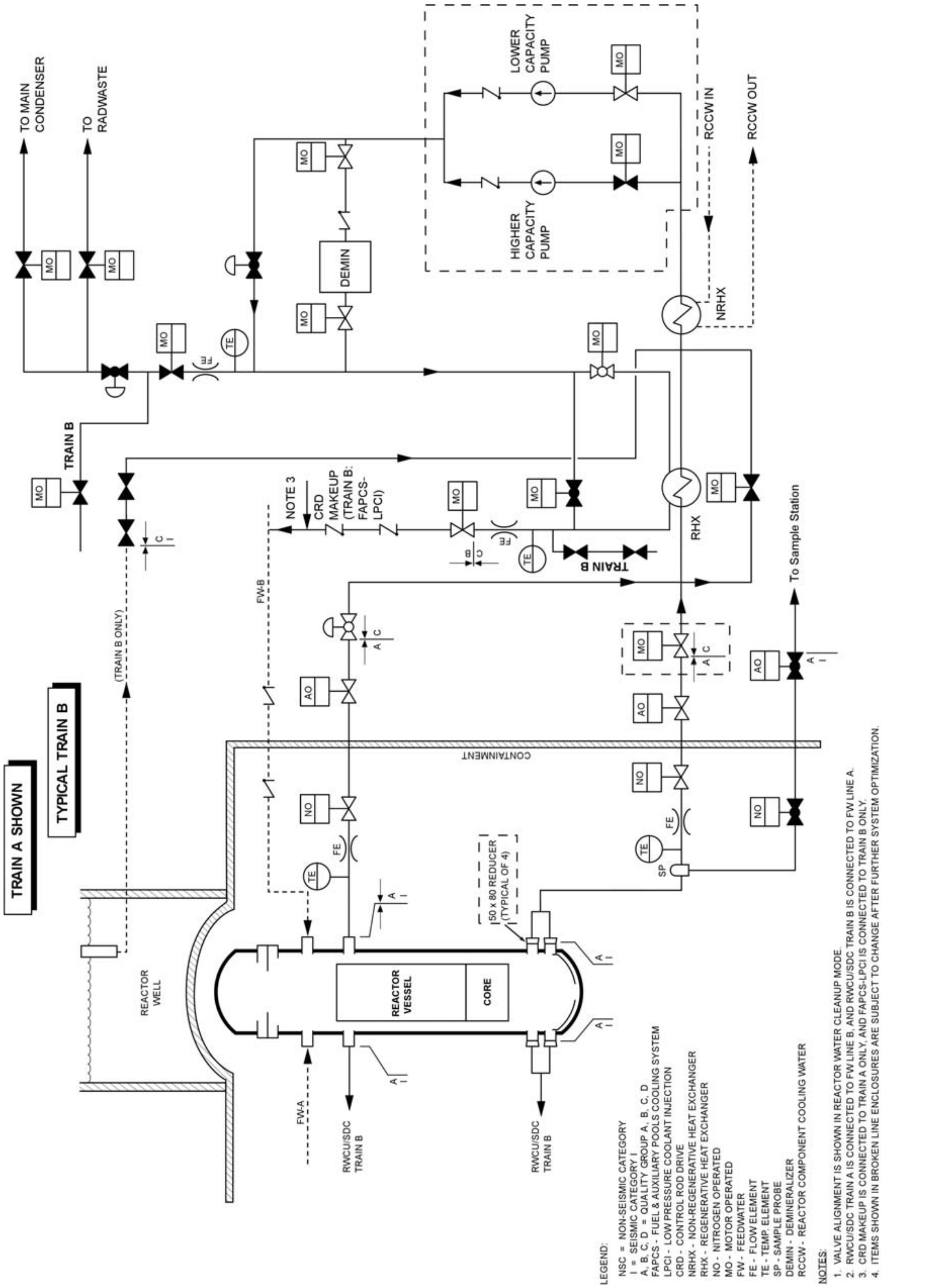


Figure 2.6.1-1. Reactor Water Cleanup/Shutdown Cooling System

2.6.2 Fuel And Auxiliary Pools Cooling System

Design Description

The Fuel and Auxiliary Pools Cooling System (FAPCS) is a nonsafety-related system with the exception of those piping and components required to provide containment isolation and flow paths for emergency makeup of the Isolation Condenser and Passive Containment Cooling (IC/PCCS) pool and the spent fuel pool with water supplies from offsite following an accident. The safety-related FAPCS piping and components and those that are required to support accident recovery functions are constructed to Quality Group B or C and designed to seismic Category I requirements.

The Fuel and Auxiliary Pools Cooling System (FAPCS) consists of two redundant cooling and cleaning (C/C) trains, each with a pump, a heat exchanger and a water treatment unit for cooling and cleaning of pools except the IC/PCC pools. A separate subsystem with its own pump, heat exchanger and water treatment unit is dedicated for cooling and cleaning of the IC/PCC pools independent of the FAPCS C/C train operation during normal plant operation. [Figure 2.6.2-1]

The primary design function of FAPCS is to cool and clean pools located in the containment, reactor building and fuel building, during normal plant operation. Through its piping system, FAPCS provides flow paths for filling and makeup of these pools during normal plant operation and under post-accident condition, as necessary.

FAPCS is also designed to provide the following accident recovery functions:

- Spent fuel pool cooling;
- Suppression pool cooling (SPC);
- Drywell spray;
- Low pressure coolant injection (LPCI) of suppression pool water into the reactor vessel; and
- Alternate shutdown cooling.

At least one FAPCS C/C train is available for continuous operation to cool and clean the water of the spent fuel pool during normal plant operation. The other train can be placed in standby mode or another operating mode. During refueling outages, both trains may be used to provide maximum cooling capacity for cooling the spent fuel pool, if needed.

Each FAPCS C/C train has sufficient flow and cooling capacity to maintain spent fuel pool bulk water temperature below the limit under normal spent fuel pool heat load conditions. Under the maximum spent fuel pool heat load conditions associated with a full core off-load and irradiated fuel in the spent fuel pool for 10 years of plant operations, both trains are needed to maintain the bulk temperature below the limit.

All FAPCS operating modes, except the SPC mode, are manually initiated and controlled by the operator from the main control room. The SPC mode is initiated either manually, or automatically on a high suppression pool water temperature signal. Proper instruments are provided for indication of operating conditions to aid the operator during the initiation and

control of system operation. Provisions are included in the design to prevent inadvertent draining of the pools during FAPCS operation.

Containment isolation valves are provided on the lines that penetrate the primary containment. Containment isolation valves are powered from independent safety-related sources. Air-operated valves with containment isolation function are designed to close upon loss of its electric power supply.

The containment isolation valves that are not required to open for performing an accident recovery function are automatically closed upon receipt of a containment isolation signal from the Leakage Detection and Isolation system (LD&IS). As a result, the containment isolation valves on the suppression pool suction and return lines and drywell spray lines do not receive containment isolation signal to close because these valves must be able to open when FAPCS is initiated to perform an accident recovery function described above. Normally closed isolation valves consisting of an air-operated check valve and a motor-operated gate valve are provided on the LPCI line to separate the low pressure FAPCS piping from the high pressure condition in the RWCU/SDC pipe during reactor power operation.

The isolation valves are provided with a reactor pressure interlock that closes these valves and prevents them from opening whenever a high reactor pressure signal from the Nuclear Boiler System (NBS) is present. Reactor pressure signals are provided to ensure high reliability that the isolation valves are closed.

Instruments and Controls

The water level signals are used to control individual makeup water inlet valves for the automatic makeup of water inventory in the skimmer surge tanks and IC/PCCS pool.

FAPCS C/C train pumps are automatically tripped on the following water level signals:

- Skimmer surge tank low water level;
- Suppression Pool low water level signal from Containment Monitoring System; and
- GDCS pool low and high water level signals from GDCS.

IC/PCCS pool C/C subsystem pump is automatically tripped on the low water level in IC/PCCS pool.

Normally closed isolation valves are provided on the FAPCS LPCI line to separate protect the FAPCS low pressure piping from an overpressurization condition in the RWCU/SDC system piping during a reactor power operation. An interlock design is provided to prevent tThese isolation valves are prevented from opening and to close them, if open, whenever a high reactor pressure signal from the Nuclear Boiler System (NBS) is present.

Upon receipt of a containment isolation signal from the LD&IS, the following valves are closed:

- Inboard and outboard isolation valves on GDCS pool suction line; and
- Outboard isolation valve on GDCS pool return line.

System Operation Modes

FAPCS modes of operation are manually initiated and controlled from the MCR, except for the SPC mode, which is initiated either manually, or automatically on high suppression pool water

temperature signal. The automatic SPC mode initiation logic selects the standby FAPCS C/C train for initiation. The major FAPCS operating modes are discussed below:

Spent Fuel Pool Cooling and Cleanup Mode – During normal plant operation or refueling outages, one of the FAPCS C/C trains operates continuously in this mode to cool and clean the water in the spent fuel pool During this mode of operation, water is drawn from the skimmer surge tanks, pumped through the heat exchanger and water treatment unit to be cooled and cleaned and then returned to the spent fuel pool. The cooling part of this mode may be initiated following an accident.

Fuel and Auxiliary Pool Cooling and Cleanup Mode - During refueling outage, one or both FAPCS C/C trains are placed in this mode of operation to cool and clean the water in the spent fuel pool and pools listed below depending on the heat load condition in these pools.

- Upper fuel transfer pool
- Buffer pool
- Reactor well
- Dryer and separator storage pool

During this mode of operation, water is drawn from the skimmer surge tanks, pumped through the heat exchanger and water treatment unit to be cooled and cleaned and then returned to these pools.

IC/PCCS Pool Cooling and Cleanup Mode – As necessary during normal plant operation, the IC/PCCS pool C/C subsystem is placed in this mode. During this mode of operation, water is drawn via a common suction header from IC/PCCS pool. Water is cooled and cleaned by the IC/PCCS pool C/C subsystem and is then returned to the pool through a common line that branches and discharges deep in the pool.

GDCS Pool Cooling and Cleanup Mode – As necessary during normal plant operation, one of the FAPCS C/C trains that is not operating in spent fuel pool cooling mode can be placed in this mode. In this mode of operation, water is drawn from GDCS pools. The water is cooled and cleaned and is then returned to the pool.

Suppression Pool Cooling and Cleanup Mode – As necessary during normal plant operation, one of the FAPCS C/C trains that is not operating in spent fuel pool cooling mode is placed in this mode. In this mode of operation, water drawn from the suppression is cooled and cleaned and then returned to the suppression pool. The cooling part of this mode may be initiated following an accident.

Low Pressure Coolant Injection (LPCI) Mode - This mode may be initiated following an accident after the reactor has been depressurized to provide reactor makeup water for accident recovery. In this mode the FAPCS pump takes suction from the suppression pool and pumps it into the reactor vessel via RWCU/SDC loop B and then Feedwater loop A.

Drywell Spray Mode - This mode may be initiated following an accident for accident recovery. During this mode of operation, FAPCS draws water from the suppression pool, cools and then spays the cooled water to drywell air space to reduce the containment pressure.

Alternate Shutdown Cooling Mode – This mode may be initiated following an accident for accident recovery. In this mode, FAPCS operates in conjunction with other systems to provide reactor shutdown cooling in the event of loss of other shutdown cooling methods. During this mode of operation, FAPCS flow path is similar to that of LPCI mode. Water is drawn from the suppression pool, cooled and then discharged back to the reactor vessel via LPCI injection flow path. The warmer water in the reactor vessel rises and then overflows into the suppression pool via two opened safety-relief valves on the main steam lines, completing a closed loop for this mode operation.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.2-1 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria, which will be undertaken for the FAPCS.

Table 2.6.2-1

ITAAC For The Fuel and Auxiliary Pools Cooling Cleanup System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The FAPC System configuration is as described in Design Description of Subsection 2.6.2.	1. Inspections of the as-built system will be conducted.	1. The FAPCS configuration is as shown on Figure 2.6.2-1.
2. The safety-related FAPCS piping and components and those that are required to support accident recovery functions are constructed to Quality Group B or C.	2. Conduct hydrostatic tests on those components per ASME Code to demonstrate their ability to retain their pressure boundary integrity under internal pressures that will be experienced during service.	2. The test results meet acceptance criteria defined in ASME Code Subsection III.
3. FAPCS provides flow paths for the emergency makeup of the IC/PCCS pools and the spent fuel pool from the offsite water supplies.	3. Perform a test to confirm flow path from the offsite water sources to the pools.	3. Makeup water flow path is demonstrated and confirmed.
4. FAPCS is capable of providing its design functions.	4. Perform hydraulic tests and/or analyses to determine: • NPSH available. • System hydraulic losses. • Flow rates. • Heat removal rates.	4. Tests and/or analyses exist that demonstrate: • NPSH available is greater than NPSH required as determined by the pump manufacturer. • System hydraulic loss is less than pump developed head. • Flow rate and heat removal rate are equal to or greater than the design values for each operating mode.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. SPC mode is automatically initiated on high suppression pool water temperature.	5. Initiate SPC mode automatically with simulated high suppression pool temperature signals.	5. The control circuit successfully initiates the SPC mode.
6. FAPCS pumps are automatically tripped on trip signals:	6. Conduct a pump trip test with simulated trip signals.	6. The pump successfully tripped as designed.
7. Upon receipt of a containment isolation signal from the LD&IS, the following valves are closed: • Inboard and outboard isolation valves on GDCS pool suction line. • Outboard isolation valve on GDCS pool return line.	7. Perform the following tests: • Logic test with simulated containment isolation signals. • Valve stroke test against the design dP.	7. Test results and/or analyses demonstrate that: • The containment isolation valve automatically closes and cannot be opened. • The valve stroke time is less than the manufacturer standard stroke time.
8. Leakage of all containment isolation valves is acceptable.	8. Perform valve leakrate tests in accordance with Type C valve leakrate test of 10 CFR 50 Appendix J.	8. Leakrate is less than the acceptance criterion established per the leak rate program (or IST).
9. A reactor pressure interlock prevents opening of LPCI injection valve.	9. Perform a logic test with a simulated high reactor pressure signal.	9. The LPCI injection valves automatically close and cannot be opened.
10. Level instruments are provided for monitoring and controlling the water levels in the skimmer surge tanks and IC/PCCS pool.	10. Perform instrument calibration and simulated makeup water control test.	10. Water level indicate accurate water levels. Makeup water control valve open and close upon receipt of water level signals as designed.

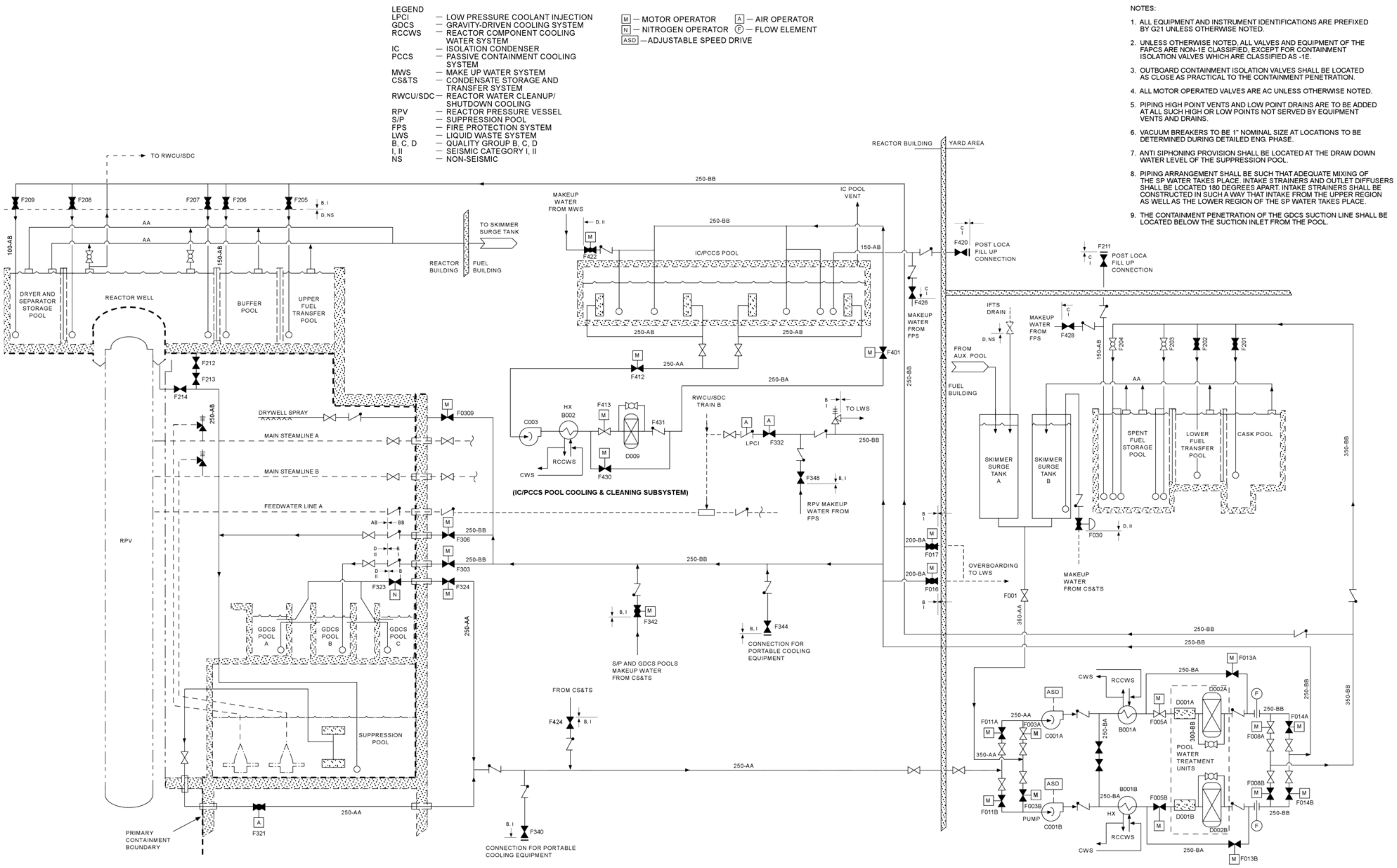


Figure 2.6.2-1. Fuel and Auxiliary Pools Cooling Cleanup System

2.7 CONTROL PANELS

The following subsections describe the different types of control panels and systems for the ESBWR.

2.7.1 Main Control Room Panels

Design Description

The main control room panel (MCR) is comprised of an integrated set of operator interface panels (e.g., main control console, large display panel). The safety-related panels are seismically qualified and provide grounding, electrical independence and physical separation between safety divisions and between safety divisions and nonsafety-related components and wiring.

The main control room panels and other MCR operator interfaces are designed to provide the operator with information and controls needed to safely operate the plant in all operating modes, including startup, refueling, safe shutdown, and maintaining the plant in a safe shutdown condition. Human factors engineering principles have been incorporated into all aspects of the MCR design.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.7.1-1 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria, which will be undertaken for the MCRP.

Table 2.7.1-1

ITAAC For Main Control Room Panels

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Independence is maintained between Class 1E divisional circuits.	1. Inspection of the as-installed Class 1E divisional circuits will be performed.	1. Separation is maintained between Class 1E divisional circuits.
2. Independence is maintained between Class 1E divisional and non-Class 1E circuits.	2. Inspection of the as-installed Class 1E divisional circuits will be performed.	2. Separation is maintained between Class 1E divisional and non-Class IE circuits.
3. Independence is maintained between Class 1E divisional circuits.	3. Tests will be performed by energizing/deenergizing one division at a time and checking for voltage in the division.	3. The voltage in only one division at a time is affected.

2.7.2 Radioactive Waste Control Panels

Design Description

The liquid and solid radwaste systems are operated from control panels in the radwaste control room. Key system alarms are repeated in the Main Control Room.

Programmable controllers are used in this application. They are not safety-related.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.7.3 Local Control Panels And Racks

Design Description

Local panels, control boxes, and instrument racks are provided as protective housings and/or support structures for electrical and electronic equipment to facilitate system operations at the local level. They are designed for uniformity using rigid steel structures capable of maintaining structural integrity as required under seismic and plant dynamic conditions. The term "local panels" includes local control boxes.

Local panels and racks containing equipment used for safety-related functions are classified as safety-related. They are located in areas, in which there are no potential sources of missiles or pipe breaks that could jeopardize modules from more than one division. Each safety-related panel/rack is Seismic Category I, qualified, and provides grounding, and electrical independence and physical separation between safety divisions and nonsafety-related components and wiring.

Electrical power to divisional panels/racks is from AC or DC power sources of the same division as that of each panel/rack itself. Power to the nonsafety-related panels/racks is from the nonsafety-related AC and/or DC sources.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.7.3-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the local panels and racks.

Table 2.7.3-1

ITAAC For Local Control Panels and Racks

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the LCPs is described in Subsection 2.7.3.	1. Inspections of the as-built system will be conducted.	1. The as-built LCPs conform with the basic configuration described in Subsection 2.7.3.
2. Safety-related LCPs are powered from their respective Class 1E divisions. Independence is provided between Class 1E divisions and between Class 1E divisions and non-Class 1E equipment.	2. a. Tests will be conducted in the LCPs by providing a test signal to only one Class 1E division at a time. b. Inspections of the as-built Class 1E divisions in the LCPs will be conducted.	2. a. A test signal exists in only the Class 1E division under test in the LCPs. b. In the LCPs, physical separation or electrical isolation exists between as-built Class 1E divisions. Physical separation or electrical isolation exists between these Class 1E divisions and non-Class 1E equipment.

2.8 NUCLEAR FUEL

The following subsections describe the fuel and control rods for the ESBWR.

2.8.1 Fuel Rods and Bundles

It is intended that the specific fuel to be used in any facility, which has adopted the certified design be in compliance with NRC approved fuel design criteria. This strategy is intended to permit future use of enhanced/improved fuel designs as they become available. However, this approach is predicated on the assumption that future fuel designs are extensions of the basic fuel technology that has been developed for boiling water reactors.

The following is a summary of the principal requirements, which must be met by the fuel supplied to any facility utilizing the certified design.

- NRC-approved analytical models and analysis procedures are applied.
- Future design features will be included in lead test assemblies.
- The generic post-irradiation fuel examination program approved by NRC is maintained.
- The fuel design thermal-mechanical analyses are performed.
- The fuel design evaluations are performed against acceptance criteria.
- Flow pressure drop characteristics are included in the calculation of the operating limit minimum critical power ratio (OLMCPR).

Inspections, Tests, Analyses and Acceptance Criteria

No entries for this topic.

2.8.2 Fuel Channel

Design Description

Any specific fuel channel to be used in any facility, which has adopted the certified design, shall be in compliance with U.S. NRC approved fuel channel design criteria. This strategy is intended to permit future use of enhanced/improved fuel channel designs as they become available. However, this approach is predicated on the assumption that future fuel channel designs are extensions of the basic technology that has been developed for boiling water reactors. The key characteristic of this established BWR fuel channel technology is the use of zirconium-based (or equivalent) fuel channels, which preclude cross-flow in the core region.

The following is a summary of the principal requirements, which must be met by the fuel channel supplied to any facility using the certified design:

- The material of the fuel channel shall be shown to be compatible with the reactor environment;
- The channel is evaluated to ensure that channel deflection does not preclude control rod drive operation; and
- The effects of channel bow are included in the fuel rod critical power evaluations.

Inspections, Tests, Analyses and Acceptance Criteria

No entries for this topic.

2.9 CONTROL RODS

Design Description

Control rods in the reactor perform the functions of power distribution shaping, reactivity control, and scram reactivity insertion for safety shutdown response and have the following design features:

- A cruciform cross-sectional envelope shape;
- A connector at the bottom for attachment to the control rod drive; and
- Contain neutron absorbing materials.

The following is a summary of the principal design criteria, which are met by the control rod:

- The control rod stresses, strains, and cumulative fatigue will be evaluated to not exceed the ultimate stress or strain limit of the material;
- The control rod will be evaluated to be capable of insertion into the core during all modes of plant operation within limits assumed in plant analyses;
- The material of the control rod will be compatible with the reactor environment;
- The reactivity worth of the control rods will be included in the plant core analyses; and
- Prior to use of new design features on a production basis, lead surveillance control rods may be used.

Inspections, Tests, Analyses and Acceptance Criteria

No entries for this system.

2.10 RADIOACTIVE WASTE MANAGEMENT SYSTEM

2.10.1 Liquid Waste Management System

Design Description

The ESBWR Liquid Waste Management System (LWMS) is designed to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences.

The LWMS does not perform or ensure any safety-related function, and is not required to achieve or maintain safe shutdown. The LWMS consists of the following subsystems:

- Equipment (low conductivity) drain subsystem;
- Floor (high conductivity) drain subsystem;
- Chemical drain subsystem; and
- Detergent drain subsystem.

A LWMS Process Diagram depicting all four subsystems is provided in Figure 2.10.1-1.

The LWMS processing equipment is located in the Radwaste Building.

Any discharge release is such that concentrations and quantities of radioactive material and other contaminants are in accord with applicable local, state, and federal regulations.

If the liquid is returned to the plant, it meets the purity requirements for condensate makeup. If the liquid is discharged, the activity concentration is consistent with the discharge criteria of 10 CFR 20 and dose commitment in 10 CFR 50, Appendix I.

Instrumentation & Controls

The Liquid Waste Management System is controlled from the Radwaste Control Room as described in Subsection 2.7.2.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10.1-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Liquid Waste Management System.

Table 2.10.1-1

ITAAC For The Liquid Waste Management System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic description LWMS is in Subsection 2.10.1.	1. Inspection of the as-built system will be conducted.	1. The as-built LWMS conforms with the functional arrangement as described in the Design Description of this Subsection 2.10.1.
2. The ASME Code components of the LWMS 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 LWMS required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the LWMS conform with the requirements in the ASME Code, Section III.
3. Main control room alarms provided for key parameters of the LWMS.	3. Testing will be performed on the main control room alarms for the LWMS.	3. Alarm displays exist or can be retrieved in the main control room.
4. MOVs, having an active safety-related function, shall close under design basis differential pressure, fluid flow, and temperature conditions.	4. Tests of safety-related isolation valves for closing will be conducted under preoperational differential pressure, fluid flow, and temperature conditions.	4. Tests and/or analyses demonstrate that upon receipt of the actuating signal, each MOV closes and is capable of closing under design basis differential pressure, fluid flow, and temperature conditions..
5. The liquid waste system has a discharge line equipped with a radiation monitor. Discharge flow is terminated on receipt of a high radiation signal from this monitor.	5. Tests will be conducted on the as-built liquid waste system using a simulated high radiation signal.	5. The discharge flow terminates upon receipt of a simulated high radiation signal.

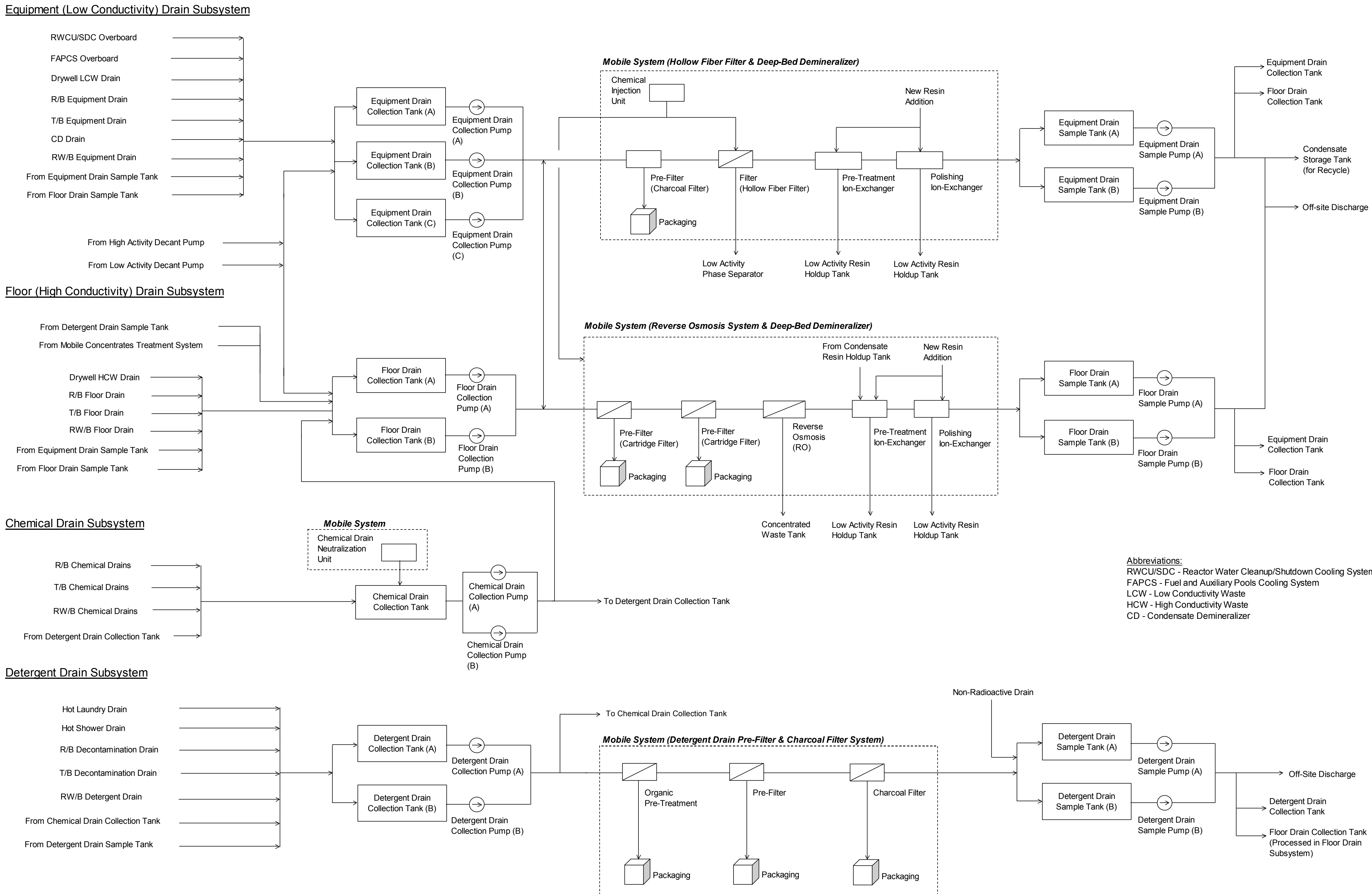


Figure 2.10.1-1. LWMS Process Diagram

2.10.2 Solid Waste Management System

Design Description

The Solid Waste Management System (SWMS) is designed to control, collect, handle, process, package, and temporarily store prior to shipment solid radioactive waste generated as a result of normal operation, including anticipated operational occurrences, that includes filter backwash sludges and bead resins generated by the LWMS, Reactor Water Cleanup/Shutdown Cooling System (RWCU/SDC), Fuel and Auxiliary Pools Cooling System (FAPCS), and Condensate Purification System. Contaminated solids such as High Efficiency Particulate Air and cartridge filters, rags, plastic, paper, clothing, tools, and equipment are also processed in the SWMS. There is no liquid plant discharge from the SWMS.

The SWMS is designed to package the radioactive solid waste for off-site shipment and burial, in accordance with the requirements of applicable NRC and DOT regulations, including Regulatory Guide 1.143, 10 CFR 61, 10 CFR 71, and 49 CFR 170 through 178.

The SWMS is located in the Radwaste Building.

Instrumentation & Controls

The Solid Waste Management System is controlled from the Radwaste Control Room as described in Subsection 2.7.2.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system

2.10.3 Gaseous Waste Management System

Design Description

The gaseous waste management system minimizes and controls the release of gaseous radioactive effluents by delaying, filtering, or diluting various offgas process and leakage gaseous releases that may contain the radioactive isotopes of krypton, xenon, iodine, and nitrogen. The Offgas System (OGS) is the principal gaseous waste management subsystem. The various building HVAC systems perform other gaseous waste functions.

The OGS provides for holdup and decay of radioactive gases in the offgas from the main condenser evacuation system and consists of process equipment along with monitoring instrumentation and control components.

The OGS design minimizes the explosion potential in the offgas process stream through recombination of radiolytic hydrogen and oxygen under controlled conditions. Although the OGS is nonsafety-related, it is capable of withstanding an internal hydrogen explosion without loss of integrity and is designed to ASME Code Section VIII-Division I and the ANSI B31.1 Piping Code.

The OGS process equipment is housed in a reinforced-concrete structure to provide adequate shielding. Charcoal adsorbers are installed in a temperature monitored and controlled vault. The facility is located in the Turbine Building.

The OGS includes redundant hydrogen/oxygen catalytic recombiners and ambient temperature charcoal beds to provide for process gas volume reduction and radionuclide retention/decay. The system processes the main condenser evacuation system discharge during plant startup and normal operation before discharging the air flow to the plant stack.

Control and Monitoring

Control and monitoring of the OGS process equipment is performed both locally and remotely from the main control room. Critical and essential information is available in the main control room. Generally, system control is from the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10.3-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Gaseous Waste Management System.

Table 2.10.3-1

ITAAC For The Gaseous Waste Management System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the OGS is described in Subsection 2.10.3.	1. Inspections will be conducted for the configuration of the OGS.	1. The as-built configuration of the OGS is in accordance with the Design Description in Section 2.10.3.
2. The OGS is designed to withstand internal hydrogen explosions.	2. A hydrostatic test of the OGS will be conducted in the plant in accordance with the ASME VIII-1 and ANSI B31.1 requirements.	2. The hydrostatic test results conform with the ASME VIII-1 and ANSI B31.1 requirements.
3. The OGS is designed to reduce radioactivity leakage through the OGS valve seats and externally into the plant.	3. Leak tests will be performed according to ANSI NDE Testing Standards.	3. The leak test results conform with the ANSI requirements.
4. The OGS automatically controls the OGS flow bypassing or through the charcoal adsorber beds depending on the radioactivity levels in the OGS process gas downstream of the charcoal beds.	4. Tests will be performed as follows: a. A simulated high charcoal gas discharge radioactivity signal will give a Main Control Room (MCR) alarm. b. If the OGS process gas flow is bypassing the main charcoal beds, a simulated high-high charcoal gas discharge radioactivity signal will give a MCR alarm and direct the gas flow through the charcoal beds. c. If a simulated OGS gas discharge radioactivity signal reaches a high-high-high level, a MCR alarm will sound and the off-gas system discharge valve will close.	4.Test results demonstrate that: a. A Main Control Room alarm sounds on an OGS discharge line high radiation signal. b. The OGS charcoal bed valves operate in the main absorber treat mode alignment on a high-high OGS discharge radioactivity signal. c. The OGS discharge valve closes on a high-high-high OGS discharge radioactivity signal.

2.11 POWER CYCLE

The following subsections describe the major power cycle (i.e., generation) systems for the ESBWR.

2.11.1 Turbine Main Steam System

Design Description

The Turbine Main Steam System (TMSS) supplies steam generated in the reactor to the turbine, Moisture Separator Reheaters, steam auxiliaries and turbine bypass valves. The TMSS does not include the seismic interface restraint nor main turbine stop or bypass valves.

The TMSS:

- Accommodates operational stresses such as internal pressure and dynamic loads without failures.
- Provides a seismically analyzed fission product leakage path to the main condenser.
- Has suitable access to permit in-service testing and inspections.
- Closes the Steam Auxiliary (SA) valve(s) on a Main Steam Isolation Valve (MSIV) isolation signal. These valves fail closed on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure.

The TMSS main steam piping consists of four lines from the seismic interface restraint to the main turbine stop valves. The header arrangement upstream of the turbine stop valves allows the valves to be tested on-line and supplies steam to the power cycle auxiliaries, as needed.

The TMSS is nonsafety-related. However, the TMSS is analyzed, fabricated and examined to ASME Code Class 2 requirements, and classified as nonsafety-related. Inservice inspection shall be performed in accordance with ASME Section XI requirements for Code Class 2 piping. Inspection by an ASME authorized nuclear inspector and ASME Code stamping are not required.

Turbine MS piping, including the steam auxiliary valve(s), from the seismic interface restraint to the main stop and main turbine bypass valves is analyzed to demonstrate structural integrity under Safe Shutdown Earthquake (SSE) loading conditions.

The TMSS is located in the steam tunnel and Turbine Building.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.1-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the TMSS.

Table 2.11.1-1

Turbine Main Steam System ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the TMSS is described in Subsection 2.11.1.	1. Inspections of the as-built system will be conducted.	1. The as-built TMSS conforms with the basic configuration description of Subsection 2.11.1.
2. The ASME Code components of the TMSS 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 TMSS required to be hydrostatically tested by the ASME Code.	2. The results of the hydrostatic test of the ASME Code components of the TMSS conform with the requirements in the ASME Code, Section III.
3. Upon receipt of an MSIV closure signal, the SA valve(s) close(s).	3. Using simulated MSIV closure signals tests will be performed on the SA valves.	3. The SA valve(s) close(s) following receipt of a simulated MSIV closure signal.
4. The SA valve(s) fail(s) closed on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure. The pneumatically operated SA valve(s) close(s) when either electrical power to the valve actuating solenoid is lost or pneumatic pressure to the valve(s) is lost.	4. Test will be performed on SA valves.	4. The SA valve(s) close(s) on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure.
5. Turbine MS piping, including the SA valve(s) from the seismic interface restraint to the main stop and main turbine bypass valves are analyzed to demonstrate structural integrity under SSE loading conditions.	5. A seismic analysis of the as-built Turbine MS piping and SA valve(s) will be performed.	5. An analysis report exists which concludes that the as-built Turbine MS piping and SA valve(s) can withstand an SSE without loss of structural integrity.

2.11.2 Condensate and Feedwater System

Design Description

The function of the Condensate and Feedwater System (C&FS) is to receive condensate from the condenser hotwells, supply condensate to the Condensate Purification System (CPS), and deliver feedwater to the reactor. Condensate is pumped from the main condenser hotwell by the condensate pumps, passes through the CPS, low pressure feedwater heaters and feedwater tank to the feedwater pumps, and then is pumped through the high pressure heaters to the reactor. The C&FS boundaries extend from the main condenser outlet to (but not including) the seismic interface restraint outside the containment. The C&FS shall be designed to provide at least 135% of the rated feedwater flow to the RPV for transient situations. The feedwater pump maximum runout capacity with a dome pressure of 7.43 Mpa (1065 psig) shall be less than or equal to 155% of rated flow.

The C&FS is classified as nonsafety-related, and has no safety design basis. No failure within the C&FS could prevent safe shutdown.

The C&FS is controlled by signals from the Feedwater Control System. The C&FS is located in the steam tunnel and Turbine Building.

The C&FS has parameter displays in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.2-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Condensate and Feedwater System.

Table 2.11.2-1

Condensate and Feedwater System ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the C&FS is described in Subsection 2.11.2.	1. Inspections of the as-built system will be conducted.	1. The as-built system conforms with Subsection 2.11.2
2. The maximum runout capacity of an individual feedwater pump against rated RPV head (1065 psig) is equal to 48% $\pm$ 3% of rated feedwater flow as described in Subsection 2.11.2.	2. A test of a single feedwater pump will be conducted at preoperational conditions against the equivalent of the rated RPV head at maximum runout capacity.	2. The Maximum flow from a single feedwater pump is equal to 48% $\pm$ 3% of rated feedwater flow as 1065 psig.

2.11.3 Condensate Purification System

Design Description

The Condensate Purification System (CPS) purifies and treats the condensate, using filtration to remove insoluble solids, and ion exchange demineralizer to remove soluble solids. The CPS consists of full flow high efficiency particulate filters followed by full flow mixed bed demineralizers.

The CPS does not perform or ensure any safety-related function, is classified as nonsafety-related, and thus, has no safety design basis. No failure within the CPS could prevent safe shutdown.

Wastes for the CPS are collected in radiation controlled areas and sent to the radwaste system for processing.

The CPS is located in the Turbine Building.

The CPS has alarms and display for effluent conductivity in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.3-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the CPS.

Table 2.11.3-1

Condensate Purification System ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the CPS is as described in Subsection 2.11.3.	1. Inspections of the as-built System will be conducted.	1. The as-built CPS conforms with the basic configuration as described in Subsection 2.11.3.
2. Alarms and displays are provided in the main control room as described in Subsection 2.11.3.	2. Inspections will be performed on the main control room alarm and display for the CPS.	2. Alarms and displays exist or can be retrieved in the main control room as defined in Section 2.11.3.

2.11.4 Turbine-Generator System

Design Description

The main turbine for the ESBWR reference plant has one high pressure (HP) turbine and three low pressure (LP) turbines. Other turbine configurations may be selected by the COL applicant. The steam passes through a moisture separator reheater (MSR) prior to entering the LP turbines. Steam exhausted from the LP turbines is condensed and degassed in the condenser. Steam is bled off from each turbine and is used to heat the feedwater. The steam and power conversion system is designed to operate above the rated turbine throttle flow for transients and short-term loading conditions.

Turbine Overspeed Protection System

In addition to the normal speed control function provided by the turbine control system, a separate turbine overspeed protection system is included to minimize the possibility of turbine failure and high energy missile damage.

The following component redundancies are employed to guard against overspeed:

- Main stop valves/control valves;
- Intermediate stop valves/intercept valves (CIVs);
- Primary speed control/backup speed control;
- Fast acting solenoid valves/emergency trip fluid system (ETS); and
- Speed control/overspeed trip/backup overspeed trip.

The turbine-generator (TG) system is nonsafety-related and is not needed to effect or support a safe shutdown of the reactor. The turbine generator is orientated within the Turbine Building to be inline with the Reactor Building to minimize the potential for any high energy TG system generated missiles from damaging any safety-related equipment or structures.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.4-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Main Turbine.

Table 2.11.4-1

ITAAC For The Turbine-Generator System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The TG system will be designed to prevent the turbine generator rotor from exceeding the design overspeed with redundant instrumentation, controls and valving, as described in Subsection 2.11.4.	1. a. Inspections of the as-built system will be conducted. b. Tests of the control logic of the as-built overspeed protection system with simulated overspeed signals will be conducted.	1. a. The following provisions to prevent overspeed are in place: Main stop valves/Control valves, Intermediate stop valves/ Intercept valves (CIVs), Primary speed control/ Backup speed control, Fast acting solenoid valves/ Emergency trip fluid system (ETS), and Speed control/Overspeed trip/ Backup overspeed trip b. Valves that supply steam to turbine close upon receipt of overspeed signal.
2. The turbine generator will be orientated to reduce the potential for low trajectory high energy TG system missiles from damaging safety-related equipment or structures, as described in Subsection 2.11.4.	2. Inspections of the as-built Turbine Building and plant arrangements will be conducted.	2. The turbine generator is in line with the Reactor and Control Building.

2.11.5 Turbine Gland Seal System

No entry for this system.

2.11.6 Turbine Bypass System

Design Description

A Turbine Bypass System (TBS) can pass steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the turbine generator. The TBS in the ESBWR reference plant design has the capability to shed 110% of the turbine generator rated load without reactor trip or operation of a SRV. The pressure regulation system provides main turbine control valve and bypass valve flow demands, to maintain a nearly constant reactor pressure during normal plant operation.

The TBS does not perform or ensure any safety-related function, is classified as nonsafety-related, and has no failure within the TBS that could prevent safe shutdown. However, the TBS is used to mitigate anticipated operational occurrences (which per 10 CFR 50, Appendix A are defined as part of normal operations), and is analyzed to demonstrate structural integrity under safe shutdown earthquake (SSE) loading conditions.

The turbine bypass valves are opened by a signal from the Steam Bypass and Pressure Control System.

The turbine bypass valves open upon turbine trip or generator load rejection, automatically trip closed whenever the vacuum in the condenser falls below a preset value, and fail closed on loss of electrical power or hydraulic system pressure. No single failure can disable more than 50% of the installed bypass capacity.

The TBS is located in the Turbine Building.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.6-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the TBS.

Table 2.11.6-1

ITAAC For The Turbine Bypass System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the TBS is described in Subsection 2.11.6.	1. Inspections of the as-built TBS will be conducted.	1. The as-built TBS conforms with the basic configuration of Subsection 2.11.6.
2. The turbine bypass valves are opened by a signal from the Steam Bypass and Pressure Control System, as described in Subsection 2.11.6.	2. Tests will be conducted using a simulated signal.	2. Turbine bypass valves open upon receipt of simulated signal from the Steam Bypass and Pressure Control System.
3. The TBS is analyzed to demonstrate structural integrity under SSE loading conditions, as described in Subsection 2.11.6.	3. A seismic analysis of the as-built TBS will be performed.	3. An analysis report exists which concludes that the as-built TBS can withstand a SSE without loss of structural integrity.
4. No single failure can disable more then 50% of the installed bypass capacity.	4. A failure modes and effects analysis of the as-built design will be performed to ensure that no single failure can disable more then 50% of the installed bypass capacity.	4. An analysis report exists that concludes that no single failure can disable more then 50% of the installed bypass capacity.

2.11.7 Main Condenser

Design Description

The Main Condenser (MC) condenses and deaerates the exhaust steam from the main turbine, provides a heat sink for the TBS, and is a collection point for other steam cycle drains and vents.

The MC hotwell provides a holdup volume for the main steam isolation valve (MSIV) fission product leakage.

The MC is classified as nonsafety-related. However, the supports and anchors for the MC are designed to withstand a safe shutdown earthquake.

The MC is located in the Turbine Building.

The MC tubes are made from corrosion-resistant material.

The MC operates at a vacuum; consequently, leakage is into the shell side of the MC. Circulating water leakage from the tubes to the condenser is detected by measuring the conductivity of sample water extracted beneath the tube bundles. In addition, a conductivity monitor is located at the discharge of the condensate pumps, and alarms are provided in the Main Control Room.

The loss of main condenser vacuum causes a turbine trip, reactor scram, bypass valve closure, and closure of the MSIVs.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.7-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Main Condenser.

Table 2.11.7-1

ITAAC For The Main Condenser

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The supports and anchors for the MC are designed to withstand a safe shutdown earthquake	1. An analysis of the ability of the as-built condenser supports and anchors to withstand a safe shutdown earthquake will be performed.	1. An analysis report exists which concludes that the as-built main condenser supports and anchors are able to withstand a safe shutdown earthquake.
2. The loss of main condenser vacuum causes a turbine trip, reactor scram, bypass valve closure, and closure of the MSIVs.	2. The RPS pressure transmitters located on the main condenser will be tested for HIGH condenser pressure.	3. Tests of the main condenser RPS pressure transmitters causes a turbine trip, reactor scram, bypass valve closure, and closure of the MSIVs with loss of main condenser vacuum on HIGH pressure.
3. Main Control Room alarms provided for the main condenser.	3. Inspections will be performed on the Main Control Room alarms for the main condenser.	3. Verification will be performed that alarms exist in the Main Control Room.

2.11.8 Circulating Water System

Design Description

The Circulating Water System (CIRC) provides cooling water for removal of the power cycle waste heat from the main condensers and transfers this heat to the power cycle heat sink.

The CIRC does not perform, ensure or support any safety-related function, and thus, has no safety design basis.

To prevent flooding of the Turbine Building, the CIRC automatically isolates in the event of gross system leakage. The circulating water pumps are tripped and the pump and condenser valves are closed in the event of a system isolation signal from the condenser area high-high level switches. A condenser area high level alarm is provided in the MCR.

A reliable logic scheme is used (e.g., 2-out-of-3 logic) to minimize potential for spurious isolation trips.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12 AUXILIARY SYSTEMS

The following subsections describe the auxiliary systems for the ESBWR.

2.12.1 Makeup Water System

Design Description

The Makeup Water System (MWS) is comprised of two nonsafety-related subsystems: the demineralization subsystem and the storage and transfer subsystem. The demineralization subsystem produces the demineralized water that is used in non-safety applications. The storage and transfer subsystem distributes water throughout the entire plant. The MWS pumps and demineralization subsystem are only designed for normal power generation demineralized water requirements. During a shutdown/refueling condition, temporary off-site water treatment equipment and pumps are connected to the demineralized water distribution network.

The MWS is a nonsafety-related system, and has no safety design basis other than provision for safety-related containment penetrations and isolation valves.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.12.1-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the MWS.

Table 2.12.1-1

ITAAC For The Makeup Water System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the MWS is as described in Subsection 2.12.1.	1. Inspections of the as-built MWS system configuration will be conducted.	1. The as-built MWS conforms with the basic configuration as described in the Design Description of Subsection 2.12.1.

2.12.2 Condensate Storage and Transfer System

Design Description

The Condensate Storage and Transfer System (CS&TS) stores condensate grade water and transfers it to plant water systems and supply points. End users include the main condenser hotwell, CRD system, RWCU/SDC system, FAPCS, suppression pool and GDCS pools, Condensate and Feedwater System (C&FS), and liquid and solid radwaste system flushing.

The system does not perform or ensure any safety-related function, and is not required to achieve or maintain safe shutdown. Therefore the system is nonsafety-related and has no safety design basis.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12.3 Reactor Component Cooling Water System

Design Description

The Reactor Component Cooling Water System (RCCWS) cools reactor auxiliary equipment including the Chilled Water System, the RWCU/SDC non-regenerative heat exchangers, the FAPCS heat exchangers, Radwaste Building Equipment, and the Standby On-Site AC Power Supply Diesel Generators.

The RCCWS has two trains. Each train consists of parallel pumps, heat exchangers, and a surge tank. Both trains share a chemical addition tank. The Plant Service Water System cools the RCCWS heat exchangers.

A simplified schematic of RCCWS is provided in Figure 2.12.3-1.

The RCCWS is nonsafety-related and Seismic Category NS. The RCCWS is not required to achieve or maintain safe shutdown.

Instrumentation and Control

Information needed for operation and monitoring of the RCCWS is available in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.12.3-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the RCCW System.

Table 2.12.3-1

ITAAC For The Reactor Component Cooling Water System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RCCWS is as described in Subsection 2.12.3.	1. Inspections of the as-built RCCWS system configuration will be conducted.	1. The as-built RCCWS conforms with the basic configuration as described in the Design Description of this Subsection 2.12.3.
2. a. Control room indications and/or controls provided for the RCCWS are as defined in Subsection 2.12.3. b. Remote Shutdown System (RSS) indications and/or controls provided for the RCCWS are as defined in Subsection 2.12.3.	2. a. Inspections will be performed on the control room indications and/or controls for the RCCWS. b. Inspections will be performed on the RSS indications and/or controls for the RCCWS.	2. a. Indications and/or controls exist or can be retrieved in the control room as defined in Subsection 2.12.3. b. Indications and/or controls exist or the RSS as defined in Subsection 2.12.3.

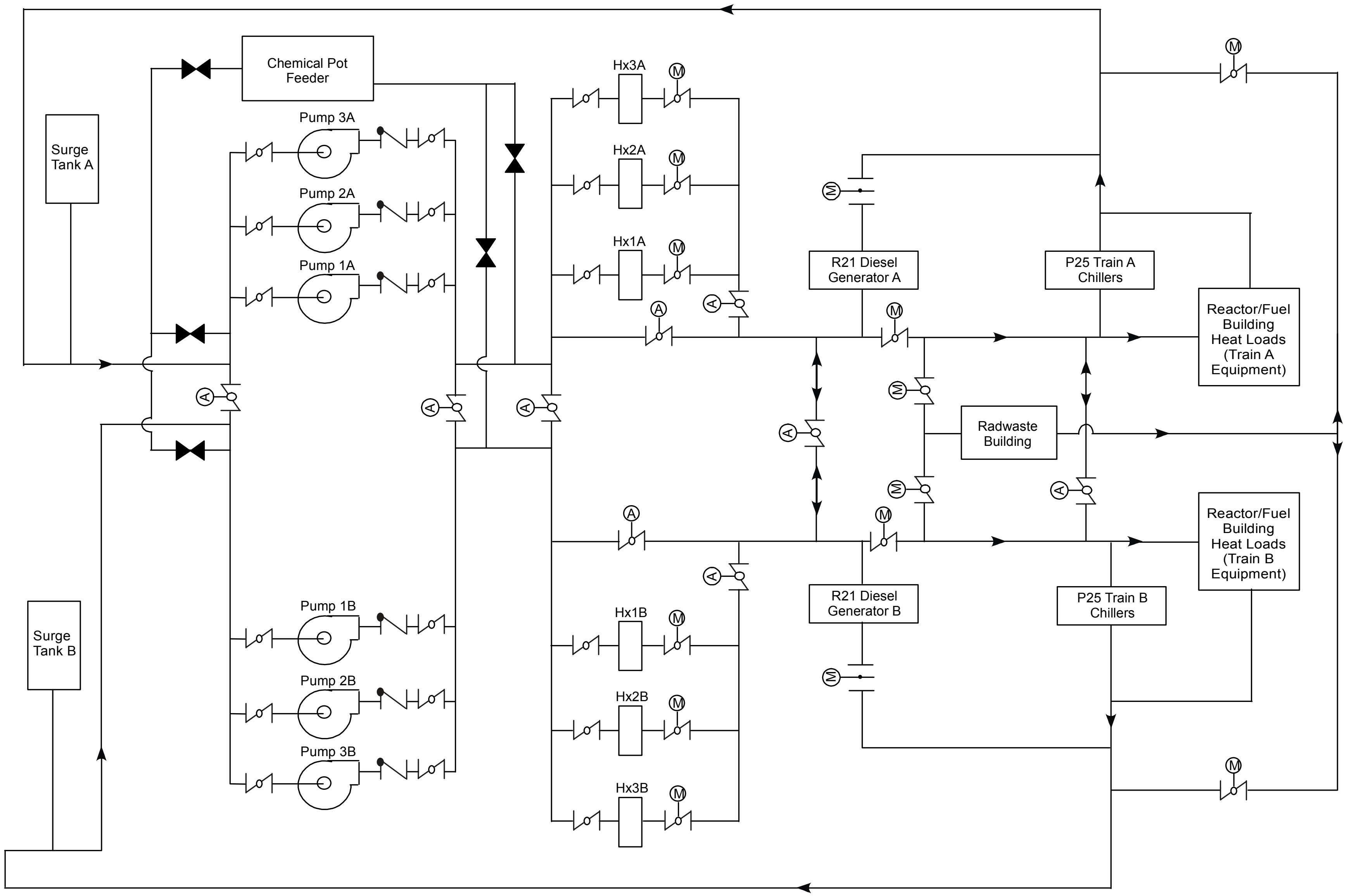


Figure 2.12.3-1. RCCWS Simplified Schematic

2.12.4 Turbine Component Cooling Water System

No entry for this system.

2.12.5 Chilled Water System

Design Description

The Chilled Water System (CWS) is made up of the Nuclear Island Chilled Water Subsystem (NICWS) and the Balance of Plant Chilled Water System (BOPCWS). The NICWS provides chilled water to the air handling units in the Fuel Building, Control Building, Reactor Building, Drywell, RCCWS equipment room, Instrument Air compressor room, NICWS chiller room, the Diesel Generator equipment rooms, and the Technical Support Center. The BOPCWS provides chilled water to the air-handling units in the Turbine Building, Radwaste Building, Electrical Building, and the Hot Machine Shop.

The NICWS consists of two 100% capacity redundant and independent loops with crossties between them. The BOPCWS consists of one 100% capacity loop with crossties between the BOPCWS and both NICWS loops. Each train has a packaged water chiller unit with local control panel, pump, head tank, air separator, and shared chemical feed tank. The NICWS condensers are cooled by the RCCWS and the BOPCWS condensers are cooled by the TCCWS.

A simplified schematic diagram of CWS is provided in Figure 2.12.5-1.

The CWS does not perform or ensure any safety-related function, and is not required to achieve or maintain safe shutdown, except for the containment penetrations, which are required to maintain containment integrity. The containment penetrations are designed to ASME Section III, Class 2, and Seismic Category I.

Instrumentation and Control

Information needed for operation and monitoring of the CWS is available in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Tables 2.12.5-1 and 2.12.5-2 provide definitions of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the NICWS and the BOPCWS.

Table 2.12.5-1

ITAAC For The Nuclear Island Chilled Water Subsystem

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the NICWS is as described in Subsection 2.12.5.	1. Inspections of the as-built system will be conducted.	1. The as-built NICWS conforms with the basic configuration as described in the Design Description of Subsection 2.12.5.
2. Standby chiller unit starts automatically upon failure of the operating unit.	2. Tests will be performed of automatic starting capability of the standby chiller unit by simulating failure of the operating chiller.	2. Chiller units acting as standby units start automatically upon actuation of a simulated failure of the operating unit .
3. The NICWS containment isolation valves automatically close upon receipt of an NICWS isolation signal from LD&IS.	3. Using simulated NICWS isolation signals; tests will be performed on the (NICWS containment isolation valves) isolation logic.	3. Upon receipt of a simulated isolation signal, the NICWS containment isolation valves automatically close.
4. The NICWS heat exchangers provide sufficient heat removal capacity to support the system loads as specified in Subsection 2.12.5.	4. Inspections and analyses will be performed to verify the heat removal capacities of the as-built NICWS heat exchangers.	4. Heat removal capacities of the NICWS heat exchangers are greater than or equal to [7.8×10^6W (26.62×10^6 Btu/h)].

Table 2.12.5-2

ITAAC For The Balance of Plant Chilled Water Subsystem

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the BOPCWS is as described in Subsection 2.12.5.	1. Inspections of the as-built system will be conducted.	1. The as-built BOPCWS conforms with the basic configuration as described in the Design Description of Subsection 2.12.5.
2. BOPCWS chillers are automatically powered from on-site diesel generators during a LOPP.	2. Tests will be performed of the automatic operation of each loop during LOPP with only on-site diesel generator power available.	2. Chiller units acting as standby units are automatically powered from the on-site diesel generators during LOPP and simulated failure of the operating unit.

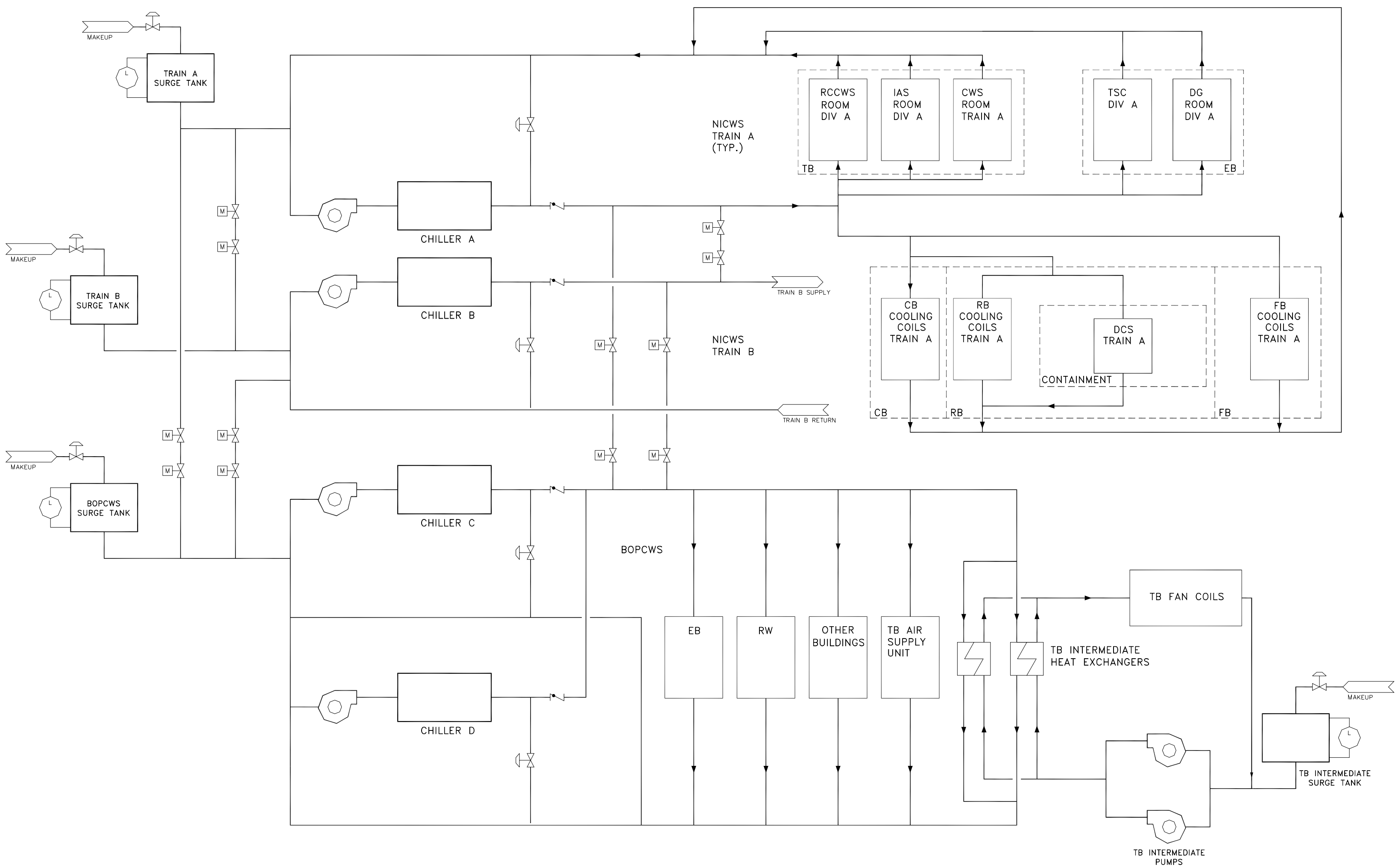


Figure 2.12.5-1. Basic Configuration of the Chilled Water System

2.12.6 Oxygen Injection System

No entry for this system.

2.12.7 Plant Service Water System

Design Description

The Plant Service Water System (PSWS) consists of two independent and 100% redundant open trains that continuously circulate water through the RCCWS and TCCWS heat exchangers. The heat removed is rejected to either the normal power heat sink (NPHS) or to the auxiliary heat sink (AHS).

In the event of a LOPP, the PSWS supports the RCCWS in bringing the plant to cold shutdown conditions in 36 hours assuming the most limiting single active failure. The Plant Service Water System (PSWS) consists of redundant pumps supplying cooling water to the Reactor Component Cooling Water (RCCW) and Turbine Component Cooling Water (TCCW) system heat exchangers.

The PSWS does not perform or ensure any safety-related function, and is not required to achieve or maintain safe shutdown. Therefore the system is not safety-related and has no safety design basis.

Instrumentation and Control

Informatrion needed for operation and monitoring of the PSWS is available in the main control room .

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12.8 Service Air System

Design Description

During normal operation, the Service Air System (SAS) provides a continuous supply of compressed air for general plant use and service air outlets. The SAS consists of two identical trains in parallel, one normally operating, and the other in standby. Each compressor train is equipped with an intercooler, aftercooler, moisture separator, and a service air receiver. Both air compressor trains are connected to a common header, which distributes air to the breathing air purifiers, Turbine Building, Electrical Building, Nuclear Island, and the Radwaste Building. SAS provides a backup source of compressed air for IAS.

The system is nonsafety-related and Seismic Category NS,

Instrumentation and Control

Essential information is available in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.12.8-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Service Air System.

Table 2.12.8-1

ITAAC For The Service Air System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of SAS is as described in Subsection 2.12.8.	1. Inspections of the as-built system configuration will be conducted.	1. The as-built system conforms with the basic configuration described in the Design Description of Subsection 2.12.8.
2. The ASME portions shall retain their integrity under internal pressures that will be experienced during service.	2. A pressure test will be conducted on those portions that are required to be pressure tested by the ASME Code.	2. Results of the pressure test conform with the requirements in ASME Code Section III.

2.12.9 Instrument Air System

Design Description

During normal operation, the Instrument Air System (IAS) provides dry, oil free, filtered compressed air for valve actuators, nonsafety-related instrument control functions, and general instrumentation and valve services outside of containment. The instrument and control systems inside containment are supplied by gaseous nitrogen from the High Pressure Nitrogen Supply System (HPNSS) during normal plant operation. During maintenance outages, the IAS provides compressed air to the nitrogen users located inside containment by way of the HPNSS piping. The IAS includes features that ensure operation over the full range of normal plant operations. The IAS operates during normal plant operation, plant startup and plant shutdown.

The system is nonsafety-related and Seismic Category NS, however, the IAS is designed to be functional after a Safe Shutdown Earthquake (SSE).

Instrumentation and Control

Pressure indication and alarms are provided in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.12.9-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Instrument Air System.

Table 2.12.9-1

ITAAC For The Instrument Air System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of IAS is as described in Subsection 2.12.9.	1. Inspections of the as-built system configuration will be conducted.	1. The as-built system conforms with the basic configuration described in the Design Description of Subsection 2.12.9.
2. The ASME portions shall retain their integrity under internal pressures that will be experienced during service.	2. A pressure test will be conducted on those portions that are required to be pressure tested by the ASME Code.	2. Results of the pressure test conform with the requirements in ASME Code Section III.

2.12.10 High Pressure Nitrogen Supply System

Design Description

The High Pressure Nitrogen Supply System (HPNSS) consists of distribution piping between the Containment Inerting System (CIS) and the containment nitrogen users. The HPNSS is a backup to the CIS.

The containment high pressure nitrogen consumers include the Nuclear Boiler System (NBS) Safety Relief Valve (SRV) Automatic Depressurization System (ADS) accumulators, Isolation Condenser steam and condensate line isolation valve accumulators, and the Main Steam Line Isolation Valve (MSIV) accumulators. These high pressure nitrogen consumers are normally served by the CIS. The HPNSS provides high pressure nitrogen gas to the nitrogen consumers during periods when the Containment Inerting System fails to maintain the required nitrogen supply pressure. The HPNSS provides a stored supply of high-pressure nitrogen gas to the SRV ADS function accumulators to compensate for nitrogen leakage during SRV actuation .

This system is nonsafety-related and Seismic Category NS except for safety-related containment penetrations, and isolation valves. These components are safety-related, and Seismic Category I. The SRV ADS accumulators and piping are part of the Nuclear Boiler System.

Instrumentation and Control

Essential information is available in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.12.10-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the High Pressure Nitrogen Supply System.

Table 2.12.10-1

ITAAC For The High Pressure Nitrogen Supply System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of HPNSS is as described in Subsection 2.12.10.	1. Inspections of the as-built system configuration will be conducted.	1. The as-built system conforms with the basic configuration as described in the Design Description of Subsection 2.12.10.
2. The outboard containment isolation valves can be manually actuated from the main control room.	2. Manual actuation testing of the outboard containment isolation valves will be performed.	2. The outboard containment isolation valves manually opens and closes from the main control room.

2.12.11 Auxiliary Boiler System

No entry for this system.

2.12.12 Hot Water System

No entry for this system.

2.12.13 Hydrogen Water Chemistry System

COL holder scope, and there is no entry for this system.

2.12.14 Process Sampling System

Design Description

The Process Sampling System (PSS) collects representative liquid samples for monitoring water quality and measuring system and equipment performance. The PSS provides for continuous and periodic sampling of principal fluid process streams associated with plant operation. Process samples requiring continuous monitoring or special conditioning are routed to one of the PSS sample stations. These sample stations also include provisions for the collection of grab samples to be taken for further laboratory analyses as required.

PSS sample stations are located in the Reactor, Fuel, Turbine, Auxiliary Boiler, and Radwaste Buildings.

The PSS does not perform or ensure any safety-related function, and is not required to achieve or maintain safe shutdown. Therefore the system is nonsafety-related and has no safety design basis. The Post Accident Sampling System is a subsystem of the Containment Monitoring System and is described in Subsection 2.3.3.

Instrumentation and Control

Essential information is available in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12.15 Zinc Injection System

No entry for this system.

2.12.16 Freeze Protection

No entry for this system.

2.13 ELECTRICAL SYSTEMS

The following subsections describe the electrical systems for the ESBWR.

2.13.1 Electrical Power Distribution System

Design Description

On-site power is supplied from either the plant turbine generator or an off-site power source depending on the plant operating status. During normal operation, plant loads are supplied from the main generator through the unit auxiliary transformers. A generator breaker allows the unit auxiliary transformers to stay connected to the grid to supply loads by backfeeding from the switchyard when the turbine is not online.

The isolated phase bus duct provides the electrical interconnection between the main generator output terminals and the low voltage terminals of the main transformers. Non-segregated phase bus ducts provide for the electrical interconnection between the unit auxiliary transformers and the 13.8 kV unit auxiliary switchgear busses. Non-segregated phase bus ducts also provide for the electrical interconnection between the 6.9kV (split secondary transformer) unit auxiliary transformer and the Plant Investment Protection busses. The two unit auxiliary transformers secondary windings are each associated with two unit auxiliary switchgear busses and two Plant Investment Protection (PIP) load busses and are physically separated and electrically isolated to minimize the likelihood of simultaneous failure. Two reserve auxiliary transformers supply power identical to the unit auxiliary transformers, through segregated phase bus ducts, to both the unit auxiliary switchgear busses and the PIP busses in the event the off-site normal preferred power supply fails and must be switched to the off-site alternate preferred power supply.

Multiple individual voltage regulating transformers supply nonsafety-related control and instrument power.

The PIP (6.9 KV second tier non-safety buses) supply power through parallel isolation step-down transformers and breakers to the four (4) nonsafety-related, 480VAC, isolation power center buses.

The plant design and circuit layout of the UPS provide physical separation of the equipment, cabling, and instrumentation essential to plant safety. Equipment of each division of the Class 1E UPS distribution system is located in an area separated physically from the other divisions. No provisions exist for the interconnection of Class 1E UPS busses of one division with those of another division or non-division. All components of Class 1E AC systems are housed in Seismic Category I structures.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.13.1-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Electrical Power Distribution System.

Table 2.13.1-1

ITAAC For The Electrical Power Distribution System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the Electrical Power Distribution System is described in Subsection 2.13.1.	1. Inspection of the as-built system will be conducted.	1. The as-built Electrical Power Distribution System conforms with the basic configuration described in Subsection 2.13.1.
2. Unit Auxiliary Transformers (UATs) are sized to supply their load requirements.	2. Tests and/or analyses will be performed to determine the as-built transformer ratings. Analyses of as-installed loads of the transformers will be performed.	2. The as-built ratings of each UAT is equal to or greater than its as-installed load assignment.
3. The Reserve Auxiliary Transformer (RAT) is sized to provide its load requirement.	3. Tests and/or analyses will be performed to determine the as-built transformer ratings. Analyses of as-installed loads of the transformers will be performed.	3. The as-built ratings of the RAT are equal to or greater than its as-installed load assignment.
4. a. Electrical Power Distribution System independence is maintained between safety-related divisions. b. Electrical Power Distribution System independence is maintained between nonsafety-related load groups. c. Electrical Power Distribution System independence is maintained between safety-related divisions and nonsafety-related load groups.	4. a. Tests will be performed by energizing/de-energizing one division at a time and checking for voltage of divisions. b. Tests will be performed by energizing/de-energizing one nonsafety-related load group at a time and checking for voltage of the load groups. c. Tests will be performed by energizing/de-energizing one safety-related division at a time and checking the voltage in nonsafety-related load groups.	4. a. Only components in the energized division receive voltage. b. Only components in the energized load group receive voltage. c. Only components in the energized load group receive voltage.

2.13.2 Electrical Wiring Penetrations

Design Description - Containment Penetrations

All power, control and instrument circuits pass through the containment wall in electrical penetration assemblies. Separate penetrations shall be provided for medium-voltage, low-voltage power, lighting, control, and instrument circuits. Containment electrical penetrations comply with Regulatory Guide 1.63.

Class 1E circuit separation groups designated Division 1, 2, 3, 4, and Non-Class 1E circuits run through separate penetration assemblies. These penetrations are located so that the physical separation is maintained between separation groups.

Containment electrical penetrations are rated and protected so that a failure of any circuit of a penetration does not result in exceeding the maximum current versus time capability of the penetration in the event of a single failure of a protective device.

All medium and low voltage power circuits passing through electrical penetrations are provided with primary and backup protective devices.

The control circuits, control power circuits, and instrumentation circuits passing through electrical penetrations minimize the need to protect the penetration from the effects of fault or overload currents.

Electrical cables penetrating containment are provided with redundant devices in series with circuit breakers when the maximum fault current can exceed the continuous current rating of the penetration. Where protective devices are used to protect the penetrations, the penetrations can withstand the maximum possible fault and overload currents for the time sufficient for operation of the backup protective devices in case of failure of the primary protective devices.

All Class 1E components are environmentally and seismically qualified to ensure the execution of their safety functions.

Design Description - Fire Barrier Penetrations

Electrical penetrations are provided for conduit and other raceways between fire areas, and the bottom entry through fire barriers into panels and switchgear. Fire integrity is maintained between fire areas by filling the penetration area around cables and around the raceway with a fire retardant material.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.13.2-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the electrical wiring penetrations.

Table 2.13.2-1

ITAAC For Electrical Wiring Penetrations

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Electrical Wiring Penetration is described in Subsection 2.13.2.	1. Inspections of the as-built Electrical Wiring Penetration will be conducted.	1. The as-built Electrical Wiring Penetration conforms with the basic configuration described in Subsection 2.13.2.
2. Electrical penetrations are protected against overcurrent.	2. Analyses for the as-built electrical penetrations and protective features will be performed to assure the penetrations are protected against overcurrent.	2. Analyses for the as-built electrical penetrations and protective features exist and conclude either 1) that the maximum overcurrent of the circuits does not exceed the continuous current rating of the penetration, or 2) that the circuits have redundant overcurrent protective devices in series and that the redundant overcurrent protection devices are coordinated with the penetration's rated short circuit thermal capacity data and prevent over-current from exceeding the continuous current rating of the electrical penetrations.
3. Divisional electrical penetrations only contain cables of one Class 1E division.	3. Inspections of the as-built divisional electrical penetrations will be conducted.	3. Verify that as-built divisional electrical penetrations only contain cables of one Class 1E division.

2.13.3 Direct Current Power Supply

Design Description

The plant direct current (DC) power supply system shall consist of five non-divisional 250 VDC power supplies, two non-divisional 125 VDC power supplies, four 24-hour divisional Class 1E 250 VDC power supplies and two 72-hour divisional Class 1E 250 VDC power supplies.

The four safety-related, (Class 1E) 24 hour DC, power supplies provide power to the Class 1E uninterruptible AC busses through inverters, and to the loads required for safe shutdown. The two 72-hour Class 1E DC power supplies provide power to 120Vac Class 1E inverters for post-accident monitoring after safe shutdown.

Each of the four divisions of Class 1E DC power is separate and independent. Divisions 1 and 2 have both a 72-hour battery and a 24-hour battery while Divisions 3 and 4 only have a 24-hour battery. The DC systems operate ungrounded (with ground detection circuitry) for increased reliability. Each division has a battery and a battery charger fed from its divisional Motor Control Center (MCC). This system is designed so that no single failure in any division prevents safe shutdown of the plant.

The Class 1E DC power supply is designed to permit periodic testing for operability and functional performance.

Nonsafety-related DC power is supplied through four non-Class 1E MCCs in the same manner as the Class 1E DC power. Each of the two load groups receives power from two of the non-Class 1E MCCs. One MCC in each group provides power to a bus through a battery charger.

Alarms annunciate in the Main Control Room to indicate loss of battery chargers and inverters. Computer inputs can then be monitored to determine the source of the problem. Annunciator and computer inputs from Class 1E equipment or circuits are treated as Class 1E and retain their divisional identification up through their Class 1E isolation device. The output circuit from this isolation device is classified as non-Class 1E. The plant design and circuit layout of the DC systems provide physical separation of the equipment, cabling, and instrumentation essential to plant safety. Each 250VDC battery is separately housed in a ventilated room apart from its charger, distribution, and ground detection panels. Equipment of each division of the DC distribution system is located in an area separated physically from the other divisions. All the components of Class 1E 250 VDC systems are housed in Seismic Category I structures. The battery charger output is of a current limiting design. The battery charger output voltage is protected against over voltage by a high voltage shutdown circuit. The over voltage protection feature is incorporated to protect equipment from damage caused by high voltage. When high voltage occurs, the unit disconnects the auxiliary voltage transformer, which results in charger shutdown. An initial composite test of the on-site DC power systems is called for as a prerequisite to initial fuel loading. This test verifies that each battery capacity is sufficient to satisfy a safety load demand profile under conditions of a LOCA and loss of preferred power. Battery capacity tests are conducted. These tests ensure that the battery has the capacity to meet safety load demands.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.13.3-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Direct Current Power Supply.

Table 2.13.3-1

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Direct Current Power Supply is described in Subsection 2.13.3.	1. Inspections of the as-built system will be conducted.	1. The as-built Direct Current Power Supply conforms with the basic configuration described in Subsection 2.13.3.
2. Each Class 1E divisional (Divisions 1, 2, 3, and 4) battery is provided with a normal and a standby battery charger supplied AC power from a MCC in the same Class 1E division as the battery.	2. Inspections of the as-built Class 1E Direct Current Power Supply will be conducted.	2. Verify that each as-built Class 1E divisional (Divisions 1, 2, 3, and 4) battery is provided with a normal and a standby battery charger supplied AC power from a MCC in the same Class 1E division as the battery.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. One set of batteries in divisions one and two are sized to supply its design loads, at the end-of-installed-life, for a minimum of 72 hours without recharging. A second set of batteries in four divisions is sized to supply its design loads, at the end-of-installed-life, for a minimum of 24 hours without recharging.	3. a. Analyses for the as-built Class 1E batteries to determine battery capacities will be performed based on the design duty cycle for each battery.	3. a. Analyses for the as-built Class 1E batteries exist and conclude that one set of Class 1E batteries in divisions one and two has the capacity, as determined by the as-built battery rating, to supply its analyzed design loads, at the end-of-installed-life, for a minimum of 72 hours without recharging and a second set of Class 1E batteries in four divisions has the capacity, as determined by the as-built battery rating, to supply its analyzed design loads, at the end-of-installed-life, for a minimum of 24 hours without recharging.
	b. Tests of each as-built class 1E battery will be conducted by simulating loads which envelope the analyzed battery design duty cycle.	b. The capacity of each as-built Class 1E battery equals or exceeds the analyzed battery design duty cycle capacity.
4. Each Class 1E normal battery charger is sized to supply its respective Class 1E division's normal steady state loads while charging its respective Class 1E battery.	4. Tests of each as-built Class 1E normal battery charger will be conducted by supplying its respective Class 1E division's normal steady state loads while charging its respective Class 1E battery.	4. Each as-built Class 1E normal battery charger can supply its respective Class 1E division's normal steady state loads while charging its respective Class 1E battery.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The Class 1E DC battery and battery charger circuit breakers, DC distribution panels, and their circuit breakers and fuses, are sized to supply their load requirements.	5. a. Analyses for the as-built Class 1E DC electrical distribution system to determine the capacities of the battery and battery charger circuit breakers DC distribution panels, and their circuit breakers and fuses, will be performed. b. Tests of the as-built Class 1E battery and battery charger circuit breakers, DC distribution panels, their circuit breakers and fuses, will be conducted by operating connected Class 1E loads at greater than or equal to the minimum allowable battery voltage and at less than or equal to the maximum battery charging voltage.	5. a. Analyses for the as-built Class 1E DC electrical distribution system exist and conclude that the capacities of Class 1E battery and battery charger circuit breakers, DC distribution panels, and their circuit breakers and fuses, as determined by their nameplate ratings, exceed their analyzed load requirements. b. Connected as-built Class 1E loads operate at greater than or equal to the minimum allowable battery voltage and at less than or equal to the maximum battery charging voltage.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.	6.	6.
a. The Class 1E battery, battery chargers, and DC distribution panels are rated to withstand fault currents for the time required to clear the fault from its power source.	a. Analyses for the as-built Class 1E DC electrical distribution system to determine fault currents will be performed.	a. Analyses for the as-built Class 1E DC electrical distribution system exist and conclude that the capacities of as-built Class 1E battery, battery charger, DC distribution panel, and current capacities exceed their analyzed fault currents for the time required, as determined by the circuit interrupting device coordination analyses, to clear the fault from its power source.
b. Class 1E battery, battery charger and DC distribution panel circuit breakers and fuses are rated to interrupt fault currents.	b. Analyses for the as-built Class 1E DC electrical distribution system to determine fault currents will be performed.	b. Analyses for the as-built Class 1E DC electrical distribution system exist and conclude that the analyzed fault currents do not exceed the battery, battery charger and DC distribution panel, circuit breaker and fuse interrupt capacities, as determined by their nameplate ratings.
7. Each Class 1E battery is located in a Seismic Category I structure and in its respective divisional battery room.	7. Inspections of the as-built Class 1E batteries will be conducted.	7. Verify that each as-built Class 1E battery is located in a Seismic Category I structure and in its respective divisional battery room.
8. Class 1E DC distribution panels are identified according to their Class 1E division and are located in Seismic Category I structures and in their respective divisional areas.	8. Inspections of the as-built Class 1E DC distribution panels will be conducted.	8. As-built DC distribution panels are identified according to their Class 1E division and are located in Seismic Category I structures and in their respective divisional areas.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. Class 1E DC distribution system cables and raceways are identified according to their Class 1E division. Class 1E divisional cables are routed in Seismic Category I structures and in their respective divisional raceways.	9. Inspections of the as-built Class 1E DC distribution system cables and raceways will be conducted.	9. Verify that as-built Class 1E DC distribution system cables and raceways are identified according to their Class 1E division. Verify that Class 1E divisional cables are routed in Seismic Category I structures and in their respective divisional raceways.
10. For the Class 1E DC electrical distribution system, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	10. a. Tests will be conducted on the as-built DC electrical distribution system by providing a test signal in only one Class 1E division at a time. b. Inspections of the as-built DC electrical distribution system will be conducted.	10. a. A test signal exists in only the Class 1E division under test in the DC electrical distribution system. b. In the as-built DC electrical distribution 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.
11. MCR alarms and displays provided for the Direct Current Power Supply are as defined in Subsection 2.13.3.	11. Inspections will be conducted on the alarms and displays for the Direct Current Power Supply.	11. Alarms and displays exist or can be retrieved in the MCR as defined in Subsection 2.13.3.

2.13.4 Standby On Site Power Supply

Design Description

Two separate nonsafety-related standby AC diesel power supplies provide separate sources of on-site power for four permanent nonsafety-related plant investment protection load groups when the normal and alternate preferred 6.9kV power supplies are not available.

The plant busses are normally energized by either the main generator or the normal preferred off-site power source, backed up by the alternate preferred power source. Transfer to the standby diesel generators is automatic when all other power supplies capable of feeding the busses are not available.

On a defense-in-depth basis, the Standby On Site Power Supply can ultimately provide power to vital safety-related loads. These loads are powered by uninterruptible safety-related DC power from Class 1E plant batteries, if normal or preferred power is not available.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.13.5 Uninterruptible AC Power Supply

Design Description

The Class 1E uninterruptible AC power supply provides redundant, reliable power to the safety logic and control functions during normal, AOO and accident conditions.

Each of the four divisions of this Class 1E uninterruptible AC power is separate and independent. Each division is powered from an inverter supplied from a Class 1E DC bus. The DC bus receives its power from a divisional battery charger and battery. Provision is made for automatic switching to an alternate Class 1E interruptible supply in case of failure of the inverter.

The Class 1E UPS busses are each supplied independently from their divisional Class 1E inverters, which, in turn, are powered from one of the independent and redundant DC busses of the same division and from their isolation power center. The divisional DC bus is powered through a battery charger connected to its divisional isolation power center, and backed by the division's Class 1E batteries. A static bypass switch is provided for transferring Class 1E UPS AC load from Class 1E inverter output to a direct AC feed from the divisional isolation power center through a regulating transformer should an inverter failure occur.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.13.5-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Uninterruptible AC Power Supply.

Table 2.13.5-1

ITAAC For The Uninterruptible AC Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	1.	1.
a. The basic configuration of the Class 1E (i.e., safety-related) Uninterruptible AC Power Supply is as described in Subsection 2.13.5.	a. Inspections of the as-built systems will be conducted.	a. The as-built the Class 1E Uninterruptible AC Power Supply conforms with the basic configuration described in Subsection 2.13.5.
b. The divisional Class 1E inverters and associated distribution panels are electrically independent and physically separated.	b. Inspections will be performed to confirm that each of the four divisional Class 1E uninterruptible power supplies (UPSs) and their associated distribution panels are electrically independent and physically separate.	b. The Certified Design Commitment is met.
c. Two Class 1E inverters automatically transfer to an alternate Class 1E normal 120 VAC power supply and four inverters transfer to an alternate Class 1E normal 480vAC power supply in case of failure of an inverter.	c. Tests will be conducted to confirm the automatic transfer within each division on their inverter(s) to an alternate Class 1E normal 120 VAC power supply or 480vAC power supply upon receipt of a simulated inverter failure.	c. The Certified Design Commitment is met.
2. Each Class 1E UPS receives power from the isolation power center and 250Vdc battery in the same division as described in Subsection 2.13.5.	2. Inspections will be performed to confirm that the ac and dc power sources for each division is from its associated division.	2. Each UPS receives power from the isolation power center and 250Vdc battery in the same division.

2.13.6 Instrument and Control Power Supply

Design Description

The Instrument and Control Power Supply is nonsafety-related, and provides non-Class 1E single phase power to instrument and control loads that do not require an uninterruptible power source.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.13.7 Communication System

Design Description

The Communication Systems are not safety-related and are classified as non-class 1E. The failure of any communications system does not adversely affect safe shutdown capability.

The Communications System includes a dial telephone system, a power-actuated paging facility, a sound-powered telephone system, and an in-plant radio system . Some elements of the system (such as the off-site security radio system, crisis management radio system, and fire brigade system) are COL applicant scope.

Interface Requirements

Interface requirements are discussed within Section 4.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.13.8 Lighting Power Supply

Design Description

The lighting systems include: the normal, standby, emergency, Main Control Room Emergency Lighting and security lighting systems. The normal lighting system provides illumination under all normal plant conditions, including maintenance, testing, and refueling operations. It is powered by preferred AC from the unit auxiliary nonsafety-related busses. The standby lighting system supplements the normal lighting system and supplements the emergency lighting system in selected areas of the plant. The standby lighting system is normally supplied power from preferred AC power or alternately, from the on-site standby diesel-generators. Both lighting systems are nonsafety-related.

Upon loss of the normal lighting system, the emergency lighting system provides illumination throughout the plant and, particularly, areas where emergency operations are performed (e.g., Main Control Room (MCR), battery rooms, local control stations, ingress/egress routes). It includes self-contained DC battery-operated units for exit and stair lighting. The system supplies at least 108 lux (10 foot-candles) of lighting in those areas of the plant where emergency operations could require reading printed materials or instrument scales. In other areas it provides illumination levels adequate for safe ingress or egress. Inside the MCR, emergency lighting is integrated with standby lighting.

Preferred AC power or, alternately, the on-site standby diesel-generators normally supply the MCR emergency lighting through the 72-hour Class 1E batteries and their busses to their 120VAC Class IE inverters. If these AC sources are not available, the system (excluding self-contained battery units) is supplied by the 72-hour Class 1E batteries through Class 1E inverters. Excluding the self-contained battery lighting units, the MCR emergency lighting system is safety-related, and meets Seismic Category I requirements.

The security lighting system provides lighting for the security center, selected security areas, and the outdoor plant perimeter. The system is normally supplied power by preferred AC or, alternately, by the on-site standby diesel-generators. The security lighting system is further

backed up by a dedicated security standby diesel-generator and a dedicated uninterruptible power supply. The security lighting system is nonsafety-related.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.13.8-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Lighting Power Supply.

Table 2.13.8-1

ITAAC For The Lighting Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The safety-related Emergency Lighting System supplies MCR illumination with at least one out of two divisions of emergency lighting operable for at least 72 hours following a design basis event including the loss of all ac power sources.	1. Inspections and tests will be conducted to confirm that the safety-related 250VDC 72-hour batteries supply Emergency Lighting to the MCR.	1. The Certified Design Commitment is met.
2. Safety-related lighting systems are electrically independent and physically separated. Cables are routed in the respective divisional raceways.	2. Inspections will be performed to confirm that lighting equipment and cables are electrically independent and physically separated between safety divisions and between normal and standby lighting systems.	2. Lighting equipment and cables are electrically independent and physically separated between safety divisions and between normal and standby lighting systems.
3. DC self contained battery-operated lighting units, eight hour rated, are provided for stairways, exit routes and major control areas which could be involved in shutdown or recovery operations.	3. Inspections and tests will be conducted to confirm that self contained battery-operated lighting units are located in stairways, exit routes, and major control areas and that they contain eight hour rated batteries as appropriate.	3. The Certified Design Commitment is met.

2.14 POWER TRANSMISSION

Power Transmission is COL applicant scope.

Interface requirements for off-site power transmission are provided within Section 4.

2.15 CONTAINMENT, COOLING AND ENVIRONMENTAL CONTROL SYSTEMS

2.15.1 Containment System

Design Description

The ESBWR containment, centrally located in the Reactor Building, features a pressure suppression design. The containment consists of a steel lined right circular cylinder reinforced concrete containment vessel (RCCV) fulfilling its design basis as a fission product barrier. The RCCV supports the upper pools whose walls are integrated into the top slab of the containment to provide structural capability for LOCA and testing pressures.

Main features include the upper and lower drywell surrounding the RPV and a wetwell containing the suppression pool that serves as a heat sink during abnormal operations and accidents.

The drywell comprises two volumes: an upper drywell volume surrounding the upper portion of the reactor vessel and housing the steam and feedwater piping, the Safety-Relief Valves (SRVs), Gravity Driven Cooling System (GDCS) pools, main steam drain piping and upper drywell coolers; and a lower drywell volume surrounding the lower portion of the reactor, housing the Fine Motion Control Rod Drives (FMCRDs), neutron monitoring system, equipment platform, lower drywell coolers and drywell sumps. The drywell top opening is enclosed with a steel head removable for refueling operations.

The wetwell comprises two volumes: suppression pool; and wetwell gas space. The gas space above the suppression pool serves as the LOCA blowdown reservoir for the upper and lower drywell nitrogen and non-condensables that pass through drywell-to-wetwell vertical vents, each with three horizontal vents located below the suppression pool surface. The suppression pool water serves as the heat sink to condense steam released into the drywell during a LOCA or steam from SRV actuations.Access into the upper and lower drywells is provided through double sealed personnel locks and equipment hatches. The equipment hatch is removable only during refueling or maintenance outages. A single equipment hatch located in the Reactor Building provides access into the wetwell.

The containment structure maintains its functional integrity at the pressures and temperatures that could follow a design basis accident.

The containment structure is protected from or designed to withstand fluid jet forces associated with outflow from the postulated rupture of any pipe within the containment.

Protection against the dynamic effects from the piping failures is provided for the drywell structure. The drywell structure is also provided protection against the dynamic effects of plant-generated missiles.

The containment structure has design features to accommodate flooding to sufficient depth above active fuel to permit safe removal of fuel assemblies from the reactor core after a postulated Design Basis Accident (DBA).

The containment structure is configured to channel flow from postulated pipe ruptures in the drywell to the suppression pool through vents submerged in the suppression pool, which are designed to accommodate the energy of the blowdown fluid. The containment system's

principal internal structure consists of the structural barrier separating the drywell from the wetwell. This barrier is comprised of the wetwell ceiling (diaphragm floor) and the inboard wall (vertical vent wall) separating the drywell from the wetwell. Both of these structural components are steel structures filled with concrete.

The containment structure and penetration isolation system with concurrent operation of other accident mitigation systems, are designed to limit fission product leakage during and following a postulated DBA to values well below leakage calculated for allowable off-site doses.

Vacuum relief between the drywell volumes and the wetwell gas space is provided by vacuum breakers. Each vacuum breaker has proximity sensors that provide position indication and an alarm in the main control room. The proximity sensors have adequate sensitivity to detect the allowable suppression pool bypass leakage capability of the containment.

An all-steel reactor shield wall of appropriate thickness is provided, which surrounds the RPV to reduce gamma shine on drywell equipment during reactor operation and to protect personnel during shutdowns for maintenance and inservice inspections. The RPV insulation is supported from the internal surface of the reactor shield wall. The reactor shield wall is supported by the pedestal structure.

Table 2.15.1-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Containment System.

Table 2.15.1-1

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Containment System (CS) is as shown on Figure 2.15.1-1.	1. Inspections of the as-built system will be conducted.	1. The as-built Containment System conforms with Figure 2.15.1-1.
2. The primary containment pressure boundary defined in Subsection 2.15.1 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. Maximum calculated drywell and wetwell pressures and temperatures for the design basis accidents are less than their design values.	3. Analyses will be performed using the as-designed CS data.	3. The maximum calculated pressures and temperatures are less than design conditions.
4. The containment provides a barrier against the release of fission products.	4. A containment integrated leak rate test will be conducted as per 10 CFR 50 Appendix J.	4. The containment air leakage rate demonstrated by the integrated leak rate test at the peak containment pressure developed during the bounding case of DBA is less than or equal to 0.5% per day by weight of air in containment free volume determined at the containment design pressure 310 kPa gauge (45 psig) and standard temperature of 20^{o}C (68^{o}F).
5. The pressure boundary of the CS retain their integrity under the design pressure of 310 kPa gauge (45 psig).	5. A containment Structural Integrity Test (SIT) will be conducted per ASME requirements at a test pressure of 1.15 times the design pressure.	5. The results of SIT meet ASME Code requirements for the applied test pressure.

Table 2.15.1-1

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The design differential pressure of the diaphragm floor between the drywell and the wetwell is 241 kPa(d) [35 psid] with higher pressure value on the drywell side.	6. Part of the SIT will test the diaphragm floor and vent wall structure with a test pressure equal to 1.15 times the design differential pressure conducted with the drywell pressure greater than wetwell pressure.	6. The results of SIT meet ASME Code requirements for the applied test pressure.
7. The suppression pool water volume is equal to or greater than low water level volume used in the containment performance safety analysis.	7. Volumetric calculations will be performed using measured pool depth.	7. The suppression pool water volume is equal to or greater than low water level volume used in the containment performance safety analysis.
8. The minimum suppression pool depth is 5.4 meters (17.7 ft).	8. Inspections of the containment suppression pool water level will be performed and measurements taken of the pool depth.	8. The minimum suppression pool depth is 5.4 meters (17.7 ft).
9. The vacuum breaker proximity sensors sensitivity to detect the suppression pool bypass leakage meets its design requirements.	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 proximity sensors and will be compared to the allowable value.	9. The vacuum breaker proximity sensors have adequate sensitivity to detect the allowable suppression pool bypass leakage of the containment.
10. Control Room features provided for the Containment System are defined in Subsection 2.15.1	10. Inspections will be performed on the Control Room features for the Containment System.	10. Features are available in the Control Room as defined in Subsection 2.15.1.

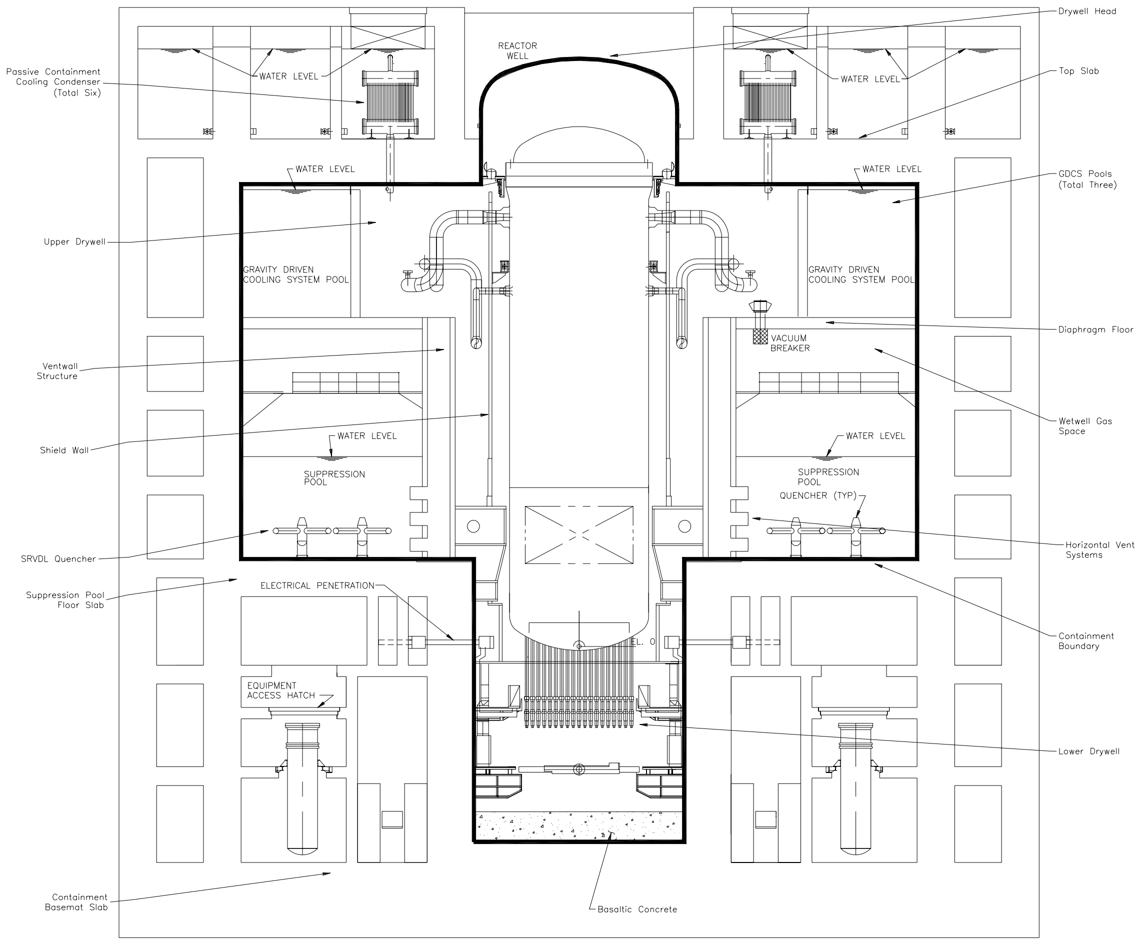


Figure 2.15.1-1. Containment System

2.15.2 Containment Vessel

Design Description

The main structure is a steel lined right circular cylindrical Reinforced Concrete Containment Vessel (RCCV). The RCCV supports the upper pools whose walls are integrated into the top slab of the containment to provide structural capability to carry LOCA and testing pressures.

Additional design details and the ITAAC are covered in Subsection 2.15.1.

2.15.3 Containment Internal Structures

Design Description

The containment internal structures consist of diaphragm floor (DF), vertical vent wall structure (VW), GDCS pool walls, Reactor Shield Wall, and Reactor Vessel Support brackets.

DF, which is the ceiling of the wetwell together with the vertical VW, separates the drywell from the wetwell. Both of these structural components are designed as steel structures filled with concrete. The vertical VW is anchored at the bottom to the RPV pedestal and is restrained at the top by the DF slab.

The GDCS pool structures are supported on top of the DF slab.

RPV support brackets are located at the junction of RPV pedestal and VW structure. These brackets provide structural support to the RPV as well as the Reactor Shield Wall.

An all-steel reactor shield wall of appropriate thickness is provided, which surrounds the RPV to reduce gamma radiation shine on drywell equipment during reactor operation and protect personnel during shutdowns for maintenance and inservice inspections. The RPV insulation is supported from the internal surface of the reactor shield wall. The reactor shield wall is supported on top of the pedestal support structure.

Additional design details and the ITAAC are covered in Subsection 2.15.1.

2.15.4 Passive Containment Cooling System

Design Description

The Passive Containment Cooling System (PCCS) maintains the containment within its pressure limits for DBAs such as a LOCA, by condensing steam from the Drywell atmosphere and returning the condensed liquid to the Gravity Driven Cooling System (GDCS) pools. The system is entirely passive, with no moving parts. No action is required for the PCCS to begin operation.

The PCCS consists of six low pressure, totally independent loops, each containing a steam condenser (passive containment cooling condenser) that condenses steam on tube side and transfers heat to water in a large cooling pool (IC/PCC pool) located outside the primary containment, which is vented to atmosphere.

Each PCCS condenser is located in a subcompartment of the IC/PCC pool. The IC/PCC pool subcompartments on each side of the reactor building communicate at their lower ends to enable full use of the collective water inventory, independent of the operational status of any given PCCS loop. There is no cross connection between the IC/PCC pools.

Each loop, which is open to the containment, contains a drain line to one of the three GDCS pool, and a vent discharge line the end of which is submerged in the pressure suppression pool.

The PCCS loops are driven by the pressure difference created between the containment drywell and the suppression pool during a LOCA, and as such require no sensing, control, logic or power actuated devices for operation.

The PCCS is classified as safety-related and Seismic Category I.

Together with the pressure suppression containment system, the six PCC condensers limit containment pressure to less than its design pressure. The IC/PCC pools contain sufficient water to support operation of the PCC for at least 72 hours after a LOCA without need to make-up to the IC/PCC pool.

The PCC condensers are closed-loop extensions of the containment pressure boundary. Therefore, there are no containment isolation valves and they are always in “ready standby”.

The PCCS can be periodically pressure-tested as part of the overall containment pressure testing program. The PCC loops can be isolated for individual pressure testing during maintenance.

During refueling outages, in-service inspection (ISI) of PCC condensers can be performed, if necessary. Ultrasonic testing of tube-to-heater welds and eddy current testing of tubes can be done with PCCs in place.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.4-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Passive Containment Cooling System.

Table 2.15.4-1

ITAAC For The Passive Containment Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration for the PCCS is as shown in Figure 2.15.4-1.	1. Inspections of the as-built system will be conducted.	1. The as-built PCCS conforms with the basic configuration shown in Figure 2.15.4-1.
2. The ASME Code components of the PCCS 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 PCC System required to be hydrostatically tested.	2. The results of the hydrostatic test of the ASME Code Components of the PCC conform with the requirements in the ASME Code, Section III.
3. The PCCSs are closed loop extensions of the containment pressure boundary to which the containment leakage limits apply.	3. A pneumatic test of the PCCS will be conducted as part of the pre-service containment integrated leak rate test.	3. The overall leakage of the containment system, which includes the PCCSs, is within 10 CFR 50. Appendix J acceptance limits.
4. The PCCS together with the pressure suppression containment system will limit containment pressure to less than its design pressure for 72 hours after a LOCA.	4. An analysis will be performed using similar or more conservative performance characteristics than those of a full scale test unit of established performance capability.	4. Analyzed containment pressure at 72 hours after a LOCA is less than containment design pressure.

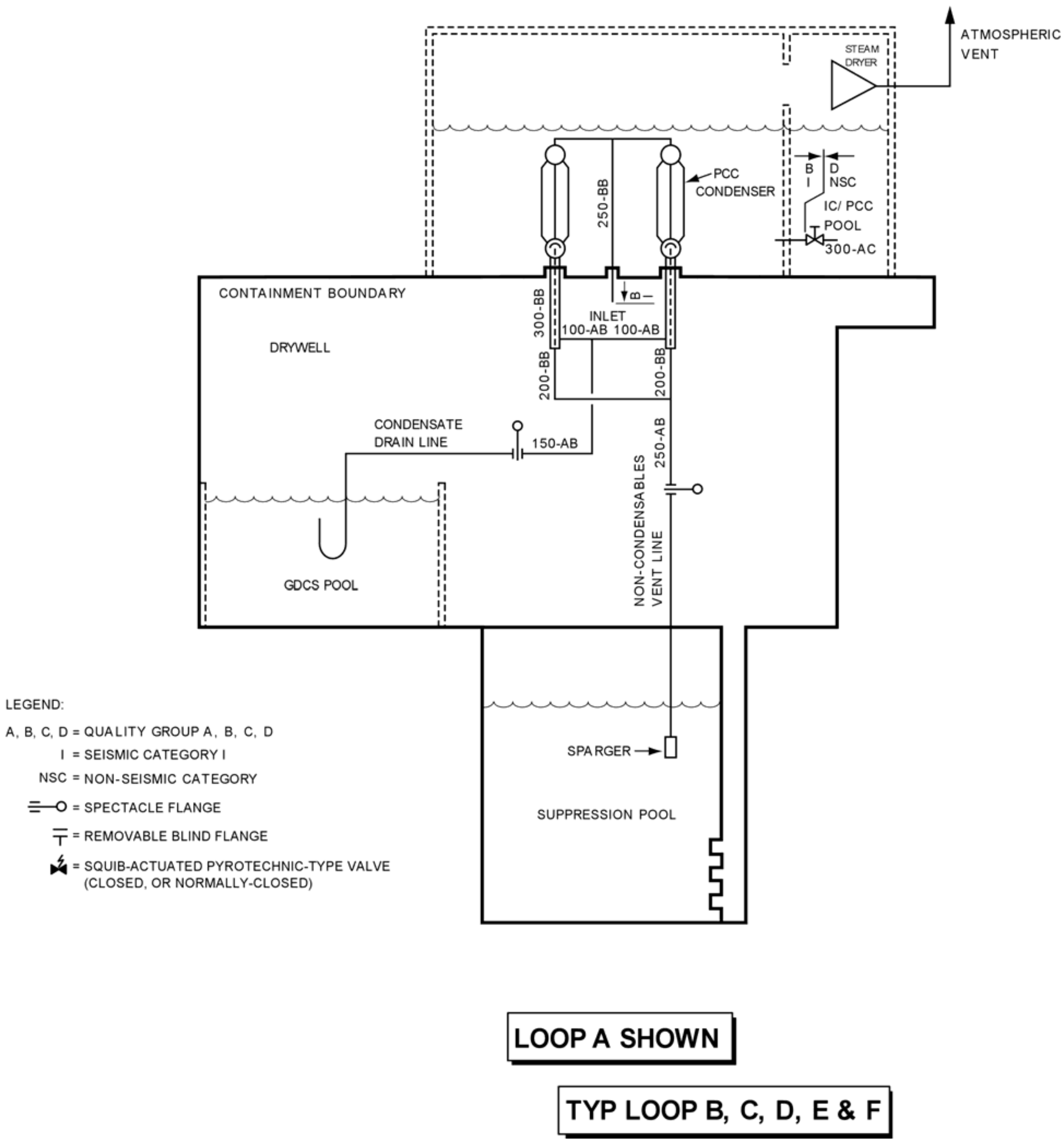


Figure 2.15.4-1. Passive Containment Cooling System Schematic

2.15.5 Containment Inerting System

Design Description

The Containment Inerting System (CIS) establishes and maintains an inert atmosphere within the containment during all plant operating modes, except during plant shutdown for refueling or equipment maintenance and during limited periods of time to permit access for inspection at low reactor power. The objective of the system is to establish conditions that help preclude combustion of hydrogen and thereby prevent damage to safety-related equipment and structures.

The CIS does not perform any safety-related function except for its containment isolation function. Failure of the CIS does not compromise any safety-related system or component nor does it prevent a safe shutdown of the plant. The containment inerting process is a nonsafety-related readiness function, which is not used after the initiation of an accident, and thus, the CIS is not a safety-related system.

The CIS establishes an inert atmosphere (i.e., an oxygen concentration ≤ 3% by volume) throughout the containment following an outage (or other occasions when the containment has become filled with air) and maintains it inert during normal conditions. The system maintains a slight positive pressure in the containment to prevent air (oxygen) in-leakage.

All CIS components are located outside the primary containment.

The CIS includes two inlet lines. A larger capacity line is used to initially inert the containment. A small capacity line is used for makeup.

The CIS includes an exhaust line at the opposite side from the injection points. The discharge line connects to the reactor building Heating Ventilation and Air-Conditioning (HVAC) system exhaust where exit gases are diverted to the plant stack. A small bleed line is also provided for manual pressure control of the containment during normal reactor heatup.

Redundant containment isolation valves provided in the inerting, makeup, exhaust and bleed lines close automatically upon receipt of an isolation signal from the Leak Detection and Isolation System (LD&IS).

During plant startup, a large flow of nitrogen is injected into the drywell and the wetwell. It is then mixed into the containment atmosphere by the drywell cooling fans. The exhaust line is kept open to displace containment resident atmosphere with nitrogen. Once the desired concentration of nitrogen is reached, the exhaust line is allowed to close. When the required inerted containment operating pressure is attained, the inerting process is terminated.

The system can inert the containment to ≤ 4% oxygen by volume within four hours and to ≤ 2% oxygen in the next eight hours. In the longer term, the system maintains the containment atmosphere at less than 3% oxygen by volume during normal operation.

Following shutdown, the containment atmosphere is de-inerted to allow safe personnel access inside the containment. Breathable air from the reactor building HVAC system is injected to the drywell and wetwell air space through the inerting injection line. The incoming air displaces containment gases (mostly nitrogen) into the exhaust line. Vented gases are served by the reactor building HVAC system exhaust fans, filters, and radiation detectors before being diverted to the plant stack. The system de-inerts the containment to an oxygen concentration of ≥ 19% by volume within twelve hours.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.5-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Containment Inerting System.

Table 2.15.5-1

ITAAC For The Containment Inerting System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Containment Inerting System is as shown in Figure 2.15.5-1.	1. Inspections of the as-built system will be conducted.	1. The as-built system conforms to basic configuration shown in Figure 2.15.5-1.
2. The Containment Inerting System isolation valves automatically close upon receipt of an isolation signal.	2. Using simulated isolation signals, tests will be performed on the (Containment Inerting System isolation valves) isolation logic.	2. Upon receipt of simulated isolation signal, the containment isolation valves close.
3. The containment is inerted to less than or equal to 4% oxygen by volume before power operation.	3. Measurement of oxygen concentration by volume will be conducted before power operation.	3. The containment is inerted to less than or equal to 4% oxygen by volume before power operation.

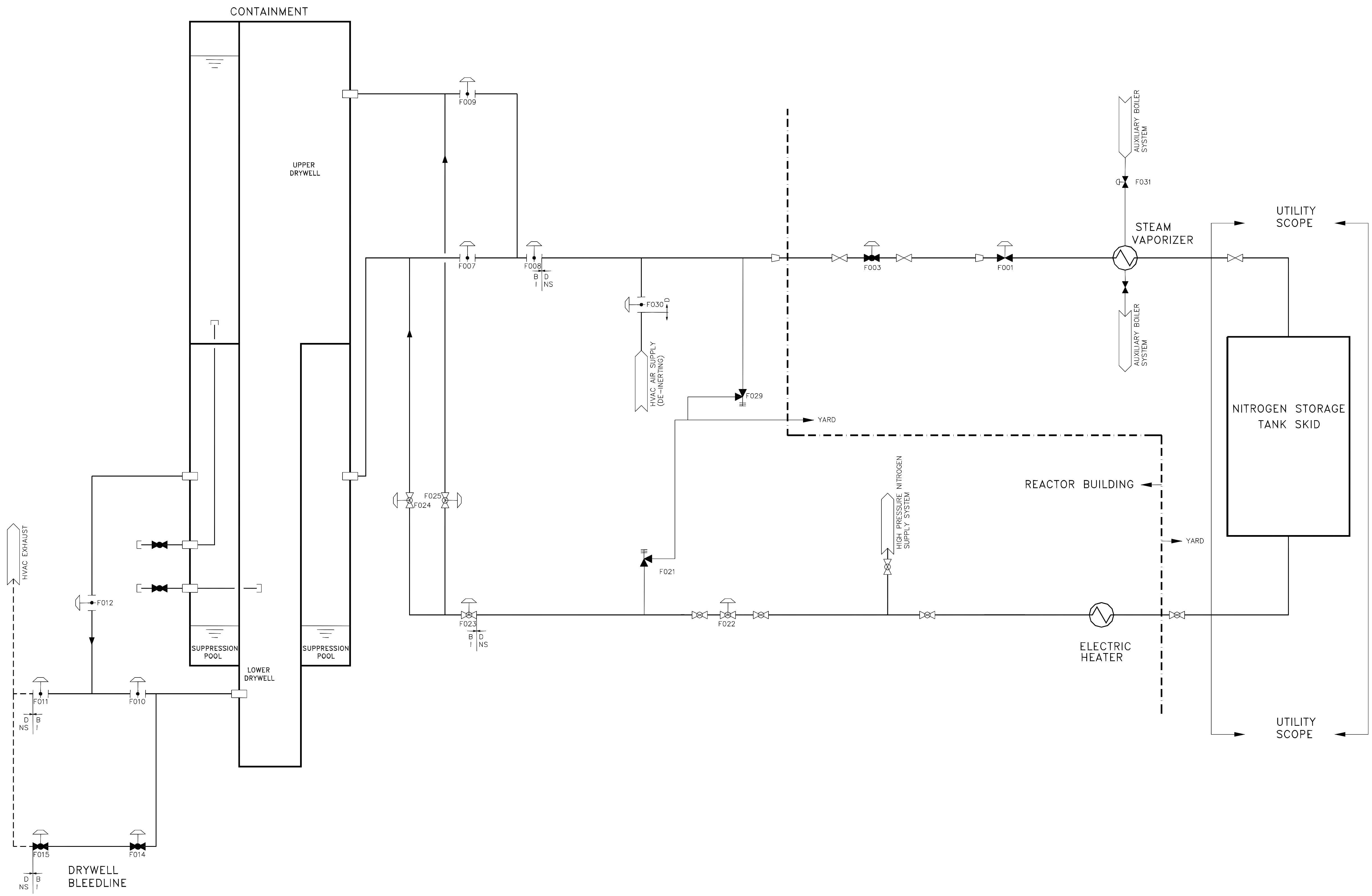


Figure 2.15.5-1. Containment Inerting System Schematic

2.15.6 Drywell Cooling System

Design Description

The Drywell Cooling System (DCS) is nonsafety-related. It consists of four fan coil units (FCUs), two located in the upper drywell, and two in the lower drywell. The system uses the FCUs to deliver cooled air/nitrogen to various areas of the upper and lower drywell through ducts/diffusers. The DCS is a closed loop, recirculating air/nitrogen, cooling system where no outside air is introduced into the system except when the containment is open. During normal plant operation, the DCS is cooled by the Chilled Water System (CWS).

Through the entire plant operating range, from startup to full load condition or from full load to shutdown, the DCS performs the following functions:

- Maintains temperature and humidity in the upper and lower drywell spaces within specified limits during normal operation
- Accelerates drywell cooldown during the period from hot reactor shutdown to cold shutdown
- Aids in complete purging of nitrogen from the drywell during shutdown
- Maintains a habitable environment for plant personnel during plant shutdowns for refueling and maintenance
- Limits drywell temperature during loss of preferred power (LOPP)

Cooled air/nitrogen leaving the FCUs is distributed to the various zones in the drywell through distribution ducts. Return ducts are not provided; the FCUs draw air/nitrogen directly from the upper or lower drywell. A condensate collection pan is provided with each FCU.

A simplified schematic diagram of DCS is provided in Figure 2.15.6-1.

The DCS does not perform any plant safety-related function. Failure of the system does not compromise any safety-related system or component nor does it prevent safe shutdown of the plant.

Instrumentation and Control

Information needed for operation and monitoring of the DCS is available in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.6-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Drywell Cooling System.

Table 2.15.6-1

ITAAC For The Drywell Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the DCS is as described in Subsection 2.15.6.	1. Inspection of the as-built system configuration will be conducted.	1. The as-built DCS conforms with the basic configuration described in the Design Description of Subsection 2.15.6.

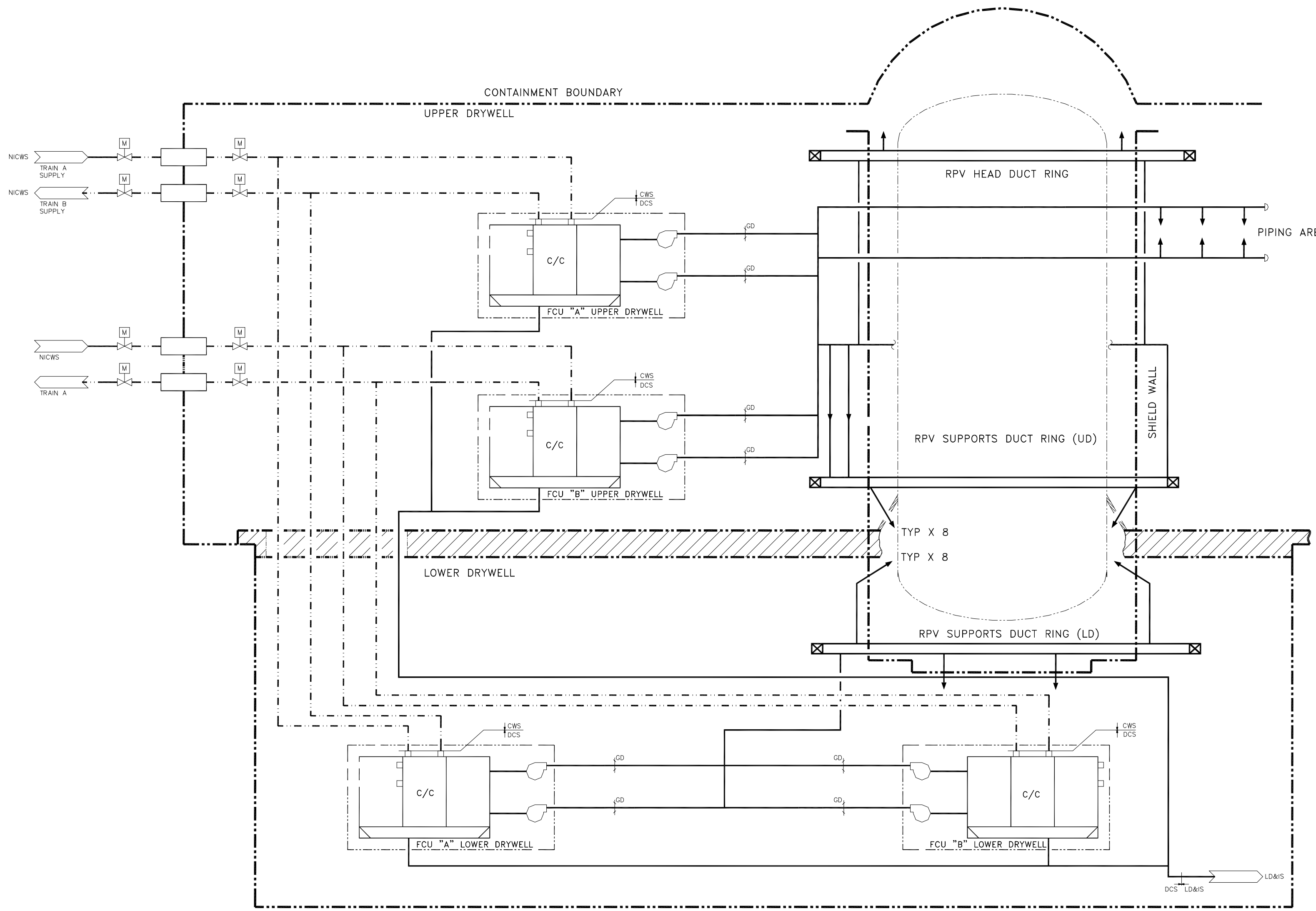


Figure 2.15.6-1 DCS Simplified System Diagram

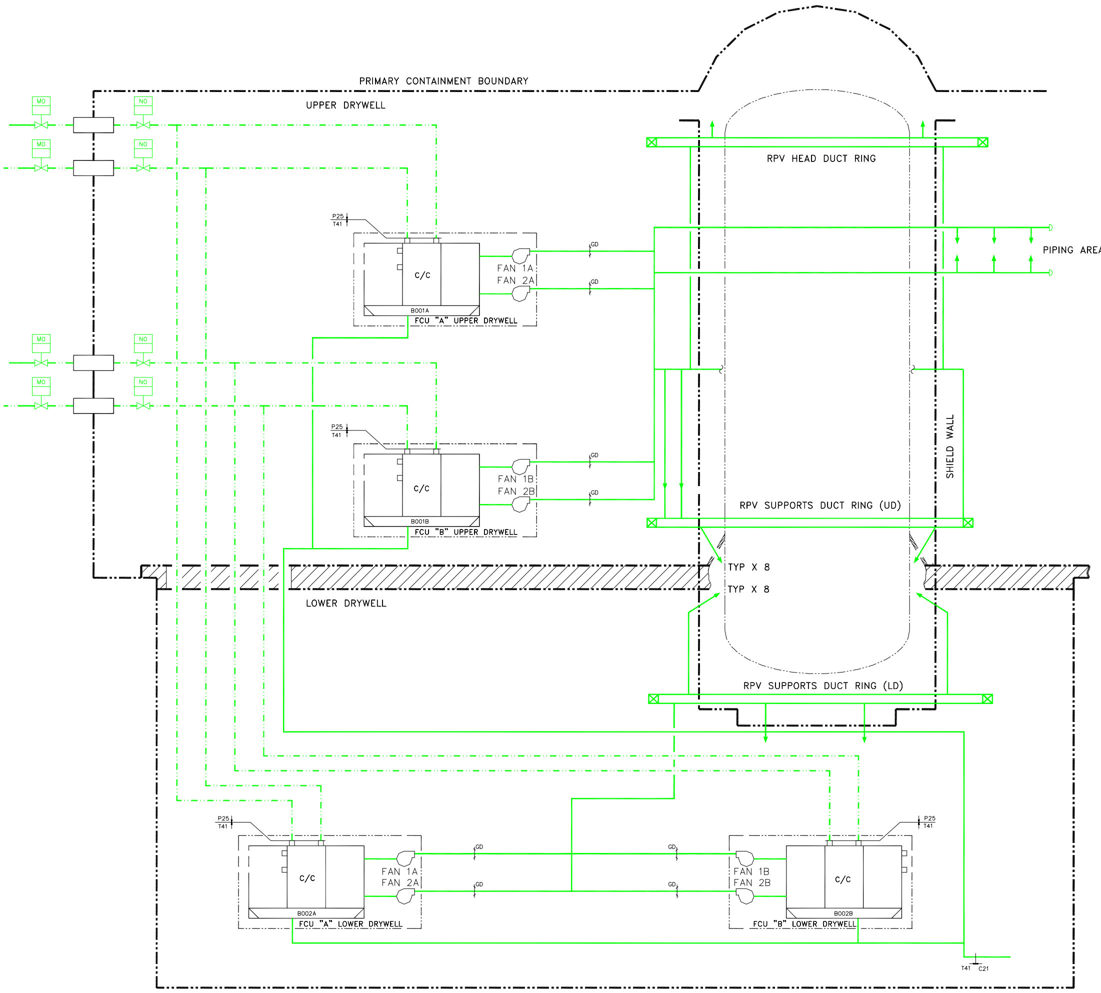


Figure 2.15.6-1
DCS Simplified System Diagram

2.15.7 Containment Monitoring System

Design Description

The Containment Monitoring System (CMS) shall provide the following functions:

- Drywell and Wetwell – Hydrogen, Oxygen concentrations and Gamma radiation levels Monitoring
- Drywell and Wetwell Pressures Monitoring
- Drywell/Wetwell Differential Pressure Monitoring
- Upper Drywell Level Monitoring
- Suppression Pool Water Level Monitoring
- Suppression Pool Temperature Monitoring
- Post-Accident Sampling (PAS)
- Lower Drywell (Post-LOCA) Pool Level Monitoring

The safety-related portions of the CMS are Seismic Category I. Power to each subsystem is provided from uninterruptible Class 1E 120 Vac divisional sources.

Containment atmospheric and drywell monitoring:

The CMS has two independent redundant divisions to monitor the gamma radiation dose rate and the concentrations of hydrogen and oxygen in the drywell and wetwell air during plant operation and following an accident. The channels, which measure gamma radiation in the drywell and wetwell air, are continuously displayed in the MCR.

The drywell pressure instruments provide signals to Leak Detection and Isolation System (LD&IS) and Reactor Protection System (RPS). A drywell pressure increase above normal values indicates the presence of reactor coolant leakage.

Safety-related differential pressure transmitters and nonsafety-related water level transmitters are connected between the drywell and the wetwell to provide, respectively, indication of proper functioning of the wetwell-drywell vacuum breaker system, and to measure containment flooding level in case of a severe accident. The differential pressure instruments are also used for post accident monitoring indications.

The Upper Drywell is monitored by two nonsafety-related channels of water level instrumentation to provide indication.

The Lower Drywell is monitored by two safety-related channels of water level instrumentation.

In the post-accident operational mode, the function of the CMS is to continuously sample the oxygen and hydrogen contents in the containment, and display the results in the main control room. This information is then used by the operator to assess containment status.

Suppression pool monitoring:

Suppression Pool Temperature Monitoring (SPTM) portion of CMS measures the suppression pool temperature and transmits the information to Safety System Logic and Control (SSLC).

SSLC which then averages the temperatures and then sends the average bulk temperature to Reactor Protection System (RPS) for reactor scram. SPTM sends a signal to Fuel and Auxiliary Pools Cooling System (FAPCS) to initiate suppression pool cooling and cleanup function. It also provides signals to Reactor Component Cooling Water System (RCCWS) and for heat load shedding to increase suppression pool cooling. The SPTM consists of four redundant divisions with four levels of temperature elements within each division.

Suppression pool water level monitoring is provided to measure the inventory of suppression pool water. The suppression pool water level is monitored during all plant operating conditions and post accident conditions. Suppression pool water level monitoring consists of eight channels of water level detection sensors distributed into four safety-related narrow range and four wide range instruments. The narrow range suppression pool water level signals are used to detect the uncovering of the first set of suppression pool temperature sensors below the pool surface.

When the suppression pool water level drops below the elevation of a particular set of temperature sensors, those sensor signals are not used in computing the average pool temperature.

Suppression pool temperature and level indications are displayed in the Main Control Room (MCR)

Post Accident Sampling Subsystem:

The Post Accident Sampling subsystem (PASS) consists of sample holding rack, sampling rack, sample conditioning rack, local control panel, and shielding casks. All valves for PASS operation are operated remotely. The sampling system isolation valves are operated from the main control room and all other valves are operated from the local control panel. After the sample vessel has been isolated and removed, the piping is flushed with demineralized water.

The sample holding rack has an enclosure around the sample vessel to contain any leaks of liquids or gases. The liquids drain to the radwaste system and the gases go to the Reactor Building exhaust system.

The PASS isolation valves are connected to a reliable source of power that is available, starting at least one hour after a LOCA. The isolation valves have Class 1E power from differnt divisions.

Gas samples are obtained from a sample line connected to the containment atmosphere monitoring system. A vacuum pump is provided to transfer the gas sample from a sample holding rack to a sampling rack.

Means to reduce radiation exposure are provided such as, shielding, remotely operated valves, and sample transporting casks.

Inspections, Tests, Analyses and Acceptance Criteria

Tables 2.15.7-1 through 2.15.7-3 provide the definitions of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the Containment Atmospheric Monitoring System, the suppression pool monitoring portions of CMS, and the PASS, respectively.

Table 2.15.7-1

ITAAC For The Containment Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of CMS is defined in Subsection 2.15.7.	1. Inspections of the as-built system configuration will be conducted.	1. The as-built CMS conforms with the basic configuration as defined in Subsection 2.15.7.
2. Each safety-related CMS subsystem is electrically and physically separated from each other.	2. Inspections of the as-built system will be performed.	2. Each safety-related CMS subsystem is located in an area physically separated from each other. Also, each safety-related CMS subsystem is powered from different electrical division.
3. The H2/O2 monitoring subsystem of CMS subsystem is activated automatically on either a signal indicating low reactor water level or a signal indicating high drywell pressure.	3. Using simulated electrical signals, CMS testing will be performed.	3. CMS meets the certified design commitment.
4. Each CMS subsystem will initiate separate alarms in the control room when levels exceed the setpoints.	4. Using simulated signal inputs, CMS testing will be performed.	4. Upon receipt of simulated signal inputs, each CMS subsystem initiates separate alarms in the control room.
5. Control room alarms and indications provided for the CMS are as defined in Subsection 2.15.7.	5. Inspections will be performed on the control room alarms and indications for the CMS.	5. Alarms and indications exist or can be retrieved in the control room as defined in Subsection 2.15.7.
6. Each safety-related CMS subsystem is powered from different divisional Class IE busses.	6. A test of the Class IE divisional power availability to each CMS subsystem will be conducted.	6. Each safety-related CMS subsystem receives electrical power from Class IE divisional electrical busses.

Table 2.15.7-2

ITAAC For The Suppression Pool Monitoring

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The equipment comprising the SPTM is defined in Subsection 2.15.7.	1. Inspection of the as-built system will be conducted	1. The as-built SPTM system conforms with the description in Subsection 2.15.7.
2. In each SPTM division, the suppression pool average temperature is calculated by the divisional Safety System Logic and Control (SSLC) logic processors using output signals from the temperature sensors. In each SPTM system division, a high suppression pool average temperature trip signal is generated by the SSLC logic processor and sent to the Reactor Protection System (RPS) when the respective calculated divisional average temperature exceeds the high suppression pool average temperature setpoint.	2. Tests will be conducted in each division of the SPTM using simulated temperature sensor signals.	2. In each SPTM division, a high suppression pool average temperature trip signal is generated by the SSLC logic processor and sent to the RPS when the calculated divisional average temperature exceeds the high suppression pool average temperature setpoint.

Table 2.15.7-2

ITAAC For The Suppression Pool Monitoring

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. Each of the four SPTM divisional logics is powered from its respective divisional Class 1E power supply. In the SPTM, independence is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E equipment.	3. a. Tests will be performed on the SPTM 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 SPTM will be performed.	3. a. A test signal exists only in the Class 1E division under test in the SPTM. b. In the SPTM, physical separation or electrical separation exists between Class 1E divisions. Physical separation or electrical isolation exists between these Class 1E divisions and non-Class 1E equipment.
4. MCR displays provided for the SPTM and suppression pool water level are as defined in Subsection 2.15.7.	4. Inspections will be conducted on the MCR displays for the SPTM and suppression pool water level.	4. Displays exist or can be retrieved in the MCR as defined in Subsection 2.15.7.
5. Remote Shutdown System (RSS) displays provided for the suppression pool level monitoring are as defined in Subsection 2.15.7.	5. Inspections will be conducted on the RSS displays for the suppression pool level monitoring.	5. Displays exist on the RSS as defined in Subsection 2.15.7.

Table 2.15.7-3

ITAAC For The Post-Accident Sampling

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The PASS isolation valves are connected to Class 1E power sources.	1. Inspections will verify the isolation valve electric power source.	1. Class 1E power sources are provided for PASS isolation valves.
2. Sampling is available following a LOCA.	2. Tests simulating a LOCA signal will be performed while the isolation valves are operated.	2. Isolation valves open for sampling in the presence of a LOCA signal.

2.16 STRUCTURES AND SERVICING SYSTEMS/EQUIPMENT

2.16.1 Cranes, Hoists and Elevators

Design Description

Cranes and Hoists are used for maintenance and refueling tasks.

The RB overhead crane is used during refueling and maintenance activities as well as when the plant is on-line. Minimum crane coverage includes the RB refueling floor lay down areas and the RB equipment storage. Minimum crane coverage includes the refueling floor and the equipment hatches (floor plugs). The RB crane is interlocked to prevent movement of heavy loads over the fuel storage pool.

The FB crane is used during refueling and maintenance activities as well as when the plant is on-line. Minimum crane coverage includes the FB floor lay down areas, cask wash down area, and the FB equipment hatch. During normal plant operation, the crane is used to handle new fuel shipping containers and the spent fuel-shipping cask. The FB crane is interlocked to prevent movement of heavy loads over the spent fuel storage pool.

The hoisting and braking systems of the RB and FB crane are redundant.

The cranes have lifting capacities equal to the maximum critical load [MCL].

The Cranes and Hoists are classified as nonsafety-related.

The upper drywell hoists and lower drywell hoists are classified as Seismic Category NS.

Elevators are installed in the RB, CB, TB and other buildings as necessary.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.1-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Cranes, Hoists and Elevators.

Table 2.16.1-1

ITAAC For The Cranes, Hoists and Elevators

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The cranes have lifting capacities equal to the maximum critical load [MCL].	1. Perform a load test at 125% of the MCL.	1. The Design Commitment is met.

2.16.2 Heating, Ventilating and Air-Conditioning Systems

Design Descriptions

Reactor Building HVAC

The Reactor Building Heating, Ventilation and Air Conditioning System (RBHVS) serves the Reactor Building.

With the following exception, the Reactor Building HVAC is nonsafety-related. The isolation dampers and ducting penetrating the Reactor Building boundary, and associated controls that provide the isolation signal are safety-related. The RBHV System performs no safety-related function except for automatic isolation of the Reactor Building boundary during accidents.

The Reactor Building HVAC consists of three Subsystems. The Reactor Building Contaminated Area HVAC Subsystem (CONAVS) serves the potentially contaminated areas of the Reactor Building. The Refueling and Pool Area HVAC Subsystem (REPAVS) serves the refueling area of the Reactor Building and the Reactor Building Clean Area HVAC Subsystem (CLAVS) serves the clean (non-radiological controlled) areas of the Reactor Building.

The CLAVS is a recirculating ventilation system with redundant AHUs, return/exhaust fans, electric heaters and smoke exhaust fans. A simplified schematic of CLAVS is provided in Figure 2.16.2-1.

The CONAVS is a once-through ventilation system. It consists of redundant AHUs, exhaust fans and building isolation dampers. It also includes a primary containment purge exhaust fan, recirculation AHUs and electric heaters. A simplified schematic of CONAVS is provided in Figure 2.16.2-2.

The REPAVS is a once-through ventilation system. It consists of redundant AHUs, exhaust fans and building isolation dampers. A simplified schematic of REPAVS is provided in Figure 2.16.2-3.

Control Building HVAC

The three Control Building HVAC subsystems are the Control Room Habitability Area HVAC (CRHAHVS), which serves the Main Control Room (MCR), the Emergency Breathing Air System (EBAS), which supplies pressurized air to the Control Room Habitability Area (CRHA) during radiological events, and the Control Building General Area HVAC (CBGAHVS), which serves the area outside the CRHA. A simplified schematic of CRHAHVS is provided in Figure 2.16.2-4.

The EBAS is safety-related, and passively maintains habitable conditions in the Control Room Habitability Area (CRHA), to ensure the safety of the control room operators. The EBAS consists of multiple independent redundant compressed breathing air tank trains. The EBAS supplies stored, compressed breathing air to the CRHA for breathing and pressurization to minimize infiltration. The EBAS is automatically initiated, and provides sufficient breathing quality air to maintain positive pressure in the CRHA for 5 occupants for 72 hours. A simplified system diagram of EBAS is provided in Figure 2.16.2-5.

The other subsystems are nonsafety-related, except for the isolation dampers and portions of the system that penetrate the CRHA. The CRHAHVS serves the Main Control Room during normal

plant operation, plant start up and plant shutdown. It consists of redundant AHUs, return/exhaust fans, CRHA air conditioning units, isolation dampers, electric heaters, and filter units. The CBGAHVS serves areas outside the CRHA. It also consists of redundant air handling units, return/exhaust fans, electric heaters and dampers. A simplified schematic of CBGAHVS is provided in Figures 2.16.2-6 and 2.16.2-7.

Turbine Building HVAC

The Turbine Building Ventilation System is nonsafety-related. The system includes an intake plenum, dampers, heating and cooling coils and supply fans. Redundant exhaust fans are also provided. Local unit coolers and fans are provided in areas with high local heat loads.

Fuel Building HVAC

The Fuel Building HVAC systems do not perform any safety-related functions, except for automatic isolation of the refueling area during accidents, and thus, both systems are classified as nonsafety-related, except for automatic isolation of the refueling and fuel pool area during accidents. Fuel Building HVAC subsystems include the Fuel Building General Area Heating and Ventilation (FBGAHV) and Fuel Building Fuel Pool Heating and Ventilation System (FBFPHV). Both Subsystems consist of redundant air handling units, exhaust fans, electric heaters, and dampers. A simplified schematic of FBGAHV is provided in Figure 2.16.2-8. A simplified schematic of FBFPHV is provided in Figure 2.16.2-9.

Other Building HVAC

Ventilation for other buildings includes the Radwaste Building, Electrical Building, Service Building, Water Treatment Building, Administration Building, Guard House, etc. All these systems are nonsafety-related, of conventional design and typically include redundant supply and exhaust fans, and air conditioning units. The Radwaste Building and Hot Machine Shop ventilation systems also include additional filtration and airborne radioactivity monitoring equipment.

Instrumentation and Control

Critical and essential information are available in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Tables 2.16.2-1, -2 and -3 provide definitions of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Reactor Building HVAC, Control Building HVAC and Fuel Building HVAC, respectively.

No entry for other HVAC systems.

Table 2.16.2-1

ITAAC For The Reactor Building HVAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Reactor Building HVAC is as described in Subsection 2.16.2.	1. Inspections of the Reactor Building HVAC configuration will be conducted.	1. The as-built system conforms with the description in Subsection 2.16.2.
2. The Reactor Building HVAC isolation dampers automatically close upon receipt of a high radiation signal from PRMS or loss of AC power.	2. Using simulated signals, tests will be preformed on the (Reactor Building HVAC isolation dampers) isolation logic.	2. Upon receipt of a simulated signal, the Reactor Building HVAC isolation dampers automatically close.
3. The Seismic Category I components (building isolation dampers and associated controls) can withstand seismic basis loads without loss of safety function.	3. i) nspection will be performed to verify Seismic Category I equipment is located in the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of Seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the testing or analyzed conditions.	3. i) The Seismic Category I equipment is located in the Nuclear Island. ii) A report exists and concludes that the equipment can withstand seismic design basis without loss of safety function. iii) A report exists and concluded that the as-installed equipment including anchorage is seismically bounded by testing or analyzed conditions.
4. The RBHVS provides cooling to the Class 1E battery rooms and Class 1E electrical equipment rooms.	4. Testing will be performed on the components using the controls in the main control room.	4. Controls in the main control room cause the components to perform the required function.

Table 2.16.2-2

ITAAC For The Control Building HVAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Control Building HVAC is as described in Subsection 2.16.2.	1. Inspections of the Control Building HVAC configuration will be conducted.	1. The as-built system conforms with the description in Subsection 2.16.2.
2. The Control Building HVAC Sealed Emergency Operating Area (CRHA) isolation dampers automatically close upon receipt of a high radiation signal from PRMS or loss of AC power.	2. Using simulated isolation signals, tests will be preformed on the (Control Building HVAC CRHA isolation dampers) isolation logic.	2. Upon receipt of a simulated isolation signal, the Control Building HVAC CRHA isolation dampers automatically close.
3. The Seismic Category I components (EBAS, CRHA isolation dampers and associated controls) can withstand seismic basis loads without loss of safety function.	3. i) Inspection will be performed to verify Seismic Category I equipment is located in the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of Seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the testing or analyzed conditions.	3. i) The Seismic Category I equipment is located in the Nuclear Island. ii) A report exists and concludes that the equipment can withstand seismic design basis without loss of safety function. iii) A report exists and concluded that the as-installed equipment including anchorage is seismically bounded by testing or analyzed conditions.
4. The CBHVS provides cooling to the CRHA and DCIS equipment rooms.	4. Testing will be performed on the components in the using the controls in the main control room.	4. Controls in the main control room cause the components to perform the required function.

Table 2.16.2-3

ITAAC For The Fuel Building HVAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Fuel Building HVAC is as described in Subsection 2.16.2.	1. Inspections of the Fuel Building HVAC configuration will be conducted.	1. The as-built system conforms with the description in Subsection 2.16.2.
2. The Fuel Building HVAC isolation dampers automatically close upon receipt of an isolation signal.	2. Using simulated isolation signals, tests will be preformed on the (Fuel Building HVAC isolation dampers) isolation logic.	2. Upon receipt of a simulated isolation signal, the Fuel Building HVAC isolation dampers automatically close.
3. The Seismic Category I components (building isolation dampers and associated controls) can withstand seismic basis loads without loss of safety function.	3. i) Inspection will be performed to verify Seismic Category I equipment is located in the Nuclear Island. ii) Type tests, analyses, or a combination of type testes and analyses of Seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the testing or analyzed conditions.	3. i) The Seismic Category I equipment is located in the Nuclear Island. ii) A report exists and concludes that the equipment can withstand seismic design basis without loss of safety function. iii) A report exists and concluded that the as-installed equipment including anchorage is seismically bounded by testing or analyzed conditions.

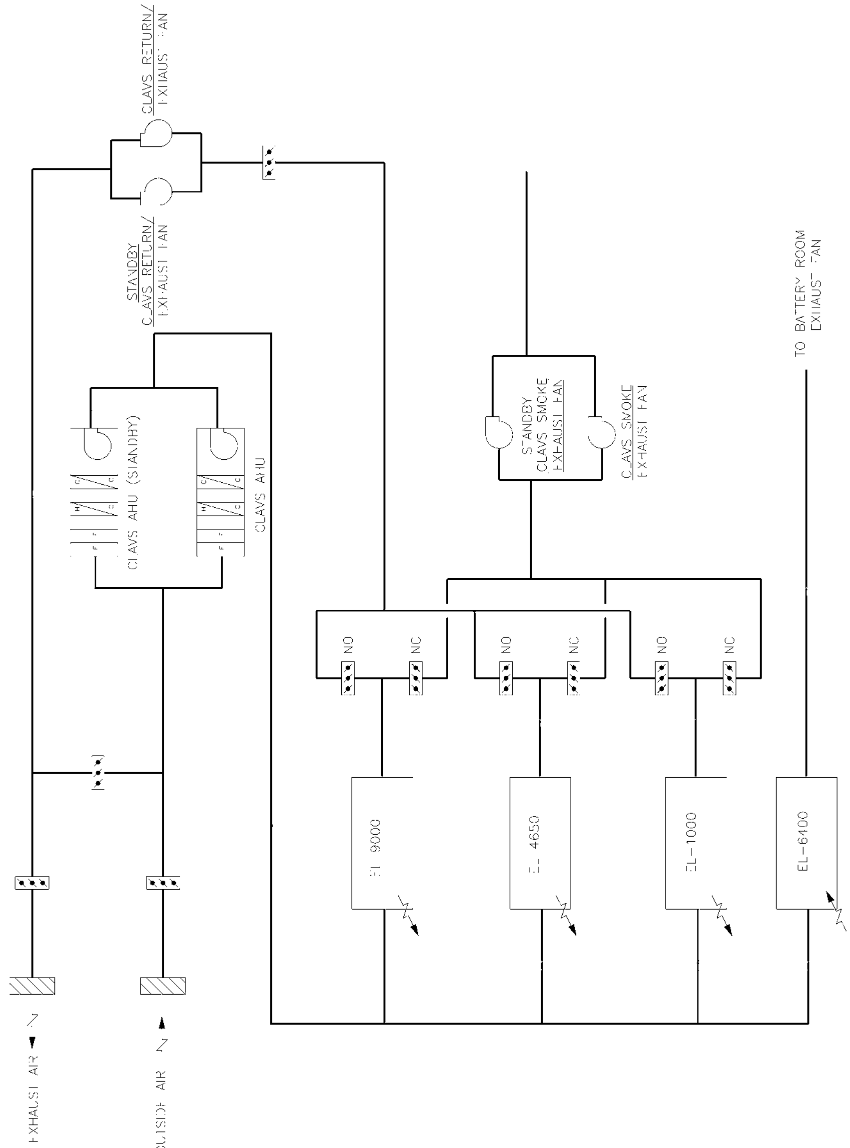


Figure 2.16.2-1. CLAVS Simplified System Diagram

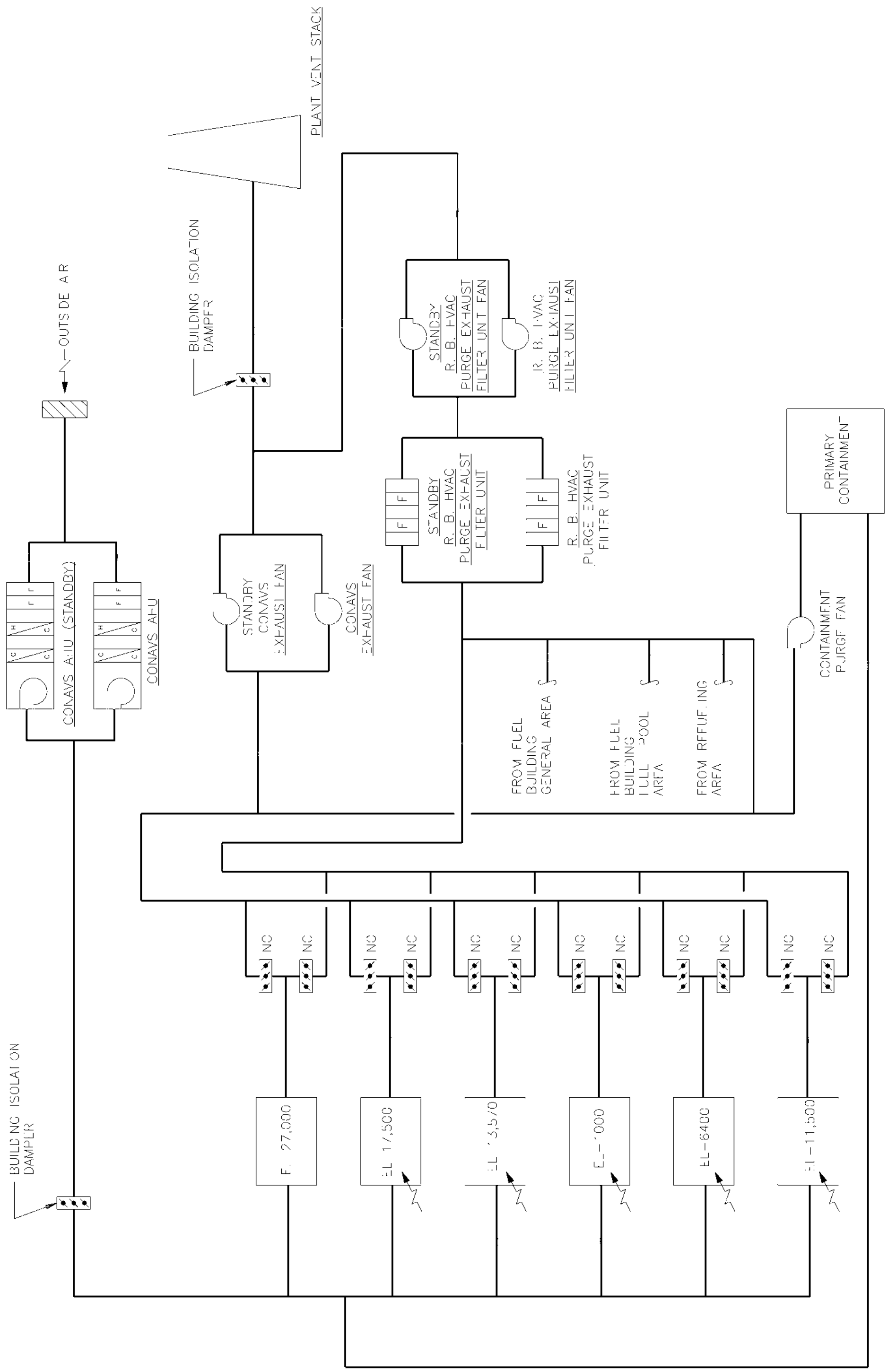


Figure 2.16.2-2. CONAVS Simplified System Diagram

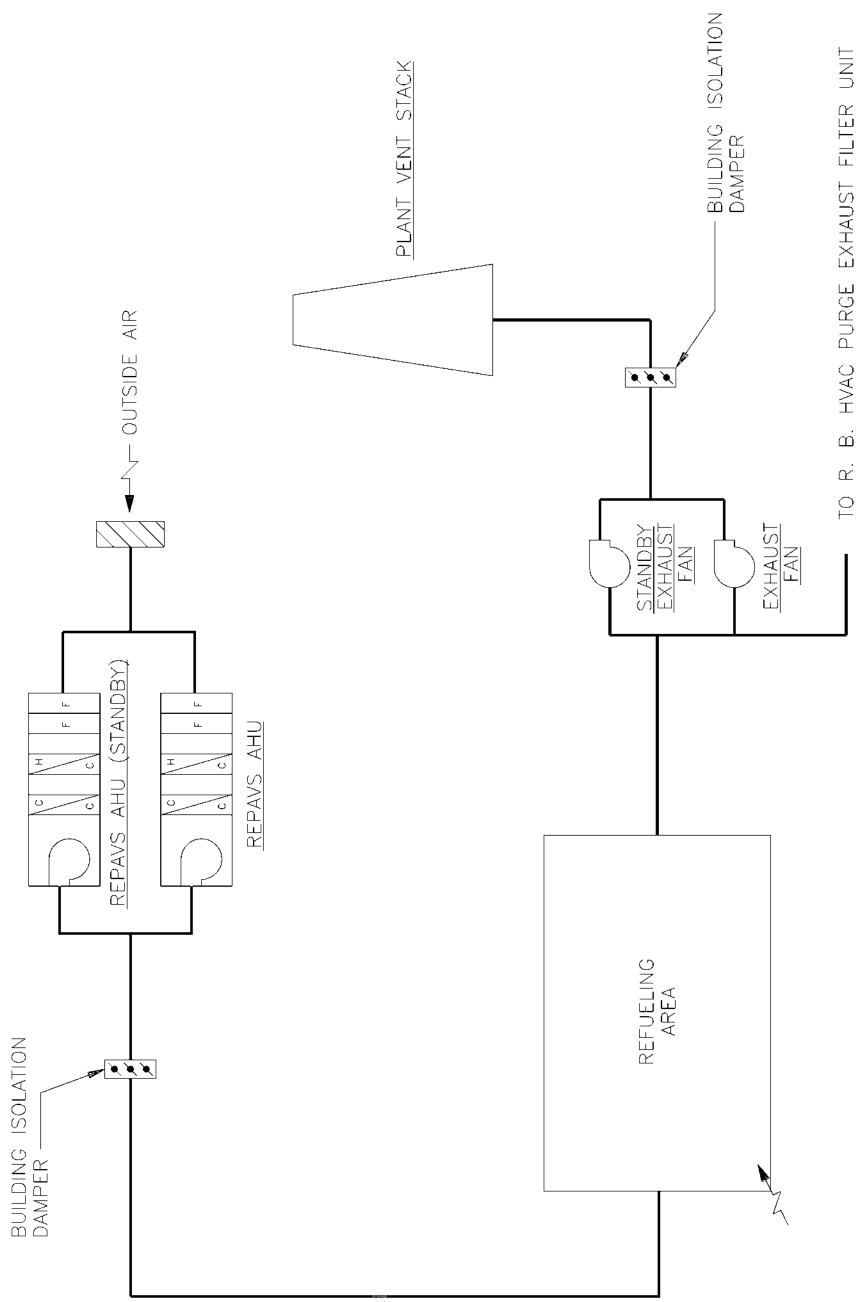


Figure 2.16.2-3. REPAVS Simplified System Diagram

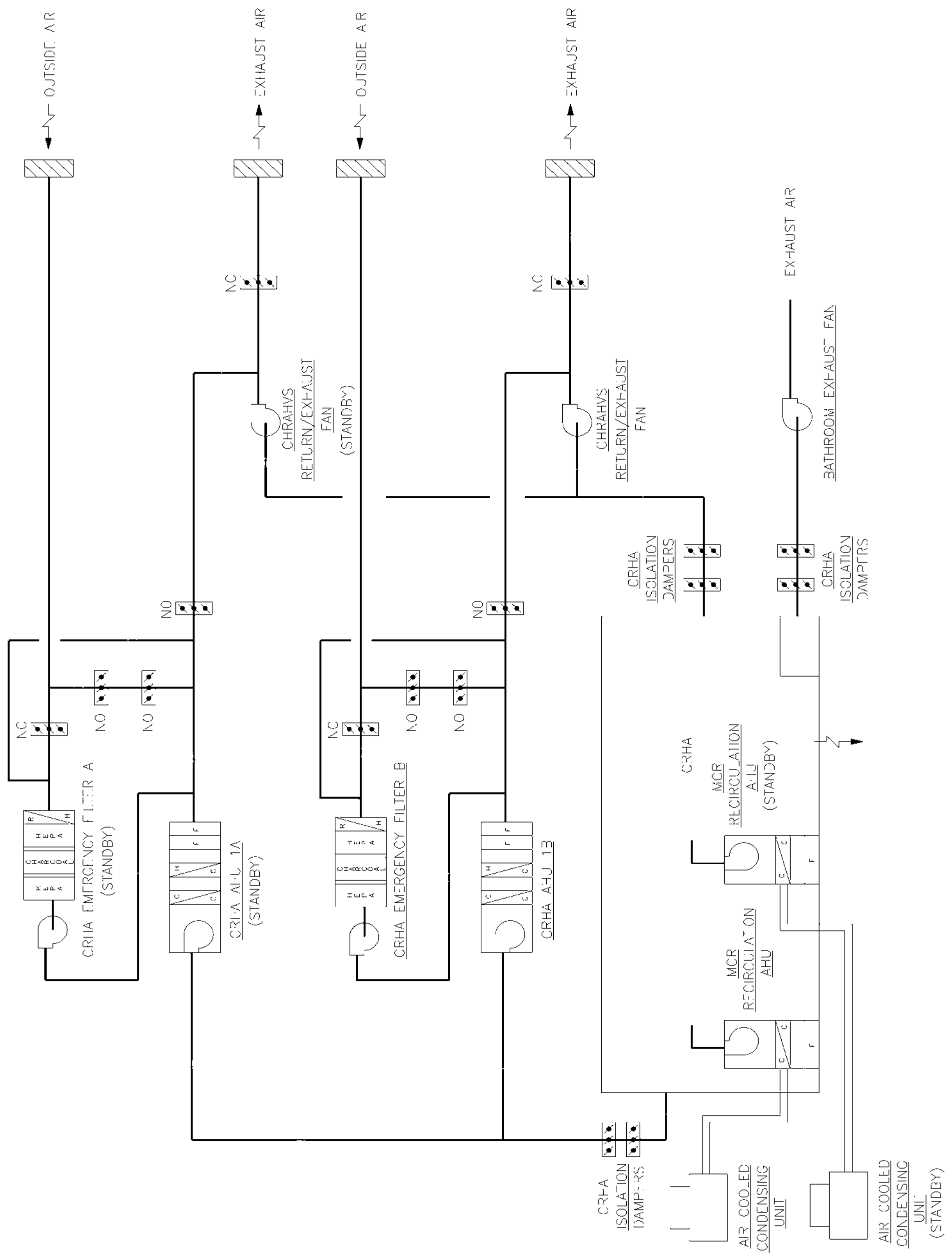


Figure 2.16.2-4. CRHAHVS Simplified System Diagram

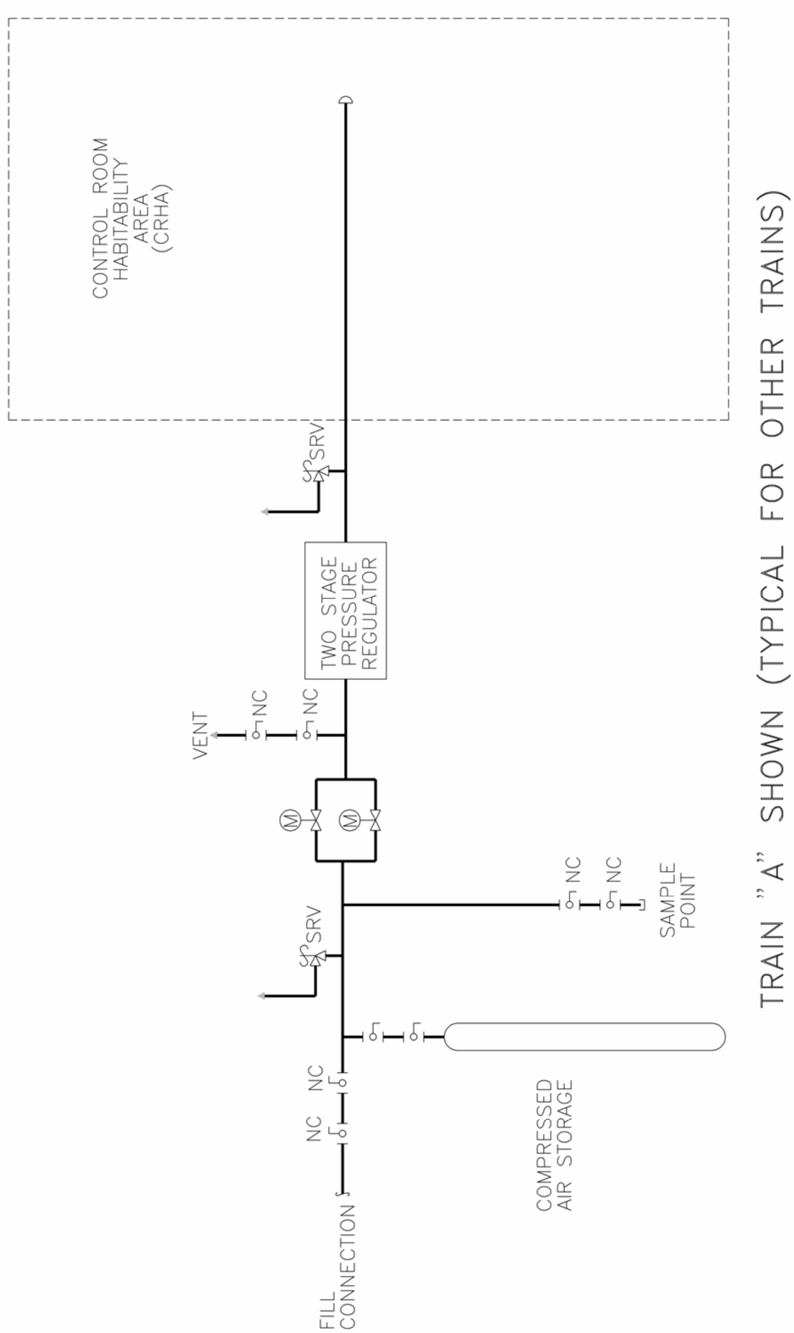


Figure 2.16.2-5. EBAS System Diagram

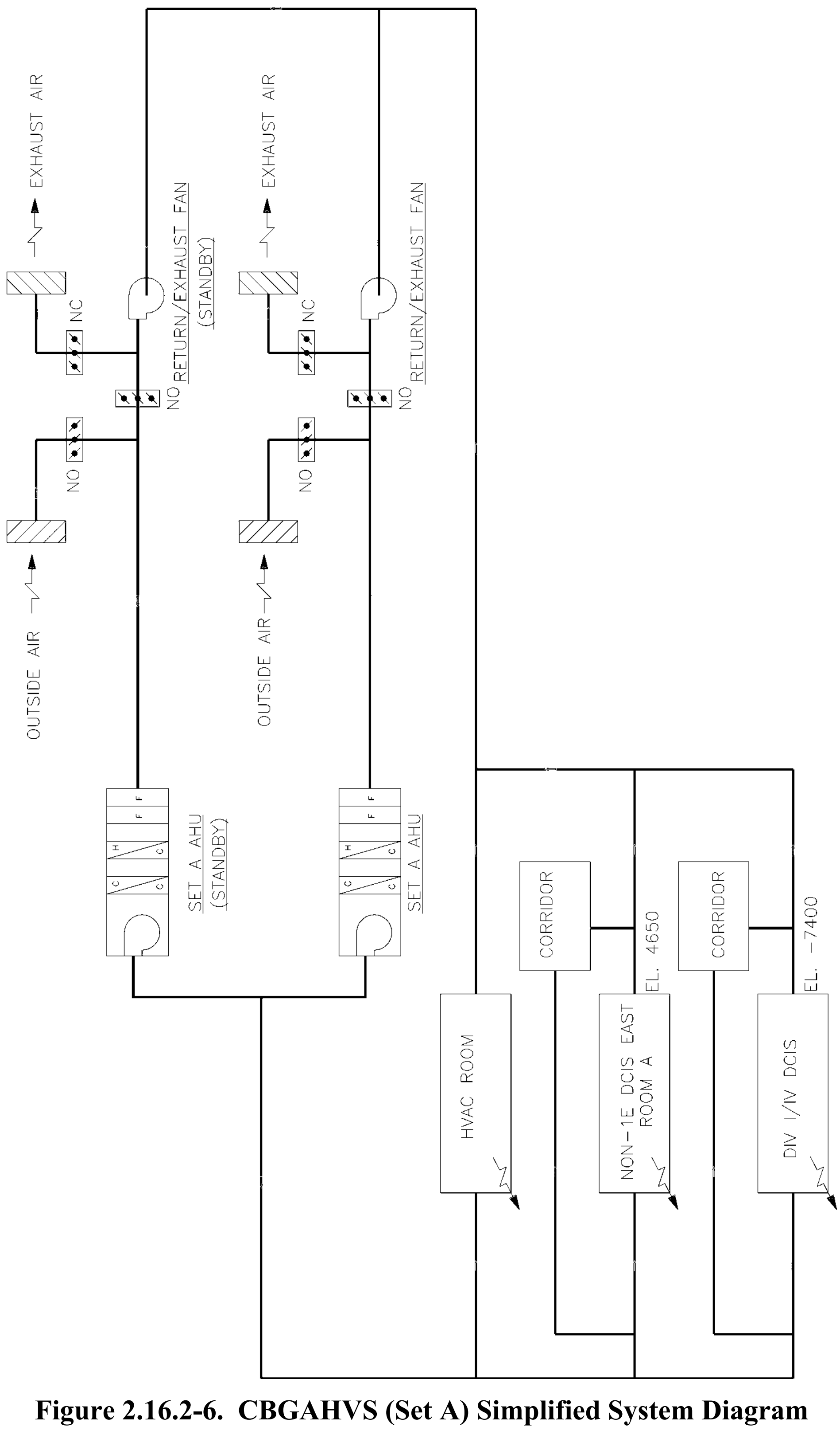


Figure 2.16.2-6. CBGAHVS (Set A) Simplified System Diagram

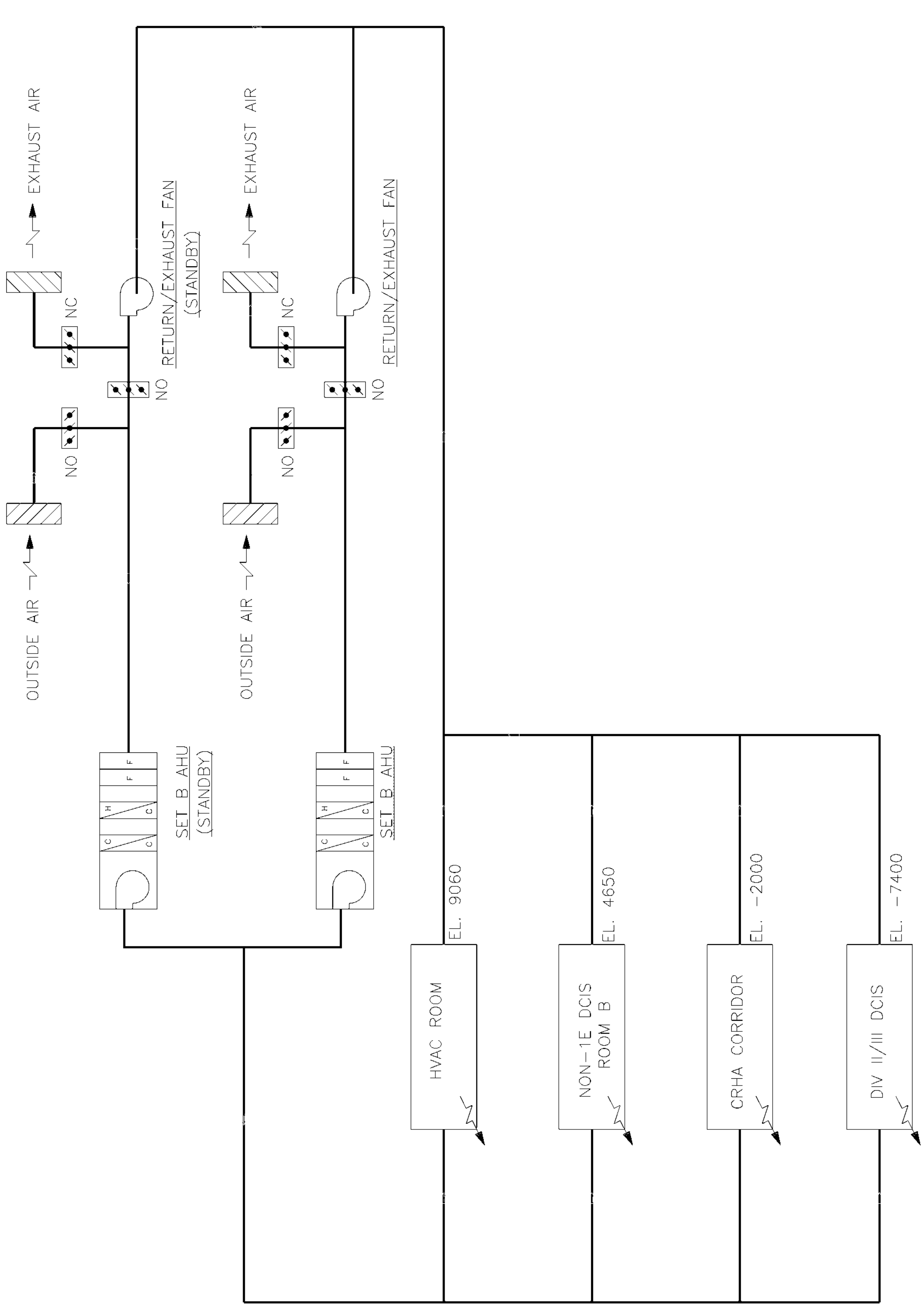


Figure 2.16.2-7. CBGAHVS (Set B) Simplified System Diagram

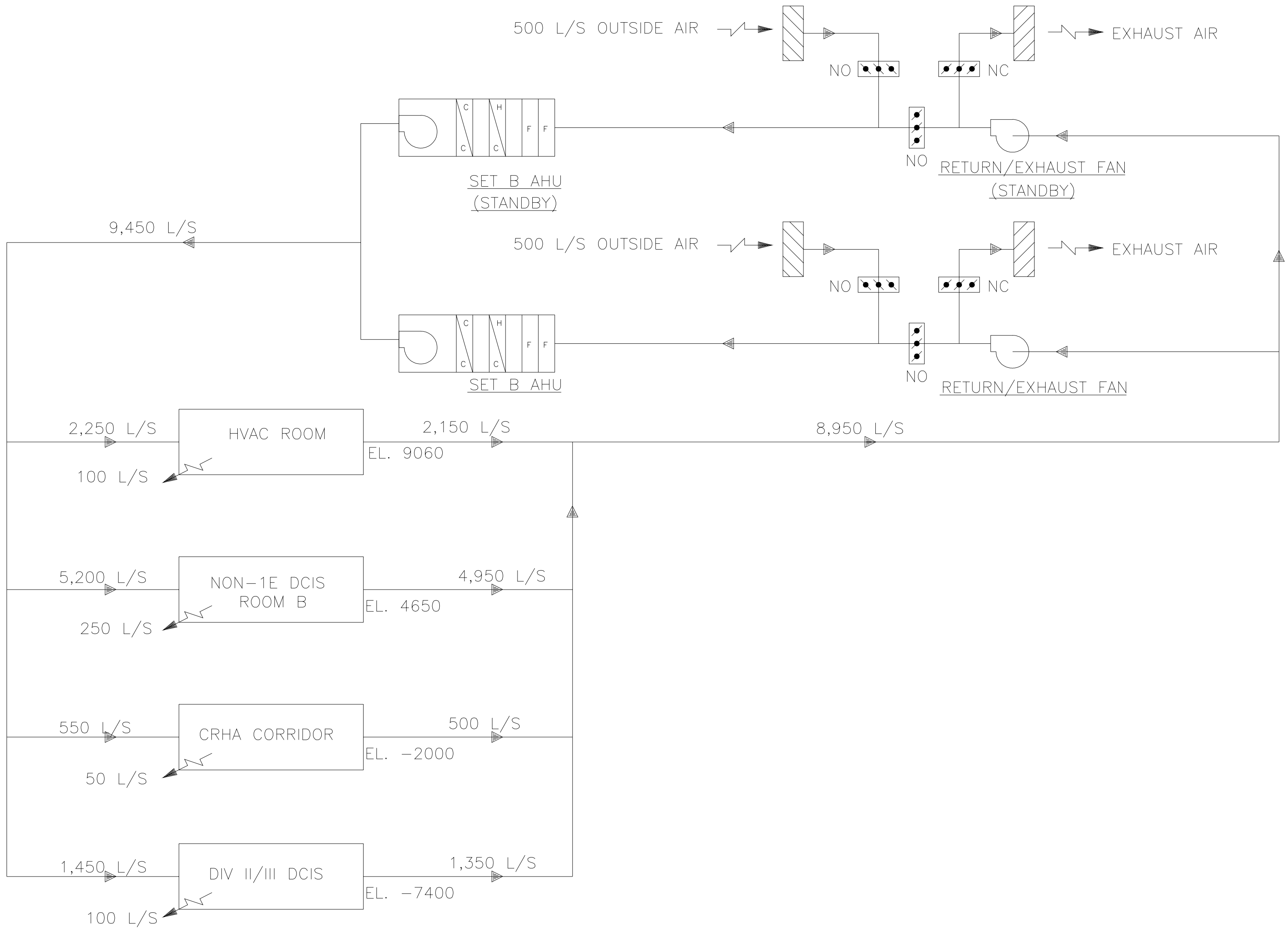


Figure 2.16.2-8. FBGAHV Simplified System Diagram

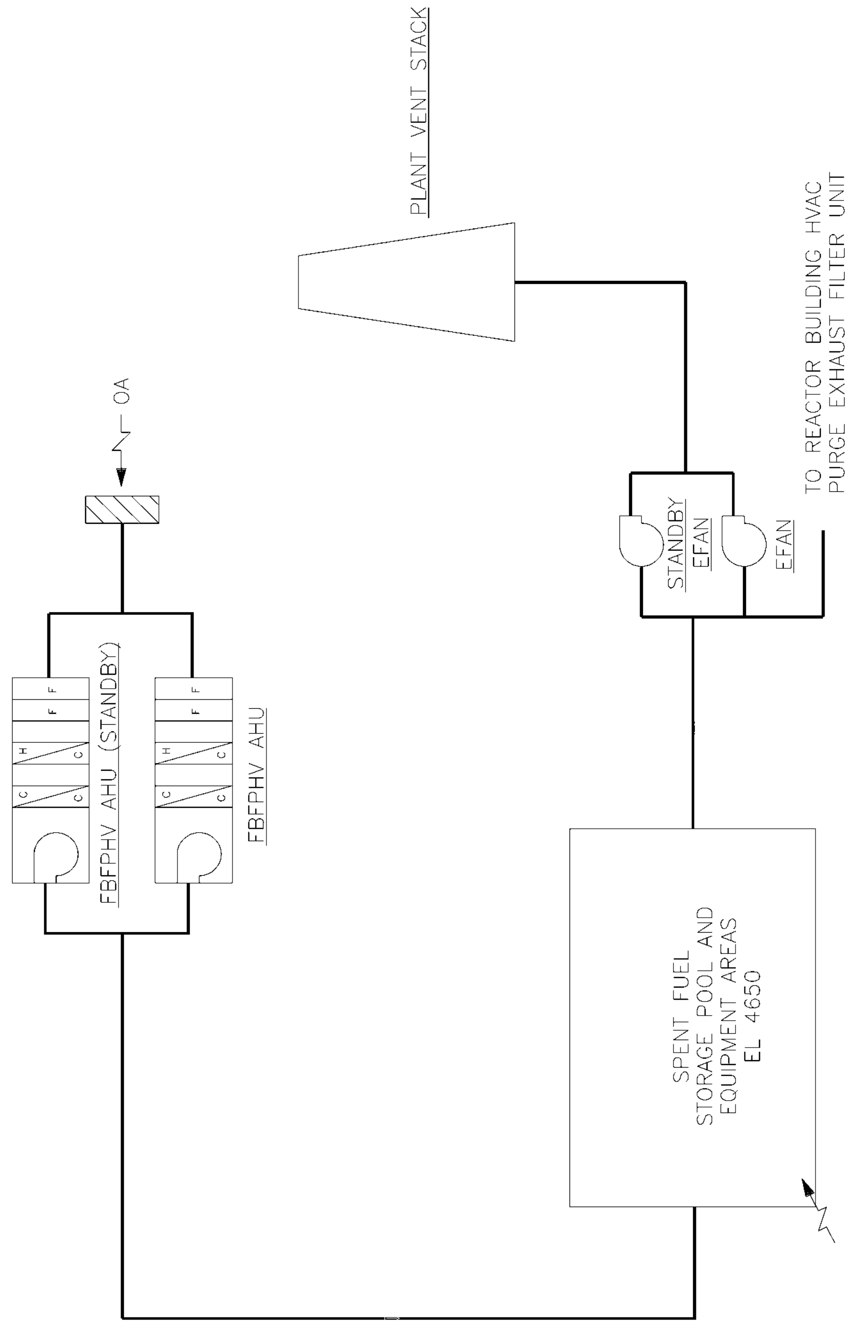


Figure 2.16.2-9. FBFPHV Simplified System Diagram

2.16.3 Fire Protection System

Design Description

The Fire Protection System (FPS) includes the fire protection water supply system, yard piping, water sprinkler, standpipe and hose systems, a foam system, smoke detection and alarm system, and fire barriers.

A simplified diagram of the FPS is provided in Figure 2.16.3-1.

Each of the three 50% capacity firewater pumps provides 100% of the firewater demand to the worst-case fire within the nuclear island (Reactor Building, Fuel Building, and Control Building) or 50% of the firewater demand to the worst-case fire within the balance of plant The pumps are capable of delivering the flow and pressure required to the location that is farthest from the firewater supply source. Two of the three pumps are located near the nuclear island power block in a fire pump enclosure. The third pump is located remote from the other two pumps to prevent a common-location failure.

For the two Nuclear Island fire pumps, the lead pump is motor-driven and the backup pump is diesel-driven. The backup pump provides firewater in the event of failure of the motor-driven pump or loss of preferred power (LOPP). The main diesel-driven fire pump, including its suction and discharge piping, meets the requirements of ASME B31.1 and remains functional after an SSE and is located in a separate fire-rated compartment from the motor-driven fire pump.

The second diesel-driven fire pump provides a back up to the other two pumps. This back-up diesel-driven fire pump may be new or existing and is connected to the main yard piping loop.

The fuel oil tanks for the diesel-driven fire pumps have sufficient capacity to allow diesel engine operation for approximately 8 hours before refilling based upon the fuel consumption and margin criteria provided in NFPA 24.

The fire water supply piping consists of a non-seismic, buried yard main and a suspended ASME B31.1 piping loop. The ASME B31.1 piping loop supplies firewater to the nuclear island buildings and remains functional following an SSE. The main firewater pumps discharge to the ASME B31.1 piping loop. A connection from the ASME B31.1 piping loop supplies firewater to the yard main. The second diesel-driven fire pump supplies firewater directly to the yard main, but is also capable of supplying water to the ASME B31.1 piping loop. Motor-operated isolation valves are provided between the buried, non-seismic, yard piping loop and the suspended, Seismic Category I, ASME B31.1 piping loop.

The FPS provides an emergency backup source of makeup water 72 hours after a LOCA for IC/PCCS pools and the spent fuel pool and reactor water inventory control.

The FPS also provides emergency backup source of makeup water for auxiliary refueling pools and reactor water inventory control through a piping connection to the Fuel and Auxiliary Pools Cleaning System (FAPCS).

The FPS is nonsafety-related. However, one source of fire water supply, one of the fire pumps and the fire water main leading to and including the standpipes and subsystems for areas containing safe shutdown equipment are analyzed to withstand the effect of a Safe Shutdown Earthquake (SSE). They shall remain functional during and after an SSE.

Instrumentation and Controls

Controls and instrumentation are provided for a fully functioning system. There are three main types of FPS instrumentation: instrumentation supporting fire detection, instrumentation supporting automatic suppression systems, and instrumentation supporting fire water delivery.

Critical and essential information and controls are provided in the main control room. In addition to automatic operation any of the fire pumps can be manually started either from MCR or local panels.

Interface Requirements

Interface requirements for the service water basins are described in Section 4.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.3-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Fire Protection System.

Table 2.16.3-1

ITAAC For The Fire Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Fire Protection System is as described in Subsection 2.16.3.	1. Inspections of the as-built system will be conducted.	1. The as-built Fire Protection System conforms to the basic configuration contained in the Design Description of Subsection 2.16.3.
2. The motor driven pump described in the Design Description for the Fire Protection System is powered from the non-Class 1E bus.	2. A test of the power availability to the motor driven pump described in the Design Description in Subsection 2.16.3 will be conducted with power supplied from the permanently installed electric power busses.	2. The motor driven pump described in the Design Description for the Fire Protection System receives power from non-Class 1E busses only.
3. Two water supply tanks with a minimum volume of about 2000 m^3 (550,000 gal) each are provided.	3. Inspection of the as-built water supply sources and volumetric calculations using as-built dimensions will be performed.	3. As-built water supply sources meet the volumetric requirements specified in the Certified Design Commitment.
4. Fire water supply system pumps independently provide a minimum flow of [454.2 m^3/hr (2,000 gpm)] at a pressure of [689 kPa gauge (100 psig)] at the most hydraulically remote 65 mm (~2.5 in) hose connections station or [448 kPa gauge (65 psig)] at the most hydraulically remote 40 mm (~1.6 in) hose station in the Reactor Building and Control Building.	4. A test of the flow rate and pressure from each pump will be conducted.	4. The fire water supply system pumps independently provide the flow and pressure specified in the Certified Design Commitment.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. No location within a fire area is more than [30.5 m (100 ft)] from a hose station.	5. Inspection of the as-built hose rack locations will be performed.	5. Standpipe and hose rack stations are located such that no location within a fire area is more than [30.5 m (100 ft)] from a hose station.
6. No safe shutdown equipment is more than [30.5 m (100 ft)] from two hose stations on separate standpipes.	6. Inspection of the as-built hose rack locations will be performed.	6. Standpipe and hose rack stations are located as such that no safe shutdown equipment is more than [30.5 m (100 ft)] from two hose stations on separate standpipes.
7. Automatic fire suppression is provided for all electrical areas exceeding that specified in codes NFPA 13 & 15.	7. Inspection of all electrical areas to identify areas requiring automatic fire suppression system per NFPA 13 & 15.	7. Confirm that an automatic fire suppression is provided for all electrical areas required per NFPA 13 & 15.
8. Automatic fire suppression is provided for all non-electrical areas exceeding that specified in codes NFPA 13 & 15.	8. Inspection of all non-electrical areas to identify areas requiring automatic fire suppression system per NFPA 13 & 15.	8. Confirm that an automatic fire suppression is provided for all non-electrical areas required per NFPA 13 & 15.
9. Automatic foam-water extinguishing systems are provided for the diesel generator and day tank rooms, per codes NFPA 11& 16.	9. Inspection of as-built systems and testing of automatic logic under simulated fire conditions will be conducted.	9. The automatic foam-water suppression systems exist and initiation logic is actuated under simulated fire conditions.
10. The fuel oil tanks for the diesel-driven fire pumps have sufficient capacity to allow diesel engine operation for approximately 8 hours as described in this Subsection 2.16.3.	10. Inspection of the fuel oil day tanks will be conducted.	10. The fuel oil tanks for the diesel-driven fire pumps have sufficient capacity to allow diesel engine operation for approximately 8 hours as defined in this Subsection 2.16.3.
11. Control room indications and controls for the Fire Protection System are as defined in Subsection 2.16.3.	11. Inspections will be performed on the control room indications and controls for the Fire Protection System.	11. Indications and controls exist or can be retrieved in the MCR as defined in Subsection 2.16.3.

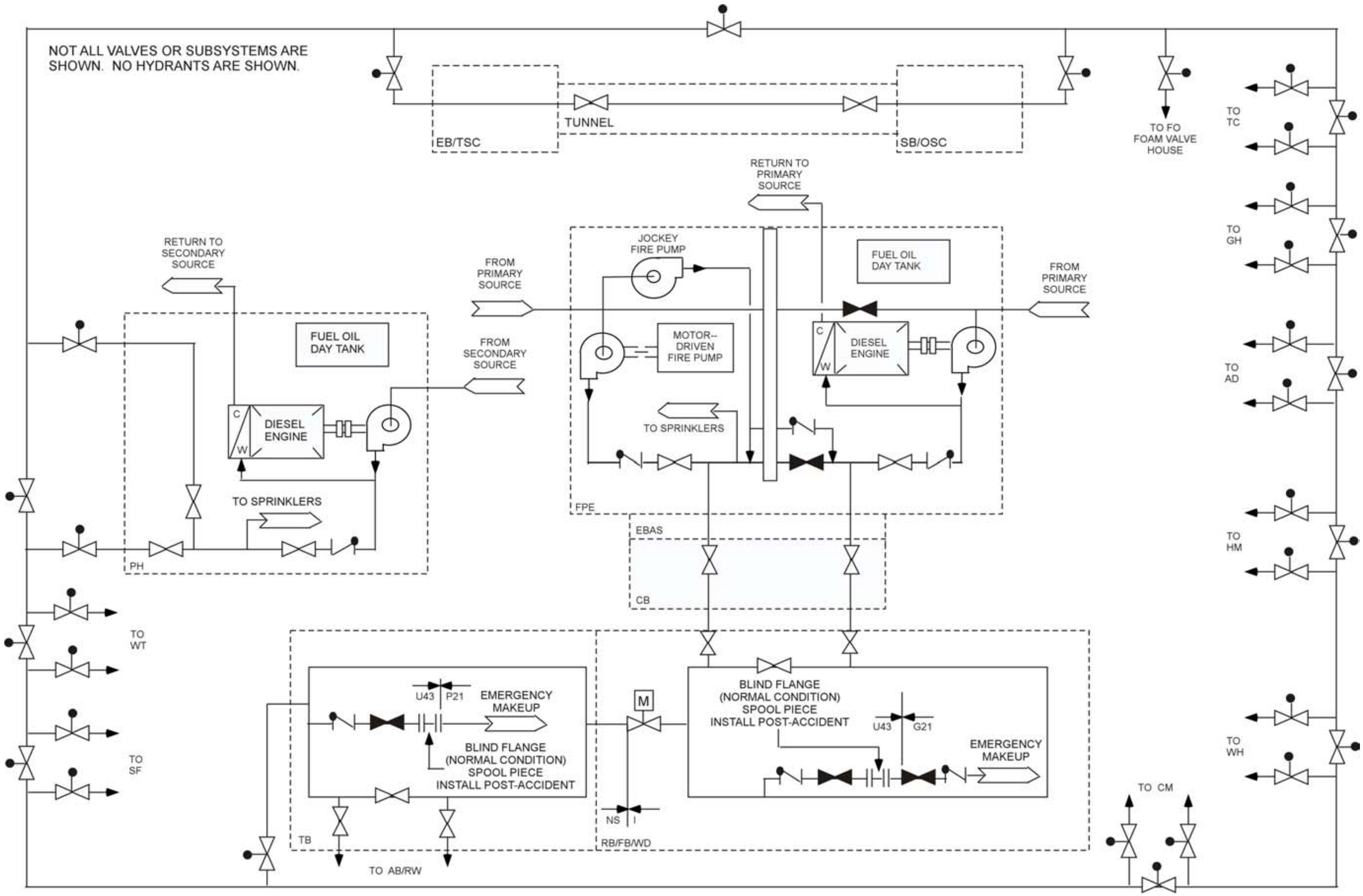


Figure 2.16.3-1. Fire Protection System

2.16.4 Equipment and Floor Drain System

Design Description

The Equipment and Floor Drain System (EFDS) consists of liquid waste collection piping, equipment drains, floor drains, vents, traps, cleanouts, collection sumps, sump pumps, tanks, valves, controls and instrumentation. The EFDS serves plant buildings (i.e., Reactor Building, Control Building, Fuel Building, Turbine Building, Electrical Building, Service Building, Radwaste Building, Service Water & Water Treatment Building, and the Fire Pump Enclosure) with floor and equipment drains and consists of the following drain subsystems: clean, low conductivity waste (LCW), high conductivity waste (HCW), detergent, and chemical waste. All potentially radioactive drains are routed to the Liquid or Solid Waste Management System for processing.

The EFDS is nonsafety-related except for containment penetrations, isolation valves, and level switches for initiating containment isolation. Other than containment isolation, the EFDS does not perform any safety-related function, nor is it required to achieve or maintain safe shutdown of the plant.

Instrumentation and Control

Instrumentation and controls for the EFDS are located at local panels, and appropriate signals are duplicated and sent to the MCR. Essential and critical information is available in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.4-1 provides definition of the inspections, test, and/or analyses, together with associated acceptance criteria, which will be undertaken for the EFDS.

Table 2.16.4-1

ITAAC For The Equipment and Floor Drain System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the EFDS is as described in Subsection 2.16.4.	1. Inspections of the EFDS configuration will be conducted.	1. The as-built system conforms with the Design Description in Subsection 2.16.4.
2. The EFDS containment isolation valves automatically close upon receipt of an EFDS isolation signal from LD&IS.	2. Using simulated EFDS isolation signals, tests will be preformed on the (EFDS isolation valves) isolation logic.	2. Upon receipt of a simulated isolation signal, the EFDS containment isolation valves automatically close.

2.16.5 Reactor Building

Design Description

The Reactor Building (RB) (Figures 2.16.5-1 through 2.16.5-11) houses the reactor system, reactor support and safety systems, concrete containment, essential power supplies and equipment, steam tunnel, and refueling area. On the upper floor of the RB are the new fuel pool and small, spent fuel storage area, dryer/separator storage pool, refueling and fuel handling systems, the upper connection to the incline fuel transfer system and the overhead crane. The Isolation Condenser/Passive Containment Cooling System pools are below the refueling floor.

The Reactor Building structure is integrated with that of a right circular cylindrical reinforced concrete containment vessel (RCCV); the RCCV is located on a common basemat with the RB. The RB is a rigid box type shear wall building. The external walls form a box surrounding a large cylindrical containment. The RB shares a common wall and sits on a large common basemat with the Fuel Building. The RB is a Seismic Category I structure. The building is partially embedded.

The RB offers some holdup and decay of fission products that may leak from the containment after an accident. Offsite dose requirements are met assuming a 100% volume change out per day in the RB volume outside of the RCCV. This holdup capability decreases releases to the atmosphere. The building and systems are also arranged to separate clean and potentially contaminated areas, with separate stairway and elevator service for each area.

To protect the RB against an external flood, penetrations in the external walls below flood level are sealed.

The RB is protected against pressurization effects associated with postulated rupture of pipes containing high-energy fluid that occur in subcompartments of the RB.

The RB is designed and constructed to accommodate the dynamic and static loading conditions associated with the various loads and load combinations, which form the structural design basis. The loads are (as applicable) those associated with:

- Natural phenomena—wind, floods, tornados (including tornado missiles), earthquakes, rain and snow.
- Internal events—floods, pipe breaks and missiles.
- Normal plant operation—live loads, dead loads, temperature effects and building vibration loads.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.5-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Reactor Building.

Table 2.16.5-1

ITAAC For The Reactor Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The RB arrangement is described in Figures 2.16.5-1 through 2.16.5-11.	1. Inspections of the as-built facility will be conducted.	1 The as-built facility conforms to the basic configuration shown in Figures 2.16.5-1 through 2.16.5-11.
2. The RB provides three-hour fire barriers for separation of the four independent safe shutdown divisions.	2. Inspections of the as-built facility will be conducted.	2. Each division is separated by barriers having three-hour fire ratings.
3. Protection is provided against external and internal floods. For external flooding, protection features are: a. Watertight access openings installed in external walls below flood level. b. Watertight penetrations installed in external walls below flood level. c. The RB floor slabs at external entrances to the RB are provided with sills. For internal flooding, protection features are: a. Flood water in one division is prevented from propagating to other divisions by walls and floors. b. Openings between divisions at the basemat have watertight doors and sills. Other openings between	3. Inspection of the as-built flood control features will be conducted.	3. Flood protection features are provided as described in the Design Commitment.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
divisions have sills. c. Each divisional area has drains to contain and route floodwater to its respective divisional area below grade. d. Equipment necessary for safe shutdown is located above the maximum flood level for that location or is qualified for flood conditions.		

Figure 2.16.5-1. Nuclear Island Plan at Elevation –11500

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

Figure 2.16.5-2. Nuclear Island Plan at Elevation –6400

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

Figure 2.16.5-3. Nuclear Island Plan at Elevation –1000

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

Figure 2.16.5-4. Nuclear Island Plan at Elevation 4650

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

Figure 2.16.5-5. Nuclear Island Plan at Elevation 9060

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

Figure 2.16.5-6. Nuclear Island Plan at Elevation 13570

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

Figure 2.16.5-7. Nuclear Island Plan at Elevation 17500

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

Figure 2.16.5-8. Nuclear Island Plan at Elevation 27000

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

Figure 2.16.5-9. Nuclear Island Plan at Elevation 34000

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

Figure 2.16.5-10. Nuclear Island Elevation Section A-A

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

Figure 2.16.5-11. Nuclear Island Elevation Section B-B

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

2.16.6 Control Building

Design Description

The Control Building (CB) (Figures 2.16.5-2 through 2.16.5-5 and Figure 2.16.5-11) houses the essential electrical, control and instrumentation equipment, the main control room for the plant, and the CB HVAC equipment.

The CB is a reinforced concrete box type shear wall structure consisting of walls and slabs and is supported on a foundation mat. The CB structure up to grade is a Seismic Category I structure that houses control equipment and operation personnel. The CB structure above grade is a Seismic Category II structure.

The main control area envelope is separated from the rest of the CB by walls, floors, doors and penetrations, which have three-hour fire ratings.

The lowest elevation in the CB is divided into separate divisional areas for instrumentation and control equipment. Interdivisional boundaries have the following features:

- Inter-divisional walls, floors, doors and penetrations, and penetrations in the external CB walls, have three-hour fire ratings.
- Watertight doors prevent flooding in one division or the adjoining corridor from propagating to other divisions.

Watertight doors between flood divisions have open/close sensors with status indication and alarms in the main control room.

The CB flooding that results from component failures in any of the CB divisions does not prevent safe shutdown of the reactor.

To protect the CB against external flooding, penetrations in the external walls below flood level are provided with watertight seals.

The CB is designed and constructed to accommodate the dynamic and static loading conditions associated with the various loads and load combinations, which form the structural design basis. The loads are those associated with:

- Natural phenomena—wind, floods, tornadoes (including tornado missiles), earthquakes, rain and snow.
- Internal events—floods, pipe breaks and missiles.
- Normal plant operation—live loads, dead loads and temperature effects.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.6-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Control Building.

Table 2.16.6-1

ITAAC For The Control Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the CB is shown on Figures 2.16.5-2 through 2.16.5-5 and 2.16.5-11.	1. Inspections of the as-built structure will be conducted.	1. The as-built CB conforms with the basic configuration shown on Figures 2.16.5-2 through 2.16.5-5 and 2.16.5-11.
2. The Main Control Room envelope is separated from the rest of the CB by walls, floors, doors and penetrations, which have a three-hour fire rating.	2. Inspections of the as-built structure will be conducted.	2. The as-built CB has a Main Control Room envelope separated from the rest of the CB by walls, floors, doors and penetrations, which have a three-hour fire rating.
3. Inter-divisional walls, floors, doors and penetrations, and penetrations in the external CB walls have a three-hour fire rating.	3. Inspections of the as-installed inter-divisional boundaries and external wall penetrations to connecting tunnels will be conducted.	3. The as-installed walls, floors, doors and penetrations that form the inter-divisional boundaries, and penetrations in the external CB walls have a three-hour fire rating.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The CB is protected against external and internal flooding. a. For external flooding, penetrations in the external walls below flood level are provided with flood protection features. b. The CB external entrances to the CB are provided with sills. c. The CB has divisional areas with watertight doors and walls to protect against inter-divisional flooding.	4. Inspections of the as-built walls, penetrations, and doors will be conducted.	4. Flood protection features are provided as, described in the Design Commitment.

2.16.7 Fuel Building

Design Description

The Fuel Building (FB) (Figures 2.16.5-1 through 2.16.5-8 and Figure 2.16.5-10) contains the spent fuel pool, cask loading area, fuel handling systems and storage areas, lower connection to the inclined fuel transfer system, overhead crane, and other plant systems and equipment.

The FB is a Seismic Category I structure except for the penthouse that houses HVAC equipment. The penthouse is a Seismic Category II structure. The FB is a rectangular reinforced concrete box type shear wall structure consisting of walls and slabs and is supported on a foundation mat. The FB is integrated with the RB, sharing a common wall between the RB and FB as well as a large common foundation mat. The building is partially embedded.

The walls forming the boundaries of the FB and penetrations through these walls have three-hour fire ratings.

The FB is designed and constructed to accommodate the dynamic and static loading conditions associated with the various loads and load combinations, which form the structural design basis. The loads are those associated with:

(1) Natural phenomena—wind, floods, tornadoes (including tornado missiles), earthquakes, rain and snow.

(2) Internal events—floods, pipe breaks and missiles.

(3) Normal plant operation—live loads, dead loads and temperature effects.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.7-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Fuel Building.

Table 2.16.7-1

ITAAC For The Fuel Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the FB is shown on Figures 2.16.5-1 through 2.16.5-8 and 2.16.5-10.	1. Inspections of the as-built structure will be conducted.	1. The as-built FB conforms with the basic configuration shown on Figures 2.16.5-1 through 2.16.5-8 and Figure 2.16.5-10.
2. Walls, doors and and penetrations in the external FB walls have a three-hour fire rating.	2. Inspections of the as-installed inter-divisional boundaries and external wall penetrations to connecting tunnels will be conducted.	2. The as-installed external walls, doors and penetrations that form the FB boundaries have a three-hour fire rating.
3. To protect the FB against an external flood: a. Penetrations in the external walls below flood level are provided with flood protection seals. b. The external entrances to the FB are provided with sills.	3. Inspections of the as-built walls, penetrations, and doors will be conducted.	3. Flood protection features are provided as, described in the Design Commitment.

2.16.8 Turbine Building

Design Description

The Turbine Building (TB) encloses the turbine-generator, main condenser, condensate and feedwater systems, condensate purification system, offgas system, turbine-generator support systems and bridge crane. The TB is a Seismic Category II nonsafety-related structure. The building is partially embedded. Shielding is provided for the turbine on the operating deck.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this building.

2.16.9 Radwaste Building

Design Description

The Radwaste Building (RW) houses the equipment and floor drain tank(s), sludge phase separator(s), resin hold up tank(s), detergent drain collection tank(s), concentrated waste tank(s), chemical drain collection tank(s), associated pumps and mobile systems for the radioactive liquid and solid waste treatment systems. Tunnels connect the Radwaste Building to the Reactor, Fuel and Turbine Buildings.

The RW is a reinforced concrete box type structure consisting of walls and slabs and is supported on a foundation mat. The RWB is a Non-Seismic Category (NS) structure. The substructure, consisting of foundation and walls up to the spill height of the building housing radioactive waste systems, is designed according to the most stringent safety classifications defined in Regulatory Guide 1.143. The building is partially embedded.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this building.

2.16.10 Other Buildings and Structures

Design Description

The Electrical Building houses the two nonsafety-related standby diesel generators, associated supporting systems and equipment, and nonsafety-related nonessential power supplies. The Electrical Building also provides space for the Technical Support Center. The building is nonsafety-related and Seismic Category NS.

The Service Water Building houses the plant service water system pumps and associated water storage, piping and valves. The building is nonsafety-related and Seismic Category NS.

The Emergency Breathing Air System (EBAS) Building is a stand-alone structure, on its own foundation mat, adjacent to the Control Building. The EBAS building houses the compressed breathing air tank trains and their supporting equipment. The EBAS building is a Seismic Category I structure.

Other facilities include, the Service Building, the Water Treatment Building, Administration Building, Training Center, Sewage Treatment Plant, warehouse, and hot and cold machine shops. These are all of conventional size and design.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for the other buildings and structures.

2.17 INTAKE STRUCTURE AND SERVICING EQUIPMENT

2.17.1 Intake and Discharge Structure

The COL applicant provides requirements for the intake and discharge structure.

No entry for this system.

2.18 YARD STRUCTURES AND EQUIPMENT

2.18.1 Oil Storage and Transfer Systems

Design Description

The major components of this system are the fuel-oil storage tank, pumps, and day tanks. Each standby diesel has its own individual supply components. Each fuel-oil pump is controlled automatically by day-tank level and feeds its day tank from the storage tank.

The Oil Storage and Transfer system is nonsafety-related. The Oil Storage and Transfer System does not perform any safety-related function and is not required to achieve or maintain safe shutdown.

Instrumentation and Control

Critical and essential information is available in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.18.2 Site Security

The site security plan and requirements for the Site Security System shall be prepared and specified by the COL applicant.

No entry for this system.

3. NON-SYSTEM BASED MATERIAL

3.1 PIPING DESIGN

Design Description

Piping associated with fluid systems is categorized as either safety-related (i.e., Seismic Category I) or nonsafety-related (i.e., non-Seismic Category I). The piping designed for a design life of 60 years. Piping systems and their components are designed and constructed in accordance with the ASME Code requirements identified in the individual system Design Descriptions.

Piping systems are designed to ASME Code class and Seismic Category I requirements.

For ASME Code Class 1 piping systems, a fatigue analysis shall be performed in accordance with the ASME Code Class 1 piping requirements. Environmental effects shall be included in the fatigue analysis. The Class 1 piping fatigue analysis shall show that the ASME Code Class 1 piping fatigue requirements have been met.

For ASME Code Class 2 and 3 piping systems, piping stress ranges due to thermal expansion shall be calculated in accordance with the ASME Code Class 2 and 3 piping requirements. The piping stress analysis shall show that the ASME Code Class 2 and 3 piping thermal expansion stress range requirements have been met. For the ASME Code Class 2 and 3 piping systems and their components, which will be subjected to severe thermal transients, the effects of these transients shall be included in the design.

Piping systems that are qualified for leak-before-break design may exclude design features to mitigate the dynamic effects from postulated high energy pipe breaks.

Inspections, Tests, Analyses and Acceptance Criteria

Table 3.1-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for the Piping Design.

Table 3.1-1

ITAAC For The Generic Piping Design

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The piping system shall be designed to meet its ASME Code Class and Seismic Category I requirements.	1. Inspections of ASME Code required documents will be conducted.	1. ASME Code Certified Stress Report for the piping system concludes that the design complies with the requirements of ASME Code, Section III.
2. Systems, structures, and components, that are required to be functional during and following an SSE, shall be protected against or qualified to withstand the dynamic and environmental effects associated with postulated failures in Seismic Category I and nonsafety-related piping systems.	2. Inspections of the pipe analysis report, or Leak-Before-Break Report (if applicable), will be conducted. An inspection of the as-built high and moderate energy pipe break mitigation features (including spatial separation) will be performed.	2. A pipe analysis report and Leak-Before-Break Report (if applicable) concludes that for each postulated piping failure, the reactor can be shut down safely
3. The as-built piping shall be reconciled with the piping design required in Section 3.1.	3. A reconciliation analysis using the as-designed and as-built information will be performed.	3. An as-built stress report concludes that the as-built piping has been reconciled with the design documents used for design analysis. For ASME Code Class piping, the as-built stress report includes the ASME Code Certified Stress Report and documentation of the results of the as-built reconciliation analysis.

3.2 SOFTWARE DEVELOPMENT

Design Description

Reg. Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," endorses IEEE Std 7-4.3.2, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations." Reg. Guide 1.152 provides guidance for safety systems that use digital computer systems. NUREG-0800, Branch Technical Position HICB-14 (BTP-14) outlines the many activities to be considered when constructing a software development program for software-based control and instrumentation (C&I) systems, herein defined as safety-related software-based products. BTP-14 divides these activities into 11 separate software development plans. The overall requirement is that software plans address and document the essential elements of each of the 11 development groups in BTP-14. GE has developed and accumulated the experience and documentation for compliance with BTP-14 expectations in GE's design and implementation of software-based products in current products, including the Advanced Boiling Water Reactor (ABWR). The ESBWR software life cycle process planning documents, based on Section 2.1 of BTP-14, were developed and submitted to the NRC for review in support of DCD Certification. GE developed the ESBWR software development program using the experience gained from use of GE's current software development plans. The development of the plans will address various aspects of the software development and quality addressed in the guidance documents of various related industry standards and regulatory guidance. In certain cases, deviation may be taken from the detailed guidance in those documents, where the GE software plans will be followed.

This section summarizes the development activities to be implemented for ESBWR safety-related software-based products. Table 3.2-1 outlines the following plans.

- Software Management Plan (SMP)
- Software Development Plan (SDP)
- Software Quality Assurance Plan (SQAP)
- Software Integration Plan (SIntP)
- Software Installation Plan (SIP)
- Software Operations and Maintenance Plan (SOMP)
- Software Training Plan (STrngP)
- Software Safety Plan (SSP)
- Software Verification and Validation Plan SVVP)
- Software Configuration Management Plan (SCMP)

Software Management Plan

The Software Management Plan (SMP) describes the management of software development, in accordance with Reg. Guide 1.173, "Development Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants." This plan, in conjunction with other plans described in this section, addresses the various elements described in the related guidance documents including Reg. Guide 1.173.

Software Development Plan

The Software Development Plan (SDP) outlined in Table 3.2-1, defines the managerial processes necessary to accomplish the design and development of the ESBWR software-based products. This plan in conjunction with other plans described in this section addresses the various elements described in the related guidance documents including IEEE-1058.

Software Quality Assurance Plan

The Software Quality Assurance Plan (SQAP) describes a systematic approach to development and implementation for ESBWR software development. This plan identifies the documentation to be prepared during the software development, verification, validation, use, and maintenance. This plan is conformed to the requirements of 10 CFR 50, Appendix B and is consistent with the requirements specified in IEEE-730, "IEEE Standard for Quality Assurance Plans." This plan, in conjunction with other plans described in this section, addresses the various elements described in the related guidance documents, including IEEE-730.

Software Integration Plan

The Software Integration Plan (SIntP) describes the software test activities to be carried out during the development of software-based products. This plan, in conjunction with other plans described in this section, addresses and meets the expectations of Reg. Guide 1.170, "Software Test Documentation for Digital Computer Software used in Safety Systems of Nuclear Power Plants," and Reg. Guide 1.171, "Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants."

Software Installation Plan

The Software Installation Plan (SIP) summarizes the management, implementation, and resource characteristics required to implement the software installation program.

Software Operational and Maintenance Plan

The Software Operations and Maintenance Plan (SOMP) complies with the software maintenance guidelines specified in IEEE Std. 828, "IEEE Standard for Software Configuration Management Plans," and part of the IEEE Std. 1042, "IEEE Guide to Software Configuration Management," in conjunction with the Software Configuration Management Plan. This plan addresses all software-based products. The Software O&M Plan describes the development of the required instructions and guidelines to operate and maintain the software-based products.

Software Training Plan

The Software Training Plan (STrngP) describes the management, implementation, and resource characteristics of the training program. The plan addresses the required training for staff working on the design, development, peer review, and testing of the software-based products, as well as requirements for the training program for the utility operating and maintaining the software-based products.

Software Safety Plan

This Software Safety Plan (SSP) establishes the processes and activities intended to ensure the safety of the safety-related software for the software-based product and to address the potential software risks. This plan, in conjunction with other plans described in this section, addresses the

various elements described in the related guidance documents. This plan, in conjunction with the other nuclear safety plans, programs, and procedures associated with the ESBWR design and development, provides compliance with the Software Safety Plan described in NUREG-0800, Chapter 7, BTP-14, and IEEE 1228, "Software Safety Plans."

Software Verification and Validation Plan

The Software Verification and Validation Plan (SVVP) defines the verification and validation process to assure the following:

- Verification and Validation (V&V) shall be performed as a controlled and documented evaluation of the conformity of the developed design to the documented design requirements, at the completion of each phase of baseline review;
- Software outputs of each life cycle phase are in compliance with the requirements defined in the previous phase;
- Final software products meet the system requirements and applicable established standards; and
- The Software Verification and Validation Plan, in conjunction with other plans described in this section, addresses the various elements and meets the expectations specified in Reg. Guide 1.168, "Verification, Validation, Reviews and Audits for Digital Computer Software Used in Safety Systems for Nuclear Power."

Software Configuration Management Plan

The Software Configuration Management Plan (SCMP), defines the specific products and systems to which it is applicable, the organizational responsibilities for software configuration management, and methods to be applied. This outline, in conjunction with other plans described in this section, addresses the various elements described in the related guidance documents including Reg. Guide 1.169, "Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants."

Inspections, Tests, Analyses and Acceptance Criteria

Table 3.2-1 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria, which will be applied to the safety-related software life-cycle.

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Software Management Plan (SMP) describes the management of the software development described in Reg. Guide 1.173, "Development Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants".	1. The Software Management Plan will be reviewed.	1. The SMP defines the organizational structure and procedural processes used to develop and maintain software. Software design requirements related to hardware specifications, configurations and validation criteria will be a part of the plan.
2. Software Development Plan (SDP) defines the managerial processes necessary to accomplish the design and development of the ESBWR software-based products. The Software Development Plan addresses the various elements described in related guidance documents including IEEE-1058.	2. The Software Development Plan will be reviewed.	2. The Software Development Plan defines the managerial processes necessary to accomplish the design and development of ESBWR software-based products and as such is updated throughout the course of the project.
3 The Software Quality Assurance Plan (SQAP) describes a systematic approach to the development and use of ESBWR software. It also identifies the documentation to be prepared during the software development, verification, validation, use, and maintenance. It is conformed to the requirements of 10 CFR 50, Appendix B and is	3. The Software Quality Assurance Plan will be reviewed.	3. The SQAP defines the quality assurance management of the software-based product. This includes the organizational structure and responsibilities, interface requirements, design standards, and procedural structure and process controls that are used to develop and use the software-based products.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
consistent with the requirements specified in IEEE-730, "IEEE Standard for Quality Assurance Plans".		
4. The Software Integration Plan (SintP) summarizes the management, implementation, and resource characteristics of the integration program.	4 The software integration plan will be reviewed.	4. The software integration plan describes the software integration organization, process, and controls required to integrate applicable software.
5. The Software Installation Plan (SIP) summarizes the management, implementation, and resource characteristics of the installation program.	5. The software installation plan will be reviewed.	5. The software installation plan describes the software installation organization, process, and controls required to install applicable software.
6. The Software Operations and Maintenance Plan (SOMP), which complies with the software maintenance guidelines specified in IEEE Std. 828, "IEEE Standard for Software Configuration Management Plans", and parts of IEEE Std. 1042, "IEEE Guide to Software Configuration Management", will be established for software-based products in conjunction with the SCMP.	6. The Software Operations and Maintenance Plan will be reviewed.	6. The Software Operations and Maintenance Plan (SOMP) in conjunction with the SCMP describes the procedure, organization, and test criteria to operate and maintain software-based products.
7. The Software Training Plan (STrngP) addresses the required training for staff working in the design, development,	7. The Software Training Plan will be reviewed.	7. The Software Training Plan contains the description of the organization supporting the software-based product,

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
peer review, and testing of the software based products, as well as requirements for the training program for the utility operating and maintaining the software based products.		training personnel qualification requirements, training methods and tools and testing criteria.
8. The Software Safety Plan (SSP) establishes the processes and activities intended to ensure the safety of the safety-related software for the software-based product and to address the potential software risks. The Software Safety Plan addresses the various elements described in IEEE 1228, "Software Safety Plans."	8. The Software Safety Plan will be reviewed.	8. The Software Safety Plan specifies the purpose and scope of the software safety plan, defining the organizational responsibilities for the plan's activities, safety analyses performed and the final documentation requirements.
9. The Software Verification and Validation Plan (SVVP) defines the verification and validation process to assure the following: a. V&V shall be performed as a controlled and documented evaluation of the conformity of the developed design to the documented design requirements at each phase of baseline review; b. Software outputs of each life cycle phase are in compliance with the requirements defined in the previous phase;	9 The Software Verification and Validation Plan will be reviewed.	9 The Software Validation and Verification Plan provides a description of the organizational structure and responsibilities of the software verification and validation organizations, methods and process used for V & V activities, and final acceptance criteria.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
c. Final software product meets the system requirements and applicable established standards; d. The Software Verification and Validation Plan meets the expectations described in Reg. Guide 1.168, "Verification. Validation, Reviews and Audits for Digital Computer Software Used in Safety Systems for Nuclear Power."		

3.3 HUMAN FACTORS ENGINEERING

Design Description

The Human Factors Engineering (HFE) design process represents a comprehensive, synergistic, iterative design approach for the development of human-centered control and information infrastructure for the ESBWR as depicted in Figure 3.3-1 Man-Machine Interface System and Human Factors Engineering Implementation Plan.

HFE Program Goals - The general objectives of the program can be stated in "human-centered" terms, which, as the HFE program develops, will be refined and used as a basis for HFE planning, test and evaluation activities. Generic "human-centered" HFE design goals include the following:

- Personnel tasks can be accomplished within time and performance criteria;
- The Human-System Interfaces (HSIs), procedures, staffing/qualifications, training and management and organizational support will support a high degree of operating crew situation awareness;
- The plant design and allocation of functions will maintain operation vigilance and provide acceptable workload levels i.e., to minimize periods of operator underload and overload;
- The operator interfaces will minimize operator error and will provide for error detection; and
- Recovery capability.

Applicable Facilities - The HFE program will address the main control room, remote shutdown facility, technical support center (TSC), emergency operations facility (EOF), and safety related local control stations.

Applicable HSIs, Procedures and Training - The applicable HSIs, procedures, and training included in the HFE program will include all operations, accident management, maintenance, test, inspection and surveillance interfaces (including procedures). This includes monitoring the designs being presented by ESBWR suppliers, to ensure that supplier design are consistent with the HFE requirements of the ESBWR HFE Program.

Applicable Plant Personnel - Plant personnel who will be addressed by the HFE program include licensed control room operators as defined in 10 CFR Part 55 and the following categories of personnel defined by 10 CFR 50.120: nonlicensed operators, shift supervisor, shift technical advisor, instrument and control technician, electrical maintenance personnel, mechanical maintenance personnel, radiological protection technician, chemistry technician, and engineering support personnel to the extent that they perform tasks that are directly related to plant safety.

The Man-Machine Interface System (MMIS) will employ modern digital technology to implement the majority of the monitoring, control, and protection functions for the ESBWR. Description of the technology is contained in the ESBWR system documentation prepared for the ESBWR DCD Chapter 7. Segmentation of major functions, separation of redundant equipment within a segment, and use of fault tolerant equipment will provide reliability and protection against the propagation of failures. Application of signal validation to selected

parameters will be used to assure plant operators have data of high quality. Multiplexed data communication will be used to reduce the cost and complexity of the instrumentation and control cable runs throughout the plant. The high accuracy and drift-free operation of the digital systems will reduce the overall maintenance calibration burden. Fiber optic cables for data transmission will be used to provide high data transmission rates with electrical isolation and protection from electromagnetic interference at reduced costs.

Standardization of hardware and software, and modularity of design will be used to simplify maintenance and provide protection against obsolescence.

It is expected that the MMIS using modern technologies will result in significant cost savings over the life of the plant through higher availability factors, lower maintenance costs, and reduced inadvertent plant trips.

The same approach to the HSI will be taken toward the design of the technical support center, emergency operations facility, remote shutdown stations, and safety-related local control stations. The general development of eleven key implementation plans, analyses, and evaluation of the following are identified:

(1) Operating experience review

(2) System Functional requirements analysis

(3) Function allocation analysis

(4) Task Analysis

(5) Staffing and qualifications

(6) Human reliability analysis

(7) HSI design implementation

(8) Procedure design

(9) Training design (COL Applicant)

(10) Human factors verification and validation

(11) Design implementation (COL Applicant)

(12) Human performance monitoring (COL Applicant)

Inspections, Tests, Analyses and Acceptance Criteria

Because the HSI technology is continually advancing, details of the HFE design will not be complete before the NRC issuance of a design certification. As such, this presentation under 10 CFR Part 52 primarily focuses on the HFE design process.

Table 3.3-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for Human Factors Engineering.

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. A multi-disciplinary Human Factors Engineering (HFE) Design Team is established and is comprised of personnel with expertise in HFE and in other technical areas relevant to the HSI design, evaluation and operation.	1. The composition of the HFE Design Team will be reviewed.	1. The HFE design team is comprised of the following expertise: i. Technical Project Management ii. Systems Engineering iii. Nuclear Engineering iv. Control and Instrumentation Engineering v. Architect Engineering vi. Human Factors vii. Plant Operations viii. Computer Systems Engineering ix. Plant Procedure Development x. Personnel Training xi. System Safety Engineering xii. Maintainability/Inspectability Engineering xiii. Reliability/Availability Engineering
2. a. An HFE Program Plan is developed which establishes that the primary human-system interfaces are developed,	2. a. The HFE Program Plan will be reviewed.	2. a. The Human Factors Engineering (HFE) Program Plan establishes: i. Human-System Interface (HSI) design

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
designed, and evaluated based upon human factors systems analysis and reflect human factors principles. The HSI scope applies to the control and monitoring interfaces of the plant operations personnel in the Main Control Room and Remote Shutdown System.		and evaluation methods and criteria. ii. The HFE Program Plan addresses: (1) The ability of the operating personnel to accomplish assigned tasks. (2) Operator workload levels and vigilance. (3) Operating personnel "situation awareness," (4) The operators' information processing requirements. (5) Operator memory requirements. (6) The potential for operator error. iii. HSI design and evaluation scope which consists of the control and monitoring interfaces of the plant operations personnel in the Main Control Room (MCR) and Remote Shutdown System (RSS). The HSI scope addresses normal, AOO and accident plant operations, including consideration of plant operations during periods when plant system/equipment or HSI equipment is undergoing test, maintenance or

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		inspection. The HSI scope also addresses the development of operating technical procedures for normal, AOO and accident plant operations and the identification of personnel training needs applicable to the HSI design. iv. The HFE Design Team as being responsible for: (1) The development of HFE plans and procedures; (2) The oversight and review of HFE design, development, test, and evaluation activities; (3) The initiation, recommendation, and provision of solutions through designated channels for problems identified in the implementation of the HFE activities; (4) Verification of implementation of solutions to problems; (5) Assurance that HFE activities comply to the HFE plans and procedures, and (6) Phasing of activities.

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		v. The methods for the identification, closure and documentation of human factors issues. vi. The HSI design configuration control procedures.
2. b. The HFE Program Plan will also establish an HFE Issue Tracking System.	2. b. The HFE Program Plan will be reviewed.	2. b. The HFE Issues Tracking System will capture the issues arising from continued HSI design, OER and HFE validation life cycle.
3. a. An Operating Experience Review Plan is developed which establishes methods, criteria and guidance for identifying, analyzing and documenting lessons learned from published reviews of past events, PRA's and other available information sources.	3. a. The Operating Experience Review Plan will be reviewed.	3. a. The Operating Experience Review (OER) Plan establishes: i. Methods for review of operating experience ii. Guidelines for evaluation of human factors issues from the OER iii. Methods for documenting the OER
3. b. A review of Operating Experience is conducted.	3. b. The results of the Operating Experience Review will be reviewed.	3. b. The review of Operating Experience, as corrected to account for nonconformances, is conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		Operating Experience Review Plan.
4. a. A System Functional Requirements Analysis Implementation Plan is developed which establishes that plant system requirements are analyzed to identify those functions that must be performed to satisfy the objectives of each functional area. System functions analysis determines the objective, performance requirements, and constraints of the design; and establishes the functions that must be accomplished to meet the objectives and required performance.	4. a. The System Functional Requirements Analysis Implementation Plan will be reviewed.	4. a. The System Functional Requirements Analysis Implementation Plan establishes: i. Methods and criteria for identifying Plant Performance Requirements. ii. Methods and criteria to perform System Level Functional Analysis based on Plant Performance Requirements. iii. The description of functions critical to safety. iv. The method for developing graphical descriptions of the critical functions. v. The method for developing detailed functions narrative descriptions. vi. The method for defining the integration of related sub-functions or division of identified sub-functions.

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. b. An analysis of system functional requirements is conducted.	4. b. The analyses of the system functional requirements will be reviewed.	4. b. The system functional requirements analyses, as corrected to account for nonconformances, are conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the System Functional Requirements Analysis Implementation Plan.
5. a. An Allocation of Functions Implementation Plan is developed which establishes the methods, criteria and guidance for allocating functions to human, software, or machine during the HSI design implementation process.	5. a. The Allocation of Functions Implementation Plan will be reviewed.	5. a. The Allocation of Functions Implementation Plan establishes: i. The basis and criteria for allocation of functions including: (1) Philosophy for allocation of function (2) Allocation of function in the ESBWR Plant Design Process (3) A framework for function allocation (4) The allocation decision space (5) Precursors to function allocation ii. Phases and the processes of function allocation including: (1) Defining and evaluating the

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		functions (2) The methods of Function Allocation (3) Evaluation of Function Allocation
5. b. An analysis of the allocation of function is conducted.	5. b. The analysis of the allocation of function will be reviewed.	5. b. The function allocation analyses, as corrected to account for nonconformance, are conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the Allocation of Functions Implementation Plan.
6. a. A Task Analysis Implementation Plan is developed which establishes methods and criteria for the performance of task analyses used to identify the specific tasks that are needed for function accomplishment, including related information, control and task-support requirements.	6. a. The Task Analysis Implementation Plan will be reviewed.	6. a. The Task Analysis Implementation Plan establishes: i. Background and scope of the Task Analysis effort. ii. Phases of Task Analysis. iii. Data sources for conducting Task Analysis. iv. Performers and basic approaches of Task Analysis. v. Methods for conducting the initial

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		(high level) task analysis. vi. Methods for developing detailed task descriptions. vii. Methods for identifying critical tasks. viii. Methods for identifying requirements for alarms, displays, controls, and data processing. ix. Methods for evaluating task analysis results. x. Methods for documenting and assembling task analysis results to provide input for the development of personnel training programs. xi. Methods for the consideration of maintenance, test, and inspection requirements in the Task Analysis effort.
6. b. An analysis of tasks is conducted.	6. b. The task analyses will be reviewed.	6. b. The task analyses, as corrected to account for nonconformance, are conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the Task Analysis Implementation Plan.

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. a. A Staffing and Qualification Plan is developed which establishes a baseline staff and their qualifications for safely operating the ESBWR during normal power operation, as well as during transient events included in the plant design basis. 7. b. An analysis of the staffing and qualifications is conducted.	7. a. The Staffing and Qualification Plan will be reviewed. 7. b. The analyses of the staffing and qualifications will be reviewed.	7. a. The Staffing and Qualification Plan establishes: i. The assumptions and initial baseline staffing for reactor control and monitoring. ii. The process and methods for evaluating staffing and qualifications including the considerations from other elements of the HFE Program Plan and Risk Assessment. 7. b. The staffing and qualification analyses, as corrected to account for , are conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the Staffing and Qualification Plan.
8. a. A Human Reliability Assessment Implementation Plan is developed which establishes how human error identified and quantified in the PRA will be analyzed to determine if new or modified HSI design features are needed to reduce, the	8. a. The Human Reliability Assessment Implementation Plan will be reviewed.	8. a. The Human Reliability Assessment Implementation Plan establishes: i. The Methods to support/conduct PRA/HRA to identify significant risk reduction improvements relating to

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
likelihood and impact of errors.		the reliability of core and containment heat removal systems that can be practically implemented during the plant design. ii. The method for identifying the risk-important human actions to be used as input to the HFE design effort. iii. The methods for analyzing the human actions with respect to the design information, human error mechanisms, and available technology. iv. Methods to integrate the PRA and HRA activities into the HFE and plant design activities. v. Methods for documenting the HRA efforts.
8. b. A Human Reiliability Analysis is conducted.	8. b. The Human Reliability Analysis will be reviewed.	8. b. The Human Reliability Analysis, as corrected to account for nonconformances, is conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the Human Reliability Analysis Implementation Plan.

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. a. HSI Design Implementation Plan is developed which establishes the methods and criteria for Human System Interface (HSI) equipment design and associated work place factors, such as illumination in the Main Control Room (MCR) and in the Remote Shutdown System (RSD) area, which are consistent with accepted Human Factors Engineering (HFE) practices and principles. This plan will incorporate HFE principles and guidelines to achieve an integrated design of the control and instrumentation systems and HSI.	9. a. The HSI Design Implementation Plan will be reviewed.	9. a. The HSI Design Implementation Plan establishes: i. A summary of the HSI Design Implementation plan including design inputs, guidance, principles, and criteria which will be employed to generate, review, test, and document the design of the ESBWR human interfaces. ii. A description of the HSI Design Criteria including: (1) A list of design features or initial assumptions for the ESBWR design. (2) General requirements for the layout of control room equipment and panels, control and information systems, and communication systems. (3) Descriptions of integrated human factors design activities. iii. A description of conformance specifications specific to the hardware and software requirements

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		for the plant process computer including the CRTs and non-safety related Flat Panel Displays. iv. A description of the HSI documentation and reporting plans.
9. b. The HSI design implementation is developed.	9. b. The HSI design implementation will be reviewed.	9. b. The HSI design implementation and analyses, as corrected to account for nonconformance, are conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the HSI Design Implementation Plan.
10. a. A Plant and Emergency Operating Procedure Development Implementation Plan is developed which establishes that plant and emergency operating procedures are developed to support and guide human interaction with plant systems and to support and guide human interactions in the control of plant operations. Human engineering principles and criteria are applied in the procedures development.	10. a. The Plant and Emergency Operating Procedure Development Implementation Plan will be reviewed.	10. a. The Plant and Emergency Operating Procedure Development Implementation Plan establishes: i. That operator actions identified in the task analysis are used as the basis for specifying the procedures for operations. ii. That the procedures to be developed address normal, abnormal, and emergency plant operations including consideration of plant operations during periods when plant

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		systems/equipment and primary operator interface (i.e., main control room) equipment is undergoing test, maintenance or inspection. iii. Methods and criteria for development of the operating technical procedures. iv. That a Writer's Guide is developed which establishes the process for developing the technical procedures for normal plant and system operation, abnormal plant operations, emergency plant operations, and for responding to plant alarm conditions. v. Method to analyze the impact of providing computer-based procedures. vi. Method to evaluate the physical means by which operators access and use procedures.
10. b. The Plant and Emergency Operating Procedures are developed.	10. b. The Plant and Emergency Operating procedure development results will be reviewed.	10. b. The development of the Plant and Emergency Operating Procedures, as corrected to account for nonconformances, is conducted in accordance with the requirements of the

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		Human Factors Engineering Program Plan and the Plant and Emergency Operating Procedure Development Implementation Plan.
11. a. A Training Program Development Implementation Plan is developed which establishes reasonable assurance plant personnel have the knowledge, skills, and abilities to properly perform their roles and responsibilities.	11. a. The Training Program Development Implementation Plan will be reviewed.	11. a. The Training Program Development Implementation Plan establishes: i. A description of the systems approach to the design and development of training programs based on the job and task requirements coordinated with the other elements of the HFE design process. ii. A description of the scope of training including categories of personnel to be trained, specific plant conditions and operational activities to be included, and control areas and other human system interface elements to be addressed. iii. Methods and guidance for the development and conduct of training including personnel qualifications, content, and delivery methods.

Table 3.3-1 ITAAC For Human Factors Engineering		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
		iv. Methods and guidance for the verification of accuracy and completeness of training materials, evaluation of training effectiveness, program modification, and periodic retraining.
11. b. The training program is developed and administered.	11. b. The development of the training program will be reviewed.	11. b. The training program, as corrected to account for nonconformances, is developed and administered in accordance with the requirements of the Human Factors Engineering Program Plan and the Training Program Implementation Plan.
12. a. A Human Factors Verification and Validation (V&V) Implementation Plan is developed which establishes the process of determining and documenting that an implemented design (a product, process, procedure, method, etc.) meets its specifications and effectively serves the purpose for which it was intended.	12. a. The Human Factors V&V Plan will be reviewed.	12. a. The Human Factors V&V Implementation Plan establishes: i. A list of the Human Factors Verification and Validation Requirements. ii. A description of the Human Factors Verification and Validation (HF V & V) Activities including: (1) Human-System Interface Task

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		Support Verification. (2) Human Factors Engineering Design Verification. (3) Integrated System Validation. (4) Human Factors Issue Resolution Verification. (5) Final Plant HFE/HSI Design Verification. iii. The implementation methods and procedures for the HF V & V activities. iv. The schedule of the HF V & V activities.
12. b. A human factors engineering analysis of the integrated HSI design is conducted.	12. b. The analyses of the integrated HSI design will be reviewed	12. b. The human factors engineering analysis of the HSI design, as corrected to account for nonconformance, is conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the Human Factors V&V Implementation Plan.
13. a. A Design Implementation Plan is developed which establishes processes and procedures for the design implementation,	13. a. The Design Implementation Plan will be reviewed.	13. a. The Design Implementation Plan establishes: i. Process and procedures for the

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
start-up, verification, validation, and testing of the ESBWR plant COL applications.		software quality program for the hardware/software design and development. ii. Process and procedures for the Electromagnetic Compatibility Compliance Plan. iii. Requirements, process, and procedures for the Set-Point Methodology Plan. iv. Methods and procedures for the qualification of safety related I&C equipment. v. Process and procedures to verify that the as-built design conforms to the verified and validated design that resulted from the HFE design process. vi. Milestones and responsibilities for the design implementation and long term HFE Life-cycle management.
13. b. The design implementation is conducted.	13. b. The plans and reviews of the Design Implementations will be reviewed.	13. b. The design implementation events, as corrected to account for nonconformances, are conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the Design

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		Implementation Plan.
14. a. A Human Performance Monitoring Plan is developed which establishes a strategy for going forward with human performance monitoring that links human factors engineering methods used during the design with methods for monitoring human performance during operation by the COL applicant.	14. a. The human performance monitoring plan will be reviewed.	14. a. The Human Performance Monitoring Plan establishes: i. Criteria and methods to identify events or processes to use in the human performance monitoring strategy. ii. Methods for establishing the human performance monitoring strategy for use during operation by the COL applicant. iii. Guidance to establish corrective actions to reduce potential for recurrence of degraded performance when indicated by the monitoring strategy. iv. Organizational responsibilities for the development and conduct of the human performance monitoring plan.
14. b. The Human Performance Monitoring Plan is conducted.	14. b. The elements of the Human Performance Monitoring will be reviewed.	14. b. The Human Performance Monitoring, as corrected to account for nonconformances, is impemented in accordance with the requirements of the Human Factors

Table 3.3-1 ITAAC For Human Factors Engineering		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
		Engineering Program Plan and the Human Performance Monitoring Plan.

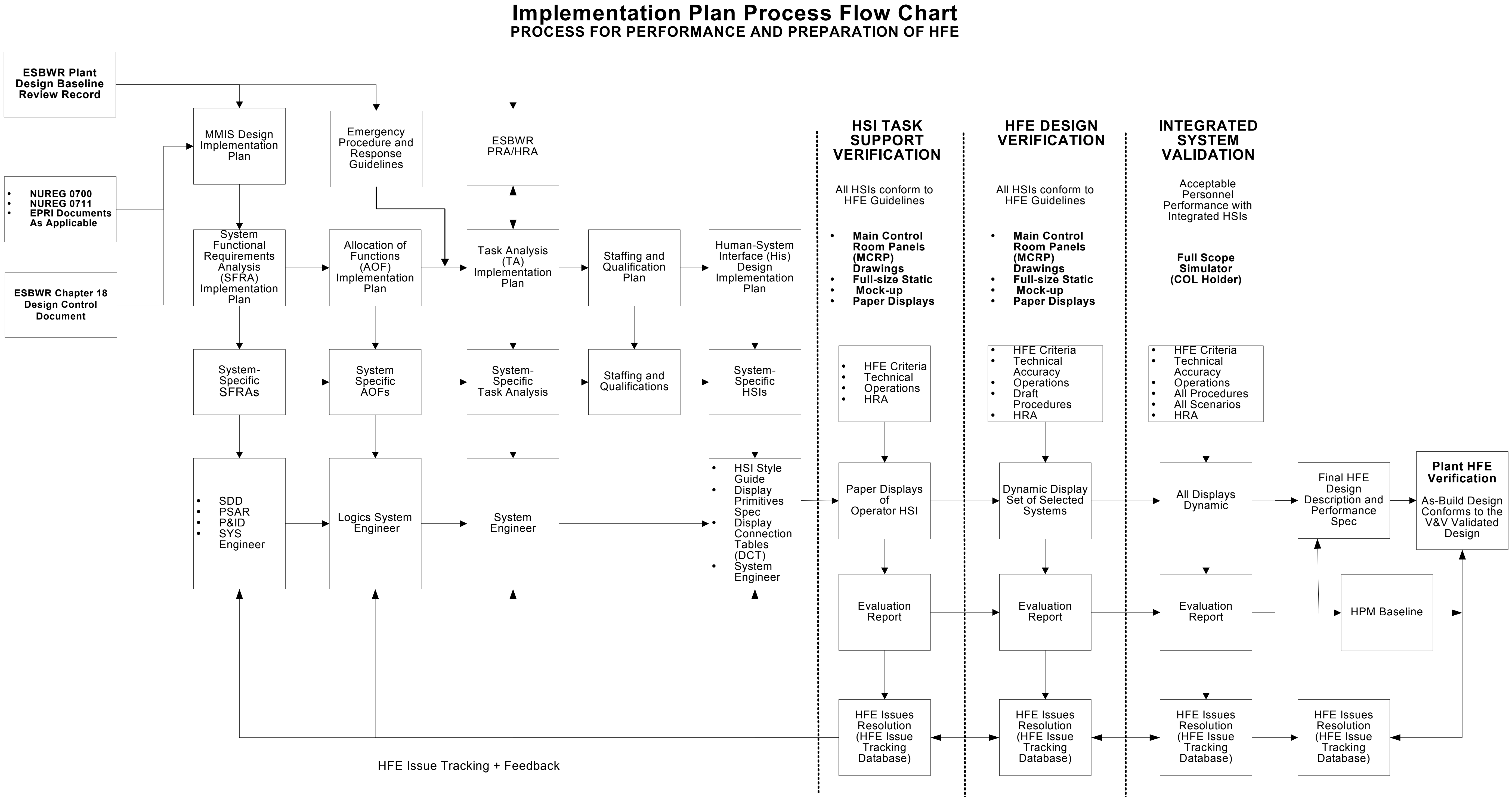


Figure 3.3-1. ESBWR MMIS HFE Implementation Plan Process Flowchart

3.4 RADIATION PROTECTION

Design Description

The ESBWR Standard Plant is designed in accordance with Regulatory Guide 8.8, i.e., to keep radiation exposures to plant personnel as low as reasonably achievable (ALARA). This section describes the component and system designs in addition to the equipment layout employed to maintain radiation exposures ALARA. Consideration of individual systems is provided to illustrate the application of these principles.

Material selection for primary coolant piping, tubing, vessel internal surfaces, and other components in contact with the primary coolant is discussed in the following paragraphs.

Carbon steel is used in a large portion of the system piping and equipment outside of the nuclear steam supply system. Carbon steel is typically low in nickel content and contains a very small amount of cobalt impurity.

Stainless steel is used in portions of the system such as the reactor internal components and heat exchanger tubes where high corrosion resistance is required. The nickel content of the stainless steels is in the 9 to 10.5% range and is controlled in accordance with applicable ASME material specifications. Cobalt content is controlled to less than 0.05% in the XM-19 alloy used in the control rod drives.

A previous review of materials certifications indicated an average cobalt content of only 0.15% in austenitic stainless steels.

Ni-Cr-Fe alloys such as Inconel 600 and Inconel X750, which have high nickel content, are used in some reactor vessel internal components. These materials are used in applications for which there are special requirements to be satisfied (such as possessing specific thermal expansion characteristics along with adequate corrosion resistance) and for which no suitable alternative low-nickel material is available. Cobalt content in the Inconel X750 used in the fuel assemblies is limited to 0.05%.

Stellite is used for hard facing of components that must be extremely wear resistant. Use of high cobalt alloys such as Stellite is restricted to those applications where no satisfactory alternative material is available. An alternative material (Colmonoy) has been used for some hard facings in the core area.

The radiation shielding protects operating personnel and the general public from radiation emanating from the reactor, the power conversion systems, the radwaste process systems, and the auxiliary systems, while maintaining appropriate access for operation and maintenance. The radiation shielding keeps radiation doses to equipment below levels at which disabling radiation damage occurs.

Specifically, the shielding requirements in the plant are designed to perform the following functions:

- limit the exposure of the general public, plant personnel, contractors, and visitors to levels that are ALARA and within 10 CFR 20 requirements;
- limit the radiation exposure of personnel, in the unlikely event of an accident, to levels that are ALARA and which conform to the limits specified in 10 CFR 50, Appendix A,

Criterion 19 to ensure that the plant is maintained in a safe condition during an accident; and

- limit the radiation exposure of critical components within specified radiation tolerances, to assure that component performance and design life are not impaired.

The radiation control aspects of the HVAC systems apply the following design objectives:

- The systems shall be designed to make airborne radiation exposures to plant personnel and releases to the environment ALARA. To achieve this objective, the guidance provided in Regulatory Guide 8.8 shall be followed.
- The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance shall be kept below the limits of 10 CFR 20 during normal power operation. This is accomplished by establishing in each area a reasonable compromise between specifications on potential airborne leakages in the area and HVAC flow through the area.

The following systems are provided to monitor area radiation and airborne radioactivity within the plant:

- The Area Radiation Monitoring System (ARMS) continuously measures, indicates and records the gamma radiation levels at strategic locations throughout the plant except within the primary containment, and activates alarms in the main control room as well as in local areas to warn operating personnel to avoid unnecessary or inadvertent exposure to radiation. This system is classified as nonsafety-related.
- The Containment Monitoring System (CMS) continuously measures, indicates, and records the gamma radiation levels within the primary containment (drywell and suppression chamber), and activates alarms in the main control room on high radiation level.
- Airborne radioactivity in effluent releases and ventilation air exhausts is continuously sampled and monitored by the Process Radiation Monitoring System (PRMS) for noble gases, air particulates and halogens. Airborne contamination is sampled and monitored at the stack common discharge, in the off-gas releases, and in the ventilation exhaust from the reactor, radwaste and turbine buildings. Samples are periodically collected and analyzed for radioactivity. In addition to this instrumentation, portable air samplers are used for compliance with 10 CFR 20 restrictions to check for airborne radioactivity in work areas prior to entry where potential radiation levels may exit that exceed the allowable limits. The radiation instrumentation that monitors airborne radioactivity is classified as nonsafety-related.

The locations requiring access to mitigate the consequences of an accident during the 100-day post-accident period are the control room, the technical support center, the remote shutdown panels, the primary containment sample station (Post Accident Sampling Subsystem), the health physics facility (counting room), the control room air bottles (Emergency Breathing Air System), the isolation condenser (IC) pool refill nozzles, and the nitrogen gas supply bottles. Each area has low post LOCA radiation levels. The dose evaluations are within regulatory guidelines.

The post-accident radiation zone maps for the areas in the Reactor Building have been developed. These zone maps represent the maximum gamma dose rates that exist in these areas during the post-accident period.

Inspections, Tests, Analyses and Acceptance Criteria

Table 3.4-1 provides definitions of the inspections, test and/or analyses, together with associated acceptance criteria, which will be undertaken for ventilation and airborne monitoring.

Table 3.4-1

ITAAC For Ventilation and Airborne Monitoring

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Plant design provides for containment of airborne radioactive materials, and the ventilation system ensures that concentrations of airborne radionuclides are maintained at levels consistent with personnel access needs.	1. Expected concentrations of airborne radioactive material will be calculated by radionuclide for normal plant operations, anticipated operational occurrences for each equipment cubicle, corridor, and operating area requiring personnel access. Calculations will consider:	1. Calculation of radioactive airborne concentration demonstrates that:
	a. Design ventilation flow rates for each area.	a. For normally occupied rooms and areas of the plant (i.e., those areas requiring routine access to operate and maintain the plant) equilibrium concentrations of airborne radionuclide will be a small fraction of the occupational concentration limits listed in 10 CFR 20 Appendix B.
	b. Typical leakage characteristics for equipment located in each area, and	b. For rooms that require infrequent access (such as for non-routine equipment maintenance), the ventilation system is capable of reducing radioactive airborne concentrations to (and maintaining them at) the occupational concentration limits listed in 10 CFR 20 Appendix B during the periods that occupancy is required.
	c. A radiation source term in each fluid system will be determined based upon an assumed off gas rate of 3,700 MBq/second (30 minute decay) appropriately adjusted	c. For rooms that seldom require access, plant design provides containment and ventilation to reduce airborne contamination spread to other areas of

Table 3.4-1

ITAAC For Ventilation and Airborne Monitoring

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	for radiological decay and buildup of activated corrosion and wear products.	lower contamination.
2. Airborne radioactivity monitoring is provided for those normally occupied areas of the plant in which there exists a significant potential for airborne contamination. The airborne radioactivity system: a. Has the capability of detecting the time integrated concentrations of the most limiting internal dose particulate and iodine radionuclides in each area equivalent to the occupational concentration limits in 10 CFR 20, Appendix B for 10 hours. b. Provides local audible alarms (visual alarms in high noise areas) with variable alarm set points, and readout/annunciation capability.	2. An analysis of the as-built airborne radioactivity monitoring system will be performed.	2. Airborne radioactivity monitoring system is installed as defined in this certified design commitment.

3.5 INITIAL TEST PROGRAM

Design Description

The ESBWR Initial Test Program (ITP) is a program that will be conducted following completion of construction and construction-related inspections and tests and extends to commercial operation. The test program will be composed of preoperational and startup test phases. The general objective of the ITP is to confirm that performance of the as-built facility is in compliance with the design characteristics used for safety evaluations.

The preoperational test phase of the ITP will consist of those test activities conducted prior to fuel loading. Preoperational testing will be conducted to demonstrate proper performance of structures, systems, components, and design features in the assembled plant. Tests will include, as appropriate, logic and interlocks test, control and instrumentation functional tests, equipment functional tests, system operational test, and system vibration and expansion measurements.

The startup test phase of the ITP will begin with fuel loading and extends to commercial operation. The primary objective of the startup phase testing will be to confirm integrated plant performance with the nuclear fuel in the reactor pressure vessel and the plant at various power levels. Startup phase testing will be conducted at five test conditions during power ascension: open vessel, heatup, low power, mid-power, and high power. The following tests will be conducted during power operation testing:

(1) Core performance analysis,

(2) Steady-state testing,

(3) Control system tuning and demonstration, and

(4) System transient tests; and

(5) Major plant transients (including trips).

Testing during all phases of the ITP will be conducted using step-by-step written procedures to control the conduct of each test. Such test procedures will delineate established test methods and applicable acceptance criteria. The test procedures will be developed from preoperational and startup test specifications. Approved test procedures will be made available to the NRC approximately 60 days prior to their intended use for preoperational tests and 60 days prior to scheduled fuel loading for startup phase tests. The preoperational and startup test specifications will also be made available to the NRC. Administratively, the ITP will be controlled in accordance with a startup administrative manual. This manual will contain the administrative requirements that govern the conduct of test program, review, evaluation and approval of test results, and test records retention.

Inspections, Tests, Analyses and Acceptance Criteria

This section represents a commitment that combined operating license applicants referencing the certified design will implement an ITP that meets the objectives presented above. ITAAC, aimed at verification of ITP implementation, are neither necessary nor required.

4. INTERFACE MATERIAL

This section provides the Tier 1 material for interface items. No Tier 1 information is provided for the conceptual design portions that are COL applicant scope.

4.1 ULTIMATE HEAT SINK

Design Description

In the event of an accident, the Ultimate Heat Sink (UHS) is provided by the Isolation Condenser/Passive Containment Cooling (IC/PCC) pools, which serve as the heat sinks for the Passive Containment Cooling System (PCCS). The IC/PCC pools have reserve capacity for 72 hours of heat removal without make-up. External resources, through safety-related emergency makeup water piping, provide makeup water for long-term heat removal. Sufficient reserve capacity is maintained on-site to extend the safe shutdown state from 72 hours through 7 days. The external connection and emergency makeup water piping are part of the Fuel and Auxiliary Pool Cooling System.

The external water source is COL applicant scope. A specific source will be designed for any facility that adopts the ESBWR certified design. This plant-specific water source will meet the interface requirements defined below.

Interface Requirements

The UHS external water source provides a backup source of water, in sufficient quantity and flowrate, for IC/PCC pool makeup and reactor water inventory control through a connection to the Fuel and Auxiliary Pools Cooling System (FAPCS).

The interface requirements identified for this system will be satisfied by providing site-specific systems, structures and components (SSCs) that are technologically similar to the SSCs for the certified ESBWR design. Consequently, verification of compliance with the interface requirements will be achieved by inspections, tests and analyses that are similar to those provided for the certified ESBWR design. These inspections, tests and analyses, together with their associated acceptance criteria will be developed by the COL applicant.

4.2 OFFSITE POWER SYSTEM

Design Description

The Offsite Power System is not within the scope of the certified design. A specific Offsite Power System will be designed for any facility, which has adopted the certified design. This plant-specific system will meet the interface requirements defined below.

Interface Requirements

The design of the ESBWR Reference Standard Plant is based on certain design bases (10 CFR 52 interface requirements), which are to be met in the design of the off-site power system. These design requirements follow:

- In case of failure of the normal preferred power supply circuit, the alternate preferred power supply circuit shall remain available through the reserve auxiliary transformers.
- The normal preferred circuit shall be electrically independent and physically separated from the alternate preferred circuit. The normal preferred and alternate preferred circuits may be connected to the same transmission systems provided the switchyard is fed by at least two transmission lines that can each supply the shutdown electrical loads and provided that both transmission lines are sufficiently separated. If the normal preferred and alternated preferred circuits are fed from separate transmission systems, each system shall be individually capable of supplying the shutdown electrical loads. They may use a common switchyard provided adequate separation exists.
- The switching station to which the main off-site circuit is connected shall have two full capacity main busses arranged such that:
  - Any incoming or outgoing transmission line can be switched without affecting another line.
  - Any circuit breaker can be isolated for maintenance without interrupting service to any circuit.
  - Faults of a single main bus are isolated without interrupting service to any circuit.
- The main, unit auxiliary and reserve auxiliary transformers shall meet the requirements of IEEE Standard C57.12.00.
- Circuit breakers and disconnecting switches are sized and designed in accordance with IEEE Standard C37.06.
- It is required that a minimum clearance of 12.7 m (50 ft) or two hours fire barrier exists between the reserve auxiliary transformers and the unit auxiliary transformers. In addition, the physical separation between transformers and oil collection systems shall be maintained.
- It is required that cables associated with the normal preferred and alternate preferred circuits be routed separately and in separate raceways apart from each other and on-site power system cables. However, they may share a common underground duct bank as indicated below.

- It is required that the alternate preferred circuit be separated from the normal preferred circuit by a minimum clear distance of 12.7m (50 ft) as measured from a plan view or by a two hour fire barrier. The associated control, instrumentation, and miscellaneous power cables of the reserve circuit shall, if located underground in the same duct bank as the cables associated in the normal preferred circuit between the switchyard and the power block, be routed in separate conduits or raceways and shall have separate manholes.
- It is required that cables associated with the alternate preferred circuit be routed in separate trenches within the switchyard from those associated with the normal preferred circuit, provided there is a common switchyard.
- The applicant shall review the proposed site specific configuration of power lines coming to the plant and the characteristics of the transmission system to which the plant is connected to determine the reliability of the off-site power system and verify that it is consistent with the probability risk analysis.
- It is required that provisions be made to permit disconnection of a failed auxiliary transformer and energizing of the rest of the auxiliary transformers in no more than 12 hours.
- It is required that the applicant provide a plant ground grid consisting of a ground mat below grade at the switchyard. The plant ground grid shall be connected with the foundation embedded loop grounding system provided for the remaining plant buildings including, but not limited to, the reactor and turbine buildings, cooling towers, unit auxiliary transformers, and the standby power source buildings.

4.3 POTABLE AND SANITARY WATER SYSTEM

Design Description

The Potable and Sanitary Water system is not within the scope of the certified design. A specific Potable and Sanitary Water system shall be designed for any facility, which has adopted the certified design. This plant-specific water system shall meet the interface requirements defined below.

Interface Requirements

Failure of the Potable and Sanitary Water system does not affect any safety-related structure, system or component.

The interface requirements identified for this system will be satisfied by providing site-specific systems, structures and components that are technologically similar to the systems, structures and components of the certified design. Consequently, verification of compliance with the interface requirements will be achieved by inspections, tests and analyses that are similar to those provided for the certified design. These inspections, tests and analyses, together with their associated acceptance criteria will be developed by the combined license applicant referencing the certified design.

4.4 PLANT SERVICE WATER SYSTEM

Design Description

The Plant Service Water System (PSWS) is the heat sink for the Reactor Component Cooling Water and the Turbine Component Cooling Water Systems. PSWS does not perform any safety-related function. There is no interface with any safety-related component.

The PSWS cooling towers and basins are not within the scope of the certified design. A specific design for this portion of the PSWS shall be selected for any facility, which has adopted the certified design. The plant-specific portion of the PSWS shall meet the interface requirements defined below.

Interface Requirements

The PSWS consists of two independent and 100% redundant open trains that continuously circulate raw water through the RCCWS and TCCWS heat exchangers. The heat removed is rejected to either the Normal Power Heat Sink (NPHS) or to the Auxiliary Heat Sink (AHS) (method is site specific).

The PSWS is designed so that neither a single active nor single passive component failure results in a complete loss of nuclear island cooling and/or plant dependence on any safety-related system. In the event of a LOPP, the PSWS supports the RCCWS in bringing the plant to cold shutdown condition in 36 hours assuming the most limiting single active failure. The PSWS pressure in the RCCWS heat exchangers is normally higher than that of the RCCWS to prevent RCCWS contamination of service water.

The design of the ESBWR Standard Plant cooling water systems is based on bounding ambient environmental conditions as well as generic BWR water quality requirements. Based on actual site conditions, the COL applicant will optimize the design of all of the cooling water systems, including, but not limited to, the following parameters:

- Main Condenser configuration (once through versus series water boxes)
- System configurations (once through versus closed loop) and materials
- Heat removal capacities of cooling water systems
- Performance characteristics of cooling water system components
- Cooling system makeup and blowdown capacities

The interface requirements identified for this system shall be satisfied by providing site-specific systems, structures and components that are technologically similar to the systems, structures and components of the certified design. Consequently, verification of compliance with the interface requirements shall be achieved by inspections, tests and analyses that are similar to those provided for the certified design. These inspections, tests and analyses, together with their associated acceptance criteria shall be developed by the combined license applicant referencing the certified design.

4.5 COOLING WATER SYSTEMS

Design Description

The cooling water systems provide the heat sink for power cycle waste heat. The cooling tower and intake and discharge structure are not within the scope of the certified design. A specific design for this portion of the cooling water systems shall be selected for any facility, which has adopted the certified design. The plant-specific portion of the cooling water systems shall meet the interface requirements defined below.

Interface Requirements

The design of the ESBWR Standard Plant Circulating Water Systems is based on bounding ambient environmental conditions as well as generic BWR water quality requirements. Based on actual site conditions, the cooling water systems will be optimized to, but not limited to, the following parameters:

- Main Condenser configuration (once through versus series water boxes);
- System configurations (once through versus closed loop) and materials;
- Heat removal capacities of cooling water systems;
- Performance characteristics of cooling water system components; and
- Cooling system makeup and blowdown capacities.

The interface requirements identified for this system shall be satisfied by providing site-specific systems, structures and components that are technologically similar to the systems, structures and components of the certified design. Consequently, verification of compliance with the interface requirements shall be achieved by inspections, tests and analyses that are similar to those provided for the certified design. These inspections, tests and analyses, together with their associated acceptance criteria shall be developed by the combined license applicant referencing the certified design.

4.6 MAKEUP WATER SYSTEM

Design Description

The Makeup Water System (MWS) is a nonsafety-related system, and has no safety design basis other than provision for safety-related containment penetrations and isolation valves. If available, the MWS can be used to provide makeup water to the Isolation Condenser / Passive Containment Cooling (IC/PCC) pools following an anticipated operational occurrence (AOO). However, this MWS function is not assumed or modeled in any safety analysis.

The Makeup Water System (MWS) demineralizer subsystem is not within the scope of the certified design. A specific MWS demineralizer subsystem shall be designed for any facility, which has adopted the certified design. This plant-specific MWS demineralizer subsystem shall meet the interface requirements defined below.

Interface Requirements

The MWS may provide makeup water to IC/PCC pools following an AOO. If available, the MWS demineralizer subsystem provides at least [1,499 1/min] of demineralized water for the Isolation Condenser (IC/PCC) pools. In addition, the MWS is designed to supply makeup for the systems listed in Table 4.6-1

The interface requirements identified for this system shall be satisfied by providing site-specific systems, structures and components that are technologically similar to the systems, structures and components of the certified design. Consequently, verification of compliance with the interface requirements shall be achieved by inspections, tests and analyses that are similar to those provided for the certified design. These inspections, tests and analyses, together with their associated acceptance criteria shall be developed by the combined license applicant referencing the certified design.

Table 4.6-1

Makeup Water System Supplied Equipment

The Makeup Water System provides water for the following:	
1	CST makeup
2	Standby Liquid Control system makeup
3	Liquid Waste System chemical addition and line flushing
4	Solid Waste System for line flushing
5	Reactor Component Cooling Water System makeup
6	Turbine Component Cooling Water System makeup
7	Chilled Water System makeup
8	Process Sampling System process use
9	Auxiliary Boiler System makeup
10	Post Accident Sampling Station flushing
11	HVAC makeup
12	Miscellaneous uses
13	IC/PCC Pool normal makeup water

4.7 COMMUNICATION SYSTEM

Design Description

Subsection 2.13.7 addresses those portions of the Communication System, which are within the scope of the certified design. All other communication system elements (i.e., off-site security radio system, crisis management radio system, and fire brigade radio system) are not within the design certification scope and will not be provided as part of the site-specific design.

Interface Requirements

No specific technical interface requirements have been identified for those portions of the plant communication system, which are not within the scope of the certified design.

Any interface requirements, which may be identified for this system, shall be satisfied by providing site-specific systems, structures and components that are technologically similar to the systems, structures and components of the certified design. Consequently, verification of compliance with the interface requirements shall be achieved by inspections, tests and analyses that are similar to those provided for the certified design. These inspections, tests and analyses, together with their associated acceptance criteria shall be developed by the combined license applicant referencing the certified design.

5. SITE PARAMETERS

5.1 SCOPE AND PURPOSE

The intent of this section is to provide Tier 1 material that complies with the 10 CFR 52 requirements to define the site parameters postulated for the ESBWR certified design.

Assuming the certified design will be referenced for a wide range of sites, it is necessary to specify a set of site parameters enveloping the conditions that could occur at most potential power plant sites in the United States. These parameters are provided in Table 5.1-1. It is intended that any facility that references the certified design will utilize a site where the actual site-specific conditions are within the defined envelope.

In the case of seismic design parameters, deviations from the defined conditions may be justified by site-specific soil-structure interaction analyses. The results may be used to confirm the seismic design adequacy of the certified design using approved methods and acceptance criteria.

Table 5.1-1

Site Parameters

Parameter	Value
Exclusion Area Boundary (EAB):	An area whose boundary has a Chi/Q less than or equal to 1.0×10^{-3} sec/m^3
Extreme Wind: Basic Wind Speed:	 49.2 m/s (1) / 62.6 m/s (2)
Design Ambient Temperatures: 1% Exceedance Values Maximum: Minimum: 0% Exceedance Values (Historical Limit) Maximum: Minimum:	 37.8°C dry bulb/26.1°C wet bulb (coincident), 27.8°C wet bulb (non-coincident) –23.3°C 46.1°C dry bulb/26.7°C wet bulb (coincident), 29.4°C wet bulb (non-coincident) –40.0°C
Precipitation (for Roof Design): Maximum rainfall rate: Maximum snow load:	 49.3 cm/hr (3) 2.39 kPa
Tornado: Maximum tornado wind speed: Translational velocity Radius: Maximum pressure drop: Rate of pressure drop: Missile Spectra:	 147.5 m/s 31.3 m/s 45.7 m 16.6 kPa 11.7 kPa/s Spectrum I of SRP 3.5.1.4
Maximum Ground Water Level:	0.61 m below grade
Maximum Flood (or Tsunami) Level:	0.30 m below grade or less
Seismology (4): SSE Response Spectra:	 See Figures 5.1-1 and 5.1-2
Soil Properties: Minimum Static Bearing Capacity: Minimum Shear Wave Velocity: Liquefaction Potential:	 718 kPa 300 m/s (5) None at the site-specific SSE level

Notes:

(1) Value to be utilized for design of nonsafety-related structures only.

(2) Value to be utilized for design of safety-related structures only.

(3) Probable maximum precipitation (PMP) based on the maximum value for one hour over 2.6 km^2 with ratio of 5 minutes to 1 hour PMP of 0.32. Maximum short-term rate: 15.7 cm in 5 min.

(4) The site-specific SSE ground response spectra of 5% damping at the foundation level in the free-field are enveloped by the larger of the 0.3g Regulatory Guide 1.60 generic site spectra and the North Anna Early Site Permit spectra, shown in Figures 5.1-1 and 5.1-2, in the horizontal and vertical directions, respectively.

(5) This is the minimum shear wave velocity at the foundation level.

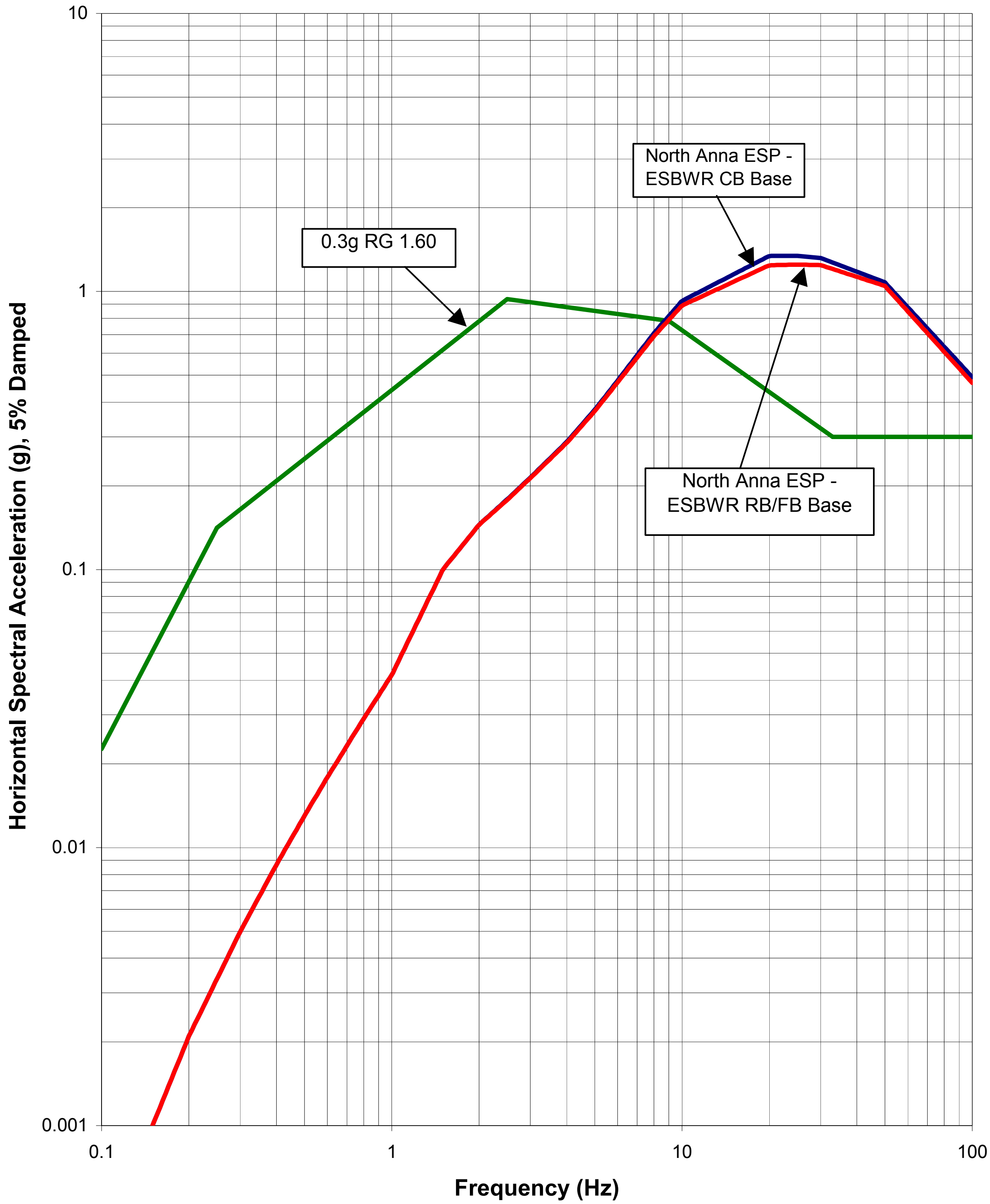


Figure 5.1-1. ESBWR Horizontal SSE Design Ground Spectra at Foundation Level

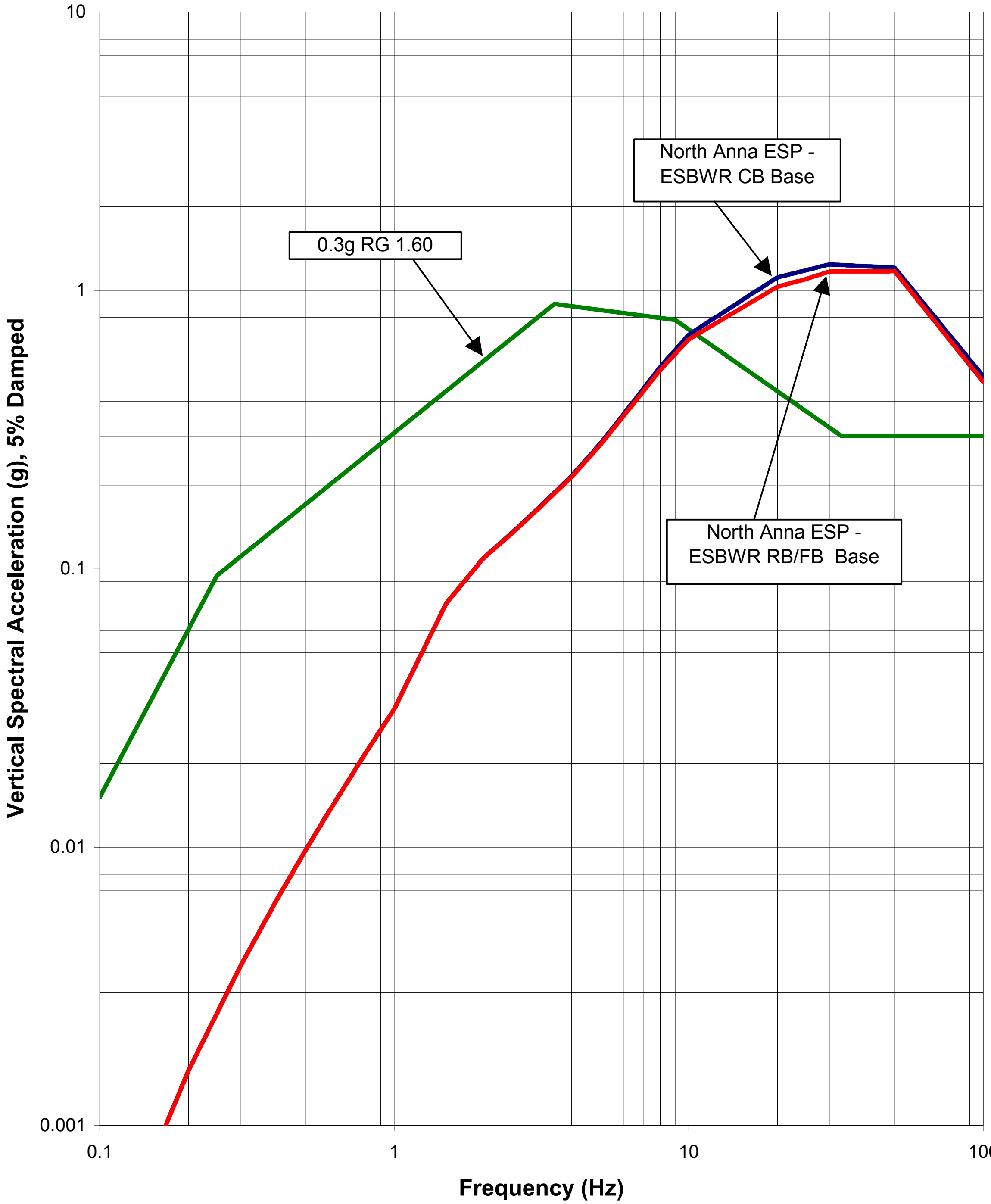


Figure 5.1-2. ESBWR Vertical SSE Design Ground Response Spectra at Foundation Level