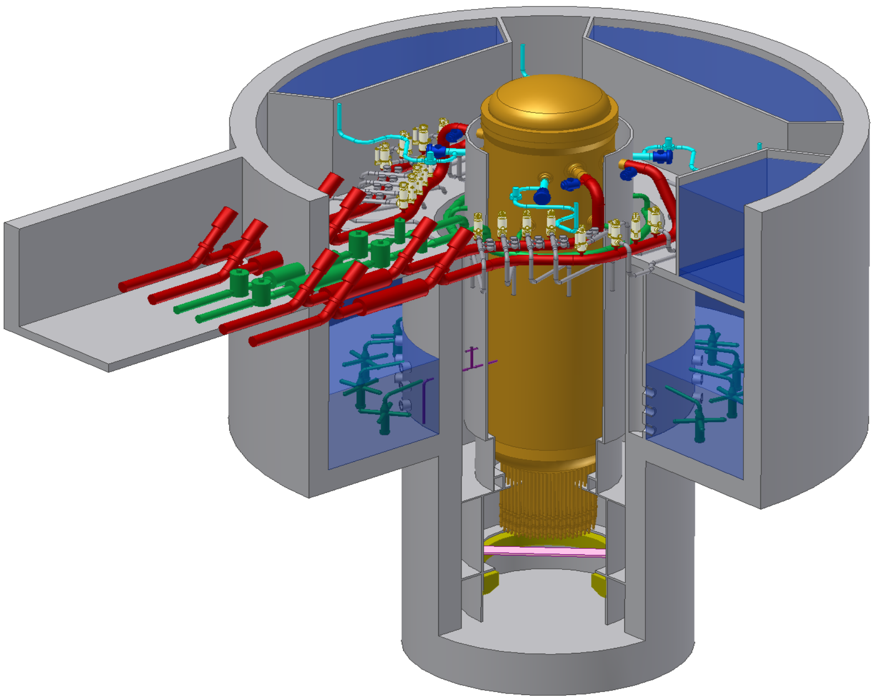
GE Energy Nuclear

26A6641AB
Revision 3
February 2007



ESBWR Design Control Document

Tier 1

Contents

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Abbreviations And Acronyms List

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
AB	Auxiliary Boiler
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
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
AOF	Allocation of Functions
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
ASA	Acoustical Society of America
ASCE	American Society of Civil Engineers
ASD	Adjustable Speed Drive

Abbreviations And Acronyms List

Term	Definition
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
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
CBGAVS	Control Building General Area HVAC Subsystem
CBVS	Control Building HVAC System
CCI	Core-Concrete Interaction
CDF	Core Damage Frequency
CDU	Condensing Unit
CEA	Consumer Electronics Association
CFR	Code of Federal Regulations
CIRC	Circulating Water System

Abbreviations And Acronyms List

Term	Definition
CIS	Containment Inerting System
CIV	Combined Intermediate Valve
CLAVS	Reactor Building Clean Area HVAC Subsystem
CM	Cold Machine Shop
CMS	Containment Monitoring System
COL	Combined Operating License
COLR	Core Operating Limits Report
CONAVS	Reactor Building Contaminated Area HVAC Subsystem
CPR	Critical Power Ratio
CPS	Condensate Purification System
CR	Control Rod
CRD	Control Rod Drive
CRDHS	Control Rod Drive Hydraulic System
CRDS	Control Rod Drive System
CRGT	Control Rod Guide Tube
CRHA	Control Room Habitability Area
CRHAVS	Control Room Habitability Area HVAC Subsystem
CS&TS	Condensate Storage and Transfer System
CSDM	Cold Shutdown Margin
CS / CST	Condensate Storage Tank
CSPP	Cyber Security Program Plan
CT	Main Cooling Tower
CTSS	Communications Continuous Tone-Controlled Squelch System
CTVCF	Constant Voltage Constant Frequency
CWS	Chilled Water System
D-RAP	Design Reliability Assurance Program
DBA	Design Basis Accident
DBE	Design Basis Event
dc / DC	Direct Current
DCD	Design Control Document
DCS	Drywell Cooling System
DCIS	Distributed Control and Information System
DF	Diaphragm Floor
DF	Decontamination Factor
DG	Diesel Generator
DGVS	Diesel Generators HVAC Subsystem
DPS	Diverse Protection System
DOI	Dedicated Operators Interface
DOT	Department of Transportation

Abbreviations And Acronyms List

Term	**Definition**
DPS	Diverse Protection System
DPV	Depressurization Valve
DTM	Digital Trip Module
DW	Drywell
EAB	Exclusion Area Boundary
EB	Electrical Building
EBVS	Electrical Building HVAC System
ECCS	Emergency Core Cooling System
EERVS	Electric and Electronic Rooms HVAC Subsystem
EFDS	Equipment and Floor Drainage System
EFU	Emergency Filter Unit
ENS	Emergency Notification System
EOF	Emergency Operations Facility
EPDS	Electric Power Distribution System
EPG	Emergency Procedure Guideline
ERIP	Emergency Rod Insertion Panel
ESBWR	Economically Simplified Boiling Water Reactor
ESF	Engineered Safety Feature
ETS	Emergency Trip System
FAPCS	Fuel and Auxiliary Pools Cooling System
FATT	Fracture Appearance Transition Temperature
FB	Fuel Building
FBGAVS	Fuel Building General Area HVAC Subsystem
FBVS	Fuel Building HVAC System
FFT	Fast Fourier Transform
FIV	Flow-Induced Vibration
FM	Factory Mutual
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analysis
FPS	Fire Protection System
FRA	Functional Requirements Analysis
FO	Diesel Fuel Oil Storage Tank
FOAKE	First-of-a-Kind Engineering
FPE	Fire Pump Enclosure
FSI	Fluid Structure Interaction
FTDC	Fault-Tolerant Digital Controller
FW	Feedwater
FWCS	Feedwater Control System
FWST	Fire Water Storage Tank

Abbreviations And Acronyms List

Term	Definition
GCS	Generator Cooling System
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GETAB	General Electric Thermal Analysis Basis
GL	Generic Letter
HCU	Hydraulic Control Unit
HCW	High Conductivity Waste
HDVS	Heater Drain and Vent System
HFE	Human Factors Engineering
HFF	Hollow Fiber Filter
HI	Hydraulic Institute
HIS	Hydraulic Institute Standards
HM	Hot Machine Shop & Storage
HP	High Pressure
HPNSS	High Pressure Nitrogen Supply System
HRA	Human Reliability Assessment
HSI	Human System Interface
HVAC	Heating, Ventilating and Air Conditioning
HVT	Horizontal Vent Test
I&C	Instrumentation and Control
IC/PCC	Isolation Condenser/Passive Containment Cooling
I/O	Input/Output
IAS	Instrument Air System
IC	Ion Chamber
IC	Isolation Condenser
ICS	Isolation Condenser System
IE	Inspection and Enforcement
IEEE	Institute of Electrical and Electronic Engineers
IESNA	Illuminating Engineering Society of North America
IFTS	Inclined Fuel Transfer System
ILRT	Integrated Leak Rate Test
IMCC	Induction Motor Control Cabinets
IPC	Isolation Power Center
ISI	In-Service Inspection
ISMA	Independent Support Motion Response Spectrum Analysis
ITA	Inspections, Tests or Analyses
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
ITP	Initial Test Program

Abbreviations And Acronyms List

Term	Definition
LCW	Low Conductivity Waste
LD	Load Driver
LD&IS	Leak Detection and Isolation System
LRF	Large Release Frequency
LFCV	Low Flow Control Valve
LLRT	Local Leak Rate Test
LOCA	Loss-of-Coolant-Accident
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
LWMS	Liquid Waste Management System
MCC	Motor Control Center
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MCRP	Main Control Room Panel
MFAP	Main Fire Alarm Panel
MLHGR	Maximum Linear Heat Generation Rate
MMIS	Man-Machine Interface Systems
MPC	Maximum Permissible Concentration
MRBM	Multi-Channel Rod Block Monitor
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSR	Moisture Separator Reheater
MWS	Makeup Water System
NBS	Nuclear Boiler System
NDE	Nondestructive Examination
N-DCIS	Non-Safety Related Distributed Control and Information System
NDRC	National Defense Research Committee
NFPA	National Fire Protection Association
NMS	Neutron Monitoring System
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

Abbreviations And Acronyms List

Term	Definition
NT	Nitrogen Storage Tank
O&M	Operation and Maintenance
OER	Operating Experience Review
OGS	Offgas System
OPRM	Oscillation Power Range Monitor
OSC	Operational Support Center
PAS	Plant Automation System
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PG	Power Generation
PGCS	Power Generation and Control Subsystem of Plant Automation System
PH	Pump House
PIP	Plant Investment Protection
PM	Preventive Maintenance
PMCS	Performance Monitoring and Control Subsystem of N-DCIS
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitoring
PSWS	Plant Service Water System
QA	Quality Assurance
RAPI	Rod Action and Position Information
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBCC	Rod Brake Controller Cabinet
RBCWS	Reactor Building Chilled Water Subsystem
RBVS	Reactor Building HVAC System
RC&IS	Rod Control and Information System
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
REPAVS	Reactor Building Refueling and Pool Area HVAC Subsystem
RG	Regulatory Guide
RHX	Regenerative Heat Exchanger
RMS	Radiation Monitoring System

Abbreviations And Acronyms List

Term	Definition
RMS	Root Mean Square
RMU	Remote Multiplexer Unit
RO	Reverse Osmosis
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRPS	Reference Rod Pull Sequence
RSM	Rod Server Module
RSS	Remote Shutdown System
RTIF	Reactor Trip and Isolation Function(s)
RW	Radwaste Building
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling System
RWGA	Radwaste Building General Area
RWVS	Radwaste Building HVAC System
RWM	Rod Worth Minimizer
S&Q	Staffing and Qualifications
SAIV	Steam Auxiliary Isolation Valve
SAS	Service Air System
SB	Service Building
SB&PC	Steam Bypass and Pressure Control System
SBO	Station Blackout
SBWR	Simplified Boiling Water Reactor
SCMP	Software Configuration Management Plan
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling
SDP	Software Development Plan
SF	Service Water Building
SFP	Spent fuel pool
SIntP	Software Integration Plan
SIP	Software Installation Plan
SIT	Structural Integrity Test
SIU	Signal Interface Unit
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SMP	Software Management Plan
SOMP	Software Operations and Maintenance Plan
SP	Setpoint
S/P	Suppression Pool
SPC	Suppression Pool Cooling

Abbreviations And Acronyms List

Term	Definition
SPTM	Suppression Pool Temperature Monitoring
SPTMS	Suppression Pool Temperature Monitoring Subsystem of Containment Monitoring System
SQAP	Software Quality Assurance Plan
SRI	Select Rod Insert
SRM	Source Range Monitor
SRNM	Short Range Neutron Monitor
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSAR	Standard Safety Analysis Report
SSC	Structure, System or Component
SSE	Safe Shutdown Earthquake
SSLC	Safety System Login and Control
SSPC	Steel Structures Painting Council
SSP	Software Safety Plan
SST	Sub-scale Test
ST	Spare Transformer
STI	Startup Test Instruction
STrngP	Software Training Plan
SVVP	Software Verification and Validation Plan
SWMS	Solid Waste Management System
TAF	Top of Active Fuel
TB	Turbine Building
TBS	Turbine Bypass System
TBVS	Turbine Building HVAC System
TC	Training Center
TCCWS	Turbine Component Cooling Water System
TCV	Turbine Control Valve
TG	Turbine Generator
TGSS	Turbine Gland Seal System
THA	Time-History Accelerograph
TLU	Trip Logic Unit
TMSS	Turbine Main Steam System
TSC	Technical Support Center
TSV	Turbine Stop Valve
UAT	Unit Auxiliary Transformer
UHS	Ultimate Heat Sink
UL	Underwriter's Laboratories Inc.
UPS	Uninterruptible Power Supply
USNRC	United States Nuclear Regulatory Commission

Abbreviations And Acronyms List

Term	Definition
V&V	Verification and Validation
Vac / VAC	Volts Alternating Current
Vdc / VDC	Volts Direct Current
VDU	Video Display Unit
WD	Wash Down Bays
WT	Water Treatment
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:

- Definitions and general provisions;
- Design descriptions;
- Inspections, tests, analyses, and acceptance criteria (ITAAC);
- Significant site parameters; and
- 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 by a change process defined in the design certification rule (typically Section VIII) that become an appendix to 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. For completeness, the major structures and systems that are not important to safety or are site-specific 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. Seven such main buildings (see Figure 1.1-1) are within the scope for the ESBWR. These are:

(1) Reactor Building – houses the safety-related mechanical systems, safety-related electrical systems, and the sensors and remote actuators for the Safety Related Distributed Control and Information System. The Reactor Building includes the reactor, containment, refueling area and auxiliary equipment, Control Rod Drives, Reactor Water Cleanup and Shutdown Cooling system, and the Nonsafety-Related Distributed Control and Information System logic cabinets.

(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 two nonsafety-related standby diesel generators and their associated auxiliary equipment.

(7) Service Building – houses the equipment and control facilities associated with personnel entry into the reactor building and turbine building, eating areas, radiation protection, changing rooms, shops and offices.

Buildings and structures not within the ESBWR Standard Plant scope include the 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 Level

The plant uses a single-cycle, natural circulation BWR with an initial rated thermal power of 4500 MW thermal.

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.

An **alarm** can be an auditory notification, visual indication and/or off-normal condition data recorder.

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.

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 to 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 Descriptions.

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

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 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 is the designation applied to a given safety-related system or set of components that enables the establishment and maintenance of physical, electrical and functional independence from other redundant sets of components.

High Regulatory Oversight systems are nonsafety-related systems that provide a significant contribution to meeting the core damage frequency and containment performance goals for advanced light water reactors.

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 and infrequent events evaluated in the Final Safety Analysis Report (FSAR) safety analyses;

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

(4) Equipment/function(s) whose failure or malfunction could 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 of installed plant equipment and their associated plant condition(s) assumed, prior to the initiation of an accident, in any accident safety analysis described in the FSAR;

(7) As described in the FSAR, nonsafety-related SSCs specifically included in the plant to control the release of radioactive wastes within 10 CFR 20 limits; and

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

An **indicated** item or a **visual indication** is any information that is visually available to an operator.

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.

A **manual control** is any operation or function that can be physically initiated, terminated or modulated by an operator.

Safe shutdown (generic definition) 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 for station blackout means bringing the plant to those shutdown conditions specified in plant Technical Specifications as Hot Standby, Hot Shutdown or Stable Shutdown.

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 Single Failure Criterion in 10 CFR 50, Appendix A.

* *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. When referring to an instrumentation and control system, train is defined as the redundant, identical sets of 2/4 trip decisions and subsequent logic (i.e., timers, permissives and interlocks) within an electrical division that actuate the series load drivers of a safety-related component. Each train utilizes the individual trip decisions from the sensor channels of each of the four divisions.

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.

Verification of the basic configuration of a system, as used in an ITAAC, means verifying the functional arrangement, welding requirements, environmental qualification and seismic qualification.

Verification of the functional arrangement of a system, as used in an ITAAC, means verifying that the system is constructed as depicted in the Tier 1 design drawings, including equipment and instrument locations.

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. For all systems with containment penetrations, ASME Section III Division 1 NE and CC requirements per design commitments shown on Table 2.15-1 are used as applicable for NDE of pressure boundary welds.

(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 safety-related 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 "safety-related 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 safety-related electrical components identified in the Design Description. Equipment located in a mild environment during or following a DBA need not be tested or analyzed.

    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 valves identified in the Design Description to demonstrate that the valves 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 require as-installed equipment). Additionally, ITA may be performed as part of the activities that are required to be performed under 10 CFR 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 with housing).

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 maintain coolant above the top of the active fuel during normal operations and 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 2.1.1-1, 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 core coolant flow. An increased flow-path length relative to most prior BWRs is provided by the chimney 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. 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, 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.

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 determined and controlled in accordance with the requirements of the ASME Code, Section III, Division 1. The Charpy upper-shelf energy in the transverse direction of the core beltline material and along the adjacent pressure retaining welds is determined according to the ASME Code. The initial minimum Charpy upper-shelf energy for the core beltline base material and the adjacent welds meets or exceeds 102 Joules. 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 the forging material of the beltline region and the weld and heat affected zone of a weld typical of those adjacent to the beltline region. The base material and weld are heat treated in a manner, which simulates the actual heat treatment performed of the beltline region of the completed vessel. The specimens together with temperature monitors and neutron flux monitors are encapsulated into the surveillance specimen holders. Brackets welded to the vessel cladding at the location of the calculated peak fluence are provided to hold the removable specimen holders and a neutron dosimeter in place.

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 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.

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 restrains 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.

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 (partial height) 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 ≤ 0.1 wt%

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 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]

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 Table 2.1.1-1 and Figure 2.1.1-1.	1. Inspections of the as-built RPV System will be conducted.	1. The inspection report(s) confirm(s) the RPV system conforms to the Certified Design Specification and the basic configuration defined in 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. Inspection report(s) confirm(s) that 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. Inspection report(s) confirm(s) that 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 inspection report(s) confirm(s) the material surveillance program for the reactor pressure vessel core beltline materials conforms to the commitments defined in Subsection 2.1.1.

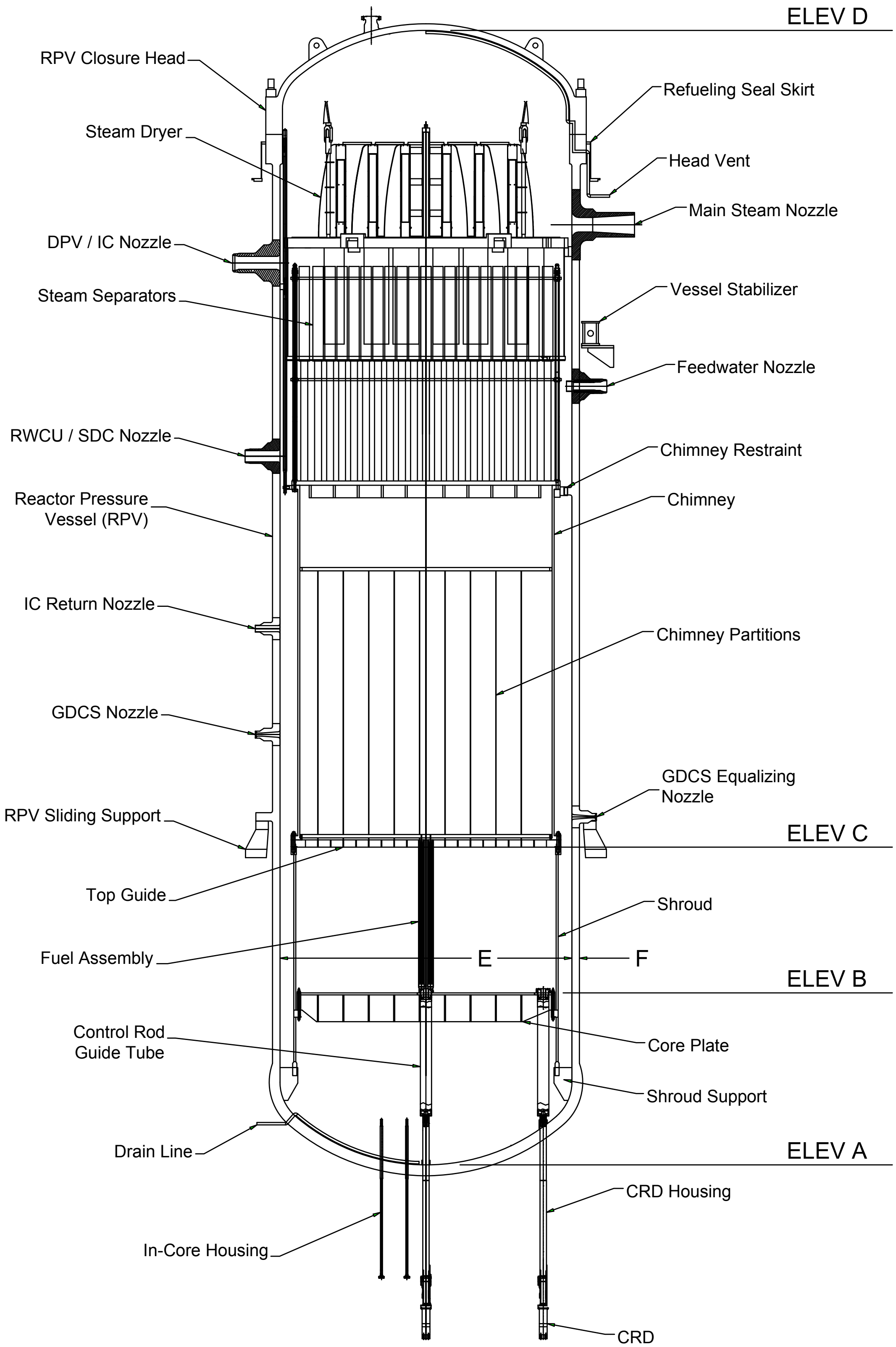
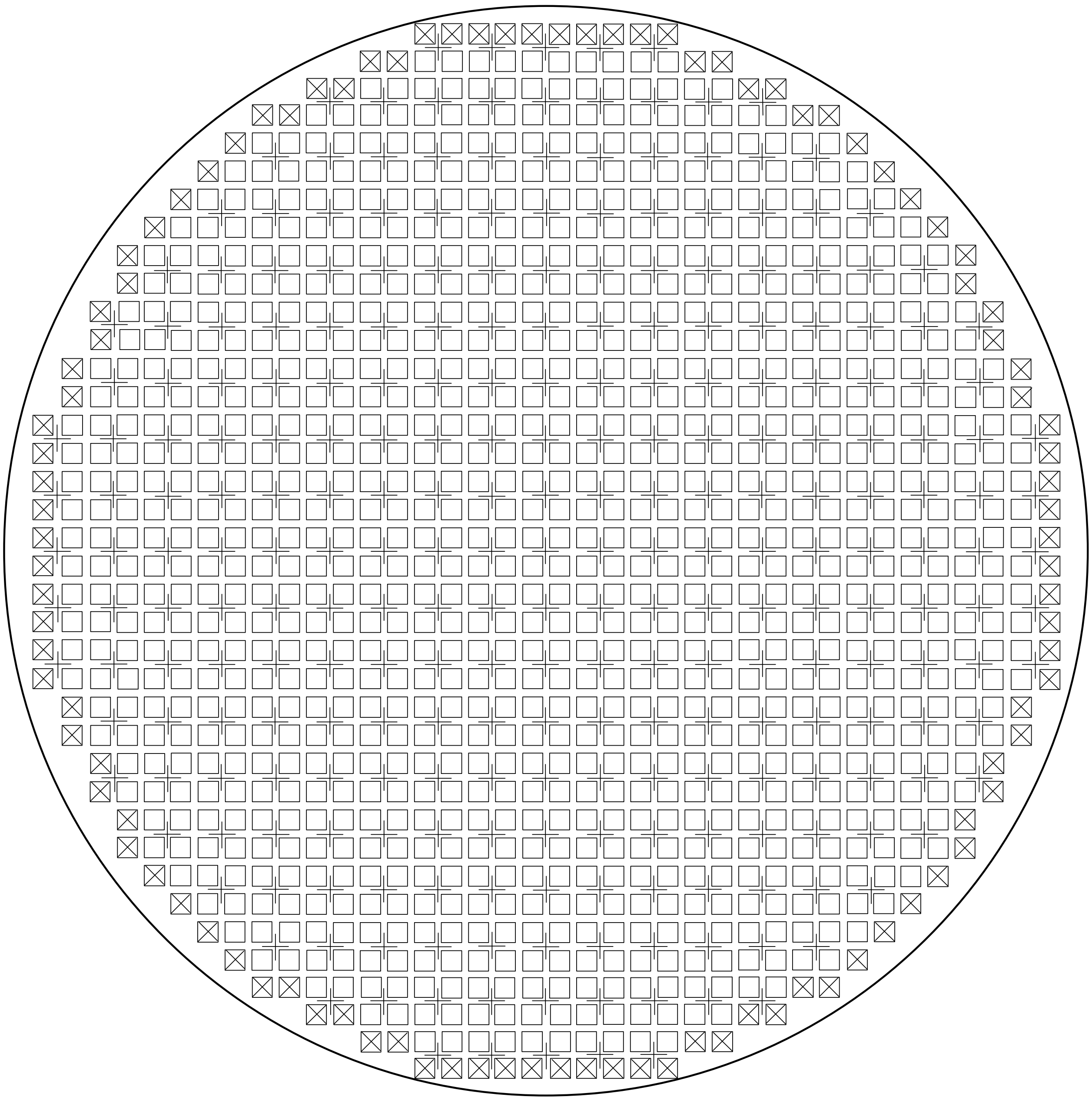


Figure 2.1.1-1. Reactor Pressure Vessel System Key Features



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 MSL minimum constructed combined volume for all four MSLs is 135 m^3 (4,767 ft^3).

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 are designated as ADS SRVs, which operate in the overpressure safety mode and the Automatic Depressurization mode and 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 are designated as Non-ADS SRVs, which operate in the overpressure safety mode. Each Non-ADS SRV discharges through its individual discharge stack that has a rupture disc at the end. Each discharge stack has a drain line that drains condensed steam leakage to the suppression pool. The Non-ADS SRVs discharge through the rupture discs into the drywell. 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. 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 seconds.

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. When actuated by an initiator, 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. Four DPVs are attached to stub tubes off of the RPV and four DPVs are attached to the main steam lines.

The NBS instrumentation consists of sensors to measure and monitor RPV pressure, temperature and water level. Additionally, there are sensors to measure and monitor turbine inlet pressure, steam line flow, main condenser vacuum and RPV metal temperatures. Low RPV water level, low turbine inlet pressure (RUN mode), low main condenser vacuum (RUN mode) and high steam line flow provide signals to close the MSIVs. The RPV water level instrumentation considers the effects of dissolved non-condensable gases in the RPV water instrument lines.

The mechanical portion of each division of the safety-related NBS instrumentation located in the Reactor Building outside the drywell is physically separated from the other divisions.

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 open 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 outboard isolation check valve and the seismic interface restraint shall be ASME Section III Class 2 and Quality Group B. Piping from the RPV to the seismic restraint upstream of the isolation shutoff 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 (14 in.) to meet the choke flow requirements. MSIVs close by venting an under piston volume and pressurizing the over piston volume. An accumulator assists MSIV closure when the make-up pneumatic supply is not available. The MSIV closes in less than or equal to 5 seconds and greater than or equal to 3 seconds when nitrogen or air is admitted to the valve actuator. When all MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters (17.4 gallons) per minute at standard temperature of 20°C (68°F) and pressure with the differential pressure across the MSIV equal to or greater than 0.269 MPaD (39 psid).

The ten ADS-SRVs and DPVs together with instrumentation and control system logic constitute the Automatic Depressurization System (ADS) of the NBS. The ADS logic is automatically initiated when a low reactor water level signal is present. When a low water level signal is present, the Confirm ECCS-LOCA Signal timer initiates and continues to time out in the continued presence of the RPV low water level signal. The time delay is less than or equal to 10 seconds. Upon time out of the Confirm ECCS-LOCA Signal timer, an actuation signal is generated to the Group 1 ADS SRVs and the Group 2 ADS timer initiates and continues to time out. Upon time out, an actuation signal is generated to the Group 2 ADS SRVs. The time delay is less than or equal to 10 seconds. Upon time out of the Confirm ECCS-LOCA Signal timer, the Group 1 DPV timer initiates and continues to time out. Upon time out, an actuation signal is generated to the Group 1 DPVs. The time delay is less than or equal to 50 seconds. Upon time out of the Confirm ECCS-LOCA Signal timer, the Group 2 DPV timer initiates and continues to time out. Upon time out, an actuation signal is generated to the Group 2 DPVs. The time delay is less than or equal to 100 seconds. Upon time out of the Confirm ECCS-LOCA Signal timer, the Group 3 DPV timer initiates and continues to time out. Upon time out, an actuation signal is generated to the Group 3 DPVs. The time delay is less than or equal to 150 seconds. Upon time out of the Confirm ECCS-LOCA Signal timer, the Group 4 DPV timer initiates and continues to time out. Upon time out, an actuation signal is generated to the Group 4 DPVs. The time delay is less than or equal to 200 seconds. Upon time out of the Confirm ECCS-LOCA Signal timer, the GDCS Injection Squib Valve timer initiates and continues to time out. The time delay is less than or equal to 150 seconds. Upon time out of the Confirm ECCS-LOCA Signal timer, the GDCS Equalization Line Squib Valve timer initiates and continues to time out. The time delay is less than or equal to 30 minutes. Upon manual actuation of the GDCS Equalization Line Squib Valve initiation logic, concurrent with an RPV low-pressure signal, the GDCS Manual

Equalization Line Squib Valve timer initiates and continues to time out. The time delay is less than or equal to 30 minutes. An accumulator opens the ADS-SRV when the pneumatic supply is not available.

The safety function of the eighteen 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.

For ATWS mitigation, ADS is inhibited automatically, based on the following signals.

- A coincident low RPV water level signal and Average Power Range Monitor (APRM) ATWS permissive signal (i.e., APRM signal above a specified setpoint) from the Neutron Monitoring System (NMS); and
- A coincident high RPV pressure and APRM ATWS permissive signal persisting for 60 seconds.
- There are controls in the MCR for the manual inhibit of the ADS under ATWS conditions.

While in the Run mode, the ADS shall automatically initiate upon detection of low water level. 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.

Safety-related components in NBS shall be powered from their respective safety-related division. In NBS, independence is provided between safety-related divisions, and between safety-related and nonsafety-related equipment.

The feedwater isolation valves shown on Figure 2.1.2-2 have an active safety-related function to close under the most severe reverse differential pressure, fluid flow, and temperature conditions they may experience.

The feedwater isolation check valves shown on Figure 2.1.2-2 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. RPV level sensors are shown on Figure 2.1.2-4.

Reactor pressure is sensed by 12 safety-related transmitters (3 in each of 4 Divisions) and 8 nonsafety-related transmitters (2 in each of 4 Groups).

RPV metal temperature is sensed by 2 nonsafety-related RPV Head Flange sensors (1 in each of 2 Groups), 2 nonsafety-related RPV Shell Flange sensors (1 in each of 2 Groups) and 2 nonsafety-related RPV Bottom Head sensors (1 in each of 2 Groups).

Drywell pressure is sensed by 4 nonsafety-related wide range transmitters (1 in each of 4 Groups), 2 safety-related wide range transmitters (1 in each of 2 Divisions), 4 safety-related narrow range transmitters (1 in each of 4 Divisions) and 2 safety-related differential pressure transmitters (drywell/wetwell) (1 in each of 2 Divisions).

Main condenser vacuum is sensed by 4 safety-related transmitters (1 in each of 4 Divisions).

Turbine inlet pressure is sensed by 4 safety-related transmitters (1 in each of 4 Divisions).

With the exception of turbine inlet pressure and main condenser vacuum sensors located in the Turbine Building, the NBS instrumentation is located in the drywell, steam tunnel and the Reactor Building.

Capacities of the SRVs and DPVs are addressed in Table 2.1.2-1.

There are displays and controls in the Remote Shutdown System for the SRVs and RPV level and pressure.

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 for the NBS.

Table 2.1.2-1

SRV Capacities

Valves	Number of Valves	ASME Rated Capacity at Analytical Lift Pressure (1)	Used For ADS
Non-ADS SRV	8	Valve size, discharge capacity and lift setpoint(s) are selected based upon analysis of the ATWS event response requirements, the AOO lift avoidance commitment, and the containment design parameters	0
ADS-SRV	10	Valve size, discharge capacity and lift setpoint(s) are selected based upon analysis of the ATWS, MSIV Full Closure and ECCS events response requirements, the AOO lift avoidance commitment, and the containment design parameters	10

DPV Capacities

Valves	Number of Valves	Rated Capacity (2)	Used For ADS
DPV	8	Valve size and discharge capacity are selected based upon analysis requirements for the ECCS events responses and the containment design parameters	8

(1) Minimum capacity is established by overpressure analysis performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III.

(2) Minimum capacity is established only in ADS mode. The DPVs are not needed to mitigate the vessel overpressure event.

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 to 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 (14 in.).	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 (14 in.).
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 of the as-built installation of the MSL flow instrumentation will be conducted to verify that it meets the design.	4. The as-built MSL flow measurement instrument lines match the instrument line design drawings and tolerances.
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 to 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 (4768 ft^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 (4768 ft^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 indicated in the main control room.
8. 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. Type testing of an MSIV will be conducted in accordance with the design and purchase specifications.	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 (17.4 gallons) 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 (39 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 (17.4 gallons) 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 (39 psid).

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. 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 seconds.	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 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. a. The SRVs are provided with instrumentation that will provide indication (i.e. by direct measurement) of valve position. b. The DPVs are provided with valve open indication instrumentation.	11. a. Inspections will be performed on the SRV position indication instrumentation. b. Inspections or tests will be performed on the DPV position indication instrumentation.	11. a. Inspection report(s) confirm that the SRV position indicators provide open and close indication. b. Inspection or test records denote that the DPV have valve open indication instrumentation.
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 a low water level signal at the input to the ADS initiation logic, the following occurs: a. The confirm ECCS-LOCA Signal timer initiates and continues to time out in the continued presence of the RPV low water level signal. The time delay is less than or equal to 10 seconds. b. Upon time out of the confirm ECCS-LOCA Signal timer, an actuation signal is generated to the Group 1 ADS SRVs and the Group 2 ADS timer initiates and continues to time

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		out. Upon time out, an actuation signal is generated to the Group 2 ADS SRVs. The time delay is less than or equal to 10 seconds. c. Upon time out of the confirm ECCS-LOCA Signal timer, the Group 1 DPV timer initiates and continues to time out. Upon time out, an actuation signal is generated to the Group 1 DPVs. The time delay is less than or equal to 50 seconds. d. Upon time out of the confirm ECCS-LOCA Signal timer, the Group 2 DPV timer initiates and continues to time out. Upon time out, an actuation signal is generated to the Group 2 DPVs. The time delay is less than or equal to 100 seconds. e. Upon time out of the confirm ECCS-LOCA Signal timer, the Group 3 DPV timer initiates and continues to time out. Upon time out, an actuation signal is generated to the Group 3 DPVs. The time delay is less than or equal to 150 seconds. f. Upon time out of the confirm ECCS-LOCA Signal timer, the Group 4 DPV timer initiates and continues to time out. Upon time out, an actuation signal is generated to the Group 4

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		DPVs. The time delay is less than or equal to 200 seconds. g. Upon time out of the confirm ECCS-LOCA Signal timer, the GDCS Injection Squib Valve timer initiates and continues to time out. The time delay is less than or equal to 150 seconds. h. Upon time out of the confirm ECCS-LOCA Signal timer, the GDCS Equalization Line Squib Valve timer initiates and continues to time out. The time delay is less than or equal to 30 minutes. i. Upon manual actuation of the GDCS Equalization Line Squib Valve initiation logic, concurrent with an RPV low-pressure signal, the GDCS Manual Equalization Line Squib Valve timer initiates and continues to time out. The time delay is less than or equal to 30 minutes.
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 an initiator, 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. Type testing 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 as addressed in Table 2.1.2-1.	16. Analyses and type 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 the conditions addressed in Table 2.1.2-1.
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.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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 (RUN mode).	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 (RUN mode).
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.
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.
21. The ADS can be initiated manually.	21. Tests will be conducted by initiating each ADS division manually.	21. Upon receipt of a manual initiation signal, an ADS actuation signal is generated to the associated ADS valve solenoids.
22. The RPV water level instrumentation considers the effects of dissolved non-condensable gasses in the RPV water instrument lines.	22. Analyses of the as-built RPV water level instrumentation will be performed using available test data and/or operating experience.	22. An analysis output exists which concludes that the RPV water level instrumentation considers the effects of dissolved non-condensable gasses in the RPV water level instrument lines.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
23. The mechanical portion of each division of the safety-related NBS instrumentation located in the Reactor Building outside the drywell is physically separated from the other divisions.	23. Inspections of the as-built NBS instrumentation will be conducted.	23. The mechanical portion of each NBS instrumentation division is physically separated from the other divisions by structural and/or fire barriers.
24. The feedwater isolation valves shown on Figure 2.1.2-2 have an active safety function to close under reverse flow differential pressure, fluid flow and temperature conditions.	24. Type testing of similar valves along with analysis and engineering evaluation will be performed. Valves will be tested as part of system checkout to demonstrate functional readiness. Type testing of critical or new valves in a dedicated test facility will be performed where the function requires pressure, flow, temperature or environmental parameters that cannot be adequately simulated under normal system operating modes.	24. Test report(s) shall validate feedwater isolation valves have an active safety function to close under reverse flow differential pressure, fluid flow and temperature conditions.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
25. Safety-related check valves shown on Figure 2.1.2-2 perform a containment isolation safety-related function closing to maintain containment integrity.	25. Type testing of similar valves along with analysis and engineering evaluation will be performed. Valves will be tested as part of system checkout to demonstrate functional readiness. Type testing of critical or new valves in a dedicated test facility will be performed where the function requires pressure, flow, temperature or environmental parameters that cannot be adequately simulated under normal system operating modes.	25. Test report confirm, based on the direction of the differential pressure across the valve, each control valve opens, closes or both opens and closes depending upon the valve's safety function.
26. The ADS accumulator can open the SRV with the drywell pressure at design pressure following failure of the pneumatic supply to the accumulator.	26. An analysis and/or type test will be performed to demonstrate the capacity of the SRV ADS accumulators.	26. The SRV ADS accumulators have the capacity to lift the stem of the SRVs to the full open position one time with the drywell pressure at the drywell design pressure.
27. Remote Shutdown System displays and controls are provided for the NBS SRVs and RPV level and pressure.	27. Inspections will be performed on the RSS displays and controls for the NBS.	27. Displays and controls exist on the RSS for the NBS SRVs and RPV level and pressure.
28. The MSIV accumulator assists in closing the MSIV with the drywell pressure at design pressure following failure of the pneumatic supply to the accumulator.	28. An analysis and/or type test will be performed to demonstrate the capacity of the MSIV accumulator.	28. The MSIV accumulator has the capacity to assist in closing the MSIV with the drywell pressure at design pressure.

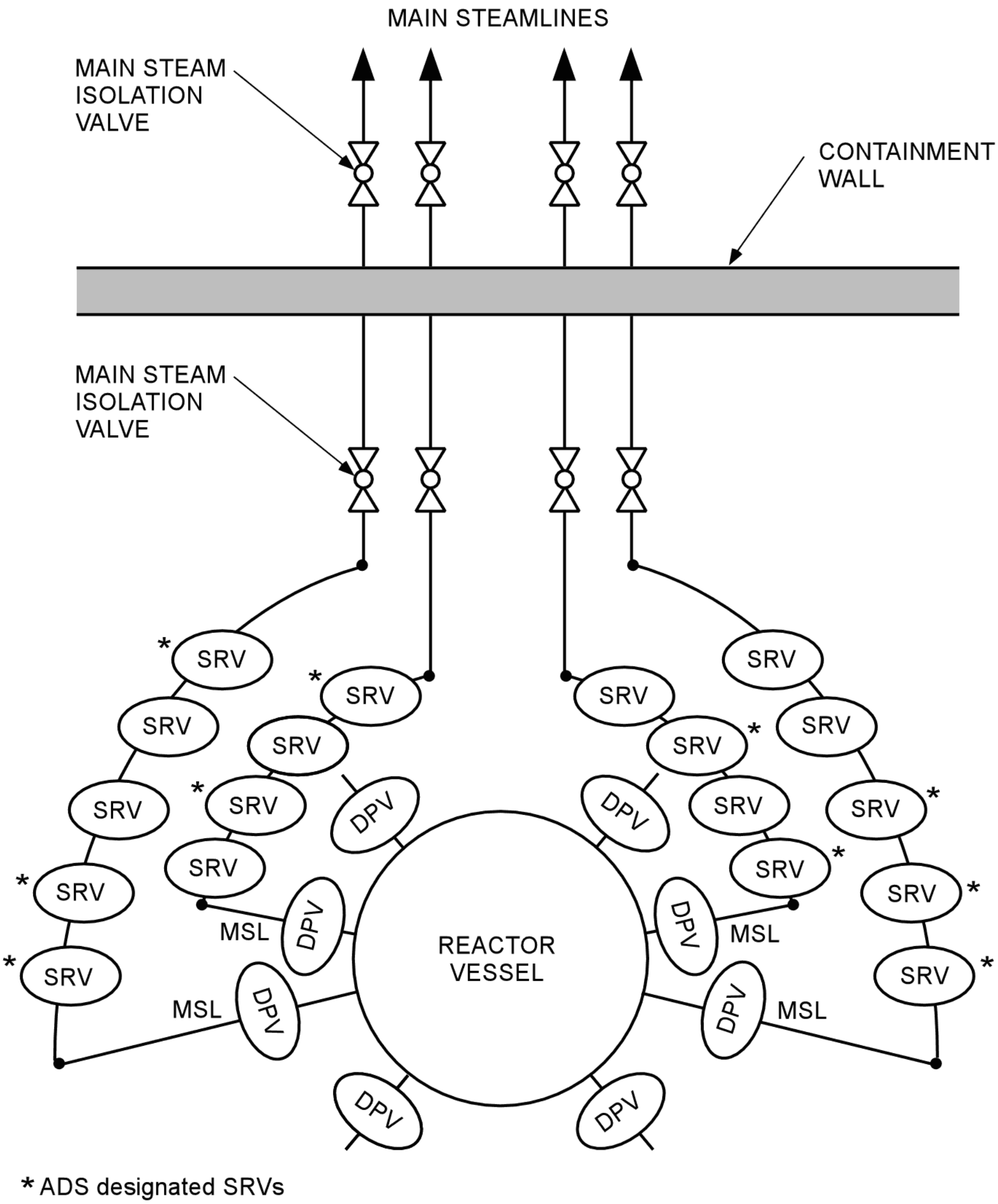


Figure 2.1.2-1. Safety Relief Valves, Depressurization Valves and Steamline Diagram

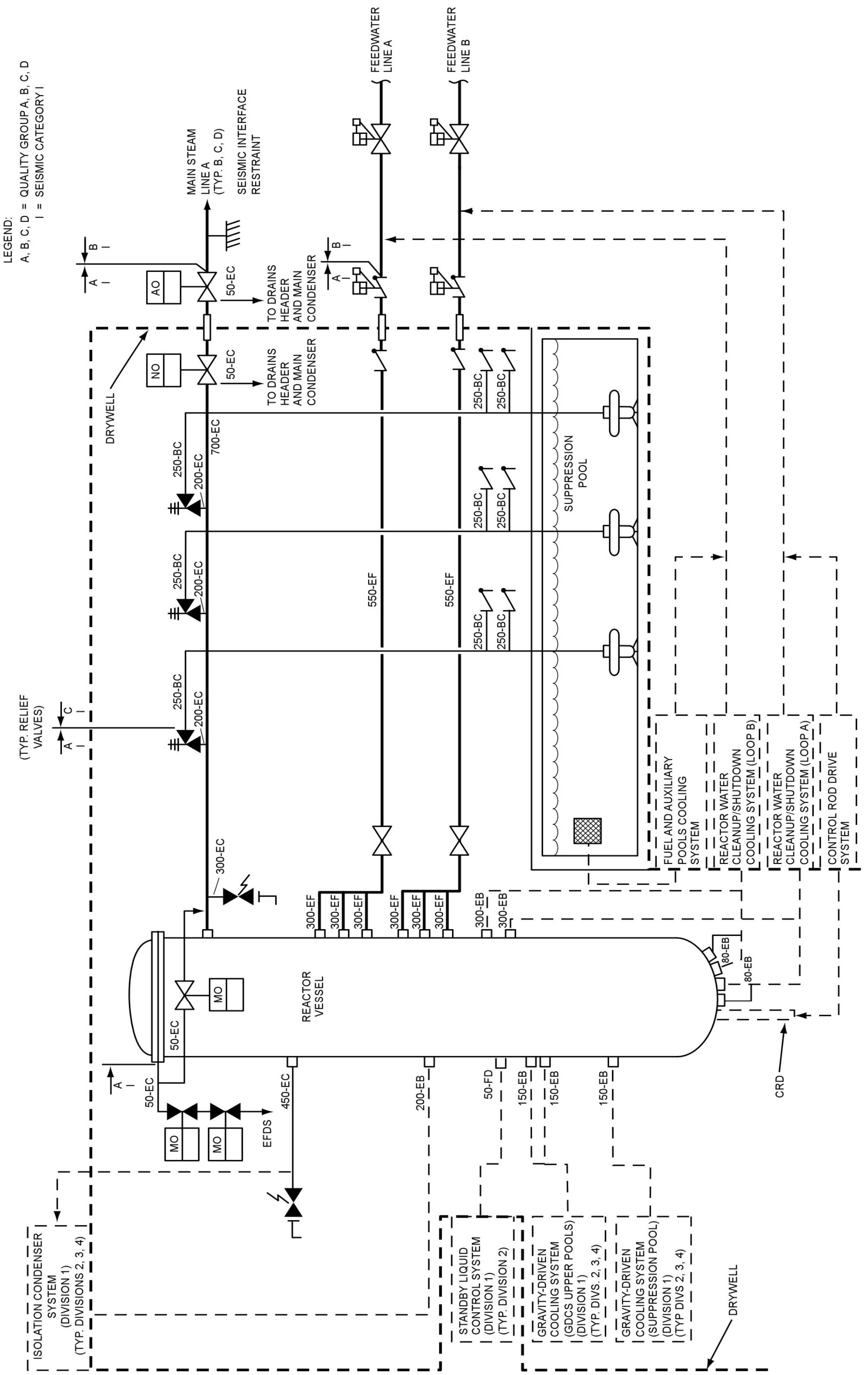


Figure 2.1.2-2. NBS Steamlines and Feedwater Lines

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

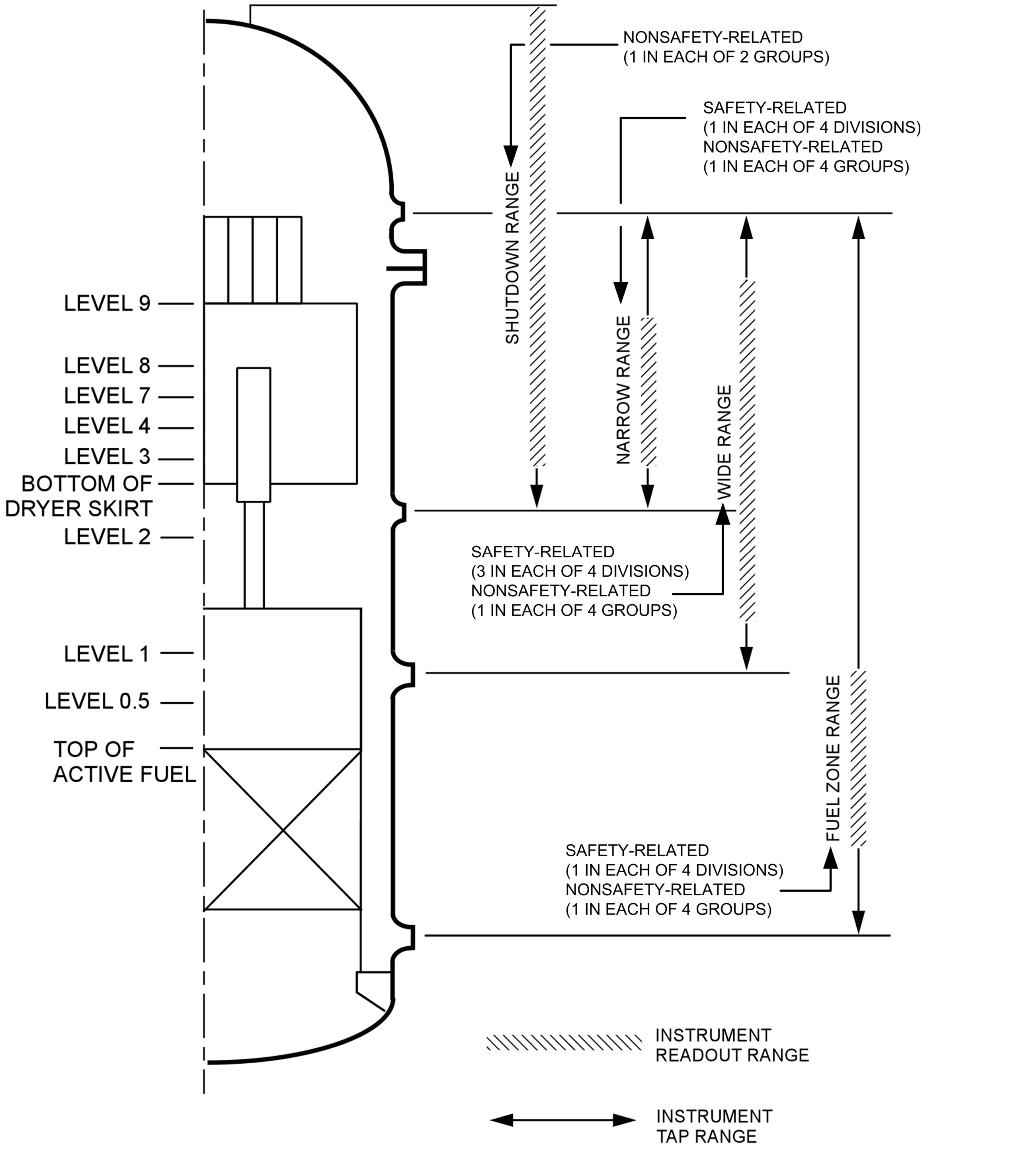


Figure 2.1.2-4. NBS Water Level Instrumentation

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 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. 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/or 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.

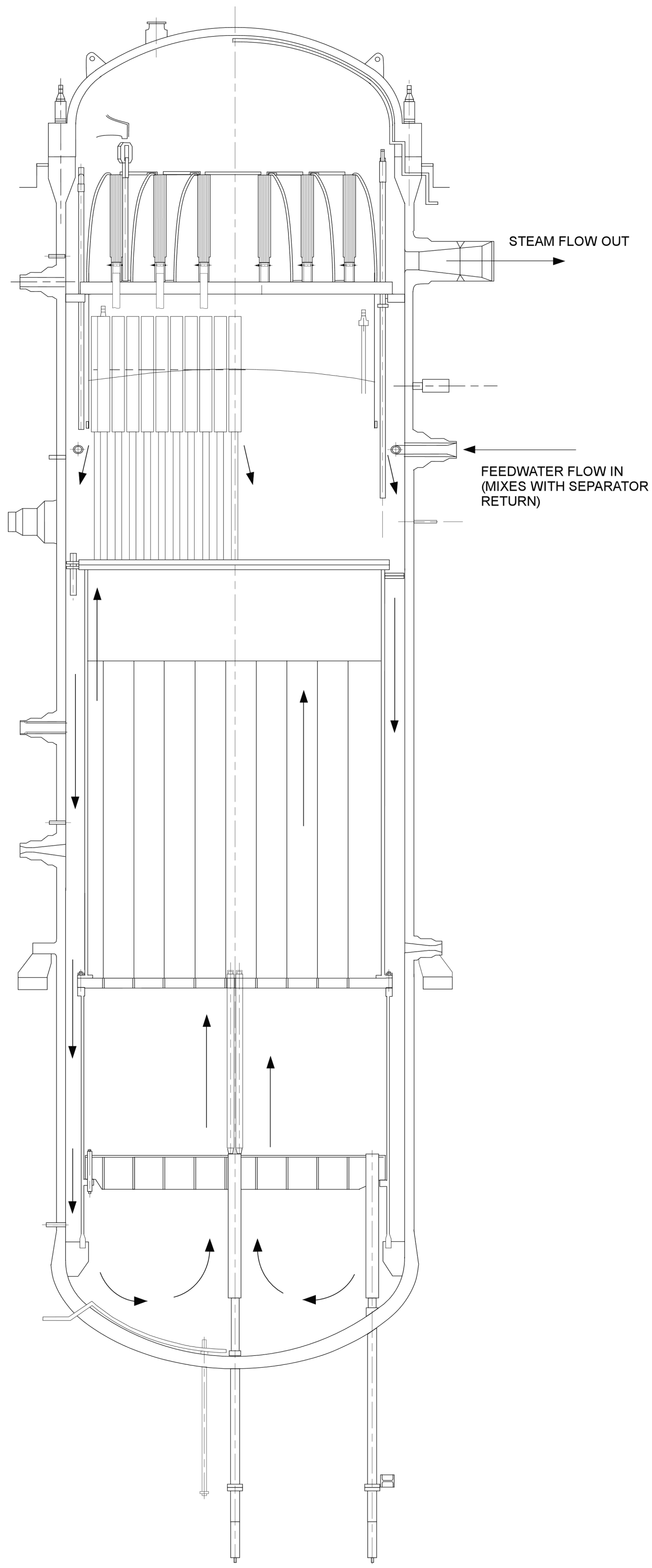


Figure 2.1.3-1. RPV Natural Circulation Process

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 Nonsafety-Related Distributed Control and Information System [N-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 N-DCIS. Upon receipt of a SCCRI signal, the Diverse Protection System (DPS) will initiate a Select Rod Insert (SRI).

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-divisional AC power sources with at least one power source being a nonsafety-related uninterruptible power supply. The Induction Motor Controller Cabinets (IMCCs), Rod Brake Controller Cabinets (RBCCs) and Emergency Rod 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 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.	5.	5.
a. 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).	a. 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.	a. The performed tests 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.
b. 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.	b. 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.	b. The performed tests verify 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.
c. 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.	c. 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.	c. The performed tests verify 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.

Table 2.2.1-1

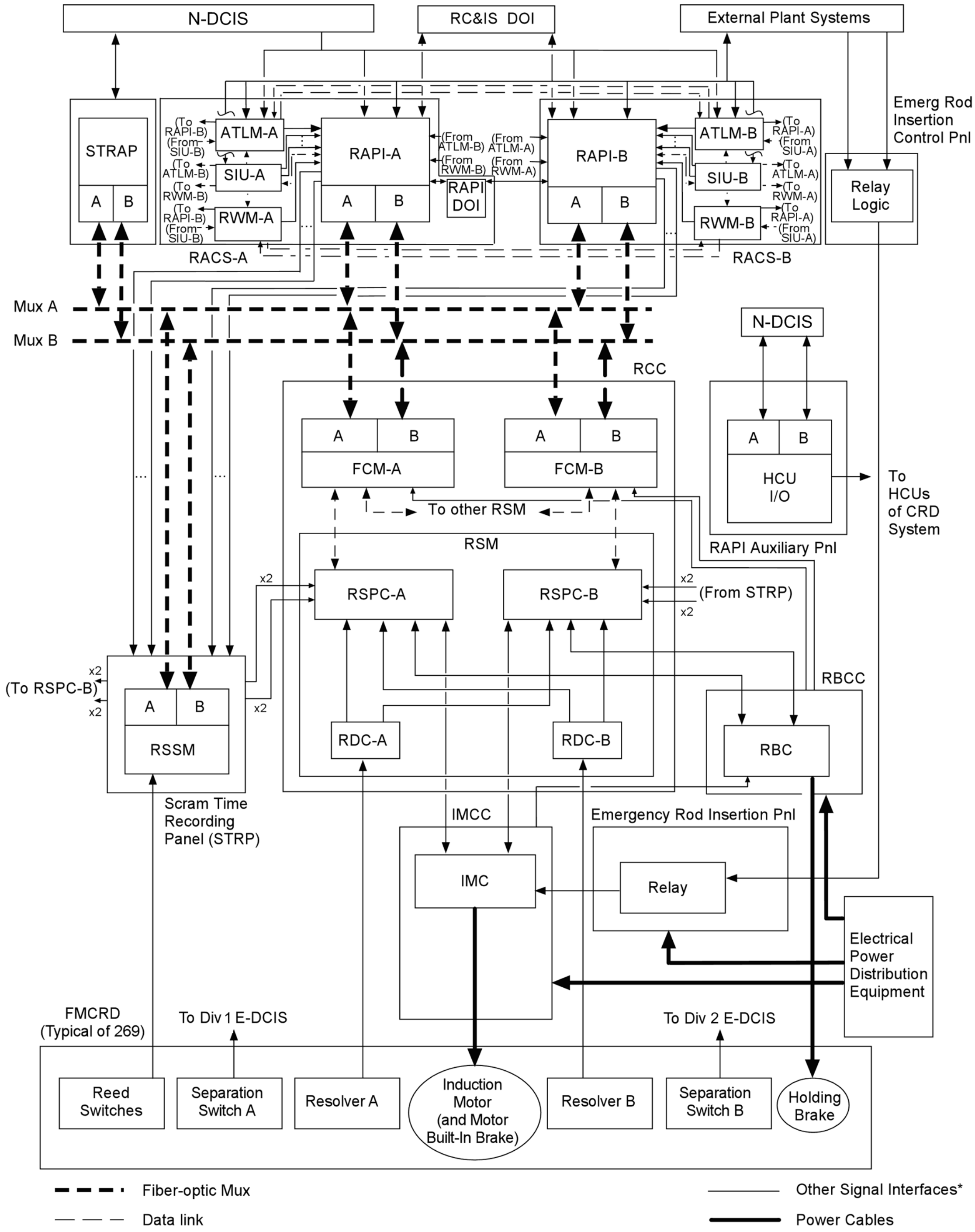
ITAAC For Rod Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. On receipt of Selected Control Rod Run In (SCRRI) signals from the N-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.
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 N-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 N-DCIS.	8. The certified design commitment is met.
9. RC&IS transmits control rod position and status data to the N-DCIS and Neutron Monitoring System	9. Tests of RC&IS will be conducted to output control rods coordinates, positions, and status to the N-DCIS and Neutron Monitoring System.	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.

Table 2.2.1-1

ITAAC For Rod Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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.
12. Rod Action Control Cabinets (RACCs), Remote Communication Cabinets (RCCs) and the DOI of RC&IS are powered from two independent nonsafety-related power sources, with at least one of the independent sources being a nonsafety-related uninterruptible power source.	12. A test of the nonsafety-related 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 to the description in Subsection 2.2.1.



* These signal interfaces may be hardwired connections and/or other signal communication links (as determined in the detailed design).

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 at a speed of 28 ± 5 mm/sec 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 maximum allowable scram insertion times for each FMCRD 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.

One of the two divisions of the RPS scram logic circuitry distributes one division of UPS 120 VAC power to the A solenoids of the HCUs and one division of 250VDC power to the solenoid of one of the two air header dump valves. The other division of scram logic circuitry distributes another division of UPS 120 VAC power to the B solenoids of the HCUs and another division of 250VDC power to the solenoid of the other air header dump valve.

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 magnetic coupling; 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 magnetic coupling; 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 CRD System is arranged in a manner that separates the safety-related equipment from the nonsafety-related portions of the system. The FMCRDs are mounted to the reactor vessel bottom head inside primary containment. The HCUs are housed in four dedicated rooms located directly outside of the primary containment at the basemat elevation of the Reactor Building. These rooms are arranged around the periphery of the primary containment wall. Each HCU room serves the FMCRDs associated with one quadrant of the reactor core. The HCUs are connected to the FMCRDs by the scram insert piping that penetrates the primary containment wall.

The balance of the nonsafety-related hydraulic system equipment (pumps, valves, filters, etc.) is physically separated from the HCUs and housed in a separate room in the reactor building. It is

connected to the HCUs by the nonsafety-related FMCRD purge water header, HCU charging water header and scram air header. These headers are classified as Seismic Category II so that they will maintain structural integrity during a seismic event and not degrade the functioning of the HCUs.

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

For the FMCRD separation switches, independence is provided between the safety-related divisions, and between the safety-related divisions and nonsafety-related 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 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 to 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 at a speed of 28 ± 5 mm/sec 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 at a speed of 28 ± 5 mm/sec.

Table 2.2.2-1

ITAAC For Control Rod Drive System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The maximum allowable scram insertion times for each FMCRD 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 maximum allowable scram times for each FMCRD 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.
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.

Table 2.2.2-1

ITAAC For Control Rod Drive System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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 safety-related power supply. For the four HCU charging water header pressure sensors, independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	8. a. Tests will be conducted on the as-built charging water header sensors by providing a test signal in only one safety-related division at a time. b. Inspections of the as-installed charging water header sensor safety-related divisions will be conducted.	8. a. The test signal exists only in the safety-related Division under test. b. Independence exists between safety-related divisions. Independence exists between these safety-related divisions and nonsafety-related equipment.
9. For the FMCRD separation switches, independence is provided between the safety-related divisions and also between the safety-related divisions and nonsafety-related equipment.	9. Inspections of the as-installed safety-related divisions in the CRD system will be performed.	9. In the CRD system, independence exists between safety-related divisions. Independence exists between safety-related divisions and nonsafety-related 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.

Table 2.2.2-1

ITAAC For Control Rod Drive System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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.

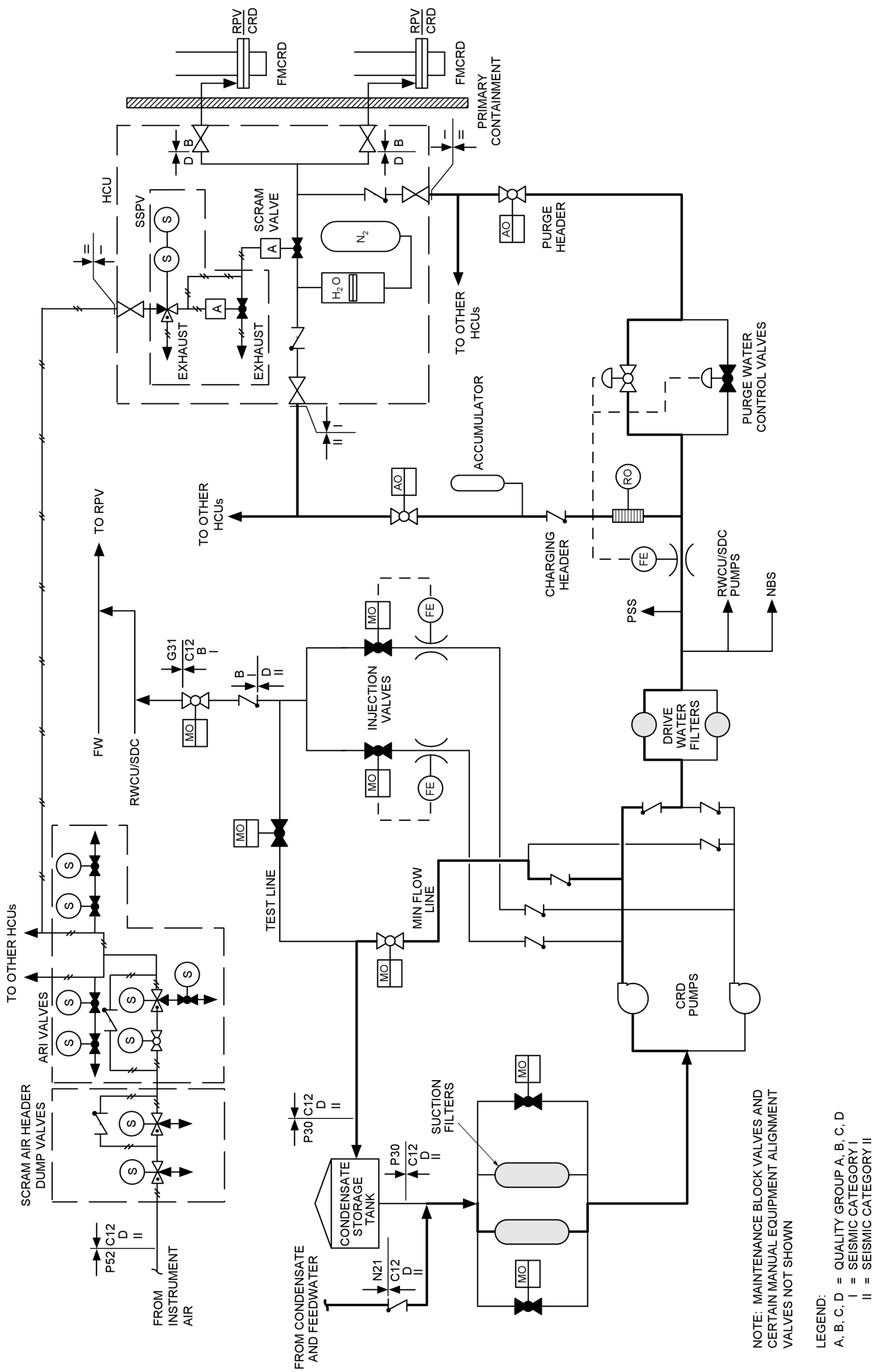


Figure 2.2.2-1. Control Rod Drive System

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. FWCS receives input from and provides output to other systems through the Nonsafety-Related Distributed Control & Information System (N-DCIS) to accomplish its control function, as shown in the feedwater control system logic functional diagram in Figure 2.2.3-1.

FWCS equipment consists of a Fault-Tolerant Digital Controller (FTDC), which is a triplicated, redundant, 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(s) 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 Low Level and High Level. If either of these limits is reached during normal operation, an alarm is generated.

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 is monitored and alarmed if found to be excessive.

If high-high water Level is reached, a signal is generated to initiate runback of the feedwater demand to zero and trip the main turbine.

Upon receipt of an Anticipated Transient Without Scram (ATWS) trip signal from the Diverse Protection Systems (DPS), FWCS initiates a runback of feedwater pump feedwater demand to zero and closes the LFCV and the RWCU/SDC Overboard flow control valve.

The FWCS operating mode is selectable from the MCR. (See Table 2.2.3-1.)

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.3-2 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria 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 (PAS), mode Not in PAS mode

Table 2.2.3-2

ITAAC For Feedwater Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Control Room controls, monitoring and alarms provided for the FWCS are as defined in Subsection 2.2.3.	1. Inspections and tests will be performed on the control room controls, monitoring and alarms for the FWCS.	1. Inspections and tests confirm that controls, monitoring and alarms exist or can be retrieved in the Control Room as defined in Subsection 2.2.3.
2. The FWCS incorporates redundant, fault tolerant digital controllers (FTDC).	2. A test will be performed by simulating failure of an operating FWCS FTDC.	2. A test confirms that there is no loss of FWCS output upon loss of any one FTDC.
3. The FWCS FTDCs identify and isolate failure of process input signals.	3. Tests will be performed by simulating input signal failures to the FWCS FTDCs.	3. A test confirms that the FWCS FTDCs output signal is based upon the remaining valid input signals.
4. The FWCS is powered by redundant, uninterruptible power supplies.	4. A test shall be performed by simulating failure of a power supply to the FWCS.	4. A test confirms that there is no loss of FWCS output upon loss of any one power supply.
5. 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.	5. Inspections and tests will be performed on the FWCS components and installed configuration. Using simulated signals, testing will be performed on the FWCS.	5. Inspections and tests confirm that FWCS is configured correctly, so that process variables from other systems can be monitored and the FWCS control demands and trip signals can be sent to other systems. The system interface function is confirmed by the above.

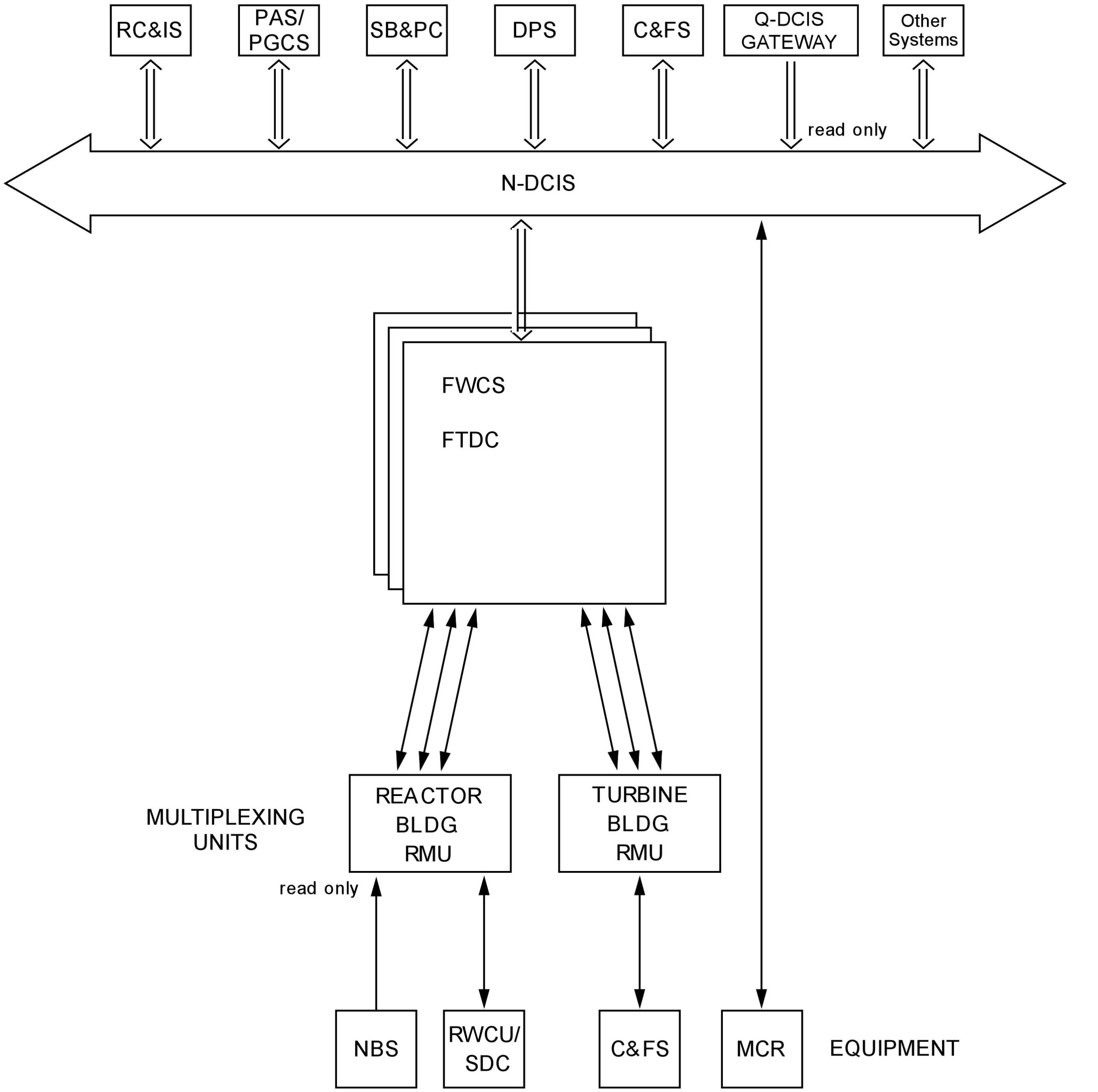


Figure 2.2.3-1. Feedwater Control System Logic Functional Diagram

2.2.4 Standby Liquid Control System

Design Description

Design Description

The Standby Liquid Control (SLC) system is used as an alternative method for reactor shutdown in the event of a failure of the control rods to insert. This is accomplished by injecting boron solution, which performs as a neutron absorber, into the reactor. This system is designed to bring the reactor from full power to a cold subcritical condition.

The SLC system has two independent 50% capacity trains, which include piping, valves, accumulators and instrumentation. The SLC system is designed to operate over the range of reactor pressure conditions up to the elevated pressures of an Anticipated Transients 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 is also designed to provide makeup water to the RPV to mitigate the consequences of a Loss-of-Coolant-Accident (LOCA). The Emergency Core Cooling System (ECCS) and the SLC are designed to flood the core during a 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.

The SLC system interfaces with safety-related divisional power for the squib-type injection valves; for the redundant shut-off valves which isolate the accumulator after injection; for accumulator solution level measurement, trip, and alarm functions; and for the particular Nuclear Boiler System (NBS) instrumentation and the ATWS/SLC mitigation logic and the Safety System Logic and Control Engineered Safety Features (SSLC/ESF) System control logic which generates the ATWS mitigation and the ECCS initiation signals respectively for automatic SLC system initiation.

The SLC system also interfaces with the Diverse Protection System (DPS), which provides diverse ECCS actuation of the SLC system, and with the LD&IS for isolation of the RWCU/SDC system on an ATWS/SLC initiation signal.

The following SLC system components required for RPV injection are classified as safety-related and Seismic Category I:

- The accumulators;
- The injection piping and valves between the accumulators and reactor vessel; and
- The accumulator level and pressure instrumentation.

<u>Instrumentation</u>

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

- Alarms for accumulator pressure and accumulator solution level;
- Displays for accumulator pressure and accumulator solution level;
- Controls and status indication for the injection squib valves and shut-off valves; and

- Dual manual system initiation switches.

The redundant injection shut-off valves are automatically closed by low accumulator solution level signals using 2-out-of-4 logic. This prevents nitrogen from entering the reactor vessel after boron injection is complete.

The accumulator nitrogen pressure and solution level low-level alarms are set to provide adequate time for recharging the accumulator with the manually operated nitrogen gas and sodium pentaborate solution supply subsystems.

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 for the SLC system.

Table 2.2.4-1

SLC ATWS Mitigation Function Parameters

Parameter	Value
Initial reactor absolute pressure	≤ 8.61 MPa (1250 psia)
Approximate initial injection flow rate per train	18.4 l/s (292 gpm)
Approximate average injection velocity* for the first 5.4 m^3 of the injection	30.5 m/s (100 ft/s)
Approximate average injection velocity* for the second 5.4 m^3 of the injection	18.4 m/s (60 ft/s)
Total solution injection (per each train) at the initial reactor absolute pressure	≥ 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 ***

*   Nozzel exit velocity into the shroud

** 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. Report(s) exist(s) and conclude(s) that 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. Analyses and tests will be performed as follows:	2. Report(s) exist(s) and conclude(s) the following:
a. Accumulator tank injectable boron solution volume ≥ 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 tank injectable boron solution volume is ≥ 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 shutdown 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	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	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

Table 2.2.4-2

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
of the RWCU/SDC System.	cooling piping and equipment of the RWCU/SDC System.	RWCU/SDC System.
d. Accumulator tank(s) with at least 12.5 wt% 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.	d. The solution is at least 12.5 wt% sodium pentaborate and the B_{10} enrichment is equal to or greater than 94%.
e. The SLC system can be manually initiated from the main control room.	e. Tests will be conducted on the as-built SLC system using the manual initiation switches.	e. The SLC system initiates when the dual manual initiation switches are actuated concurrently.
f. Both trains of the SLC system are automatically initiated during an ATWS event or during a LOCA.	f. Tests will be conducted on the as-built SLC system using simulated ATWS/SLC signals and ECCS initiation signal (from SSLC/ESF or the DPS).	f. Upon receipt of the following simulated signals: (1) ATWS/SLC initiation signal or (2) ECCS initiation signal (from SSLC/ESF or the DPS), both trains of the SLC system automatically initiate.
g. On ATWS mitigation SLC initiation signal, RWCU/SDC system is sent an isolation signal via the LD&IS.	g. Tests will be conducted on the as-built SLC system using simulated ATWS/SLC initiation signal to confirm RWCU/SDC isolation signal via the LD&IS.	g. Following a simulated ATWS mitigation SLC initiation signal, an RWCU/SDC isolation signal is processed via LD&IS.

Table 2.2.4-2

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The system shall be capable of delivering ≥ 5.4 m^3 of the injectable boron solution volume per each train at the average velocities given in Table 2.2.4-1 during an ATWS event with accumulator at 14.82 MPa and reactor at 8.61 MPa.	3. Tests will be conducted with water using an open vessel to demonstrate acceptable system performance. An analysis will be performed to establish test parameters, such as differential pressure, temperature and fluid densities, in order to simulate the design conditions associated with the SLC operation and to establish the acceptance criteria for the test.	3. Tests and analysis reports conclude that SLC system injects ≥ 5.4 m^3 of the injectable water volume per each train within a time period* such that the average velocities given in Table 2.2.4-1, against simulated differential pressure conditions associated with ATWS conditions are achieved. * Based on analysis for the actual test conditions.
4. The SLC system shall be capable of delivering ≥ 7.8 m^3 of the injectable boron solution volume per train 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. The test will be conducted using the same methods as in ITAAC 3.	4. SLC system injects a total volume of ≥ 7.8 m^3 of the injectable water volume per train, 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. Test report(s) document that the 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 safety-related 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. Test report(s) document that the safety functions of SLC system are dependent only on safety-related power supply 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. Test report(s) document that 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.
9. The SLC system logic uses four independent level instrumentation channels to monitor SLC accumulator level. An accumulator injection isolation signal is initiated when any 2-out-of-4 associated level channels have tripped.	9. The accumulator level instrument channels of the SLC system shall be tested using simulated signal inputs to confirm accumulator isolation.	9. Report(s) exist(s) and conclude(s) that the SLC accumulator injection isolation logic uses four independent and redundant instrument channels to monitor level and an injection isolation signal is initiated when any 2-out-of-4 channels have tripped.

Table 2.2.4-2

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. Independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment.	10. Analyses and tests will be performed as follows:	10.
	a. Test(s) will be performed to verify the electrical independence of each safety-related division.	a. Report(s) exist(s) and conclude(s) that electrical independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment.
	b. An inspection will be performed to verify the physical independence of the as-installed safety-related divisions and the nonsafety-related equipment.	b. Report(s) exist(s) and conclude(s) that physical independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment.
	c. Test(s) will be performed to verify communication independence on each redundant network.	c. Report(s) exist(s) and conclude(s) that) that communication independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment.

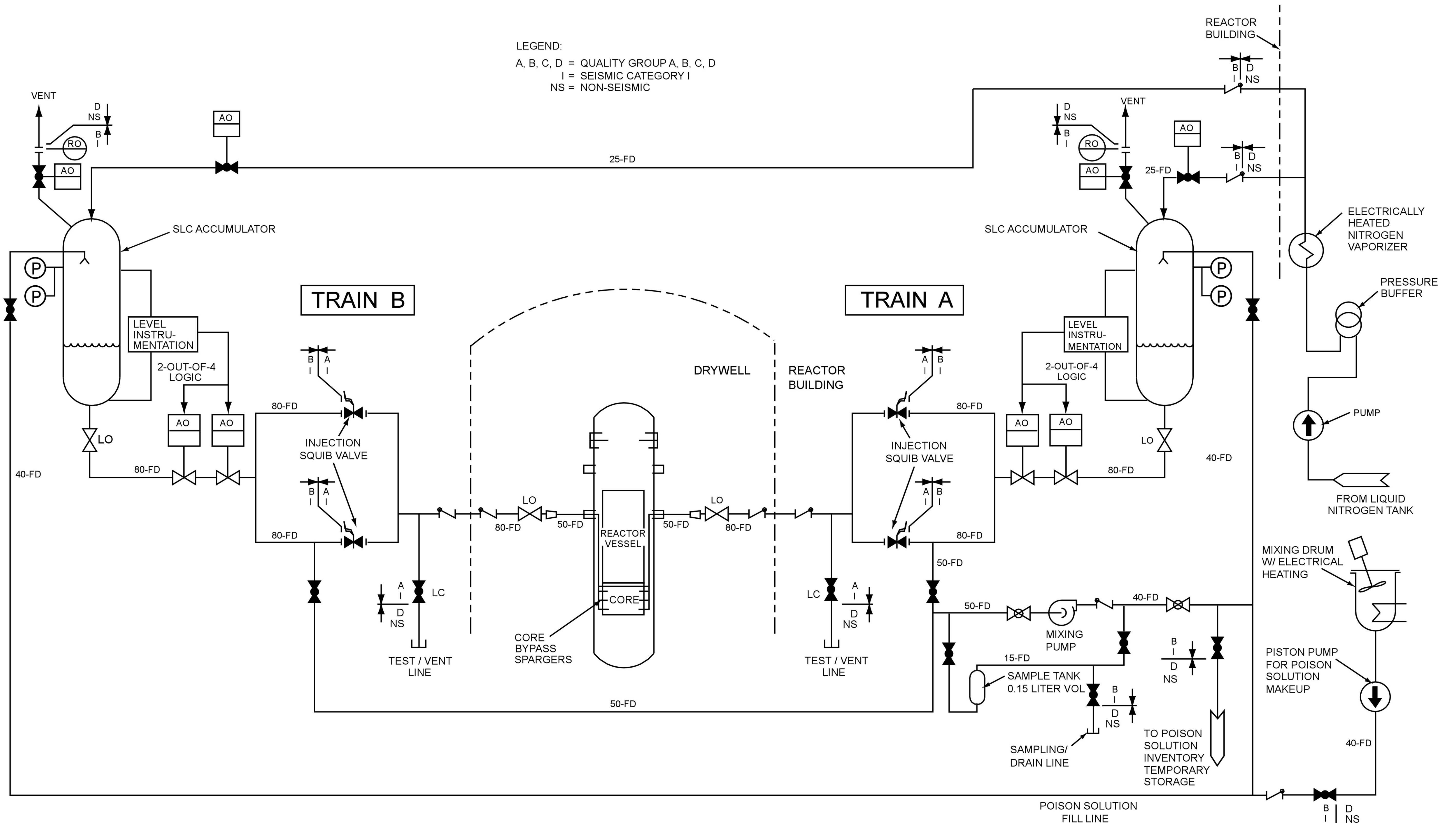


Figure 2.2.4-1. Standby Liquid Control System

2.2.5 Neutron Monitoring System

Design Description

The Neutron Monitoring System (NMS) monitors thermal neutron flux 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 and SSLC; and

(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 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 NMS safety-related components and associated electrical equipment, which are classified as safety-related are Seismic Category 1.

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 safety-related power supplies. In the NMS outside the primary containment, independence is provided between safety-related divisions, and between the safety-related divisions and nonsafety-related 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 trip functions are in effect when the RPS mode switch is <u>not</u> in the RUN position (Non-Coincidence Mode). The SRNM upscale trip setpoint is lowered in the NMS Non-Coincidence Mode (RPS Mode switch in RUN). SRNM trips shall be active only when the reactor mode switch is not in the RUN position..

The SRNM and APRMs are fail-safe in the event of loss of electrical power to any division of their logic equipment. A loss of power to any division results in a divisional trip being provided to the RPS.

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 single 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 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 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 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 and/or tests will be conducted on the as-built configuration as shown in Figures 2.2.5-1 and 2.2.5-2 and functional requirement as described in Subsection 2.2.5	1. Inspection and/or test report(s) exist(s) and conclude(s) that the system conforms to the basic configuration shown in Figures 2.2.5-1 and 2.2.5-2 and functional requirement as described in Subsection 2.2.5.
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 NMS logic is designed to avoid inadvertent initiation by requiring coincident trip of at least two divisions out of four to cause the change of the states of all actuated outputs as described in Subsection 2.2.5.	3. Tests will be conducted using simulated inputs to cause a trip condition in only one division out of the four at a time and checking the status of all other remaining divisions for a change in state of all actuated outputs of other remaining divisions.	3. Report(s) exist(s) and conclude(s) that only the status of the division under test are in a trip condition and the status of all other remaining divisions are in the non-trip condition, thereby conforms that the NMS logic is designed to avoid inadvertent initiation by requiring coincident trip of at least two divisions out of four to cause the change of the states of all actuated outputs as described in Subsection 2.2.5.
4. The NMS logic is designed to provide	4. Tests will be conducted using	4. Report(s) exist(s) and conclude(s) that

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
a trip initiation by requiring coincident trip of at least two divisions out of four to cause the trip output as described in Subsection 2.2.5.	simulated inputs to cause trip conditions in two, then three, and then four divisions at a time of the system	in all test conditions, the system output trip change state, per the system logic design, and thereby provide a trip initiation by coincident trip of at least two divisions out of four to cause the trip output as described in Subsection 2.2.5.
5. The SRNM and PRNM power supplies are provided by the four 120VAC UPS buses.	5. Tests will be conducted after installation.	5. The installed safety-related equipment is powered from the four divisional safety-related UPS.
6. 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.	6. SRNM and APRMS Bypass functions will be tested by using simulated signals.	6. The as-built SRNM and APRMS bypass logics are in accordance with Subsection 2.2.5.
7. The NMS is designed with channel bypass provisions for on-line test and repair during plant operation. When a channel is placed in by-pass condition, the NMS logic changes from two-out-of-four to two-out-of-three trip.	7. Tests will be conducted with a division in the NMS placed in bypass, and a simulated signal is initiated to cause a trip condition in each of the un-bypassed channels in this system.	7. Report(s) exist(s) and conclude(s) that all output channels change state when at least two out of three un-bypassed channels of remaining divisions of the system have tripped by the simulated signal and the channel under bypass did not change state. The trip condition remains until manually reset.
8. The NMS logic is designed fail-safe such that loss of electrical power to a division of RPS results in a trip output from that division.	8. Tests will be conducted by disconnecting electrical power to one division of NMS at a time, to verify that each division provides a trip output to RPS.	8. Report(s) exist(s) and conclude(s) that the NMS logic is designed fail-safe such that, loss of electrical power to a division of NMS, results in a trip output to RPS.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. Independence is provided between safety-related divisions and between safety-related divisions and non-safety related equipment.	9. a. Test(s) will be performed to verify the electrical independence of each safety-related division. b. An inspection will be performed to verify the physical independence of the as-installed safety-related divisions and the nonsafety-related equipment. c. Test(s) will be performed to verify communication independence on each redundant network.	9. a. Test report(s) exist(s)and conclude(s) that electrical independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment. b. Report(s) exist(s) and conclude(s) that physical independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment. c. Report(s) exist(s) and conclude(s) that) that communication independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The NMS is designed to ensure that, safety related system setpoints are defined, determined and implemented based on the setpoint methodology approved by NRC. This setpoint methodology includes the following as a minimum: a. The allowance for uncertainties between the process analytical limit and the device setpoint determined using a documented setpoint methodology.	10. Inspection, tests, and/or analysis will be performed to verify that all the setpoints of instruments associated with the NMS are in conformance with the design requirements.	10. Report(s) exist(s) and conclude(s) that the NMS is designed to ensure that, safety related system setpoints are defined, determined and implemented based on the setpoint methodology approved by NRC. This setpoint methodology includes the following as a minimum: a. The allowance for uncertainties between the process analytical limit and the device setpoint determined using a documented setpoint methodology.

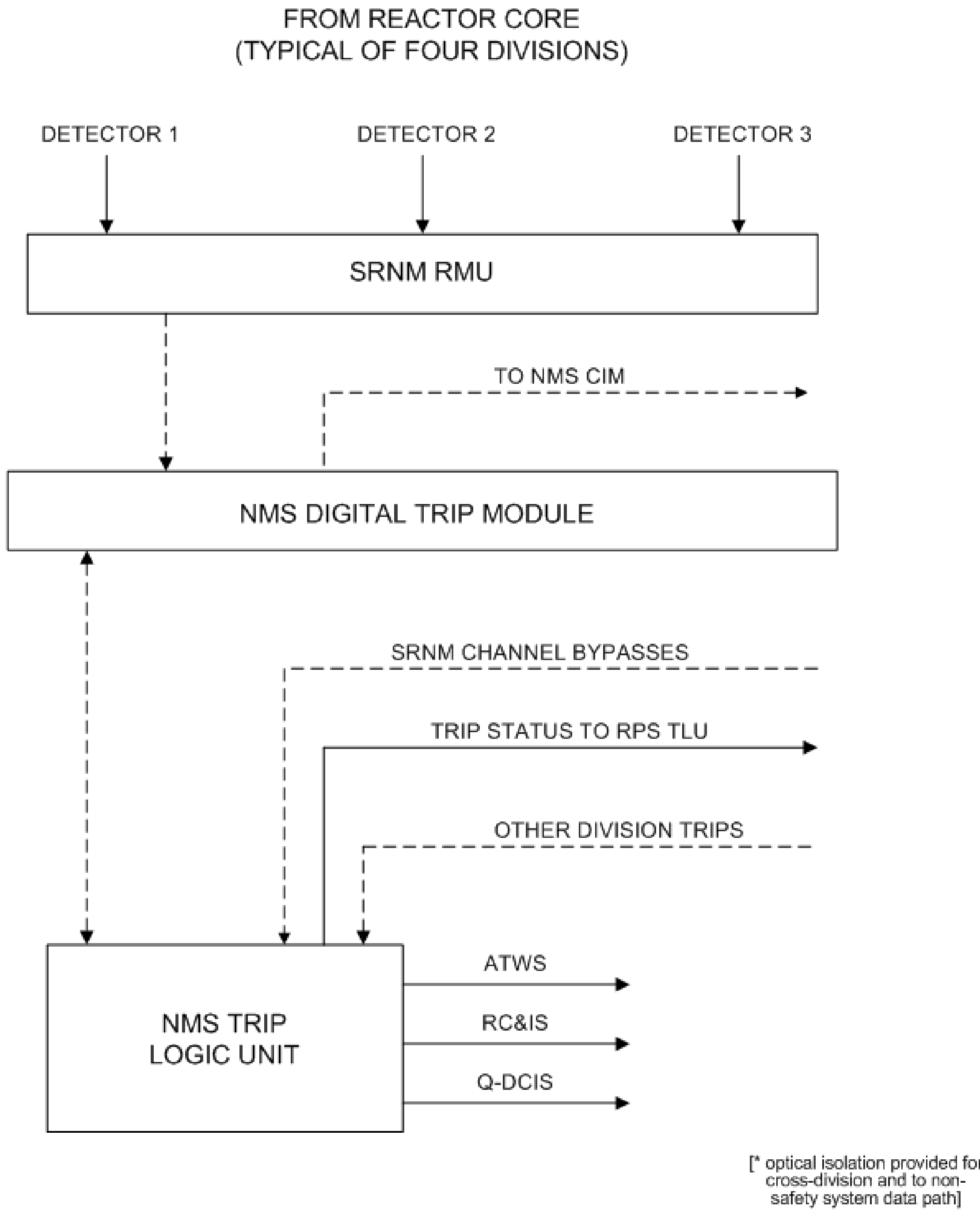


Figure 2.2.5-1. Basic Configuration of a Typical SRNM Division (Subsystem)

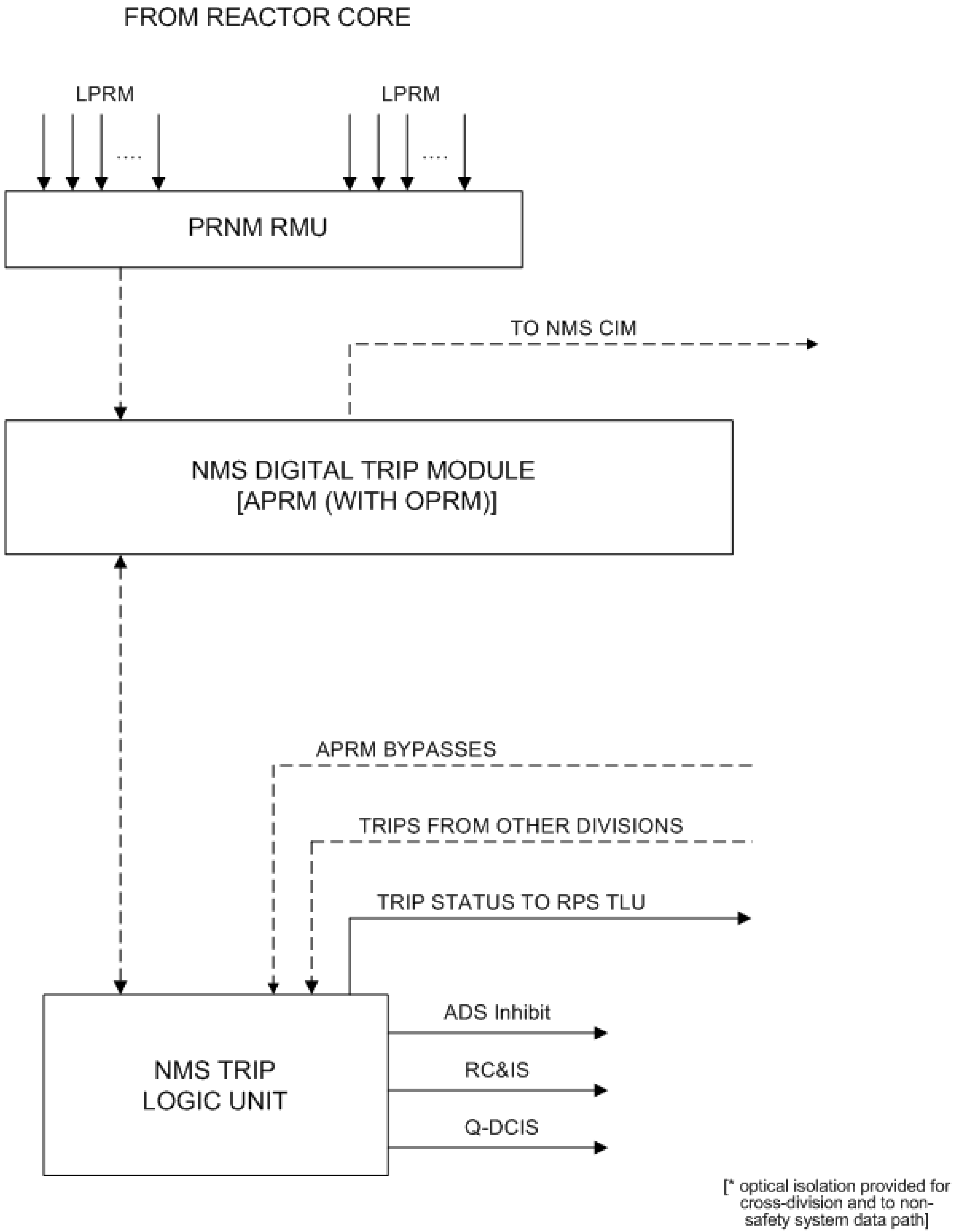


Figure 2.2.5-2. Basic Configuration of a Typical PRNM Division (Subsystem)

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), if the MCR becomes uninhabitable. 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.

The RSS is classified as a safety-related system, and includes control interfaces with both safety-related and nonsafety-related equipment. The RSS has two redundant and independent panels that are located in two separate rooms in separate divisional quadrants of the Reactor Building. Figure 2.2.6-1 shows a schematic arrangement for one RSS panel. All parameters that are indicated and/or controlled from Division 1 and Division 2 in the MCR are also indicated and/or can be controlled from any of the two RSS panels. Similarly, all nonsafety-related parameters that are indicated and/or controlled in the MCR are indicated and/or can be controlled from any of the two RSS panels. Manual scram of the reactor, and manual isolation of the Main Steamline Isolation Valves (MSIVs), is possible from any of the two RSS panels.

Each panel contains the following:

- Division 1 Manual Scram Control,
- Division 2 Manual Scram Control,
- Division 1 Manual MSIV Isolation Control,
- Division 2 Manual MSIV Isolation Control,
- Division 1 Safety-Related Video Display Unit (VDU),
- Division 2 Safety-Related VDU,
- Nonsafety-Related VDU, and
- Communications Equipment.

Division 1 and 2 VDUs on the RSS panels are connected respectively to the same Q-DCIS networks as the Division 1 and 2 VDUs at the MCR. Similarly, the nonsafety-related VDUs on the RSS panels and the MCR are connected to the same N-DCIS network. Power sources for the RSS panel VDUs are identical to the corresponding VDUs at the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

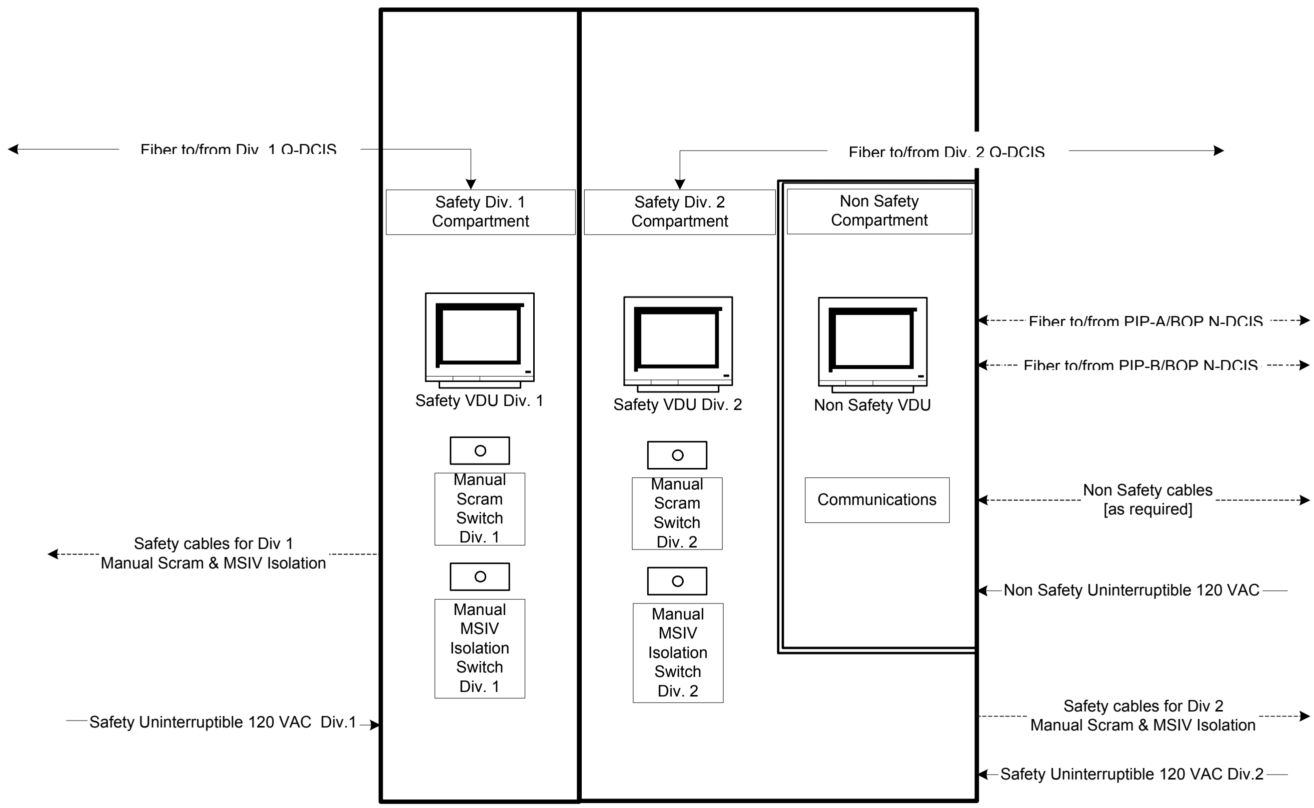
Table 2.2.6-1 provides a definition of the visual inspections, tests and/or analyses, together with associated acceptance criteria for the RSS.

Table 2.2.6-1

ITAAC For The Remote Shutdown System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The RSS has two redundant & independent panels for monitoring and controlling of the interfacing systems. The panels are located in different areas of the Reactor Building. Each panel is configured as shown in Figure 2.2.6-1.	1. Inspections will be conducted to confirm the appropriate location, isolation, and also to confirm that the as-built panels are configured as in Figure 2.2.6-1.	1. The panels conform to their requirements for safety-related divisional separation, safety-related to nonsafety-related separation, are located in separate divisional areas of the Reactor Building, and are in conformance with Figure 2.2.6-1.
2. Each RSS panel consists of two safety-related VDUs (Div 1 & 2), one nonsafety-related touch screen VDU, two manual scram switches (Div 1 & 2), two manual MSIV isolation switches (Div 1 & 2), and communications equipment.	2. Inspections will be conducted to confirm that each RSS panel has the required operator interface devices.	2. Each RSS panel has two safety-related VDUs, one nonsafety-related VDU, two manual scram switches, and two manual MSIV isolation switches. The VDUs have identical display/control provisions and power sources as the corresponding VDUs in MCR. Communication between personnel performing remote shutdown operations is established.
3. Each RSS panel provides the means to trip the reactor, close the MSIVs, and maintain safe shutdown.	3. Tests will be conducted from each RSS panel to simulate manual scram of the reactor, closure of MSIVs, and controls to maintain safe shutdown.	3. Test results indicate that actuation of manual scram, MSIV closure, and controls for shutdown/cooldown systems like RWCU/SDC generate the intended signals for the actuation of devices as applicable.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment.	4. Tests/Inspections will be performed as follows: a. To verify the electrical independence of each safety-related division. b. To verify the physical independence of the as-installed safety-related divisions and the nonsafety-related equipment. c. To verify the independence of communication network between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	4. Electrical independence, physical independence, and independence of communication networks is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.



Note: Typical of one RSS panel. Other panel is identical.

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 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 and Seismic Category I.

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
- Turbine Control Valve (TCV) fast closure
- NMS-monitored SRNM and APRM conditions exceed acceptable limits
- Reactor high pressure
- Reactor water level (Level 3) decreasing
- Reactor water level (Level 8) increasing
- Main steam isolation valve (MSIV) closure (Run mode only)
- Low Control Rod Drive HCU accumulator charging header pressure
- High suppression pool temperature
- High condenser pressure
- Loss of power generation bus (Loss of feedwater flow) (Run mode only)
- Operator-initiated manual scram
- 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 basic configuration block diagram and the signal flow paths from sensors to scram pilot valve solenoids. Each division has a separate safety-related 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).

<u>Process parameters displays</u>:

Reactor vessel pressure;

Reactor water levels;

Primary containment drywell pressures;

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

Suppression pool (local or bulk) temperatures;

Power Generation Bus voltages;

Main Condenser Pressure; and

NMS Outputs.

<u>Status alarms:</u>

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 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. 1, 2, 3, 4 automatic trip);

RPS divisional manual trip (each of the four, i.e., Div. 1, 2, 3, 4 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 Channel A (or B, C, D) and Channel A (or B, C, D) sensors bypassed (four);

Division 1 (or 2, 3, 4) TLU out-of-service bypass (four);

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

Any CRD accumulator-charging-header trip, bypass switch still in BYPASS position with plant in STARTUP or RUN mode; and

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

<u>Bypasses:</u>

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 High Reactor Water Level (Level 8);

Bypass of scram trip for High Condenser Pressure;

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).

<u>Manual Controls:</u>

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. The basic configuration is as shown in Figure 2.2.7-1 and functional requirement is as described in Subsection 2.2.7	1. Inspections and/or tests will be conducted on the as-built configuration as shown in Figure 2.2.7-1 and functional requirement as described in Subsection 2.2.7	1. Inspection and/or test report(s) exist(s) and conclude(s) that the system conforms to the basic configuration shown in Figure 2.2.7-1 and functional requirement as described in Subsection 2.2.7
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. Report(s) exist(s) and conclude(s) that the 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.
3. The RPS logic is designed to avoid inadvertent initiation by requiring coincident trip of at least two divisions out of four to cause the change of the states of all actuated outputs as described in Subsection 2.2.7.	3. Tests will be conducted using simulated inputs to cause a trip condition in only one division out of the four at a time and checking the status of all other remaining divisions for a change in state of all actuated outputs of other remaining divisions.	3. Report(s) exist(s) and conclude(s) that only the status of the division under test are in a trip condition and the status of all other remaining divisions are in the non-trip condition, thereby confirm that the RPS logic is designed to avoid inadvertent initiation by requiring coincident trip of at least two divisions out of four to cause the change of the states of all actuated outputs as described in Subsection 2.2.7.

Table 2.2.7-1

ITAAC For The Reactor Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The RPS logic is designed to provide a trip initiation by requiring coincident trip of at least two divisions out of four to cause the trip output as described in Subsection 2.2.7.	4. Tests will be conducted using simulated inputs to cause trip conditions in two, then three and then four divisions at a time of the system	4. Report(s) exist(s) and conclude(s) that in all test conditions, the system output trip change state, per the system logic design, and thereby provide a trip initiation by coincident trip of at least two divisions out of four to cause the trip output as described in Subsection 2.2.7.
5. RPS Manual trip function is designed to facilitate the operator to perform a manual scram of the reactor by initiating scram manual controls in only Divisions 1 & 2 to interrupt safety-related power to both scram solenoids as described in Subsection 2.2.7.	5. Tests will be conducted in the RPS to initiate a manual trip by initiating manual control in Division 1 and then in Division 2. At first one manual control at a time and then both simultaneously.	5. Report(s) exist(s) and conclude(s) that when manual scram manual control in Division 1 is depressed, Div 1 safety-related AC power is interrupted. When scram manual control in Division 2 is depressed, Div 2 safety-related AC power is interrupted. When both manual controls in Divisions 1 and 2 are depressed simultaneously, safety-related AC power is interrupted to both scram solenoids as described in Subsection 2.2.7.

Table 2.2.7-1

ITAAC For The Reactor Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The RPS is designed with channel bypass provisions for on-line test and repair during plant operation. When a channel is placed in by-pass condition, the RPS logic changes from two-out-of-four to two-out-of-three trip.	6. Tests will be conducted with a division in the RPS placed in bypass, and a simulated signal is initiated to cause a trip condition in each of the un-bypassed channels in this system.	6. Report(s) exist(s) and conclude(s) that all output channels change state when at least two out of three un-bypassed channels of remaining divisions of the system have tripped by the simulated signal and the channel under bypass did not change state. The trip condition remains until manually reset.
7. RPS logic is designed to ensure scram completion once initiated and inhibiting the reset of scram circuitry for a time delay as determined in design, after scram initiation.	7. a. Logic will be reviewed to verify that a proper seal-in circuit is provided to ensure that once the protective action and/or reactor scram is initiated it will be completed. b. Tests will be conducted to ensure scram completion once initiated and then attempting to reset scram circuitry during the time delay determined/identified in design, after scram initiation.	7. a. Approved logic do exists and provides appropriate seal-in circuit to ensure that once the protective action and/or reactor scram is initiated it will go for completion. b. Report(s) exist(s) and conclude(s) that logic is designed to ensure scram completion once initiated and inhibiting the reset of scram circuitry for a time delay as determined in design, after scram initiation.

Table 2.2.7-1

ITAAC For The Reactor Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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 initiating manual scram manual controls 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.
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. Report(s) exist(s) and conclude(s) that all -indications, alarms, and controls exist or can be retrieved in control room as defined in Subsection 2.2.7.
10. The RPS logic is designed fail-safe such that, loss of electrical power to a division of RPS, results in the load drivers (LDs) of that division to change their state from “close” to “open”.	10. Tests will be conducted by disconnecting electrical power to one division of RPS at a time, to verify that the load drivers (LDs) of that division to change their state from “close” to “open”.	10. Report(s) exist(s) and conclude(s) that the RPS logic is designed fail-safe such that, loss of electrical power to a division of RPS, results in the load drivers (LDs) of that division to change their state from “close” to “open”.

Table 2.2.7-1

ITAAC For The Reactor Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. Independence is provided between safety-related divisions and between safety-related divisions and non-safety related equipment.	11. a. Test(s) will be performed to verify the electrical independence of each safety-related division. b. An inspection will be performed to verify the physical independence of the as-installed safety-related divisions and the nonsafety-related equipment. c. Test(s) will be performed to verify communication independence on each redundant network.	11. a. Test report(s) exist(s)and conclude(s) that electrical independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment. b. Report(s) exist(s) and conclude(s) that physical independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment. c. Report(s) exist(s) and conclude(s) that) that communication independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment.

Table 2.2.7-1

ITAAC For The Reactor Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. The RPS is designed to ensure that, safety related system setpoints are defined, determined and implemented based on the setpoint methodology approved by NRC. This setpoint methodology includes the following as a minimum: a. The allowance for uncertainties between the process analytical limit and the device setpoint determined using a documented setpoint methodology.	12. Inspection, tests, and/or analysis will be performed to verify that all the setpoints of instruments associated with the RPS are in conformance with the design requirements.	12. Report(s) exist(s) and conclude(s) that the RPS is designed to ensure that, safety related system setpoints are defined, determined and implemented based on the setpoint methodology approved by NRC. This setpoint methodology includes the following as a minimum: a. The allowance for uncertainties between the process analytical limit and the device setpoint determined using a documented setpoint methodology.

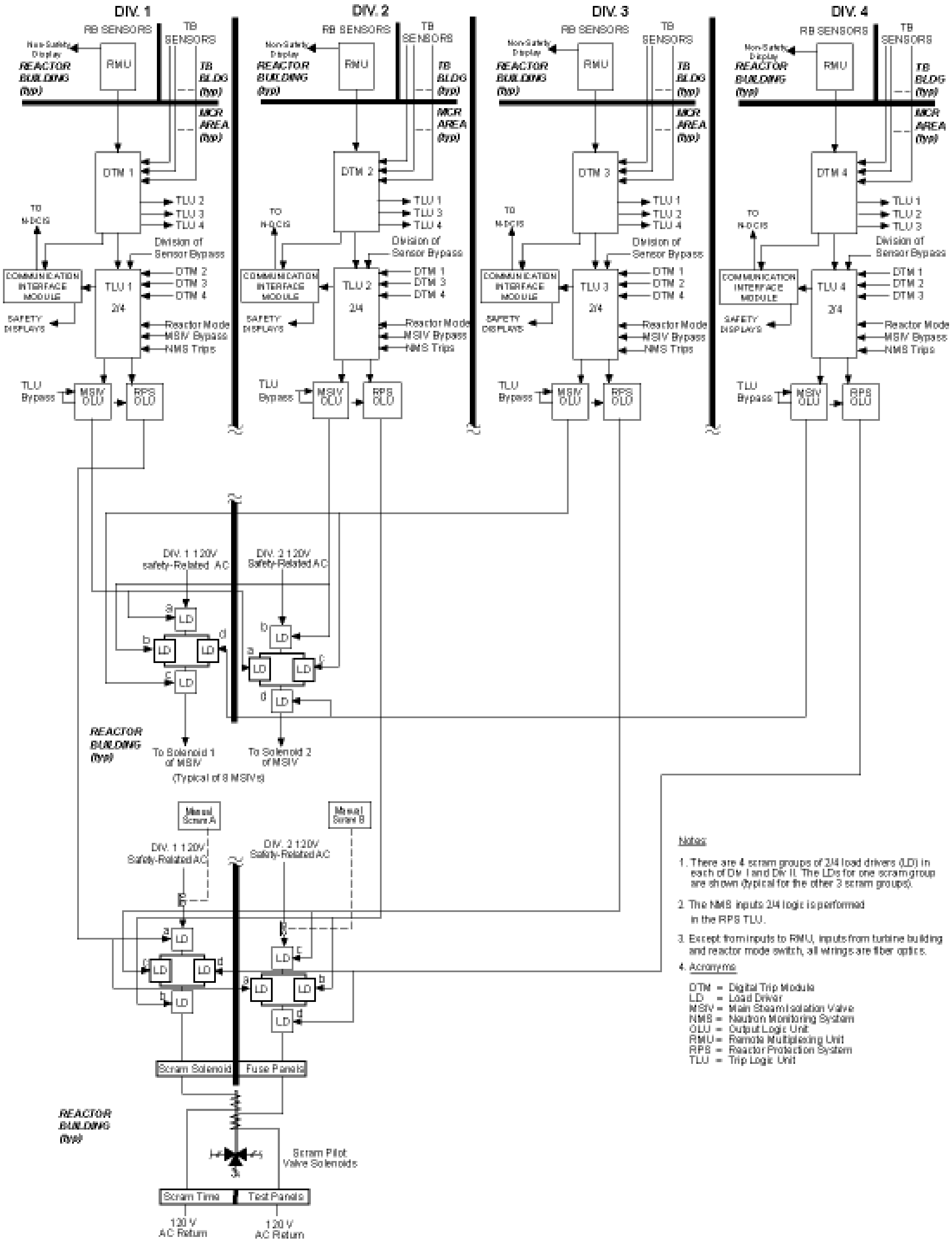


Figure 2.2.7-1. Reactor Protection System Basic Configuration Block Diagram

2.2.8 Plant Automation System

Design Description

The Plant Automation System (PAS) is classified as a power generation system, and it cannot override any safety-related or nonsafety-related function. Events requiring control rod scram are sensed and controlled by the safety-related RPS, which is completely independent of the Plant Automation System.

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

The PAS provides the capability for supervisory control of the plant systems important to power operation by supplying setpoint and task commands to independent nonsafety-related automatic control systems as changing load demands and plant conditions dictate.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this 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 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. The SB&PC system incorporates redundant, fault tolerant digital controllers (FTDC).	1. A test will be performed by simulating failure of an operating SB&PC FTDC.	1. Test report(s) document that the there is no loss of SB&PC output upon loss of any one FTDC.
2. The system incorporates redundant control channels.	2. The system will be tested by simulating failure of any one operating controller.	2. Test report(s) document that the system continues to function during loss of any one operating controller.
3. The system is powered by redundant uninterruptible power supplies.	3. A loss of one power supply test will demonstrate no loss of functions of SB&PC system.	3. Test report(s) document that there is no loss of SB&PC functions upon loss of any one power supply.

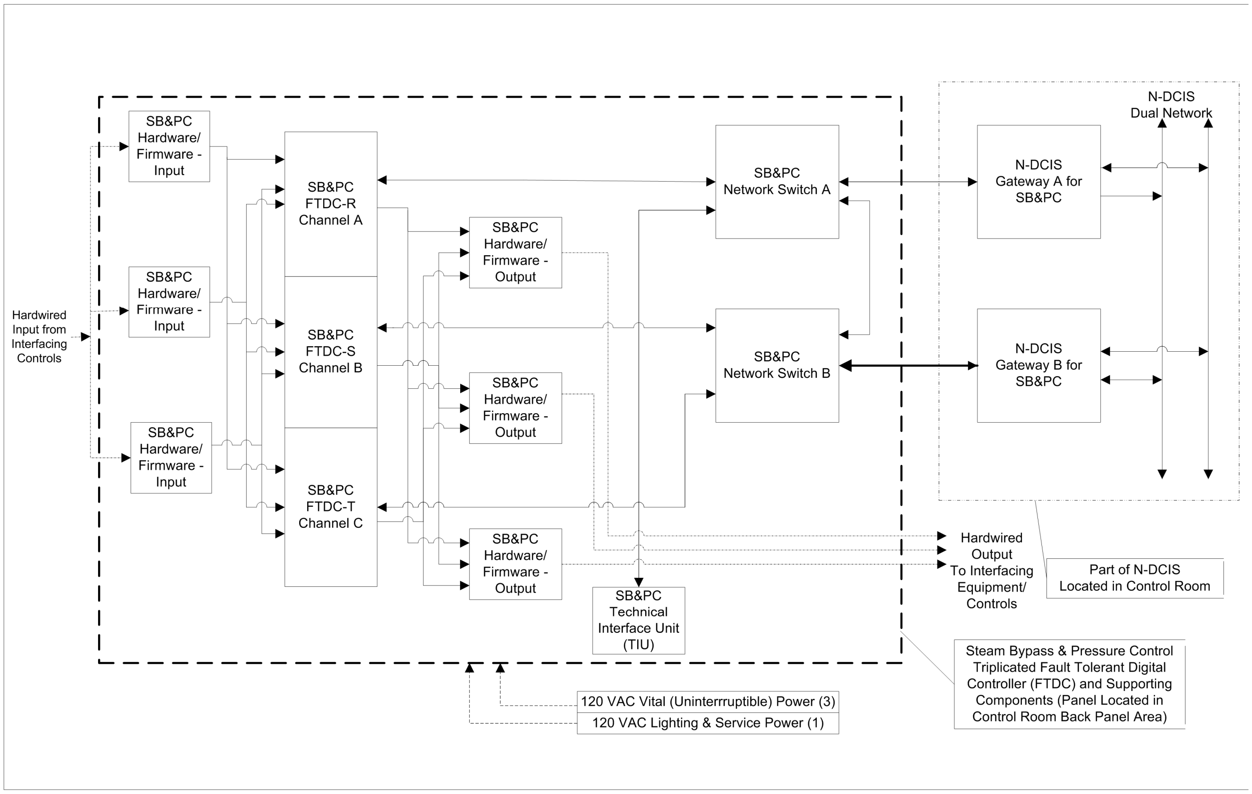


Figure 2.2.9-1. SB&PC Control Interface Simplified Block Diagram

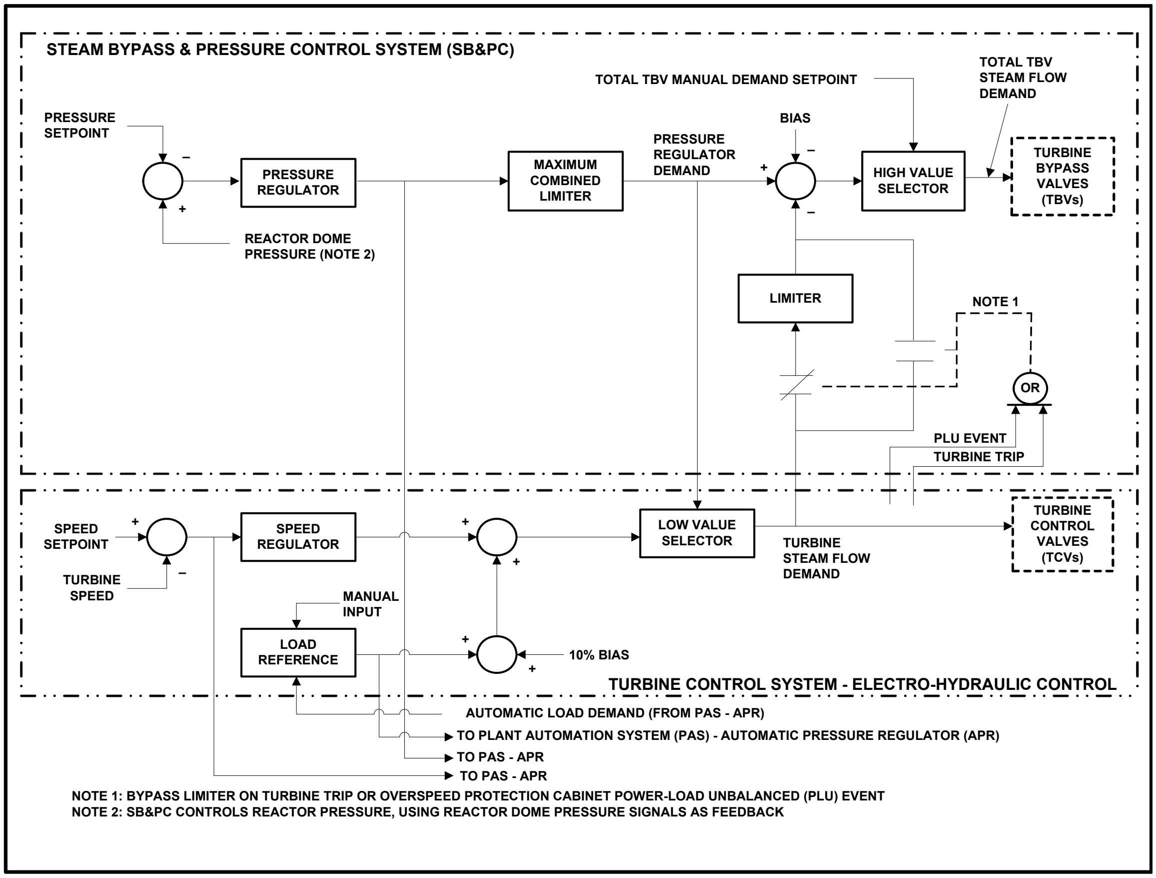


Figure 2.2.9-2. SB&PC Simplified Functional Block Diagram

2.2.10 Safety-Related Distributed Control and Information System

Design Description

The Safety-Related Distributed Control and Information System (Q-DCIS) is the designation given to the collection of hardware and software that comprise the Reactor Protection System (RPS), Neutron Monitoring System (NMS), and SSLC/ESF Systems and several other systems and/or functions that have safety-related components (post accident monitoring, containment monitoring, process radiation monitoring, control room habitability). The safety-related portions of the instrumentation, controls, and monitoring systems and/or functions that comprise Q-DCIS are listed in Table 2.2.10-1. The generic functions of the Q-DCIS include:

- Acquisition of data needed for a system to function;
- Provision of a logic platform to implement system functions;
- Provision of an operator interface to allow control and monitoring;
- Output of manual and automatic calculations and commands to system actuators;
- Communication between divisions; and
- Communication between Q-DCIS and Nonsafety-Related Distributed Control and Information System (N-DCIS).

Electrical power is from redundant safety-related sources of the DC Power Supply and Uninterruptible AC Power Supply.

Inspections, Tests, Analyses and Acceptance Criteria

The inspections, tests, and/or analyses, together with associated acceptance criteria for Q-DCIS are contained within the ITAAC tables for the systems in Table 2.2.10-1. The Q-DCIS power supplies and their ITAACs are described in Subsections 2.13.3 (DC Power Supply) and 2.13.5 (Uninterruptible AC Power Supply).

Table 2.2.10-1

Systems and Functions Comprising The Q-DCIS

System	Subsection
Standby Liquid Control	2.2.4
Neutron Monitoring System	2.2.5
Remote Shutdown System	2.2.6
Reactor Protection System	2.2.7
Leak Detection and Isolation System	2.2.12
Safety System Logic and Control • Automatic Depressurization System • Control Room Habitability System	2.2.13
Process Radiation Monitoring System	2.3.1
Isolation Condenser System	2.4.1
Gravity-Driven Cooling System	2.4.2
Nuclear Boiler System	2.1.2
Containment Monitoring System • Suppression Pool Temperature Monitoring Function	2.15.7
Post Accident Monitoring Instrumentation	3.7

2.2.11 Nonsafety-Related Distributed Control and Information System

Design Description

The Nonsafety-Related Distributed Control and Information System (N-DCIS) is the designation given to the collection of hardware and software that comprise the nonsafety-related instrumentation, controls and monitoring systems and/or functions.

N-DCIS has no safety-related function.

Electrical power is from redundant nonsafety-related sources of the DC Power Supply and Uninterruptible AC Power Supply.

N-DCIS includes the Diverse Protection System, which is addressed Subsection 2.2.14.

Inspections, Tests, Analyses and Acceptance Criteria

The inspections, tests, and/or analyses, together with associated acceptance criteria for N-DCIS are contained (as required) within the ITAAC tables for the applicable systems.

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 2.2.12-1.

The following functions are supported by the LD&IS:

- Containment isolation following a LOCA event;
- Main steamlines and drain lines isolation;
- Isolation Condenser System process lines isolation;
- RWCU/SDC system process and sampling lines isolation;
- Fuel and Auxiliary Pools Cooling system suction lines from the GDCS pools isolation;
- Chilled Water System lines to DW coolers isolation;
- Drywell sumps liquid drain lines isolation;
- Containment purge and vent lines isolation;
- Reactor Building HVAC (RBVS) air exhaust ducts isolation;
- Feedwater system process lines 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 LD&IS functions are performed in two separate safety-related platforms. The Main Steam Isolation Valve (MSIV) isolation logic functions are performed in the RPS/RTIF platform while all other containment isolation logic functions are performed in the SSLC/ESF system.

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, such that loss of electrical power to one LD&IS divisional logic channel initiates a channel trip. The logic is energized at all times and de-energizes to trip for isolation functions. The divisional LD&IS logic channels and associated sensors are powered from safety-related divisional power.

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. Electrical, communication and physical independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment.

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.

The LD&IS is designed to ensure that, safety-related system setpoints are defined, determined and implemented based on an NRC approved setpoint methodology.

Manual controls 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 N-DCIS, and are indicated 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 for the LD&IS.

Table 2.2.12-1

LD&IS Interfacing Sensor Parameters

Temperatures:

- MSL Tunnel Area Temperature
- Drywell Temperature
- RWCU/SDC Rooms Temperature
- MSL Temperature in Turbine Building

Pressures:

- MSL Turbine Inlet Pressure
- Main Condenser Pressure
- Reactor Vessel Head Flange Seal Pressure Leakage
- Drywell Pressure
- Feedwater Line Differential Pressure

Radiation Levels:

- RCCWS Intersystem Leakage
- Drywell Fission Product
- RBVS 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 Flow
- RWCU/SDC Differential 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 basic configuration is as shown in Figure 2.2.12-1 and functional requirement is as described in Subsection 2.2.12.	1. Inspections and/or tests will be conducted on the as-built configuration as shown in Figure 2.2.12-1 and functional requirement as described in Subsection 2.2.12.	1. Report(s) exist(s) and conclude(s) that the system conforms to the basic configuration shown in Figure 2.2.12-1 and functional requirement as described 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. Tests will be performed on each LD&IS instrument channel to verify that LD&IS monitors and detects leakages from the RCPB, and initiates closure of primary and secondary containment isolation valves.	2. Report(s) exist(s) and conclude(s) that LD&IS monitors and detects leakages from the RCPB, and initiates closure of primary and secondary containment isolation valves.
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. Tests will be performed on the four LD&IS isolation logic redundant instrument channels using simulated signals to verify that the isolation signal is initiated when any two-out-of-four channels have tripped.	3. Report(s) exist(s) and conclude(s) that the LD&IS isolation logic uses four redundant instrument channels to monitor each RCPB leakage parameter and the isolation signal is initiated when 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.

Table 2.2.12-2

ITAAC For Leak Detection and Isolation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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.
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 valve 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 valve 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. Report(s) exist(s) and conclude(s) that manual reset controls are provided to perform reset functions as described in Subsection 2.2.12.
8. Control room alarms, indications and/or controls are provided as defined in Subsection 2.2.12.	8. Inspections will be performed on the control room alarms, indications and/or controls for the LD&IS.	8. Report(s) exist(s) and conclude(s) that all alarms, indications and/or controls are present or can be retrieved in the control room as defined in Subsection 2.2.12.

Table 2.2.12-2

ITAAC For Leak Detection and Isolation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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. Report(s) exist(s) and conclude(s) that LD&IS logic design is fail-safe, such that loss of electrical power to one LD&IS divisional logic channel initiates a channel trip.
10. The divisional LD&IS logic channels and associated sensors are powered from safety-related divisional power.	10. Tests will be performed on the LD&IS system by providing a test signal in only one safety-related division at a time and verify that the divisional LD&IS logic channels and associated sensors are powered from safety-related divisional power.	10. Report(s) exist(s) and conclude(s) that the divisional LD&IS logic channels and associated sensors are powered from safety-related divisional power.

Table 2.2.12-2

ITAAC For Leak Detection and Isolation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. Independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment.	11.	11.
	a. Test(s) will be performed to verify the electrical independence of each safety-related division.	a. Report(s) exist(s)and conclude(s) that electrical independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment.
	b. An inspection will be performed to verify the physical independence of the as-installed safety-related divisions and the nonsafety-related equipment.	b. Report(s) exist(s) and conclude(s) that physical independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment.
	c. Test(s) will be performed to verify communication independence on each redundant network.	c. Report(s) exist(s) and conclude(s) that) that communication independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment.

Table 2.2.12-2

ITAAC For Leak Detection and Isolation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. The LD&IS is designed to ensure that, safety related system setpoints are defined, determined and implemented based on the NRC approved setpoint methodology. This setpoint methodology includes the following as a minimum: The allowance for uncertainties between the process analytical limit and the device setpoint determined using an NRC approved setpoint methodology.	12. Inspection, tests, and/or analysis will be performed to verify that all the setpoints of instruments associated with the LD&IS are in conformance with the design requirements.	12. Report(s) exist(s) and conclude(s) that the LD&IS is designed to ensure that, safety-related system setpoints are defined, determined and implemented based on the NRC approved setpoint methodology. This setpoint methodology includes the following as a minimum: The allowance for uncertainties between the analytical limit and the device setpoint determined using an NRC approved setpoint methodology.

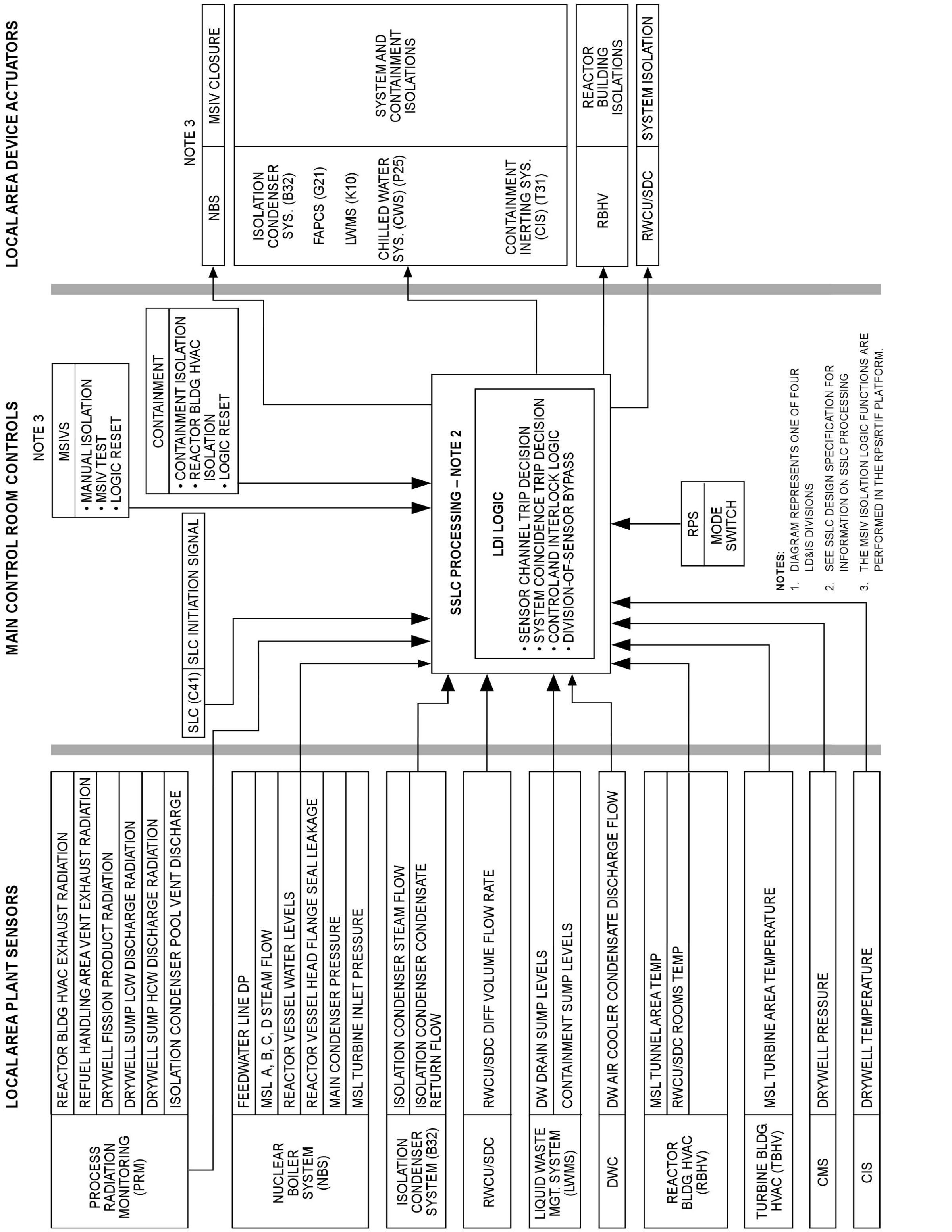


Figure 2.2.12-1. Leak Detection and Isolation System Basic Configuration Block Diagram

2.2.13 Engineered Safety Features Safety System Logic and Control

Design Description

The Safety System Logic and Control for the Engineered Safety Features systems (SSLC/ESF) is the decision-making control logic segment of the engineered safety features systems indicated on Figure 2.2.13-2. SSLC/ESF processes automatic and manual demands for nuclear system isolation, and engineered safety features actuation based upon sensed plant process parameters or operator request.

The SSLC/ESF logic does not require operator intervention during normal operation. The SSLC/ESF 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/ESF uses "energized-to-trip" and "fail-as-is" logic. The logic is designed such that once initiated automatically or manually, the intended sequence of protective actions will continue until completion.

The SSLC/ESF include the emergency core cooling system (ECCS) initiation logic, the isolation logic for the control room habitability system (CRHS), and the isolation function logic for the leak detection and isolation system (LD&IS) excluding the main steam isolation valve (MSIV) isolation logic (MSIV isolation logic is implemented in the safety-related logic processing platform for the Reactor Protection System). The SSLC/ESF also includes the safety-related logic for the safe shutdown function of the ICS.

The ECCS initiation logic actuates the Automatic Depressurization System (ADS), which is comprised of the Safety Relief Valves (SRVs) and Depressurization Valves (DPVs), the Gravity-Driven Cooling System (GDCS), the Isolation Condenser System (ICS), and the Standby Liquid Control System.

The safety-related instrumentation for the CRHS is designed to isolate the control room envelope and re-align to emergency filtration mode on high inlet ventilation radiation or detection of smoke at the inlet ventilation. The LD&IS is described in Subsection 2.2.12.

The SSLC/ESF is configured as a four-division data acquisition and control system, with each physically isolated and redundant instrumentation division containing an independent set of microprocessor-based, software-controlled logic processors. The basic configuration of the SSLC/ESF is depicted on Figure 2.2.13-1. 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/ESF 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 Q-DCIS or directly hardwired from transmitters or sensors. Trip signals from the four redundant sensor divisions are processed in a two-out-of-four 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. The bypass condition is indicated in the control room.

- (For SSLC/ESF) safety critical automatic operations are provided with manual controls in each division for equipment initiation.
- SSLC/ESF provides alarm and status outputs to operator indications, alarms and the plant computer.

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 for the SSLC/ESF system.

Table 2.2.13-1

ITAAC For Safety System Logic and Control/ESF System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration is as shown in Figure 2.2.13-1 and functional requirement is as described in Subsection 2.2.13.	1. Inspection and/or tests will be conducted in the as-built-configuration as shown in Figure 2.2.13-1 and functional requirement as described in Subsection 2.2.13.	1. Report(s) exist(s) and conclude(s) that the system conforms to the basic configuration shown in Figure 2.2.13-1 and functional requirement as described in Subsection 2.2.13.
2. The SSLC/ESF 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/ESF safety function shall be tested using simulated signal inputs.	2.Report(s) exist(s) and conclude(s) that the SSLC/ESF logic uses four independent and redundant instrument channels to monitor each safety-related parameter and a trip signal is initiated when any two-out-of-four channels have tripped.
3. The SSLC/ESF 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/ESF logic function will be tested using simulated signal inputs, with the one channel under bypass condition. No trip signal shall be resulted from the bypassed channel.	3. Report(s) exist(s) and conclude(s) that isolation 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/ESF provides separate manual controls in the control room for each safety-related ESF function.	4. Tests will be performed using the SSLC/ESF functions in the control room.	4. Report(s) exist(s) and conclude(s) that SSLC/ESF provides separate manual controls in the control room for each safety-related ESF function.

Table 2.2.13-1

ITAAC For Safety System Logic and Control/ESF System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	5. a. Test(s) will be performed to verify the electrical independence of each safety-related division. b. An inspection will be performed to verify the physical independence of the as-installed safety-related divisions and the nonsafety-related equipment.	5. a. Report(s) exist(s) and conclude(s) that electrical independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment. b. Report(s) exist(s) and conclude(s) that physical independence exists between each of safety-related divisions, and also between safety-related divisions and nonsafety-related equipment.
6. SSLC/ESF equipment in each division is powered from the divisional, safety-related power sources.	6. Test(s) will be performed applying divisional power to the assigned equipment division to perform self-test on each SSLC/ESF controller in the division to verify that SSLC/ESF equipment in each division is powered from the divisional, safety-related power sources.	6. Report(s) exist(s) and conclude(s) that SSLC/ESF equipment in each division is powered from the divisional, safety-related power sources.

Table 2.2.13-1

ITAAC For Safety System Logic and Control/ESF System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. Bypass Implementation SSLC/ESF provides the following bypass functions: a. Division-of-sensors bypass b. Output trip logic bypass	7. Bypass Implementation Tests will be performed to verify the SSLC bypass functions.	7. Bypass Implementation Report(s) exist(s) and conclude(s) that: 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. For the squib valves, keylock switches provide effective bypass at the actuator level. Bypass status is indicated at main control panel.
8. Control room alarms, indications and/or controls are provided as defined in Subsection 2.2.13.	8. Inspections will be performed on the control room alarms, indications and/or controls for the SSLC/ESF system.	8. Report(s) exist(s) and conclude(s) that all alarms, indications and/or controls are present or can be retrieved in the control room as defined in Subsection 2.2.13.
9. The SSLC/ESF system is designed to ensure that, safety-related system setpoints are defined, determined and implemented based on the setpoint methodology approved by the NRC. This setpoint methodology includes the following as a minimum: The allowance for uncertainties	9. Inspections, tests, and/or analysis will be performed to verify that all the setpoints of instruments associated with the SSLC/ESF system are in conformance with the design requirements.	9. Report(s) exist(s) and conclude(s) that the SSLC/ESF system is designed to ensure that, safety-related system setpoints are defined, determined and implemented based on the NRC approved setpoint methodology. This setpoint methodology includes the following as a minimum:

Table 2.2.13-1

ITAAC For Safety System Logic and Control/ESF System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
between the process analytical limit and the device setpoint determined using a documented NRC approved setpoint methodology.		The allowance for uncertainties between the process analytical limit and the device setpoint determined using an NRC approved setpoint methodology.

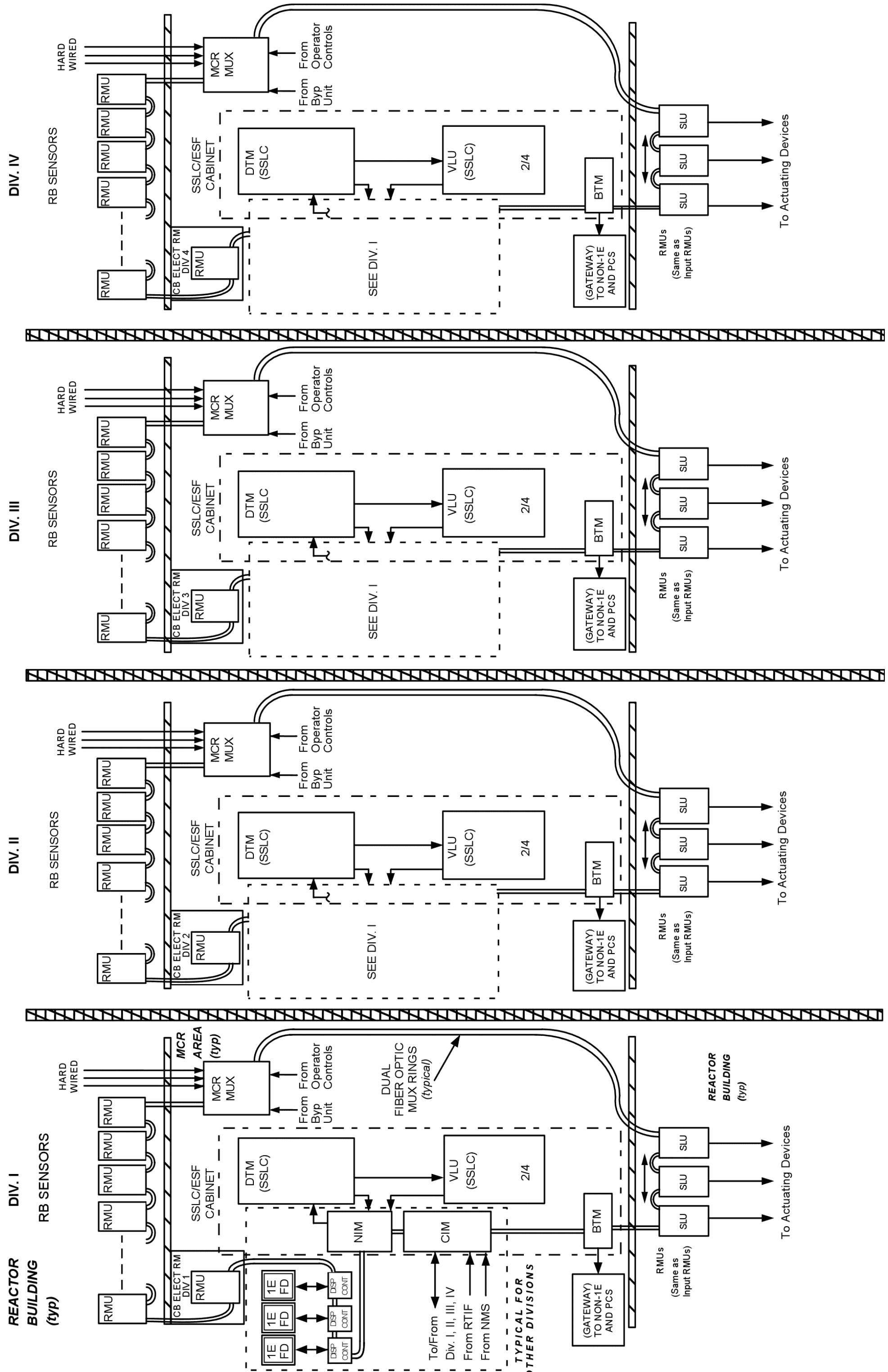
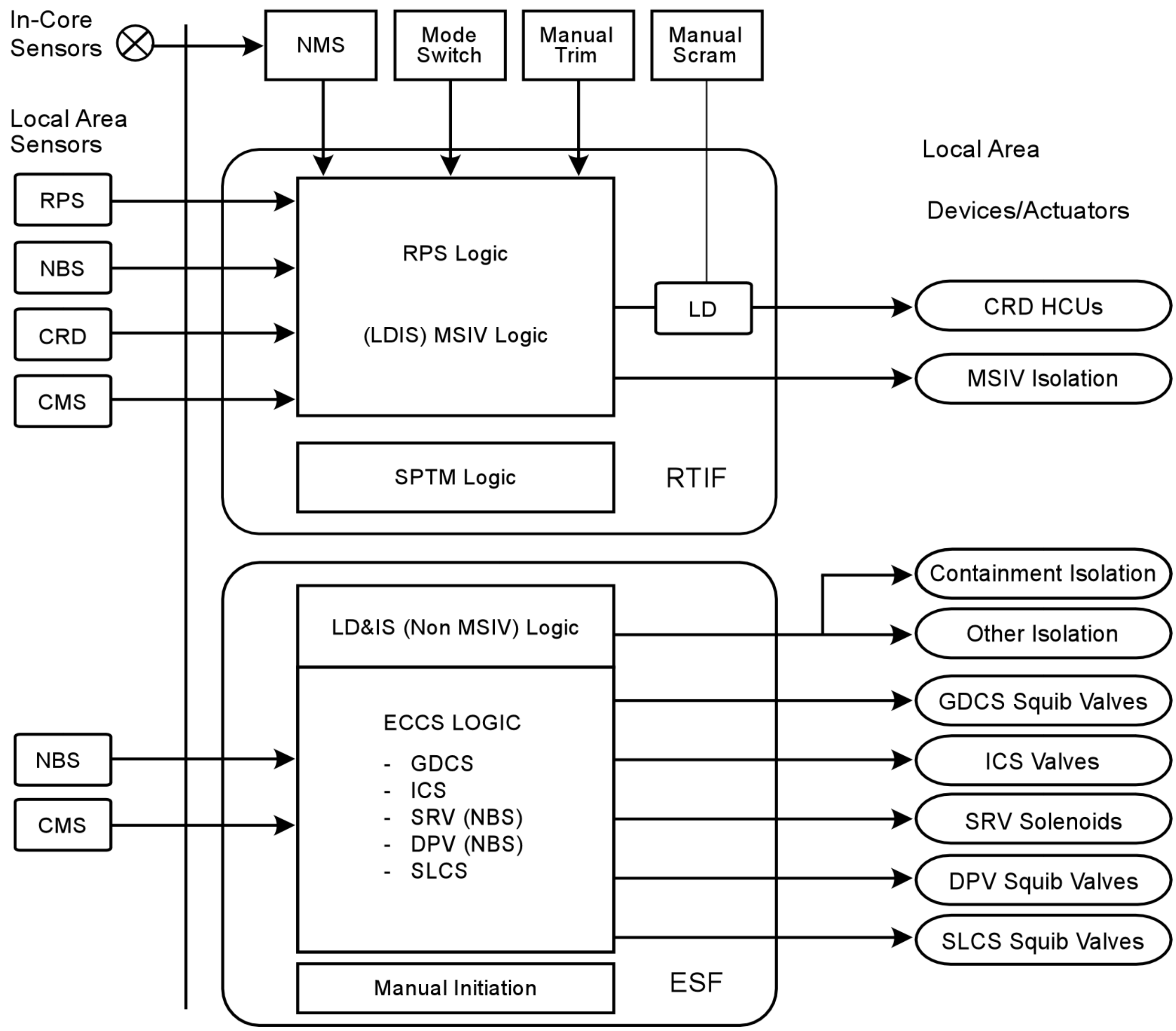


Figure 2.2.13-1. Safety System Logic and Control System Block Diagram – ESF Portion



Note: 1) 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, suppression pool temperature
2) Manual Scram interrupts power to the circuit
3) LD&IS resides in SSLC and shares sensor inputs with RTIF (for MSIV isolation) and ESF

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

2.2.14 Diverse Instrumentation and Controls

Design Description

The ATWS mitigation logic is implemented with safety-related and nonsafety-related portions. The safety-related ATWS/SLC mitigation logic, initiates liquid boron injection for emergency shutdown and initiates feedwater runback. The Automatic Depressurization System is inhibited on conditions indicative of an ATWS.

The ATWS mitigation logic, which initiates alternate rod insertion (ARI), is implemented in the nonsafety-related DPS. The DPS instrumentation and controls consist of the following major 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 safety-related 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 safety-related SSLC/ESF systems.
- Alternate rod insertion (ARI) and associated logics (e.g., fine motion control rod run in) to scram the plant. The ARI logic is part of the Anticipated Transient Without Scram (ATWS) mitigation function.
- Select Rod Insert (SRI) to hydraulically scram selected control rods on transients, which require Selected Control Rod Run-in (SCRRI) with power remaining elevated.
- Feedwater runback logic in addition to the ATWS/SLC logic mitigate certain ATWS events.

Manual controls are available to initiate the various diverse I&C functions. Monitoring and indication is available in the main control room to support manual operations. 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 SRM on SECY 93-087 and BTP HICB 19. The following scram signals are included in the DPS instrumentation and controls:

- High Reactor Pressure;
- High Reactor Water Level (L8);
- Low Reactor Water Level (L3);
- High Drywell Pressure;

- High Suppression Pool Temperature; and
- Closure of the MSIVs.

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-4 sensor voting confirmed by any 2-out-of-3 redundant processors.

Backup of ESF Functions:

To provide adequate diverse vessel depressurization and core cooling functions, the DPS includes initiation logic for the Automatic Depressurization system (ADS) which uses safety relief valves (SRVs) and depressurization valves (DPVs), Gravity Driven Cooling System (GDCS), Isolation Condenser System (ICS) and the Standby Liquid Control (SLC) system that is diverse from the primary ESF function logic.

The DPS also performs the following major isolations:

- Closure of the MSIVs on detection of high steam flow, low reactor pressure, or low reactor level;
- Closure of the ICS isolation valves on high steam flow or excessive condensate flow;
- Closure of the RWCU/SDC isolation valves o high differential flow; and
- Isolation of the Feedwater System on a feedwater line break inside containment on high differential pressure between feedwater lines coincident with high drywell pressure

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 SRM on SECY 93-087 and 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-4 sensor voting confirmed by any 2-out-of-3 redundant processors. Both series output load drivers are required to operate to cause an actuation.

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 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 mitigation initiation signal.
- A signal to the Rod Control and Information System (RCIS) to initiate electrical insertion of all control rods, via Fine Motion Control Rod Drives (FMCRDs), Run-In 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 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 basic configuration is as shown in Figure 2.2.14-1 and functional requirements are as described in Subsection 2.2.14.	1. Inspection and/or tests will be conducted on the as-built system configuration as shown in Figure 2.2.14-1 and functional requirements as described in Subsection 2.2.14.	1. Report(s) exist(s) and conclude(s) that the system conforms to the basic configuration shown in Figure 2.2.14-1 and functional requirements as described in Subsection 2.2.14.
2. The diverse I&C systems (i.e., ATWS mitigation logic and the DPS) are designed with signal initiation as follows: a. The diverse instrumentation and controls logic uses redundant instrument channels to monitor each parameter. A diverse reactor scram signal is initiated when any two channels have tripped. A diverse ESF actuation signal is initiated when any two channels have tripped. An ATWS/SLC initiation is processed when any two channels have tripped. An ARI/FMCRD Run-In signal is initiated when any two channels have tripped. An SRI signal is initiated when any two channels have tripped.	2. Tests will be performed to using simulated signals to verify that the diverse I&C systems (i.e., ATWS mitigation logic and the DPS) are designed with signal initiation as follows: a. The instrument channels of each diverse instrumentation and control function shall be tested using simulated signal inputs.	2. Report(s) exist(s) and conclude(s) that the diverse I&C systems (i.e., ATWS mitigation logic and the DPS) are designed with signal initiation as follows: a. The diverse I&C logic uses 4 redundant instrument channels to monitor each parameter and: (1) A diverse reactor scram signal is initiated when any 2-out-of-4 channels have tripped. (2) A diverse ESF actuation signal is initiated when any 2-out-of-4 channels have tripped. (3) A simulated SLC actuation signal is initiated when any 2-out-of-4 ATWS/SLC channels have tripped. (3) an ARI/FMCRD Run-In signal is initiated when any 2-out-of-4 channels have tripped. (4) an SRI signal is initiated on any 2-out-of-3 channels have tripped. (5) a feedwater

Table 2.2.14-1

ITAAC For Diverse Instrumentation and Controls

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
Feedwater runback is initiate when any two channels have tripped. b. Operation of redundant series output load drivers is required to cause a DPS actuation for ESF functions.	b. The DPS logic for diverse ESF actuation requires operation of series load drivers to cause actuation of end devices.	runback signal is initiated when any 2 channels have tripped. b. Each DPS diverse ESF function is actuated only when the DPS logic energizes the respective series load drivers.
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. Report(s) exist(s) and conclude(s) that the diverse instrumentation and Each trip signal is initiated when associated manual controls have been operated.

Table 2.2.14-1

ITAAC For Diverse Instrumentation and Controls

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment.	4. a. Test(s) will be performed to verify the electrical independence of each safety-related division. b. An inspection will be performed to verify the physical independence of the as-installed safety-related divisions and the nonsafety-related equipment. c. Test(s) will be performed to verify communication independence on each redundant network.	4. a. Report(s) exist(s) and conclude(s) that electrical independence is provided between safety related divisions and between safety-related divisions and nonsafety-related equipment. b. Report(s) exist(s) and conclude(s) that physical independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment. Interconnections among divisions and outputs to nonsafety-related systems use an isolating transmission medium. c. Report(s) exist(s) and conclude(s) that communication independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment.
5. Nonsafety-related DPS equipment in each division is powered from the nonsafety-related, multi load group plant power sources.	5. Tests will be performed to verify that nonsafety-related DPS equipment in each division is powered from the nonsafety-related, multi load group plant power sources.	5. Report(s) exist(s) and conclude(s) that nonsafety-related DPS equipment in each division is powered from the nonsafety-related, multi load group plant power sources.

Table 2.2.14-1

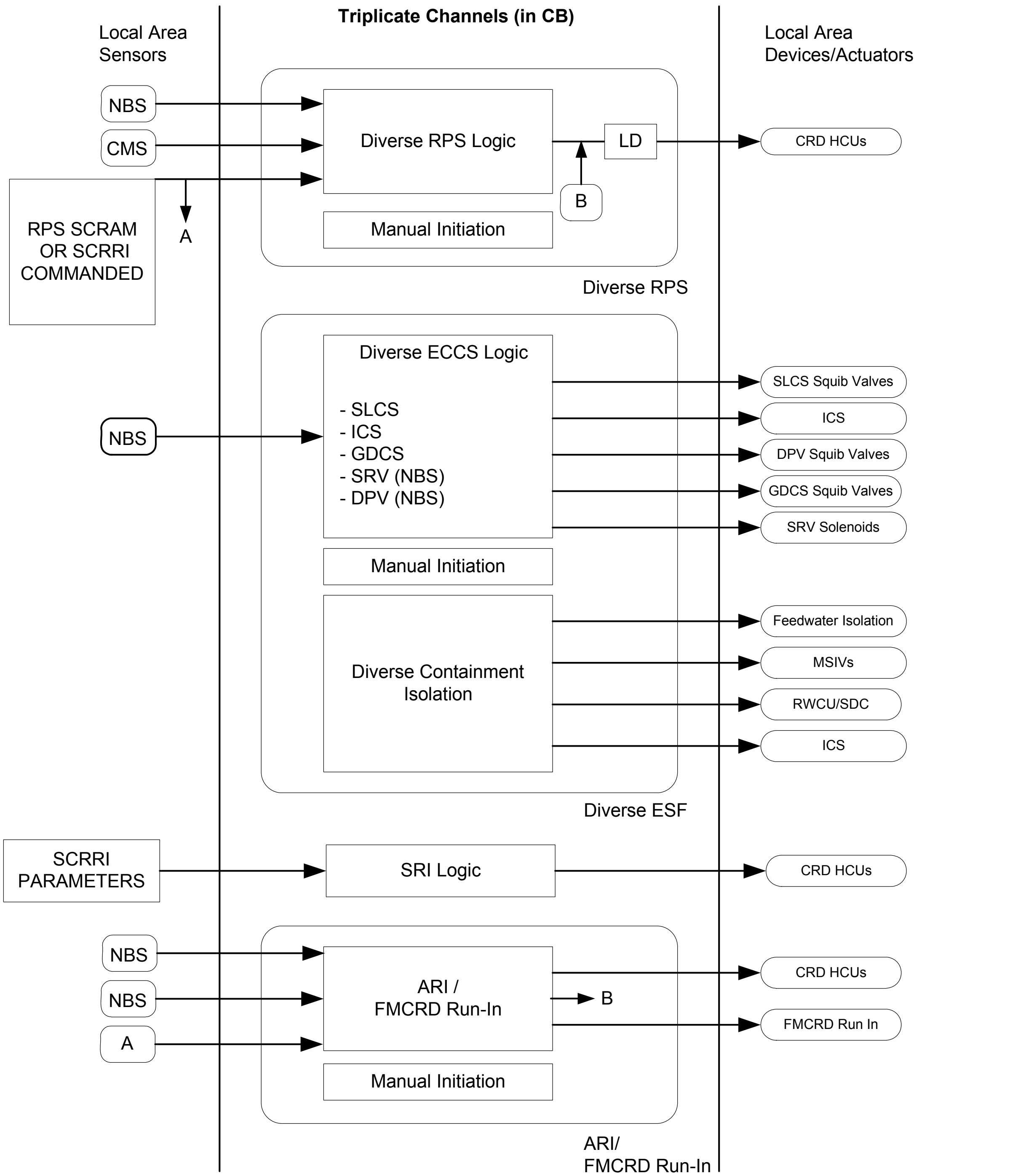
ITAAC For Diverse Instrumentation and Controls

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Control room alarms, indications and/or controls are provided as defined in Subsection 2.2.14.	6. Inspections will be performed on the control room alarms, indications and/or controls for the diverse I&C systems (ATWS and DPS).	6. Report(s) exist(s) and conclude(s) that all alarms, indications and/or controls are present or can be retrieved in the control room as defined in Subsection 2.2.14.
7. The diverse I&C systems (ATWS and DPS) are designed to ensure that, safety related system setpoints are defined, determined and implemented based on the setpoint methodology approved by NRC. This setpoint methodology includes the following as a minimum: The allowance for uncertainties between the process analytical limit and the device setpoint determined using a documented NRC approved setpoint methodology.	7. Inspection, tests, and/or analysis will be performed to verify that all the safety-related setpoints of instruments associated with the diverse I&C systems (ATWS and DPS) are in conformance with the design requirements.	7. Report(s) exist(s) and conclude(s) that the diverse I&C systems (ATWS and DPS) are designed to ensure that, safety related system setpoints are defined, determined and implemented based on the setpoint methodology approved by NRC. This setpoint methodology includes the following as a minimum: The allowance for uncertainties between the process analytical limit and the device setpoint determined using an NRC approved setpoint methodology.
8. Confirmatory analyses to support and validate the DPS design scope.	8. Analyses will be performed to validate the DPS design based on the assumptions used in LTR NEDO-33251. The DPS design will be validated using the radiological release acceptance criteria outlined in BTP HICB-19 have been satisfied.	8. Report(s) exist(s) and conclude(s) that the DPS design ensures releases during a common mode protection system failure coincident with the design basis events discussed in the Safety Analyses are within 10 CFR 100 limits (or percentage thereof) as specified in BTP HICB-19.

Table 2.2.14-1

ITAAC For Diverse Instrumentation and Controls

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. Failure Modes and Effects Analysis (FMEA) per NUREG/CR-6303 of safety-related protection system platforms (RPS and SSLC/ESF) completed to validate the DPS diverse protection function.	9. Complete FMEA per NUREG/CR-6303 to validate the DPS protection functions described in LTR NEDO-33251.	9. Report(s) exist(s) and conclude(s) that the completed FMEA (which address NUREG/CR-6303 Type 1-3 failures) for the RPS and SSLC/ESF safety-related platforms have been addressed in the DPS design scope.



NOTE:
LOCAL AREA SENSORS FOR CONTAINMENT ISOLATION FUNCTIONS NOT SHOWN.

Figure 2.2.14-1. Simplified Diverse Logic and Controls Basic Configuration 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 indication 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, and Signal Conditioning Units (SCU). The PRMS consists of independent subsystems, each of which contains between one and sixteen 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 indication, 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.

The SCUs for all PRMS subsystems are provided with indications for alarms and radiation levels.

The following PRMS provides instrumentation for radiological monitoring 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 eight items, which are safety-related, provide safety related functions. The remaining PRMS subsystems provide nonsafety-related functions.

- The <u>Reactor Building HVAC System (RBVS) Exhaust RMS</u> is safety-related and has 4 channels. It continuously monitors the gross gamma quantity of radioactivity being exhausted from the contaminated area served by Reactor Building Contaminated Area (HVAC) Subsystem. 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 RBVS dampers are closed, and exhaust fans are stopped.
- The <u>Control Building Air Intake HVAC RMS</u> is safety-related. It consists of eight channels. Four divisionalized 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 Exhaust RMS is safety-related. It 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 HVAC Exhaust RMS is safety-related. It 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 (RBVS) dampers are closed, and exhaust fans are tripped off.
- The Fuel Building General Area HVAC RMS is safety-related, and has 4 channels. It monitors the gross gamma radiation level in the Fuel Building HVAC exhaust duct for the general area. In the event of an abnormal radioactivity release, Fuel Building HVAC exhaust dampers are closed, and fans are stopped.
- The Fuel Building Fuel Pool HVAC RMS is safety-related. It consists of four channels that monitor the gamma radiation level of the air exiting the spent fuel pool and equipment areas. In the event of radioactive releases due to an accident while handling spent fuel, Fuel Building HVAC exhaust dampers are closed, and fans are stopped.
- The Drywell Sump LCW/HCW Discharge RMS is safety-related. It 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 is safety-related. It 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 RBVS isolation dampers, the RB HVAC exhaust fans are stopped.
- The Main Steamline (MSL) RMS is nonsafety-related, and has 4 channels are 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.
- The Offgas Pre-Treatment RMS is nonsafety-related and 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 is nonsafety-related. It 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. One skid contains provisions for continuous gaseous, particulate and halogen radioactivity monitoring of the offgas post treatment process. The second skid contains only provisions for continuous gaseous monitoring. 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, is nonsafety-related. It monitors the radioactivity exhausting in the ventilation air from the charcoal vault.
- The Turbine Building Normal Ventilation Air HVAC RMS is nonsafety-related. It 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 is nonsafety-related. It consists of two non-divisional channels that continuously monitor the air in the compartment area HVAC in the Turbine Building for gross radiation levels.
- The Turbine Building Combined Ventilation Exhaust RMS is nonsafety-related. It monitors the Turbine Building Combined Ventilation exhaust for halogens, particulates and noble gas releases during normal and accident conditions.
- The Main Turbine Gland Seal Steam Condenser Exhaust RMS is nonsafety-related. It is continuously monitors the gland seal steam offgas, discharged into the Turbine Building Ventilation System, for radioactive noble gases.
- The Radwaste Building Ventilation Exhaust RMS is nonsafety-related. It continuously monitors halogens, particulates and noble gas releases from the Radwaste Building vent to the atmosphere for both normal and accident conditions.
- The Liquid Radwaste Discharge RMS, consisting of a single channel, is nonsafety-related. It 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 is nonsafety-related. It consists of two channels that monitor the drywell air space radiation levels for leakage detection. One channel continuously monitors radioactive particulates, while the other channel monitors noble gases. The subsystem is utilized to detect reactor coolant leakage.
- The Reactor Component Cooling Water (RCCW) Intersystem Leakage RMS is nonsafety-related. It consists of two channels. These channels monitor for gross radiation levels that are indicative of leakage through the heat exchangers in the RCCW system.
- The Technical Support Center HVAC Air Intake RMS is a single channel nonsafety-related radiation monitor continuously monitoring 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 Combined Ventilation Exhaust RMS is nonsafety-related. It 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 Plant Stack RMS is nonsafety-related. It 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.

Figure 2.3.1-1 in conjunction with Table2.3.1-1 shows the locations of the PRMS monitors.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.3.1-2 provides a definition of the inspections, tests and/or analyses, together with the associated acceptance criteria for the Process Radiation Monitoring System.

Table 2.3.1-1

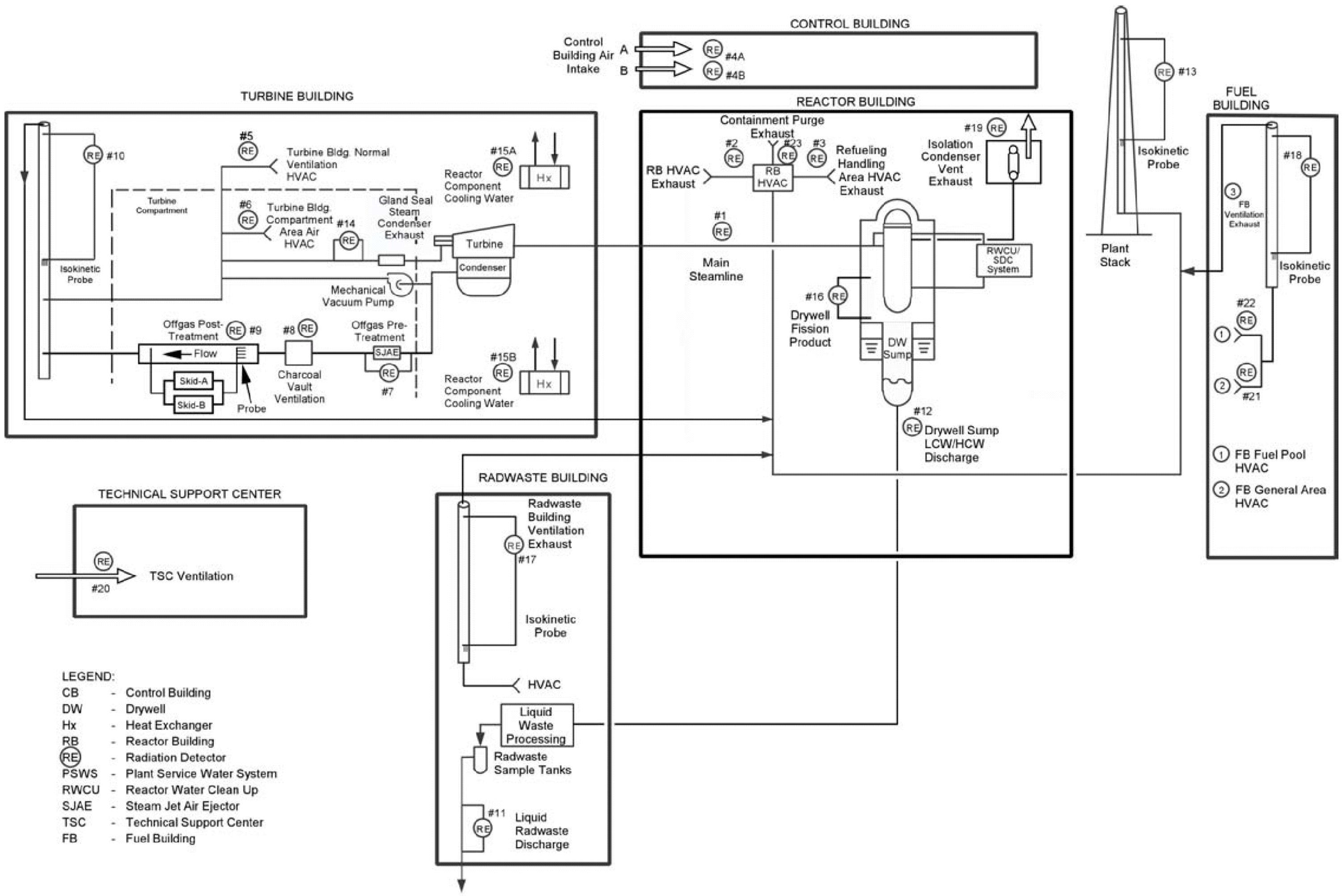
Key to Radiation Monitors shown on Figure 2.3.1-1

ID on Figure 2.3.1-1	Description
1	MSL
2	Reactor Building HVAC Exhaust
3	Refuel Handling Area HVAC Exhaust
4A, 4B	Control Building Air Intake HVAC
5	TB Normal Ventilation Air HVAC
6	TB Compartment Area Air HVAC
7	Offgas Pre-treatment
8	Charcoal Vault Ventilation
9A, 9B	Offgas Post-treatment
10	TB Combined Ventilation Exhaust
11	Liquid Radwaste Discharge
12	Drywell Sump LCW/HCW Discharge
13	Plant Stack
14	Main Turbine Gland Seal Steam Condenser Exhaust
15A, 15B	Reactor Component Cooling Water Intersystem Leakage
16	Drywell Fission Product
17	Radwaste Building Ventilation Exhaust
18	FB Combined Ventilation Exhaust
19	Isolation Condenser Vent Exhaust
20	TSC HVAC Air Intake
21	FB General Area HVAC
22	FB Fuel Pool HVAC
23	Containment Purge Exhaust

Table 2.3.1-2

ITAAC For The Process Radiation Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Figure 2.3.1-1 in conjunction with Table 2.3.1-1 shows the locations of the PRMS subsystems. Subsection 2.3.1 describes the configuration and functional requirements of the subsystems.	1. Inspections shall be conducted on each as-built PRMS subsystem as shown in Figure 2.3.1-1 in conjunction with Table 2.3.1-1, and as described in Subsection 2.3.1.	1. PRMS subsystems conform to the basic configuration shown in Figure 2.3.1-1 in conjunction with Table 2.3.1-1, and as described in Subsection 2.3.1.
2. The safety-related PRMS subsystems as identified in Subsection 2.3.1 are powered from uninterruptible safety-related power sources.	2. Inspections will be conducted to confirm that the PRMS safety-related subsystems described in Subsection 2.3.1 are powered from uninterruptible safety-related power sources.	2. The safety-related PRMS subsystems described in Subsection 2.3.1 receive electrical power from uninterruptible safety-related buses.
3. The nonsafety-related PRMS subsystems as identified in Subsection 2.3.1 are powered from uninterruptible nonsafety-related power sources.	3. Inspections will be conducted to confirm that the PRMS nonsafety-related subsystems described in Subsection 2.3.1 are powered from uninterruptible nonsafety-related power sources.	3. The safety-related PRMS subsystems described in Subsection 2.3.1 receive electrical power from uninterruptible nonsafety-related buses.
4. PRMS subsystems provide the following: a. Indications in MCR for radiation levels b. Indications on SCUs for radiation levels c. Alarms in MCR on radiation level exceeding setpoint d. Indications on SCUs on radiation level exceeding setpoint	4. Tests will be conducted by simulating a high radiation signal that exceeds a setpoint value that is preset for the testing. Inspections will be conducted to confirm that indication and alarm requirements are met as listed in the Design Commitment.	4. Indication and alarm requirements are met as listed in the Design Commitment.



Note: See Table 2.3.1-1 for radiation detectors numbers.

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. ARM locations are shown in Table 2.3.2-1.

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 indication 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 nonsafety-related 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 2.3.2-2 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria for the Area Radiation Monitoring system.

Table 2.3.2-1
ARM Locations

Area	Description & Location
Reactor Building	Refueling Floor Area #1, EL 34000
Reactor Building	Refueling Floor Area # 2, El 34000
Reactor Building	New Fuel Storage Pool, EL 27000
Reactor Building	New Fuel Storage Pool, EL 27000
Reactor Building	RWCU/SDC Pump, EL -11500
Reactor Building	RB Sump Pumps, EL -11500
Reactor Building	RWCU/SDC Train A Heat Exchanger EL -11500
Reactor Building	RWCU/SDC Train B Heat Exchanger EL -11500
Reactor Building	Equipment Hatch Pathway, EL -6400
Reactor Building	Personnel Hatch Pathway, EL -6400
Reactor Building	FMCRD HCU Area #1, EL -6400
Reactor Building	FMCRD HCU Area # 3, EL -6400
Reactor Building	RWCU/SDC Filter Demineralizer Area (Near Equip. Hatch), EL -1000
Reactor Building	Radiological Control Area Entrance, EL 17500
Reactor Building	Hydrogen/Oxygen Monitoring (CMS), Skid EL 13570
Reactor Building	Hydrogen/Oxygen Monitoring (CMS), Skid EL 13570
Reactor Building	Instrument Rack Area #1, EL -11500
Reactor Building	Instrument Rack Area #2, EL -11500
Reactor Building	Instrument Rack Area #3, EL -11500
Reactor Building	Instrument Rack Area #4, EL -11500
Reactor Building	Instrument Rack Area #5, EL -11500
Reactor Building	Instrument Rack Area #6, EL -11500
Reactor Building	Instrument Rack Area #7, EL -11500
Reactor Building	Instrument Rack Area #8, EL -11500
Reactor Building	Fuel Transfer System (FTS) Maintenance Room (Multiple), EL 17500
Reactor Building	Fuel Handling Machine (IFTS), EL 34000
Reactor Building	Remote Shutdown Panel A Area, El. –1000
Reactor Building	Remote Shutdown Panel B Area, El. –1000
Fuel Building	Spent Fuel Floor, EL 4650
Fuel Building	Fuel Handling Machine, EL 4650

Table 2.3.2-1

ARM Locations

Area	Description & Location
Fuel Building	Fuel Transfer Cask Area, EL 4650
Fuel Building	FAPCS Heat Exchangers, EL -11500
Fuel Building	FAPCS System Transfer Pumps, EL -11500
Fuel Building	Sump Pumps, EL -11500 H
Fuel Building	Ground Grade Access Pathway, EL 4650
Fuel Building	Wash Down Bay Entry Door, EL 4650 (Truck)
Fuel Building	Fuel Transfer System (FTS) Maintenance Rooms (Multiple) EL 4650
Radwaste Building	Electrical Board Room El -9350
Radwaste Building	Control Room
Radwaste Building	High Activity Resin Recirculation Pump Room, El –9350
Radwaste Building	High Activity Resin Transfer Pump Room, El –2350
Radwaste Building	Trailer Access Area, El 4650
Radwaste Building	Liquid Radioactive Waste Treatment Area (Hallover Fiber deep-Bed Demineralizer, Reverse Osmosis System, etc.) El 4650
Radwaste Building	Wet Solid Radioactive Waste Treatment Area (Dewatering Equipment, Concentrate Treatment System, etc.) EL4650
Radwaste Building	Dry Solid Waste Treatment Area (High Dose Rate Waste Storage Area, etc.) El 4650
Radwaste Building	Packaged Waste Staging Area, El 4650
Turbine Building	Main Condenser Floor Area EL -1400
Turbine Building	Drain Cooler Area EL 4650
Turbine Building	Offgas Sampling Area EL 4650
Turbine Building	Condensate Pumps Area EL -1400
Turbine Building	Low Pressure Heater Area EL 20000
Turbine Building	Deaerator Area, EL 28000
Turbine Building	SRV/MSIV Maintenance Area EL 20000
Turbine Building	Steam Jet Air Ejector (SJAE) B Area EL 4650
Turbine Building	SJAE A Area EL 4650
Turbine Building	High Pressure Heater Area EL 20000
Turbine Building	Filters and Demineralizers Area EL 4650
Turbine Building	Turbine Operating Floor Area EL 28000
Turbine Building	Turbine Operating Floor Area EL 28000

Table 2.3.2-1
ARM Locations

Area	Description & Location
Turbine Building	Crane Travel Area (Various)
Turbine Building	Equipment Main Access Area, EL 4650
Turbine Building	RCCW System Area Entrance EL 4650
Turbine Building	Offgas Charcoal Adsorber Room Entrance Area EL -1400
Turbine Building	Backwash Transfer Pumps Entrance Area EL -1400
Turbine Building	Condensate Hollow Fiber Filter Valve Room EL -1400
Turbine Building	Sample Room Area EL -1400
Turbine Building	Filters and Demineralizers Area EL 4650
Turbine Building	Turbine Operating Floor Area EL 28000
Turbine Building	Turbine Operating Floor Area EL 28000
Turbine Building	Crane Travel Area (Various)
Turbine Building	Equipment Main Access Area, EL 4650
Turbine Building	RCCW System Area Entrance EL 4650
Turbine Building	Offgas Charcoal Adsorber Room Entrance Area EL -1400
Turbine Building	Backwash Transfer Pumps Entrance Area EL -1400
Turbine Building	Condensate Hollow Fiber Filter Valve Room EL -1400
Turbine Building	Sample Room Area EL -1400
Turbine Building	Condensate D/B Demineralizer Entrance Area, EL 4650
Turbine Building	Offgas Hydrogen Recombiner A, EL 12000
Turbine Building	Offgas Hydrogen Recombiner B, EL 4650
Turbine Building	Instrument Air Compressor Area, EL 12000
Turbine Building	MCC Water Chiller Room A, EL 28000
Turbine Building	MCC Water Chiller Room B, EL 28000
Turbine Building	Turbine Building Exhaust Duct Area EL 33000
Turbine Building	RCCWS Area Entrance EL 4650
Control Building	Main Control Room, EL -1000

Table 2.3.2-2

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 and Table 2.3.2-1.	1. Inspection of the as-built system will be conducted.	1. Inspections confirm that the as-built ARM system conforms to the description in Subsection 2.3.2 and Table 2.3.2-1.
2. Each ARM channel listed in Table 2.3.2-1 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 a simulated high radiation level signal to verify that the MCR alarm and local alarm (if provided) is on when the simulated signal exceeds a preset setpoint.	2. Tests confirm that the MCR alarm and local audible alarm (if provided) are initiated when the simulated radiation level exceeds a preset limit.
3. Each ARM channel listed in Table 2.3.2-1 is provided with indication of radiation level.	3. Tests will be conducted using a simulated high radiation signal to verify that the indications for each ARM channel responds to the simulated high radiation signal.	3. Inspections confirm that the indications for each ARM channel responds to the simulated high radiation signal.

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 initiates automatically on any of the following:

- RPV high pressure following a time delay
- RPV water level below level 2 following a time delay
- RPV water level below level 1
- Loss of power to 2 of 4 reactor feed pumps with the reactor mode switch in RUN
- 2 of 4 MSIVs not fully open with the reactor mode switch in RUN.

To start an IC into operation, a condensate return valve and condensate return bypass valve are 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 Q-DCIS power is lost.

An in-line vessel is located on the condensate return line, downstream of the nitrogen motor operated valve. The in-line vessel is located on each ICS train to provide the additional condensate volume for the RPV.

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 Dryer/Separator pool and Reactor Well shall be designed to have sufficient water volume to provide makeup water to the IC/PCC pools for the initial 72 hours of a LOCA.

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)
- 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 manual switches that actuate the isolation valves and the condensate return valves. The opening and closure of the valves is verified by their status indicators.

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 pressure within the reactor coolant pressure boundary following any abnormal event that results in containment isolation.
- 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 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 inspection reports of the as-built ICS conforms to the basic configuration shown in Figure 2.4.1-1.
a. 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.	a. Inspections of the as-built 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.	a. The design reports of the as-built 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.
b. The IC/PCC pools are safety-related and Seismic Category I.	b. Inspections of the as-built portions of the IC/PCC pools that they are safety-related and Seismic Category I.	b. The design reports of the as-built IC/PCC pools confirm that they are safety-related and Seismic Category I.
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.	2. Inspection will be performed to confirm that a flow limiter is included in the branch line.	2. Inspection report(s) document that 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 ICS 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 to the requirements of the ASME Code, Section III.

Table 2.4.1-1

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Each safety-related isolation valve, shown on Figure 2.4.1-1, closes on an isolation signal.	4. Opening and/or closing tests of valves will be conducted under pre-operational fluid flow.	4. Test reports document that each safety-related isolation valve, closes on an isolation signal.
5. Each condensate return valve shown on Figure 2.4.1-1 will open to initiate the ICS.	5. Opening and/or closing tests of valves will be conducted under pre-operational differential pressure, fluid flow and temperature conditions.	5. Test reports document that each condensate return valve opens to initiate the IC system.
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. Test report(s) document that the ICS isolation valves close upon receipt 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. Test report(s) document that the ICS isolation valves close upon receipt of signals from the LD&IS.

Table 2.4.1-1

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Each ICS train normally closed condensate return valve opens upon receipt of the following automatic actuation signals: • RPV high pressure following a time delay • RPV water level below level 2 following a time delay • RPV water level below level 1 • Loss of power to 2 of 4 reactor feed pumps with the reactor mode switch in RUN • 2 of 4 MSIVs not fully open with the reactor mode switch in RUN.	8. Valve opening tests will be performed using simulated automatic initiation signals.	8. Test report(s) document that the condensate return valves open upon receipt of automatic initiation 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 the following automatic actuation signals: • RPV high pressure following a time delay • RPV water level below level 2 following a time delay • RPV water level below level 1 • Loss of power to 2 of 4 reactor feed pumps with the reactor mode switch in RUN • 2 of 4 MSIVs not fully open with the reactor mode switch in RUN.	9. Valve opening tests will be performed using simulated automatic initiation signals.	9. Test report(s) document that the condensate return valves open upon receipt of automatic initiation 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. Test report(s) document that the two-series, solenoid-operated vent line valves open on a simulated high RPV pressure signal after a time delay 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
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 tests will be performed that manually opens the vent valves during pre-operational testing following condensate return or condensate bypass valve opening signals.	11. Test report(s) document that 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.
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. Test report(s) document that 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 times with the drywell at atmospheric pressure, and an analysis or test that shows the 2 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. Safety-related loads for the ICS are powered from the correct safety-related Divisions.	14. Tests will be performed on the IC System by providing a test signal in only one safety-related Division at a time.	14. Test report(s) document that the test signal exists only in the specific safety-related Division.

Table 2.4.1-1

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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.
16. Each ICS train minimum heat removal capacity is 33.75 MWt with reactor above normal operating pressure.	16. Using prototype test data and as-built IC unit information, an analysis will be performed to establish the heat removal capacity of the IC unit.	16. Tests-analysis report(s) document that the ICS train unit heat removal capacity is ≥ 33.75 MWt for a reactor at above normal operating pressure.
17. The ICS 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 ICS provides at least [13.88 m^3 (490 ft^3)] of liquid available for return to the RPV.
18. The mechanical portion of each division of the safety-related ICS instrumentation located in the Reactor Building is physically separated from the other divisions.	18. Inspection of the as-built ICS instrumentation will be conducted.	18. Test report(s) document the mechanical portion of each ICS instrumentation division is physically separated from the other divisions by structural and/or fire barriers.
19. The divisional equipment in the ICS is powered from its respective safety-related divisions. In the ICS, independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	19. a. Tests will be performed in the ICS by providing a test signal in only one safety-related division at a time. b. Inspection of the as-installed safety-related divisions in the ICS will be performed.	19. a. The test signal exists only in the safety-related division under test in the ICS. b. Physical separation or electrical isolation exists between safety-related divisions in the ICS. Physical separation or electrical isolation exists between safety-related divisions and nonsafety-related equipment.

Table 2.4.1-1

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
20. The Dryer/Separator Pool will provide sufficient makeup water volume to the IC/PCC expansion pool on a low water signal in the initial 72 hours of a LOCA.	20. a. A valve-opening test will be performed using simulated low-level water signal from the IC/PCC expansion pool. b. An analysis will be performed to demonstrate the Dryer/Separator Pool will provide sufficient makeup water volume to the IC/PCC expansion pool on a low water signal in the initial 72 hours of a LOCA.	20. a. Test report(s) document that the two-series, valves open on a simulated low-level water signal from the IC/PCC expansion pool. b. Analyzed Dryer/Separator Pool has sufficient makeup water volume to the IC/PCC expansion pool on a low water signal in the initial 72 hours of a LOCA.

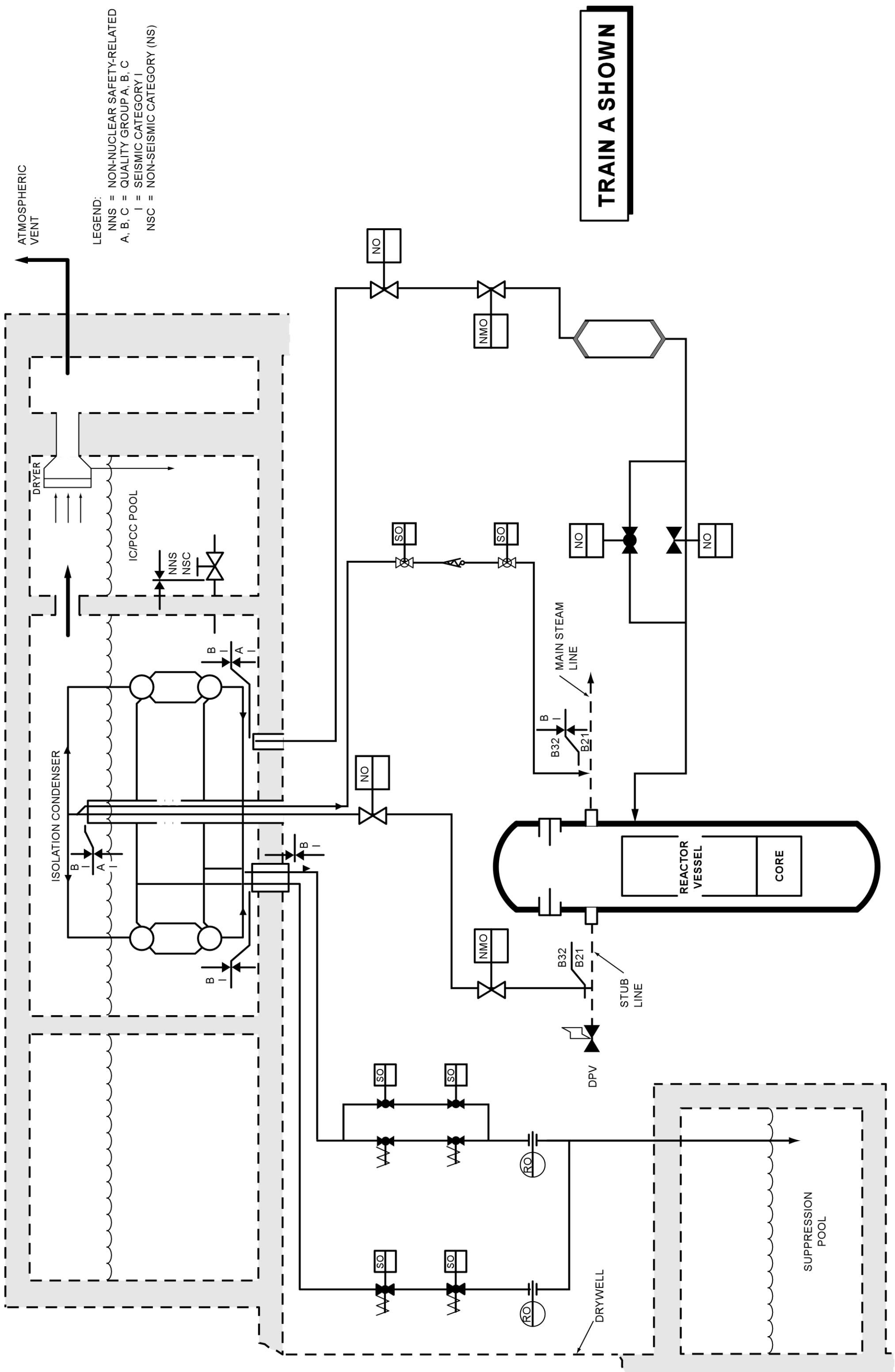


Figure 2.4.1-1. Isolation Condenser System Schematic

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) located within containment 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.

The GDCS is an engineered safety feature (ESF) system. It is classified as safety-related and Seismic Category I. GDCS instrumentation and power supply are safety-related. All GDCS safety-related components are qualified to withstand the harsh environments postulated for design basis accidents.

Basic system features and functions 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. 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: B
- 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 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 of the GDCS is as shown on Figure 2.4.2-1.	1. Inspections of the as-built system will be conducted.	1. The as-built GDCS conforms to the basic configuration described in Subsection 2.4.2 and shown in Figure 2.4.2-1.
2. a. The GDCS injections lines will provide sufficient flow to maintain water coverage one meter above TAF for 72 hours following a design basis LOCA. b. The GDCS equalizing lines will provide sufficient flow to maintain water coverage one meter above TAF for 72 hours following a 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 injection 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 GDCS 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 will open as designed.	3. A vendor type test will be performed on a squib valve to open as designed.	3. Records of vendor type test will conclude GDCS squib valves used in the injection and equalization will open as designed.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Check valves designated in Figure 2.4.2-1 as having an active safety-related function open, close, or both open and also close under system pressure, fluid flow, and temperature conditions.	4. Type tests of valves for opening, closing, or both opening and also closing, will be conducted.	4. Based on the direction of the differential pressure across the valve, each check valve opens, closes, or both opens and closes, depending upon the valve's safety functions.
5. (Deleted)		
6. (Deleted)		
7. Control Room indications and controls are provided for the GDCS.	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 leak tightness 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 ft^2).	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).
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 to the requirements in the ASME Code, Section III.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. Portions of the GDCS are designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III.	12. Data, analysis and inspection reports will be inspected for the ASME components.	12. Inspections confirm that the ASME Code components are designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III.
13. Each of the GDCS divisions is powered from their respective safety-related power divisions.	13. Tests will be performed on the GDCS by providing a test signal in only one safety-related power division at a time.	13. The test signal exists only in the safety-related power division under test in the GDCS.
14. Each mechanical division of the GDCS is physically separated from the other divisions with the exception of divisions B and C connected to pool B/C as shown in Figure 2.4.2-1.	14. Inspections of the as-built GDCS will be performed.	14. Each mechanical division of the GDCS is physically separated from other mechanical divisions of the GDCS by structural and /or fire barriers with the exception of divisions B and C connected to pool B/C as shown in Figure 2.4.2-1.
15. The combined minimum drainable water volume for GDCS pools A, B/C, and D is 1661 m^3 (58658 ft^3).	15. An analysis of combined minimum drainable volume for GDCS pools A, B/C, and D will be performed.	15. Analysis will show the combined minimum drainable water volume for GDCS pools A, B/C, and D is 1661 m^3 (58658 ft^3).
16. The minimum water level in GDCS pools A, B/C, and D is 6.5 m (11.49 ft).	16. An analysis of minimum water level in GDCS pools A, B/C, and D will be performed.	16. Analysis will show the minimum water level in GDCS pools A, B/C, and D is 6.5 m (11.49 ft).
17. The minimum elevation change between minimum water level of GDCS pools and the centerline of GDCS injection line nozzles is 13.5 m (44.3 ft).	17. An analysis of minimum elevation change between minimum water level of GDCS pools and the centerline of GDCS injection line nozzles will be performed.	17. Analysis will show the minimum elevation change between minimum water level of GDCS pools and the centerline of GDCS injection line nozzles is 13.5 m (44.3 ft).

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
18. The minimum drainable volume from the suppression pool to the RPV is 799 m^3 (28,216 ft^3).	18. An analysis of minimum drainable volume from the suppression pool to the RPV will be performed.	18. Analysis will show the minimum drainable volume from the suppression pool to the RPV is 799 m^3 (28,216 ft^3).
19. The minimum equalizing driving head is 1 meter (3.28 ft).	19. An analysis of the minimum equalizing driving head will be performed.	19. Analysis will show the minimum equalizing driving head is 1 meter (3.28 ft).

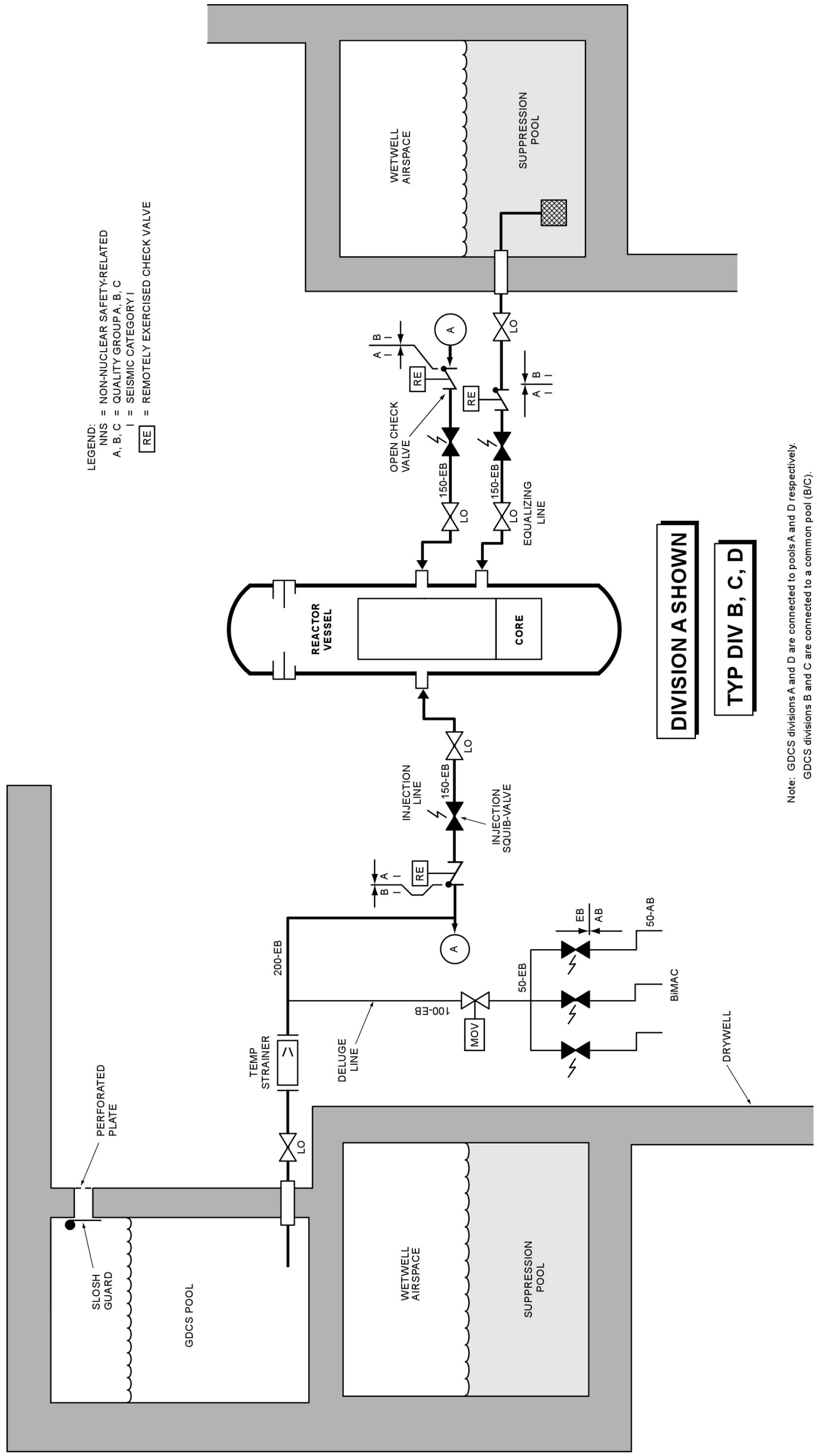


Figure 2.4.2-1. Gravity-Driven Cooling System

2.5 REACTOR SERVICING EQUIPMENT

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

2.5.1 Fuel Servicing Equipment

Design Description

This equipment does not perform or ensure any safety-related function, is not required to achieve or maintain safe shutdown, and is not subject to high regulatory oversight. Therefore the equipment is nonsafety-related and has no safety design basis.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this equipment.

2.5.2 Miscellaneous Servicing Equipment

Design Description

This equipment does not perform or ensure any safety-related function, is not required to achieve or maintain safe shutdown, and is not subject to high regulatory oversight. Therefore the equipment is nonsafety-related and has no safety design basis.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this equipment.

2.5.3 Reactor Pressure Vessel Servicing Equipment

Design Description

This equipment does not perform or ensure any safety-related function, is not required to achieve or maintain safe shutdown, and is not subject to high regulatory oversight. Therefore the equipment is nonsafety-related and has no safety design basis.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this equipment.

2.5.4 RPV Internals Servicing Equipment

Design Description

This equipment does not perform or ensure any safety-related function, is not required to achieve or maintain safe shutdown, and is not subject to high regulatory oversight. Therefore the equipment is nonsafety-related and has no safety design basis.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this equipment.

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 II.

As an operational aid, a position indicating system and travel limit computer are available 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 refueling machine may be manually operated, in the event that the position indicating system and travel limit computer is not available.

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 II.

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 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 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 shall be performed with actual or simulated signals to demonstrate that the interlocks function as required.	2. The tests have been completed and the results demonstrate that the required interlocks function as required.
3. The FB fuel handling machine has 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 fuel handling machine is provided with controls interlocks which: a. Maintain water shielding over fuel when grappled on mast. b. (Deleted) 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.	4. Test will be performed with actual or simulated signals to demonstrate that the interlocks function as required.	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 may initially be 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. Alternatively new fuel may be moved directly to the reactor building buffer pool fuel storage racks. New fuel storage in the reactor building buffer pool shall maintain sub-criticality in stainless steel racks without neutron absorbing material by maintaining the proper bundle to bundle spacing. 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 contain 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) at rack exit under normal conditions. Temperatures in excess of 100°C (212°F) are allowed during accident conditions. All stresses within the rack remain under allowable limits when temperatures are in excess of 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 for the new and spent fuel storage racks.

Table 2.5.6-1

ITAAC For The Fuel Storage Racks (Spent and New)

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 of to determine k_{eff} for full spent fuel pool storage racks.	1. Analysis records confirm that the maximum calculated $k_{eff} \leq 0.95$.
2. The maximum rack water coolant flow temperature at the rack exit shall be ≤ 100°C (212°F).	2. Calculations will be performed to determine the maximum temperature of the spent fuel racks.	2. Analysis records confirm that the maximum temperature in the spent fuel racks is < 100°C (212°F) at rack exit under normal operating conditions.
3. The maximum stresses in the racks do not exceed design allowable during accident conditions.	3. Calculations will be performed to determine the maximum temperature of the spent fuel racks and allowable stress under maximum rack temperature.	3. Analysis records confirm that the maximum stresses in the racks do not exceed design allowable during accident conditions.

2.5.7 Under-Vessel Servicing Equipment

Design Description

This equipment does not perform or ensure any safety-related function, is not required to achieve or maintain safe shutdown, and is not subject to high regulatory oversight. Therefore the equipment is nonsafety-related and has no safety design basis.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this equipment.

2.5.8 FMCRD Maintenance Area

Design Description

This area does not perform or ensure any safety-related function, is not required to achieve or maintain safe shutdown, and is not subject to high regulatory oversight. Therefore, the area is nonsafety-related and has no safety design basis.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this area.

2.5.9 Fuel Cask Cleaning

Design Description

This fuel cask cleaning does not perform or ensure any safety-related function, is not required to achieve or maintain safe shutdown, and is not subject to high regulatory oversight. Therefore, fuel cask cleaning is nonsafety-related and has no safety design basis.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for fuel cask cleaning.

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. There are no modes of normal or abnormal operation that will trap fuel assemblies without the ability to add water or prevent unconditional venting of pressure that may develop due to boiling.

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.

A procedure provides instructions to the IFTS operators on how to maintain the IFTS filled with water in the event (for any reason) the fuel transport cart with fuel loaded within the IFTS cannot be moved (i.e., fuel cannot be removed from the within the IFTS).

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 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 passes functional testing.
2. For personnel radiation protection purposes, the IFTS is designed to control access (as described in Subsection 2.5.10) to areas immediately adjacent to the IFTS.	2. Interlocks will be individually tested using simulated or actual signals.	2. The as-built IFTS passes functional testing of the radiation access interlocks prior to and during IFTS 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 as-built IFTS is equipped with a combination of physical controls and interlocks of the water tight barriers that prevent any combination of barriers from being opened allowing an uncontrolled drain path.	3. A report confirms that there are no creditable combinations of events that will allow uncontrolled draining during normal or accident conditions.
4. Operating procedures shall include an affective "stopped transport cart fuel cooling procedure" on how to maintain the IFTS filled with water in the event the fuel transport cart with fuel loaded within the IFTS cannot be moved.	4. Testing confirms that the use of "stopped transport cart fuel cooling procedure" can maintain the IFTS filled with water.	4. Test results show that use of the "stopped transport cart fuel cooling procedure" would maintain the IFTS filled with water.

2.5.11 (Deleted)

(This system has been deleted)

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 nonsafety-related primary functions:

- Purifies reactor coolant during normal operation and shutdown;
- Provides shutdown cooling to bring the reactor to cold shutdown, and removes core decay heat to maintain cold shutdown;
- 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; and
- 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 safety-related and are powered from safety-related buses. 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. Normally, one train is in operation while the other is in standby.

During shutdown cooling, the RPV water is cooled through the NRHX, purified by the demineralizer, 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. Containment isolation valves are automatically or manually actuated.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.1-1 provides the inspections, tests, and/or analyses that 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 to 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 analyses 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 Code Section III 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 Section III.	3. The results of the hydrostatic test of the ASME portions of the system conform to the requirements in the ASME Code, Section 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 safety-related buses.	6. A test of the power availability to safety-related components will be conducted with power supplied from the permanently installed electric power buses.	6. Safety-related components receive electrical power from safety-related buses only.

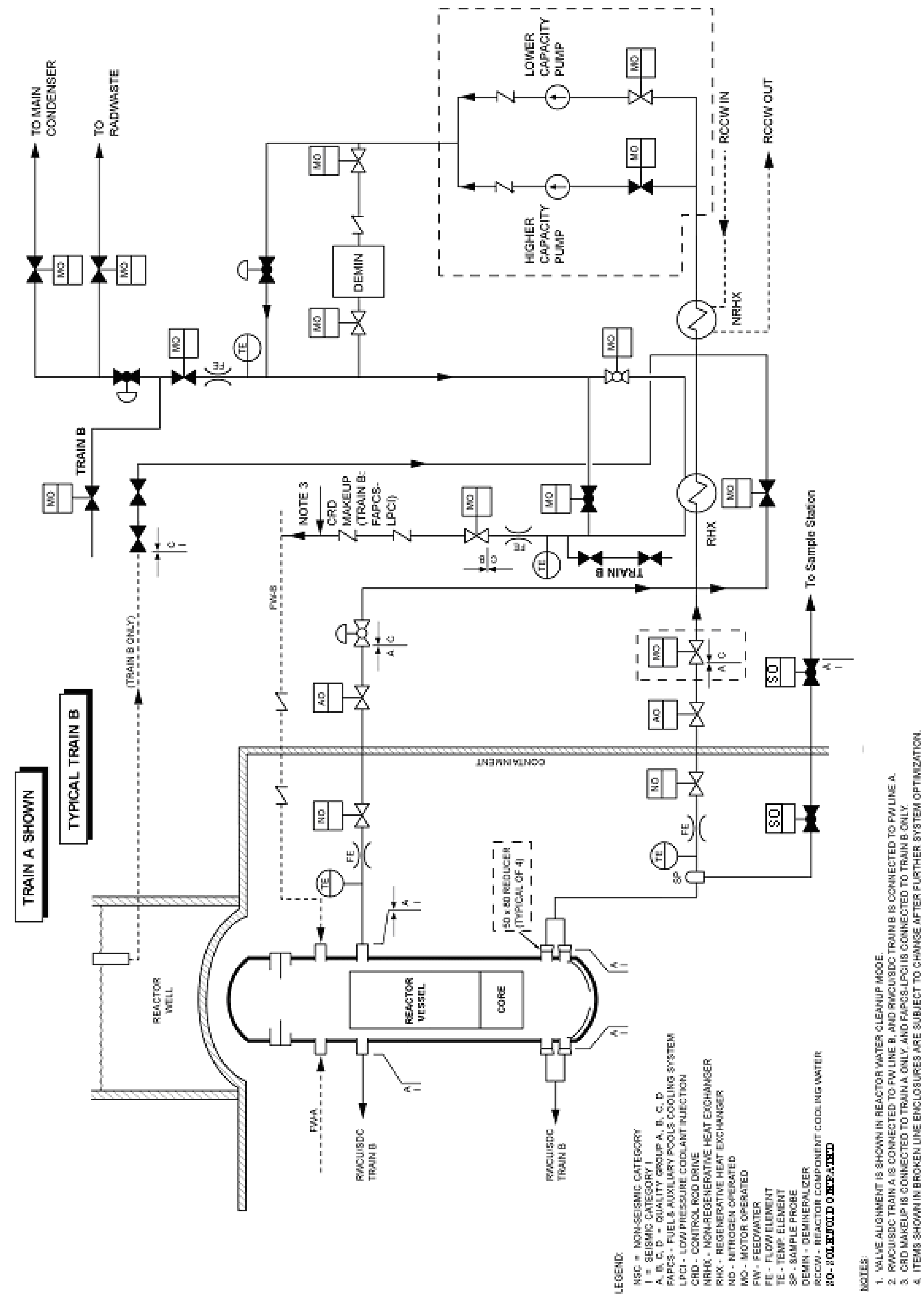


Figure 2.6.1-1. Reactor Water Cleanup/Shutdown Cooling System

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;
- 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 the Fire Protection System or from a separate offsite source following an accident; and
- An interface with the safety-related RWCU/SDC system piping used for low pressure coolant injection.

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 or II requirements.

The Fuel and Auxiliary Pools Cooling System (FAPCS) consists of two physically separated cooling and cleaning (C/C) trains, each with 100% capacity for normal operation. Both trains contain 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. A bypass line exists around the water treatment units in each train allowing them to be bypassed manually when desired. The water treatment units are bypassed automatically when a high temperature set point is exceeded downstream of the heat exchanger.

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 20 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. Pneumatic-operated valves with containment isolation function are designed to close within 30 seconds upon loss of its electric power or gas supply, except for containment isolation valves on the suppression pool supply and return lines, which fail as-is.

The containment isolation valves that are not required to perform a post-accident recovery function are automatically closed upon receipt of a containment isolation signal from the Leakage Detection and Isolation system (LD&IS).

Normally closed safety-related isolation valves consisting of two parallel redundant air-operated check valves and two parallel, redundant, nitrogen-motor-operated, fail as-is gate valves 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. This line is safety-related up to and including the nitrogen-motor-operated gate valves due to its interface with safety-related RWCU/SDC system piping. The redundant valves are contained in separate fire zones for improved reliability.

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.

Upon receipt of a containment isolation signal from the LD&IS, containment isolation valves that are not required to perform a post-accident recovery function are closed.

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 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 FAPCS configuration is as described in Design Description of Subsection 2.6.2 and Figure 2.6.2-1.	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 and adequate flow capacity for the emergency makeup of the IC/PCCS pools and the Spent Fuel Pool from the Fire Protection System and offsite water supplies.	3. Perform a test to confirm flow path and flow capacity from the Fire Protection System and offsite water sources to the pools.	3. Makeup water flow path is demonstrated and confirmed by operation of the function. Flow capacity for offsite water supplies is demonstrated to be no less than the capacity of the Fire Protection System, which is tested under the criteria in Table 2.16.3-2.
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 Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		design values for each operating mode.
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 normal operating dP.	7. Test results and/or analyses demonstrate that: • The containment isolation valve automatically closes. • The valve stroke time is less than 30 seconds
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 while a high pressure single is present.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. Level instruments with adequate operating ranges are provided for monitoring and controlling the water levels in the Spent Fuel Pool (SFP) and IC/PCCS pools.	10. Perform instrument calibration and simulated makeup water control test.	10. Water level instruments indicate accurate water levels. Makeup water control valves open and close upon receipt of water level signals as designed. Operating range for SFP level instruments spans the normal water level down to the top of the active fuel. Operating range for the IC/PCCS pools spans the normal water level down to the midpoint of the IC heat exchanger tube.

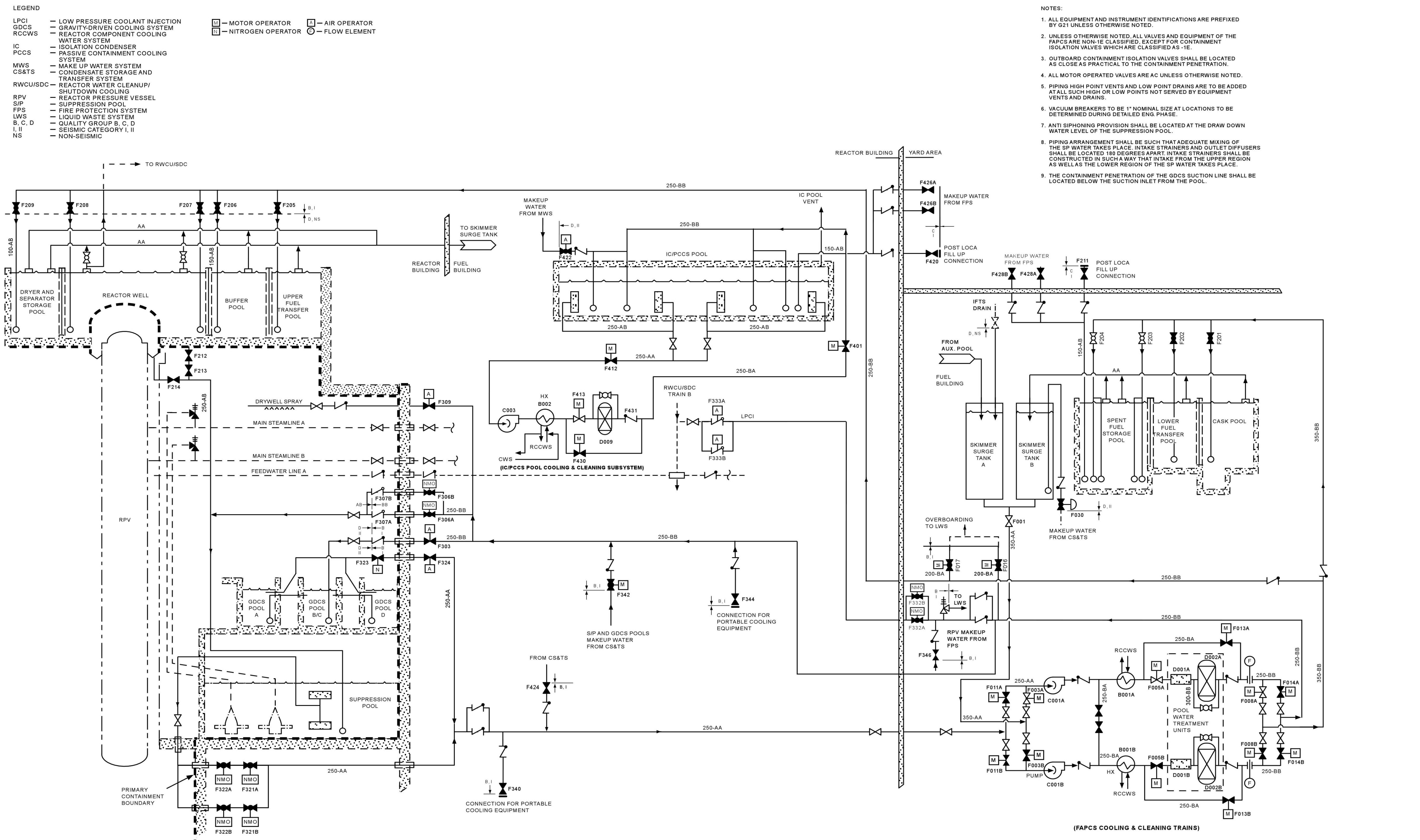


Figure 2.6.2-1. Fuel and Auxiliary Pools Cooling Cleanup System

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

The main control room panels (MCRPs) 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, and maintain the plant in a safe shutdown condition. Human factors engineering principles are 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 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 among safety-related divisional circuits.	1. Inspection of the as-installed safety-related divisional circuits is performed.	1. Independence is maintained between safety-related divisional circuits.
2. Independence is maintained between safety-related divisional and nonsafety-related circuits.	2. Inspection of the as-installed safety-related divisional and nonsafety-related circuits is performed.	2. Independence is maintained between safety-related divisional and nonsafety-related circuits.
3. Independence is maintained among safety-related divisional circuits.	3. Tests are performed by energizing/deenergizing components of one division at a time and checking for voltage in all divisions.	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 either from control panels in a control room or locally. Key system parameters and alarms are repeated in the Main Control Room.

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 to maintain 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. Because of their fail-safe design, no potential sources of missiles or pipe breaks prevent modules from performing their safety function(s). Each panel/rack containing equipment used for safety-related functions is classified as Seismic Category I, and constructed/located to assure electrical independence and physical separation among safety-related 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 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 are conducted.	1. The as-built LCPs conform to the basic configuration described in Subsection 2.7.3.
2. Safety-related LCPs are powered from their respective safety-related divisions. Independence is provided among safety-related divisions and between safety-related divisions and nonsafety-related equipment.	2. a. Tests are conducted in the LCPs by providing a test signal to only one safety-related division at a time. b. Inspections of the as-built safety-related divisions in the LCPs are conducted.	2. a. A test signal exists in only the safety-related division under test in the LCPs. b. In the LCPs, physical and electrical independence exists among as-built safety-related divisions. Physical and electrical independence exists between these safety-related divisions and nonsafety-related 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.

- Fuel rod failure is predicted to not occur as a result of normal operation and anticipated operational occurrences.
- Control rod insertion will not be prevented as a result of normal operation, anticipated operational occurrences or postulated accident.
- The number of fuel rod failures will not be underestimated for postulated accidents.
- Coolability will be maintained for all design basis events, including seismic and LOCA events.
- Specified acceptable fuel design limits (thermal and mechanical design limits) will not be exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.
- In the power operating ranges, the prompt inherent nuclear feedback characteristics will tend to compensate for a rapid increase in reactivity.
- The reactor core and associated coolant, control and protection systems will be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

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.

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:

- During any design basis events including the mechanical loading from safe shutdown earthquake event combined with LOCA event, fuel channel damage will not be so severe as to prevent control rod insertion when it is required.
- Coolability will be maintained for all design basis events.
- Channel bowing will not cause specified acceptable fuel design limits to be exceeded during normal operation and anticipated operational occurrences.

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, structure, or welded connection;
- 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.

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 three depicted mobile systems boxes are placeholders that indicate examples of unit operations and chemical reactors that could process radioactive waste. Actual mobile system unit operations and chemical reactors may differ based on improvements in radwaste technology.

The LWMS processing equipment is located in the Radwaste Building.

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 either from the Radwaste Control Room or locally 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 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 functional description of LWMS is in Subsection 2.10.1.	1. Inspections of the as-built system will be conducted.	1. The as-built LWMS conforms to the basic functional description 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 API or ASME Code per Regulatory Guide 1.143 Revision 2.	2. The results of the hydrostatic test of the ASME Code components of the LWMS conform with the requirements in the API or ASME Code per Regulatory Guide 1.143 Revision 2.
3. Main control room alarms are provided for unit-specific key parameters of the LWMS.	3. Inspections or tests will be performed on the main control room alarms for the LWMS.	3. Inspections or tests confirm that there alarms for the unit-specific key parameters in the main control room.
4. 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.	4. Tests will be conducted on the as-built liquid waste system using a simulated high radiation signal.	4. The discharge flow terminates upon receipt of a simulated high radiation signal.

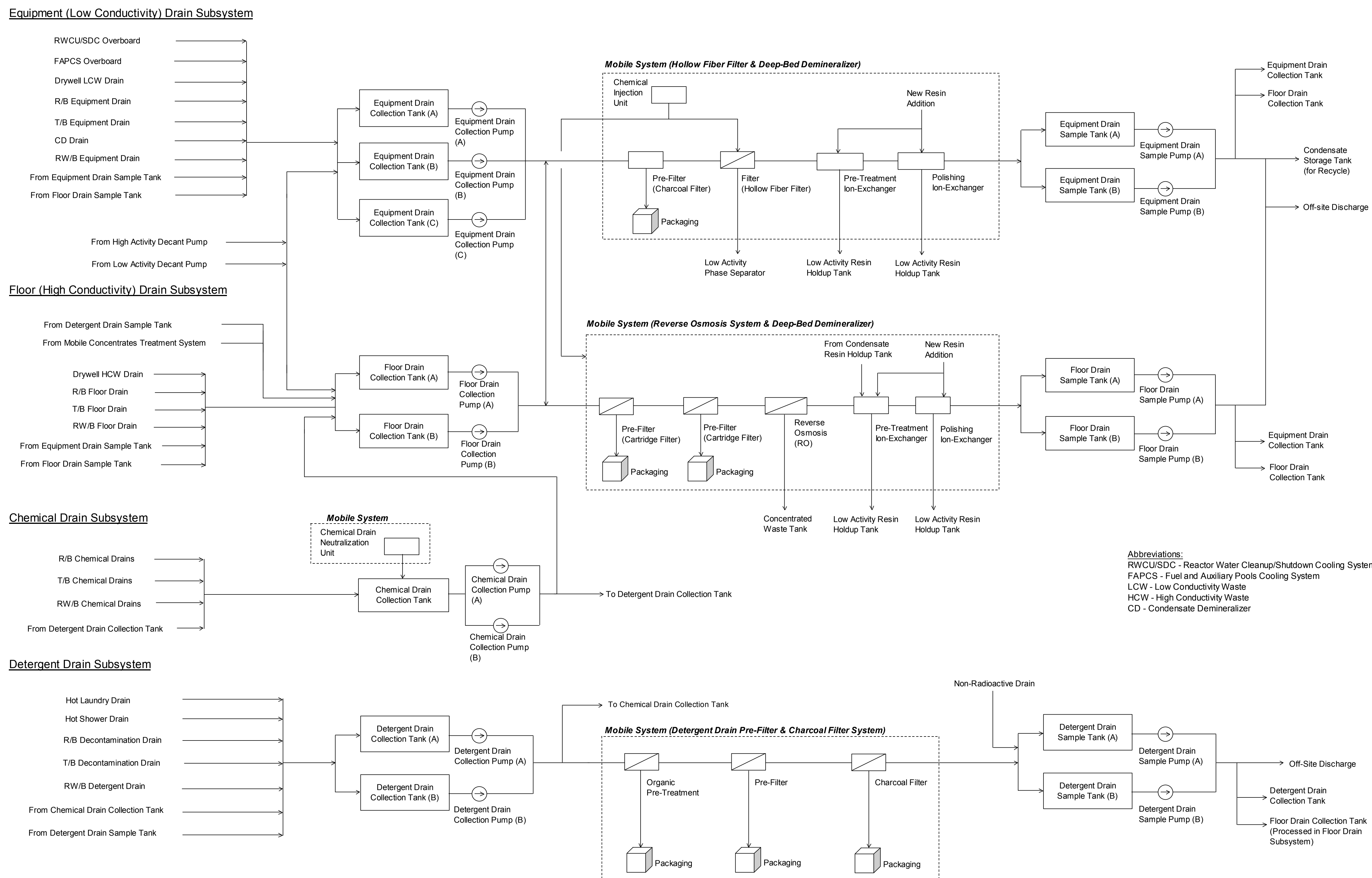


Figure 2.10.1-1. LWMS Process Diagram*

* Similar to or similar

2.10.2 Solid Waste Management System

Design Description

The Solid Waste Management System (SWMS) is designed to control, collect, handle, process, package, and 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 either from the Radwaste Control Room or locally 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 thereby, decay of radioactive gases in the offgas from the main condenser air removal 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 1 or Division 2 and the ANSI B31.3 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 manages the main condenser air removal 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. 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 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 leak test of the OGS will be conducted in the plant in accordance with API-620, ASME VIII Division 1 or Division 2 and ANSI B31.3 requirements per Regulatory Guide 1.143 Rev. 2 Table 1	2. The leak test results conform to the API-620, ASME VIII Division 1 or Division 2 and ANSI B31.3 requirements per Regulatory Guide 1.143 Revision 2 Table 1.
3. The OGS is designed to reduce radioactivity leakage through the OGS valve seats and mechanical joints and externally into the plant.	3. "Soap bubble" tests will be performed. This requirement does not apply to in-line process valves.	3. A test report of the "soap bubble" tests will show no detectable leakage.
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, the off-gas system discharge valve will close.	4. Test reports document that: a. A Main Control Room alarm activates on an OGS discharge line high radiation signal. b. The OGS charcoal bed valves operate in the main adsorber 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 Generator, moisture separator reheaters, steam auxiliaries and turbine bypass system. The TMSS does not include the seismic interface restraint, main turbine stop valves 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 Isolation Valve(s) (SAIVs) on branch lines between the Main Steam Isolation Valves (MSIVs) and main turbine stop valves on an MSIV isolation signal. These valves fail closed on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure.
- Opens the drain valve(s) on a MSIV isolation signal that are required to change position to provide the MSIV leakage path to the main condenser. The required drain valve(s) are equipped with reliable power sources or designed to fail to the open position 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 turbine bypass system and power cycle auxiliaries, as needed.

The TMSS is nonsafety-related. However, the TMSS is analyzed, fabricated and examined to ASME Code Class 2 requirements. 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.

TMSS piping, including the Steam Auxiliary Isolation 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 MSIV fission product leakage path to the main condenser is analyzed to demonstrate structural integrity under Safe Shutdown Earthquake (SSE) loading conditions. The drain valve(s), that are required to change position to provide the MSIV leakage path to the main condenser are equipped with reliable power sources or designed to fail to the required position on loss of power or air.

The TMSS is located in the Reactor Building 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 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 to the basic configuration description of Subsection 2.11.1.
2. The ASME Code Section III 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 Section III.	2. The results of the hydrostatic test of the ASME Code components of the TMSS satisfy the applicable requirements in the ASME Code, Section III.
3. Upon receipt of an MSIV closure signal, the SAIV(s) close(s) and required MSIV fission product leakage path TMSS drain valve(s) open(s).	3. Tests will be performed on the SAIV(s) and required MSIV fission product leakage path TMSS drain valve(s) using simulated MSIV closure signals.	3. The SAIVs close(s) and required MSIV fission product leakage path TMSS drain valve(s) open(s) following receipt of a simulated MSIV closure signal.
4. The SAIV(s) fail(s) closed and required MSIV fission product leakage path TMSS drain valve(s) open(s) on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure.	4. A functional test will be performed on SAIV(s) and required MSIV fission product leakage path TMSS drain valve(s).	4. The SAIV(s) close(s) and required MSIV fission product leakage path TMSS drain valve(s) open(s) on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure.
5. TMSS piping, including the SAIV(s) from the seismic interface restraint to the main stop and main turbine bypass valves and the required MSIV fission product leakage path, is analyzed to demonstrate structural integrity under SSE loading conditions.	5. A seismic analysis of the as-built TMSS piping and SAIV(s) and required MSIV fission product leakage path will be performed.	5. An analysis report exists which concludes that the as-built TMSS piping SAIV(s) and required MSIV fission product leakage path can withstand a SSE without loss of structural integrity.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The integrity of the as-built main steam valve leakage path to the condenser (main steam piping, bypass piping, required drain piping, and main condenser) is not compromised by non-seismically designed systems, structures and components.	6. Inspections (e.g., walk down) of non-seismically designed systems, structures and components overhead, adjacent to, and attached to the main steam line leakage path (i.e., the main steam piping, bypass piping, required drain piping and main condenser) will be performed.	6. A report exists that documents that the non-seismically designed systems, structures and components overhead, adjacent to, and attached to the main steam line leakage path to the condenser will not compromise the integrity of the main steam piping, bypass piping, required drain piping and main condenser.

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(s) by the condensate pumps, passes through the CPS, auxiliary condenser/coolers, low-pressure feedwater heaters and into the feedwater tank. The feedwater pumps take a suction from the feedwater tank and pump feedwater through the high-pressure feedwater heaters to the reactor. The C&FS boundaries extend from the main condenser outlet to (but not including) the seismic interface restraint outside containment. The C&FS is designed to provide at least 135% of the rated feedwater flow to the RPV during anticipated operational occurrences. The feedwater pump maximum runout capacity at 7.34 MPa gauge (1065 psig) is 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 Reactor Building 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 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 to Subsection 2.11.2
2. The maximum runout capacity of three feedwater pumps at 7.34 MPa gauge (1065 psig) is less than or equal to 155% of rated feedwater flow.	2. An analysis of the as-built feedwater system will be performed to show that the maximum runout capacity of three pumps is less than 155% of rated feedwater flow at 7.34 MPa gauge (1065 psig).	2. An analysis exists which concludes that the maximum runout capacity of three pumps from the as-built feedwater system is less than or equal to 155% of rated feedwater flow at 7.34 MPa gauge (1065 psig).
3. Three operating feedwater pumps are capable of supplying 135% of the rated feedwater flow at 7.34 MPa gauge (1065 psig).	3. An analysis of the as-built feedwater system will be performed to show that three feedwater pumps are capable of supplying 135% of the rated feedwater flow at 7.34 MPa gauge (1065 psig).	3. An analysis exists which shows that three feedwater pumps are capable of supplying 135% of the rated feedwater flow at 7.34 MPa gauge (1065 psig).

2.11.3 Condensate Purification System

Design Description

The Condensate Purification System (CPS) purifies and treats the condensate as required to maintain reactor feedwater purity, using filtration to remove suspended solids, including corrosion products, and ion exchange to remove dissolved solids. The CPS consists of full flow high efficiency particulate filters and full flow ion exchange demineralizers.

The CPS does not perform, ensure or support 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 controlled areas and sent to the radwaste system for processing.

The CPS is located in the Turbine Building.

The CPS has alarms and indication(s) for effluent conductivity in the main control room or on local panels.

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 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 to the basic configuration as described in Subsection 2.11.3.
2. Alarms and indication(s) are provided in the main control room or on local panels as described in Subsection 2.11.3.	2. Inspections will be performed on the alarms and indication(s) for the CPS.	2. Alarms and indication(s) exist in the main control room or on local panels as described in Section 2.11.3.

2.11.4 Turbine Generator System

Design Description

The main turbine for the ESBWR standard plant has one high pressure (HP) turbine and three low pressure (LP) turbines. Other turbine configurations may be selected on a unit-specific basis. The steam passes through moisture separator reheater(s) (MSRs) 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 condensate and feedwater.

The steam and power conversion system is designed to operate above the rated turbine throttle flow. The system's physical layout provides protection to essential systems and components, as required, from the effects of high and moderate energy Turbine Generator (TG) system piping failures or failure of the connection(s) from the low pressure turbine exhaust hoods to the condenser. Essential systems and components are as defined in BTP SPLB 3-1.

Turbine Disk Integrity

Turbine disk integrity is provided through the combined use of selected materials with suitable toughness, analyses, testing, inspections, and operating procedures. Turbine components and valves will be in-service tested and inspected at intervals in accordance with industry practice (i.e., BWR Owners Group guidance) or as required to meet turbine missile generation probability requirements.

Turbine Overspeed Protection System

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

The following component redundancies are employed to guard against overspeed:

- Main stop valves/control valves;
- Intermediate stop valves/intercept valves
- Normal speed control/Primary Overspeed control/Emergency overspeed control;
- Fast acting solenoid valves/emergency trip fluid system (ETS); and
- Speed control signals/primary overspeed trip/emergency overspeed trip.

The Turbine Generator system is nonsafety-related and is not needed to effect or support a safe shutdown of the reactor. The Turbine Generator is oriented within the Turbine Building to be inline with the Reactor Building to minimize the potential of turbine missiles damaging any safety-related equipment or structures.

The probability of turbine material and overspeed related failures, resulting in external turbine missiles, is $< 1 \times 10^{-4}$ per turbine year.

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 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 TG 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, Normal speed control/Primary overspeed control/Emergency overspeed Control, Fast acting solenoid valves/ Emergency trip fluid system (ETS), and Speed control signals/Primary overspeed trip/Emergency overspeed trip. b. Valves that supply steam to turbine close upon receipt of overspeed signal.
2. The Turbine Generator will be oriented to reduce the potential for low trajectory turbine 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 Building. The Control Building is outside the low-trajectory turbine missile strike zone.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The physical layout of the system assures that protection is provided to essential systems and components, as required, from the effects of high and moderate energy TG system piping failures or failure of the connection(s) from the low pressure turbine exhaust hood(s) to the condenser.	3. Inspections of the as-built Turbine Building and plant arrangements will be conducted.	3. Essential systems and components are protected, as required, from the effects of high and moderate energy TG system piping failures or failure of the connection(s) from the low pressure turbine exhaust hood to the condenser.
4. Turbine disk integrity is provided through the combined use of selected materials with suitable toughness, analyses, design, testing, inspections, and operating procedures.	4. a. An analysis of turbine material property data, turbine rotor and blade design, and pre-service inspection and testing will be conducted. This information will be available for review greater than one year before loading the fuel. b. Review of turbine and turbine valve in-service test and inspection, and operating procedures will be conducted.	4. a. An analysis exists that includes turbine material property data, rotor and blade design analyses (including loading combinations, assumptions and warm-up time) demonstrating sufficient safety margin to withstand loadings from overspeed events, and pre-service testing and inspection information (including scope, methods and acceptance criteria). b. The turbine and turbine valve in-service test and inspection program includes scope, frequency, methods, acceptance criteria, disposition of reportable indications, corrective actions, and technical basis for inspection frequency. In-service test, inspection and operating procedures are in accordance with industry practice and meet OEM requirements for turbine missile probability.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The probability of turbine material and overspeed related failures, resulting in external turbine missiles, is < 1×10^{-4} per turbine year.	5. A material and overspeed failures analysis will be performed on the as-built turbine design.	5. An analysis exists that documents that the probability of turbine material and overspeed related failures, resulting in external turbine missiles, is < 1×10^{-4} per turbine year.

2.11.5 Turbine Gland Seal System

The following is only applicable to a plant, if that plant's turbine needs a Turbine Gland Seal System (TGSS) to meet GDC 60.

Design Description

The TGSS prevents the escape of radioactive steam from the turbine shaft/casing penetrations and valve stems and prevents air in-leakage through sub-atmospheric turbine glands. Specifically, the TGSS

(1) Limits atmospheric air leakage into the turbine casings and minimizes radioactive steam leakage out of the casings of the Turbine Generator;

(2) Returns the condensed steam to the condenser and exhausts the noncondensable gases to the plant vent stack; and

(3) Has sufficient capacity to handle steam and air flows resulting from greater than normal gland clearances.

The TGSS has main control room monitors and/or alarms for gland steam condenser exhauster suction pressure, water level in the gland steam condenser drain leg, and continuous radiation monitoring of its effluents.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.11.5-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria for the TGSS.

Table 2.11.5-1

Turbine Gland Seal System ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic functions for the TGSS are as described in Subsection 2.11.5.	1. Inspections of the as-built system will be conducted.	1. The as-built system conforms to Subsection 2.11.5.
2. Main Control Room instrumentation for the TGSS is as described in Subsection 2.11.5.	2. Inspections of the as-built Main Control Room TGSS instrumentation will be conducted.	2. The as-built Main Control Room instrumentation conforms to Subsection 2.11.5.

2.11.6 Turbine Bypass System

Design Description

The Turbine Bypass System (TBS) passes steam directly to the main condenser via the Turbine Main Steam System (TMSS) under the control of the Steam Bypass and Pressure Control (SB&PC) system. Steam is bypassed to the condenser whenever the amount of steam generated by the reactor cannot be entirely used by the turbine. The TBS in the ESBWR standard plant design has the capability to accommodate a full load rejection and 110% of rated main steam flow without the operation of a SRV. The SB&PC 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, ensure, or support any safety-related function and is classified as nonsafety-related. However, the TBS is used to mitigate anticipated operational occurrences (AOOs), which are defined in 10 CFR 50, Appendix A as part of normal operations), and is analyzed to demonstrate structural integrity under safe shutdown earthquake (SSE) loading conditions.

The turbine bypass valves are controlled by signal(s) from the SB&PC System.

The turbine bypass valves normally open upon turbine trip or generator load rejection, automatically trip closed whenever the main condenser pressure increases to preset value(s), and fail closed on loss of electrical power or hydraulic pressure to their actuator. No single active 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 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 to the basic configuration of Subsection 2.11.6.
2. The turbine bypass valves are controlled by signal(s) from the Steam Bypass and Pressure Control System, as described in Subsection 2.11.6.	2. Tests will be conducted using a simulated signal(s).	2. Turbine bypass valves operate upon receipt of simulated signal(s) 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 active 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 active single failure can disable more then 50% of the installed bypass capacity.	4. An analysis report exists that concludes that no single active failure can disable more then 50% of the installed bypass capacity.
5. The TBS has the capability to accommodate 110% of rated mainsteam flow without the operation of a SRV.	5. A design analysis of the unit-specific TBS will be performed, to confirmation that the TBS can accommodate 110% of rated mainsteam flow without the operation of a SRV.	5. An analysis record exists that concludes that the TBS has the capability to accommodate 110% of rated mainsteam flow without the operation of a SRV.

2.11.7 Main Condenser

Design Description

The Main Condenser (MC) condenses and deaerates the exhaust steam from the main turbine, provides a hold-up volume for N^{16} decay, provides a heat sink for the Turbine Bypass System (TBS), and is a collection point for other steam cycle drains and vents.

The MC hotwell provides a hold-up volume for main steam isolation valve (MSIV) fission product leakage.

The MC is classified as nonsafety-related. However, the condenser supports and anchors are designed to maintain condenser integrity following a safe shutdown earthquake.

The MC is located in the Turbine Building.

The MC tubes are made from corrosion-resistant material.

The MC normally operates at a vacuum; consequently, air in-leakage is into the shell side of the MC. Circulating water leakage into the condenser shell is detected by measuring the conductivity of sample water extracted at select locations in the hotwell. In addition, conductivity monitor(s) in the condensate system provide alarms in the Main Control Room (MCR).

An increase in main condenser pressure above preset level(s) causes a MCR alarm, 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 for the Main Condenser.

Table 2.11.7-1

ITAAC For The Main Condenser

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The condenser supports and anchors are designed to maintain condenser integrity following a safe shutdown earthquake.	1. An analysis of the ability of the as-built condenser supports and anchors to maintain condenser integrity following 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 maintain condenser integrity following a safe shutdown earthquake.
2. An increase in main condenser pressure above preset level(s) causes a MCR alarm, turbine trip, reactor scram, bypass valve closure, and closure of the MSIVs.	2. The main condenser pressure transmitters and associated logic will be tested with a simulated increase in main condenser pressure for HIGH condenser pressure function.	2. Tests of the main condenser pressure transmitters cause a MCR alarm, turbine trip, reactor scram, bypass valve closure, and closure of the MSIVs with a simulated increase in main condenser pressure above preset level(s).

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 Normal Power Heat Sink (NPHS).

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 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 water 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 a nonsafety-related system, and has no safety design basis other than provision for safety-related containment penetrations and isolation valves. The containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.

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 for the MWS.

Table 2.12.1-1

ITAAC For The Makeup Water System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The MWS has safety-related containment penetrations and isolation valves.	1. Inspections of the as-built MWS safety-related containment penetrations and isolation valves will be conducted.	1. Records of inspections confirm that the as-built MWS has safety-related containment penetrations and isolation valves.
2. The MWS containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.	2. Inspections of the as-built MWS containment penetrations and isolation valves design documents will be conducted.	2. Records of inspections confirm that the as-built MWS containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.

2.12.2 Condensate Storage and Transfer System

Design Description

This system does not perform or ensure any safety-related function, is not required to achieve or maintain safe shutdown, and is not subject to high regulatory oversight. 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

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

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12.4 Turbine Component Cooling Water System

Design Description

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

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system

2.12.5 Chilled Water System

Design Description

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

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.12.5-1 provides definitions of the inspections, tests, and/or analyses, together with associated acceptance criteria for the CWS.

Table 2.12.5-1

ITAAC For The Chilled Water Subsystem

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The CWS has safety-related containment penetrations and isolation valves.	1. Inspections of the as-built CWS safety-related containment penetrations and isolation valves will be conducted.	1. Records of inspections confirm that the as-built CWS has safety-related containment penetrations and isolation valves.
2. The CWS containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.	2. Inspections of the as-built CWS containment penetrations and isolation valves design documents will be conducted.	2. Records of inspections confirm that the as-built CWS containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.

2.12.6 Oxygen Injection System

Design Description

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

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12.7 Plant Service Water System

Design Description

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

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12.8 Service Air System

Design Description

The Service Air System (SAS) is a nonsafety-related system, and has no safety design basis other than provision for safety-related containment penetrations and isolation valves. The containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.

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 for the SAS.

Table 2.12.8-1

ITAAC For The Service Air System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The SAS has safety-related containment penetrations and isolation valves.	1. Inspections of the as-built SAS safety-related containment penetrations and isolation valves will be conducted.	1. Records of inspections confirm that the as-built SAS has safety-related containment penetrations and isolation valves.
2. The SAS containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.	2. Inspections of the as-built SAS containment penetrations and isolation valves design documents will be conducted.	2. Records of inspections confirm that the as-built SAS containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.

2.12.9 Instrument Air System

Design Description

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

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12.10 High Pressure Nitrogen Supply System

Design Description

The High Pressure Nitrogen Supply System (HPNSS) is a nonsafety-related system, and has no safety design basis other than provision for safety-related containment penetrations and isolation valves. The containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.

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 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 HPNSS has safety-related containment penetrations and isolation valves.	1. Inspections of the as-built HPNSS safety-related containment penetrations and isolation valves will be conducted.	1. Inspections confirm that the as-built HPNSS has safety-related containment penetrations and isolation valves.
2. The HPNSS containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.	2. Inspections of the as-built HPNSS containment penetrations and isolation valves design documents will be conducted.	2. Inspections confirm that the as-built HPNSS containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.

2.12.11 Auxiliary Boiler System

Design Description

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

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12.12 Hot Water System

Design Description

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

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12.13 Hydrogen Water Chemistry System

This system is optional and is not within the scope of the design certification. There is no entry for this system.

2.12.14 Process Sampling System

Design Description

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

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.12.15 Zinc Injection System

This system is optional and is not within the scope of the design certification. Therefore, there is no entry for this system.

2.12.16 Freeze Protection

This system is optional and is not within the scope of the design certification. Therefore, there is no entry for this system.

2.12.17 Station Water System

Design Description

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

Inspections, Tests, Analyses and Acceptance Criteria

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 main 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 (UATs) and the 13.8 kV unit auxiliary Power Generation (PG) switchgear buses. Non-segregated phase bus ducts also provide for the electrical interconnection between the 6.9kV (split secondary transformer) UAT and the Plant Investment Protection (PIP) buses. The two UATs secondary windings are each associated with two PG switchgear buses and two PIP load buses and are physically separated and electrically isolated to minimize the likelihood of simultaneous failure. Two reserve auxiliary transformers (RATs) supply power identical to the UATs, through non-segregated phase bus ducts, to both the PG switchgear buses and the PIP buses. In the event the off-site normal preferred power supply fails, the PG and PIP buses are automatically transferred to the off-site alternate preferred power supply, through the RATs. Return back to the normal preferred power source is a manually performed transfer.

If both the normal and alternate preferred power supplies are lost (LOOP) then the standby diesel generators will start and automatically connect to the PIP buses and sequentially load the buses.

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

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 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 to 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. a. The as-built ratings of each UAT are equal to or greater than its as-installed load assignment. b. The as-built and tested transformer parameters agree with those used in the electric load analysis, within acceptable tolerances. c. The as-built and tested transformer name plate data agrees with the as-tested data, within acceptable tolerances.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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. a. The as-built ratings of the RAT are equal to or greater than its as-installed load assignment. b. The as-built and tested transformer parameters agree with those used in the electric load analysis, within acceptable tolerances. c. The as-built and tested transformer name plate data agrees with the as-tested data, within acceptable tolerances.
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 specific division are energized. b. Only components in the specific load group are energized. c. Only components in the specific division are energized.
5. The four Isolation Phase Buses and mountings conform to Seismic Category 1 requirements.	5. An inspection will be performed of the isolation phase buses to verify that the installed equipment including anchorage is seismically bounded by the tested and/or analyzed condition.	5. A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested and/or analyzed conditions.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Controls, displays and alarms exist in the MCR to cause the PG and PIP circuit breakers to perform their required functions.	6. Tests will be performed to verify the controls, displays and alarms for the PIP and PG circuit breakers perform their required functions.	6. All alarms, displays and/or controls are present or can be retrieved in the control room for the PG and PIP circuit breakers.

2.13.2 Electrical Wiring Penetrations

Design Description – Primary Electrical Containment Penetrations

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

Safety-related circuit separation groups designated Division 1, 2, 3, 4, and nonsafety-related circuits run through separate safety-related 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.

The control circuits, control power circuits, and instrumentation circuits passing through electrical penetrations are normally low current cables. This minimizes the need to protect the penetration from the effects of fault or overload currents.

Redundant over-current interrupting devices are provided for all electrical circuits routed through containment penetrations, if the maximum available fault current (including failure of upstream devices) is greater than the continuous rating of the penetration. This avoids penetration damage in the event of failure of any single over-current device to clear a fault within the penetration or beyond it.

All safety-related electrical penetrations are environmentally and seismically qualified to ensure the execution of their safety-related functions.

Fire Barrier Penetrations, moved to Subsection 2.16.3.1.

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 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 Penetrations is described in Subsection 2.13.2.	1. Inspections of the as-built Electrical Wiring Penetrations will be conducted.	1. The as-built Electrical Wiring Penetrations conform to 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 safety-related 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 safety-related 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, and eight 72-hour divisional safety-related 250 VDC power supplies.

The eight 72-hour safety-related DC power supplies provide power to 120Vac safety-related inverters for post-accident monitoring, MCR emergency lighting and safe shutdown loads.

Each of the four divisions of safety-related DC power is separate and independent. Divisions 1, 2, 3 and 4 each have two 72-hour batteries. The DC systems operate ungrounded (with ground detection circuitry) for increased reliability. Each division has a battery charger fed from its divisional Isolation Power Center (IPC). This system is designed so that no single failure in any division prevents safe shutdown of the plant.

The safety-related DC power supply is designed to permit periodic testing for operability and functional performance.

Nonsafety-related DC power is supplied through four nonsafety-related IPCs in the same manner as the safety-related DC power. Each of the two load groups receives power from two of the nonsafety-related IPCs. One IPC in each group provides power to a bus through a battery charger.

Alarms initiate in the Main Control Room (MCR) to indicate loss of battery chargers and inverters. Computer inputs can then be monitored to determine the source of the problem. Alarm and computer inputs from safety-related equipment or circuits are treated as safety-related and retain their divisional identification up through their safety-related isolation device. The output circuit from this isolation device is classified as nonsafety-related. The plant design and circuit layout of the DC systems provide physical separation of the safety-related equipment, cabling and instrumentation. 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 safety-related 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. 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-related 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 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 design documents confirm that the Direct Current Power Supply conforms to the basic configuration described in Subsection 2.13.3.
2. Each safety-related divisional (Divisions 1, 2, 3, and 4) battery is provided with a normal and a standby battery charger supplied AC power from a IPC in the same safety-related division as the battery.	2. Inspections of the as-built safety-related Direct Current Power Supply will be conducted.	2. Inspection report(s) conclude that each as-built safety-related divisional (Divisions 1, 2, 3, and 4) battery is provided with a normal and a standby battery charger supplied AC power from a IPC in the same safety-related division as the battery.
3. Two sets of 72-hour batteries in each division are sized to supply its design loads, at the end-of-installed-life, for a minimum of 72 hours without recharging.	3. a. Analyses for the as-built safety-related batteries to determine battery capacities will be performed based on the design duty cycle for each battery. b. Tests of each as-built safety-related battery will be conducted by simulating loads which envelope the analyzed battery design duty cycle.	3. a. Analysis reports of the as-built batteries exist and conclude that two sets of safety-related batteries in each division have the capacity, as determined by the vendor performance specification, to supply its rated constant current , for a minimum of 72 hours without recharging. b. Test report(s) conclude that the capacity of each as-built safety-related battery equals or exceeds the analyzed battery design duty cycle capacity.

Table 2.13.3-1

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Each safety-related normal battery charger is sized to supply its respective safety-related division's normal steady state loads while charging its respective safety-related battery.	4. Tests of each as-built safety-related normal battery charger will be conducted by supplying its respective safety-related division's normal steady state loads while charging its respective safety-related battery.	4. Test report(s) conclude that each as-built safety-related normal battery charger can supply its respective safety-related division's normal steady state loads while recharging its respective safety-related battery in 24 hours.

Table 2.13.3-1

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The safety-related 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 safety-related 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.	5. a. Analysis reports of the as-built safety-related DC electrical distribution system exist and conclude that the capacities of safety-related 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 and DC interrupting current requirements as determined by their DC nameplate ratings.
	b. Tests of the as-built safety-related battery and battery charger circuit breakers, DC distribution panels, their circuit breakers and fuses, will be conducted by operating connected safety-related loads at greater than or equal to the minimum allowable battery voltage and at less than or equal to the maximum equalizing battery charging voltage.	b. Test report(s) conclude that the connected as-built safety-related loads operate at greater than or equal to the minimum allowable battery voltage and at less than or equal to the maximum equalizing battery charging voltage.

Table 2.13.3-1

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.	6.	6.
a. The safety-related 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 safety-related DC electrical distribution system to determine fault currents will be performed.	a. Analysis reports of the as-built safety-related DC electrical distribution system exist and conclude that the capacities of as-built safety-related 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. Safety-related battery, battery charger and DC distribution panel circuit breakers and fuses are rated to interrupt fault currents.	b. Analyses for the as-built safety-related DC electrical distribution system to determine fault currents will be performed.	b. Analysis reports of the as-built safety-related 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 safety-related battery is located in a vented Seismic Category I structure and in its respective divisional battery room.	7. Inspections of the as-built safety-related batteries will be conducted.	7. Verify that each as-built safety-related battery is located in a vented Seismic Category I structure and in its respective divisional battery room.

Table 2.13.3-1

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Safety-related DC distribution panels are identified according to their safety-related division and are located in Seismic Category I structures and in their respective divisional areas.	8. Inspections of the as-built safety-related DC distribution panels will be conducted.	8. Inspection reports conclude that the as-built DC distribution panels are identified according to their safety-related division and are located in Seismic Category I structures and in their respective divisional areas.
9. Safety-related DC distribution system cables and raceways are identified according to their safety-related division. safety-related divisional cables are routed in Seismic Category I structures and in their respective divisional raceways.	9. Inspections of the as-built safety-related DC distribution system cables and raceways will be conducted.	9. Inspection reports conclude that the as-built safety-related DC distribution system cables and raceways are identified according to their safety-related division. Verify that safety-related divisional cables are routed in Seismic Category I structures and in their respective divisional raceways.

Table 2.13.3-1

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. For the safety-related DC electrical distribution system, independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	10. a. Tests will be conducted on the as-built DC electrical distribution system by providing a test signal in only one safety-related division at a time. b. Inspections of the as-built DC electrical distribution system will be conducted.	10. a. Test reports document that a test signal exists in only the safety-related division under test in the DC electrical distribution system. b. Inspection reports conclude that the as-built DC electrical distribution system, physical separation or electrical isolation exists between safety-related divisions. Physical separation or electrical isolation exists between these safety-related divisions and nonsafety-related 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. Inspection reports conclude that 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 two permanent nonsafety-related plant investment protection load groups when the normal and alternate preferred 6.9kV power supplies are not available.

On a defense-in-depth basis, the Standby On Site Power Supply can ultimately provide power to safety-related loads. These loads are powered by safety-related DC power from safety-related 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 safety-related uninterruptible 120 VAC power is supplied by the four divisions of DC power and the eight safety-related batteries, two per division, that provide redundant, reliable power for 72 hours to the monitoring, safety logic and control functions during normal operations including AOOs and accident conditions.

Each of the four divisions of this safety-related uninterruptible 120 VAC power is separate and independent. Each division has two inverters supplied from two safety-related DC buses.

A static bypass switch is provided for transferring safety-related UPS AC load from safety-related 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 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 safety-related (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 safety-related Uninterruptible AC Power Supply conforms to the basic configuration described in Subsection 2.13.5.
b. The divisional safety-related inverters and associated distribution panels are electrically independent and physically separated.	b. Inspections will be performed to confirm that each of the four divisional safety-related uninterruptible power supplies (UPSs) and their associated distribution panels are electrically independent and physically separate.	b. Inspections confirm that each of the four divisional safety-related uninterruptible power supplies (UPSs) and their associated distribution panels are electrically independent and physically separate.
c. Two safety-related inverters per division automatically transfer to an alternate safety-related normal 120 VAC power supply in case of failure of an inverter.	c. Tests will be conducted to confirm the automatic transfer within each division on their inverters to an alternate safety-related normal 120 VAC power supply upon receipt of a simulated inverter failure.	c. Tests confirm that the automatic transfer within each division on their inverters to an alternate safety-related normal 120 VAC power supply occurs upon receipt of a simulated inverter failure.
2. Each safety-related 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 nonsafety-related 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 classified as nonsafety-related. The failure of any communications system does not adversely affect safe shutdown capability.

The Communications System may include a 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 site-specific.

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 buses. The standby lighting system supplements the normal lighting system and 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 8-hour 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 safety-related batteries and their buses 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 safety-related batteries through safety-related inverters. Excluding the self-contained battery lighting units, the MCR emergency lighting system is safety-related, and meets Seismic Category I requirements for mountings.

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 for the Lighting Power Supply.

Table 2.13.8-1

ITAAC For The Lighting Power Supply

Design Commitments	Inspections, Tests, Analyses	Acceptance Criteria
1. The safety-related Emergency Lighting System supplies MCR illumination with four divisions of emergency lighting operable for at least 72 hours following a design basis event including the loss of all ac power sources.	1. Tests will be conducted to confirm that the safety-related 250VDC 72-hour batteries supply Emergency Lighting to the MCR.	1. Tests confirm that the safety-related 72-hour batteries supply Emergency Lighting to the MCR.
2. Safety-related lighting systems are electrically independent and physically separated. Cables are routed in the respective divisional raceways.	2. Inspections will be performed by operating one division at a time to confirm that lighting equipment and cables are electrically independent and physically separated between safety divisions and between normal and standby lighting systems.	2. Inspections confirm 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 tested.	3. Inspections and tests confirm DC self contained battery-operated lighting units a. Are tested for eight hours, and b. Are provided for stairways, exit routes and major control areas, which could be involved in shutdown or recovery operations.
4. The MCR emergency lighting system meets Seismic Category I requirements for mountings.	4. Inspections will be conducted of the analyses and installation records demonstrating that the MCR emergency lighting system mountings meet Seismic Category I requirements.	4. Inspections confirm that the MCR emergency lighting system mountings meet Seismic Category I requirements.

2.14 POWER TRANSMISSION

Power Transmission is site-specific.

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 is provided with an isolation valve in series, which automatically closes if the vacuum breaker fails to fully close when required. Each vacuum breaker has proximity sensors that provide position of open/close indication and an alarm in the main control room.

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 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 described in Subsection 2.15.1.	1. Inspections of the as-built system will be conducted.	1. Inspection reports document that the as-built Containment System conforms with the description in Subsection 2.15.1.
2. The primary containment pressure boundary defined in Subsection 2.15.1 is designed to meet ASME Code, Section III requirements. The reinforced concrete containment vessel (RCCV) and its liners are designed to meet the requirements in Article CC-3000 of ASME Code, Section III, Div. 2. The steel components of the RCCV are designed to meet the requirements in Article NE-3000 of ASME Code, Section III, Div. 1.	2. Inspections of ASME Code required documents will be conducted.	2. An ASME Code Certified Stress Report exists for the pressure boundary components. For ASME Section III, Division 2 construction, stress reports demonstrate compliance to NCA-3350 through NCA-3380, and NCA-3454. For ASME Section III, Division 1 construction, stress reports demonstrate compliance to NCA-3550.
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.

Table 2.15.1-1

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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 retains its 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. The first prototype containment structure will be instrumented to measure strains per ASME Code Section III, Div. 2, CC-6370.	5. Test results demonstrate compliance to ASME Code Section III, Div. 2, CC-6000, Structural Integrity Test of Concrete Containments.
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.

Table 2.15.1-1

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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 calculated 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 has proximity sensors to detect open/close position.	9. Inspections will be performed with the vacuum breakers in the open and closed positions, with indication of the open/close position.	9. Inspection report show that the vacuum breaker proximity sensors detect open/close position of vacuum breakers and provide indication.
10. Each vacuum breaker isolation valve automatically closes if the vacuum breaker does not fully close when required.	10. A test will be performed by simulating a not-fully closed vacuum breaker signal originating from the closed position proximity sensor.	10. Each vacuum breaker isolation valve automatically closes.
11. Control Room has indication of the open/close position for vacuum breakers.	11. Inspections will be performed on the Control Room for indicators of open/close position of vacuum breakers.	11.Indicators of open/close for vacuum breakers exist in the Control Room.

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 Pressure 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.

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), in conjunction with the suppression pool, 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, independent trains, 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 train.

Each train, 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, and designed to ASME Code Section III, Class 2, Quality Class B.

Together with the suppression pool, the six PCC condensers limit containment pressure to less than its design pressure. The Dryer/Separator pool and Reactor Well shall be designed to have sufficient water volume to provide makeup water to the IC/PCC pools for the initial 72 hours of a LOCA.

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 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. a. The PCCS is classified as safety-related and Seismic Category I, and designed to ASME Code Section III, Class 2, Quality Class B.	1. Inspections of the as-built system will be conducted. a. Inspections of the as-built PCCS is classified as safety-related and Seismic Category I, and designed to ASME Code Section III, Class 2, Quality Class B.	1. The as-built PCCS conforms to the basic configuration shown in Figure 2.15.4-1. a. The design reports of the as-built PCCS is classified as safety-related and Seismic Category I, and designed to ASME Code Section III, Class 2, Quality Class B.
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 to the requirements in the ASME Code, Section III.
3. The PCCSs are closed train 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. Test report(s) document that 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.

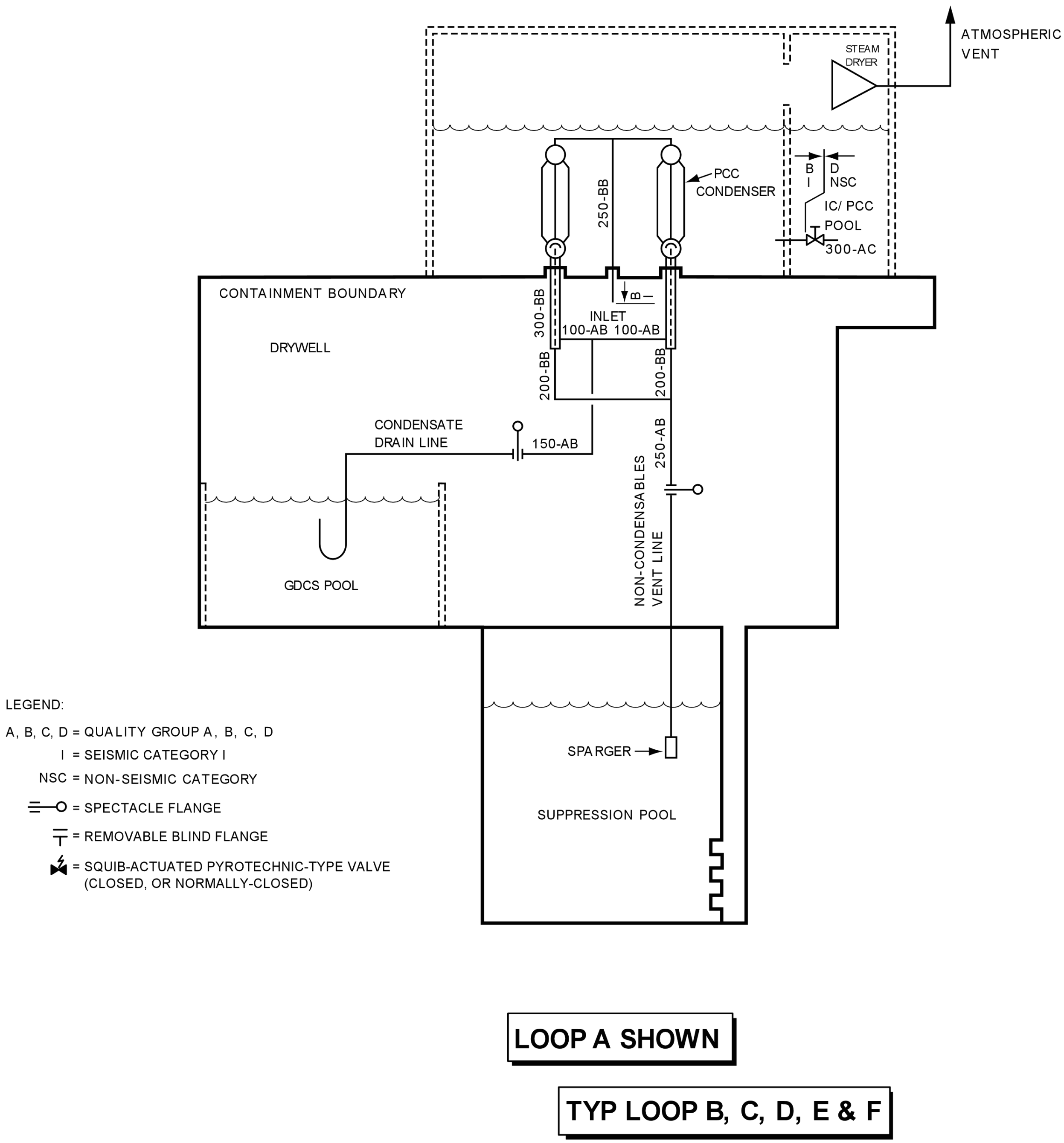


Figure 2.15.4-1. Passive Containment Cooling System Schematic

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 containment shall be inerted to $\leq$ 4% oxygen by volume prior to and during power operation.

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 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 Containment Inerting System has safety-related containment penetrations and isolation valves.	1. Inspections of the as-built Containment Inerting System safety-related containment penetrations and isolation valves will be conducted.	1. Inspections confirm that the as-built Containment Inerting System has safety-related containment penetrations and isolation valves.
2. The Containment Inerting System containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.	2. Inspections of the as-built Containment Inerting System containment penetrations and isolation valves design documents will be conducted.	2. Inspections confirm that the as-built Containment Inerting System containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.
3. The containment can be inerted to less than or equal to 4% oxygen by volume.	3. Test of the containment in an inerted state will be conducted to determine oxygen concentration by volume.	3. Test report concludes that the containment can be inerted to less than or equal to 4% oxygen by volume.

2.15.6 Drywell Cooling System

Design Description

The Drywell Cooling System (DCS) does not perform or ensure any safety-related function, is not required to achieve or maintain safe shutdown, and is not subject to high regulatory oversight. Therefore the system is nonsafety-related and has no safety design basis.

Instrumentation and Control

Drywell temperature indications are provided in the main control room (MCR).

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 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 drywell temperature indications are retrievable in the main control room.	1. Inspections of main control room indications will be conducted and verified for retrievability of drywell temperature indications.	1. Inspection report documents that Drywell temperature indications are provided in the MCR.

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; and
- 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 safety-related 120 Vac divisional sources.

Containment atmospheric and drywell monitoring:

Each CMS subsystem is specified to meet environmental and radiological requirements for its location and intended post-accident operations. Design of the sampling ports will allow a representative containment atmosphere sample (drywell and wetwell) to be drawn for analysis.

The CMS has two independent redundant divisions to monitor the gamma radiation dose rate (nonsafety-related) and the concentrations of hydrogen and oxygen (safety-related) in the drywell and wetwell air during plant operation and following an accident. These channels, which measure parameters in the drywell and wetwell air, are continuously indicated in the MCR. The hydrogen and oxygen monitoring subsystem is capable of measuring hydrogen/oxygen concentration over the required range. In addition, the H2 monitor is specified to be in operation within a timeframe (to be determined) after occurrence of an accident in accordance with USNRC Regulatory Guide requirements.

The safety-related 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 safety-related 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 indicated in the Main Control Room (MCR)

Inspections, Tests, Analyses and Acceptance Criteria

Tables 2.15.7-1 through 2.15.7-2 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 and the suppression pool monitoring portions of CMS.

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. a. The as-built CMS conforms to the basic configuration as defined in Subsection 2.15.7. b. The hydrogen monitoring subsystem is capable of measuring hydrogen concentration over the required range. c. The oxygen monitoring subsystem is capable of measuring oxygen concentration over the required range. d. Design of the sampling ports allow a representative containment atmosphere sample (drywell and wetwell) to be drawn for analysis.
2. Each safety-related CMS channel is electrically and physically separated from each other.	2. Inspections of the as-built system will be performed.	2. Each safety-related CMS channel is located in an area physically separated from each other. Also, each safety-related CMS channel is powered from separate safety-related electrical division.

Table 2.15.7-1

ITAAC For The Containment Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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. a. CMS conforms to the basic configuration as defined in Subsection 2.15.7. b. H2/O2 monitor to be in operation within [90] minutes, including warm-up time, after occurrence of an accident, which requires the monitor to be functional.
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 channel is powered from different divisional safety-related buses.	6. A test of the safety-related divisional power availability to each CMS channel will be conducted.	6. Each safety-related CMS channel receives electrical power from separate safety-related divisional electrical buses.
7. Each CMS subsystem is specified to meet environmental and radiological requirements for its location and intended post-accident operations.	7. Test and/or analysis are available to verify environmental and radiological requirements for each CMS component location and intended post-accident operations are met.	7. Vendor data and/or test report to provide indication of acceptable operation in the postulated post-accident environment in the installed location.

Table 2.15.7-1

ITAAC For The Containment Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Independence is provided between safety-related divisions and between safety-related divisions and non-safety related equipment.	8.	8.
	a. Test(s) will be performed to verify the electrical independence of each safety-related division.	a. Test report(s) exist(s)and conclude(s) that electrical independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment.
	b. An inspection will be performed to verify the physical independence of the as-installed safety-related divisions and the nonsafety-related equipment.	b. Report(s) exist(s) and conclude(s) that physical independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment.
	c. Test(s) will be performed to verify communication independence on each redundant network.	c. Report(s) exist(s) and conclude(s) that) that communication independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment.

Table 2.15.7-1

ITAAC For The Containment Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. The CMS is designed to ensure that, safety related system setpoints are defined, determined and implemented based on the setpoint methodology approved by NRC. This setpoint methodology includes the following as a minimum: a. The allowance for uncertainties between the process analytical limit and the device setpoint determined using a documented setpoint methodology.	9. Inspection, tests, and/or analysis will be performed to verify that all the setpoints of instruments associated with the CMS are in conformance with the design requirements.	9. Report(s) exist(s) and conclude(s) that the CMS is designed to ensure that, safety related system setpoints are defined, determined and implemented based on the setpoint methodology approved by NRC. This setpoint methodology includes the following as a minimum: a. The allowance for uncertainties between the process analytical limit and the device setpoint determined using a documented setpoint methodology.

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. Inspections of the as-built system configuration will be conducted.	1. The as-built SPTM conforms to the basic configuration as defined 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 safety-related power supply. In the SPTM, independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	3. a. Tests will be performed on the SPTM by providing a test signal in only one safety-related division at a time. b. Inspections of the as-built safety-related divisions in the SPTM will be performed.	3. a. A test signal exists only in the safety-related division under test in the SPTM. b. In the SPTM, physical separation or electrical separation exists between safety-related divisions. Physical separation or electrical isolation exists between these safety-related divisions and nonsafety-related 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.

Table 2.15.7-2

ITAAC For The Suppression Pool Monitoring

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Independence is provided between safety-related divisions and between safety-related divisions and non-safety related equipment.	5. a. Test(s) will be performed to verify the electrical independence of each safety-related division. b. An inspection will be performed to verify the physical independence of the as-installed safety-related divisions and the nonsafety-related equipment. c. Test(s) will be performed to verify communication independence on each redundant network.	5. a. Test report(s) exist(s)and conclude(s) that electrical independence is provided between safety-related divisions and between safety-related divisions and nonsafety-related equipment. b. Report(s) exist(s) and conclude(s) that physical independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment. c. Report(s) exist(s) and conclude(s) that) that communication independence exists between each of safety-related divisions and also between safety-related divisions and nonsafety-related equipment.

Table 2.15.7-2

ITAAC For The Suppression Pool Monitoring

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The CMS is designed to ensure that, safety related system setpoints are defined, determined and implemented based on the setpoint methodology approved by NRC. This setpoint methodology includes the following as a minimum: a. The allowance for uncertainties between the process analytical limit and the device setpoint determined using a documented setpoint methodology.	6. Inspection, tests, and/or analysis will be performed to verify that all the setpoints of instruments associated with the CMS are in conformance with the design requirements.	6. Report(s) exist(s) and conclude(s) that the CMS is designed to ensure that, safety related system setpoints are defined, determined and implemented based on the setpoint methodology approved by NRC. This setpoint methodology includes the following as a minimum: a. The allowance for uncertainties between the process analytical limit and the device setpoint determined using a documented setpoint methodology.

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 and FB cranes are classified as Seismic Class II and meet the requirements of NUREG-0612 and NUREG-0554.

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.

Cranes and other lifting devices are designed for their heaviest expected loads, and are load tested at 125% of the rated capacity.

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 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 each crane has a lifting capacity greater than its heaviest expected load.	1. Perform a load test at 125% of the rated capacity.	1. Each crane is successfully load tested at 125% of its rated capacity.
2. The RB and FB cranes are interlocked to prevent movement of heavy loads over spent fuel storage in the FB and over new or spent fuel in the RB.	2. Tests will be conducted of the as-built RB and FB crane movement using a heavy load.	2. The RB and FB crane interlock prevents the carrying of a load greater than one fuel assembly and its associated handling device over spent fuel storage in the FB and over new or spent fuel in the RB.
3. The basic configuration of the RB and FB cranes are as described in Subsection 2.16.1.	3. Procurement specifications apply key industry standards and appropriate NUREGs such as NUREG-0612 and NUREG-0554. Inspection of the as-built equipment will be performed.	3. The as-built RB and FB cranes comply with the description in Subsection 2.16.1.
4. Special Lifting Devices designed to be used in conjunction with the RB and FB cranes have sufficient lifting capacity.	4. Procurement specifications apply key industry standards and appropriate NUREGs such as NUREG-0612 and NUREG-0554. Perform a load test at 125% of the rated capacity.	4. Each lifting device is successfully load tested at 125% of its rated capacity.

2.16.2 Heating, Ventilating and Air-Conditioning Systems

2.16.2.1 Reactor Building HVAC

Design Description

The Reactor Building HVAC System (RBVS) serves the Reactor Building. Safety-related components for the RBVS are listed in Table 2.16.2-1.

With the following exception, the RBVS 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 RBVS performs no safety-related function except for automatic isolation of the Reactor Building boundary during accidents.

The RBVS consists of three subsystems. The Reactor Building Contaminated Area HVAC Subsystem (CONAVS) serves the potentially contaminated areas of the Reactor Building. The Reactor Building 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-radiologically controlled) areas of the Reactor Building.

The CONAVS and REPAVS maintain potentially contaminated areas of the building at a negative pressure with respect to adjacent clean areas to minimize exfiltration of potentially contaminated air. The RBVS maintains the hydrogen concentration levels in the battery rooms below 2% by volume. The reactor building exhaust is continuously monitored for radiological contamination prior to discharge to the plant vent stack.

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.

Instrumentation and Controls

The safety-related instrumentation and controls for RBVS are available in the main control room (MCR).

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.2-2 provides the design commitments, inspections, tests, analyses and acceptance criteria for the RBVS system.

2.16.2.2 Control Building HVAC System

Design Description

The Control Building HVAC consists of two independent subsystems. The Control Room Habitability Area HVAC Subsystem (CRHAVS) serves the MCR and associated areas bounded by the Control Room Habitability Area (CRHA) envelope. The Control Building General Area HVAC (CBGAVS) serves the areas inside the Control Building but outside the CRHA. Table 2.16.2-3 lists the major Control Building HVAC system safety-related components.

Both of these subsystems are nonsafety-related except for that portion of the CRHAVS that forms the CRHA boundary envelope, and the CRHAVS Emergency Filter Units (EFU) and associated components, which are safety-related. This safety-related CRHA boundary envelope consists of the CRHA structure, doors, penetrations, redundant boundary isolation dampers, valves, and that portion of transition ductwork, piping, or tubing that is located between the CRHA boundary structure and the redundant CRHA isolation dampers or valves. The CRHA isolation dampers are the major components discussed in this Subsection. Additional systems, structures, and components (such as EFUs) that are necessary for habitability are discussed in other subsections. The mechanical cooling of the Control Building General Areas and the CRHA is not provided as a safety-related function during a CRHA boundary isolation. Passive means of limiting CRHA and general area temperature rise to acceptable levels have been provided by the ESBWR design.

The CRHAVS serves the MCR and associated support areas during normal plant operations, plant start-up and plant shutdown. It consists of redundant AHU's, CRHA isolation dampers, system dampers, electric heaters and EFUs. A simplified schematic of the CRHAVS is provided in Figure 2.16.2-4.

The CBGAVS serves the areas outside the CRHA. It also consists of redundant air handling units, return/exhaust fans, electric heaters and system dampers. A simplified schematic of CBGAVS is provided in Figures 2.16.2-5 and 2.16.2-6.

Instrumentation and Controls

The safety-related instrumentation and controls for Control Building HVAC are available in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.2-4 provides definitions of the inspections, test and/or analyses, together with associated acceptance criteria for the Control Building HVAC.

2.16.2.3 Emergency Filter Units

Design Descriptions

The Emergency Filter Units (EFU) supply pressurized breathing air to the Control Room Habitability Area (CRHA) during isolation of the CRHA boundary envelope. The EFUs are safety-related and maintain habitable conditions in the CRHA to ensure the safety of the control room operators. An EFU is automatically initiated upon CRHA isolation to provide breathing air and pressurization of the CRHA to minimize infiltration. There are two independent redundant EFU trains capable of supplying sufficient air and CRHA pressurization for up to 21 operators

for 72 hours. The EFUs are part of the CRHAVS, and a simplified system diagram is provided in Figure 2.16.2-4. Design information on safety-related equipment is provided in Table 2.16.2-5.

Instrumentation and Controls

The safety-related instrumentation and controls for the EFUs are available in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.2-6 provides the design commitments, inspections, tests, analyses and acceptance criteria for the EFUs.

2.16.2.4 Turbine Building HVAC System

Design Description

The Turbine Building Ventilation System (TBVS) is nonsafety-related. The system includes an intake plenum, dampers, heating and cooling coils and supply fans. Redundant exhaust fans are also provided. Turbine building exhaust is directed to the plant vent stack where it is monitored for radiation prior to being discharged to the atmosphere. Local unit coolers and fans are provided in areas with high local heat loads. The Turbine Building Ventilation System is designed to minimize exfiltration of air to adjacent areas by maintaining a slightly negative pressure in the Turbine Building (by exhausting more air than is supplied) relative to adjacent areas.

Instrumentation and Controls

The instrumentation and controls for TBVS isolation are available in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

No entree for this system.

2.16.2.5 Fuel Building HVAC System

Design Description

The Fuel Building HVAC system (FBVS) does not perform any safety-related functions, except for automatic isolation of the Fuel Building boundary during accidents, and thus, the system is classified as nonsafety-related. The FBVS maintains the fuel building at a minimum negative pressure of 62 Pa (-1/4 inch W.G.) relative to surrounding areas to minimize exfiltration of potentially contaminated air. Fuel Building HVAC subsystems include the Fuel Building General Area HVAC Subsystem (FBGAVS) and the Fuel Building Fuel Pool HVAC Subsystem (FBFPVS). Both Subsystems consist of redundant air handling units, exhaust fans, electric heaters, and dampers. A simplified schematic of FBGAVS is provided in Figure 2.16.2-7. A simplified schematic of FBFPVS is provided in Figure 2.16.2-8. Safety-related components for the FBVS are listed in Table 2.16.2-7.

Instrumentation and Controls

The safety-related instrumentation and controls for Fuel Building HVAC are available in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.2-7 provides the design commitments, inspections, tests, analyses and acceptance criteria for the Fuel Building HVAC.

2.16.2.6 Radwaste Building HVAC System

Design Description

The two Radwaste Building HVAC (RWVS) subsystems are the Radwaste Building Control Room HVAC (RWCRVS) and the Radwaste Building General Area HVAC Subsystem (RWGAVS). The combined function of these two systems is to provide a controlled environment for personnel comfort and for proper operation and integrity of equipment. The RWGAVS maintains the Radwaste Building general area at a slight negative pressure relative to adjacent areas and outside atmosphere so as to prevent the exfiltration of air to adjacent areas. The Radwaste Building exhaust is monitored prior to discharging it through the plant vent stack.

Instrumentation and Controls

Instrumentation and controls for Radwaste Building HVAC are located at a local panel in the Radwaste Building. Selected HVAC alarm signals are transmitted to a general trouble alarm in the Radwaste Building Control Room.

Inspections, Tests, Analyses and Acceptance Criteria

No entree for this system.

2.16.2.7 Electrical Building HVAC System

Design Description

The Electrical Building HVAC System (EBVS) includes three subsystems. The Electric and Electronic Rooms HVAC Subsystem (EERVS), the Technical Support Center HVAC Subsystem (TSCVS), and the Diesel Generators HVAC Subsystem (DGVS). Each of these systems provides adequate ventilation and conditioned air for equipment and personnel comfort. In addition, the EERVS limits the buildup of hydrogen in the nonsafety-related battery rooms to less than 2% hydrogen by volume. The TSCVS normally maintains the TSC at a slightly positive pressure with respect to the adjacent rooms and the outside environment to minimize the infiltration of air. The EBVS does not perform or ensure any safety-related functions. The major areas served by the EERVS are Electric and Electronic Rooms and Battery Rooms. The major areas served by TSCVS are the TSC, General Rooms/Areas, the kitchen, and toilet facilities. The major areas served by the DGVS are the DG Rooms, Electronic Equipment Areas, and Diesel Oil Day Tank Rooms.

Instrumentation and Controls

The functional instrumentation and controls for EBVS are available in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for this system.

2.16.2.8 Other Building HVAC Systems

Design Description

Ventilation systems for other buildings include the 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

The functional information is available in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

No entry for other HVAC systems.

Table 2.16.2-1

Reactor Building HVAC System Safety-Related Components

Component	Seismic Category	ASME Code Classification	Fail Safe Position
CONAVS building supply air isolation dampers	I	N/A	Closed
CONAVS building exhaust air isolation dampers	I	N/A	Closed
REPAVS building supply air isolation dampers	I	N/A	Closed
REPAVS building exhaust air isolation dampers	I	N/A	Closed

Table 2.16.2-2

ITAAC For The Reactor Building HVAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the RBVS is as described in Subsection 2.16.2.1.	1. Inspections of the RBVS configuration will be conducted.	1. The as-built system conforms to the description in Subsection 2.16.2.1.
2. The RBVS 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 performed on the (RBVS isolation dampers) isolation logic.	2. Upon receipt of a simulated signal, the RBVS isolation dampers automatically close.
3. The safety-related components (building isolation dampers and associated controls) can withstand Seismic Category I loads without loss of safety-related function.	3. a. Type tests, analyses, or a combination of type tests and analyses of Seismic Category I equipment will be performed. b. 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. a. A report exists and concludes that the equipment can withstand seismic design basis without loss of safety-related function. b. A report exists and concluded that the as-installed equipment including anchorage is seismically bounded by testing or analyzed conditions.
4. The RBVS provides cooling to the safety-related battery rooms and safety-related electrical equipment rooms.	4. Testing will be performed on the components using the controls in the MCR.	4. Controls in the MCR cause the components to perform the required function.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. CONAVS maintains served areas of the reactor building at a minimum negative pressure of 62 Pa (-1/4 inch W.G.) relative to surrounding clean areas to minimize the exfiltration of potentially contaminated air.	5. a. Testing will be performed to confirm that the contaminated areas of the reactor building served by CONAVS maintains a minimum negative pressure of 62 Pa (-1/4 inch W.G.) when operating CONAVS supply and exhaust fans in the normal system fan lineup. a. Testing will be performed to confirm the ventilation flow rate through the contaminated areas of the reactor building served by CONAVS when operating CONAVS supply and exhaust fans in the normal system fan lineup.	5. a. The time average pressure differential in the CONAVS served areas of the reactor building as measured by each of the pressure differential indicators is minimum negative pressure of 62 Pa (-1/4 inch W.G.). b. Test report indicating the exhaust flow rate is greater than or equal to the CONAVS supply flow rate.
6. REPAVS maintains served areas of the reactor building at a minimum negative pressure of 62 Pa (-1/4 inch W.G.) relative to surrounding clean areas to minimize the exfiltration of potentially contaminated air.	6. a. Testing will be performed to confirm that the refueling area of the reactor building served by REPAVS maintains a minimum negative pressure of 62 Pa (-1/4 inch W.G.) when operating REPAVS supply and exhaust fans in the normal system fan lineup. b. Testing will be performed to confirm the ventilation flow rate through the refueling area of the reactor building served by REPAVS when operating REPAVS supply and exhaust fans in the normal system fan lineup.	6. a. The time average pressure differential in the REPAVS served areas of the reactor building as measured by each of the pressure differential indicators is minimum negative pressure of 62 Pa (-1/4 inch W.G.). b. Test report indicating the exhaust flow rate is greater than or equal to the REPAVS supply flow rate.

Table 2.16.2-3

Control Building HVAC System Safety-Related Components

Component	Seismic Category	ASME Code Classification	Notes
CRHA supply air isolation dampers	I	N/A	Fail Closed
EFU downstream isolation dampers	I	N/A	Fail Closed
CRHA Restroom Exhaust isolation dampers	I	N/A	Fail Closed
CRHA Smoke Exhaust intake isolation dampers	I	N/A	Fail Closed
CRHA Smoke Exhaust output isolation dampers	I	N/A	Fail Closed
CRHAVS EFUs	I	N/A	N/A

Table 2.16.2-4

ITAAC For The Control Building Habitability HVAC Subsystem

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the CRHAVS is as described in Subsection 2.16.2.2.	1. Inspections of the CRHAVS configuration will be conducted.	1. The as-built system conforms to the description in Subsection 2.16.2.2.
2. The CRHA isolation dampers automatically close upon receipt of a high radiation signal from PRMS, i.e. a. high radiation in the CRHAVS intake; b. high radiation downstream of an Emergency Filter Unit (EFU) during emergency operation, and c. low airflow through an EFU during emergency operation, or d. loss of AC power.	2. Using simulated high radiation isolation signals, tests will be performed on the (CRHA isolation dampers) isolation logic. A loss of AC power test will be performed.	2. Upon receipt of each simulated isolation signal; a. high radiation in the CRHAVS intake, b. high radiation downstream of the Emergency Filter Unit (EFU) during emergency operation, and c. low airflow through an EFU during emergency operation, or d. loss of AC power, the CRHA isolation dampers automatically close.
3. The safety-related components (EFUs, CRHA isolation dampers and associated components, instrumentation and controls) can withstand Seismic Category I loads without loss of safety-related function.	3. a. Type tests, analyses, or a combination of type tests and analyses of safety-related Seismic Category I equipment will be performed. b. 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. a. A report exists and concludes that the equipment can withstand seismic design basis without loss of safety-related function. b. A report exists and concluded that the as-installed equipment including anchorage is seismically bounded by testing or analyzed conditions.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The CRHAVS provides cooling to the CRHA.	4. Testing will be performed on the components using the controls in the MCR.	4. Controls in the MCR cause the components to perform the required function.
5. Independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	5. a. Tests will be performed on CRHA isolation damper and EFU operation by providing a test signal in only one safety-related division at a time. b. Inspection of the as-installed safety-related divisions in the system will be performed.	5. a. The test signal exists only in the safety-related division under test in CRHA isolation damper and EFU control. b. Physical separation or electrical isolation exists between CRHA isolation dampers and EFU safety-related divisions. Physical separation or electrical isolation exists between safety-related divisions and nonsafety-related equipment.
6. Instrumentation showing the status of CRHA isolation damper and EFU operational status (Open/Closed) indication will be installed in the MCR.	6. a. Inspection will be performed to verify CRHA isolation damper and EFU operational status indication is installed in the MCR. b. Testing will be performed to show that the operational status indication in the MCR accurately depicts the operational status of the CRHA isolation dampers and EFUs.	6. a. The CRHA isolation damper and EFU operational status indication located in the MCR. b. A report exists and concludes that the operational status indication accurately depicts the operational status of the CRHA isolation dampers and EFUs.

Table 2.16.2-5

Emergency Filter Units

Component	Seismic Category	ASME Code	Notes
EFU	I	AG-1	Minimum flow rate of 200 l/s (9.5 l/s per person for up to 21 persons), Independent trains with N-1 redundancy
EFU dampers	I	AG-1	Redundant dampers in each independent train

Table 2.16.2-6

ITAAC For Emergency Filter Units

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the EFU is as described in Subsection 2.16.2.3	1. Inspections of the EFU configuration will be conducted.	1. The as-built system conforms with the description in Subsection 2.16.2.3
2. The selected redundant EFU dampers open upon receipt of a control room habitability envelope isolation signal.	2. Using simulated isolation signals, tests will be performed on the isolation logic.	2. Upon receipt of a simulated isolation signal, selected EFU dampers automatically open.
3. The safety-related EFU components can withstand Seismic Category I loads without loss of safety-related function.	3. a. Type tests, analyses, or a combination of type tests and analyses of safety-related Seismic Category I equipment will be performed. b. 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. a. A report exists and concludes that the equipment can withstand seismic design basis without loss of safety-related function. b. A report exists and concluded that the as-installed equipment including anchorage is seismically bounded by testing or analyzed conditions.

Table 2.16.2-6

ITAAC For Emergency Filter Units

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Independence for the EFU trains is provided between safety-related Divisions, and between safety-related Divisions and nonsafety-related equipment.	4. a. Tests will be performed on EFUs by providing a test signal in only one safety-related Division at a time. b. Inspection of the as installed safety-related Divisions in the System will be performed.	4. a. The test signal exists only in the safety-related Division under test. b. For the EFU trains, physical separation or electrical isolation exists between these safety-related Divisions. Physical separation or electrical isolation exists between safety-related Divisions and nonsafety-related equipment.
5. EFUs are capable of providing 200 l/s (9.5 l/s per person for up to 21 control room occupants for 72 hours).	5. Inspections will be performed to verify that adequate breathing air capacity is provided.	5. a. EFU flow rate ≥ 200 l/s
6. a. EFUs maintain the CRHA at a minimum positive pressure of 31 pascals (0.125 inch water gauge) with respect to the surrounding areas at the required air addition flow rate of 200 l/s (424 scfm). b. The in-leakage does not exceed the assumed unfiltered in-leakage assumed by control room operator dose analysis.	6. a. Testing will performed to measure the differential between the CRHA and surrounding areas. b. Tracer gas testing in accordance with ASTM E741 will be performed to measure the in-leakage into the CRHA with EFUs operating.	6. a. A minimum positive pressure of 31 pascals (0.125 inch water gauge) with respect to the surrounding areas is maintained with EFU operating. b. The in-leakage measured by tracer gas testing does not exceed the assumed unfiltered in-leakage assumed by control room operator dose analysis.

Table 2.16.2-6

ITAAC For Emergency Filter Units

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. a. Temperature rise on a loss of normal cooling will not exceed 8.3°C (15°F) for 72 hours. b. The CRHA heat sink is maintained at or below 25.56°C (78°F).	7. a. The temperature rise in the CRHA will be analyzed. The internal heat loads will be verified to be no greater than that assumed in the analysis. b. The CRHA air temperature will be analyzed to be maintained at or below the maximum assumed initial air and heat sink temperature.	7. a. The internal heat loads will be verified to be no greater than that assumed in the analysis. b. The average ambient air temperature in the CRHA is ≤ 25.56°C (78°F).
8. The powered EFU dampers can be remotely operated from the MCR.	8. EFU dampers will be opened and closed using manually initiated signals from the MCR.	8. The EFU dampers open and close when manually imitated signals are sent from the MCR.
9. The powered EFU dampers identified as having Distributed Control and Information System (DCIS) control perform their active function.	9. Tests will be performed to verify that EFU dampers identified as having DCIS control perform their active function.	9. The EFU dampers having DCIS control perform their safety function when simulated DCIS signals are initiated.
10. The EFU fans can be remotely operated from the MCR.	10. EFU fans will be started and stopped using manually initiated signals from the MCR.	10. The EFU fans start and stop when manually initiated signals are sent from the MCR.
11. EFUs meet the in-place leakage testing requirements of ASME AG-1 and RG 1.52.	11. EFUs will be in-place leak tested in accordance with ASME AG-1, Section TA, to meet the requirements of RG 1.52.	11. EFUs meet the acceptance criteria for in-place testing per RG 1.52 when tested in accordance with RG 1.52 and ASME AG-1, Section TA.

Table 2.16.2-7

Fuel Building HVAC System Safety-Related Components

Component	Seismic Category	ASME Code Classification	Fail Safe Position
FBGAVS building supply air isolation dampers	I	N/A	Closed
FBGAVS building exhaust air isolation dampers	I	N/A	Closed
FBFPVS building supply air isolation dampers	I	N/A	Closed
FBFPVS building exhaust air isolation dampers	I	N/A	Closed

Table 2.16.2-8

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 2.16.2.5.	1. Inspections of the Fuel Building HVAC configuration will be conducted.	1. The as-built system conforms to the description in Subsection 2.16.2.5.
2. The Fuel Building HVAC isolation dampers automatically close upon receipt of an isolation signal.	2. Using simulated isolation signals, tests will be performed 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 safety-related components (building isolation dampers and associated controls) can withstand Seismic Category I loads without loss of safety-related function.	3. a. Type tests, analyses, or a combination of type testes and analyses of safety-related Seismic Category I equipment will be performed. b. 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. a. A report exists and concludes that the equipment can withstand seismic design basis without loss of safety-related function. b. A report exists and concluded that the as-installed equipment including anchorage is seismically bounded by testing or analyzed conditions.

Table 2.16.2-8

ITAAC For The Fuel Building HVAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The FBVS maintains the fuel building at a minimum negative pressure of 62 Pa (-1/4 inch W.G.) relative to surrounding areas.	4. a. Testing will be performed to confirm that the FBVS maintains a minimum negative pressure of 62 Pa (-1/4 inch W.G.) when operating FBVS supply and exhaust AHUs in the normal system fan lineup b. Testing will be performed to confirm the ventilation flow rate through the fuel building area when operating the FBVS supply and exhaust fans in the normal system fan lineup.	4. a. The time average pressure differential in the served areas of the fuel building as measured by the pressure differential indicators is a minimum negative pressure of 62 Pa (-1/4 inch W.G.). b. Test report indicating the exhaust flow rate is greater than or equal to the FBVS supply flow rate.

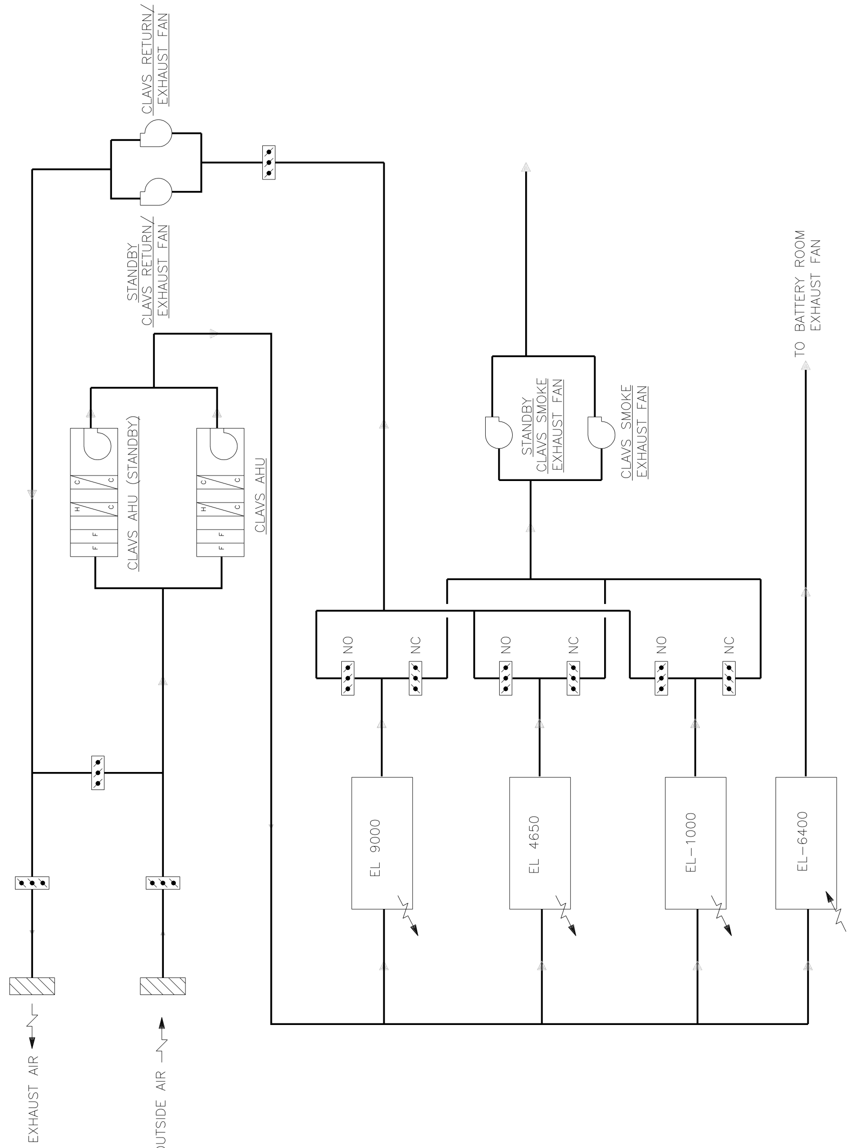


Figure 2.16.2-1. CLAVS Simplified System Diagram

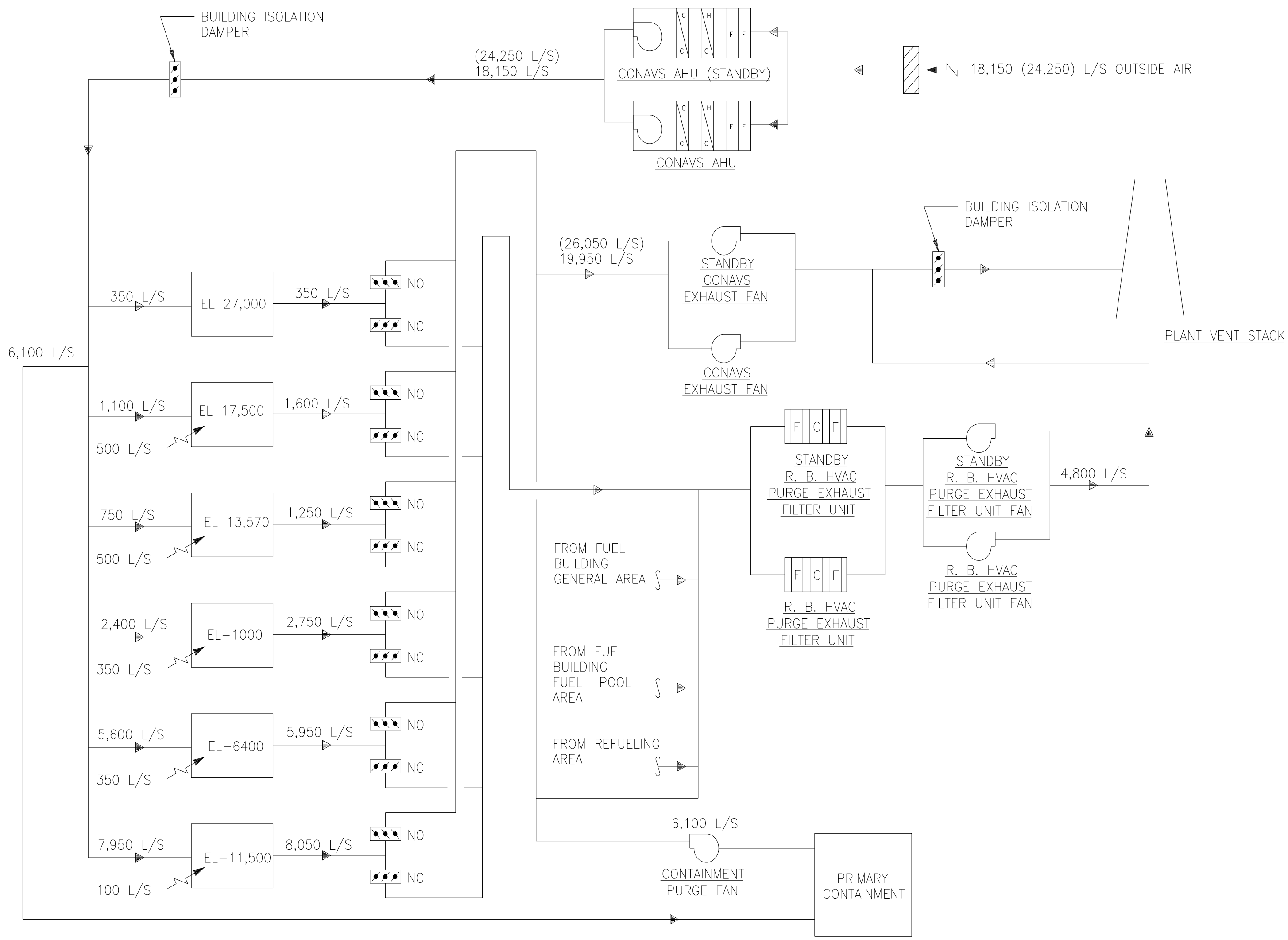


Figure 2.16.2-2. CONAVS Simplified System Diagram

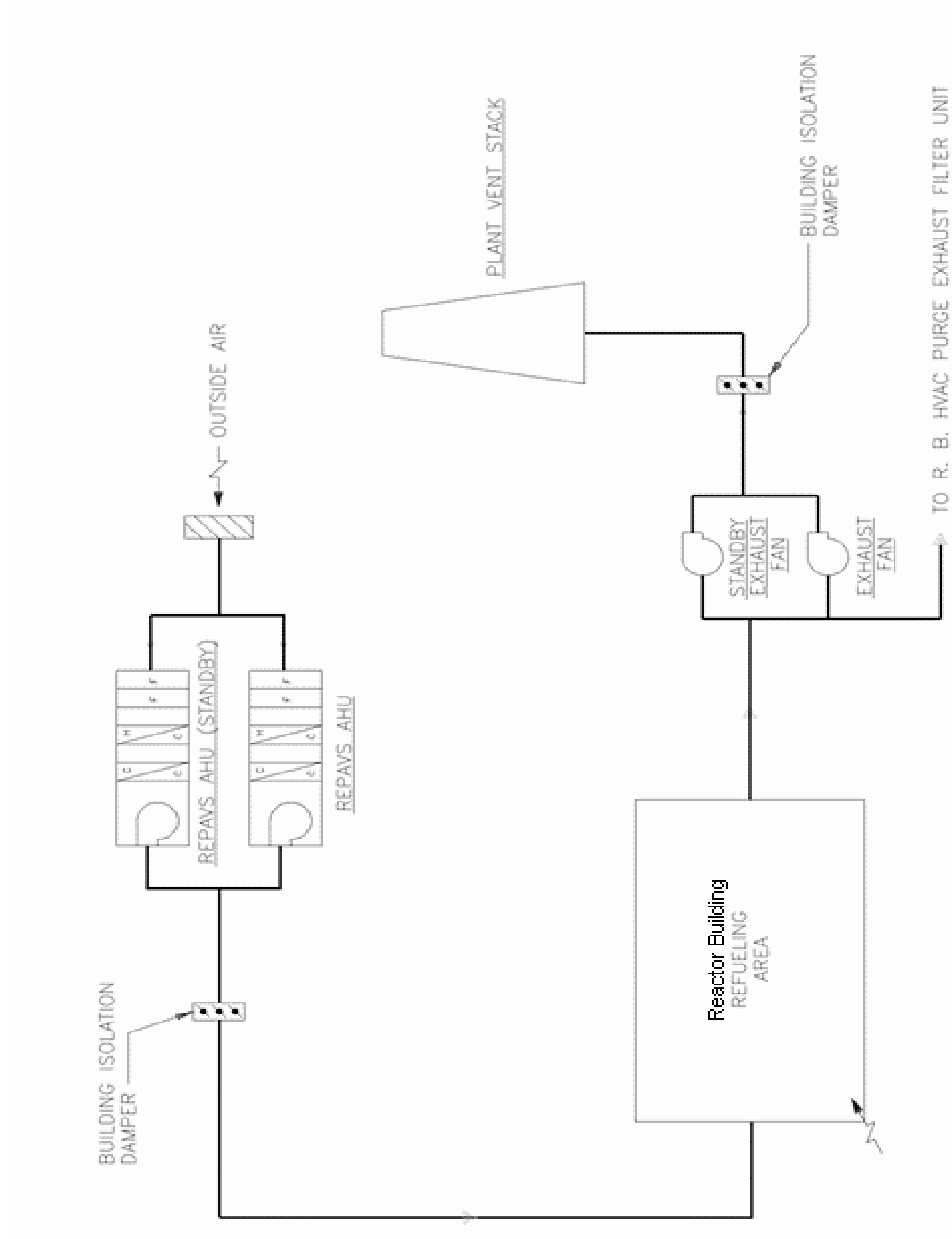


Figure 2.16.2-3. REPAVS Simplified System Diagram

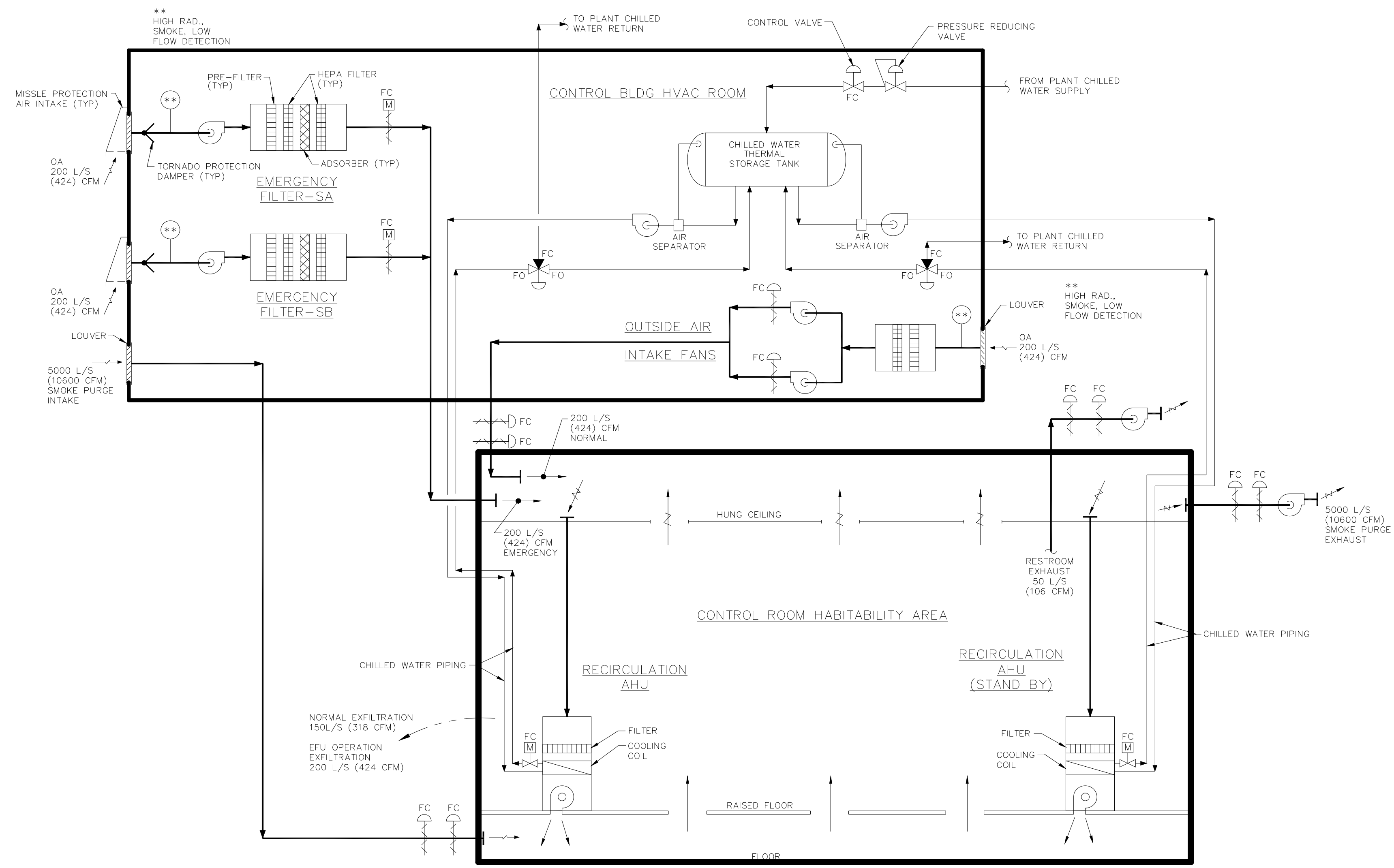


Figure 2.16.2-4. CRHAVS Simplified System Diagram

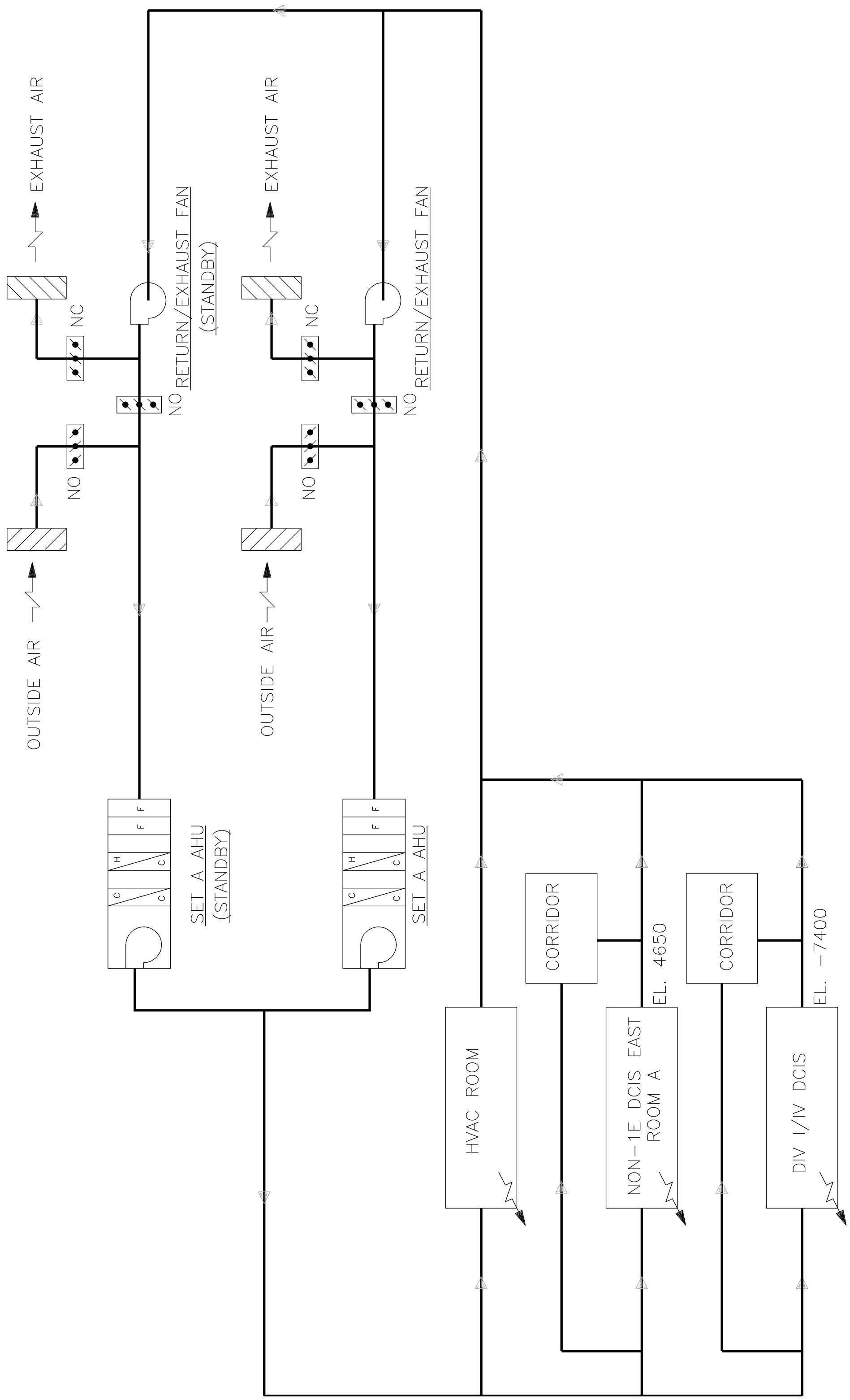


Figure 2.16.2-5. CBGAVS (Set A) Simplified System Diagram

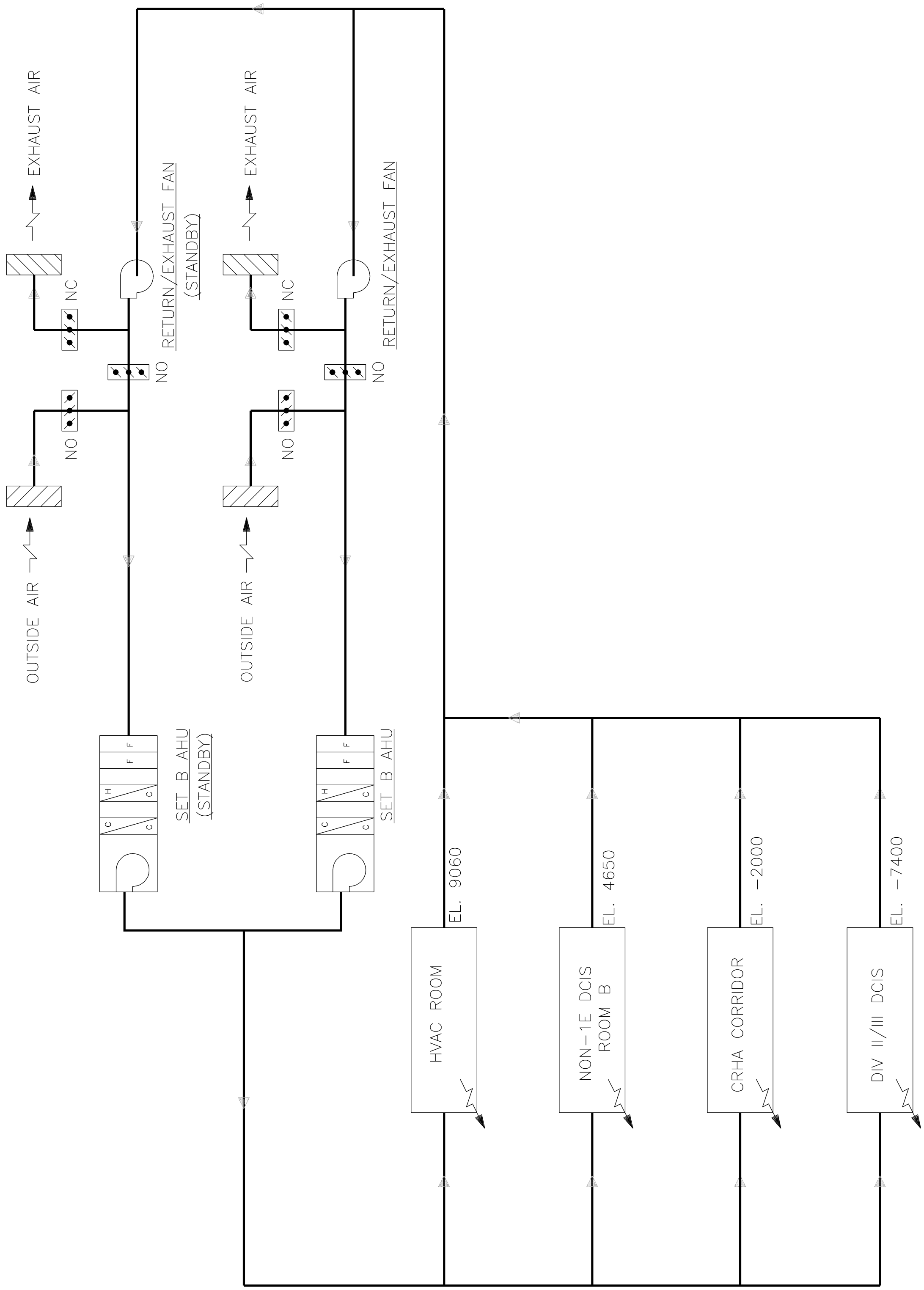


Figure 2.16.2-6. CBGAVS (Set B) Simplified System Diagram

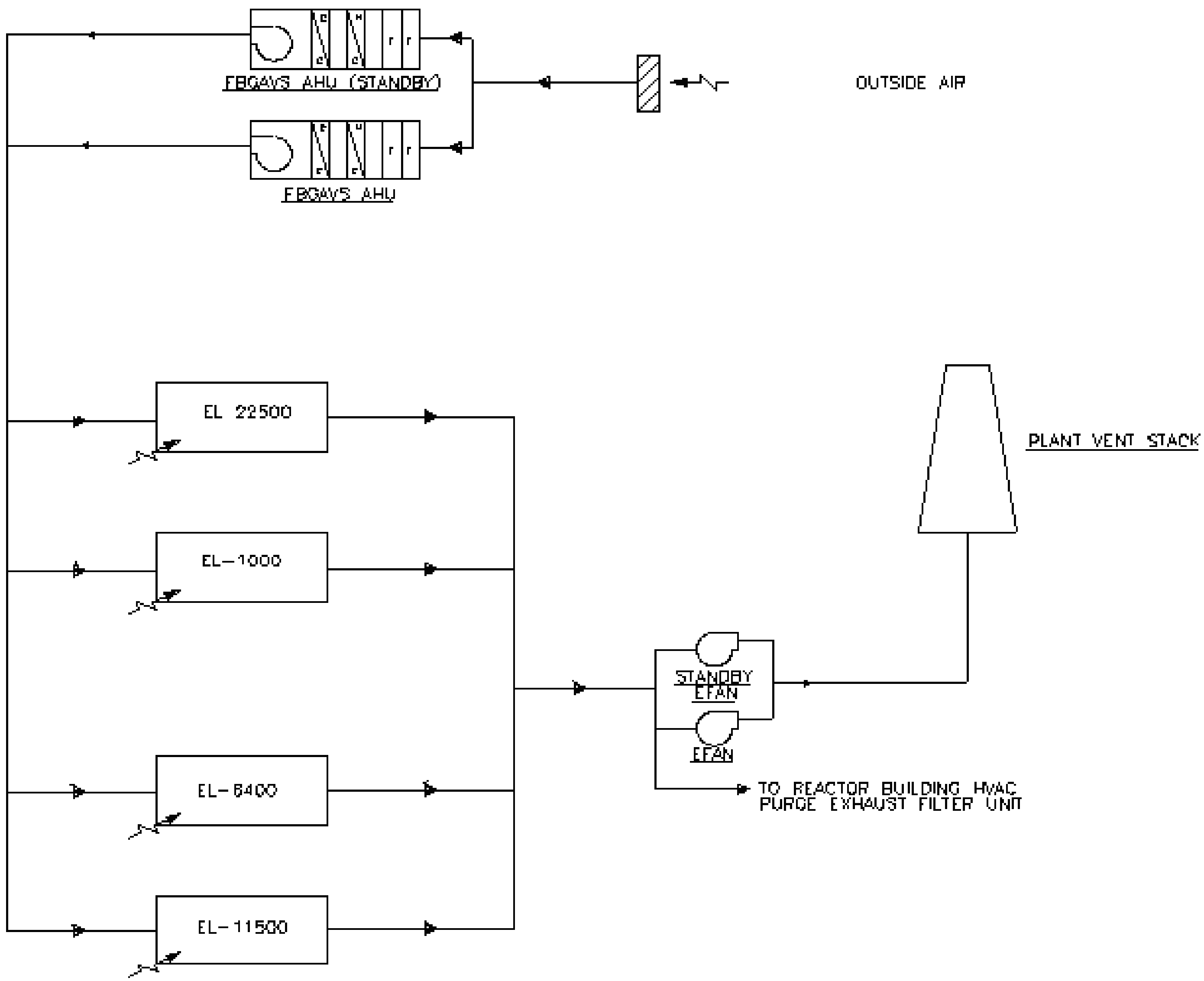


Figure 2.16.2-7. FBGAVS Simplified System Diagram

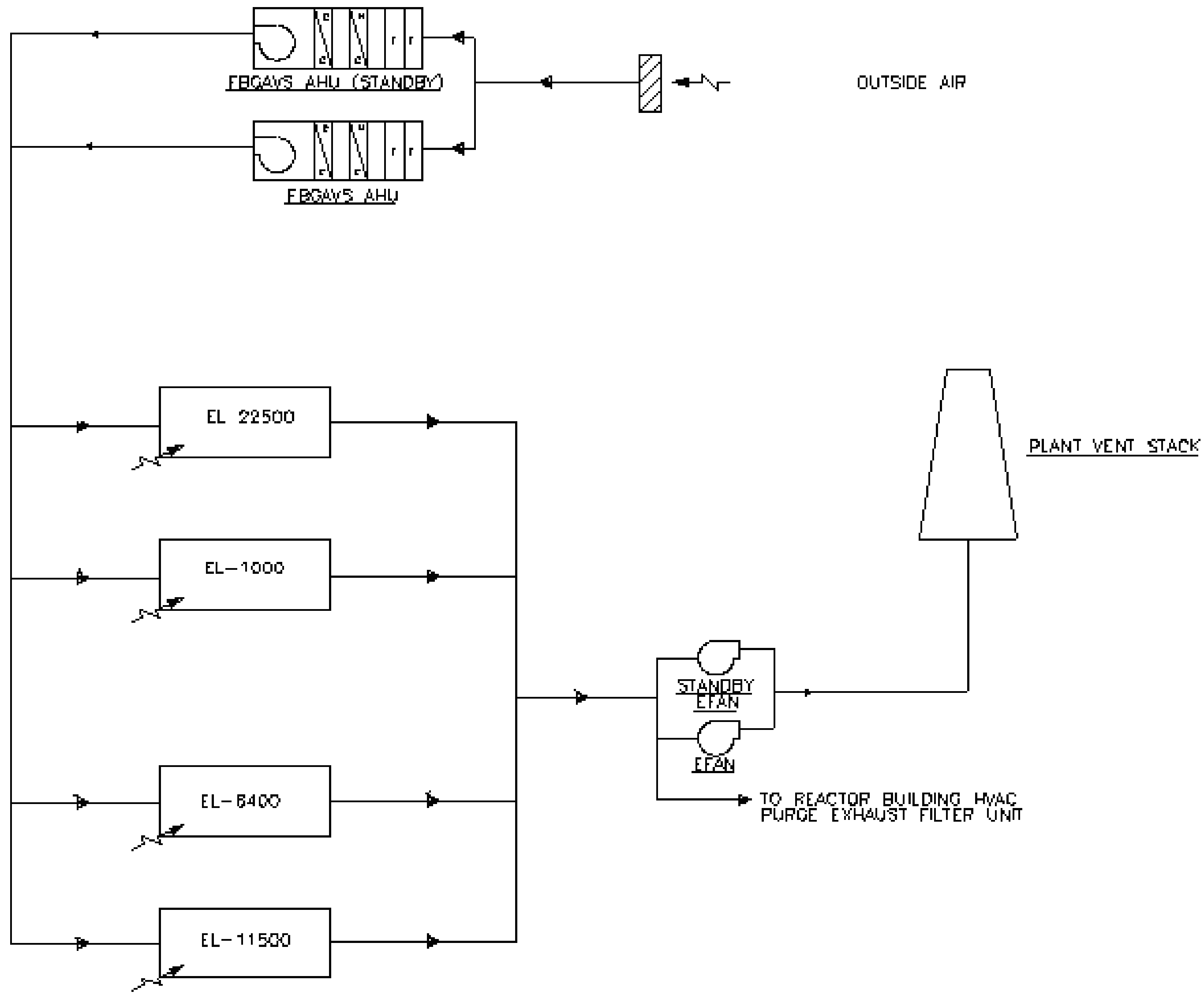


Figure 2.16.2-8. FBFPVS Simplified System Diagram

2.16.3 Fire Protection System

Design Description

The Fire Protection System (FPS) is nonsafety-related, and thus, does not require a safety-related power source. However, because of nonsafety-related to safety-related interfaces, some equipment (see Table 2.16.3-1) has elevated seismic and quality classifications. It 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. Table 2.16.3-2 provides the important component design characteristics.

Two primary fire pumps each provide 100% of the firewater demand to the worst-case fire within the nuclear island (Reactor Building, Fuel Building, and Control Building) and 50% of the firewater demand to the worst-case fire within the Turbine Building (TB). Two secondary fire pumps each provide 50% of the firewater demand for the worst-case fire in the TB and 100% of the firewater demand for the worst-case fire in the remainder of the balance of plant (BOP). All fire pumps are capable of delivering the flow and pressure required to the location that is farthest from the firewater supply source. The two primary fire pumps are located near the nuclear island power block in a fire pump enclosure (FPE). The two secondary fire pumps are located remote from the other two pumps to avoid any common-location failures. For the two primary nuclear island fire pumps, the lead fire pump is motor-driven and the backup is a Seismic Category I diesel driven fire pump. The backup diesel-driven fire pump provides firewater in the event of failure of the motor-driven fire pump or loss of preferred power (LOPP).

For the two secondary fire pumps, the lead fire pump shall be motor-driven and the backup fire pump shall be Non-Seismic (NS) diesel-driven. The secondary diesel-driven fire pump provides firewater in the event of failure of the motor-driven fire pump.

The fuel oil tanks for the diesel-driven fire pumps have sufficient capacity to allow diesel engine operation for approximately 96 hours before refilling based upon the fuel consumption and margin criteria provided in NFPA 24.

The combustible loading limit for electrical areas is conservatively determined as 1400 MJ/m^2, and the combustible loading limit for all other indoor areas is conservatively determined as 700 MJ/m^2. Rooms that exceed these limits require automatic fire suppression.

The firewater supply piping consists of a buried non-seismic, yard main loop and a suspended ASME B31.1, Seismic Category I nuclear island piping loop. The Seismic Category I loop is designed to remain functional following a SSE. The primary fire pumps supply firewater to the Seismic Category I, loop, supplying firewater within the structures of the nuclear island (Reactor Building, Control Building, and Fuel Building). The secondary fire pumps supply firewater directly to the yard main loop. Isolation valves are provided between the buried, non-seismic, yard piping loop and the suspended ASME B31.1 Seismic Category I piping loop.

The FPS can perform a nonsafety-related defense-in-depth function of being backup source of makeup water (through a piping connection to the Fuel and Auxiliary Pools Cooling System) 72 hours after a LOCA for IC/PCCS pools and the spent fuel pool and reactor water inventory control.

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.

Commission papers SECY 90-016, SECY 93-087 and SECY 94-084 provide enhanced fire protection criteria for advanced reactor designs. These criteria are directed toward plants with active safety-related systems, however, within the constraints of the active-to-passive design differences, the ESBWR design meets the intent of those criteria.

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 MCR. In addition to automatic operation any of the fire pumps can be manually started either from MCR or local panels.

Instrumentation for the Fire Detection System:

Instrumentation for the fire detection system provides signals for early detection and warning of fires. Local fire alarm panels per National Fire Protection Association (NFPA) 72 supervise fire and smoke detectors. The local fire alarm panels are in turn connected to the main fire alarm panel (MFAP) via a dedicated data link. Signals transmitted include detector status (normal, alarm, supervisory, trouble) as well as local fire alarm panel status.

Upon receipt of a signal from any of the area fire detectors, alarms and visual indications are activated at the MFAP in the MCR and at the local fire alarm panel.

Instrumentation for fire detection is either Factory Mutual (FM) approved or Underwriter Labs (UL) listed, where available.

Instrumentation Supporting Fire Suppression Systems:

Each fire suppression system automatically actuated by a fire detection system has the control logic and capability for manual actuation available at the local fire alarm panel for the protected area. Remote manual actuation of these suppression systems is also available from the MCR. Automatic sprinkler systems that do not require separate detection systems for actuation are not equipped with manual actuation means.

Instrumentation for fixed fire suppression systems provides local and remote monitoring capability for the suppression system status. All instruments for automatic suppression systems are wired to the local fire alarm panels for control. Dedicated data links transmit command and status information to and from the local fire alarm panels and the MFAP in the MCR.

All instrumentation for automatically actuated fire suppression systems is either FM approved or UL listed, as specified in NFPA code.

Instrumentation Supporting Fire Water Delivery:

Instrumentation supporting firewater delivery provides status indication of fire water tank level, fire water main pressure, jockey pump status, and main fire pump status conditions.

When a portion of the firewater system activates, the motor-driven fire pump automatically starts on low-pressure. If the motor-driven pump fails to start or cannot maintain pressure, the main diesel-driven pump starts from a different pressure switch. The second diesel-driven pump is designed to start last if the two main pumps fail to start or cannot maintain the required system pressure. All pumps are stopped manually. Any pump can be started manually from the MFAP in the MCR or locally.

A pressure switch is used to automatically start and stop the motor-driven jockey pump.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.3-3 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria for the Fire Protection System.

Table 2.16.3-1

Fire Protection System Equipment Classifications

Principal Components	Safety Class.	Location(s)	Quality Group	Seismic Category
Non-seismic yard piping loop and valves including supports	N	OO, OL	D	NS
Seismic Category I piping loop and valves including supports	N	OO, RB, CB, FB	D	I
Fire water storage tank	N	OO	D	I
Fire pump enclosure	N	OO	—	II
Seismic Category I pump including diesel-engine drive	N	OO	D	I
Other pumps and motors	N	OO	D	NS
Electrical modules and cables for RB pre-action sprinklers	N	RB	—	NS
All other electrical modules and cables	N	ALL	—	NS
CO_2 actuation modules	N	TB	—	NS
Sprinklers	N	RB, TB, RW, SB, EB, OL	D	NS
Foam, pre-action or deluge	N	EB, TB, OO	—	NS

Location codes: ALL = all

CV	=	Containment Vessel	RW	=	Radwaste Building
CB	=	Control Building	CP	=	Circulating Water Pump House
RB	=	Reactor Building	SF	=	Service Water Building
OO	=	Outdoors Onsite	TB	=	Turbine Building
OL	=	Any Other Location	EB	=	Electrical Building
FB	=	Fuel Building	SB	=	Services Building

Table 2.16.3-2

FPS Component Design Characteristics

Firewater Pumps	
Primary motor-driven fire pump	454.2 m^3/hr (2,000 gpm) ***
Primary diesel-driven fire pump	454.2 m^3/hr (2,000 gpm) ***
Secondary motor-driven fire pump*	454.2 m^3/hr (2,000 gpm) ***
Secondary diesel-driven fire pump*	454.2 m^3/hr (2,000 gpm) ***
Primary motor-driven jockey pump	4.54 m^3/hr (20 gpm) minimum as required to maintain the NI firewater loop pressure 34.4 kPa (5 psi) above the start pressure of the fire pumps
Secondary motor-driven jockey pump	4.54 m^3/hr (20 gpm) minimum as required to maintain the yard main loop pressure 34.4 kPa (5 psi) above the start pressure of the fire pumps
Required minimum total makeup flow rate to IC/PCC and spent fuel pools at 72 hours into an event	46 m^3/hr (200 gpm)
Firewater Storage	
Primary storage tanks combined minimum usable firewater storage	3900 m^3 (1,030,000 gallons)
Secondary storage minimum firewater storage**	2081.8 m^3 (550,000 gallons)

* Secondary fire pump may be new or existing depending upon available site-specific provisions.

** Secondary firewater storage may be a tank, cooling tower basin, or a large body of water depending upon available site-specific provisions. Storage volume listed is the minimum storage volume to be dedicated for fire protection use.

*** Based on 50% of the largest firewater demand of 967 m^3/hr (4256 gpm) for Turbine Building, including hose stream.

Table 2.16.3-3

ITAAC For The Fire Protection System

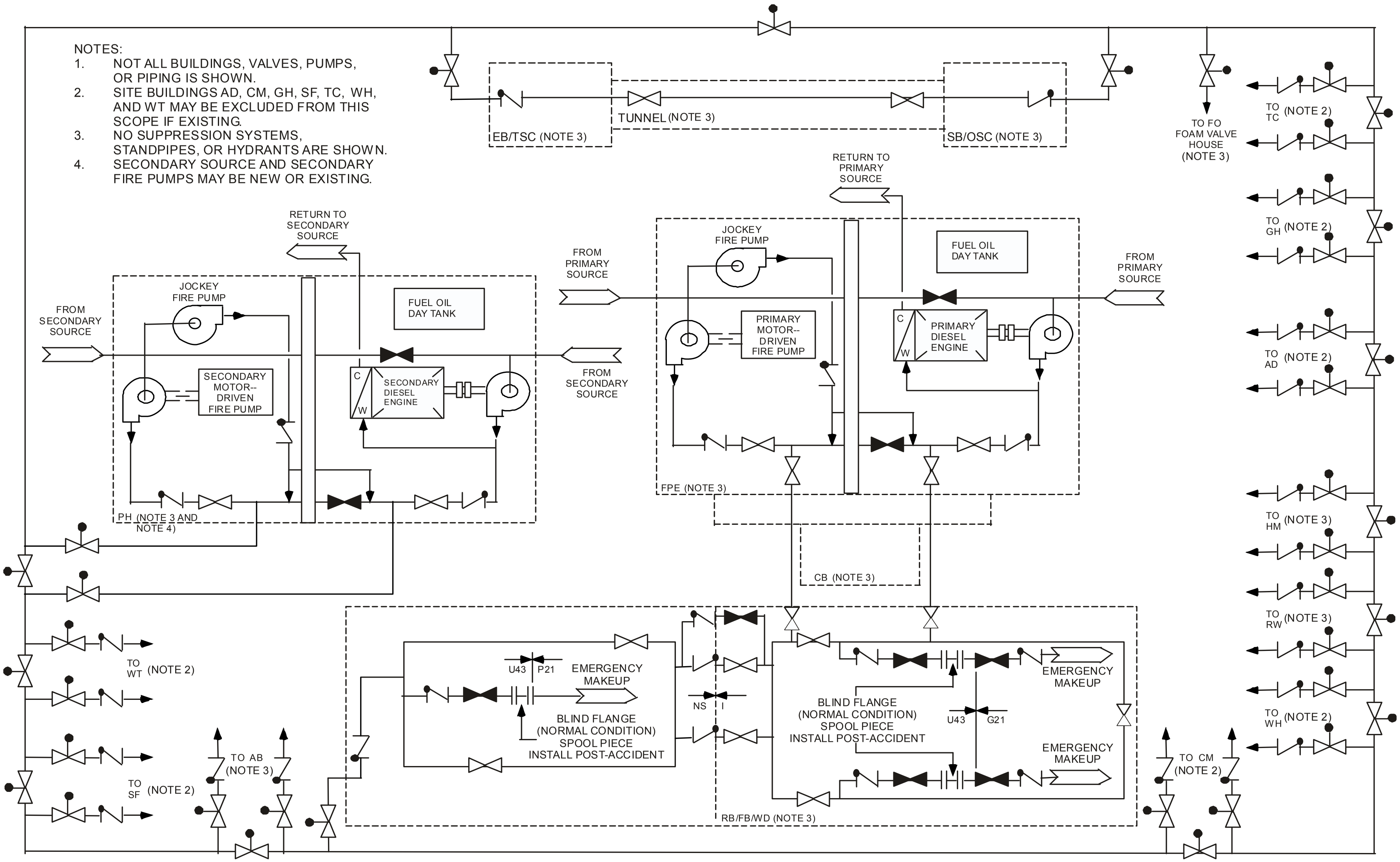
Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	1.	1.
a. The basic configuration of the Fire Protection System is as described in Subsection 2.16.3 and Figure 2.16.3-1.	a. Inspections of the as-built system will be conducted.	a. The as-built Fire Protection System conforms to the basic configuration contained in the Design Description of Subsection 2.16.3 and Figure 2.16.3-1.
b.1 Fire protection equipment will meet the quality and seismic requirements, as shown in Table 2.16.3-1.	b.1 Inspections of the as-built equipment design documentation will be conducted.	b.1 Fire protection equipment meet the quality and seismic requirements, as shown in Table 2.16.3-1.
b.2 For areas containing safe shutdown equipment, Fire Protection Equipment (i.e.; one source of fire water supply, one of the fire water pumps, and fire main leading to and including standpipes and subsystems) is analyzed to withstand the effect of an SSE to remain functional during and after an SSE.	b.2 Inspection of the as-built equipment design documentation for fire protection equipment for areas containing safe shutdown equipment will be conducted.	b.2 Fire protection equipment for areas containing safe shutdown equipment is designed and installed to withstand the effect of SSE to remain functional during and after an SSE.
c. Fire protection pumps will have the flow capabilities shown in Table 2.16.3-2.	c. Test of the as-built pumps will confirm pump flow capabilities.	c. Fire protection pumps will have the flow capabilities shown in Table 2.16.3-2.
2. (Deleted)		
3. Two water supplies one with a minimum volume of 3900 m^3 (1,030,000 gal) and another with a minimum volume of 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.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.	4.	4.
a. The fire water pumps independently will provide their required flow at a pressure of 689 kPa gauge (100 psig) at the most hydraulically remote 65 mm (~2.5 in) hose connections station in the Reactor Building and Control Building.	a. A test of the flow rate and pressure from each pump will be conducted.	a. The fire water pumps independently provide their required flow at a pressure of 689 kPa gauge (100 psig) at the most hydraulically remote 65 mm (~2.5 in) hose connections station in the Reactor Building and Control Building.
b. Fire water pumps independently will provide their required flow at 448 kPa gauge (65 psig) at the most hydraulically remote 40 mm (1.57 in) hose station in the Reactor Building and Control Building.	b. A test of the flow rate and pressure from each pump will be conducted.	b. The fire water pumps independently provide their required flow at 448 kPa gauge (65 psig) at the most hydraulically remote 40 mm (1.57 in) hose station in the Reactor Building and Control Building.
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 and safe shutdown equipment locations will be performed.	6. Standpipe, hose rack stations and safe shutdown equipment are located as such that no safe shutdown equipment is more than [30.5 m (100 ft)] from two hose stations on separate standpipes.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. a. Automatic fire suppression complying with NFPA 13 & 15 is provided for all electrical areas exceeding the combustible load limit of 1400 MJ/m^2. b. Automatic fire suppression for all electrical areas exceeding the combustible load limit of 1400 MJ/m^2 should comply with inspection and test of automatic logic as required by applicable NFPA standard.	7. a. Inspections to assure that of all electrical areas, exceeding the combustible load limit of 1400 MJ/m^2, have automatic fire suppression, per NFPA 13 & 15. b. Inspection of as-built systems and testing of automatic logic under simulated fire condition will be conducted.	7. a. Confirm that of all electrical areas, exceeding the combustible load limit of 1400 MJ/m^2, have fire suppression, per NFPA 13 & 15. b. For all electrical areas exceeding the combustible load limit of 1400 MJ/m^2, the Automatic fire suppression system initiation logic is actuated under simulated fire conditions, per NFPA standard.
8. a. Automatic fire suppression complying with NFPA 13 & 15 is provided for all non-electrical areas exceeding the combustible load limit of 700 MJ/m^2. b. Automatic fire suppression for all non-electrical areas exceeding the combustible load limit of 700 MJ/m^2 should comply with inspection and test of automatic logic as required by applicable NFPA standard.	8. a. Inspections to assure that all non-electrical areas, exceeding the combustible load limit of 700 MJ/m^2, have fire suppression, per NFPA 13 & 15. b. Inspection of as-built systems and testing of automatic logic under simulated fire condition will be conducted.	8. a. Confirm that of all non-electrical areas, exceeding the combustible load limit of 700 MJ/m^2, have fire suppression, per NFPA 13 & 15. b. For all non-electrical areas exceeding the combustible load limit of 700 MJ/m^2, the Automatic fire suppression system initiation logic is actuated under simulated fire conditions, per NFPA standard.
9. Automatic foam-water extinguishing systems are provided for the 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.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The fuel oil tanks for the diesel-driven fire pumps have sufficient capacity to allow diesel engine operation for a minimum of 96 hours as described in this Subsection 2.16.3.	10. a. Testing will confirm fuel consumption rates of the as-built diesel engines. b. Analysis will confirm the as-built fuel oil tank volume(s). c. Analysis will confirm that there is sufficient fuel oil tank volume for the diesel engines to operation for 96 hours.	10. The fuel oil tanks for the diesel-driven fire pumps have sufficient capacity to allow diesel engine operation for a minimum of 96 hours before refilling based upon the as built fuel tanks and fuel consumption rates and margin criteria provided in NFPA 24.
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/displays and controls for the Fire Protection System. Tests of the displays and controls will be performed to assure that the displays and controls function properly.	11. Indications/displays and controls exist or can be retrieved in the MCR as defined in Subsection 2.16.3, and that the displays and controls function properly.
12. The fire water supply system shall be capable of supplying a total makeup flow rate of ≥ 46 m^3/hr (200 gpm) to the IC/PCC and spent fuel pools.	12. A test of the flow rate from each pump will be conducted.	12. 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
13.Fire Detection	13.	13.
a. Local fire alarm panels supervise fire and smoke detectors.	a. Tests will confirm that local fire alarm panels supervise each fire and smoke detector.	a. Each fire and smoke detector is supervised by a local fire alarm panel.
b. The local fire alarm panels are connected to the alarm MFAP via a dedicated data link.	b. Inspections will confirm that local fire alarm panels are connected to the alarm MFAP via a dedicated data link.	b. A dedicated data link connects the local fire alarm panels to the MFAP.
c. Transmitted signals include detector status (normal, alarm, supervisory, trouble) and local fire alarm panel status.	c. Inspections will confirm that transmitted signals include detector status (normal, alarm, supervisory, trouble) and local fire alarm panel status.	c. Transmitted signals include detector status (normal, alarm, supervisory, trouble) and local fire alarm panel status.
d. Instrumentation for fixed and automatic fire suppression systems provides local and remote monitoring capabilities for the suppression system status.	d. Tests of the fixed and automatic fire suppression system instrumentation confirm local and remote monitoring capabilities.	d. Fixed and automatic fire suppression system instrumentation have local and remote monitoring capabilities.
e. All instrumentation for automatically actuated fire suppression systems is either FM approved or UL listed, as specified in NFPA code.	e. Inspections will confirm that all instrumentation for automatically actuated fire suppression systems is either FM approved or UL listed, as specified in NFPA code.	e. All instrumentation for automatically actuated fire suppression systems is either FM approved or UL listed, as specified in NFPA code.



G21 - Fuel and Auxiliary Pools Cooling System
P21 - Reactor Component Cooling Water
U43 - Fire Protection System

Figure 2.16.3-1. Fire Protection System

2.16.3.1 Fire Barriers

Design Description

Fire barriers of 3-hour fire resistance rating are provided separating:

- Safety-related systems from any potential fires in nonsafety-related areas that could affect the ability of safety-related systems to perform their safety function.
- Redundant divisions or trains of safety-related systems from each other to prevent damage that could adversely affect a safe shutdown function from a single fire.
- Components within a single safety-related electrical division that present a fire hazard to components in another safety-related division.
- Electrical circuits (safety-related and nonsafety-related) whose fire-induced failure could cause a spurious actuation that could adversely affect a safe shutdown function.

Penetrations through fire barriers are sealed or closed to provide fire resistance ratings at least equal to that of the barriers. Only noncombustible materials qualified per ASTM E-119 are used for construction of fire barriers. Fire dampers protect ventilation duct openings in fire barriers as required by NFPA 90A.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.3.1-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria for the Fire Barriers.

Table 2.16.3.1-1

ITAAC For Fire Barriers

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. 3-hour fire barriers shall be installed in all locations listed in Subsection 2.16.3.1.	1. Inspections will assure 3-hour fire barriers are installed.	1. All locations listed in Subsection 2.16.3.1 are protected by 3-hour fire barriers.
2. Penetrations through fire barriers are sealed or closed to provide fire resistance ratings at least equal to that of the barriers.	2. Inspections will confirm that penetrations through fire barriers are sealed or closed to provide fire resistance ratings at least equal to that of the barriers.	2. Penetrations through fire barriers provide fire resistance ratings at least equal to that of the barriers.
3. Only noncombustible materials qualified per ASTM E-119 are used for construction of fire barriers.	3. Inspections of material records will confirm that Only noncombustible materials qualified per ASTM E-119 are used for construction of fire barriers.	3. Only noncombustible materials qualified per ASTM E-119 are used for construction of fire barriers
4. Fire dampers protect ventilation duct openings in fire barriers as required by NFPA 90A.	4. Inspections confirm that fire dampers in ventilation duct openings meet NFPA 90A.	4. Fire dampers in ventilation duct openings meet NFPA 90A.

2.16.4 Equipment and Floor Drain System

Design Description

The Equipment and Floor Drain System (EFDS) is a nonsafety-related system, and has no safety design basis other than provision for safety-related containment penetrations and isolation valves. The containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.

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 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 EFDS has safety-related containment penetrations and isolation valves.	1. Inspections of the as-built EFDS safety-related containment penetrations and isolation valves will be conducted.	1. Inspections confirm that the as-built EFDS has safety-related containment penetrations and isolation valves.
2. The EFDS containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.	2. Inspections of the as-built EFDS containment penetrations and isolation valves design documents will be conducted.	2. Inspections confirm that the as-built EFDS containment penetrations and isolation valves are designed to ASME Section III, Class 2, and Seismic Category I.

2.16.5 Reactor Building

Design Description

The Reactor Building (RB) (equivalent to as shown in 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 critical dimensions are provided in Table 2.16.5-1.

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 safety-related, 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.

The RB provides three-hour fire barriers for separation of the four independent safe shutdown divisions.

The RB is protected against external and internal floods. In regards to external flooding, the RB incorporates structural provisions into the plant design to protect the structures, systems and components from postulated flood and groundwater conditions.

This approach provides:

- Wall thicknesses below flood level designed to withstand hydrostatic loads;
- Water stops provided in all expansion and construction joints below flood and groundwater levels;
- Waterproofing of below flood and groundwater levels external surfaces;
- Water seals at pipe penetrations below flood and groundwater levels; and
- Roofs designed to prevent pooling of large amounts of water in accordance with Regulatory Guide 1.102.

Protective features used to mitigate or eliminate the consequences of internal flooding are:

- Structural enclosures or barriers;
- Curbs and sills;
- Leakage detection components; and

- Drainage systems.

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 2.16.5-2 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria for the RB.

Table 2.16.5-1

Critical Dimensions of Reactor Building

(To be included in Rev. 4)

Table 2.16.5-2

ITAAC For The Reactor Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The RB critical dimensions are provided in Table 2.16.5-1.	1. Inspections of the as-built facility will be conducted.	1. A structural analysis report exists which concludes that the as-built building with dimensions in Table 2.16.5-1 is able to withstand the structural design basis loads.
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. Inspection reports exist that show each division is separated by barriers having three-hour fire ratings.
3. The RB is protected against external and internal flooding. For external flooding, protection features are: a. Exterior access openings sealed in external walls below flood and groundwater levels. b. Water seals at pipe penetrations installed in external walls below flood and groundwater levels. c. Water stops provided in expansion and construction joints below flood and groundwater levels. For internal flooding, protection features are: d. Flood water in one division is prevented from propagating to other	3. Inspection of the as-built flood control features will be conducted.	3. As-built documentation exists for the following flood protection features. For external flooding: a. Exterior access openings are sealed in external walls below flood and groundwater levels. b. Water seals at pipe penetrations are installed in external walls below flood and groundwater levels. c. Water stops provided in expansion and construction joints are below flood and groundwater levels. For internal flooding: d. Flood water in one division is prevented from propagating to other division(s) by divisional walls, sills

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
division(s) by divisional walls, sills and watertight doors. e. Equipment necessary for safe shutdown is located above the maximum flood level for that location or is qualified for flood conditions.		and watertight doors. e. 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

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

Figure 2.16.5-8. Nuclear Island Plan at Elevation 27000

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

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 MCR for the plant, and the CB HVAC equipment. The critical dimensions are provided in Table 2.16.6-1.

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 is a Seismic Category I structure that houses control equipment and operations personnel.

The main control room 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 MCR.

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, and missiles.
- Normal plant operation—live loads, dead loads and temperature effects.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.6-2 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria for the Control Building.

Table 2.16.6-1

Critical Dimensions of Control Building

(To be included in Rev. 4)

Table 2.16.6-2

ITAAC For The Control Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The CB critical dimensions are provided in Table 2.16.6-1.	1. Inspections of the as-built structure will be conducted.	1. A structural analysis report exists which concludes that the as-built building with dimensions in Table 2.16.6-1 is able to withstand the structural design basis loads.
2. The MCR 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. Inspection reports exist that show that the as-built CB has a MCR envelope separated from the rest of the CB by walls, floors, doors and penetrations, 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. Inspection reports exist that show that 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. For external flooding, protection features are: a. Exterior access openings sealed in external walls below flood and groundwater levels. b. Water seals at pipe penetrations installed in external walls below flood and groundwater levels. c. Water stops provided in expansion and construction joints below flood and groundwater levels. For internal flooding, protection features are: d. Flood water in one division is prevented from propagating to other division(s) by divisional walls, sills and watertight doors. e. Equipment necessary for safe shutdown is located above the maximum flood level for that location or is qualified for flood conditions.	4. Inspections of the as-built flood control features will be conducted.	4. As-built documentation exists for the following flooding protection features. For external flooding: a. Exterior access openings are sealed in external walls below flood and groundwater levels. b. Water seals at pipe penetrations are installed in external walls below flood and groundwater levels. c. Water stops are provided in expansion and construction joints below flood and groundwater levels. For internal flooding: d. Flood water in one division is prevented from propagating to other division(s) by divisional walls, sills and watertight doors. e. Equipment necessary for safe shutdown is located above the maximum flood level for that location or is qualified for flood conditions.

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 critical dimensions are provided in Table 2.16.7-1.

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.

There is no safety-related component in the FB that could be affected by flooding in this structure. Flooding in the FB could not affect the RB because the connection points in the lower elevation are watertight. To protect the FB against external flooding, penetrations in the external walls below flood level are provided with watertight seals.

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 and missiles; and

(3) Normal plant operation—live loads, dead loads and temperature effects.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.7-2 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria for the Fuel Building.

Table 2.16.7-1

Critical Dimensions of Fuel Building

(To be included in Rev. 4)

Table 2.16.7-2

ITAAC For The Fuel Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The FB critical dimensions are provided in Table 2.16.7-1	1. Inspections of the as-built structure will be conducted.	1. A structural analysis report exists which concludes that the as-built building with dimensions in Table 2.16.7-1 is able to withstand the structural design basis loads.
2. Walls, doors 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. Inspection reports exist that show that the as-installed external walls, doors and penetrations that form the FB boundaries have a three-hour fire rating.
3. The FB is protected against an external flooding. Protection features are: a. Exterior access openings sealed in external walls below flood and groundwater levels. b. Water seals at pipe penetrations installed in external walls below flood and groundwater levels. c. Water stops are in expansion and construction joints below flood and groundwater levels.	3. Inspection of the as-built flood control features will be conducted.	3. As-built documentation exists for the following flood protection features. Protection features are: a. Exterior access openings are sealed in external walls below flood and groundwater levels. b. Water seals at pipe penetrations are installed in external walls below flood and groundwater levels. c. Water stops are provided in expansion and construction joints below flood and groundwater levels.

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 structure is Seismic Category II nonsafety-related, and thus, it is constructed to prevent a structural failure that could impair the ability of nearby safety-related SSCs to perform their safety-related functions. The building is partially embedded. Shielding is provided for the turbine on the operating deck.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.8-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria for the Turbine Building.

Table 2.16.8-1

ITAAC for The Turbine Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Turbine Building is a Seismic Category II structure.	1. Inspections of the as-built Turbine Building design documentation issued for construction will be performed.	1. A Structural analysis report exists which concludes that the as-built Turbine Building can withstand Seismic Category II design loads.

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 RW is a Non-Seismic Category (NS) structure. The RW is designed according to the safety classifications defined in Regulatory Guide 1.143 Category RW-IIa.

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.

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 intake and discharge structure is site-specific.

No entry for this system.

2.18 YARD STRUCTURES AND EQUIPMENT

2.18.1 Oil Storage and Transfer Systems

Design Description

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

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 on a site-specific basis.

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 has a design life of 60 years. Piping systems and their components are designed and constructed in accordance with their applicable design code requirements identified in the individual system design specifications.

Safety-related 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 for the Piping Design.

Table 3.1-1

ITAAC For The Generic Piping Design

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Each safety-related 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. On an individual safety-related system basis, an ASME Code Certified Stress Report 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 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 concludes that for each postulated piping failure, the reactor can be shut down safely.
3. On an individual system basis, 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. On an individual system basis, 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 approach is that software plans address and document the 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, will be 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 addresses various aspects of the software development and quality outlined in the related industry standards and regulatory guidance. In certain cases, deviation is taken from the detailed guidance in those documents, in which case 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)
- Cyber Security Program Plan (CSPP)

Software Management Plan

The Software Management Plan (SMP) 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-1058.1.

Software Development Plan

The Software Development Plan (SDP) 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 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."

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) describes the instruction and guideline to operate and maintain the software-based product. IEEE Std. 1219, "IEEE Standard for Software Maintenance" provides an acceptance approach for management and for execution of the software maintenance activities, which this SOMP uses as a guide.

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 the training needs for the utility plant staff, including operators and I&C engineers and technicians in operation and maintenance 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, addresses 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) describes the independent V&V organization responsible for executing the V&V tasks to ensure the design requirements of each life cycle phase are traceable to a relevant requirement defined in the previous phase, and the developed software-based product meets its specified requirements, performs its intended functions correctly and performs no unintended functions. The 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."

Cyber-Security Program Plan

The cyber security program plan is developed using a structured design process to protect digital assets from cyber attack, which provides for specific documentation and reviews during the following waterfall lifecycle phases:

- Concepts Phase
- Requirement Phase
- Design Phase
- Implementation Phase
- Test Phase
- Installation, Checkout and Acceptance Testing Phase
- Operation Phase
- Maintenance Phase
- Retirement Phase

The objective of inspecting and testing cyber-security functions is to verify the process used to design the hardware and software, and to ensure that the system cyber-security requirements are validated by execution of integration, system, and acceptance tests, respectively. Testing includes tests on system hardware configuration (including all external connectivity), software integration, software qualification, system integration, system qualification, and system factory acceptance.

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) defines the managerial processes necessary to accomplish the design and development of the ESBWR software-based products and addresses the various elements described in related guidance documents including IEEE-1058.1.	1. Inspection of the Software Management Plan will be performed.	1. Report(s) exist(s) and conclude(s) that the Software Management Plan (SMP) defines the managerial processes necessary to accomplish the design and development of the ESBWR software-based products and addresses the various elements described in related guidance documents including IEEE-1058.1.
2. Software Development Plan (SDP) 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".	2. Inspection of the Software Development Plan will be performed.	2. Report(s) exist(s) and conclude(s) that the Software Development Plan (SDP) 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".

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 consistent with the requirements specified in IEEE-730, "IEEE Standard for Quality Assurance Plans".	3. Inspection of the Software Assurance Plan will be performed.	3. Report(s) exist(s) and conclude(s) that 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 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 Inspection of the Software Integration Plan will be performed.	4. Report(s) exist(s) and conclude(s) that the Software Integration Plan (SIntP) summarizes the management, implementation, and resource characteristics of the integration program.
5. The Software Installation Plan (SIP) summarizes the management, implement of software operations and maintenance, and resource characteristics of the installation program.	5. Inspection of the Software Integration Plan will be performed.	5. Report(s) exist(s) and conclude(s) that the Software Installation Plan (SIP) summarizes the management, implementation, and resource characteristics of the installation program.

6. The Software Operations and Maintenance Plan (SOMP), which uses IEEE Std. 1219, "IEEE Standard for Software Maintenance" to provide an acceptance approach for management and for executing the software maintenance activities, will be established for software-based products in conjunction with the SCMP.	6. Inspection of the Software Operations and Maintenance Plan (SOMP),	6. Report(s) exist(s) and conclude(s) that the Software Operations and Maintenance Plan (SOMP), which uses IEEE Std. 1219, "IEEE Standard for Software Maintenance" to provide an acceptance approach for management and for executing the software maintenance activities.
7. The Software Training Plan (STrngP) addresses the required training for staff working in 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.	7. Inspection of the Software Training Plan (STrngP) will be performed.	7. Report(s) exist(s) and conclude(s) that the Software Training Plan (STrngP) addresses the required training for staff working in 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.
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. Inspection of the Software Safety Plan (SSP) will be performed.	8. Report(s) exist(s) and conclude(s) that 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."

9. The Software Verification and Validation Plan (SVVP) describes the independent V&V organization responsible for executing the V&V tasks to ensure the design requirements of each life cycle phase are traceable to a relevant requirement defined in the previous phase, and the developed software-based product meets its specified requirements, performs its intended functions correctly and performs no unintended functions. The 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.	9 Inspection of the Software Verification and Validation Plan (SVVP) will be performed.	9 Report(s) exist(s) and conclude(s) that the Software Verification and Validation Plan (SVVP) describes the independent V&V organization responsible for executing the V&V tasks to ensure the design requirements of each life cycle phase are traceable to a relevant requirement defined in the previous phase, and the developed software-based product meets its specified requirements, performs its intended functions correctly and performs no unintended functions. The 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."
10. The Software Configuration Management Plan (SCMP), defines the management, the implementation of the configuration control and the specific documents, files and systems to which it is applicable.	10. Inspection of the Software Configuration Management Plan (SCMP) will be performed.	10. Report(s) exist(s) and conclude(s) that the Software Configuration Management Plan (SCMP), defines the management, the implementation of the configuration control and the specific documents, files and systems to which it is applicable.

11. The cyber security program plan is developed using a structured design process to protect digital assets from cyber attack, which provides for specific documentation and reviews during the following waterfall lifecycle phases: • Concepts Phase • Requirement Phase • Design Phase • Implementation Phase • Test Phase • Installation, Checkout, and Acceptance Testing Phase • Operation Phase • Maintenance Phase • Retirement Phase	11 The following are performed: • Inspection of the process used to design the hardware and software. • Tests on system hardware configuration (including all external connectivity), software integration, software qualification, system integration, system qualification, and system factory acceptance.	11. Inspection and test reports exist and conclude(s) that the cyber-security program plan is developed using a structured design process to protect digital assets from cyber attack, which provides for specific documentation and reviews during the following waterfall lifecycle phases: • Concepts Phase • Requirement Phase • Design Phase • Implementation Phase • Test Phase • Installation, Checkout, and Acceptance Testing Phase • Operation Phase • Maintenance Phase • Retirement Phase

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.

HFE Program Goals - The general objectives of the program can be stated in "human-centered" terms, which, as the HFE program develops, is refined and used as a basis for HFE planning, test and evaluation activities. HFE design goals include ensuring that:

- Personnel tasks can be accomplished within time and performance criteria;
- Human-System Interfaces (HSIs), procedures, staffing/qualifications, training and management and organizational variables support a high degree of operating crew situation awareness;
- Allocation of functions accommodates human capabilities and limitations;
- Operator vigilance is maintained;
- Acceptable operator workload is met;
- Operator interfaces contribute to an error free environment; and
- Error detection and recovery capabilities are provided.

Applicable Facilities - The HFE program addresses the Main Control Room (MCR), Remote Shutdown System (RSS), Technical Support Center (TSC), Emergency Operations Facility (EOF), and Local Control Stations (LCSs) with a safety-related function or as defined by high level task analysis.

Applicable HSIs, Procedures and Training - The applicable HSIs, procedures, and training included in the HFE program include operations, accident management, maintenance, test, inspection and surveillance interfaces (including procedures) for those systems that have safety significance. 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, both licensed and unlicensed, addressed by the HFE program include licensed control room operators as defined in 10 CFR Part 55 and the categories of personnel defined by 10 CFR 50.120. In addition any other plant personnel who perform tasks that are directly related to plant safety, are addressed in the HFE program.

Man-Machine Interface System (MMIS) employs digital technology to implement the majority of the monitoring, control, and protection functions for the ESBWR.

Standardization of hardware and software, and modularity of design will be used to simplify maintenance and provide protection against obsolescence.

The elements of the ESBWR HFE Program Management are provided in the plan entitled "Man-Machine Interface System and Human Factors Engineering Implementation Plan (MMIS and HFE Implementation Plan). In the plan the following are described:

- HFE goals/objectives
- A technical program to accomplish the objectives
- The system to track HFE issues
- The HFE design team
- Management and organizational structure for the technical program

The activities of the HFE technical program described in the MMIS and HFE Implementation Plan are:

(1) Operating Experience Review
(2) Functional Requirements Analysis
(3) Allocation of Functions
(4) Task Analysis
(5) Staffing and Qualifications
(6) Human Reliability Analysis
(7) Human System Interface Design
(8) Procedure Development
(9) Training Development
(10) Human Factors Verification and Validation
(11) Design Implementation
(12) Human Performance Monitoring

The proposed methodologies for the conducts of the HFE activities are described in separate implementation plans. The results and outcomes of the activities are summarized in individual results summary reports.

The MMIS and HFE Implementation Plan and supporting HFE activity implementation plans are submitted for NRC staff review in the pre-design project phase. The result summary reports contain the main sources of information, are available for the NRC staff review, and are included in the list of items for Inspections, Tests, Analyses, and Acceptance Criteria.

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.

Table 3.3-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria for Human Factors Engineering.

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Operating Experience Review (OER) is performed in accordance with the MMIS and HFE Implementation Plan and its requirements.	1. OER activity is conducted and a results summary report is completed describing the personnel and methodology employed in the conduct of the activity and summarizing the OER outcomes and results.	1. A results summary report is completed describing the following: a. The OER team members and backgrounds b. The scope of the OER c. The sources of operating experience reviewed and documented results d. The process for issue analysis, tracking, and review
2. Functional Requirements Analysis (FRA) and Allocation of Functions (AOF) is performed in accordance with the MMIS and HFE Implementation Plan and its requirements.	2. FRA and AOF activities are conducted and a results summary report is completed describing the personnel and methodology employed in the conduct of the activities and summarizing the FRA and AOF outcomes and results.	2. A results summary report is completed describing the following: a. The FRA and AOF team members and backgrounds b. Plant functional requirements c. Function allocations d. The methodology and implementation of the FRA and AOF activities concluding that the activities were performed in accordance with implementation plans.

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. Task Analysis is performed in accordance with the MMIS and HFE Implementation Plan and its requirements.	3. Task Analysis activity is conducted and a results summary report is completed describing the personnel and methodology employed in the conduct of the activity and summarizing the Task Analysis outcomes and results	3. A results summary report is completed describing the following: a. The Task Analysis team members and backgrounds b. The scope of the Task Analysis c. High level task descriptions d. Detailed task descriptions e. The methodology and implementation of the Task Analysis concluding that the activity was performed in accordance with implementation plans
4. Staffing and Qualifications (S&Q) is performed in accordance with the MMIS and HFE Implementation Plan and its requirements	4. Staffing and Qualifications activity is conducted and a results summary report is completed describing the personnel and methodology employed in the conduct of the activity and summarizing the Staffing and Qualifications outcomes and results.	4. A results summary report is completed describing the following: a. The S&Q team members and backgrounds b. The scope of the S&Q activity c. Final staffing levels and qualifications d. The basis for the S&Q concluding that issues and concerns raised in other HFE activities are addressed. e. The methodology and implementation of the S&Q activity concluding that the activity was performed in accordance with implementation plans.

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Human Reliability Analysis (HRA) is performed in accordance with the MMIS and HFE Implementation Plan and its requirements.	5. HRA activity is conducted and a results summary report is completed describing the personnel and methodology employed in the conduct of the activity and summarizing the HRA outcomes and results.	5. A results summary report is completed describing the following: a. The HRA team members and backgrounds b. The scope of the HRA c. Risk important human actions and how these are addressed in the HF design process d. The methodology and implementation of the HRA activity concluding that the activity was performed in accordance with implementation plans.
6. Human System Interface (HSI) Design is performed in accordance with the MMIS and HFE Implementation Plan and its requirements	6. HSI Design activity is conducted and a results summary report is completed describing the personnel and methodology employed in the conduct of the activity and summarizing the HSI Design outcomes and results	6. A results summary report is completed describing the following: a. The HSI Design team members and backgrounds b. HFE standards and guideline documents used in the activity c. Style Guide and design specifications for HSI design d. List of instruments that complies with RG 1.97 and supporting analysis e. The methods used for the evaluation and verification of the HSI f. The methodology and implementation of

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		the HSI Design activity concluding that the activity was performed in accordance with implementation plans
7. Procedure Development is performed in accordance with the MMIS and HFE Implementation Plan and its requirements	7. Procedure Development activity is conducted and a results summary report is completed describing the personnel and methodology employed in the conduct of the activity and summarizing the Procedure Development outcomes and results. Plant procedures (and supporting development material) are available for inspection.	7. a. Effective plant procedures are approved. A results summary report is completed describing the following: b. The Procedure Development team members and backgrounds. c. The scope of the procedures development process. d. Final procedures and procedure support equipment. e. Technical basis for severe accident management. f. The methodology and implementation of the procedures development activity concluding that the activity was performed in accordance with implementation plans
8. Training Development is performed in accordance with the MMIS and HFE Implementation Plan and its requirements	8. Training Development activity is conducted and a results summary report is completed describing the personnel and methodology employed in the conduct of the activity and summarizing	8. A results summary report is completed describing the following: a. The Training Development team members and backgrounds b. The purpose and scope of the Training

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	the Training Development outcomes and results	Development c. The roles of organizations involved and the facilities and resources needed to satisfy the needs of the training d. The organization and content of the Training Program e. The learning objectives f. The methods for evaluating the effectiveness of the training program and trainee mastery of training g. The methods for verifying the accuracy and completeness of training course materials h. Procedures for refining and updating the content and conduct of training i. The plan for periodic retraining of personnel j. The methodology and implementation of the Training Development activity concluding that the activity was performed in accordance with implementation plans
9. Human Factors Verification and Validation (HF V&V) is performed in accordance with the MMIS and HFE Implementation Plan	9. HF V&V activity is conducted and a results summary report is completed describing the personnel and methodology employed in the conduct	9. A results summary report is completed describing the following: a. The HF V&V team members and

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
and its requirements	of the activity and summarizing the HF V&V outcomes and results	backgrounds b. The scope of the V&V c. Sample of operational conditions used for the V&V d. HSI Inventory and characterization e. HSI Task Support Verification f. HFE Design Verification g. Integrated System Validation h. The methodology and implementation for the HF V&V activity concluding that the activity was performed in accordance with implementation plans.
10. Design Implementation is performed in accordance with the MMIS and HFE Implementation Plan and its requirements	10. Design Implementation activity is conducted and a results summary report is completed describing the personnel and methodology employed in the conduct of the activity and summarizing the Design Implementation outcomes and results	10. A results summary report is completed describing the following: a. The Design Implementation team members and backgrounds b. The HSI Verification (As-built) c. The Procedures and Training Confirmation (As-Built) d. The evaluation of aspects of the design not addressed in the HF V&V e. Resolution of HEDs and Open issues concluding that all HFE-related issues in the issue tracking system (HFEITS) are

Table 3.3-1
ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		corrected or justified f. The methodology and implementation for the Design Implementation activity concluding that the activity was performed in accordance with implementation plans.
11. Human Performance Monitoring (HPM) is performed in accordance with the MMIS and HFE Implementation Plan and its requirements	11. HPM activity is initiated and a results summary report is completed describing the personnel and methodology employed in the conduct of the activity and summarizing the HPM strategy, initial outcomes and results	11. A results summary report is completed describing the following: a. The HPM team members and backgrounds b. The HPM strategy including the scope, structure, and provisions for specific cause determination, trending of performance degradation and failures, and corrective actions c. The methodology and implementation of the HPM activity concluding that the activity was performed in accordance with implementation plans

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.

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 will be below the concentrations that define an airborne radioactive area in 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 MCR 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 MCR 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 sampling locations, the health physics facility (counting room), 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 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: • Design ventilation flow rates for each area; • Typical leakage characteristics for equipment located in each area; and • 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 for radiological decay and buildup of activated corrosion and wear products.	1. Calculation of radioactive airborne concentration demonstrates that: 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. 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. 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
		lower contamination.
2. Airborne radioactivity monitoring provides local alarms (visual alarms in high noise areas) with variable alarm setpoints, and readout/alarm capability.	2. Inspections and/or tests will confirm that airborne radioactivity monitoring has local alarms (visual alarms in high noise areas) with variable alarm setpoints, and readout/alarm capability.	2. Inspections/test reports document that the airborne radioactivity monitoring has local alarms (visual alarms in high noise areas) with variable alarm setpoints, and readout/alarm capability.

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.

3.6 DESIGN RELIABILITY ASSURANCE PROGRAM

Design Description

The GE ESBWR Design Reliability Assurance Program (D-RAP) is used during detailed design and specific equipment selection phases to assure that the important ESBWR reliability assumptions of the probabilistic risk assessment (PRA) will be considered throughout the plant life. The PRA is used to evaluate plant responses to abnormal event initiations and the corresponding plant mitigation functions, to ensure potential plant damage scenarios pose a very low probability and risk to the public.

The objectives of the D-RAP are to provide reasonable assurance that risk significant SSCs are designed such that:

(1) Assumptions from the risk analysis are utilized;

(2) SSCs when challenged, function in accordance with the assumed reliability;

(3) SSCs whose failure results in a reactor trip, function in accordance with the assumed reliability; and

(4) Maintenance actions to achieve the assumed reliability are identified.

The scope of the ESBWR D-RAP includes risk-significant SSCs, both safety-related and nonsafety-related, that provide defense-in-depth or result in significant improvement in the PRA evaluations. The D-RAP provides reasonable assurance that the design of risk-significant SSCs is consistent with their risk analysis assumptions.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.6-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the D-RAP.

Table 3.6-1

ITAAC For Design Reliability Assurance Program

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The D-RAP provides reasonable assurance that the design of risk-significant SSCs is consistent with their risk analysis assumptions.	1. Inspect plant operating and maintenance procedures, and Maintenance Rule program.	1. Reports and/or specifications exist that identify reliability assurance strategies, i.e., operational, maintenance, and/or performance monitoring activities, to provide reasonable assurance that the estimated reliability of each safety significant SSC is at least equal to the assumed reliability in the plant specific PRA.

3.7 POST ACCIDENT MONITORING INSTRUMENTATION

Design Description

The post accident monitoring instrumentation provides information required to monitor variables and systems over their anticipated ranges for post-accident conditions as appropriate to ensure adequate safety. The post accident monitoring instrumentation receives information on variables from other systems and much of the instrumentation is within the boundaries of these systems. This information may be safety-related or nonsafety-related.

The ESBWR Distributed Control and Information System (DCIS) provides the required signal paths to process this information. The ESBWR DCIS is subdivided into the Safety-related DCIS (Q-DCIS) and the Nonsafety-related DCIS (N-DCIS). For variables associated with critical safety functions and powered from safety-related sources the safety related Q-DCIS provides the required signal paths to process this information. This information is then displayed on Q-DCIS divisional safety-related displays. The safety-related information can also be transmitted via isolated nonsafety-related gateways to the nonsafety-related N-DCIS for input to nonsafety-related displays, plant computer functions, and the Alarm Management System. Type A, Type B, and Type C variables are powered from safety-related sources. Type D and Type E variables will have their power source determined as part of the design process.

For variables that are powered from nonsafety-related sources the N-DCIS provides the required signal paths to process this information. This information is used for input to nonsafety-related displays, plant computer functions, and the Alarm Management System.

There is a defined process to determine the appropriate variables and types (A, B, C, D, or E).

For each variable and type the process determines additional characteristics appropriate to that variable as outlined below:

Performance criteria

- Range
- Accuracy
- Response time
- Required instrument duration
- Reliability
- Performance assessment documentation

Design criteria

- Single failure
- Common cause failure
- Independence and separation
- Isolation
- Information ambiguity

- Power supply
- Calibration
- Testability
- Direct measurement
- Control of access
- Maintenance and repair
- Auxiliary supporting features
- Portable instruments
- Documentation of Design Criteria

Qualification criteria

- Type A variables
- Type B variables
- Type C variables
- Type D variables
- Type E variables
- Portable instruments
- Post Event operating time
- Documentation of qualification criteria

Display criteria

- Information characteristics
- Human factors
- Anomalous indications
- Continuous vs. on-demand display
- Trend or rate information
- Display identification
- Type of monitoring channel display
- Display location
- Information ambiguity
- Recording
- Digital display signal validation
- Display criteria documentation

Quality assurance

All safety-related equipment is provided under 10 CFR 50 Appendix B quality programs.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.7-1 specifies the inspections, tests, analyses, and associated acceptance criteria for post accident monitoring instrumentation.

Table 3.7-1

ITAAC For The Post Accident Monitoring Instrumentation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The post accident monitoring instrumentation is designed with the requirements (variables, types, performance criteria, design criteria, qualification criteria, display criteria, and quality assurance) as described in Section 3.7.	1. Inspections tests and/or analysis will be performed to verify that the post accident monitoring instrumentation is in conformance with the requirements as described in Section 3.7.	1. Report(s) exists and conclude(s) that the post accident monitoring instrumentation is in conformance with the requirements as described in Section 3.7.

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 site-specific. A specific source will be designated 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 on a site-specific basis.

4.2 OFFSITE POWER SYSTEM

Design Description

The Offsite Power System is not within the scope of the certified design. A site-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 alternate 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 buses 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 affecting connection to the grid.
    - Faults of a single main bus are isolated without interrupting service to the plant.
- 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.
- 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 PRA.

- 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

Potable and Sanitary Water systems are not connected to any safety-related structure, system or component or any system that could potentially contain radioactive material.

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 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) (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 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 PSWS will be optimized, including, but not limited to, the following parameters:

- System configurations and materials; 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.

4.5 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 cooling 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, including, but not limited to, the following parameters:

- System configurations and materials; 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 (396 gpm)] 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 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.

5.1.1 References

5.1-1 American Society of Civil Engineers, Minimum Design Loads for Buildings and Other Structures, ASCE 7-02, 2002.

5.1-2 National Weather Source Publication Hydrometeorology Report No. 52 (HMR-52)

5.1-3 Electric Power Research Institute, "Advanced Light Water Reactor Utility Requirements Document," Revision 6, May 1997.

Table 5.1-1

Envelope of ESBWR Standard Plant Site Design Parameters [1]

Maximum Ground Water Level:	0.61 m (2 ft) below plant grade
Extreme Wind:	**Seismic Category I and II Structures** - 100-year Wind Speed (3-sec gust): 67.1 m/s (150 mph) - Exposure Category: D **Non-Seismic Standard Plant Structures** - Extreme wind 49.2 m/s (110 mph)
Maximum Flood (or Tsunami) Level: [2]	0.3 m (1 ft) below plant grade
Tornado:	- Maximum Tornado Wind Speed: [3] 147.5 m/s (330 mph) - Maximum Rotational Speed: 116.2 m/s (260 mph) - Translational Velocity: 31.3 m/s (70 mph) - Radius: 45.7 m (150 ft) - Maximum Pressure Differential: 16.6 kPa (2.4 psi) - Rate of Pressure Change: 11.7 kPa/s (1.7 psi/s) - Missile Spectra: [3] Spectra I of SRP 3.5.1.4, Rev 2 applied to full building height.
Precipitation (for Roof Design):	- Maximum Rainfall Rate: [4] 49.3 cm/hr (19.4 in/hr) - Maximum Short Term Rate: 15.7 cm (6.2 in) in 5 minutes - Maximum Roof Load: [5] 2873 Pa (60 lbf/ft^2)
Ambient Design Temperature: [6]	**2% Exceedance Values** - Maximum: 35.6°C (96°F) dry bulb 26.1°C (79°F) wet bulb (coincident) 27.2°C (81°F) wet bulb (non-coincident) - Minimum: -23.3°C (-10°F) **1% Exceedance Values** - Maximum: 37.8°C (100°F) dry bulb 26.1°C (79°F) wet bulb (coincident) 27.8°C (82°F) wet bulb (non-coincident) - Minimum: -23.3°C (-10°F) **0% Exceedance Values** - Maximum: 46.1°C (115°F) dry bulb 26.7°C (80°F) wet bulb (coincident) 29.4°C (85°F) wet bulb (non-coincident) - Minimum: -40°C (-40°F)
Soil Properties:	- Minimum Static Bearing Capacity: [7] ≥ 718 kPa (15000 lbf/ft^2) - Minimum Shear Wave Velocity: [8] 300 m/s (1000 ft/s) - Liquefaction Potential: None under footprint of Seismic Category I or II structures. - Angle of Internal Friction ≥ 30 degrees

Table 5.1-1

Envelope of ESBWR Standard Plant Site Design Parameters (continued)

Seismology:	- SSE Horizontal Ground Response Spectra: (9)	See Figure 5.1-1
	- SSE Vertical Ground Response Spectra: (9)	See Figure 5.1-2
Hazards in Site Vicinity:	- Site Proximity Missiles and Aircraft:	≤ 10^{-7} per year
	- Toxic Gases:	None *
	- Volcanic Activity:	None
* Maximum toxic gas concentrations at the Main Control Room (MCR) and Technical Support Center (TSC) HVAC intakes:	< toxicity limits	
Required Stability of Slopes: (10)	- Factor of safety for static (non-seismic) loading	1.5
	- Factor of safety for dynamic (seismic) loading	1.1
Maximum Settlement Values for Seismic Category I Buildings (see Tier 2 Subsections 3G.1.5.5.4 and 3G.2.5.5.1)**:**		
Maximum Settlement at any corner of basemat	- Under Reactor/Fuel Building Mat	103 mm (4.0 inches)
	- Under Control Building	18 mm (0.7 inches)
Averaged Settlement at four corners of basemat	- Under Reactor/Fuel Building Mat	65 mm (2.6 inches)
	- Under Control Building	11 mm (0.4 inches)
Maximum Differential Settlement along the longest mat foundation dimension	- within Reactor/Fuel Building	77 mm (3.0 inches)
	- within Control Building	13 mm (0.5 inches)
Maximum Differential Displacement between Reactor/Fuel Buildings and Control Building		85 mm (3.3 inches)

Maximum Differential Displacement between Reactor/Fuel Buildings and Control Building			85 mm (3.3 inches)
Meteorological Dispersion (X/Q): (11)	EAB X/Q:		
	0-2 hours:	2.00E-03 s/m^3	
	LPZ X/Q:		
	0-8 hours:	1.90E-04 s/m^3	
	8-24 hours:	1.40E-04 s/m^3	
	1-4 days:	7.50E-05 s/m^3	
	4-30 days:	3.00E-05 s/m^3	
* First value is for unfiltered inleakage. Second value is for filtered air intake (emergency and normal)	Control Room X/Q: *		
	Reactor Building		
	0-2 hours:	1.90E-03 s/m^3	1.50E-03 s/m^3
	2-8 hours:	1.30E-03 s/m^3	1.10E-03 s/m^3
	8-24 hours:	5.90E-04 s/m^3	5.00E-04 s/m^3
	1-4 days:	5.00E-04 s/m^3	4.20E-04 s/m^3
	4-30 days	4.40E-04 s/m^3	3.80E-04 s/m^3
	Passive Containment Cooling System / Reactor Building Roof		
	0-2 hours:	3.40E-03 s/m^3	3.00E-03 s/m^3
	2-8 hours:	2.70E-03 s/m^3	2.50E-03 s/m^3
	8-24 hours:	1.40E-03 s/m^3	1.20E-03 s/m^3
	1-4 days:	1.10E-03 s/m^3	9.00E-04 s/m^3
	4-30 days	7.90E-04 s/m^3	7.00E-04 s/m^3
	Turbine Building		
	Turbine Building		
	0-2 hours:	1.20E-03 s/m^3	1.20E-03 s/m^3
	2-8 hours:	9.80E-04 s/m^3	9.80E-04 s/m^3
	8-24 hours:	3.90E-04 s/m^3	3.90E-04 s/m^3
	1-4 days:	3.80E-04 s/m^3	3.80E-04 s/m^3
	4-30 days	3.20E-04 s/m^3	3.20E-04 s/m^3
Long Term Dispersion Estimates: (12)	X/Q:	2.0E-06 s/m^3	
	D/Q:	4.0E-09 m^{-2}	

Notes:

(1) The design of the Radwaste Building uses a set of design parameters that are specified in Regulatory Guide 1.143, Table 2, Class RW IIa instead of the corresponding values given in this table.

(2) Probable maximum flood level (PMF), as defined in Table 1.2-6 of Volume III of Reference 5.1-3.

(3) Maximum speed selected is based on NRC Interim Position on Regulatory Guide 1.76. Concrete structures designed to resist Spectrum I missiles of SRP 3.5.1.4, Rev. 2, will also resist missiles postulated in Draft Guide DG-1143.

(4) Based on probable maximum precipitation (PMP) for one hour over 2.6 km^2 (one square mile) with a ratio of 5 minutes to one hour PMP of 0.32 as found in Reference 5.1-2. Roof scuppers are designed to handle the PMP. When used in combination with snow pack, the roof and drainage design is for 2873 Pa (60 lbf/ft^2) extreme load.

(5) Maximum design roof load accommodates snow load and probable maximum winter precipitation in References 5.1-1 and 5.1-2.

(6) Zero percent exceedance values are based on conservative estimates of historical high and low values for potential sites. One and two percent exceedance values were selected in order to bound the values presented in Reference 5.1-3 and available Early Site Permit applications.

(7) At foundation level of Seismic Category I structures.

(8) This is the equivalent uniform shear wave velocity (V_{eq}) at seismic strains after the soil property uncertainties have been applied. V_{eq} is calculated to achieve the same wave traveling time over the depth equal to the embedment depth plus 2 times the largest foundation plan dimension below the foundation as follows:

$$V_{eq} = \frac{\sum d_i}{\sum \frac{d_i}{V_i}}$$

where d_i and V_i are the depth and shear wave velocity, respectively, of the ith layer. The ratio of the largest to the smallest shear wave velocity over the mat foundation width at the foundation level does not exceed 1.7.

(9) Safe Shutdown Earthquake (SSE) design ground response spectra are defined as free-field outcrop spectra at the foundation level (bottom of the base slab) of Seismic Category I structures.

(10) Values reported here are actually design criteria rather than site design parameters. They are included here because they do not appear elsewhere in Tier 1.

(11) If a selected site has a X/Q value that exceeds the ESBWR reference site value, then the radiological consequences associated with the controlling design basis accidents will be analyzed on a site-specific basis, to demonstrate that the dose reference values provided in 10 CFR 50.34(a) and control room operator dose limits provided in General Design Criterion 19 (using site-specific X/Q values) will be met.

(12) If a selected site has a X/Q value that exceeds the ESBWR reference site value, then the release concentrations will be adjusted proportionate to the change in X/Q. In addition, for a site selected that exceeds the bounding X/Q or D/Q values, the process of how the resulting annual average doses will continue to meet the dose reference values provided in 10 CFR 50 Appendix I (using site-specific X/Q and D/Q values) will be provided on a site-specific basis.

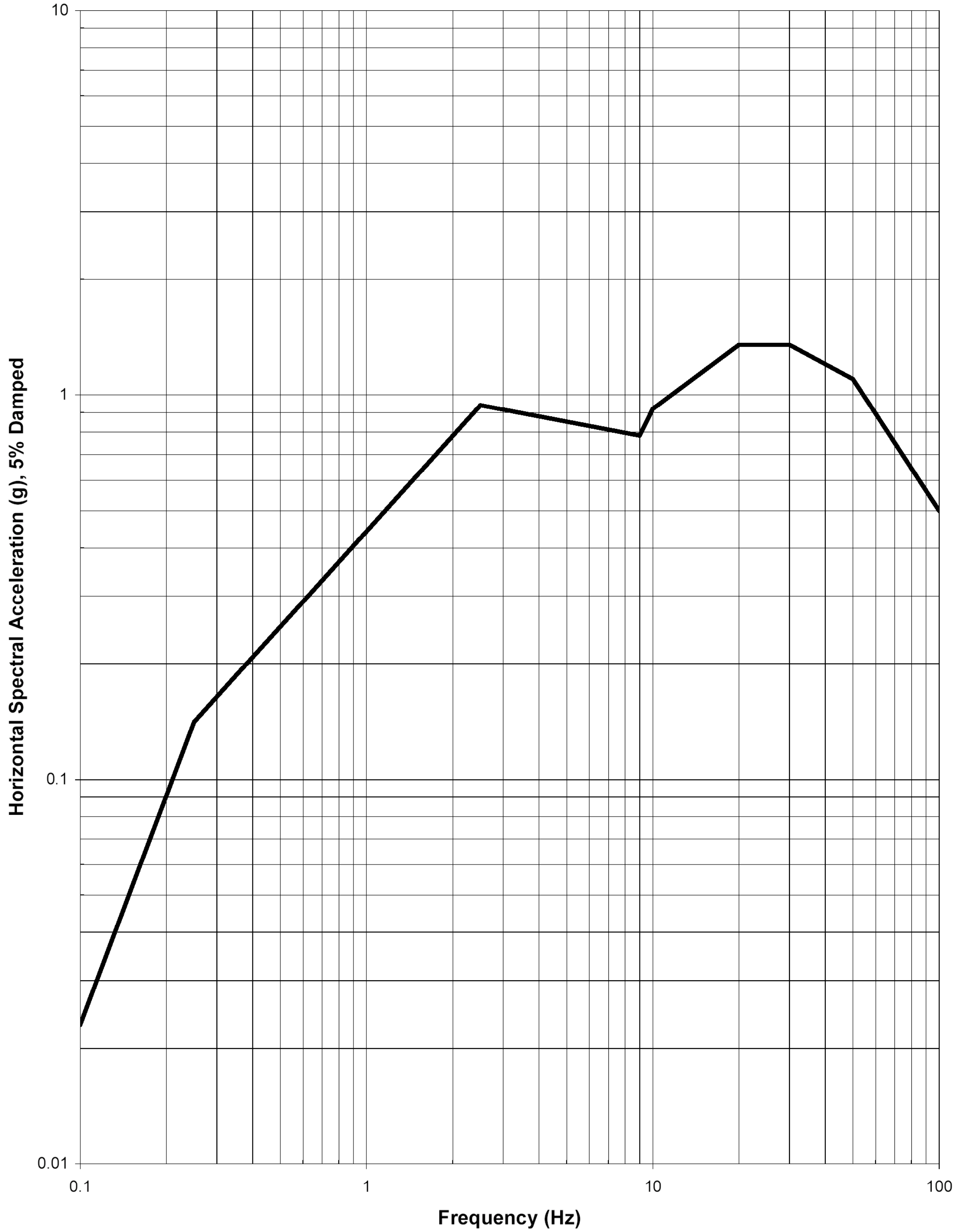


Figure 5.1-1. ESBWR Horizontal SSE Design Ground Spectra at Foundation Level

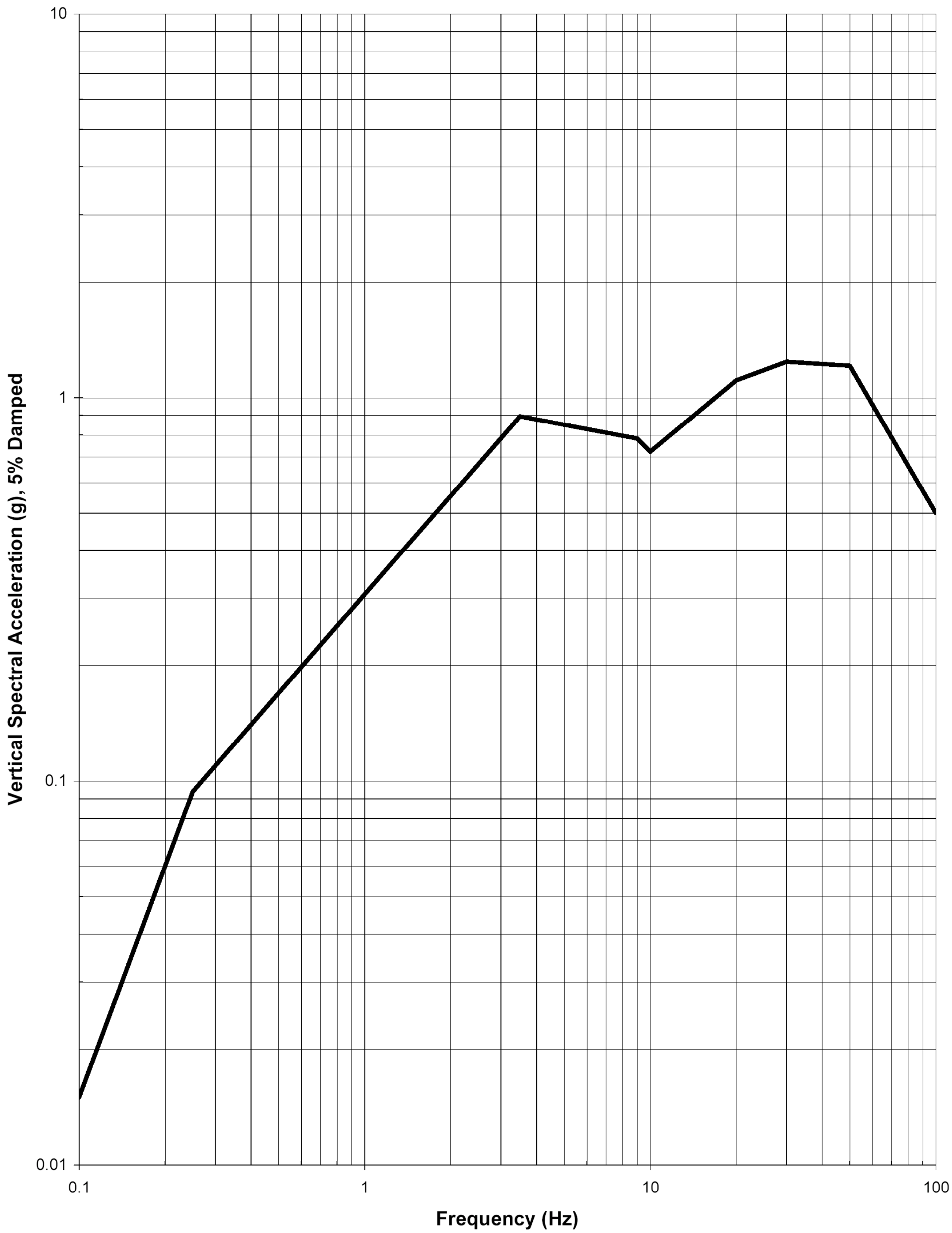


Figure 5.1-2. ESBWR Vertical SSE Design Ground Response Spectra at Foundation Level