GE-Hitachi Nuclear Energy

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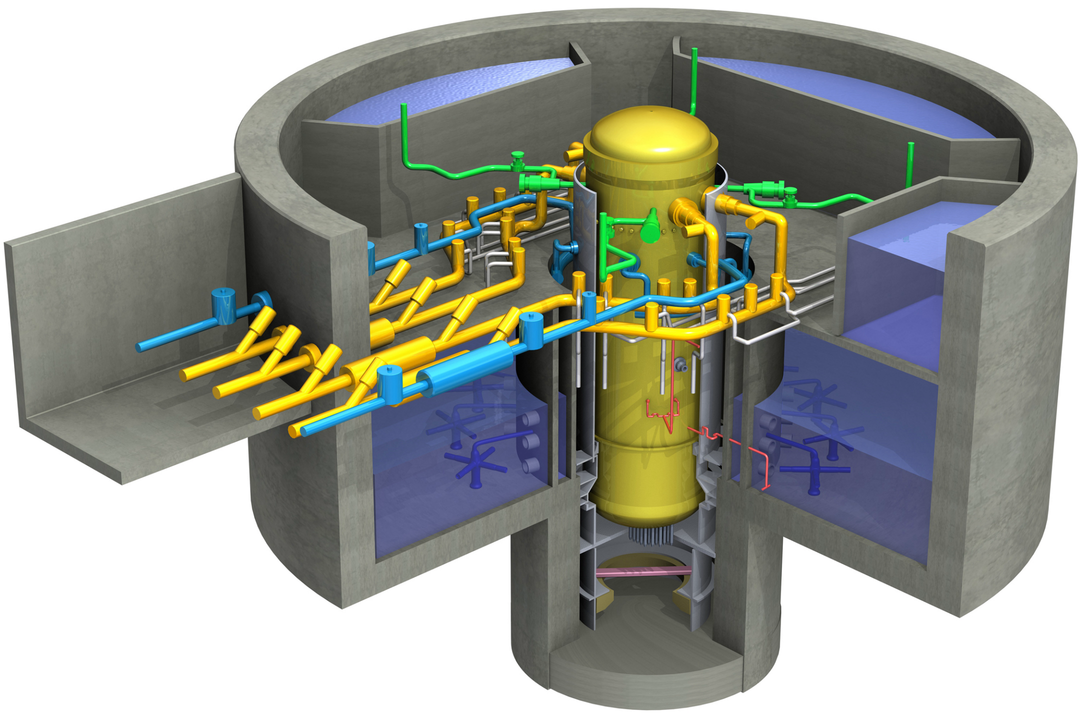

ESBWR Design Control Document
Tier 1

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Abbreviations And Acronyms List

ABWR	Advanced Boiling Water Reactor
AC	Air Conditioning
ADS	Automatic Depressurization System
AFIP	Automated Fixed In-Core Probe
AG	Analysis Guides
AHS	Auxiliary Heat Sink
AHU	Air Handling Units
AISC	American Institute of Steel Construction
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOF	Allocation of Function
AOO	Anticipated Operational Occurrence
API	American Petroleum Institute
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARM	Area Radiation Monitor
ARMS	Area Radiation Monitoring System
ASCE	American Society of Civil Engineers
ASD	Adjustable Speed Drive
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATLM	Automated Thermal Limit Monitor

ATWS	Anticipated Transients Without Scram
B21	Nuclear Boiler System
B32	Isolation Condenser System
BOP	Balance of Plant
BTP	Backfit Test Program
BTU	British Thermal Unit
BWR	Boiling Water Reactor
C	Celsius (Centigrade)
C21	Leak Detection and Isolation System
C41	Standby Liquid Control System
CB	Control Building
CBGAVS	Control Building General Area HVAC Subsystem
CFR	Code of Federal Regulations
CIRC	Circulating Water System
CIS	Containment Inerting System
CLAVS	Clean Area Ventilation Subsystem of Reactor Building HVAC
CMS	Containment Monitoring System
COL	Combined Operating License
CONAVS	Controlled Area Ventilation Subsystem of Reactor Building HVAC
COTS	Commercial Off-the-Shelf Software
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
CRHS	Control Room Habitability System
CS	Containment Safety
CSPP	Cyber Security Program Plan
CST	Condensate Storage Tank
CV	Check Valve
CWS	Chilled Water System
D11	Process Radiation Monitoring System
DAC	Design Acceptance Criteria
DBA	Design Basis Accident
DBE	Design Basis Event
DBT	Design-Basis Tornado
DC	Direct Current
DCD	Design Control Document
DCIS	Distributed Control and Information System
DCS	Drywell Cooling System
DF	Decontamination Factor
DG	Diesel Generator
DGVS	Diesel Generators HVAC Subsystem

DICS	Diverse instrumentation and Control System
DOI	Dedicated Operators Interface
DOT	Department of Transportation
DPS	Diverse Protection System
DW	Drywell
E50	Gravity Driven Cooling System
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
EL	Elevation
EMI	Electromagnetic Interference
EOF	Emergency Operations Facility
EPEN	Electrical Penetration
ERICP	Emergency Rod Insertion Control Panel
ERIP	Emergency Rod Insertion Panel
ESF	Engineered Safety Feature
ETS	Emergency Trip System
FAPCS	Fuel and Auxiliary Pools Cooling System
FB	Fuel Building

FBFPVS	Fuel Building Fuel Pool HVAC Subsystem
FBGAVS	Fuel Building General Area HVAC Subsystem
FBVS	Fuel Building HVAC System
FIV	Flow-Induced Vibration
FM	Factory Mutual
FMCRD	Fine Motion Control Rod Drive
FMEA	Factory Mutual Engineering Association
FPE	Fire Pump Enclosure
FPS	Fire Protection System
FRA	Functional Requirements Analysis
FSAR	Final Safety Analysis Report
FTDC	Fault-Tolerant Digital Controller
FW	Feedwater
FWCS	Feedwater Control System
FWRB	Feedwater Runback
G21	Fuel and Auxiliary Pools Cooling System
G31	Reactor Water Cleanup/Shutdown Cooling System
GDC	General Design Criteria/Criterion
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GEH	GE-Hitachi Nuclear Energy
GWSR	Ganged Withdrawal Sequence Restriction
HCU	Hydraulic Control Unit

HCW	High Conductivity Waste
HEPA	High-Efficiency Particulate Air/Absolute
HF	High Frequency
HFE	Human Factors Engineering
HFEITS	Human Factors Engineering Issue Tracking System
HP	High Pressure
HPM	Human Performance Monitoring
HPNSS	High Pressure Nitrogen Supply System
HRA	Human Reliability Assessment
HSI	Human System Interface
HVAC	Heating, Ventilation and Air Conditioning
IC	Ion Chamber
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
IPC	Isolation Power Center
ISI	In-Service Inspection
ISLOCA	Interfacing-Systems Loss-of-Coolant Accident
ITA	Inspections, Tests or Analyses
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
ITP	Initial Test Program

LCW	Low Conductivity Waste
LD	Load Driver
LFCV	Low Flow Control Valve
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
LPSP	Low-Power Set Point
LPZ	Low-Population Zone
LWMS	Liquid Waste Management System
MC	Motor Controllers
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MCR	Main Control Room
MCRP	Main Control Room Panel
MFAP	Main Fire Alarm Panel
MMI	Man-Machine Interface
MMIS	Man Machine Interface System
MPEN	Penetration, Mechanical
MRBM	Multi-Channel Rod Block Monitor
MSIV	Main Steam Isolation Valve

MSL	Main Steam Line
MT	Main Turbine
MVP	Mechanical Vacuum Pump
MW	Megawatt
MWS	Makeup Water System
NA	Not Applicable
NBS	Nuclear Boiler System
NCA	Nuclear Compliance Archives
NDE	Nondestructive Examination
NFPA	National Fire Protection Association
NICWS	Nuclear Island Chilled Water Subsystem
NMS	Neutron Monitoring System
NPHS	Normal Power Heat Sink
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NS	Non-seismic
NSSS	Nuclear Steam Supply System
NUREG	Nuclear Regulatory Commission technical report designation
OBCV	Overboard Control Valve
OER	Operating Experience Review
OGS	Offgas System
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLMLHGR	Operating Limit Minimum Linear Heat Generation Rate

OPRM	Oscillation Power Range Monitor
P10	Make Up Water System
P25	Chilled Water System
P52	Instrument Air System
P54	High Pressure Nitrogen Supply System
PAS	Plant Automation System
PC	Passive Component
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PDS	Previously Developed Software
PG	Power Generation
PIP	Plant Investment Protection
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PMWP	Probable Maximum Winter Precipitation
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitoring
PSWS	Plant Service Water System
QA	Quality Assurance
R	Roentgen
R31	Raceway System
RACS	Rod Action Control Subsystem

RAP	Reliability Assurance Plan
RAPI	Rod Action and Position Information
RB	Reactor Building
RBCC	Rod Brake Controller Cabinet
RBVS	Reactor Building HVAC System
RCC	Remote Communication Cabinet
RCCV	Reinforced Concrete Containment Vessel
RCCW	Reactor Component Cooling Water
RCCWS	Reactor Component Cooling Water System
RCIS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
REPAVS	Refueling and Pool Area Ventilation System
RF	Radio Frequency
RG	Regulatory Guide
RHX	Regenerative Heat Exchanger
RMS	Radiation Monitoring System/Subsystem
RPS	Reactor Program System
RPV	Reactor Pressure Vessel
RRPS	Reference Rod Pull Sequence
RSS	Reactor Shutdown System
RTA	Requirements Traceability Analysis
RTIF	Reactor Trip and Isolation Function(s)
RTNSS	Regulatory Treatment of Non-safety Systems

RW	Radioactive Waste
RWCRVS	Radwaste Building Control Room HVAC
RWCU	Reactor Water Cleanup
RWGAVS	Radwaste Building General Area HVAC Subsystem
RWM	Rod Worth Minimizer
RWVS	Radwaste Building HVAC System
SAIV	Steam Auxiliary Isolation Valve
SAS	Service Air System
SB	Service Building
SCMP	Software Configuration Management Plan
SCRRI	Selected Control Rod Run-In
SCU	Signal Conditioning Units
SDC	Shutdown Cooling
SDD	Software Design Documentation
SDP	Software Development Plan
SECY	Secretary of the Commission, Office of the (NRC)
SF	Service Water Building
SIP	Software Installation Plan
SIT	Structural Integrity Test
SIU	Signal Interface Unit
SJAE	Steam Jet-Air Ejector
SLC	Standby Liquid Control
SMP	Software Management Plan

SOMP	Software Operations and Maintenance Plan
SP	Setpoint
SPC	Suppression Pool Cooling
SPE	Software Project Engineering
SPEN	Penetration, Hatch, Equip or Personnel
SPTM	Suppression Pool Temperature Monitoring
SQAP	Software Quality Assurance Plan
SRI	Safety Related Items
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSC	Structure, System or Component
SSE	Safe Shutdown Earthquake
SSLC	Safety System Logic and Control
SSP	Software Safety Plan
SSPV	Scram Solenoid Pilot Valve
STPM	Simulated Thermal Power Monitor
STRAP	Scram Time Recording and Analysis Panel
SV	Safety Valve
SVVP	Software Verification and Validation Plan
SW	Software
SWMS	Solid Waste Management System

T11	Containment Vessel
T15	Passive Containment Cooling System
T31	Containment Inerting System
T62	Containment Monitoring System
TAF	Top of Active Fuel
TB	Turbine Building
TBD	To Be Determined
TBS	Terminal Board Scram
TBV	Turbine Block Valve
TBVS	Turbine Building Ventilation System
TCCWS	Turbine Component Cooling Water System
TCV	Temperature Control Valve
TG	Turbine Generator
TGCS	Turbine Generator Control System
TGSS	Turbine Gland Seal/Sealing System
TLU	Trip Logic Unit
TMSS	Turbine Main Steam System
TSC	Technical Support Center
TSCVS	Technical Support Center HVAC Subsystem
TSV	Turbine Stop Valve
U50	Equipment and Floor Drain System
UAT	Unit Auxiliary Transformer
UHS	Ultimate Heat Sink

UL	Underwriter's Laboratories Inc.
UPS	Uninterruptible Power Supply
USNRC	United States Nuclear Regulatory Commission
V	Volt
VAC	Volts Alternating Current
VDC	Volts Direct Current
VDU	Video Display Unit
VW	Vertical Wall
WW	Wetwell

1. INTRODUCTION

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

1.1 DEFINITIONS AND GENERAL PROVISIONS

1.1.1 Definitions

The definitions below apply to terms which may be 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.

Analysis means a 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.

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 (for electrical systems or equipment) is the designation applied to a given safety-related system or set of components that is physically, electrically, and functionally independent from other redundant sets of components.

Equipment Identification Number as used in Tier 1 means the designation on a Tier 1 figure and is not representative of an actual equipment number or tag number.

Equipment Qualification

For purposes of ITAAC:

> ***Environmental Qualification:*** Type tests, or type tests and/or analyses, of the safety-related mechanical components and electrical equipment demonstrate qualification to applicable normal, abnormal and design basis accident conditions without loss of the safety-related function for the time needed to perform the safety-related function. These harsh 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 the associated ITAAC, 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 as being subject to the environmental qualification requirements.

As used in this paragraph, “safety related mechnical components” refers to mechanical parts, subassemblies or assemblies that are categorized as Quality Group A, B or C. Mechanical components qualification also may be by type tests, analyses or a combination of tests and analyses of individual parts or subassemblies or of complete assemblies rather than by testing the individual parts or subassemblies separately.

Safety-related equipment located in a mild environment will be qualified for their environmental requirements through specifications and certifications to the environments; however, for a mild environment, only safety-related digital instrumentation and control equipment will be addressed by ITAAC in Tier 1, consistent with NRC guidance in NUREG-0800, Section 14.3. Additionally, EMI susceptibility and emissions qualification is performed by type testing for the safety-related digital instrumentation and control equipment and is not specifically addressed in an ITAAC. ITAAC address analyses of material data for safety-related mechanical equipment located in a harsh environment.

Seismic Qualification: 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) may be used 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.

Functional Arrangement/Physical Arrangement (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.

Functional Arrangement (for a System) means the physical arrangement of systems and components to provide the service for which the system in intended, and which is described in the system Design Description.

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.

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, but are not limited to, walk-downs, configuration checks, measurements of dimensions, and non-destructive examinations.

Inspect for Retrievability of a display means to visually observe that the specified information appears on a monitor when summoned by the operator.

Operate means the actuation, control, running, and/or shutting down (*e.g.,* closing, turning off) of equipment.

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).

Test or **Testing** 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 functional arrangement of a system, as used in an ITAAC, means verifying that the system is constructed as depicted in the Tier 1 design description and design drawings, including equipment and instrument locations, if applicable.

1.1.2 General Provisions

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

1.1.2.1 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.

If an inspection, test, or analyses requirement does not specify the temperature or other conditions under which a test must be run, then the test conditions are not constrained.

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.

1.1.2.2 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.

Many of the Acceptance Criteria include the words "A report exists and concludes that …." This concept indicates that the ITAAC for that Design Commitment will be met when it is confirmed that appropriate documentation exists and the documentation shows that the Design Commitment is met. Appropriate documentation may be a single document or a collection of documents that show that the stated Acceptance Criteria are met. Examples of appropriate documentation include design reports, test reports, inspection reports, analysis reports, evaluation reports, design and manufacturing procedures, certified data sheets, commercial dedication procedures and records, quality assurance records, calculation notes, and equipment qualification data packages.

Many entries in the ITA column of the ITAAC tables include the words “Inspection will be performed for the existence of a report verifying…” When these words are used it indicates that the ITA is tests, type tests, analyses, or a combination of tests, type tests, and analyses and a report will be produced documenting the results. This report will be available for inspection.

Many ITAAC are only a reference to another Tier 1 location, either a section, subsection, or ITAAC table entry (for example, “See Tier 1, Section…”). A reference to another ITAAC location is always in both the ITA and acceptance criteria columns for a design commitment. This reference is an indication that the ITA and Acceptance Criteria for that Design Commitment are satisfied when the referenced ITA are completed and the Acceptance Criteria for the referenced Tier 1 sections, subsections, or table entries are satisfied.

For those nonsystem-based ITAAC, which address piping and equipment qualification, the ITA and Acceptance Criteria may be satisfied on a system-by-system basis so as not to delay completion of ITAAC for a particular system. In this manner, a system may be turned over for operation following verification of the information needed to satisfy the nonsystem-based ITAAC. Documentation of completion of the ITAAC for a particular system will be retained in a manner that will allow verification of completion of the ITAAC for the nonsystem-based ITAAC. Notification to the NRC of completion of the nonsystem-based ITAAC also may be on a system basis throughout construction; however, a separate notification to the NRC will be made upon final completion of the nonsystem-based ITAAC for purposes of ensuring that the Acceptance Criteria have been met.

1.1.2.3 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).

1.1.2.4 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 another general illustration. For instrumentation and control 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.

1.1.2.5 Rated Reactor Core Thermal Power

The initial rated reactor core thermal power for the standard ESBWR is 4500 megawatts thermal (MWt).

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.

2.1 NUCLEAR STEAM SUPPLY

The following subsections describe the major Nuclear Steam Supply Systems (NSSS) components of the Reactor Pressure Vessel System and the Nuclear Boiler System. This section also describes the natural circulation process for the ESBWR.

2.1.1 Reactor Pressure Vessel System

Design Description

The RPV system generates heat and boils water to steam in a direct cycle. The functional arrangement of the RPV system is that it includes the reactor core and reactor internals (see Figure 2.1.1-1). The chimney provides an additional elevation head (or driving head) necessary to sustain natural circulation flow through the RPV. The chimney also forms an annulus separating the subcooled recirculation flow returning downward from the steam separators and feedwater from the upward steam-water mixture flow exiting the core. The steam is separated from the steam-water mixture by passing the mixture sequentially through an array of steam separators attached to a removable cover on the top of the chimney assembly, and through the steam dryer, resulting in outlet dry steam. The water mixes with the feedwater as it comes into the RPV through the feedwater nozzle. RPV internals consist of core support structures and other equipment.

The RPV is located in the containment. Internal component locations are shown on Figure 2.1.1-1.

(1) The functional arrangement of the RPV system is as described in the Design Description of this Subsection 2.1.1, Table 2.1.1-1 and Figure 2.1.1-1.

(2) The key dimensions (and acceptable variations) of the as-built RPV are as described in Table 2.1.1-2.

(3) The RPV components identified in Table 2.1.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

(4) Pressure boundary welds in components identified in Table 2.1.1-1 as ASME Code Section III meet ASME Code Section III requirements.

(5) The components identified as ASME Code Section III retain their pressure boundary integrity under internal pressure that will be experienced during service.

(6) The seismic Category I equipment identified in Table 2.1.1-1 can withstand seismic design basis loads without loss of safety function.

(7) RPV surveillance 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. 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.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.1-3 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

Reactor Pressure Vessel System Mechanical Equipment

Equipment Name	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated Valve	Loss of Motive Power Position	MCR Alarms
RPV	Yes	Yes	Yes	-	-	-	-
Core support structures (shroud, shroud support, top guide, core plate, control rod guide tubes and fuel supports) which have a support function	Yes	Yes	-	-	-	-	-
Chimney and Partitions	-	-	-	-	-	-	-
Chimney head and steam separators assembly	-	-	-	-	-	-	-
Steam dryer assembly	-	-	-	-	-	-	-

Table 2.1.1-2

Key Dimensions of RPV Components and Acceptable Variations

Description	**Dimension/ Elevation** (Figure 2.1.1-1)	**Nominal Value** (mm, in.)	**Acceptable Variation(s)** (mm, in.)
RPV bottom head inside invert elevation	A	0	Reference 0
Top of core plate elevation	B	[4178, 164.5]	[±16, 0.63]
Bottom of top guide elevation	C	[7718, 303.9]	[±16, 0.63]
RPV top head inside invert elevation	D	[27560, 1085]	[±100, 3.94]
RPV inside diameter (inside cladding)	E	[7112, 280.0]	[±51, 2.01]]
RPV wall thickness in beltline (including cladding)	F	[182, 7.17]	[190.5 max, 7.50 max]

Table 2.1.1-3

ITAAC For Reactor Pressure Vessel System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the RPV system is as described in the Design Description of this Section 2.1.1, Table 2.1.1-1 and Figure 2.1.1-1.	Inspections of the as-built RPV System will be conducted.	Report(s) document that the RPV system and core arrangement conforms to the functional arrangement described in the Design Description of this Section 2.1.1, Table 2.1.1-1 and Figure 2.1.1-1.
2. The key dimensions (and acceptable variations) of the as-built RPV are as described in Table 2.1.1-2.	Inspection of the as-built RPV key dimensions (and acceptable variations thereof) will be conducted.	Report(s) document that the RPV conforms to the key dimensions (and acceptable variations) described in Table 2.1.1-2.
3. The RPV components identified in Table 2.1.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspections will be conducted of the as-built components as documented in the ASME Code design reports.	Report(s) document that the ASME Code Section III design reports exist for the as-built components identified in Table 2.1.1-1 as ASME Code Section III.
4. Pressure boundary welds in components identified in Table 2.1.1-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	Report(s) document that a report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
5. The components in Table 2.1.1-1 identified as ASME Code Section III retain their pressure boundary integrity under internal pressure that will be experienced during service.	A hydrostatic test will be conducted on those components of the RPV system required to be hydrostatically tested by the ASME Code.	Report(s) document that 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.

Table 2.1.1-3

ITAAC For Reactor Pressure Vessel System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The seismic Category I equipment identified in Table 2.1.1-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.1.1-1 is located on a seismic structure. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	Report(s) document that: i) The seismic Category I equipment identified in Table 2.1.1-1 is located on a seismic structure. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
7. RPV surveillance 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. 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.	Inspections of the as-built RPV system will be conducted for implementation of the RPV surveillance specimens, neutron dosimeter, and brackets. An analysis is performed to determine the location of the peak fluence.	Report(s) document the RPV surveillance specimens and neutron dosimeters are provided and that brackets are installed at the location(s) of calculated peak fluence determined by an analysis of the as-built configuration.

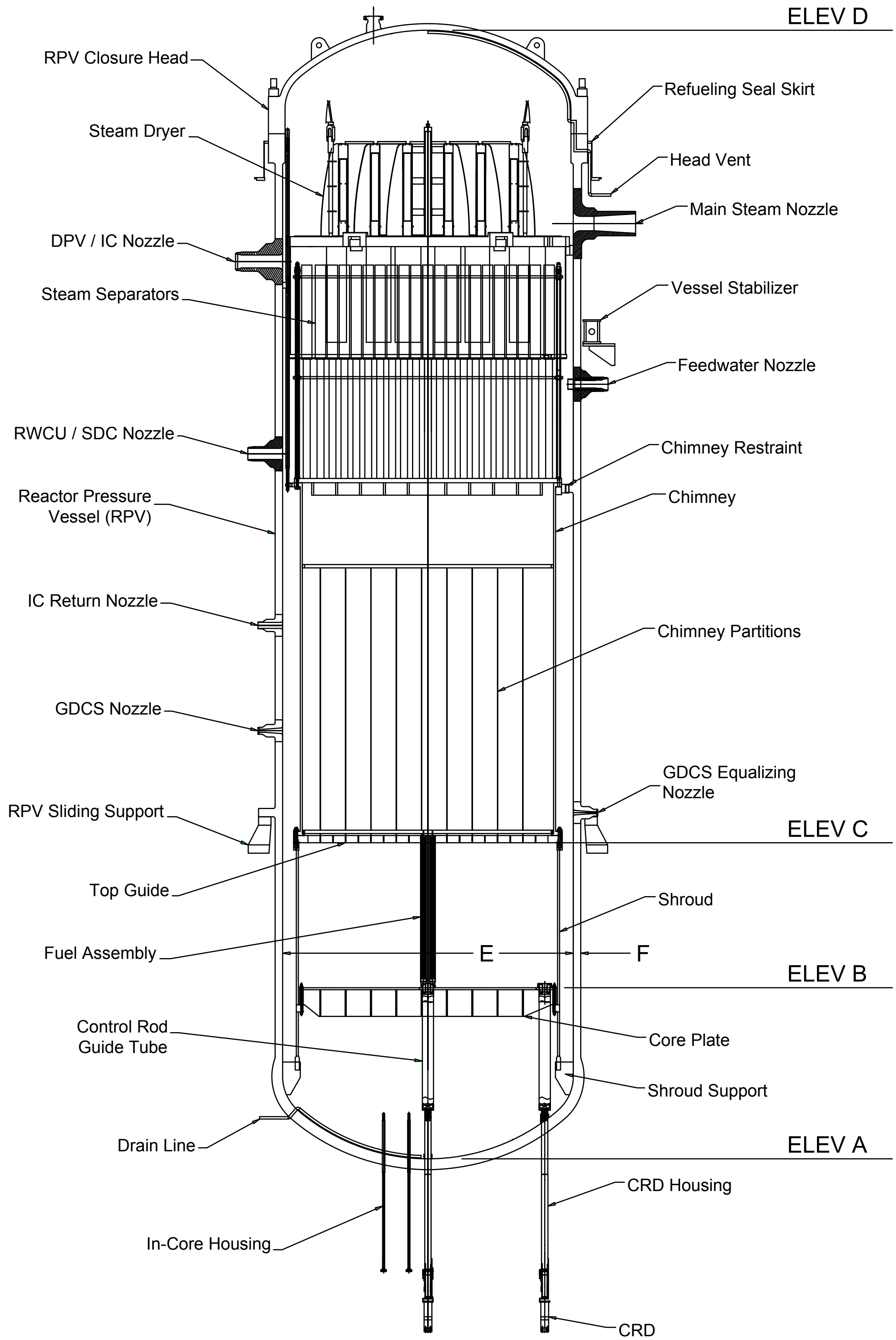


Figure 2.1.1-1. Reactor Pressure Vessel System Key Features Layout

2.1.2 Nuclear Boiler System

Design Description

The NBS generates steam from feedwater and transports steam from the RPV to the main turbine.

(1) The functional arrangement of the NBS System is as described in the Design Description of this Section 2.1.2, Tables 2.1.2-1 and 2.1.2-2, and Figures 2.1.2-1, 2.1.2-2, and 2.1.2-3.

(2) ASME Code Section III

a. The components identified in Table 2.1.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

b. The piping identified in Table 2.1.2-1 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

(3) Pressure Boundary Welds

a. Pressure boundary welds in components identified in Table 2.1.2-1 as ASME Code Section III meet ASME Code Section III requirements.

b. Pressure boundary welds in piping identified in Table 2.1.2-1 as ASME Code Section III meet ASME Code Section III requirements.

(4) Pressure Boundary Integrity

a. The components identified in Table 2.1.2-1 as ASME Code Section III retain their pressure boundary integrity at internal pressures that will be experienced during service.

b. The piping identified in Table 2.1.2-1 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

(5) Seismic Capability

a. The seismic Category I equipment identified in Tables 2.1.2-1 and 2.1.2-2 can withstand seismic design basis loads without loss of safety function.

b. Each of the lines identified in Table 2.1.2-1 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

(6) a. Each of the NBS System safety-related divisions identified in Table 2.1.2-2 is powered from its respective safety-related division

b. Separation is provided between NBS System safety-related divisions, and between safety-related divisions and nonsafety-related cable.

(7) Each mechanical train of safety-related NBS equipment located in the Reactor Building outside the drywell is physically separated from the other trains.

(8) Instrumentation and Control

a. Control Room alarms, displays, and/or controls provided for the NBS System are defined in Table 2.1.2-2.

b. The MSIVs close upon any of the following conditions:
   - Main Condenser Vacuum Low (Run mode)
   - Turbine Area Ambient Temperature High
   - MSL Tunnel Ambient Temperature High
   - MSL Flow Rate High
   - Turbine Inlet Pressure Low
   - Reactor Water Level Low

(9) Repositional valves (not including the DPVs (squib-activiated valves)) designated in Table 2.1.2-2 as having an active safety-related function open, close, or both open and also close under design differential pressure, fluid flow, and temperature conditions.

(10) The pneumatically operated valve(s) shown in Figure 2.1.2-2 closes (opens) if either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve(s) is lost.

(11) Check valves designated in Table 2.1.2-1 as having an active safety-related function open, close, or both open and also close under design system pressure, fluid flow, and temperature conditions.

(12) The throat diameter of each MSL flow restrictor is sized for design choke flow requirements.

(13) Each MSL flow restrictor has taps for two instrument connections to be used for monitoring the flow through each MSL.

(14) The combined steamline volume from the RPV to the main steam turbine stop valves and steam bypass valves is sufficient to meet the assumptions for AOOs and infrequent events.

(15) The MSIVs are capable of fast closing under design differential pressure, fluid flow and temperature conditions.

(16) When all MSIVs are closed by normal means, the combined leakage through the MSIVs for all four MSLs will be less than or equal to the design bases assumption value.

(17) The opening pressure for the SRVs mechanical lift mode satisfies the overpressure protection analysis.

(18) The opening time for the SRVs (in the overpressure operation of self-actuated or mechanical lift mode) from when the pressure exceeds the valve set pressure to when the valve is fully open shall be less than or equal to the design opening time.

(19) The steam discharge capacity of each SRV satisfies the overpressure protection analysis.

(20) The opening pressure for the SVs satisfies the overpressure protection analysis.

(21) The opening time for the SVs from when the pressure exceeds the valve set pressure to when the valve is fully open shall be less than or equal to the design opening time.

(22) The steam discharge capacity of each SV satisfies the overpressure protection analysis.

(23) The relief-mode actuator (and safety-related appurtenances) can open each SRV with the drywell pressure at design pressure.

(24) When actuated by an initiator, the booster assembly opens each DPV in less than or equal to the design opening time and design conditions.

(25) Each DPV minimum flow capacity is sufficient to support rapid depressurization of the RPV.

(26) The equipment qualification of the NBS components is addressed in Tier 1, Section 3.8.

(27) The containment isolation portions of the NBS are addressed in Tier 1, Subsection 2.15.1.

Refer to Subsection 2.2.15 for "Instrumentation and Controls Compliance with IEEE Standard 603."

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.1.2-3 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria for the NBS.

Table 2.1.2-1

Nuclear Boiler System Mechanical Equipment

Equipment Name	Equipment ID on Figure 2.1.2-2	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position
Main steam lines to the seismic restraint in the steam tunnel	-	Yes	Yes	Yes	-	-	-
Inboard Main steam isolation valves	V8 (Typ. of 4)	Yes	Yes	Yes	Yes	Yes	Closed
Outboard Main steam isolation valves	V9 (Typ. of 4)	Yes	Yes	Yes	Yes	Yes	Closed
MSIV actuator and support hardware and associated structural supports	For valves V8, V9 (Typ. of 8)	Portions may be connected to ASME Code systems	Yes	Portions may be connected to RCPB	-	-	-
Main steam flow restrictors	-	Yes	Yes	Yes	-	-	-
Steam line drain/bypass subsystem	-	Yes	Yes	Portions	Inboard drains valves (see Tier 1, Subsection 2.15.1)	-	-
Feedwater piping from RPV to seismic restraint upstream of isolation shutoff valve	-	Yes	Yes	Yes	-	-	-
Safety valves (SV)	V7 (Typ. of 8)	Yes	Yes	Yes	No	No	-
Safety relief valves	V6	Yes	Yes	Yes	No	Yes (in	Closed for relief

Table 2.1.2-1

Nuclear Boiler System Mechanical Equipment

Equipment Name	Equipment ID on Figure 2.1.2-2	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position
(SRV)	(Typ. of 10)					relief mode)	mode
Depressurization valves	V5 (Typ. of 8 total)	Yes	Yes	Yes	No	Yes	Fail as is
RPV head vent subsystem	V1, V2, and V3	Yes	Yes	Yes	No	Yes (no active safety function)	Fail as is
System instrumentation – detection and monitoring (indication in Main Control Room): • Position of MSIVs • Position of DPVs • Position of SRVs • Differential pressure between two feedwater lines • Continuity circuit for each DPV squib device • Continuity circuit for each	-	Safety-related fluid portions only	Yes	Safety-related fluid portions only	Safety-related fluid portions only (see Tier 1, Subsection 2.15.1)	-	-

Table 2.1.2-1

Nuclear Boiler System Mechanical Equipment

Equipment Name	Equipment ID on Figure 2.1.2-2	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position
SRV							
SRV discharge lines	-	Yes	Yes	-	-	-	-
SRV discharge line vacuum relief valves	V18, V19 (Typ.)	Yes	Yes	No	No	-	-
SRV discharge line quencher	Q1 (Typ.)	Yes	Yes	No	-	-	-
Feedwater isolation valves	V14, V17	Yes	Yes	No	No	Yes	Closed (gravity close on loss of system pressure)
Feedwater inboard isolation check valves	V12, V15	Yes	Yes	Yes	Yes (will close on reverse flow)	No	-
Feedwater outboard isolation check valves	V13, V16	Yes	Yes	Yes	Yes (active function to close on LOCA and will close on reverse flow)	Yes	- (gravity close on loss of system pressure)

Note: A dash means not applicable.

Table 2.1.2-2

Nuclear Boiler System Electrical Equipment

Equipment Name	Equipment ID on Figure 2.1.2-2	Control Q-DCIS/ DPS[1]	Safety-Related Electrical Equipment	Safety-Related Display	Active Function	Seismic Category I	Remotely Operated	Containment Isolation Valve Actuator
Inboard Main steam isolation valves	V8 (Typ. of 4)	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Outboard Main steam isolation valves	V9 (Typ. of 4)	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Safety relief valves (SRV)	V6 (Typ. of 10)	Yes (ADS – See Section 2.2.16)	Yes	Yes	Yes	Yes	Yes	No
Safety valves (SV)	V7 (Typ. of 8)	No	Yes – Position Indicator Only	Yes	No	Yes	No	No
Depressurization valves	V5 (Typ. of 8 total)	Yes	Yes	Yes	Yes	Yes	Yes	No
Feedwater isolation valves	V14, V17	Yes	Yes	Yes	Yes	Yes	Yes	No
Feedwater utboard isolation check valves	V13, V16	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Reactor Pressure Transmitters (1 each in 4 divisions)	-	Yes	Yes	Yes	Yes	Yes	-	-
Reactor water level transmitters (1 each in 4 divisions)	-	Yes	Yes	Yes	Yes	Yes	-	-

Table 2.1.2-2

Nuclear Boiler System Electrical Equipment

Equipment Name	Equipment ID on Figure 2.1.2-2	Control Q-DCIS/ DPS[1]	Safety-Related Electrical Equipment	Safety-Related Display	Active Function	Seismic Category I	Remotely Operated	Containment Isolation Valve Actuator
MSIV Isolation Logic	-	Yes	Yes	Yes	Yes	Yes	-	-
Leak Detection and Isolation System Logic	-	Yes	Yes	Yes	Yes	Yes	-	-

Note: A dash means not applicable.

Note 1: See Section 2.2.7.

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the NBS System is as described in the Design Description of this Section 2.1.2, Tables 2.1.2-1 and 2.1.2-2 and Figures 2.1.2-1, 2.1.2-2, and 2.1.2-3.	Inspection of the as-built system will be performed.	Report(s) document that the as-built NBS System conforms to the functional arrangement described in the Design Description of this Section 2.1.2, Tables 2.1.2-1 and 2.1.2-2 and Figures 2.1.2-1, 2.1.2-2, and 2.1.2-3.
2. ASME Code Section III		
a) The components identified in Table 2.1.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	Report(s) document that the ASME Code Section III design reports exist for the as-built components identified in Table 2.1.2-1 as ASME Code Section III.
b) The piping identified in Table 2.1.2-1 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	Report(s) document that the ASME code Section III design reports exist for the as-built piping identified in Table 2.1.2-1 as ASME Code Section III.

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. Pressure Boundary Welds		
a) Pressure boundary welds in components identified in Table 2.1.2-1 as ASME Code Section III meet ASME Code Section III requirements.	Pressure boundary welds in components identified in Table 2.1.2-1a as ASME Code Section III meet ASME Code Section III requirements.	Report(s) document that a report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
b) Pressure boundary welds in piping identified in Table 2.1.2-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	Report(s) document that a report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4. Pressure Boundary Integrity		
a) The components identified in Table 2.1.2-1 as ASME Code Section III retain their pressure boundary integrity at internal pressures that will be experienced during service.	A hydrostatic test will be conducted on those code components of the NBS System required to be hydrostatically tested by the ASME Code	Report(s) document that the results of the hydrostatic test of the ASME Code components of the NBS System conform to the requirements in the ASME Code, Section III.
b) The piping identified in Table 2.1.2-1 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be conducted on those code components of the System required to be hydrostatically tested by the ASME Code.	Report(s) document that the results of the hydrostatic test of the ASME Code components of the System conform with the requirements in the ASME Code, Section III.

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Seismic Capability		Report(s) document that:
a) The seismic Category I equipment identified in Tables 2.1.2-1 and 2.1.2-2 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.1.2-1a are located on the Nuclear Island.	i) The seismic Category I equipment identified in Tables 2.1.2-1 and 2.1.2-2 is located on a seismic structure.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
b) Each of the lines identified in Table 2.1.2-1 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	Report(s) document that a report exists and concludes that each of the as-built lines identified in Table 2.1.2-1 for which functional capability is required meets the requirements for functional capability.

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6a). Each of the NBS System safety-related divisions identified in Table 2.1.2-2 is powered from its respective safety-related division. b) Separation is provided between NBS System safety-related divisions, and between safety-related divisions and nonsafety-related cable.	See Tier 1, Subsections 2.13.1, 2.13.3, or 2.13.5, as appropriate. See Tier 1, Subsection 2.2.15.	See Tier 1, Subsection 2.13.1, 2.13.3, or 2.13.5, as appropriate. See Tier 1, Subsection 2.2.15.
7. Each mechanical train of safety-related NBS equipment located in the Reactor Building outside the drywell is physically separated from the other trains.	Inspections of the as-built NBS equipment trains will be performed.	Report(s) document that each mechanical train of NBS equipment located in the Reactor Building outside the drywell is physically separated from the other trains by structural and/or fire barriers.

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Instrumentation and Control a) Control Room alarms, displays, and/or controls provided for the NBS System are defined in Table 2.1.2-2.	Inspections will be performed on the as-built Control Room alarms, displays, and/or controls for the NBS System.	Report(s) document that alarms, displays, and/or controls exist or can be retrieved in the Control Room as defined in Table 2.1.2-2.
b) The MSIVs close upon any of the following conditions: - Main Condenser Vacuum Low (Run mode) - Turbine Area Ambient Temperature High - MSL Tunnel Ambient Temperature High - MSL Flow Rate High - Turbine Inlet Pressure Low - Reactor Water Level Low	Valve closure tests will be performed on the as-built MSIVs using simulated signals.	Report(s) document that the MSIVs close upon generation of any of the following simulated signals: - Main Condenser Vacuum Low (Run mode) - Turbine Area Ambient Temperature High - MSL Tunnel Ambient Temperature High - MSL Flow Rate High - Turbine Inlet Pressure Low - Reactor Water Level Low
9. Repositional valves (not including DPVs (squib-activated valves)) designated in Table 2.1.2-2 as having an active safety-related function to open, close, or both open and also close under design differential pressure, fluid flow, and temperature conditions.	Tests of installed valves will be performed for opening, closing, or both opening and also closing under system preoperational differential pressure, fluid flow, and temperature conditions.	Report(s) document that, upon receipt of the actuating signal, each valve opens, closes, or both opens and also closes, depending upon the valve's safety function.

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. The pneumatically operated valve(s) shown in Figure 2.1.2-2 closes (opens) if either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve(s) is lost.	Tests will be conducted on the as-built valve(s).	Report(s) document that the pneumatically operated valve(s) shown in Figure 2.1.2-2 closes (opens) when either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve(s) is lost.
11. Check valves designated in Table 2.1.2-1 as having an active safety-related function open, close, or both open and also close under design system pressure, fluid flow, and temperature conditions.	Tests of installed valves for opening, closing, or both opening and also closing, will be conducted under system preoperational pressure, fluid flow, and temperature conditions.	Report(s) document that, based on the direction of the differential pressure across the valve, each CV opens, closes, or both opens and also closes, depending upon the valve's safety functions.
12. The throat diameter of each MSL flow restrictor is sized for design choke flow requirements.	Inspection of the as-built MSL flow restrictor will be performed and measurements taken.	Report(s) document that the throat diameter of each MSL flow restrictor is less than or equal to 355 mm (14 in.).
13. Each MSL flow restrictor has taps for two instrument connections to be used for monitoring the flow through each MSL.	Inspections of the as-built installation of the MSL flow restrictor will be conducted to verify that it provides for two instrument connections.	Report(s) document that the as-built MSL flow restrictor provides for two instrument connections.
14. The combined steamline volume from the RPV to the main steam turbine stop valves and steam bypass valves is sufficient to meet the assumptions for AOOs and infrequent events.	Analyses/calculations will be performed using the as-built dimensions of the steamlines to determine the combined steam line volume. The calculational results will be documented in a report.	Report(s) document that the combined steamline volume is greater than or equal to 135 m^3 (4767 ft^3).

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
15. The MSIVs are capable of fast closing under design differential pressure, fluid flow and temperature conditions.	Tests of the as-built MSIV will be conducted under preoperational test conditions or type testing of an MSIV will be conducted in accordance with the design and purchase specifications to demonstrate that the MSIVs will fast close.	Report(s) document that testing demonstrates MSIVs are capable of fast closure in not less than 3 seconds and not more than 5 seconds.
16. When all MSIVs are closed by normal means, the combined leakage through the MSIVs for all four MSLs will be less than or equal to the design bases assumption value.	Tests at preoperational conditions along with analysis will be performed on the as-built MSIVs to determine the leakage as adjusted to the specified design conditions.	Report(s) document that, when all MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to a total combined leakage (corrected to standard conditions) of ~0.0623 m^3/minute (~2.2 ft^3/minute) for post-LOCA leakage.
17. The opening pressure for the SRVs mechanical lift mode satisfies the overpressure protection analysis.	Type test (at a facility) or setpoint test will be conducted in accordance with the ASME Code to certify the valve.	Report(s) document that testing/type testing verifies the mechanical lift nominal setpoint pressure of 8.366 $\pm$ 0.251 MPa gauge (1213 $\pm$ 36.39 psig).
18. The opening time for the SRVs (in the overpressure operation of self-actuated or mechanical lift mode) from when the pressure exceeds the valve set pressure to when the valve is fully open shall be less than or equal to the design opening time.	Analysis and type tests (at a test facility) will be conducted in accordance with the ASME Code to ensure that the valves open within the design opening time.	Report(s) document that tests and analyses exist and conclude that opening time for the SRVs for the overpressure operation mode is less than or equal to 0.5 seconds.

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
19. The steam discharge capacity of each SRV satisfies the overpressure protection analysis.	Type tests (at a facility) will be conducted in accordance with the ASME Code for relief valve certification.	Report(s) document that valve capacity stamping on each SRV records the certified capacity at rated setpoint of 138 kg/s (304 lbm/s) minimum.
20. The opening pressure for the SVs satisfies the overpressure protection analysis.	Type tests (at a facility) or setpoint tests will be conducted in accordance with the ASME Code to certify the valve.	Report(s) document that testing/type testing verifies the mechanical lift nominal setpoint pressure of 8.503 ± 0.255 MPa gauge (1233 ± 36.99 psig).
21. The opening time for the SVs from when the pressure exceeds the valve set pressure to when the valve is fully open shall be less than or equal to the design opening time.	Analysis and type tests (at a test facility) will be conducted in accordance with the ASME Code to ensure that the valves open within the design opening time.	Report(s) document that tests and analyses exist and conclude that opening time for the SVs is less than or equal to 0.5 seconds.
22. The steam discharge capacity of each SV satisfies the overpressure protection analysis.	Type tests (at a facility) will be conducted in accordance with the ASME Code for relief valve certification.	Report(s) document that valve capacity stamping on each SV records the certified capacity at rated setpoint of 140.2 kg/s (309 lbm/s) minimum.
23. The relief-mode actuator (and safety-related appurtenances) can open each SRV with the drywell pressure at design pressure.	An analysis and/or type test will be performed to demonstrate the capacity of the relief-mode actuation for each SRV.	Test/analysis report(s) conclude that the relief-mode actuation has the capacity to lift the SRVs to the full open position one time with the drywell pressure at the drywell design pressure.

Table 2.1.2-3

ITAAC For The Nuclear Boiler System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
24. When actuated by an initiator, the booster assembly opens each DPV in less than or equal to the design opening time and design conditions.	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 each DPV within the design opening time and design conditions. Type testing and analyses (as needed) will verify minimum and maximum capacity at vessel pressure.	Report(s) document that tests and analyses conclude that each 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.
25. Each DPV minimum flow capacity is sufficient to support rapid depressurization of the RPV.	Analyses and type tests (at a test facility) will be performed.	Test reports and analyses exist and conclude that the DPV flow capacity is greater than or equal to [239 kg/s (527 lbm/s) at an inlet pressure of 7.48 Mpa gauge (1085 psig)].
26. The equipment qualification of the NBS components is addressed in Tier 1, Section 3.8.	See Tier 1 Section 3.8.	See Tier 1 Section 3.8.
27. The containment isolation portions of the NBS are addressed in Tier 1, Subsection 2.15.1.	See Tier 1 Subsection 2.15.1.	See Tier 1 Subsection 2.15.1.

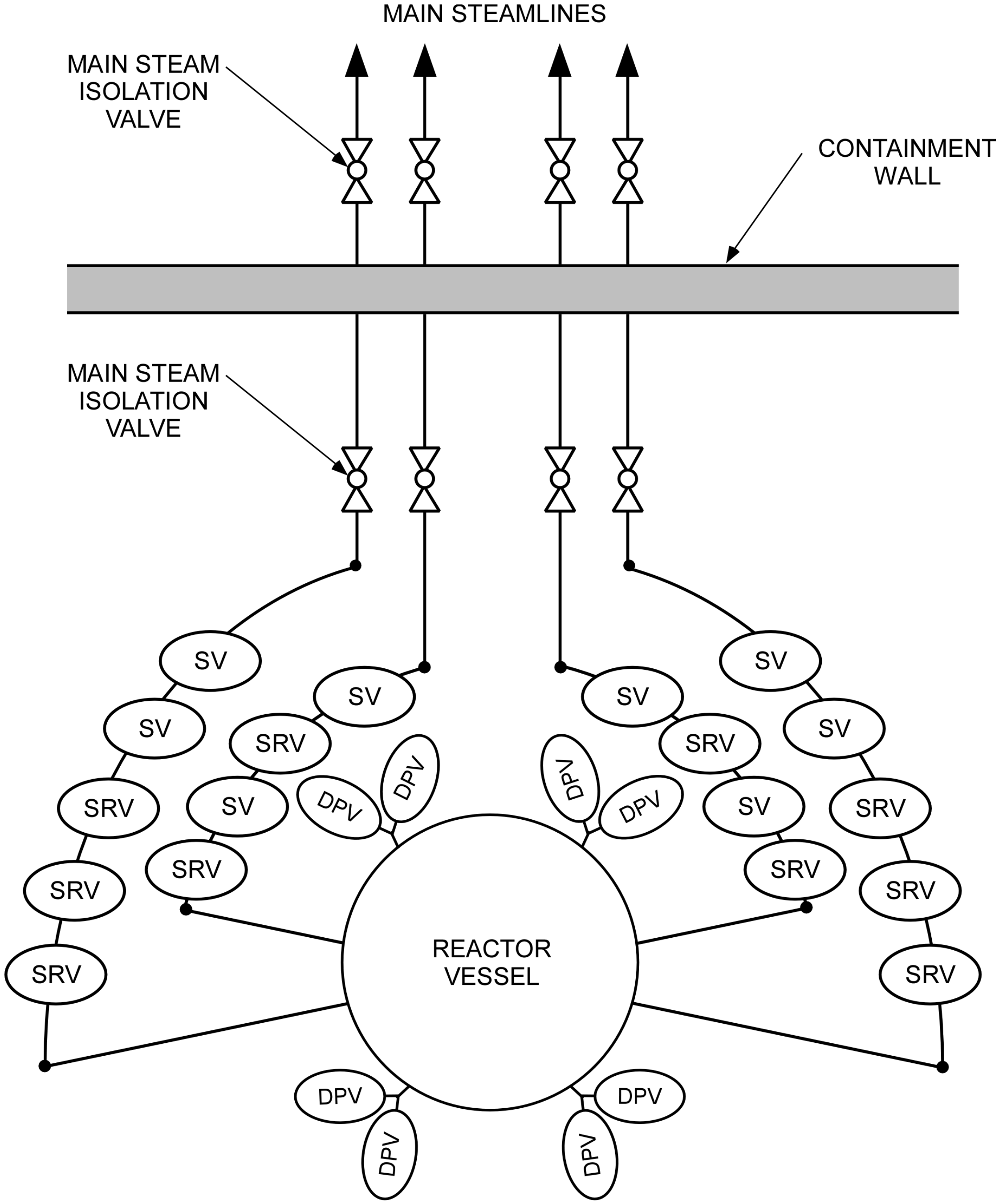


Figure 2.1.2-1. Safety Relief Valves, Depressurization Valves and Steamline Diagram

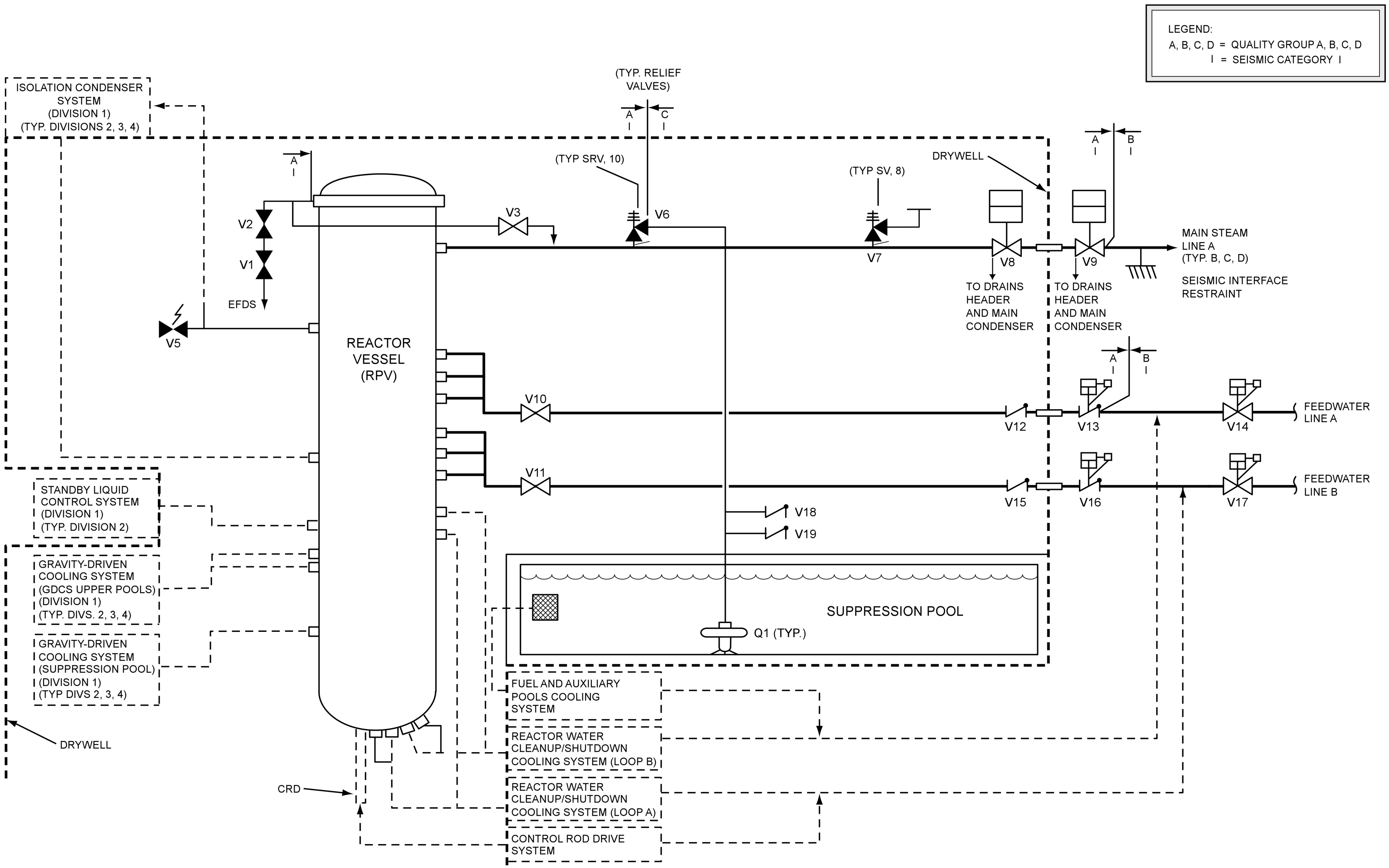


Figure 2.1.2-2. NBS Steamlines and Feedwater Lines

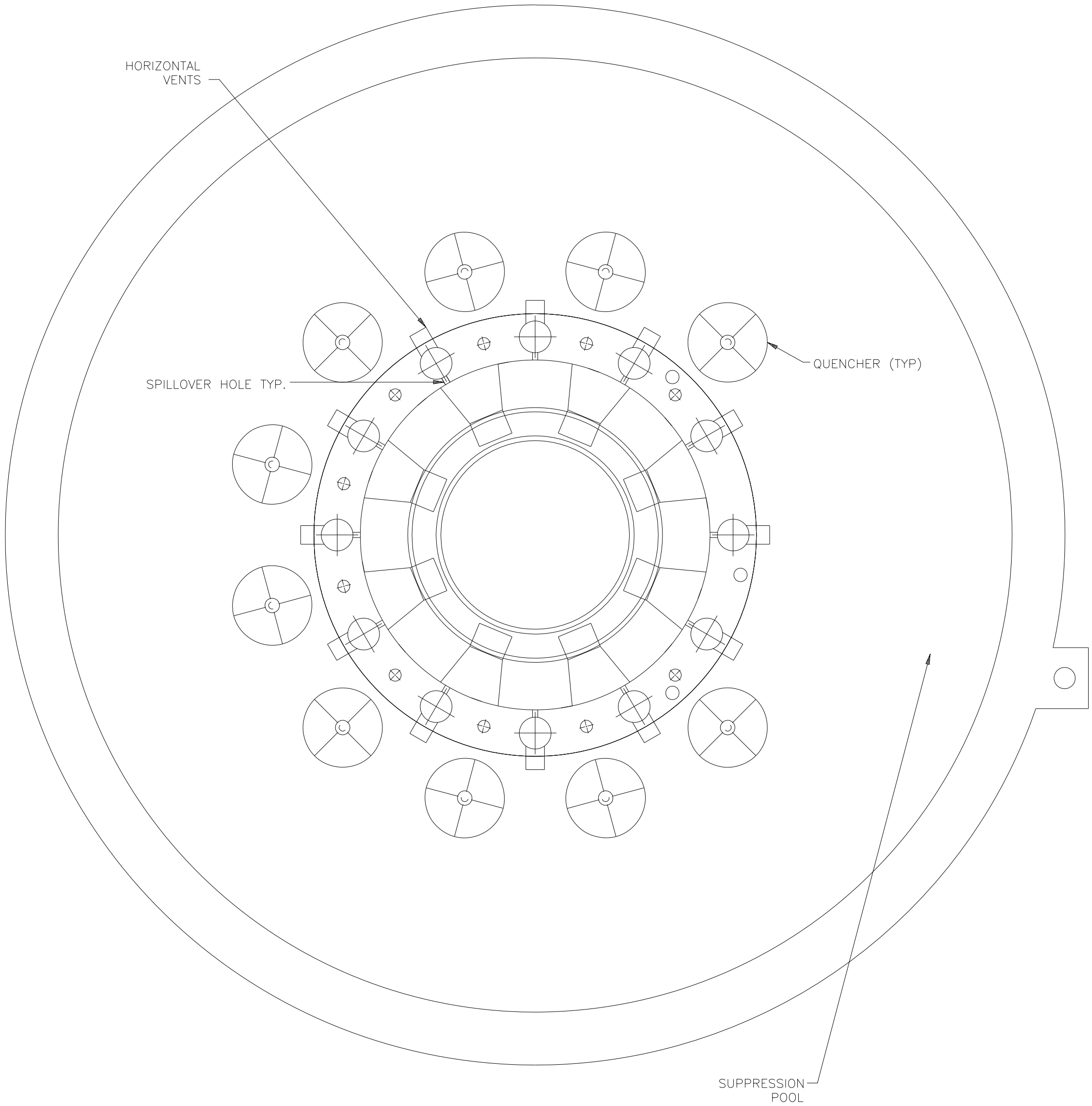


Figure 2.1.2-3. Safety-Relief Valve Discharge Line Quencher Arrangement

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) automatically controls and monitors, and provides manual control capability for, positioning of the control rods in the reactor by the Control Rod Drive System (CRDS).

Functional Arrangement

(1) RC&IS functional arrangement is defined in Table 2.2.1-1.

(2) RC&IS is divided into major functional groups as defined in Table 2.2.1-2.

Functional Requirements

(3) RC&IS automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.1-3.

(4) RC&IS rod block functions are defined in Table 2.2.1-4.

(5) RC&IS controls, interlocks, and bypasses are defined in Table 2.2.1-5.

(6) RC&IS minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.1-6 defines the inspections, tests, and/or analyses, together with associated acceptance criteria for the RC&IS.

Table 2.2.1-1

RC&IS Functional Arrangement

RC&IS is dual-redundant architecture divided into major functional groups as defined in Table 2.2.1-2. RC&IS Dedicated Operator Interface (DOI) is located in the MCR. RC&IS equipment is located in a mild environment rooms within the Reactor Building (RB) and Control Building (CB). RC&IS equipment is powered by separate, non-divisional AC power sources with at least one power source being a nonsafety-related uninterruptible power supply. RC&IS equipment provides the capability to perform the FMCRD-related surveillance tests, including periodic individual CRDS HCU scram performance testing. RC&IS is capable of continued operation when different subsystems of RC&IS are bypassed.

Table 2.2.1-2

RC&IS Major Functional Groups

Major Functional Group	Functions
RC&IS DOI	Provides control rod position, FMCRD status, RC&IS status, and CRDS HCU status information to the plant operator in MCR. Provides controls for performing normal rod movement functions, bypassing major RC&IS subsystems, performing CRD surveillance tests, and resetting RC&IS trips and abnormal status conditions.
Rod Action Control System (RACS)	Comprises the following subsystems: • Rod Action and Position Information (RAPI) • RAPI DOI that displays the same information that is available on the RC&IS DOI • RAPI Signal Interface Unit (SIU) • RWM • ATLM
RAPI	Performs manual, semi-automatic, and automatic rod movement commands. Performs rod blocks, as defined in Table 2.2.1-4, based upon internal RC&IS signals from either channel: • ATLM • RWM • RAPI SIU Performs rod blocks, as defined in Table 2.2.1-4, based upon external input signals: • Safety-related RPS Reactor Mode Switch (RMS) position. • Safety-related NMS SRNM. • Safety-related NMS APRM.

Table 2.2.1-2

RC&IS Major Functional Groups

Major Functional Group	Functions
RAPI (Continued)	• Nonsafety-related NMS MRBM. • Safety-related FMCRD CR separation switches. • Nonsafety-related refueling platform position. • Nonsafety-related refueling hoist load condition. Maintains a mirror copy of the N-DCIS reference rod pull sequence (RRPS) in memory. Enforces adherence to RRPS; deviation from the RRPS causes RAPI logic to issue the following: • Issue rod block as defined in Tables 2.2.1-3 and 2.2.1-4. • Switch to RC&IS manual mode; disable automatic and semi-automatic modes of operation. • Send alarm signal to MCR that RC&IS is in manual mode. Provides control rod position and FMCRD status information to the N-DCIS, the NMS, the RWM, and the ATLM. Performs the scram-follow function. Performs the SCRRI function. Sends a SCRRI signal to the DPS to initiate the SRI function. Performs ARI motor run-in function. Sends/receives rod movement commands, rod position, CRDS FMCRD status information, and RC&IS-related status information. Sends CRDS HCU purge water valve control signals Sends and receives CRDS HCU status signals. Provides capability to perform the following CRDS surveillance tests: • Scram Time Test, • Coupling Check Test, and • Double-Notch Test.

Table 2.2.1-2

RC&IS Major Functional Groups

Major Functional Group	Functions
RAPI SIU	Handles RAPI inter-channel communication between ATLM, RWM, and RAPI A and B channels and external communication with the nonsafety-related NMS MRBM.
RWM	Enforces absolute rod pattern restrictions, called the Ganged Withdrawal Sequence Restrictions (GWSR) when reactor power is below the low power setpoint (LPSP) and the RPS RMS is in either the STARTUP or RUN position. Supports shutdown margin testing
ATLM	Microprocessor-based subsystem of the RC&IS Enforces Operating Limit Minimum Critical Power Ratio (OLMCPR) Enforces the Operating Limit Minimum Linear Heat Generation Rate (OLMLHGR) Issues rod withdrawal block signals
Remote Communication Cabinets (RCCs)	Houses the redundant microprocessor-based communication system that interfaces with the RAPI, MCC, and RBCC.
Motor Controller Cabinets (MCCs)	Houses the CRDS FMCRD motor controllers (MC). Interfaces with RCC, RBCC and Emergency Rod Insertion Panel (ERIP)
Rod Brake Controller Cabinets (RBCCs)	Operates the CRDS FMCRD holding brakes Interfaces with RCC.
ERICP	Located in CB. Relay-hardware based, nonsafety-related control system that alternatively commands scram follow, ARI, and SCRRI.
ERIP	Interface with the MCC FMCRD MCs.
Scram Time Recording Panels (STRPs)	Monitors the CRDS FMCRD position switch status Automatically records and time tags CRDS FMCRD scram timing position switch status changes Transmits recorded scram timing data to the scram time recording and analysis panel (STRAP) Communicates with the RAPI

Table 2.2.1-2

RC&IS Major Functional Groups

Major Functional Group	Functions
STRAP	Performs scram timing performance analysis
RAPI Auxiliary Panels	Open CRDS HCU purge water valve. Monitor scram valve position. Monitor CRDS HCU accumulator water pressure. Monitor CRDS HCU accumulator water level. Send data to RAPI subsystem.

Table 2.2.1-3

RC&IS Automatic Functions, Initiators, and Associated Interfacing Systems

Function	Initiator	Interfacing System
Initiate Rod Block and Terminate Rod Withdrawal (See Table 2.2.1-4 for a complete list of rod blocks.)	ATLM Operating Limit Minimum Critical Power Ratio (MCPR) parameter greater than or equal to setpoint.	NMS
	ATLM Operating Limit Minimum Linear Heat Generation Rate (MLHGR) parameter greater than or equal to setpoint.	NMS
	SRNM period greater than or equal to setpoint.	NMS
	RWM function sequence error.	NMS
	Refueling platform over core and fuel on hoist.	The RB refueling machine
	Reactor Mode Switch (RMS) in SHUTDOWN position	RPS
	CRD charging water low pressure	CRDS
	CRD charging water low-pressure trip bypass	CRDS
	RWM function parameter greater than or equal to setpoint.	NMS
	Large deviation of CR positions from RRPS in selected gang.	-
	Any attempt to withdraw an additional rod beyond the original control rod pair.	-
	RAPI trouble	-
	RAPI Signal Interface Unit trouble	-

Table 2.2.1-4

RC&IS Rod Block Functions

Rod Block	Permissive Condition	Description
Rod separation detection	RMS: STARTUP or RUN	Rod withdrawal block only for those selected rod(s) for which the separation condition is detected and are not in the Inoperable Bypass condition.
RMS in SHUTDOWN position	RMS: SHUTDOWN	Rod withdrawal block for all control rods.
SRNM withdrawal block	RMS: SHUTDOWN, REFUEL, or STARTUP	Rod withdrawal block for all control rods.
APRM withdrawal block	None	Rod withdrawal block for all control rods.
CRD charging water low pressure	None	Rod withdrawal block for all control rods.
CRD charging water low-pressure trip bypass	None	Rod withdrawal block for all control rods.
RWM withdrawal block	Reactor power less than setpoint	Rod withdrawal block for all control rods.
RWM insert block	Reactor power less than setpoint	Rod insertion block for all control rods.
ATLM withdrawal block	Reactor power greater than setpoint	Rod withdrawal block for all control rods.
MRBM withdrawal block	Reactor power greater than setpoint	Rod withdrawal block for all control rods.
Gang large deviation	RC&IS Mode Switch: GANG:	Rod withdrawal block for all operable control rods of the selected gang upon detection of: – Large deviation of CR positions from RRPS in selected gang. – Any attempt to withdraw an additional rod beyond the original control rod pair.

Table 2.2.1-4

RC&IS Rod Block Functions

Rod Block	Permissive Condition	Description
Refuel mode withdrawal block	RMS: REFUEL, refueling platform over RPV, and fuel bundle on crane	Rod withdrawal block for all control rods.
Startup mode withdrawal block	RMS: STARTUP and refueling platform over RPV	Rod withdrawal block for all control rods.
RAPI trouble	RRPS active	Rod withdrawal block and rod insertion block for all control rods.
RAPI Signal Interface Unit trouble	None	Rod withdrawal block for all control rods when difference detected between any pair of input or output A and B channels.
Electrical group power abnormal	None	Rod withdrawal block and rod insertion block for all control rods.

Table 2.2.1-5

RC&IS Controls, Interlocks, and Bypasses

Function	Description
Control	Single / Ganged mode selection. Automatic / semi-automatic / manual mode selection. Normal / notch / continuous CR movement mode Insert / Withdraw. SCRRI/SRI manual initiation. ARI manual initiation (DPS)
Interlock	Single / Dual Rod Sequence Restriction Override (S/DRSRO) allows an operator to place up to two CR associated with the same HCU in S/DRSRO for scram time surveillance testing. Rod Inoperable Bypass condition allows up to [16] CR to be selected with the Reactor Mode Switch (RMS) in RUN position (RPS). Rod Inoperable Bypass condition allows up to [54] CR to be selected with the RMS in REFUEL position (RPS).
Bypass	Rod position detector channel bypass. S/DRSRO. Rod Inoperable Bypass selection. RCC (communication) channel bypasses. ATLM channel bypass. RWM channel bypass. RAPI channel bypass.

Table 2.2.1-6

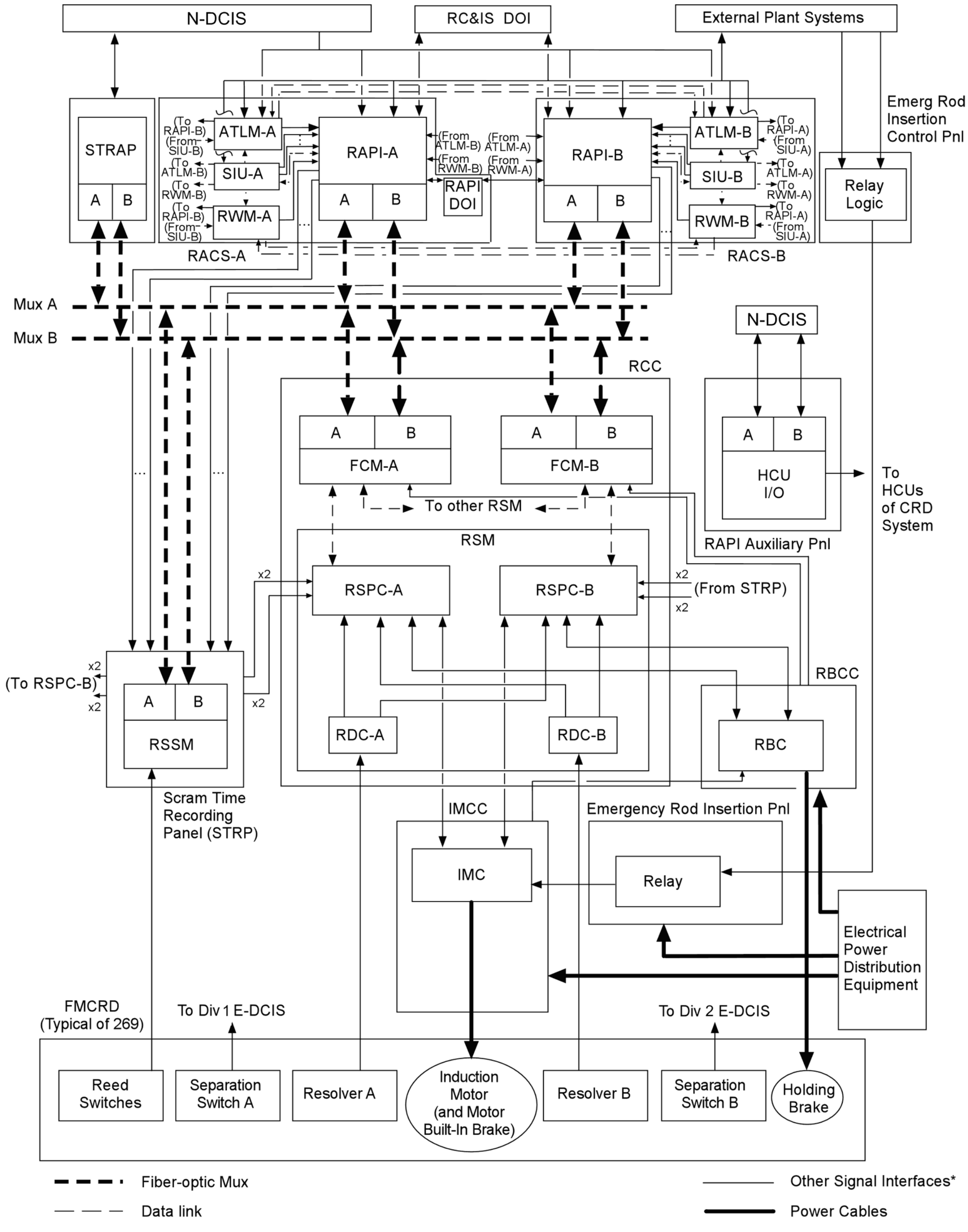
ITAAC For Rod Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. RC&IS functional arrangement is defined in Table 2.2.1-1.	Test(s) and inspection(s) of the as-built system will be performed.	Test and inspection report(s) document that the as-built system conforms with the functional arrangement defined in Table 2.2.1-1.
2. RC&IS is divided into major functional groups as defined in Table 2.2.1-2.	Test(s) and inspection(s) of the as-built system will be performed.	Test and inspection report(s) document that the as-built system functions as defined in Table 2.2.1-2.
3. RC&IS automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.1-3.	Test(s) and type test(s) will be performed on the as-built system using simulated signals.	Test and type test report(s) document the system is capable of performing the functions defined in Table 2.2.1-3.
4. RC&IS rod block functions and the permissive conditions under which the rod block is active are defined in Table 2.2.1-4.	Test(s) and type test(s) will be performed using simulated signals and manual actions to confirm that the rod withdrawal and insertion commands are blocked as defined in Table 2.2.1-4.	Test and type test report(s) document that the rod block functions defined in Table 2.2.1-4 are performed in response to simulated signals and manual actions.
5. RC&IS controls, interlocks, and bypasses are defined in Table 2.2.1-5.	Inspection(s), test(s) and type test(s) will be performed on the as-built system using simulated signals and manual actions.	Inspection, test and type test report(s) document that the system controls, interlocks, and bypasses exist, can be retrieved in the main control room, or are performed in response to simulated signals and manual actions as defined in Table 2.2.1-5.

Table 2.2.1-6

ITAAC For Rod Control and Information System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. RC&IS minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.	See Section 3.3.	See Section 3.3.



* 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, manually and automatically upon command from the RPS, DPS, and RC&IS, executes rapid control rod (CR) insertion (scram), performs fine CR positioning (reactivity control), detects CR separation (prevent rod drop accident), limits the rate of CR ejection due to a break in the CR pressure boundary (prevent fuel damage), and supplies high pressure makeup water to the reactor during events in which the feedwater system is unable to maintain reactor water level.

Functional Arrangement

(1) CRD system functional arrangement comprises three major functional groups: fine motion control rod drive (FMCRD), hydraulic control unit (HCU), and CRD hydraulic subsystem (CRDHS), as defined in Table 2.2.2-1 and shown in Figure 2.2.2-1.

Functional Requirements

(2) The components and piping defined in Table 2.2.2-5 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

(3) Pressure boundary welds in components and piping defined in Table 2.2.2-5 as ASME Code Section III meet ASME Code Section III requirements.

(4) The components and piping defined in Table 2.2.2-5 as ASME Code Section III retain their pressure boundary integrity at rated pressures.

(5) The Seismic Category I equipment defined in Table 2.2.2-1 can withstand seismic design basis loads without loss of structural integrity and safety function.

(6) The FMCRD is capable of positioning CR incrementally and continuously over its entire range.

(7) Valves defined in Table 2.2.2-5 and 2.2.2-6 as having an active safety-related function open, close, or both open and close under differential pressure, fluid flow, and temperature conditions.

(8) For the high pressure makeup mode of operation, the minimum flow supplied to the reactor is 3920 l/min (1036 gpm) with both CRD pumps operating and 1960 l/min (518 gpm) with one pump operating with reactor pressure less than or equal to 8.62 MpaG (1250 psig).

(9) CRD system automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.2-3.

(10) CRD system controls and interlocks are defined in Table 2.2.2-4.

(11) CRD system minimum inventory of alarms, displays, controls, and status indications in the main control room are addressed in Section 3.3.

(12) CRD maximum allowable scram times for vessel bottom pressures below 7.481 MPa gauge (1085 psig) are defined in Table 2.2.2-2.

(13) Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Tables 2.2.2-1, 2.2.2-6, and 2.2.2-7, is addressed in Subsection 2.2.15.

(14) The equipment qualification of CRDS components defined in Tables 2.2.2-1, 2.2.2-6, and 2.2.2-6, is addressed in Section 3.8.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.2-7 defines the inspections, tests, and/or analyses, together with associated acceptance criteria for the CRD system.

Table 2.2.2-1

CRDS Functional Arrangement

FMCRDs, including the equipment defined in Table 2.2.2-5 and 2.2.2-6, are safety-related, Seismic Category I.
FMCRDs are capable of rapid hydraulic insertion of the CRs during ATWS peak reactor pressure transient.
FMCRDs are capable of maintaining RCPB continuously at the RPV design pressure and briefly during the ATWS peak reactor pressure transient.
FMCRD has continuous CR position indication sensors that detect CR position based on motor rotation.
FMCRD has scram position indication switches that detect intermediate and scram completion CR positions.
FMCRD has a bayonet CR coupling mechanism that requires a minimum rotation to decouple.
FMCRD rotation, sufficient to decouple the CR, is precluded when fuel bundles are present.
FMCRD have spring-loaded latches in the hollow piston that engage slots in the guide tube to prevent rotation of the bayonet coupling except at predefined positions.
FMCRD has redundant safety-related rod separation switches that detect separation of the FMCRD from the CR.
FMCRD has a magnetic coupling that provides seal-less, leak-free operation of the CRD mechanism.
FMCRD has safety-related holding brakes that engage on loss of power.
FMCRD has safety-related scram inlet port check valves that close under reverse flow.
FMCRD has passive safety-related integral internal blowout support.
FMCRD hydraulic scram feature moves a CR pair (except for one single CR) to defined scram positions, starting from loss of signal to the scram solenoid pilot valves in the HCUs, using only the stored energy in the CRDS HCU scram accumulators, in time spans equal to or less than the times defined in Table 2.2.2-2.
HCUs are safety-related, Seismic Category I.
HCUs are located in four dedicated rooms in the Reactor Building (RB).
HCU scram charging water header pressure instrumentation is safety-related, Seismic Category I.
HCU scram pilot solenoid valves transfer open to vent on loss of power to both solenoids.

Table 2.2.2-1

CRDS Functional Arrangement

HCU air header dump valves transfer open to vent on loss of power.
HCU ARI solenoid valves are closed on loss of power and transfer open to vent when energized.
Each HCU contains a nitrogen-water accumulator charged to a sufficiently high pressure and with the necessary valves and components to fully insert two CRs.
HCUs provide a flow path for purge water to the associated FMCRDs during normal operation.
HCU scram accumulators are continuously monitored for water leakage by level instruments.
HCU has a test port to allow connection of temporary test equipment for the conduct of FMCRD ball check valve testing and drive friction testing.
CRDHS FMCRD purge water header, HCU charging header, and scram air header, are classified Seismic Category II.
Divisional safety-related power supplies power safety-related FMCRD and HCU equipment.

Table 2.2.2-2

CRD Maximum Allowable Scram Times for

Vessel Bottom Pressures Below 7.481 MPa gauge (1085 psig)

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

Table 2.2.2-3

CRD System Automatic Functions, Initiators, and Associated Interfacing Systems

Function	Initiator	Interfacing System
Hydraulic scram	RPS scram signal	RPS and DPS
Scram follow/ARI motor run-in	RPS scram signal / DPS scram signal	RPS and DPS
Provide make up water to RPV	RPV water level low (L2)	RPS
SCRRI (electric)	SCRRI signals	RC&IS
ARI (hydraulic)	DPS ARI signal	DPS
SRI (hydraulic)	DPS SRI signals	DPS

Table 2.2.2-4

CRD System Controls and Interlocks

Parameter	Description
Control	Manual start (CRD pumps)
Interlock	High-pressure makeup mode (RPV water level low (Level 2)) • The standby CRD pump is started. Both pumps operate in parallel to deliver the required makeup flow capacity to the reactor. • The two pump suction filter bypass valves are opened. • The charging water header isolation valve and purge water header isolation valve are closed. • The pump minimum flow bypass line isolation valve closes. • The flow control valves in the high-pressure makeup lines open to regulate the makeup water flow rate to the reactor. • The test valve in the high-pressure makeup line to the RWCU/SDC system opens if it is closed at the start of the event and the test valve in the return line to the CST closes if it is open at the start of the event. • The high-pressure makeup flow control valves close to stop flow to the reactor at high reactor water Level 8. • The pump minimum flow bypass line isolation valve opens and both pumps continue to operate in a low flow condition by directing their flow back to the CST through the pump minimum flow lines.

Table 2.2.2-4(Continued)

CRD System Controls and Interlocks

Parameter	Description
	• The control valves reopen and the pump minimum flow bypass isolation valve closes to restart high-pressure makeup flow if a subsequent Level 2 signal should occur. Normal operation mode (CRD common pump discharge line pressure low) • Start standby CRD pump. Normal operation mode (CRD pump inlet pressure low) • Trip running CRD pump after expiration of an adjustable time delay. Normal operation mode (pump lube oil pressure low) • Trip running CRD pumps and remove CRD pump start permissive condition. Normal operation mode (rod separation detection) • Send individual rod block initiate signal to RC&IS. Normal operation mode (scram charging header pressure low) • Send all rods block initiate signal to RC&IS. Normal operation mode (rod gang misalignment) • Send all rods in gang block initiate signal to RC&IS. High-pressure makeup mode (inboard FW maintenance valve closed) • Inhibit opening (hp makeup water) injection valves. High-pressure makeup mode (at least 2 GDCS pool levels low) • Trip CRD pumps. CRD common pump discharge line pressure low • Starts the standby pump.

Table 2.2.2-4(Continued)

CRD System Controls and Interlocks

Parameter	Description
	CRD common pump discharge line flow • Modulates purge water control valves. Injection flow • Modulates injection valves.

Table 2.2.2-5

Control Rod Drive System Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.2.2-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Functional Capability Required	MCR Alarms
FMCRD components required for scram	FMCRD	No	Yes	No	Active	-
FMCRD reactor coolant primary pressure boundary components	FMCRD	Yes	Yes	Yes	-	-
HCU components required for scram	HCU	No	Yes	No	Active	1. HCU gas pressure low 2. HCU accumulator leakage high
Scram inlet piping	-	No	Yes	No	-	-
Internal drive housing supports	-	-	Yes	No	-	-
FMCRD magnetic coupling	FMCRD	-	Yes	No	Passive	-
FMCRD ball check valves	FMCRD	-	Yes	No	Active	-
HCU charging water supply line check valve	HCU	Yes	Yes	No	Active	-
HCU purge water supply line check valve	HCU	Yes	Yes	No	Active	-

Table 2.2.2-6

Control Rod drive system Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.2.2-1	Control Q-DCIS/ DPS See Note 1	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Loss of Motive Power Position	Remotely Operated Valve
HCU scram solenoid pilot valves	SSPV	Y	Y	Y	Associated scram valve position status	Energize to apply air to HCU scram valve	Vent HCU scram valve	By RPS system logic
FMCRD brakes	FMCRD	Y	Y	Y	Y	Release brake	Apply brake	-
FMCRD reed switches	FMCRD	Y	N	Y	Y	-	-	-
HCU charging water header pressure transmitters	P Div 1-4	Y	Y	Y	MCR alarm	-	-	-
FMCRD separation switches	FMCRD	Y	Y	Y	MCR alarm	-	-	-

NOTE 1: See Tables 2.2.2-3, 2.2.2-4, and 2.2.2-5 for control functions and initiating conditions.

Table 2.2.2-7

ITAAC For Control Rod Drive System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. CRD system functional arrangement comprises three major functional groups: fine motion control rod drive (FMCRD), hydraulic control unit (HCU), and CRD hydraulic subsystem (CRDHS), as defined in Table 2.2.2-1 and shown in Figure 2.2.2-1.	Inspection(s), test(s), and type test(s) of the as-built system will be conducted.	Inspection report(s) document that the as-built CRD system conforms to the functional arrangement defined in Table 2.2.2-1 and as shown in Figure 2.2.2-1.
2. The components and piping defined in Table 2.2.2-5 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components and piping as documented in the ASME design reports.	Inspection report(s) document that the ASME Code Section III design reports exist for the as-built components and piping defined in Table2.2.2-5 as ASME Code Section III.
3. Pressure boundary welds in components and piping defined in Table 2.2.2-5 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	Inspection report(s) document that a report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4. The components and piping defined in Table 2.2.2-5 as ASME Code Section III retain their pressure boundary integrity at rated pressures.	A hydrostatic test will be conducted on those code components and piping of the Control Rod Drive System required to be hydrostatically tested by the ASME code.	Inspection report(s) document that the results of the hydrostatic test of the ASME Code components and piping of the CRD System conform with the requirements in the ASME Code Section III.

Table 2.2.2-7

ITAAC For Control Rod Drive System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The Seismic Category I equipment defined in Table 2.2.2-1, 2.2.2-5, and 2.2.2-6, can withstand seismic design basis loads without loss of structural integrity and safety function.	a) Type tests and/or analyses of Seismic Category I equipment will be performed.	a) Report(s) document(s) that the Seismic Category I equipment can withstand seismic design basis dynamic loads without loss of structural integrity and safety function.
	b) Inspections will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	b) The as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
	c) Inspections of the equipment defined in Table 2.2.1-1, 2.2.2-5, and 2.2.2-6, will be performed to verify that the equipment is housed in Seismic Category I structures.	c) The Seismic Category I equipment defined in Table 2.2.1-1, 2.2.2-5, and 2.2.2-6, is housed in a Seismic Category I structure.
6. The FMCRD is capable of positioning CR incrementally and continuously over its entire range.	Type test(s) will be performed of the motor run-in and withdrawal function on the FMCRD using a simulated CR.	Type test report(s) document that FMCRD is capable of positioning CR incrementally in minimum increments of [36.5] mm (1.44 in.) and continuously over its entire range at a speed of [28 ± 5] mm/sec.

Table 2.2.2-7

ITAAC For Control Rod Drive System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. Valves defined in Table 2.2.2-5 and 2.2.2-6 as having an active safety-related function open, close, or both open and close under differential pressure, fluid flow, and temperature conditions.	Tests of installed valves will be performed for opening, closing, or both opening and closing under system preoperational differential pressure, fluid flow, and temperature conditions.	Test report(s) document that, upon receipt of the actuating signal, each valve opens, closes, or both opens and closes, depending upon the valve's safety function.
8. For the high pressure makeup mode of operation, the minimum flow supplied to the reactor is [3920] l/min ([1036] gpm) with both CRD pumps operating and [1960] l/min ([518] gpm) with one pump operating with reactor pressure less than or equal to [8.62] MpaG ([1250] psig).	Test(s) of the high pressure makeup flow capacity of the CRD pumps will be conducted in a test facility.	Test report(s) document that the CRDHS delivers a minimum flow of [3920] l/min ([1036] gpm) with both CRD pumps operating and [1960] l/min ([518] gpm) with one CRD pump operating against a pressure less than or equal to [8.62] MPaG ([1250] psig).
9. CRD system automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.2-3.	Test(s) and type test(s) will be performed on the as-built system using simulated signals.	Test and type test report(s) document the system is capable of performing the functions defined in Table 2.2.2-3.
10. CRD system controls and interlocks are defined in Table 2.2.2-4.	Test(s) and type test(s) will be performed on the as-built system using simulated signals.	Test and type test report(s) document that the system controls and interlocks exist, can be retrieved in the main control room, or are performed in response to simulated signals and manual actions as defined in Table 2.2.2-4.
11. CRD system minimum inventory of alarms, displays, controls, and status indications in the main control room are addressed in Section 3.3.	See Section 3.3.	See Section 3.3.

Table 2.2.2-7

ITAAC For Control Rod Drive System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. CRD maximum allowable scram times for vessel bottom pressures below 7.481 MPa gauge (1085 psig) are defined in Table 2.2.2-2.	Test(s) will be performed of each CRD control rod pair scram function using simulated signals.	Test report(s) document that scram times for each control rod pair are less than or equal to the maximum allowable scram times defined in Table 2.2.2-2.
13. Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Tables 2.2.2-1, 2.2.2-5, and 2.2.2-6, is addressed in Subsection 2.2.15.	See Subsection 2.2.15.	See Subsection 2.2.15.
14. The equipment qualification of CRDS components defined in Tables 2.2.2-1, 2.2.2-5, and 2.2.2-6, is addressed in Section 3.8.	See Section 3.8.	See Section 3.8.

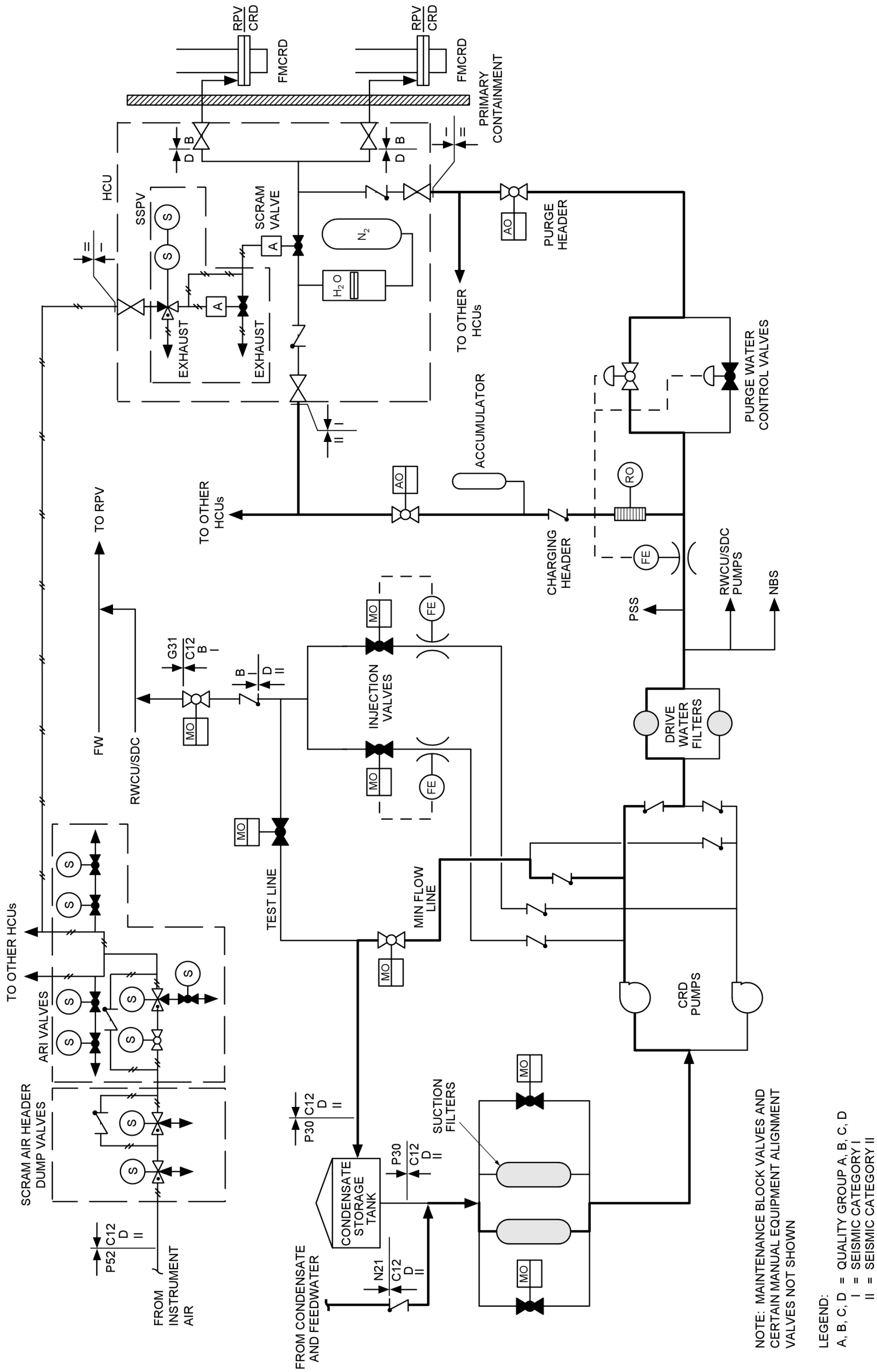


Figure 2.2.2-1. Control Rod Drive System

2.2.3 Feedwater Control System

Design Description

The Feedwater Control System (FWCS), automatically or manually, controls RPV water level by modulating the supply of feedwater flow to the RPV, the low flow control valve (LFCV), individual reactor feed pump ASD, or the RWCU/SDC system overboard control valve (OBCV).

Functional Arrangement

(1) FWCS functional arrangement is defined in Table 2.2.3-1.

Functional Requirements

(2) FWCS automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.3-2.

(3) FWCS controls are defined in Table 2.2.3-3.

(4) FWCS minimum inventory of alarms, displays, and status indications in the main control room are addressed in Section 3.3.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.3-4 defines the inspections, tests, and/or analyses, together with associated acceptance criteria for the FWCS.

Table 2.2.3-1

Feedwater Control Modes

FWCS is nonsafety-related.
FWCS is a triple-redundant, fault tolerant digital controller (FTDC)

Table 2.2.3-2

FWCS Automatic Functions, Initiators, and Associated Interfacing Systems

Functions	Initiators	Interfacing System
Perform FW runback	RPV water level high (Level 8)	NBS
Send signal to N-DCIS to initiate SCRRI / SRI function	FW temperature low.	RC&IS
Reduce speed of other FW pumps	FW flow high.	-
Start standby reactor feed pump.	Reactor feed pump trip.	-
Open the steam line condensate drain valves	Steam flow less than predefined value of rated flow.	-
Perform FW runback.	FW temperature low	-
Trip all FW pumps.	RPV water level high-high (Level 9)	-
Perform FW runback.	RPV water level high (Level 8)	-
Perform FW runback and close the LFCV and the RWCU/SDC overboard flow control valve.	ATWS trip signal	DICS

Table 2.2.3-3

FWCS Controls

Parameter	Description
Control	Manual speed control (reactor feed pump) Manual start/stop (reactor feed pump) Automatic / manual mode (reactor feed pump control) Manual control (No. 7 high pressure FW heater string bypass valve and isolation valves) Automatic Control Modes: Single element control: (enable at predefined value below rated reactor power) RPV water level: • Modulate either the low flow control valve (LFCV) or individual reactor feed pump ASD. • Modulate RWCU/SDC system overboard control valve (OBCV) Three element control: (enable during normal power operation) Three process variables generate master feedwater flow demand signal (for output to individual reactor feed pump loop trim controller): • Total steam flow • Total FW flow • RPV water level Reactor feed pump loop trim controller modulates individual reactor feed pump ASD: • Master FW flow demand signal • Individual reactor feed pump flow signals

Table 2.2.3-4

ITAAC For Feedwater Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The FWCS functional arrangement is defined in Table 2.2.3-1.	Inspections and tests will be performed on the FWCS functional arrangement using simulated signals and simulated actuators.	Inspection and test report(s) document(s) that FWCS functional arrangement is as defined in Table 2.2.3-1.
2. FWCS automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.3-2.	Test(s) and type test(s) will be performed on the as-built system using simulated signals.	Test and type test report(s) document the system performs the functions defined in Table 2.2.3-2.
3. FWCS controls are defined in Table 2.2.3-3.	Inspection(s), test(s) and type test(s) will be performed on the as-built system using simulated signals and manual actions.	Test and type test report(s) document that the system controls and interlocks exist, can be retrieved in the main control room, or are performed in response to simulated signals and manual actions as defined in Table 2.2.3-3.
4. FWCS minimum inventory of alarms, displays, and status indications in the main control room are addressed in Section 3.3.	See Section 3.3.	See Section 3.3.

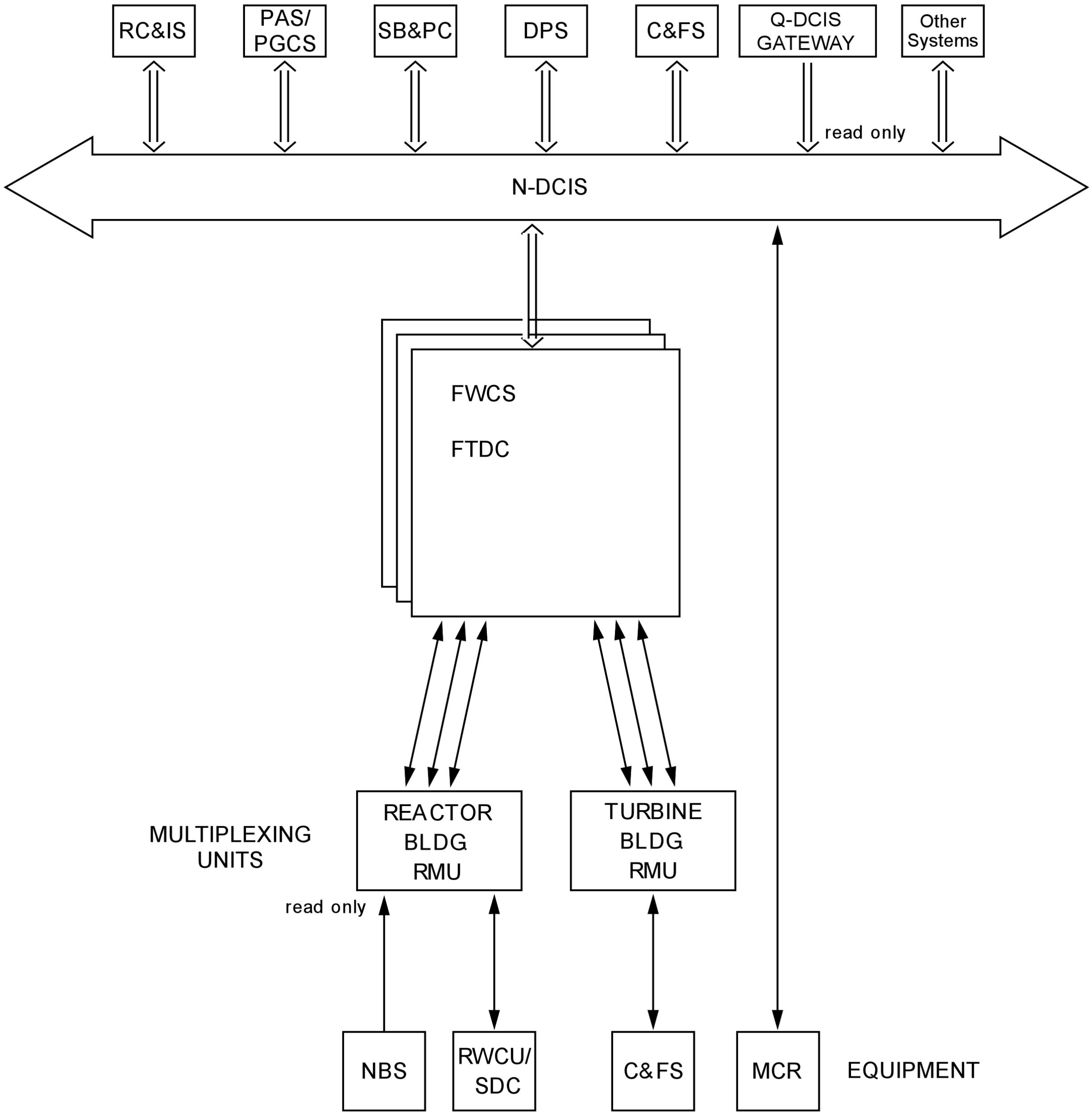


Figure 2.2.3-1. Feedwater Control System Logic Functional Diagram

2.2.4 Standby Liquid Control System

Design Description

The Standby Liquid Control (SLC) system is an alternative means to reduce core reactivity to ensure complete shutdown of the reactor core from the most reactive conditions at any time in core life, and provides makeup water to the RPV to mitigate the consequences of a Loss-of-Coolant-Accident (LOCA).

Functional Arrangement

(1) The SLC system functional arrangement is defined in Table 2.2.4-1.

Functional Requirements

(2) The SLC system automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.4-2.

(3) The SLC system controls and interlocks in the main control room are defined in Table 2.2.4-3.

(4) The SLC system minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.

(5) Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Table 2.2.4-1 is addressed in Subsection 2.2.15.

(6) The equipment qualification of SLC system components defined in Table 2.2.4-1 is addressed in Section 3.8.

(7) During an ATWS, the SLC system shall be capable of injecting borated water into the RPV at flowrates that assure rapid power reduction.

(8) The SLC system shall be capable of injecting borated water for use as makeup water to the RPV in response to a Loss-of-Coolant-Accident (LOCA).

(9) The redundant injection shut-off valves shown in Figure 2.2.4-1 as V1, V2, V3, and V4 are automatically closed by low accumulator level signals.

(10) ASME Code Section III

a. The components identified in Table 2.2.4-4 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

b. The piping identified in Table 2.2.4-4 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

(11) Pressure boundary welds

a. Pressure boundary welds in components identified in Table 2.2.4-4 as ASME Code Section III meet ASME Code Section III requirements.

b. Pressure boundary welds in piping identified in Table 2.2.4-4 as ASME Code Section III meet ASME Code Section III requirements.

(12) Pressure boundary integrity

a. The components identified in Table 2.2.4-4 as ASME Code Section III retain their pressure boundary integrity at under internal pressures that will be experienced during service.

b. The piping identified in Table 2.2.4-4 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

(13) The Seismic Category I equipment identified in Tables 2.2.4-4 and 2.2.4-5 can withstand seismic design basis loads without loss of safety function.

(14) Each of the components identified in Table 2.2.4-4 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

(15) Each of the SLC System divisions (or safety-related loads/components) identified in Tables 2.2.4-4 and 2.2.4-5 is powered from its respective safety-related division.

(16) In the SLC System, independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.

(17) Each mechanical train of the SLC System is physically separated from the other trains outside of the Containment.

(18) Re-positionable (not squib) valves designated in Table 2.2.4-4 as having an active safety-related function open, close, or both open and close under differential pressure, fluid flow, and temperature conditions.

(19) The pneumatically operated valve(s) designated in Table 2.2.4-4 fail in the mode listed if either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve(s) is lost.

(20) Check valves designated in Table 2.2.4-4 as having a safety-related function open, close, or both open and close under system pressure, fluid flow, and temperature conditions.

(21) The SLC System injection squib valve will open as designed.

(22) The equivalent natural boron concentration at cold shutdown conditions for the total solution injection volume is based on the liquid inventory in the RPV at the main steam line nozzle elevation plus the liquid inventory in the reactor shutdown cooling piping and equipment of the RWCU/SDC system.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.4-6 defines the inspections, tests, and/or analyses, together with associated acceptance criteria for the SLC system.

Table 2.2.4-1

SLC System Functional Arrangement

The SLC system is safety-related, Seismic Category I.
The SLC system comprises two 50% capacity trains with an accumulator tank for each train.
Each accumulator tank has an injectable liquid volume of at least 7.8 m^3 (2061 gal).
Each accumulator tank has a cover gas volume above the liquid of at least 14.8 m^3 (523 ft^3).
Each accumulator tank is capable of maintaining an initial nitrogen cover gas absolute pressure of least 14.82 MPa (2150 psia).
Each SLC train has redundant squib-type injection valves, installed in parallel, connected to a common injection line.
Each injection line is connected to manifolds within the RPV core region, which have injection nozzles in each quadrant.
Each accumulator has redundant level and pressure instrumentation.
Each accumulator has redundant shut-off valves, installed in series, which automatically close after injection.
Each accumulator has a pressure relief line and valve.
Each accumulator has vent piping and valves to permit depressurization of each accumulator.
Each accumulator has piping and valves used for initial and periodic solution gas (nitrogen) and charging.
Each SLC system train is powered by a separate safety-related power supply.
The SLC accumulators and piping upstream of the injection valves conforms with ASME Section III, Class NC.
The injection valves and piping downstream of the injection valves conforms with ASME Section III, Class NB.
Electronic equipment, including instrumentation, located in SLC tank room and SLC tank instrumentation room is qualified for a mild environment.

Table 2.2.4-2

SLC System Automatic Functions, Initiators, and Associated Interfacing Systems

Function	Initiator	Interfacing System
Open SLC injection valves	DPV Group 1 timer expired.	SSLC/ESF ADS
	RPV pressure high and Startup Range Neutron Monitor (SRNM), i.e., the SRNM ATWS permissive, exist for specified time delay period.	NBS and NMS
	SRNM ATWS permissive and RPV water level low (L2) exist for specified time delay period.	NBS and NMS
	SRNM ATWS permissive and Manual ARI/FMCRD run-in signals exist for specified time delay period.	NBS, NMS, RPS, and DPS
Close SLC accumulator shut-off valves	SLC accumulator level low following injection.	

Table 2.2.4-3

SLC System Controls and Interlocks

Parameter	Description
Control	Manual initiation of SLC injection valves (ATWS/SLC)
Interlock	ATWS trip signal (from ATWS/SLC trip signal to open SLC injection squib valves and send RWCU/SDC isolation signal to LD&IS)
	ECCS initiation signal (from SSLC/ESF to open SLC injection squib valves and send signal to LD&IS for RWCU/SDC isolation)
	ECCS initiation signal (from DPS to open SLC injection squib valves and send signal to LD&IS for RWCU/SDC isolation signal)

Table 2.2.4-4
SLC System Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.2.4-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
Train A Nitrogen Supply Line Check Valve	-	Y	Y	N	N	N	-	N
Train B Nitrogen Supply Line Check Valve	-	Y	Y	N	N	N	-	N
Train A Nitrogen Supply Isolation Valve	-	Y	Y	N	N	Y	Closed	N
Train B Nitrogen Supply Isolation Valve	-	Y	Y	N	N	Y	Closed	N
Train A Accumulator Relief Valve	-	Y	Y	N	N	N	-	N
Train B Accumulator Relief Valve	-	Y	Y	N	N	N	-	N
Train A Accumulator Vent Valve	-	Y	Y	N	N	Y	Closed	N
Train A Accumulator Vent Valve	-	Y	Y	N	N	Y	Closed	N
Train B Accumulator Vent Valve	-	Y	Y	N	N	Y	Closed	N

Table 2.2.4-4

SLC System Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.2.4-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
Train B Accumulator Vent Valve	-	Y	Y	N	N	Y	Closed	N
Train A Accumulator Solution Fill Isolation Valve	-	Y	Y	N	N	N	-	N
Train B Accumulator Solution Fill Isolation Valve	-	Y	Y	N	N	N	-	N
Train A Accumulator Isolation valve	-	Y	Y	N	N	N	-	N
Train A Accumulator Shut-off valve	V3	Y	Y	N	N	Y	Fail-as-is	Y
Train A Accumulator Shut-off valve	V4	Y	Y	N	N	Y	Fail-as-is	Y
Train B Accumulator Isolation valve	-	Y	Y	N	N	N	-	N
Train B Accumulator Shut-off valve	V1	Y	Y	N	N	Y	Fail-as-is	Y

Table 2.2.4-4

SLC System Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.2.4-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
Train B Accumulator Shut-off valve	V2	Y	Y	N	N	Y	Fail-as-is	Y
Train A Injection Squib Valve	-	Y	Y	Y	Y	Y	Fail-as-is	Y
Train A Injection Squib Valve	-	Y	Y	Y	Y	Y	Fail-as-is	Y
Train B Injection Squib Valve	-	Y	Y	Y	Y	Y	Fail-as-is	Y
Train B Injection Squib Valve	-	Y	Y	Y	Y	Y	Fail-as-is	Y
Train A Accumulator Injection Test/Vent Valve	-	Y	Y	Y	N	N	-	N
Train B Accumulator Injection Test/Vent Valve	-	Y	Y	Y	N	N	-	N
Train A Injection Check Valve	-	Y	Y	Y	Y	N	-	N
Train A Injection Check Valve	-	Y	Y	Y	Y	N	-	N

Table 2.2.4-4

SLC System Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.2.4-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
Train B Injection Check Valve	-	Y	Y	Y	Y	N	-	N
Train B Injection Check Valve	-	Y	Y	Y	Y	N	-	N
Train A Vessel Isolation Valve	-	Y	Y	Y	N	N	-	N
Train B Vessel Isolation Valve	-	Y	Y	Y	N	N	-	N
Train A Mixing Pump Suction Isolation Valve	-	Y	Y	N	N	N	-	N
Train B Mixing Pump Suction Isolation Valve	-	Y	Y	N	N	N	-	N
Main Mixing Pump Suction Isolation Valve	-	Y	Y	N	N	N	-	N
Main Mixing Pump Discharge Isolation Valve	-	Y	Y	N	N	N	-	N
Poison Solution Check Valve	-	Y	Y	N	N	N	-	N

Table 2.2.4-4

SLC System Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.2.4-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
Poison Solution Batch Mixing Isolation Valve	-	Y	Y	N	N	N	-	N

Table 2.2.4-5

SLC System Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.2.4-1	Control Q-DCIS / DPS See Note 1	Seismic Cat. I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated	Containment Isolation Valve Actuator
Train A Accumulator Pressure Sensor	PT	Y	Y	Y	Y	-	-	N
Train B Accumulator Pressure Sensor	PT	Y	Y	Y	Y	-	-	N
Train A Accumulator Level Sensor	LT	Y	Y	Y	Y	Isolate Shut-off Valves	-	N
Train B Accumulator Level Sensor	LT	Y	Y	Y	Y	Isolate Shut-off Valves	-	N
Train A Injection Squib Valve Initiator(s)	-	Y	Y	Y	Y	SLC Injection	Y	Y
Train B Injection Squib Valve Initiator(s)	-	Y	Y	Y	Y	SLC Injection	Y	Y

Table 2.2.4-5

SLC System Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.2.4-1	Control Q-DCIS / DPS See Note 1	Seismic Cat. I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated	Containment Isolation Valve Actuator
SLC Logic Controllers	-	Y	Y	Y	-	Isolate Accumulators on low level	-	N

NOTE 1: Squib valve initiators allow independent SLC injection from different safety-related divisions and/or DPS

Table 2.2.4-6

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the SLC system is as described in Table 2.2.4-1 and shown in Figure 2.2.4-1.	Inspection(s), test(s), and type test(s) of the as-built system will be performed.	Report(s) document(s) that the as-built system conforms to the functional arrangement described in Table 2.2.4-1 and shown in Figure 2.2.4-1.
2. The SLC system automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.4-2.	See Subsection 2.2.15	See Subsection 2.2.15
3. The SLC system controls and interlocks in the main control room are defined in Table 2.2.4-3.	See Subsection 2.2.15	See Subsection 2.2.15
4. The SLC system minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.	See Section 3.3.	See Section 3.3.
5. Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Table 2.2.4-1 is addressed in Subsection 2.2.15.	See Subsection 2.2.15.	See Subsection 2.2.15.
6. The equipment qualification of SLC system components defined in Table 2.2.4-1 is addressed in Section 3.8.	See Section 3.8.	See Section 3.8.

Table 2.2.4-6

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. During an ATWS, the SLC system shall be capable of injecting borated water into the RPV at flowrates that assure rapid power reduction.	Tests are conducted to measure injection time of the as-built SLC system by injecting demineralized water from both accumulators into the open RPV. The initial differential pressure ([6.21] MPa) between the accumulators and the RPV are set to that expected at the beginning of an ATWS by adjusting the accumulator pressures. Analyses are performed to correlate test results to as-built SLC system performance during an actual ATWS.	Test and analysis reports exist and conclude that during an ATWS the as-built SLC system (both accumulators) injects borated water into the RPV within the following time frames: • The first 5.4 m^3 of solution injects in ≤ [196] seconds. • The first and second 5.4 m^3 of solution injects in ≤ [519] seconds.
8. The SLC system shall be capable of injecting borated water for use as makeup water to the RPV in response to a Loss-of-Coolant-Accident (LOCA).	Tests are conducted with the as-built SLC system to measure the total volume of demineralized water injected from both accumulators into the open RPV. These tests utilize the continuation of the tests conducted in ITAAC #3. Analyses are performed to correlate test results to as-built SLC system performance during an actual LOCA.	Test and analysis reports exist and conclude that the as-built SLC system (both accumulators) injects a total volume of ≥15.6 m^3 of borated water in response to a LOCA.
9. The redundant injection shut-off valves shown in Figure 2.2.4-1 as V1, V2, V3, and V4 are automatically closed by low accumulator level signals.	Test(s) will be performed using a simulated low accumulator level signal to close the injection shut-off valves V1, V2, V3, and V4.	Test report(s) document that the as-built injection shut-off valves identified in Figure 2.2.4-1 as V1, V2, V3, and V4 close upon receipt of a simulated low accumulator level signal

Table 2.2.4-6

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. ASME Code Section III a. The components identified in Table 2.2.4-4 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. b. The piping identified in Table 2.2.4-4 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	a. Inspection will be conducted of the as-built components as documented in the ASME design reports. b. Inspection will be conducted of the as-built components as documented in the ASME design reports.	a. Report(s) document that the ASME Code Section III design reports exist for the as-built components identified in Table 2.2.4-4 as ASME Code Section III. b. Report(s) document that the ASME code Section III design reports exist for the as-built piping identified in Table 2.2.4-4 as ASME Code Section III.
11. Pressure boundary welds a. Pressure boundary welds in components identified in Table 2.2.4-4 as ASME Code Section III meet ASME Code Section III requirements. b. Pressure boundary welds in piping identified in Table 2.2.4-4 as ASME Code Section III meet ASME Code Section III requirements.	a. Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III. b. Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	a. Report(s) document that a report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds. b. Report(s) document that a report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.

Table 2.2.4-6

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. Pressure boundary integrity		
a. The components identified in Table 2.2.4-4 as ASME Code Section III retain their pressure boundary integrity at under internal pressures that will be experienced during service.	a. A hydrostatic test will be conducted on those code components of the SLC System required to be hydrostatically tested by the ASME code.	a. Report(s) document that the results of the hydrostatic test of the ASME Code components of the SLC System conform to the requirements in the ASME Code, Section III.
b. The piping identified in Table 2.2.4-4 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	b. A hydrostatic test will be conducted on those code components of the System required to be hydrostatically tested by the ASME code.	b. Report(s) document that the results of the hydrostatic test of the ASME Code components of the System conform to the requirements in the ASME Code, Section III.

Table 2.2.4-6

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. The seismic Category I equipment identified in Tables 2.2.4-4 and 2.2.4-5 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Tables 2.2.4–4 and 2.2.4–5 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	Report(s) document that: i) The seismic Category I equipment identified in Tables 2.2.4–4 and 2.2.4–5 is located on a seismic structure. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
14. Each of the components identified in Table 2.2.4-4 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	Report(s) document that a report exists and concludes that each of the as-built lines identified in Table 2.2.4–4 for which functional capability is required meets the requirements for functional capability.

Table 2.2.4-6

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
15. Each of the SLC System divisions (or safety-related loads/components) identified in Tables 2.2.4-4 and 2.2.4-5 is powered from its respective safety-related division.	Testing will be performed on the SLC System by providing a test signal in only one safety-related division at a time.	Report(s) document that a test signal exists in the safety-related division (or at the equipment identified in Table 2.2.4–4 powered from the safety-related division) under test in the SLC System.
16. In the SLC System, independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	i) Tests will be performed on the SLC System by providing a test signal in only one safety-related division at a time. ii) Inspection of the as-installed safety-related divisions in the SLC System will be performed.	Report(s) document that i) The test signal exists only in the safety-related division under test in the System. ii) In the SLC System, 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.
17. Each mechanical train of the SLC System is physically separated from the other trains outside of the Containment.	Inspections of the as-built SLC System will be performed.	Report(s) document that each mechanical train of the SLC System is physically separated from other mechanical trains of the system by structural and/or fire barriers outside of the Containment.

Table 2.2.4-6

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
18. Re-positionable (not squib) valves designated in Table 2.2.4-4 as having an active safety-related function open, close, or both open and close under differential pressure, fluid flow, and temperature conditions.	Tests of installed valves will be performed for opening, closing, or both opening and closing under system preoperational differential pressure, fluid flow, and temperature conditions.	Report(s) document that, upon receipt of the actuating signal, each valve opens, closes, or both opens and closes, depending upon the valve's safety function.
19. The pneumatically operated valve(s) designated in Table 2.2.4-4 fail in the mode listed if either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve(s) is lost.	Tests will be conducted on the as-built valve(s).	Report(s) document that the pneumatically operated valve(s) identified in Table 2.2.4–4 fail in the listed mode when either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve(s) is lost.
20. Check valves designated in Table 2.2.4-4 as having a safety-related function open, close, or both open and close under system pressure, fluid flow, and temperature conditions	Tests of installed valves for opening, closing, or both opening and closing, will be conducted under system preoperational pressure, fluid flow, and temperature conditions.	Report(s) document that, based on the direction of the differential pressure across the valve, each CV opens, closes, or both opens and closes, depending upon the valve's safety functions.
21. The SLC System injection squib valve will open as designed.	A vendor type test will be performed on a squib valve to open as designed.	Records of vendor type test will conclude SLC injection squib valves used in the injection and equalization will open as designed.

Table 2.2.4-6

ITAAC For The Standby Liquid Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
22. The equivalent natural boron concentration at cold shutdown conditions for the total solution injection volume is based on the liquid inventory in the RPV at the main steam line nozzle elevation plus the liquid inventory in the reactor shutdown cooling piping and equipment of the RWCU/SDC system.	An analysis of the as-built system will be performed to determine the equivalent natural boron concentration at cold shutdown conditions for the total solution injection volume.	The equivalent natural boron concentration at cold shutdown conditions for the total solution injection volume is > [1100 ppm].

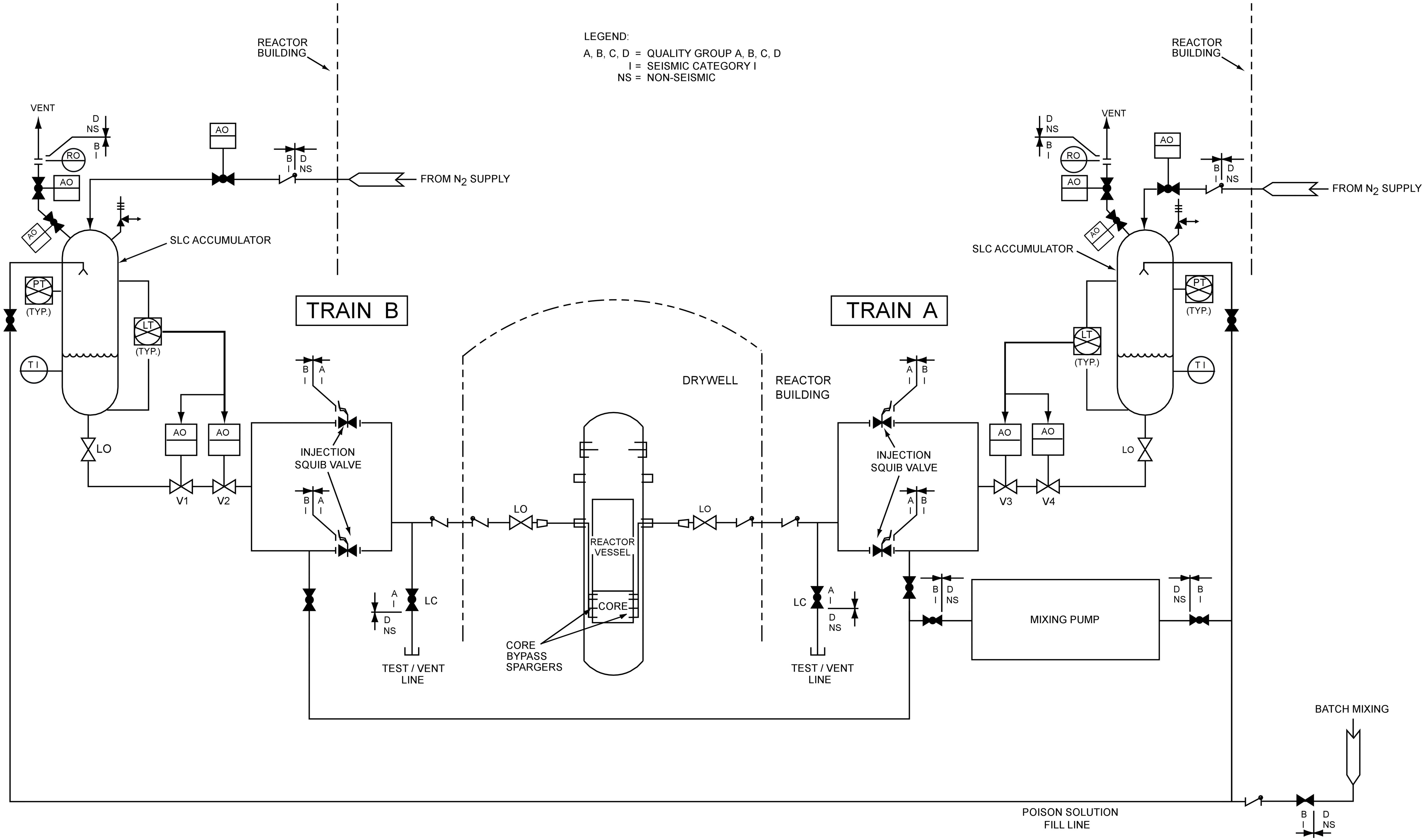


Figure 2.2.4-1. Standby Liquid Control System Simplified Diagram

2.2.5 Neutron Monitoring System

Design Description

Design Description

The Neutron Monitoring System (NMS) monitors thermal neutron flux and supports the Reactor Protection System (RPS).

Functional Arrangement

(1) NMS functional arrangement is defined in Table 2.2.5-1.

Functional Requirements

(2) NMS automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.5-2.

(3) NMS controls, interlocks, and bypasses are defined in Table 2.2.5-3.

(4) NMS minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.

(5) Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Table 2.2.5-1 is addressed in Subsection 2.2.15.

(6) The equipment qualification of NMS components defined in Table 2.2.5-1 is addressed in Section 3.8.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.5-4 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria for the NMS.

Table 2.2.5-1

NMS Functional Arrangement

NMS comprises the safety-related startup range neutron monitor (SRNM) subsystem and the power range neutron monitor (PRNM) subsystem; and the nonsafety-related automatic fixed in-core probe (AFIP) subsystem and multi-channel rod block monitor (MRBM) subsystem. NMS SRNM and PRNM subsystems are safety-related. NMS is a four division, redundant, microprocessor-based, non-volatile memory system. NMS divisions fail-safe to a trip condition on critical hardware failure, power failure, or loss of communication failure. NMS controllers are located in mild environments in divisionally separate rooms in the Control Building (CB) and Reactor Building (RB). NMS logic is designed to provide a trip initiation by requiring coincident trip of at least two divisions to cause the trip output when non-coincident logic is not imposed. Each NMS division is powered by its divisional safety-related UPS power supply. The PRNM subsystem comprises the local power range monitors (LPRM), the average power range monitors (APRM), and the oscillating power range monitors (OPRM). SRNM trip signal logic is interlocked with Coincident/Non-coincident switch and the Reactor Mode Switch. The SRNM subsystem has 12 SRNM channels, each channel having one fixed in-core regenerative fission chamber sensor. The SRNM subsystem monitors neutron flux from the source range to 15% of the reactor rated power. The LPRM detector assemblies, SRNM detector assemblies, wiring, cables, and connector are located in a harsh environment within the lower drywell in the RB. LPRM provides signals that are proportional to the local neutron flux. LPRM subsystem comprises 64 assemblies, divided into four divisions, distributed uniformly throughout the core, each assembly having four uniformly spaced fixed in-core fission chamber detectors and seven AFIP gamma thermometer sensors. The LPRM detector assemblies have a design pressure of 8.62 MPa g (1250 psig). The LPRM subsystems monitor neutron flux from 1 % to 125 % of reactor rated power.
Each APRM division generates the PRNM trip signals for the associated RPS division based on LPRM signals, APRM calculations, OPRM calculations.

Table 2.2.5-1

NMS Functional Arrangement

OPRM provides neutron flux oscillation trip signals for the APRM trip signal.

Table 2.2.5-2

NMS Functions, Initiators, and Associated Interfacing Systems

Function	Initiator	Interfacing System
SRNM Trip	SRNM short period	RC&IS, DPS, RPS
	SRNM upscale	RC&IS, DPS, RPS
	SRNM inoperable	RC&IS, DPS, RPS
	SRNM Downscale	RC&IS, DPS, RPS
	SRNM intermediate upscale flux (non-coincident mode)	RPS
	SRNM non-coincident upscale [A non-coincident SRNM trip with Reactor Mode Switch in SHUTDOWN, REFUEL, or STARTUP, position, the NMS Coincident/Non-coincident switch is in the NON-COINCIDENT position, and a single SRNM exceeds count setpoint]	RC&IS, DPS, RPS
PRNM Trip	APRM upscale flux	RC&IS, DPS, RPS, SSLC/ESF
	APRM inoperative	RC&IS, DPS, RPS
	APRM downscale (only in RUN)	RC&IS, DPS, RPS
	APRM upscale simulated reactor thermal power	RC&IS, DPS, RPS
	OPRM oscillation detection	DPS, RPS

Table 2.2.5-3

NMS Controls, Interlocks, and Bypasses

MCR Parameter	Description
Control	APRM Channel Bypass Control (one for each division) (hardware).
	SRNM Channel Bypass Controls (one for each bypass group) (hardware).
	MRBM Main Channel Bypass
	Coincident/Non-coincident switch.
Interlock	APRM ATWS Permissive (for ATWS ADS inhibit.)
	Reactor Mode Switch (RPS)
	SRNM ATWS Permissive (ATWS/SLC)
	APRM Signal (RC&IS)
	SRNM Signal (RC&IS)
Bypass	MRBM Main Channel Bypass (one for each division)
	APRM Channel Bypass Control (one for each division)
	SRNM Channel Bypass Controls (one for each bypass group)

Table 2.2.5-3

NMS Controls, Interlocks, and Bypasses

MCR Parameter	Description
Control	APRM Channel Bypass Control (one for each division) (hardware)
	SRNM Channel Bypass Controls (one for each bypass group) (hardware)
	MRBM Main Channel Bypass
	Coincident/Non-coincident switch
Interlock	APRM ATWS Permissive (for ADS inhibit)
	Reactor Mode Switch (RPS)
	SRNM ATWS Permissive (ATWS/SLC)
	APRM Signal (RC&IS)
	SRNM Control Rod Withdrawal Permissive (RC&IS)
	MRBM Rod Block
Bypass	MRBM Main Channel Bypass (one for each division)
	APRM Channel Bypass Control (one for each division)
	SRNM Channel Bypass Controls (one for each bypass group)

Table 2.2.5-4

ITAAC For The Neutron Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. NMS functional arrangement is defined in Table 2.2.5-1.	Inspection(s), test(s), and/or type test(s) will be performed on the as-built configuration as described in Table 2.2.5-1.	Inspection, test, and/or type test report(s) document(s) that the system conforms to the functional arrangement as described in Table 2.2.5-1.
2. NMS automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.5-2.	See Subsection 2.2.15.	See Subsection 2.2.15.
3. NMS controls, interlocks, and bypasses are defined in Table 2.2.5-3.	See Subsection 2.2.15.	See Subsection 2.2.15.
4. The SLC system minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.	See Section 3.3.	See Section 3.3.
5. Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Table 2.2.5-1 is addressed in Subsection 2.2.15.	See Subsection 2.2.15.	See Subsection 2.2.15.
6. The equipment qualification of NMS components defined in Table 2.2.5-1 is addressed in Section 3.8.	See Section 3.8.	See Section 3.8.

2.2.6 Remote Shutdown System

Design Description

The Remote Shutdown System (RSS) provides remote manual control of the systems necessary to: (a) perform a prompt shutdown (scram) of the reactor, (b) perform safe (hot) shutdown of the reactor after a scram, (c) perform subsequent cold shutdown of the reactor, and (d) monitor the reactor to ensure safe conditions are maintained during and following a reactor shutdown.

Functional Arrangement

(1) RSS functional arrangement is described in Subsection 2.2.6 and defined in Table 2.2.6-1.

Functional Requirements

(2) RSS controls are defined in Table 2.2.6-2.

(3) RSS minimum inventory of alarms, displays, controls, and status indications is addressed in Section 3.3.

(4) Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Table 2.2.6-1 is addressed in Subsection 2.2.15.

(5) The equipment qualification of RSS components defined in Table 2.2.6-1 is addressed in Section 3.8.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.6-3 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria for the RSS.

Table 2.2.6-1

RSS Functional arrangement

RSS is safety-related, Seismic Category I
RSS has two redundant, independent, panels
RSS panels are located in two separate rooms in different divisional quadrants of the Reactor Building
RSS panels have a safety-related Division 1 visual display unit (VDU), a safety-related Division 2 VDU, and a nonsafety-related VDU
Safety-related systems in each RSS panel receive power from divisionally separate safety-related uninterruptible power supplies
Nonsafety-related systems in each RSS panel receive power from nonsafety-related power supplies

Table 2.2.6-2
RSS Controls

Division 1 Manual Scram Control
Division 2 Manual Scram Control
Division 1 Manual MSIV Isolation Control
Division 2 Manual MSIV Isolation Control
Safety-related Division 1 VDU displays
Safety-related Division 2 VDU displays
Nonsafety-related VDU displays

Table 2.2.6-3

ITAAC For The Remote Shutdown System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. RSS functional arrangement is described in Subsection 2.2.6 and defined in Table 2.2.6-1.	Inspection(s) and test(s) will be performed to confirm that the as-built panels are configured as described in Subsection 2.2.6 and defined in Table 2.2.6-1.	Test report(s) document(s) that the as-built panels are configured as described in Subsection 2.2.6 and defined Table 2.2.6-1.
2. RSS controls are defined in Table 2.2.6-2.	Test(s) and type test(s) will be performed on the controls defined in Table 2.2.6-2.	Test report(s) document(s) that the RSS panels are capable of issuing control signals from the controls defined in Table 2.2.6-2.
3. RSS minimum inventory of alarms, displays, controls, and status indications is addressed in Section 3.3.	See Section 3.3	See Section 3.3.
4. Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Table 2.2.6-1 is addressed in Subsection 2.2.15.	See Subsection 2.2.15.	See Subsection 2.2.15.
5. The equipment qualification of DICS components defined in Table 2.2.6-1 is addressed in Section 3.8.	See Section 3.8.	See Section 3.8.

2.2.7 Reactor Protection System

Design Description

The Reactor Protection System (RPS) initiates a reactor trip (scram) automatically whenever selected plant variables exceed preset limits or by manual operator action.

Functional Arrangement

(1) RPS functional arrangement is defined in Table 2.2.7-1.

Functional Requirements

(2) RPS automatic trip initiators and associated interfacing systems are defined in Table 2.2.7-2.

(3) RPS controls, interlocks (system interfaces), and bypasses are defined in Table 2.2.7-3.

(4) Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components is addressed in Subsection 2.2.15.

(5) RPS minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.

(6) The equipment qualification of RPS components is addressed in Section 3.8.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.7-3 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

RPS Functional Arrangement

The RPS comprises four redundant safety-related, Seismic Category I, divisions of sensor channels, trip logics and trip actuators.
The RPS comprises two divisions of manual scram controls and scram logic circuitry.
RPS logic is designed to provide a trip initiation by requiring a coincident trip of at least two divisions to cause the trip output.
RPS trip actuator load drivers interrupt circuit power to scram pilot solenoids and scram air header dump valves.
RPS is fail-safe such that on loss of redundant divisional electrical power supplies the load drivers of that division change to the tripped state.
Redundant safety-related power supplies are provided for each division.

Table 2.2.7-2

RPS Automatic Functions, Initiators, and Associated Interfacing Systems

Function	Initiator	Interfacing System
Reactor scram	NMS PRNM trip condition	NMS
	NMS SRNM trip condition	NMS
	CRD charging header pressure low	CRDS
	Turbine stop valve closed position	-
	Turbine control valve control oil pressure low	-
	Condenser pressure high	-
	Power Generation Bus Loss	-
	MSIV closed position	NBS
	Reactor Pressure high	NBS
	RPV reactor level low (Level 3)	NBS
	RPV reactor level low (Level 8)	NBS
	Drywell pressure high	CMS
	Suppression pool average temperature high	CMS

Table 2.2.7-3

RPS Controls, Interlocks (System Interfaces), and Bypasses

Parameter	Description
Control	Manual divisional trip switches
	Manual scram trip switches
	Reactor Mode Switch
	Divisional actuator trip manual switches
	RPS trip reset manual switches
	RPS scram test switch (to RC&IS)
Interlock (System Interface)	RPS full scram condition (to RC&IS, CRDS)
	Turbine bypass valves open position indication
	APRM Simulated Thermal Power (to NMS)
	Reactor Mode Switch positions: -RUN (to NMS, ICS, PAS, LD&IS) -STARTUP (to PAS, NMS) -SHUTDOWN (to CRDS) -REFUEL (to CRDS, PAS, NMS)[DW381]
	Reactor Mode Switch in the SHUTDOWN position automatic bypass after a time delay
	Drywell pressure signal (to LD&IS)
	CRD charging header pressure signal (to RC&IS)
	Loss of Power Generation Bus (Loss of Feedwater Flow) signal (to ICS)
	MSIV closure bypass (to LD&IS)
Bypass	Special MSIV operational bypass switches
	Reactor Mode Switch in Shutdown scram manual bypass switches
	CRD HCU accumulator charging header pressure trip manual bypass switches (to RC&IS)
	MSIV closure trip signals manual bypass switches (to LD&IS)
	RPS TLU output manual divisional bypass switches
	Division of sensors channel inputs to each RPS division manual bypass switches

Table 2.2.7-4

ITAAC For The Reactor Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement is as described in Table 2.2.7-1.	Inspection(s), test(s), and/or type test(s) will be performed on the as-built configuration as described in Table 2.2.7-1.	Inspection, test, and/or type test report(s) document(s) that the system conforms to the functional arrangement as described in Table 2.2.7-1.
2. RPS automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.7-2.	Test(s) and type test(s) will be performed on the as-built system using simulated signals.	Test and type test report(s) document the system is capable of performing the functions defined in Table 2.2.7-2.
3. RPS controls, interlocks (system interfaces), and bypasses are defined in Table 2.2.7-3.	Test(s) and type test(s) will be performed on the as-built system using simulated signals.	Test and type test report(s) document that the system controls and interlocks exist, can be retrieved in the main control room, or are performed in response to simulated signals and manual actions as defined in Table 2.2.7-3.
4. Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components is addressed in Subsection 2.2.15.	See Subsection 2.2.15.	See Subsection 2.2.15.
5. RPS minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.	See Section 3.3.	See Section 3.3.
6. The equipment qualification of RPS components is addressed in Section 3.8.	See Section 3.8.	See Section 3.8.

2.2.8 Plant Automation System

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 Generator Control System (TGCS), which controls the turbine control valves.

Functional Arrangement

(1) SB&PC System functional arrangement is defined in Table 2.2.9-1.

Functional Requirements

(2) SB&PC System functions and initiating conditions are defined in Table 2.2.9-2.

(3) SB&PC System minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.9-3 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

SB&PC System Functional Arrangement

SB&PC System is a triple-redundant, fault-tolerant, digital controller.
SB&PC System fault-tolerant digital controllers are located in a mild environment in the Control Building.
SB&PC System has three redundant nonsafety-related AC uninterruptible power supplies designed so that loss of one power supply or incoming power source does not affect SB&PC system functional operation.
SB&PC System interfaces with Nuclear Boiler System receiving reactor steam dome pressure signals.
SB&PC System interfaces with the TGCS receiving (1) turbine power load unbalance signal, (2) turbine trip signal, and (3) turbine steam flow demand signal.
SB&PC System interfaces with main condenser receiving main condenser pressure signal.

Table 2.2.9-2

SB&PC System Functions and Initiating Conditions

Function	Initiating Condition
Close TBV	Main condenser pressure high
Modulate TBV	SB&PC System normal pressure control function

Table 2.2.9-3

ITAAC For The Steam Bypass and Pressure Control System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the SB&PC System is as defined in Table 2.2.9-1.	Inspections of the as-built system will be conducted.	Inspection reports(s) document that the as-built SB&PC system conforms to the functional arrangement as defined in Table 2.2.9-1.
2. SB&PC system functions and initiating conditions are as defined in Table 2.2.9-2.	Tests will be performed on the SB&PC system using simulated signals.	Test report(s) confirm that the SB&PC system is capable of performing the functions defined in Table 2.2.9-2.
3. SB&PC system minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.	See Section 3.3	See Section 3.3.

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 safety-related portions of the systems listed in Table 2.2.10-1.

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.

Table 2.2.10-1

Systems and Functions Comprising The Q-DCIS

System [1]	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	2.2.13
Diverse Instrumentation and Control System	2.2.14
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

(1) Safety-related portions only

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.

N-DCIS includes the systems listed in Table 2.2.11-1.

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 systems in Table 2.2.11-1.

Table 2.2.11-1

Systems and Functions Comprising The N-DCIS

System (1)	Subsection
Rod Control and Information System	2.2.1
Control Rod Drive System • CRD Hydraulic System	2.2.2
Feedwater Control System	2.2.3
Remote Shutdown System • Nonsafety-related VDUs	2.2.6
Plant Automation System	2.2.8
Steam Bypass and Pressure Control System	2.2.9
Diverse Instrumentation and Control System • Diverse Protection System	2.2.14
Reactor Water Cleanup/Shutdown Cooling System	2.6.1
Fuel and Auxiliary Pools Cooling System	2.6.2
Main Control Room Panels • Nonsafety-related VDUs	2.7.1
Condensate and Feedwater System	2.11.2
Condensate Purification System	2.11.3
Turbine Generator System • Turbine Generator Control System	2.11.4
Reactor Component Cooling Water System	2.12.3
Turbine Component Cooling Water System	2.12.4
Chilled Water System	2.12.5
Plant Service Water System	2.12.7
Instrument Air System	2.12.9
Electrical Power Distribution System	2.13.1
Standby On Site Power Supply	2.13.4
Drywell Cooling System	2.15.6
Plant Service Water System • Cooling Tower	4.4

(1) Nonsafety-related portions only

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, and initiates closure of inboard and outboard main steamline isolation valves (MSIVs), containment isolation valves (CIVs), and Reactor Building (RB) isolation dampers by the safety-related reactor trip and isolation function (RTIF) and SSLC/ESF programmable logic controller platforms.

Functional Arrangement

(1) LD&IS functional arrangement is defined in Tables 2.2.12-1.

Functional Requirements

(2) LD&IS isolation function monitored variables are defined in Table 2.2.12-2.

(3) LD&IS leakage source monitored variables are defined in Table 2.2.12-3.

(4) LD&IS controls, interlocks, and bypasses are defined in Table 2.2.12-4.

(5) Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components is addressed in Subsection 2.2.15.

(6) The equipment qualification of LD&IS components defined in Table 2.2.12-1 is addressed in Section 3.8.

(7) LD&IS minimum inventory of alarms, displays, and status indications in the main control room are addressed in Section 3.3.

(8) The containment isolation components that correspond to the isolation functions defined in Tables 2.2.12-2 and 2.2.12-3 are addressed in Subsection 2.15.1.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.12-5 defines the inspections, tests, and/or analyses, together with associated acceptance criteria for the LD&IS.

Table 2.2.12-1

LD&IS Functional Arrangement

LD&IS is safety-related, Seismic Category I. LD&IS functions, other than MSIV isolation function, is implemented by SSLC/ESF platform. LD&IS MSIV isolation function is implemented by the RTIF platform. LD&IS isolation functions logic is designed to provide an actuation by requiring coincident trip of at least two divisions to cause the trip output. LD&IS logic is de-energized to initiate the isolation function (i.e., fail-safe). LD&IS logic channels and associated sensors are powered from safety-related power supplies. Drywell floor drain high conductivity waste (HCW) sump instrumentation is designed with the sensitivity to detect a leakage step-change (increase) of 3.8 liters/min (1.0 gpm) within one hour and to alarm at sump flow rates in excess of 19 liters/min (5 gpm).

Table 2.2.12-2

LD&IS Isolation Function Monitored Variables

Monitored Variables	**LD&IS Isolation Functions** (1)									
	Main Steam & Drain Lines	RWCU/ SDC Lines	IC System Lines	Fission Products Sampling Lines	DW LCW Sump Drain Line	DW HCW Sump Drain Line	Containment Purge & Vent Valves	CWS Lines to DW Air Coolers	FAPCS Process Lines	R /B HVAC Exhaust Ducts
RWCU/SDC Flow High	–	X	–	–	–	–	–	–	–	–
HCW Drain Line Radiation High	–	–	–	–	–	X	–	–	–	–
LCW Drain Line Radiation High	–	–	–	–	X	–	–	–	–	–
SLC Initiation Signal	–	X	–	–	–	–	–	–	–	–
Refueling Area Air Exhaust Radiation High	–	–	–	–	–	–	X	–	–	X
Reactor Building Air Exhaust Radiation High	–	–	–	–	–	–	X	–	–	X
IC Condensate Flow High	–	–	X	–	–	–	–	–	–	–
IC Steam Flow High	–	–	X	–	–	–	–	–	–	–
DW Pressure High	–	–	–	X	X	X	X	X	X	X

Table 2.2.12-3

LD&IS leakage Source Monitored Variables

Monitored Variables	**Leakage Source** (2)																					
	Main Steam-lines		IC Steam-lines		IC Conden-sate Lines		CWS Lines		FAPCS Lines		RWCU/ SDC Lines		Feed-water Lines		GDCS Water		Reactor Vessel Head Seal		Valve Stem Packing		Misc. Leaks	
Location (1)	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O
Intersystem Leakage Radiation High	–	–	–	–	–	–	–	–	–	–	–	X	–	–	–	–	–	–	–	–	–	–
RWCU/SDC Flow High	–	–	–	–	–	–	–	–	–	–	–	X	–	–	–	–	–	–	–	–	–	–
Equip. Areas Differential Temperature High	–	X	–	–	–	–	–	–	–	–	–	X	–	–	–	–	–	–	–	–	–	–
MSL Tunnel or Turbine Building Area Ambient Temperature High	–	X	–	–	–	–	–	–	–	–	–	X	–	X	–	–	–	–	–	–	–	–
MSL or IC Steamline Flow High	X	X	X	X	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–
DW Air Cooler Cond. Flow High	X	–	X	–	–	–	X	–	X	–	X	–	X	–	X	–	–	–	–	–	–	–
Vessel Head Flange Seal Pressure High	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	X	–	–	–	–	–
RB Equip. / Floor Drain Sump Pump Activity	–	X	–	X	–	X	–	X	–	X	–	X	–	X	–	–	–	–	–	X	–	X

Table 2.2.12-3 (Continued)
LD&IS leakage Source Monitored Variables

Monitored Variables	**Leakage Source** [2]																					
	Main Steam-lines		IC Steam-lines		IC Conden-sate Lines		CWS Lines		FAPCS Lines		RWCU/ SDC Lines		Feed-water Lines		GDCS Water		Reactor Vessel Head Seal		Valve Stem Packing		Misc. Leaks	
Location [1]	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O	I	O
SRV Discharge Line Temperature High	X	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–
DW Temperature High	X	–	X	–	–	–	X	–	X	–	X	–	X	–	X	–	–	–	–	–	X	–
DW Fission Product Radiation High	X	–	X	–	X	–	–	–	–	–	X	–	X	–	–	–	–	–	–	–	–	–
DW Equip. Drain Sump Level Change High	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	X	–	X	–	–	–
DW Floor Drain Sump Level Change High	X	–	X	–	X	–	X	–	X	–	X	–	X	–	X	–	–	–	–	–	X	–
DW Pressure High	X	–	X	–	–	–	–	–	–	–	X	–	X	–	–	–	–	–	–	–	–	–
RPV Water Level Low (L1, L2)	X	X	X	X	X	X	X	–	X	–	X	X	X	–	–	–	–	–	–	–	–	–

(1) "I" means inside DW leakage; "O" means outside DW leakage.

(2) "X" means control/alarm is associated with the monitored variable; "–" means not applicable.

Table 2.2.12-4

LD&IS Controls, Interlocks, and Bypasses

Parameter	Description
Control	Manual isolation (individually transfer open/close each CIV and MSIV) Manual reset (individually reset CIV and MSIV isolation logic to enable CIV and MSIV manual open function) MSIV test switches Isolation logic reset switches
Interlock	Reactor Mode Switch RPV water level low (Level 2) time delay (RTIF) RPV water level low (Level 2) time delay (SSLC/ESF) Turbine stop valve closed position RPV pressure Drywell pressure
Bypass	RTIF Division of sensors channel inputs to each division manual bypass switches RTIF Divisional actuator trip manual switches SSLC/ESF Division of sensors channel inputs to each manual bypass switch SSLC/ESF manual control bypasses (disables) the load driver/discrete output

Table 2.2.12-5

ITAAC For Leak Detection and Isolation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. LD&IS functional arrangement is defined in Tables 2.2.12-1.	Inspections and/or tests will be conducted on the as-built configuration as defined in Tables 2.2.12-1.	Report(s) document(s) that the system conforms to the functional arrangement defined in Tables 2.2.12-1.
2. LD&IS isolation function monitored variables are defined in Table 2.2.12-2.	See Subsection 2.2.15.	See Subsection 2.2.15.
3. LD&IS leakage source monitored variables are defined in Table 2.2.12-3.	See Subsection 2.2.15.	See Subsection 2.2.15.
4. LD&IS controls, interlocks, and bypasses are defined in Table 2.2.12-4.	See Subsection 2.2.15.	See Subsection 2.2.15.
5. Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components is addressed in Subsection 2.2.15.	See Subsection 2.2.15.	See Subsection 2.2.15.
6. The equipment qualification of LD&IS components defined in Table 2.2.12-1 is addressed in Section 3.8.	See Section 3.8.	See Section 3.8.
7. LD&IS minimum inventory of alarms, displays, and status indications in the main control room are addressed in Section 3.3.	See Section 3.3.	See Section 3.3.

Table 2.2.12-5

ITAAC For Leak Detection and Isolation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The containment isolation components that correspond to the isolation functions defined in Table 2.2.12-2 are addressed in Subsection 2.15.1.	See Subsection 2.15.1.	See Subsection 2.15.1.

2.2.13 Engineered Safety Features Safety System Logic and Control [VJ458]

Design Description

The Safety System Logic and Control for the Engineered Safety Features systems (SSLC/ESF) addressed in this subsection performs the safety-related Emergency Core Cooling System (ECCS) control logic, the isolation logic for the control room habitability system (CRHS), and the safe shutdown function of the Isolation Condenser System (ICS).

Functional Arrangement

(1) The SSLC/ESF functional arrangement is described in Table 2.2.13-1.

Functional Requirements

(2) SSLC/ESF automatic functions, initiators, and associated interfacing systems are described in Table 2.2.13-2.

(3) SSLC/ESF controls, interlocks, and bypasses in the main control room (MCR) are described in Table 2.2.13-3.

(4) Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components is addressed in Subsection 2.2.15.

(5) SSLC/ESF minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.

(6) The equipment qualification of SSLC/ESF components described in Table 2.2.13-1 is addressed in Section 3.8.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.13-4 defines the inspections, tests, and/or analyses, together with associated acceptance criteria for the SSLC/ESF system.

Table 2.2.13-1

SSLC/ESF Functional Arrangement

SSLC/ESF comprises four redundant, safety-related, Seismic Category I, divisions of trip logics and trip actuators. SSLC/ESF receives the inputs from, and sends outputs to interfacing systems as defined in Tables 2.2.13-2 and 2.2.13-3. SSLC/ESF logic is designed to provide a trip initiation by requiring coincident trip of at least two divisions to cause the trip output. Redundant safety-related power supplies are provided for each division. SSLC/ESF uses “energized-to-trip” and “fail-as-is” logic. ADS (SRVs and DPVs), GDCS, and SLC are actuated sequentially and in groups.

Table 2.2.13-2

SSLC/ESF Automatic Functions, Initiators, and Associated Interfacing Systems

Function	Initiator	Interfacing System
ADS	RPV reactor water level low (Level 1)	RPS
GDCS Injection	RPV reactor water level low (Level 1)	RPS
GDCS Equalizing Lines	RPV reactor water level low (Level 1)	RPS
ICS	RPV reactor water level low (Level 1)	RPS
SLC	RPV reactor water level low (Level 1)	RPS
CRHAVS emergency filtration mode	CRHA inlet air supply radiation high from PRMS	PRMS
CRHAVS isolation	Smoke detectors	FPS

Table 2.2.13-3

SSLC/ESF Controls, Interlocks, and Bypasses

Parameter	Description
Control	ADS sequence actuation from VDUs in the MCR (one arm/fire switch per division)
	GDCS sequence actuation from VDUs in the MCR (one arm/fire switch per division)
Interlock	ECCS-LOCA confirmation time delay for ADS
	Group 1 SRV open time delay
	Group 2 SRV open time delay
	Group 1 DPV open and SLC initiation time delay
	Group 2 DPV open time delay
	Group 3 DPV open time delay
	Group 4 DPV open time delay
	GDCS manual initiation interlock on low reactor pressure signal
	GDCS injection squib valve open time delay
	GDCS equalization line squib valve open time delay
	GDCS equalization line squib valve open interlock (RPV water level low (Level 0.5)
	GDCS equalization line squib valve open time delay - manual actuation
Bypass	SSLC/ESF Division of sensors division manual bypass switch
	SSLC/ESF manual control bypasses (disables) the load driver/discrete output

Table 2.2.13-4

ITAAC For Safety System Logic and Control/ESF System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The SSLC/ESF functional arrangement is described in Table 2.2.13-1.	Inspection and/or tests will be conducted in the as-built configuration as described in Table 2.2.13-1.	Inspection and/or test report(s) exist(s) and conclude(s) that the system conforms to the functional arrangement as described in Table 2.2.13-1.
2. SSLC/ESF automatic trip initiators and associated interfacing systems are described in Table 2.2.13-2.	See Subsection 2.2.15.	See Subsection 2.2.15.
3. SSLC/ESF controls, interlocks, and bypasses in the main control room (MCR) are defined in Table 2.2.13-3.	See Subsection 2.2.15.	See Subsection 2.2.15.
4. Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components is addressed in Subsection 2.2.15.	See Subsection 2.2.15.	See Subsection 2.2.15.
5. SSLC/ESF minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.	See Section 3.3.	See Section 3.3.
6. The equipment qualification of SSLC/ESF components is addressed in Section 3.8.	See Section 3.8.	See Section 3.8.

2.2.14 Diverse Instrumentation and Controls

Design Description

The diverse instrumentation and control system (DICS) comprises the Anticipated Transients Without Scram Standby Liquid Control (ATWS/SLC) system and the diverse protection system (DPS).

Functional Arrangement

(1) DICS functional arrangement is defined in Table 2.2.14-1 and Figure 2.2.14-1.

Functional Requirements

(2) DICS automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.14-2.

(3) DICS controls, interlocks and bypasses in the main control room (MCR) are defined in Table 2.2.14-3.

(4) DICS minimum inventory of alarms, displays, controls, and status indications in the main control room are addressed in Section 3.3.

(5) The equipment qualification of DICS components defined in Table 2.2.14-1 is addressed in Section 3.8.

(6) The containment isolation components that correspond to the isolation functions defined in Table 2.2.14-2 are addressed in Subsection 2.15.1. [TCF510]

(7) Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Table 2.2.14-1 is addressed in Subsection 2.2.15.

(8) Confirmatory analyses to support and validate the DPS design scope. [SMK511]

(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. [SMK512]

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.2.14-4 defines the inspections, tests, and/or analyses, together with associated acceptance criteria for the DICS.

Table 2.2.14-1

DICS Functional Arrangement

ATWS/SLC system is safety-related.
ATWS/SLC system is housed within each of the divisional safety-related reactor trip and isolation function (RTIF) cabinets on a separate chassis from the other equipment within the cabinet.
RTIF cabinets housing the ATWS/SLC system are located within the divisional electrical rooms in the control building (CB) in a mild environment.
ATWS/SLC system logic is designed to provide a trip initiation by requiring coincident trip of at least two divisions to cause the trip output.
Each ATWS/SLC system is powered by a divisionally separated safety-related power supply.
ATWS/SLC RPV has dome pressure sensors and RPV water level sensors that are hardwired to their respective divisional controller.
DPS uses a triple redundant, digital controller, powered from nonsafety-related load group power supplies.
DPS triple redundant digital controllers require agreement in at least two channels for a coincident trip actuation.
DPS is housed in a cabinet located in a mild environment in the CB.
DPS scram initiation logic is “energize-to-actuate” applied at the power return side of the control circuit going to the scram pilot valve solenoids.
DPS logic is “energize-to-actuate”.

Table 2.2.14-2

DICS Functions, Initiators, and Interfacing Systems

Function	Initiator	Interfacing System
SLC system initiation (ATWS/SLC)	RPV dome pressure high and Startup Range Neutron Monitor (SRNM) signal greater than ATWS setpoint (SRNM ATWS permissive) with time delay	NMS, NBS
	RPV water level low (Level 2) and SRNM ATWS permissive with time delay	NBS, NMS
	RPV water level low (Level 1)	NBS
FWRB (ATWS/SLC)	RPV dome pressure high and SRNM ATWS permissive	NBS, NMS
ADS inhibit (ATWS/SLC)	RPV water level low (Level 2) and APRM ATWS permissive	NBS, NMS
	RPV dome pressure high and APRM ATWS permissive with time delay	NBS, NMS
ATWS ARI and FMCRD motor run-in (DPS)	RPV dome pressure high	NBS
	RPV water level low (Level 2)	NBS
	Manual ATWS mitigation signal present	-
	RPS scram signal	RPS
	SCRRI/SRI signal and power levels remain elevated	RPS
	Manual DPS scram signal	-
SRI (DPS)	SCRRI signal.	RC&IS

Table 2.2.14-2

DICS Functions, Initiators, and Interfacing Systems

Function	Initiator	Interfacing System
	Generator load rejection signal.	TGCS
	Loss of FW heating.	FWCS
	Turbine trip signal.	TGCS
	OPRM thermal neutron flux oscillation	NMS
	Manual SCRRI / SRI	-
Delayed FWRB (DPS)	SCRRI/SRI signal and power levels remain elevated	NMS,
DPS Scram (DPS)	RPV dome pressure high.	NBS
	RPV water level high (Level 8).	NBS
	RPV water level low (Level 3).	NBS
	Drywell pressure high.	CMS
	Suppression pool temperature high.	CMS
	MSIV closure	NBS
ADS initiation	RPV water level low (Level 1)	NBS
GDCS initiation	RPV water level low (Level 1)	NBS
ICS initiation (DPS)	RPV water level low (Level 2)	NBS
	RPV water level low (Level 1)	NBS
	MSIV closure	NBS
SLC system initiation	RPV water level low (Level 1)	NBS

Table 2.2.14-2

DICS Functions, Initiators, and Interfacing Systems

Function	Initiator	Interfacing System
MSIV closure (DPS)	Steam flow high	NBS
	RPV pressure low	NBS
	RPV water level low (Level 2)	NBS
ICS isolation valve closure (DPS)	Steam flow high	ICS
	Condensate flow high	ICS
RWCU/SDC isolation valve closure (DPS)	Differential flow high	RWCU/SDC
FWRB (DPS)	RPV water level high (Level 8)	NBS
FW pump trip (DPS)	RPV water level high (Level 9)	NBS
FW isolation (DPS)	Line differential pressure high	NBS

Table 2.2.14-3

DICS Controls, Interlocks and Bypasses

Control	Manual initiation of ADS
	Manual initiation of ICS
	Manual initiation of GDCS squib-initiated injection valves
	Manual initiation of GDCS squib-initiated equalization valves
	Manual initiation of ATWS SLC
	Manual initiation of ATWS ARI
	Manual initiation of ATWS FWRB
	Manual scram
Interlock	APRM ATWS Permissive
	SRNM ATWS Permissive
	Reactor Mode Switch position
	Time Delays
Bypass	Division of sensor bypass

Table 2.2.14-4

ITAAC For Diverse Instrumentation and Controls

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. DICS functional arrangement is defined in Table 2.2.14-1 and Figure 2.2.14-1.	Inspection(s), test(s), and/or type test(s) will be conducted on the as-built system configuration defined in Table 2.2.14-1 and Figure 2.2.14-1.	Inspection(s), test(s), and/or type test(s) document(s) that the system conforms to the functional arrangement defined in Table 2.2.14-1 and Figure 2.2.14-1.
2. DICS automatic functions, initiators, and associated interfacing systems are defined in Table 2.2.14-2.	a. Tests will be performed on the DICS nonsafety-related components will be conducted on the as-built system configuration using simulated signals. b. For safety-related DCIS components, see Subsection 2.2.15.	a. Test report(s) confirm that the DICS is capable of performing the functions defined in Table 2.2.14-2. b. For safety-related DCIS components, see Subsection 2.2.15.
3. DICS interlocks and controls are defined in Table 2.2.14-3.	a. Test(s) and type test(s) will be performed on the DICS nonsafety-related logic process interlocks and controls defined in Table 2.2.14-3. b. For safety-related DCIS components, see Subsection 2.2.15.	a. Test report(s) document(s) that the DICS logic process interlocks and issue control signals defined in Table 2.2.14-3. b. For safety-related DCIS components, see Subsection 2.2.15.
4. DICS minimum inventory of alarms, displays, controls, and status indications in the main control room are addressed in Section 3.3.	See Section 3.3.	See Section 3.3.

Table 2.2.14-4

ITAAC For Diverse Instrumentation and Controls

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The equipment qualification of DICS components defined in Table 2.2.14-1 is addressed in Section 3.8.	The equipment qualification of DICS components defined in Table 2.2.14-1 is addressed in Section 3.8.	See Section 3.8.
6. The containment isolation components that correspond to the isolation functions defined in Table 2.2.14-2 are addressed in Subsection 2.15.1.	The containment isolation components that correspond to the isolation functions defined in Table 2.2.14-2 are addressed in Subsection 2.15.1.	See Subsection 2.15.1.
7. Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Table 2.2.14-1 is addressed in Subsection 2.2.15.	Conformance with IEEE Std. 603 requirements by the safety-related control system structures, systems, and components defined in Table 2.2.14-1 is addressed in Subsection 2.2.15.	See Subsection 2.2.15.
8. Confirmatory analyses to support and validate the DPS design scope.	Confirmatory analyses to support and validate the DPS design scope.	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.
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.	Complete FMEA per NUREG/CR-6303 to validate the DPS protection functions described in LTR NEDO-33251.	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.

2.2.15 Instrumentation & Control Compliance With IEEE Std. 603

Design Description

The design descriptions related to IEEE Std. 603 criteria are provided below. Safety-related Instrumentation and Control systems are designed to the following criteria from IEEE Std. 603 as listed in Table 2.2.15-1. An X in the table identifies the system for which an ITAAC applies. Refer to the Tier 1 Subsections cited in the table for additional design descriptions applicable to the listed systems. Note that only the safety-related portions of the listed systems are addressed.

(1) Criterion 5.1, Single Failure: The listed systems are designed to ensure that safety-related functions required for design basis events (DBE) are performed in the presence of: (a) single detectable failures within safety-related systems concurrent with identifiable but non-detectable failures; (b) failures caused by the single failure; and (c) failures and spurious system actions that cause or are caused by the design basis event requiring the safety-related functions, as identified in the applicable failure modes and effects analysis (FMEA).

(2) Criteria 5.2 and 7.3, Completion of Protective Actions: The listed systems are designed so that, (a) once initiated (automatically or manually), the intended sequences of safety-related functions of the execute features continue until completion, and (b) after completion, deliberate operator action is required to return the safety-related systems to normal.

(3) Criterion 5.6, Independence: For the listed systems, there is physical, electrical, and communications independence between redundant portions of safety-related systems, between safety-related systems and the effects of a DBE, and between safety-related systems and nonsafety-related systems, as identified in the applicable FMEA.

(4) Criteria 5.7 and 6.5, Capability for Test & Calibration: The listed systems have the capability to have their equipment tested and calibrated while retaining their capability to accomplish their safety-related functions.

(5) Criterion 5.9, Control of Access: The listed systems have features that permit administrative control of access to safety-related system equipment.

(6) Criteria 6.1 and 7.1, Automatic Control: The listed systems provide the means to automatically initiate and control the required safety-related functions.

(7) Criteria 6.2 and 7.2, Manual Control: The listed systems have features in the main control room to manually initiate and control the automatically initiated safety-related functions at the division level.

(8) Criteria 6.6 and 7.4, Operating Bypasses: The listed systems automatically (1) prevent the activation of an operating bypass, whenever the applicable permissive conditions for an operating bypass are not met, and (2) remove activated operating bypass(es), if the plant conditions change so that an activated operating bypass is no longer permissible.

(9) Criteria 6.7 and 7.5, Maintenance Bypasses: The listed systems are capable of performing their safety-related functions, when one division is in maintenance bypass.

(10) Criterion 6.8, Setpoint: The listed system setpoints for safety-related functions are determined by a defined setpoint methodology.

Inspections, Tests, Analysis and Acceptance Criteria

Table 2.2.15-2 provides a definition of the inspections, tests, and/or analyses, together with and acceptance criteria for the systems listed in Table 2.2.15-1.

Table 2.2.15-1

ITAAC Applicability Matrix (1)

IEEE Std. 603 Criterion	Applicable System (Tier 1 Subsection) (2)												
	NBS (2.1.2)	CRDS (2.2.2)	SLC System (2.2.4)	NMS (2.2.5)	RSS (2.2.6)	RPS (2.2.7)	LD&IS (2.2.12)	SSLC/ESF (2.2.13)	PRMS (2.3.1)	ICS (2.4.1)	GDCS (2.4.2)	CMS (2.15.7)	SPTM (2.15.7)
5.1	-	-	-	X	-	X	X	X	-	X	X	-	-
5.2 and 7.3	-	-	-	-	-	X	X	X	-	-	-	-	-
5.6	X	X	X	X	X	X	X	X	X	X	X	X	X
5.7 and 6.5	X	X	X	X	-	X	X	X	-	X	X	X	X
5.9	-	-	-	X	X	X	X	X	-	-	-	-	-
6.1 and 7.1	-	-	-	-	-	X	-	X	-	-	-	-	-
6.2 and 7.2	-	-	-	-	-	X	-	X	-	-	-	-	-
6.6 and 7.4	-	-	-	X	-	X	-	X	-	-	-	-	-
6.7 and 7.5	-	-	-	X	-	X	-	X	-	-	-	-	-
6.8	X	-	X	X	-	X	X	X	X	-	-	X	X

(1) A dash means not applicable.

(2) Safety-related portions only

Table 2.2.15-2

ITAAC For IEEE Std. 603 Compliance Confirmation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Criterion 5.1, Single Failure: The Criterion 5.1 systems listed in Table 2.2.15-1 are designed to ensure that safety-related functions required for design basis events (DBE) are performed in the presence of: (a) single detectable failures within safety-related systems concurrent with identifiable but non-detectable failures; (b) failures caused by the single failure; and (c) failures and spurious system actions that cause or are caused by the DBE requiring the safety-related functions, as identified in the applicable FMEA.	Criterion 5.1, Single Failure: Block level FMEA of the Criterion 5.1 systems listed in Table 2.2.15-1 show that they perform safety-related functions required for design basis events in the presence of: (a) single detectable failures within safety-related systems concurrent with identifiable but non-detectable failures; (b) failures caused by the single failure; and (c) failures and spurious system actions that cause or are caused by the DBE requiring the safety-related functions, as identified in the applicable FMEA. {{DAC}}	Criterion 5.1, Single Failure: Analysis report(s) conclude(s) that the systems identified in Table 2.2.15-1 for Criterion 5.1 ensure(s) that safety-related functions required for design basis events are performed in the presence of: (a) single detectable failures within safety-related systems concurrent with identifiable but non-detectable failures; (b) failures caused by the single failure; and (c) failures and spurious system actions that cause or are caused by the DBE requiring the safety-related functions, as identified in the applicable FMEA. {{DAC}}

Table 2.2.15-2

ITAAC For IEEE Std. 603 Compliance Confirmation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. Criteria 5.2 and 7.3, Completion of Protective Actions: The Criteria 5.2 and 7.3 systems listed in Table 2.2.15-1 are designed so that, (a) once initiated (automatically or manually), the intended sequences of safety-related functions of the execute features continue until completion, and (b) after completion, deliberate operator action is required to return the safety-related systems to normal.	Criteria 5.2 and 7.3, Completion of Protective Actions: a. Inspection of the simplified logic diagrams (SLDs) for the Criteria 5.2 and 7.3 systems listed in Table 2.2.15-1 verifies that the design shows (a) "seal-in" features that are provided to enable system-level safety-related functions to go to completion, and (b) "manual reset" features that are provided to require deliberate operation action to return the safety-related systems to normal. {{DAC}}	Criteria 5.2 and 7.3, Completion of Protective Actions: a. Inspection report(s) conclude(s) that the SLDs show (a) "seal-in" features, and (b) "manual reset" features. {{DAC}}
	b. Test(s) for the Criteria 5.2 and 7.3 systems listed in Table 2.2.15-1 will be performed to show that (a) once initiated (automatically or manually), the intended sequences of safety-related functions of the "execute features" continue until completion, and (b) after completion, deliberate operator action is required to return the safety-related systems to normal.	b. Test report(s) conclude(s) that for the Criteria 5.2 and 7.3 systems listed in Table 2.2.15-1, (a) once initiated (automatically and manually), the intended sequences of safety-related functions of the "execute features" continue until completion, and (b) after completion, deliberate operator action is required to return the safety-related systems to normal.

Table 2.2.15-2

ITAAC For IEEE Std. 603 Compliance Confirmation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. Criterion 5.6, Independence: For the Criterion 5.6 systems listed in Table 2.2.15-1, there is physical, electrical, and communications independence between redundant portions of a safety-related system, between safety-related systems and the effects of a DBE, and between safety-related systems and nonsafety-related systems, as identified in the applicable FMEA.	Criterion 5.6, Independence: a. Block level FMEA will be performed to verify that the designs of the Criterion 5.6 systems listed in Table 2.2.15-1 have physical, electrical, and communications independence between redundant portions of a safety-related system, between safety-related systems and the effects of a DBE, and between safety-related systems and nonsafety-related equipment, as identified in the applicable FMEA. {{DAC}} b. Inspection(s) will be performed to demonstrate that the Criterion 5.6 systems listed in Table 2.2.15-1 have physical independence between redundant portions of a safety-related system, between safety-related systems and the effects of a DBE, and between safety-related systems and nonsafety-related equipment, as identified in the applicable FMEA.	Criterion 5.6, Independence: a. Analysis report(s) conclude(s) that the designs of the Criterion 5.6 systems listed in Table 2.2.15-1 have physical, electrical, and communications independence between redundant portions of a safety-related system, between safety-related systems and the effects of a DBE, and between safety-related systems and nonsafety-related equipment, as identified in the applicable FMEA. {{DAC}} b. Inspection report(s) conclude(s) that the Criterion 5.6 systems listed in Table 2.2.15-1 have physical independence between redundant portions of a safety-related system, between safety-related systems and the effects of a DBE, and between safety-related systems and nonsafety-related equipment, as identified in the applicable FMEA.

Table 2.2.15-2

ITAAC For IEEE Std. 603 Compliance Confirmation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	c. Type test(s), test(s), and / or analysis(es) will be performed to demonstrate that the Criterion 5.6 systems communication interface modules listed in Table 2.2.15-1 have electrical and communications independence between redundant portions of a safety-related system, between safety-related systems and the effects of a DBE, and between safety-related systems and nonsafety-related equipment.	c. Type test(s), test(s), and / or analysis(es) report(s) conclude(s) that the Criterion 5.6 systems communication interface modules listed in Table 2.2.15-1 have electrical and communications independence between redundant portions of a safety-related system, between safety-related systems and the effects of a DBE, and between safety-related systems and nonsafety-related equipment.

Table 2.2.15-2

ITAAC For IEEE Std. 603 Compliance Confirmation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Criteria 5.7 and 6.5, Capability for Test and Calibration: The Criteria 5.7 and 6.5 systems listed in Table 2.2.15-1 have the capability to have their equipment tested and calibrated while retaining their capability to accomplish their safety-related functions.	Criteria 5.7 and 6.5, Capability for Test and Calibration: a. Inspection(s) of the SLDs of the Criteria 5.7 and 6.5 systems listed in Table 2.2.15-1 will be performed to verify that both the automatic and manual circuitry have the capability to have the safety-related systems' equipment tested and calibrated while retaining the safety-related systems' capability to accomplish their safety-related functions. {{DAC}} b. Test(s) of Criteria 5.7 and 6.5 systems listed in Table 2.2.15-1 will be performed to demonstrate that the design allows for tripping or bypass of individual functions in each safety-related system channel. c. Test(s) of Criteria 5.7 and 6.5 systems listed in Table 2.2.15-1, will be performed to demonstrate that the digital computer-based I&C systems' self-test features confirm computer system operation on system initiation.	Criteria 5.7 and 6.5, Capability for Test and Calibration: a. Inspection report(s) conclude(s) that the SLDs of the Criteria 5.7 and 6.5 systems listed in Table 2.2.15-1 have the capability to have the safety-related systems' equipment tested and calibrated while retaining the safety-related systems' capability to accomplish their safety-related functions. {{DAC}} b. Test report(s) conclude(s) that for the Criterion 5.7 and 6.5 systems listed in Table 2.2.15-1 individual functions in each safety-related system channel can be tripped or bypassed. c. Test report(s) conclude(s) that for the Criteria 5.7 and 6.5 systems listed in Table 2.2.15-1, the digital computer-based I&C systems' self-test features confirm computer system operation on system initiation.

Table 2.2.15-2

ITAAC For IEEE Std. 603 Compliance Confirmation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Criterion 5.9, Control of Access: The design of the Criterion 5.9 systems listed in Table 2.2.15-1 have features that permit administrative control of access to safety-related system equipment.	Criterion 5.9, Control of Access: Inspection of system design specification(s) for the Criterion 5.9 systems listed in Table 2.2.15-1 will be performed to confirm that access control features are specified for safety-related systems equipment. {{DAC}}	Criterion 5.9, Control of Access: Inspection report(s) conclude(s) that within the system design specification(s) of the Criterion 5.9 systems listed in Table 2.2.15-1, access control features are specified for safety-related systems equipment. {{DAC}}
6. Criteria 6.1 and 7.1, Automatic Control: The Criteria 6.1 and 7.1 systems listed in Table 2.2.15-1 provide the means to automatically initiate and control the required safety-related functions.	Criteria 6.1 and 7.1, Automatic Control: a. Inspection(s) will be performed of the SLDs for the Criteria 6.1 and 7.1 systems listed in Table 2.2.15-1 to verify that the design automatically initiates and controls the required safety-related functions. {{DAC}} b. Test(s) will be performed to demonstrate that the Criteria 6.1 and 7.1 systems listed in Table 2.2.15-1 automatically initiate and control the required safety-related functions.	Criteria 6.1 and 7.1, Automatic Control: a. Inspection report(s) conclude(s) that the SLDs for the Criteria 6.1 and 7.1 systems listed in Table 2.2.15-1 show(s) that the design automatically initiates and controls the required safety-related functions. {{DAC}} b. Test report(s) conclude(s) that the Criteria 6.1 and 7.1 systems listed in Table 2.2.15-1 automatically initiate and control the required safety-related functions.

Table 2.2.15-2

ITAAC For IEEE Std. 603 Compliance Confirmation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. Criteria 6.2 and 7.2, Manual Control: The Criteria 6.2 and 7.2 systems listed in Table 2.2.15-1 have features in the main control room to manually initiate and control the automatically initiated safety-related functions at the division level.	Criteria 6.2 and 7.2, Manual Control: a. Inspection(s) will be performed of the SLDs for the Criteria 6.2 and 7.2 systems listed in Table 2.2.15-1 to verify that they have main control room features that are capable of manually initiating and controlling automatically initiated safety-related functions at the division level. {{DAC}} b. Test(s) will be performed to demonstrate that the Criteria 6.2 and 7.2 systems listed in Table 2.2.15-1 have main control room features that manually initiate and control automatically initiated safety-related functions at the division level.	Criteria 6.2 and 7.2, Manual Control: a. Inspection report(s) conclude(s) that the SLDs for the Criteria 6.2 and 7.2 systems listed in Table 2.2.15-1 have main control room features that are capable of manually initiating and controlling automatically initiated safety-related functions at the division level. {{DAC}} b. Test report(s) conclude(s) that the Criteria 6.2 and 7.2 systems listed in Table 2.2.15-1 have main control room features that manually initiate and control automatically initiated safety-related functions at the division level exist(s).

Table 2.2.15-2

ITAAC For IEEE Std. 603 Compliance Confirmation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Criteria 6.6 and 7.4, Operating Bypasses: The Criterion 6.6 and 7.4 systems listed in Table 2.2.15-1 automatically (1) prevent the activation of an operating bypass, whenever the applicable permissive conditions for an operating bypass are not met, and (2) remove activated operating bypass(es), if the plant conditions change so that an activated operating bypass is no longer permissible.	Criteria 6.6 and 7.4, Operating Bypasses: a. Inspections(s) will be performed of the SLDs for the Criterion 6.6 and 7.4 systems listed in Table 2.2.15-1 to verify that the systems are capable of automatically (1) preventing the activation of an operating bypass, whenever the applicable permissive conditions for an operating bypass are not met, and (2) removing activated operating bypasses, if the plant conditions change so that an activated operating bypass is no longer permissible. {{DAC}} b. Test(s) will be performed to demonstrate that the Criterion 6.6 and 7.4 systems listed in Table 2.2.15-1 automatically (1) prevent the activation of an operating bypass, whenever the applicable permissive conditions for an operating bypass are not met, and (2) remove activated operating bypass(es), if the plant conditions change so that an activated operating bypass is no longer permissible.	Criteria 6.6 and 7.4, Operating Bypasses: a. Inspection report(s) conclude that the SLDs for the Criterion 6.6 and 7.4 systems listed in Table 2.2.15-1 show that the systems are capable of automatically (1) preventing the activation of an operating bypass, whenever the applicable permissive conditions for an operating bypass are not met, and (2) removing activated operating bypasses, if the plant conditions change so that an activated operating bypass is no longer permissible. {{DAC}} b. Test report(s) conclude(s) that the Criterion 6.6 and 7.4 systems listed in Table 2.2.15-1 automatically (1) prevent the activation of an operating bypass, whenever the applicable permissive conditions for an operating bypass are not met, and (2) remove activated operating bypass(es), if the plant conditions change so that an activated operating bypass is no longer permissible.

Table 2.2.15-2

ITAAC For IEEE Std. 603 Compliance Confirmation

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. Criteria 6.7 and 7.5, Maintenance Bypasses: The Criterion 6.7 and 7.5 systems listed in Table 2.2.15-1 are capable of performing their safety-related functions, when one division is in maintenance bypass.	Criteria 6.7 and 7.5, Maintenance Bypasses: a. Inspections(s) will be performed of the SLDs for the Criterion 6.7 and 7.5 systems listed in Table 2.2.15-1 to verify that the safety-related systems are capable of performing their safety-related functions, when one division is in maintenance bypass. {{DAC}} b. Test(s) will be performed to demonstrate that the Criterion 6.7 and 7.5 systems listed in Table 2.2.15-1 perform their safety-related functions, when one division is in maintenance bypass.	Criteria 6.7 and 7.5, Maintenance Bypasses: a. Inspection report(s) conclude(s) that the SLDs for the Criterion 6.7 and 7.5 systems listed in Table 2.2.15-1 show that the safety-related systems are capable of performing their safety-related functions, when one division is in maintenance bypass. {{DAC}} b. Test report(s) conclude(s) that the Criterion 6.7 and 7.5 systems listed in Table 2.2.15-1 perform their safety-related functions, when one division is in maintenance bypass.
10. Criterion 6.8, Setpoint: For the Criterion 6.8 systems listed in Table 2.2.15-1, setpoints for safety-related functions are defined, determined and implemented based on a defined setpoint methodology.	Criterion 6.8, Setpoint: Inspection(s), test(s), and/or analysis(es) for the Criterion 6.8 systems listed in Table 2.2.15-1 will be performed to verify that the setpoints for safety-related functions are defined, determined and implemented based on a defined setpoint methodology.	Criterion 6.8, Setpoint: Inspection(s), test(s), or analysis(es) report(s) for the Criterion 6.8 systems listed in Table 2.2.15-1 conclude(s) that the safety-related systems' setpoints for safety-related functions are defined, determined and implemented based on a defined setpoint methodology.

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) monitors 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. Alarms are also activated when a monitor becomes inoperative or goes upscale/downscale. 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.

(1) The functional arrangement of the PRMS is as described in the Design Description of this Subsection 2.3.1 and Figure 2.3.1-1 in conjunction with Table 2.3.1-1.

(2) a. The safety-related PRMS subsystems as identified in Table 2.3.1-1 are powered from uninterruptible safety-related power sources.

b. The safety-related PRMS subsystems identified in Table 2.3.1-1 have electrical divisional separation.

(3) The safety-related process radiation monitors listed in Table 2.3.1-1 are seismic Category I and can withstand seismic design basis loads without loss of safety function.

(4) Safety-related 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.

e. Alarms in MCR on upscale/downscale or inoperative conditions.

(5) The nonsafety-related process monitors listed in Table 2.3.1-1 are provided.

Refer to Subsection 2.2.15 for "Instrumentation and Controls Compliance with IEEE Standard 603."

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 PRMS. As appropriate, each of the ITAAC in Section 2.3.1 may be closed on a system-by-system basis throughout construction, in order that the PRMS subsystems may be placed in service.

Table 2.3.1-1

Process Radiation Monitors (Shown on Figure 2.3.1-1)

ID on Figure 2.3.1-1	Safety-Related	Description	Safety Function
1	Yes	MSL	Performs no safety-related closure function
2	Yes	Reactor Building HVAC Exhaust	Continuously monitors gross gamma quantity of radioactivity being exhausted from the contaminated area served by Reactor Building Contaminated Area (HVAC) Subsystem. The discharge point from 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, RBVS dampers are closed, and exhaust fans are stopped.
3	Yes	Refuel Handling Area HVAC Exhaust	Continuously monitors gamma radiation levels in exhaust plenum of HVAC exhaust ducts in Refuel Handling Area of 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, Reactor Building HVAC (RBVS) dampers are closed and exhaust fans are stopped.

Table 2.3.1-1

Process Radiation Monitors (Shown on Figure 2.3.1-1)

ID on Figure 2.3.1-1	Safety-Related	Description	Safety Function
4A, 4B	Yes	Control Building Air Intake 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.
5	No	TB Normal Ventilation Air HVAC	NA
6	No	TB Compartment Area Air HVAC	NA
7	No	Offgas Pre-treatment	NA
8	No	Charcoal Vault Ventilation	NA
9	No	Offgas Post-treatment	NA
10	No	TB Combined Ventilation Exhaust	NA
11	No	Liquid Radwaste Discharge	NA

Table 2.3.1-1

Process Radiation Monitors (Shown on Figure 2.3.1-1)

ID on Figure 2.3.1-1	Safety-Related	Description	Safety Function
12	Yes	LCW Drywell Sump Discharge HCW Drywell Sump Discharge	Continuously monitors gamma radiation levels in transfer pipes from Drywell Low Conductivity Waste (LCW) and High Conductivity Waste (HCW) sumps to the Radwaste System. The two monitored locations are (1) downstream of the Drywell LCW sump discharge pipe isolation valve and (2) 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.
13	No	Plant Stack	NA
14	No	Main Turbine Gland Seal Steam Condenser Exhaust	NA
15A, 15B	No	Reactor Component Cooling Water Intersystem Leakage	NA
16	No	Drywell Fission Product	NA
17	No	Radwaste Building Ventilation Exhaust	NA
18	No	FB Combined Ventilation Exhaust	NA

Table 2.3.1-1

Process Radiation Monitors (Shown on Figure 2.3.1-1)

ID on Figure 2.3.1-1	Safety-Related	Description	Safety Function
19	Yes	Isolation Condenser Vent Exhaust	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.
20	No	TSC HVAC Air Intake	NA
21	Yes	FB General Area HVAC	Monitors the gross gamma radiation level in 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.
22	Yes	FB Fuel Pool HVAC	Monitors the gamma radiation level of air exiting 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.

Table 2.3.1-1

Process Radiation Monitors (Shown on Figure 2.3.1-1)

ID on Figure 2.3.1-1	Safety-Related	Description	Safety Function
23	Yes	Containment Purge Exhaust	Monitors gross radiation level in exhaust duct leading from the primary containment. In the event of radioactive releases, monitors initiate closure of ventilation isolation dampers prior to exceeding radioactive effluent limits. In addition to closure of the RBVS isolation dampers, the RB HVAC exhaust fans are stopped.

Table 2.3.1-2

ITAAC For The Process Radiation Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the PRMS is as described in the Design Description of this Subsection 2.3.1 and Figure 2.3.1-1 in conjunction with Table 2.3.1-1.	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.	Inspection report(s) document that the as-built PRMS subsystems conform to the functional arrangement as described in the Design Description of this Subsection 2.3.1 and shown in Figure 2.3.1-1 in conjunction with Table 2.3.1-1.
2a. The safety-related PRMS subsystems as identified in Table 2.3.1-1 are powered from uninterruptible safety-related power sources. b. The safety-related PRMS subsystems identified in Table 2.3.1-1 have electrical divisional separation.	a. Inspections will be conducted to confirm that the PRMS safety-related subsystems identified in Table 2.3.1-1 are powered from uninterruptible safety-related power sources. b. Inspections of the as-built divisions will be conducted.	a. Inspection report(s) document that the safety-related PRMS subsystems identified in Table 2.3.1-1 receive electrical power from uninterruptible safety-related buses. b. Inspection report(s) document that the each subsystem division is physically separated from the other division.

Table 2.3.1-2

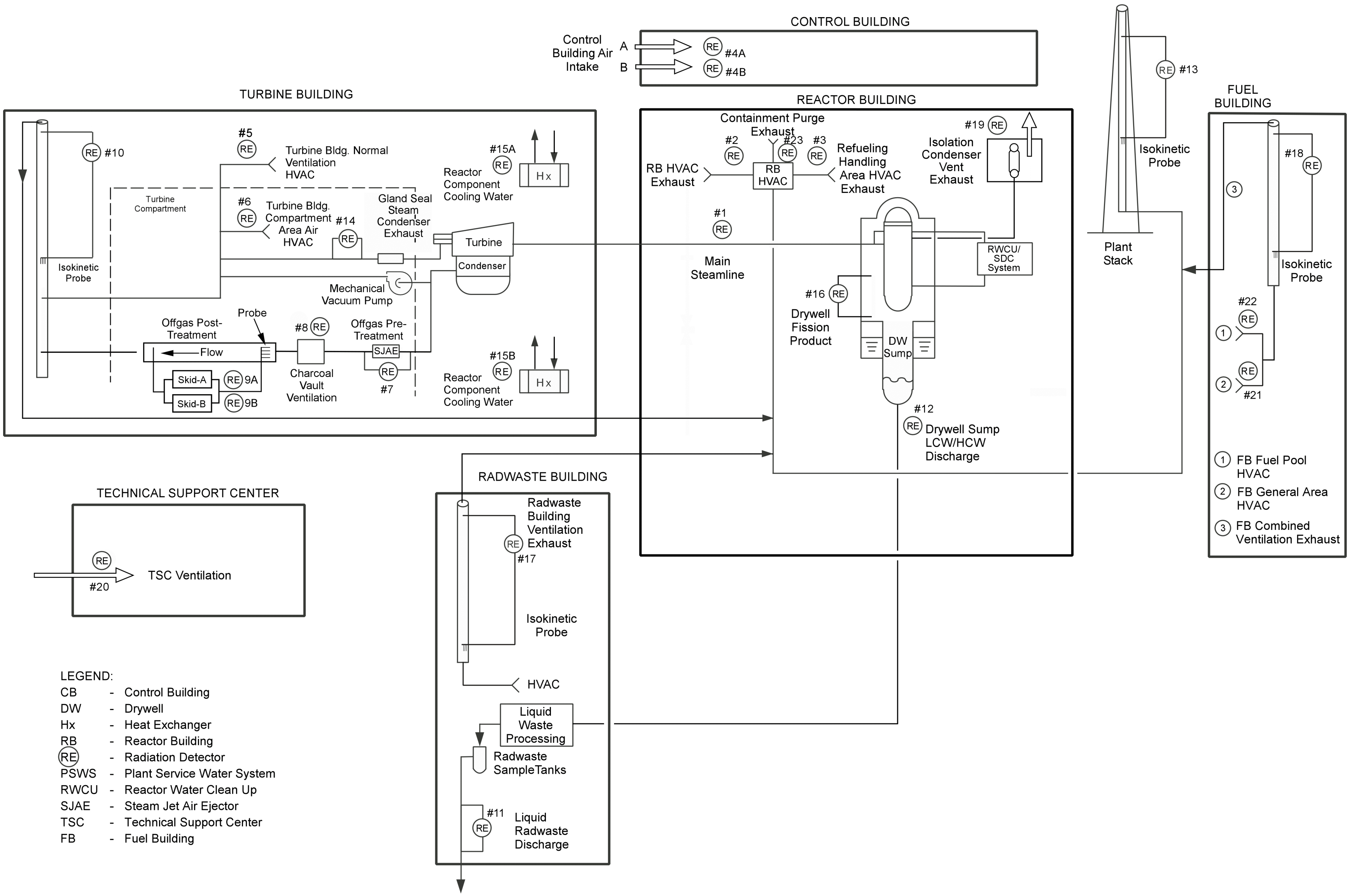
ITAAC For The Process Radiation Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The safety-related process radiation monitors listed in Table 2.3.1-1 are seismic Category I and can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I process radiation monitors identified in Table 2.3.1-1 are located on a seismic structure. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I process radiation monitors will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed process radiation monitors including anchorage are seismically bounded by the tested or analyzed conditions.	i) Inspection report(s) document that the seismic Category I process radiation monitors identified in Table 2.3.1-1 are located on a seismic structure. ii) Test/analysis reports exist and conclude that the seismic Category I process radiation monitors can withstand seismic design basis loads without loss of safety function. iii) Inspection reports exist and conclude that the as-installed process radiation monitors including anchorage are seismically bounded by the tested or analyzed conditions.

Table 2.3.1-2

ITAAC For The Process Radiation Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Safety-related 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 e. Alarms in MCR on upscale/downscale or inoperative conditions.	Tests will be conducted by simulating a high radiation signal or portable gamma source that exceeds a setpoint value that is preset for the testing. Inspections will be conducted to confirm that the as-built indication and alarm requirements are met.	Test/inspection reports exist and document that the as-built indication and alarm requirements are met.
5. The nonsafety-related process monitors listed in Table 2.3.1-1 are provided.	Inspection for the existence of the monitors will be performed.	Inspection reports document that the nonsafety-related monitors exist.



Note: See Table 2.3.1-1 for radiation detector 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.

(1) The functional arrangement (location) of the ARMS equipment is as listed on 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.

(3) Each ARM channel listed in Table 2.3.2-1 is provided with indication of radiation level.

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
Reactor Building	New Fuel Storage Pool
Reactor Building	RWCU/SDC Pump
Reactor Building	RB Sump Pumps
Reactor Building	RWCU/SDC Train A & B Heat Exchanger
Reactor Building	Equipment Hatch Pathway
Reactor Building	Personnel Hatch Pathway
Reactor Building	FMCRD HCU Area
Reactor Building	RWCU/SDC Filter Demineralizer Area (Near Equip. Hatch)
Reactor Building	Radiological Control Area Entrance
Reactor Building	Hydrogen/Oxygen Monitoring (CMS)
Reactor Building	Instrument Rack Area
Reactor Building	Fuel Transfer System (FTS) Maintenance Room (Multiple)
Reactor Building	Fuel Handling Machine (IFTS)
Reactor Building	Remote Shutdown Panel A & B Area
Fuel Building	Spent Fuel Floor
Fuel Building	Fuel Handling Machine
Fuel Building	Fuel Transfer Cask Area
Fuel Building	FAPCS Heat Exchangers
Fuel Building	FAPCS System Transfer Pumps
Fuel Building	Sump Pumps
Fuel Building	Ground Grade Access Pathway
Fuel Building	Wash Down Bay Entry Door
Fuel Building	Fuel Transfer System (FTS) Maintenance Rooms (Multiple)
Radwaste Building	Electrical Board Room
Radwaste Building	Control Room
Radwaste Building	High Activity Resin Recirculation Pump Room
Radwaste Building	High Activity Resin Transfer Pump Room
Radwaste Building	Trailer Access Area

Table 2.3.2-1

ARM Locations

Area	Description & Location
Radwaste Building	Liquid Radioactive Waste Treatment Area (Deep-Bed Demineralizer, Reverse Osmosis System, etc.)
Radwaste Building	Wet Solid Radioactive Waste Treatment Area (Dewatering Equipment, Concentrate Treatment System, etc.)
Radwaste Building	Dry Solid Waste Treatment Area (High Dose Rate Waste Storage Area, etc.)
Radwaste Building	Packaged Waste Staging Area,
Turbine Building	Main Condenser Floor Area
Turbine Building	Drain Cooler Area
Turbine Building	Offgas Sampling Area
Turbine Building	Condensate Pumps Area
Turbine Building	Low Pressure Heater Area
Turbine Building	Deaerator Area,
Turbine Building	SRV/MSIV Maintenance Area
Turbine Building	Steam Jet Air Ejector (SJAE) B Area
Turbine Building	High Pressure Heater Area
Turbine Building	Filters and Demineralizers Area
Turbine Building	Turbine Operating Floor Area
Turbine Building	Crane Travel Area (Various)
Turbine Building	Equipment Main Access Area
Turbine Building	RCCW System Area Entrance
Turbine Building	Offgas Charcoal Adsorber Room Entrance Area
Turbine Building	Backwash Transfer Pumps Entrance Area
Turbine Building	Condensate Hollow Fiber Filter Valve Room
Turbine Building	Sample Room Area
Turbine Building	Filters and Demineralizers Area
Turbine Building	Turbine Operating Floor Area
Turbine Building	Crane Travel Area (Various)
Turbine Building	Equipment Main Access Area
Turbine Building	RCCW System Area Entrance
Turbine Building	Offgas Charcoal Adsorber Room Entrance Area
Turbine Building	Backwash Transfer Pumps Entrance Area

Table 2.3.2-1

ARM Locations

Area	Description & Location
Turbine Building	Condensate Hollow Fiber Filter Valve Room
Turbine Building	Sample Room Area
Turbine Building	Condensate D/B Demineralizer Entrance Area
Turbine Building	Offgas Hydrogen Recombiner A & B
Turbine Building	Instrument Air Compressor Area
Turbine Building	MCC Water Chiller Room
Turbine Building	Turbine Building Exhaust Duct Area
Turbine Building	RCCWS Area Entrance
Control Building	Main Control Room

Table 2.3.2-2

ITAAC For The Area Radiation Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement (location) of the ARMS equipment is as listed on Table 2.3.2-1.	Inspection of the as-built system will be conducted.	Report(s) document that inspections confirm the as-built ARM system locations conform to 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.	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.	Reports(s) document tests results which 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.	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.	Reports(s) document tests results which 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 from the RPV when the reactor is isolated. Decay heat removal keeps the RPV pressure below the SRV pressure setpoint. ICS consists of four independent trains, each containing a heat exchanger that condenses steam on the tube side and transfers heat by heating and boiling water in the IC/PCC pool, which is then vented to the atmosphere.

To commence operation of an ICS train, 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 exposing cooler tube surfaces to the hot steam.

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
- MSIVs in 2 of 4 steam lines less than fully open (≤92% open) with the reactor mode switch in RUN.

The operator from the MCR can also initiate the ICS manually. A fail-open pnuematically operated condensate return bypass valve in each train opens if the control power is lost.

An in-line vessel is located on the condensate return line. The in-line vessel is located on each ICS train to provide additional condensate volume to the RPV. The amount of water volume contained in each train of ICS is at least [13.88 m^3 (490 ft^3)].

An ICS train is isolated automatically when either a high radiation level in the IC compartment is detected or excess flow is detected in the steam supply line or condensate return line.

The IC/PCC pool is divided into sub compartments that are interconnected at their lower ends to provide full use of the water inventory for heat removal by any IC.

The IC/PCC pools in combination with the Dryer/Separator pool and Reactor Well have an installed capacity that provides at least 72 hours of reactor decay heat removal capability without makeup. Two normally closed valves connect the Dryer/Separator pool and the IC/PCC expansion pool. The heat rejection process can be continued indefinitely by replenishing the IC/PCC pool inventory. An 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 shutdown reactor decay heat with 3 of 4 ICs operating and to reduce reactor pressure and temperature to safe shutdown conditions, with occasional venting of 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 the SRV setpoints.

The portions of the ICS steam supply (P-1), condensate return (P-2) and purge lines (including isolation valves), which are located inside the containment and out to and including the IC flow restrictors, are designed to ASME Code Section III, Class 1, Quality Class A. Other portions of the ICS including the vent lines 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 below the SRV septoints 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.. The ICS also provides an initial depressurization of the reactor on loss of feedwater flow.
- With an intact RCPB, the ICS in conjunction with the water 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 and Anticipated Transient Without Scram (ATWS).

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.4.1-3 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

ICS Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve.	Remotely Operated	Loss of Motive Power Position
Train A Isolation Condenser	---	---	---	---	---	---	---
IC (A) Heat Exchanger	---	Yes	Yes	No	No	N/A	N/A
Inline Vessel (A)	---	Yes	Yes	Yes	No	N/A	N/A
IC (A) Steam Supply Line	P-1(A)	Yes	Yes	Yes	No	N/A	N/A
IC (A) Steam Supply Line Isolation Valve	V-1(A)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (A) Steam Supply Line Isolation Valve	V-2(A)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (A) Condensate Return Line	P-2(A)	Yes	Yes	Yes	No	N/A	N/A
IC (A) Condensate Return Line Isolation Valve	V-3(A)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (A) Condensate Return Line Isolation Valve	V-4(A)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (A) Condensate Return Line Valve	V-5(A)	Yes	Yes	Yes	Yes	Yes	As-Is

Table 2.4.1-1

ICS Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve.	Remotely Operated	Loss of Motive Power Position
IC (A) Condensate Return Line Bypass Valve	V-6(A)	Yes	Yes	Yes	Yes	Yes	Open
Upper IC (A) Header Vent Line	---	Yes	Yes	No	No	N/A	N/A
Upper IC (A) Header Vent Line Valve	V-7(A)	Yes	Yes	No	Yes	Yes	Closed
Upper IC (A) Header Vent Line Valve	V-8(A)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (A) Header Vent Line	---	Yes	Yes	No	No	N/A	N/A
Lower IC (A) Header Vent Line Valve	V-9(A)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (A) Header Vent Line Valve	V-10(A)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (A) Header Vent Line Valve	V-11(A)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (A) Header Vent Line Valve	V-12(A)	Yes	Yes	No	Yes	Yes	Closed

Table 2.4.1-1
ICS Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve.	Remotely Operated	Loss of Motive Power Position
Train B Isolation Condenser	---	---	---	---	---	---	---
IC (B) Heat Exchanger	---	Yes	Yes	No	No	N/A	N/A
Inline Vessel (B)	---	Yes	Yes	Yes	No	N/A	N/A
IC (B) Steam Supply Line	P-1(B)	Yes	Yes	Yes	No	N/A	N/A
IC (B) Steam Supply Line Isolation Valve	V-1(B)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (B) Steam Supply Line Isolation Valve	V-2(B)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (B) Condensate Return Line	P-2(B)	Yes	Yes	Yes	No	N/A	N/A
IC (B) Condensate Return Line Isolation Valve	V-3(B)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (B) Condensate Return Line Isolation Valve	V-4(B)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (B) Condensate Return Line Valve	V-5(B)	Yes	Yes	Yes	Yes	Yes	As-Is

Table 2.4.1-1

ICS Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve.	Remotely Operated	Loss of Motive Power Position
IC (B) Condensate Return Line Bypass Valve	V-6(B)	Yes	Yes	Yes	Yes	Yes	Open
Upper IC (B) Header Vent Line	---	Yes	Yes	No	No	N/A	N/A
Upper IC (B) Header Vent Line Valve	V-7(B)	Yes	Yes	No	Yes	Yes	Closed
Upper IC (B) Header Vent Line Valve	V-8(B)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (B) Header Vent Line	---	Yes	Yes	No	No	N/A	N/A
Lower IC (B) Header Vent Line Valve	V-9(B)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (B) Header Vent Line Valve	V-10(B)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (B) Header Vent Line Valve	V-11(B)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (B) Header Vent Line Valve	V-12(B)	Yes	Yes	No	Yes	Yes	Closed

Table 2.4.1-1

ICS Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve.	Remotely Operated	Loss of Motive Power Position
Train C Isolation Condenser	---	---	---	---	---	---	---
IC (C) Heat Exchanger	---	Yes	Yes	No	No	N/A	N/A
Inline Vessel (C)	---	Yes	Yes	Yes	No	N/A	N/A
IC (C) Steam Supply Line	P-1(C)	Yes	Yes	Yes	No	N/A	N/A
IC (C) Steam Supply Line Isolation Valve	V-1(C)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (C) Steam Supply Line Isolation Valve	V-2(C)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (C) Condensate Return Line	P-2(C)	Yes	Yes	Yes	No	N/A	N/A
IC (C) Condensate Return Line Isolation Valve	V-3(C)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (C) Condensate Return Line Isolation Valve	V-4(C)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (C) Condensate Return Line Valve	V-5(C)	Yes	Yes	Yes	Yes	Yes	As-Is

Table 2.4.1-1

ICS Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve.	Remotely Operated	Loss of Motive Power Position
IC (C) Condensate Return Line Bypass Valve	V-6(C)	Yes	Yes	Yes	Yes	Yes	Open
Upper IC (C) Header Vent Line	---	Yes	Yes	No	No	N/A	N/A
Upper IC (C) Header Vent Line Valve	V-7(C)	Yes	Yes	No	Yes	Yes	Closed
Upper IC (C) Header Vent Line Valve	V-8(C)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (C) Header Vent Line	---	Yes	Yes	No	No	N/A	N/A
Lower IC (C) Header Vent Line Valve	V-9(C)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (C) Header Vent Line Valve	V-10(C)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (C) Header Vent Line Valve	V-11(C)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (C) Header Vent Line Valve	V-12(C)	Yes	Yes	No	Yes	Yes	Closed

Table 2.4.1-1

ICS Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve.	Remotely Operated	Loss of Motive Power Position
Train D Isolation Condenser	---	---	---	---	---	---	---
IC (D) Heat Exchanger	---	Yes	Yes	No	No	N/A	N/A
Inline Vessel (D)	---	Yes	Yes	Yes	No	N/A	N/A
IC (D) Steam Supply Line	P-1(D)	Yes	Yes	Yes	No	N/A	N/A
IC (D) Steam Supply Line Isolation Valve	V-1(D)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (D) Steam Supply Line Isolation Valve	V-2(D)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (D) Condensate Return Line	P-2(D)	Yes	Yes	Yes	No	N/A	N/A
IC (D) Condensate Return Line Isolation Valve	V-3(D)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (D) Condensate Return Line Isolation Valve	V-4(D)	Yes	Yes	Yes	Yes	Yes	As-Is
IC (D) Condensate Return Line Valve	V-5(D)	Yes	Yes	Yes	Yes	Yes	As-Is

Table 2.4.1-1

ICS Mechanical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve.	Remotely Operated	Loss of Motive Power Position
IC (D) Condensate Return Line Bypass Valve	V-6(D)	Yes	Yes	Yes	Yes	Yes	Open
Upper IC (D) Header Vent Line	---	Yes	Yes	No	No	N/A	N/A
Upper IC (D) Header Vent Line Valve	V-7(D)	Yes	Yes	No	Yes	Yes	Closed
Upper IC (D) Header Vent Line Valve	V-8(D)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (D) Header Vent Line	---	Yes	Yes	No	No	N/A	N/A
Lower IC (D) Header Vent Line Valve	V-9(D)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (D) Header Vent Line Valve	V-10(D)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (D) Header Vent Line Valve	V-11(D)	Yes	Yes	No	Yes	Yes	Closed
Lower IC (D) Header Vent Line Valve	V-12(D)	Yes	Yes	No	Yes	Yes	Closed

Note: N/A = Not Applicable

Table 2.4.1-2

ICS Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	Control Q-DCIS / DPS See Note 1	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated Valve	Containment Isolation Valve Actuator
Train A Isolation Condenser	---	---	---	---	---	---	---	---
IC (A) Steam Supply Line Isolation Valve	V-1(A)	***	Yes	Yes	Position	Close	Yes	Yes
IC (A) Steam Supply Line Isolation Valve	V-2(A)	***	Yes	Yes	Position	Close	Yes	Yes
IC (A) Condensate Return Line Isolation Valve	V-3(A)	***	Yes	Yes	Position	Close	Yes	Yes
IC (A) Condensate Return Line Isolation Valve	V-4(A)	***	Yes	Yes	Position	Close	Yes	Yes
IC (A) Condensate Return Line Valve	V-5(A)	***	Yes	Yes	Position	Open / Close	Yes	No
IC (A) Condensate Return Line Bypass Valve	V-6(A)	***	Yes	Yes	Position	Open / Close	Yes	No

Table 2.4.1-2

ICS Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	Control Q-DCIS / DPS See Note 1	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated Valve	Containment Isolation Valve Actuator
Upper IC (A) Header Vent Line Valve	V-7(A)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Upper IC (A) Header Vent Line Valve	V-8(A)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (A) Header Vent Line Valve	V-9(A)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (A) Header Vent Line Valve	V-10(A)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (A) Header Vent Line Valve	V-11(A)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (A) Header Vent Line Valve	V-12(A)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Train B Isolation Condenser	---	---	---	---	---	---	---	---
IC (B) Steam Supply Line Isolation Valve	V-1(B)	***	Yes	Yes	Position	Close	Yes	Yes
IC (B) Steam Supply Line Isolation Valve	V-2(B)	***	Yes	Yes	Position	Close	Yes	Yes

Table 2.4.1-2

ICS Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	Control Q-DCIS / DPS See Note 1	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated Valve	Containment Isolation Valve Actuator
IC (B) Condensate Return Line Isolation Valve	V-3(B)	***	Yes	Yes	Position	Close	Yes	Yes
IC (B) Condensate Return Line Isolation Valve	V-4(B)	***	Yes	Yes	Position	Close	Yes	Yes
IC (B) Condensate Return Line Valve	V-5(B)	***	Yes	Yes	Position	N/A	Yes	No
IC (B) Condensate Return Line Bypass Valve	V-6(B)	***	Yes	Yes	Position	Open / Close	Yes	No
Upper IC (B) Header Vent Line Valve	V-7(B)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Upper IC (B) Header Vent Line Valve	V-8(B)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (B) Header Vent Line Valve	V-9(B)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (B) Header Vent Line Valve	V-10(B)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No

Table 2.4.1-2

ICS Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	Control Q-DCIS / DPS See Note 1	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated Valve	Containment Isolation Valve Actuator
Lower IC (B) Header Vent Line Valve	V-11(B)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (B) Header Vent Line Valve	V-12(B)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Train C Isolation Condenser	---	---	---	---	---	---	---	---
IC (C) Steam Supply Line Isolation Valve	V-1(C)	***	Yes	Yes	Position	Close	Yes	Yes
IC (C) Steam Supply Line Isolation Valve	V-2(C)	***	Yes	Yes	Position	Close	Yes	Yes
IC (C) Condensate Return Line Isolation Valve	V-3(C)	***	Yes	Yes	Position	Close	Yes	Yes
IC (C) Condensate Return Line Isolation Valve	V-4(C)	***	Yes	Yes	Position	Close	Yes	Yes
IC (C) Condensate Return Line Valve	V-5(C)	***	Yes	Yes	Position	Open / Close	Yes	No

Table 2.4.1-2

ICS Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	Control Q-DCIS / DPS See Note 1	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated Valve	Containment Isolation Valve Actuator
IC (C) Condensate Return Line Bypass Valve	V-6(C)	***	Yes	Yes	Position	Open / Close	Yes	No
Upper IC (C) Header Vent Line Valve	V-7(C)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Upper IC (C) Header Vent Line Valve	V-8(C)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (C) Header Vent Line Valve	V-9(C)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (C) Header Vent Line Valve	V-10(C)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (C) Header Vent Line Valve	V-11(C)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (C) Header Vent Line Valve	V-12(C)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Train C Isolation Condenser	---	---	---	---	---	---	---	---
IC (D) Steam Supply Line Isolation Valve	V-1(D)	***	Yes	Yes	Position	Close	Yes	Yes

Table 2.4.1-2

ICS Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.1-1	Control Q-DCIS / DPS See Note 1	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated Valve	Containment Isolation Valve Actuator
IC (D) Steam Supply Line Isolation Valve	V-2(D)	***	Yes	Yes	Position	Close	Yes	Yes
IC (D) Condensate Return Line Isolation Valve	V-3(D)	***	Yes	Yes	Position	Close	Yes	Yes
IC (D) Condensate Return Line Isolation Valve	V-4(D)	***	Yes	Yes	Position	Close	Yes	Yes
IC (D) Condensate Return Line Valve	V-5(D)	***	Yes	Yes	Position	Open / Close	Yes	No
IC (D) Condensate Return Line Bypass Valve	V-6(D)	***	Yes	Yes	Position	Open / Close	Yes	No
Upper IC (D) Header Vent Line Valve	V-7(D)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Upper IC (D) Header Vent Line Valve	V-8(D)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (D) Header Vent Line Valve	V-9(D)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No

Table 2.4.1-2

ICS Electrical Equipment

Equipment Name (Description)	**Equipment Identifier See Figure 2.4.1-1**	**Control Q-DCIS / DPS See Note 1**	**Seismic Category I**	**Safety-Related**	**Safety-Related Display**	**Active Function**	**Remotely Operated Valve**	**Containment Isolation Valve Actuator**
Lower IC (D) Header Vent Line Valve	V-10(D)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (D) Header Vent Line Valve	V-11(D)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No
Lower IC (D) Header Vent Line Valve	V-12(D)	Yes/No	Yes	Yes	Position	Open / Close	Yes	No

Notes

*** The following valve pairs must have a total of four control inputs to the pair; V-1 & V-2, V-3 & V-4, and V-5 & V-6.

The minimum control inputs for the pairs must include 2 different divisions to one of the valve pair and third different division and DPS to the other valve pair. The design is such that any combination of 2 of 4 divisions or DPS can cause the desired action to occur.

N/A = Not Applicable

Table 2.4.1-3

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the IC System is as described in the Design Description of this Section 2.4.1, Tables 2.4.1-1 and 2, and Figure 2.4.1-1.	Inspection of the as-built system will be performed.	Report(s) document that the as-built IC System conforms with the functional arrangement described in the Design Description of this Section 2.4.1, Tables 2.4.1-1 and 2, and Figure 2.4.1-1.
2a. The components identified in Table 2.4.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	Report(s) document that the ASME Code Section III design reports exist for the as-built components identified in Table 2.4.1-1 as ASME Code Section III.
b. The piping identified in Table 2.4.1-1 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	Report(s) document that the ASME code Section III design reports exist for the as-built piping identified in Table 2.4.1-1 as ASME Code Section III.
3a. Pressure boundary welds in components identified in Table 2.4.1-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	Report(s) document that a report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
b. Pressure boundary welds in piping identified in Table 2.4.1-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	Report(s) document that a report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.

Table 2.4.1-3

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4a. The components identified in Table 2.4.1-1 as ASME Code Section III retain their pressure boundary integrity at under internal pressures that will be experienced during service.	A hydrostatic test will be conducted on those code components of the IC System required to be hydrostatically tested by the ASME code.	Report(s) document that the results of the hydrostatic test of the ASME Code components of the IC System conform with the requirements in the ASME Code, Section III.
b. The piping identified in Table 2.4.1-1 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be conducted on those code components of the System required to be hydrostatically tested by the ASME code	Report(s) document that the results of the hydrostatic test of the ASME Code components of the System conform with the requirements in the ASME Code, Section III.

Table 2.4.1-3

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5a. The seismic Category I equipment identified in Tables 2.4.1-1 and 2 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Tables 2.4.1-1 and 2 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	Report(s) document that: i) The seismic Category I equipment identified in Table Tables 2.4.1-1 and 2 is located in a seismic structure. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
b. Each of the lines identified in Table 2.4.1-1 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	Report(s) document that a report exists and concludes that each of the as-built lines identified in Table 2.4.1-1 for which functional capability is required meets the requirements for functional capability.

Table 2.4.1-3

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6a. Each of the IC System divisions (or safety-related loads/components) identified in Table 2.4.1-2 is powered from its respective safety-related division.	Testing will be performed on the IC System by providing a [simulated] test signal in only one safety-related division at a time.	Report(s) document that a [simulated] test signal exists in the safety-related division (or at the equipment identified in Table 2.4.1-2 powered from the safety-related division) under test in the IC System.
b. In the IC System, independence is provided between safety-related divisions, and between safety-related divisions and non-safety related equipment.	i) Tests will be performed on the IC System by providing a test signal in only one safety-related division at a time. ii) Inspection of the as-installed safety-related divisions in the IC System will be performed.	Report(s) document that: i) The test signal exists only in the safety-related Division under test in the System. ii) In the IC System, physical separation or electrical isolation exists between these safety-related divisions. Physical separation or electrical isolation exists between safety-related divisions and non-safety related equipment.
7. Each mechanical train of the IC System is physically separated from the other trains. Physical separation is not required in the Primary Containment.	Inspections of the as-built IC System will be performed.	Report(s) document that the each mechanical train of the IC System is physically separated from other mechanical trains of the system by structural and/or fire barriers.

Table 2.4.1-3

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Control Room displays provided for the IC System are defined in Table 2.4.1-2	Inspections will be performed on the Control Room displays for the IC System.	Report(s) document that displays exist or can be retrieved in the Control Room as defined in Table 2.4.1-2
9. Re-positionable (NOT squib) valves designated in Table 2.4.1-1 as having an active safety-related function open, close, or both open and also close under differential pressure, fluid flow, and temperature conditions.	Tests of installed valves will be performed for opening, closing, or both opening and also closing under system preoperational differential pressure, fluid flow, and temperature conditions.	Report(s) document that, upon receipt of the actuating signal, each valve opens, closes, or both opens and also closes, depending upon the valve's safety function.
10. The pneumatically operated valve(s) designated in Table 2.4.1-1 fail in the mode listed if either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve(s) is lost.	Tests will be conducted on the as-built valve(s).	Report(s) document that the pneumatically operated valve(s) identified in Table 2.4.1-1 fail in the listed mode when either electric power to the valve actuating solenoid is lost, or pneumatic pressure to the valve(s) is lost.
11. The equipment qualification of IC system components is addressed in Tier 1 Section 3.8.	See Tier 1 Section 3.8	See Tier 1 Section 3.8
12. The containment isolation portions of the IC System are addressed in Tier 1 Subsection 2.15.1.	See Tier 1 Subsection 2.15.1.	See Tier 1 Subsection 2.15.1.
13. Each condensate return valve (V-5 and V-6) shown on Figure 2.4.1-1 will open to initiate the ICS.	Opening and/or closing tests of valves will be conducted under pre-operational differential pressure, fluid flow and temperature conditions.	Test reports document that each condensate return valve opens under pre-operational differential pressure, fluid flow, and temperature conditions to initiate the IC system.

Table 2.4.1-3

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
14. The normally open ICS isolation valves (V-1, V-2, V-3 and V-4) in the steam supply and condensate return lines close automatically on receipt of high vent line radiation from the Process Radiation Monitoring System (PRMS).	An isolation valve closure test will be performed using simulated signals.	Test report(s) document that the ICS isolation valves close upon receipt of signals from the PRMS.
15. The normally open ICS isolation valves (V-1, V-2, V-3 and V-4) in the steam supply and condensate return lines close automatically on receipt of signals from the LD&IS.	An isolation valve closure test will be performed using simulated signals.	Test report(s) document that the ICS isolation valves close upon receipt of signals from the LD&IS.

Table 2.4.1-3

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
16. Each ICS train normally closed condensate return valve (V-5) 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 • MSIVs in 2 of 4 steam lines less than fully open (≤92%) with the reactor mode switch in RUN.	Valve opening tests will be performed using simulated automatic actuation signals.	Test report(s) document that the condensate return valves open upon receipt of automatic actuation signals.

Table 2.4.1-3

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
17. Each ICS train normally closed condensate return bypass valve (V-6) 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 less than fully open (≤92%) with the reactor mode switch in RUN.	Valve opening tests will be performed using simulated automatic actuation signals.	Test report(s) document that the condensate return valves open upon receipt of automatic actuation signals.
18. The two-series, solenoid-operated bottom vent line valves (V-9, and V-10) open on high RPV pressure after time delay following condensate return or condensate bypass valve opening signals.	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.	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 2.4.1-3

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
19. The three vent lines with two-series, solenoid-operated top and bottom vent line valves (V-7 & V-8: V-9 & V-10; V-11 & V-12) open on manual actuation only if condensate return or condensate bypass valve is not closed.	A test(s) will be performed that manually opens the vent valves during pre-operational testing following condensate return or condensate bypass valve opening signals.	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.
20. The accumulator for the pneumatic isolation valves (V-1, V-2, V-3 and V-4) 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.	An analysis and/or test will be performed to demonstrate the capacity of the isolation valve accumulators.	Test report(s) document that isolation valve accumulators have the capacity to close the valves three times with the drywell pressure at the design pressure.
21. Upon loss of pneumatic pressure to the condensate bypass valve (V-6) (fail open), the valve strokes to the fully open position.	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.	The condensate bypass valve fully opens when pneumatic pressure is removed from the condensate bypass valve.
22. Each ICS train minimum heat removal capacity is 33.75 MWt with reactor at or above normal operating pressure.	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.	Test and/or analysis report(s) document that the ICS train unit heat removal capacity is greater than or equal to the committed value for the reactor at or above normal operating pressure.
23. Each ICS train provides at least [13.88 m^3 (490 ft^3)] drainable liquid volume available for return to the RPV.	An analysis will be performed for the as-built isolation condenser system.	An analysis exists and demonstrates that the as-built ICS trains provides the required volume of liquid available for return to the RPV.

Table 2.4.1-3

ITAAC For The Isolation Condenser System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
24. The Dryer/Separator Pool and Reactor Well provide sufficient makeup water volume to the IC/PCC expansion pool to support operation of the ICS and PCCS for the first 72 hours..	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 as-built Dryer/Separator Pool and Reactor Well provide sufficient makeup water volume to the IC/PCC expansion pool on a low water signal in the initial 72 hours of a LOCA.	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). An analysis exists and demonstrates that the as-built Dryer/Separator Pool and Reactor Well provide sufficient makeup water volume to the IC/PCC expansion pool on a low water signal in the initial 72 hours of a LOCA.

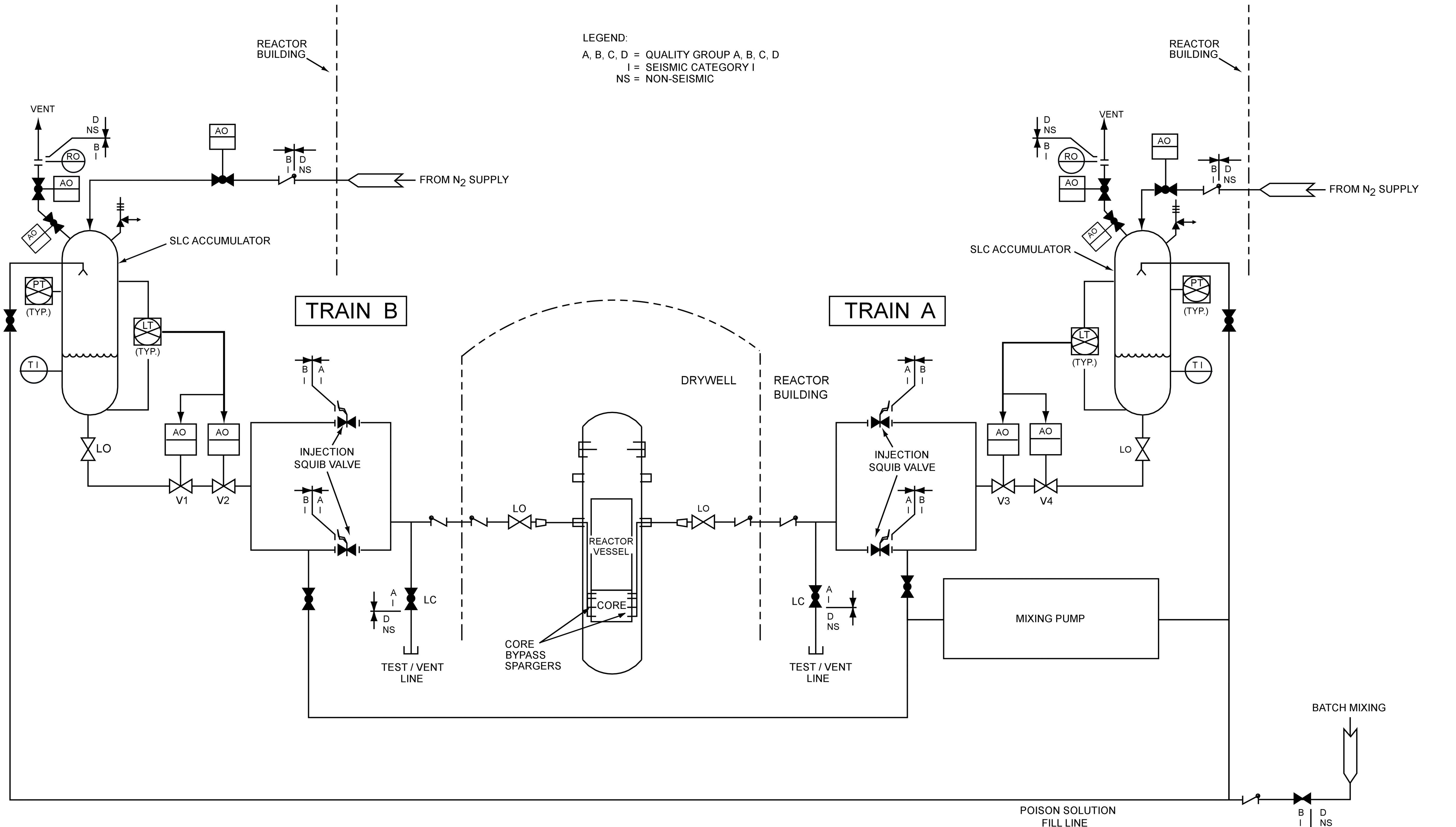


Figure 2.4.1-1. Isolation Condenser System Schematic

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.

(1) The functional arrangement of the GDCS is as listed in Table 2.4.2-1 and shown on Figure 2.4.2-1.

(2) a. Components identified in Table 2.4.2-1 as ASME Code Section III are designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III requirements.

b. Piping identified in Table 2.4.2-1 as ASME Code Section III are designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III requirements.

(3) a. Pressure boundary welds in components identified in Table 2.4.2-1 as ASME Code Section III meet ASME Code Section III requirements.

b. Pressure boundary welds in piping identified in Table 2.4.2-1 as ASME Code Section III meet ASME Code Section III requirements.

(4) a. Each component identified in Table 2.4.2-1 as ASME Code Section III retains its pressure boundary integrity at under internal pressures that will be experienced during service.

b. The piping identified in Table 2.4.2-1 as ASME Code Section III retains its pressure boundary integrity at design pressure.

(5) a. The seismic Category I equipment identified in Table 2.4.2-1 can withstand seismic design basis loads without loss of safety function.

b. Each of the lines identified in Table 2.4.2-1 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

(6) The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations and emergency procedure guidelines, is available in the main control room.

(7) The equipment qualification of GDCS components is addressed in Tier 1 Section 3.8.

(8) a. The GDCS injections lines provide sufficient flow to maintain water coverage one meter above TAF for 72 hours following a design basis LOCA.

b. The GDCS equalizing lines provide sufficient flow to maintain water coverage one meter above TAF for 72 hours following a design basis LOCA.

(9) The GDCS squib valve used in the injection and equalization open as designed.

(10) 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.

(11) Control Room indications and controls are provided for the GDCS.

(12) GDCS squib valves maintain RPV backflow leak tightness and maintain reactor coolant pressure boundary integrity during normal plant operation.

(13) Each GDCS injection line includes a nozzle flow limiter to limit break size.

(14) Each GDCS equalizing line includes a nozzle flow limiter to limit break size.

(15) Each of the GDCS divisions is powered from their respective safety-related power divisions.

(16) Each mechanical division of the GDCS outside the drywell 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.

(17) The GDCS pools A, B/C, and D are sized to hold a minimum drainable water volume.

(18) The GDCS pools A, B/C, and D are of sized for holding a specified minimum water level.

(19) The minimum elevation change between minimum water level of GDCS pools and the centerline of GDCS injection line nozzles is sufficient to provide gravity-driven flow.

(20) The minimum drainable volume from the suppression pool to the RPV is sufficient to meet long-term post-LOCA core cooling requirements.

(21) The long-term GDCS minimum equalizing driving head is based on RPV Level 0.5.

(22) The GDCS Deluge squib valves open as designed.

Refer to Subsection 2.2.15 for “Instrumentation and Controls Compliance with IEEE Standard 603.”

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.4.2-3 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

GDCS Mechanical Equipment[1]

Equipment Name (Description)	Equipment Identifier See Figure 2.4.2-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
GDCS Pool Supply Line to V-2	P-1(A)	Yes	Yes	No	-	-	-	-
GDCS Pool Injection Line Check Valve	V-1(A)	Yes	Yes	No	No	No	-	-
GDCS Pool Injection Line Squib Valve	V-2(A)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Pool Injection Line Squib Valve	V-3(A)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Pool Injection Line Check Valve	V-4(A)	Yes	Yes	No	No	No	-	-
GDCS Injection Line from V-2 (V-3) to RPV	P-2(A)	Yes	Yes	Yes	-	-	-	-
GDCS Suppression Pool Supply line to V-5	P-5(A)	Yes	Yes	No	-	-	-	-
GDCS Suppression Pool Injection Line Check Valve	V-6(A)	Yes	Yes	No	No	No	-	-
GDCS Suppression Pool Injection Line Squib Valve	V-5(A)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Suppression Pool Equalizing Line from V-5 to RPV	P-3(A)	Yes	Yes	Yes	-	-	-	-
GDCS Deluge Line	P-4(A)	Yes	Yes	No	-	-	-	-
GDCS Deluge Line Squib Valve	V-7(A)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Deluge Line Squib Valve	V-8(A)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Deluge Line Squib Valve	V-9(A)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Pool Supply Line to V-2	P-1(B)	Yes	Yes	No	-	-	-	-
GDCS Pool Injection Line Check Valve	V-1(B)	Yes	Yes	No	No	No	-	-

[1] A "-" means not applicable.

Table 2.4.2-1

GDCS Mechanical Equipment[1]

Equipment Name (Description)	Equipment Identifier See Figure 2.4.2-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
GDCS Pool Injection Line Squib Valve	V-2(B)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Pool Injection Line Squib Valve	V-3(B)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Pool Injection Line Check Valve	V-4(B)	Yes	Yes	No	No	No	-	-
GDCS Injection Line from V-2 (V-3) to RPV	P-2(B)	Yes	Yes	Yes	-	-	-	-
GDCS Suppression Pool Supply line to V-5	P-5(B)	Yes	Yes	No	-	-	-	-
GDCS Suppression Pool Injection Line Check Valve	V-6(B)	Yes	Yes	No	No	No	-	-
GDCS Suppression Pool Injection Line Squib Valve	V-5(B)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Suppression Pool Equalizing Line from V-5 to RPV	P-3(B)	Yes	Yes	Yes	-	-	-	-
GDCS Deluge Line	P-4(B)	Yes	Yes	No	-	-	-	-
GDCS Deluge Line Squib Valve	V-7(B)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Deluge Line Squib Valve	V-8(B)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Deluge Line Squib Valve	V-9(B)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Pool Supply Line to V-2	P-1(C)	Yes	Yes	No	-	-	-	-
GDCS Pool Injection Line Check Valve	V-1(C)	Yes	Yes	No	No	No	-	-
GDCS Pool Injection Line Squib Valve	V-2(C)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Pool Injection Line Squib Valve	V-3(C)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Pool Injection Line Check Valve	V-4(C)	Yes	Yes	No	No	No	-	-
GDCS Injection Line from V-2 (V-3) to RPV	P-2(C)	Yes	Yes	Yes	-	-	-	-

Table 2.4.2-1

GDCS Mechanical Equipment[1]

Equipment Name (Description)	Equipment Identifier See Figure 2.4.2-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
GDCS Suppression Pool Supply line to V-5	P-5(C)	Yes	Yes	No	-	-	-	-
GDCS Suppression Pool Injection Line Check Valve	V-6(C)	Yes	Yes	No	No	No	-	-
GDCS Suppression Pool Injection Line Squib Valve	V-5(C)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Suppression Pool Equalizing Line from V-5 to RPV	P-3(C)	Yes	Yes	Yes	-	-	-	-
GDCS Deluge Line	P-4(C)	Yes	Yes	No	-	-	-	-
GDCS Deluge Line Squib Valve	V-7(C)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Deluge Line Squib Valve	V-8(C)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Deluge Line Squib Valve	V-9(C)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Pool Supply Line to V-2	P-1(D)	Yes	Yes	No	-	-	-	-
GDCS Pool Injection Line Check Valve	V-1(D)	Yes	Yes	No	No	No	-	-
GDCS Pool Injection Line Squib Valve	V-2(D)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Pool Injection Line Squib Valve	V-3(D)	Yes	Yes	Yes	No	Yes	As-Is	Yes
GDCS Pool Injection Line Check Valve	V-4(D)	Yes	Yes	No	No	No	-	-
GDCS Injection Line from V-2 (V-3) to RPV	P-2(D)	Yes	Yes	Yes	-	-	-	-
GDCS Suppression Pool Supply line to V-5	P-5(D)	Yes	Yes	No	-	-	-	-
GDCS Suppression Pool Injection Line Check Valve	V-6(D)	Yes	Yes	No	No	No	-	-
GDCS Suppression Pool Injection Line Squib Valve	V-5(D)	Yes	Yes	Yes	No	Yes	As-Is	Yes

Table 2.4.2-1

GDCS Mechanical Equipment[1]

Equipment Name (Description)	Equipment Identifier See Figure 2.4.2-1	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
GDCS Suppression Pool Equalizing Line from V-5 to RPV	P-3(D)	Yes	Yes	No	-	-	-	-
GDCS Deluge Line	P-4(D)	Yes	Yes	No	-	-	-	-
GDCS Deluge Line Squib Valve	V-7(D)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Deluge Line Squib Valve	V-8(D)	Yes	Yes	No	No	Yes	As-Is	Yes
GDCS Deluge Line Squib Valve	V-9(D)	Yes	Yes	No	No	Yes	As-Is	Yes

Table 2.4.2-2

Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.2-1	Control Q-DCIS/ DPS	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated	Containment Isolation Valve Actuator
GDCS Injection Line Check Valve	V-1(A)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Injection Line Squib Valve	V-2(A)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Injection Line Squib Valve	V-3(A)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Injection Line Check Valve	V-4(A)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Suppression Pool Injection Line Squib Valve	V-5(A)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Suppression Pool Injection Line Check Valve	V-6(A)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Deluge Line Squib Valve	V-7(A)	-	Yes	No	Yes	Open	Yes	No
GDCS Deluge Line Squib Valve	V-8(A)	-	Yes	No	Yes	Open	Yes	No
GDCS Deluge Line Squib Valve	V-9(A)	-	Yes	No	Yes	Open	Yes	No
GDCS Injection Line Check Valve	V-1(B)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Injection Line Squib Valve	V-2(B)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Injection Line Squib Valve	V-3(B)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Injection Line Check Valve	V-4(B)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Suppression Pool Injection Line Squib Valve	V-5(B)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Suppression Pool Injection Line Check Valve	V-6(B)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Deluge Line Squib Valve	V-7(B)	-	Yes	No	Yes	Open	Yes	No
GDCS Deluge Line Squib Valve	V-8(B)	-	Yes	No	Yes	Open	Yes	No
GDCS Deluge Line Squib Valve	V-9(B)	-	Yes	No	Yes	Open	Yes	No
GDCS Injection Line Check Valve	V-1(C)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Injection Line Squib Valve	V-2(C)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Injection Line Squib Valve	V-3(C)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Injection Line Check Valve	V-4(C)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Suppression Pool Injection Line Squib Valve	V-5(C)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No

Table 2.4.2-2

Electrical Equipment

Equipment Name (Description)	Equipment Identifier See Figure 2.4.2-1	Control Q-DCIS/ DPS	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated	Containment Isolation Valve Actuator
GDCS Suppression Pool Injection Line Check Valve	V-6(C)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Deluge Line Squib Valve	V-7(C)	-	Yes	No	Yes	Open	Yes	No
GDCS Deluge Line Squib Valve	V-8(C)	-	Yes	No	Yes	Open	Yes	No
GDCS Deluge Line Squib Valve	V-9(C)	-	Yes	No	Yes	Open	Yes	No
GDCS Injection Line Check Valve	V-1(D)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Injection Line Squib Valve	V-2(D)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Injection Line Squib Valve	V-3(D)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Injection Line Check Valve	V-4(D)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Suppression Pool Injection Line Squib Valve	V-5(D)	Yes / Yes	Yes	Yes	Yes	Open	Yes	No
GDCS Suppression Pool Injection Line Check Valve	V-6(D)	-	Yes	Yes	Yes	Open/Close	No	No
GDCS Deluge Line Squib Valve	V-7(D)	-	Yes	No	Yes	Open	Yes	No
GDCS Deluge Line Squib Valve	V-8(D)	-	Yes	No	Yes	Open	Yes	No
GDCS Deluge Line Squib Valve	V-9(D)	-	Yes	No	Yes	Open	Yes	No

Table 2.4.2-3

ITAAC For The Gravity-Driven Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the GDCS is as listed in Table 2.4.2-1 and shown on Figure 2.4.2-1.	Inspections of the as-built system will be conducted.	The as-built GDCS conforms to the functional arrangement as listed in Table 2.4.2-1 and shown in Figure 2.4.2-1.
2a. Components identified in Table 2.4.2-1 as ASME Code Section III are designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III requirements.	Inspections will be conducted of the as-built components as documented in the ASME design reports.	Inspections confirm that the ASME Code components are designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III.
b. Piping identified in Table 2.4.2-1 as ASME Code Section III are designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III requirements.	Inspections will be conducted of the as-built piping as documented in the ASME design reports.	Inspections confirm that the ASME Code piping is designed, fabricated, installed, and inspected in accordance with the ASME Code, Section III.
3a. Pressure boundary welds in components identified in Table 2.4.2-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.

Table 2.4.2-3

ITAAC For The Gravity-Driven Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
b. Pressure boundary welds in piping identified in Table 2.4.2-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4a. Each component identified in Table 2.4.2-1 as ASME Code Section III retains its pressure boundary integrity at under internal pressures that will be experienced during service.	A hydrostatic test will be conducted on those code components of the GDCS required to be hydrostatically tested by the ASME code.	Report(s) document that the results of the hydrostatic test of the ASME Code components of the GDCS conform to the requirements in the ASME Code, Section III.
b. The piping identified in Table 2.4.2-1 as ASME Code Section III retains its pressure boundary integrity at design pressure.	A hydrostatic test will be conducted on those code components of the GDCS required to be hydrostatically tested by the ASME code.	Report(s) document that the results of the hydrostatic test of the ASME Code components of the GDCS conform to the requirements in the ASME Code, Section III.

Table 2.4.2-3

ITAAC For The Gravity-Driven Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5a. The seismic Category I equipment identified in Table 2.4.2-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.4.2-1. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) Report(s) document that the seismic Category I equipment identified in Table 2.4.2-1 is located on a seismic structure. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
b. Each of the lines identified in Table 2.4.2-1 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	A report exists and concludes that each of the as-built lines identified in Table 2.4.2-1 for which functional capability is required meets the requirements for functional capability.

Table 2.4.2-3

ITAAC For The Gravity-Driven Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations and emergency procedure guidelines, is available in the main control room.	Inspections will be performed on the main control room alarms, displays, and/or controls for the GDCS.	Report(s) document that alarms, displays, and/or controls exist or can be retrieved in the main control room.
7. The equipment qualification of GDCS components is addressed in Tier 1 Section 3.8.	See Tier 1 Section 3.8.	See Tier 1 Section 3.8.
8a. The GDCS injections lines provide sufficient flow to maintain water coverage one meter above TAF for 72 hours following a design basis LOCA.	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.	An analysis exists that demonstrates that the observed flow rate, in conjunction with vessel depressurization and other modes of GDCS operation, maintains water coverage one meter above TAF for 72 hours following the design basis LOCA.

Table 2.4.2-3

ITAAC For The Gravity-Driven Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
b. The GDCS equalizing lines provide sufficient flow to maintain water coverage one meter above TAF for 72 hours following a design basis LOCA.	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.	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.
9. The GDCS squib valve used in the injection and equalization open as designed.	A vendor type test will be performed on a squib valve to open as designed.	Records of vendor type test concludes GDCS squib valves used in the injection and equalization open as designed.
10. 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.	Type tests of valves for opening, closing, or both opening and also closing, will be conducted.	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.
11. Control Room indications and controls are provided for the GDCS.	Inspections will be performed on the Control Room indications and controls for the GDCS.	Indications and controls exist or can be retrieved in the control room as defined in Subsection 2.4.2.
12. GDCS squib valves maintain RPV backflow leak tightness and maintain reactor coolant pressure boundary integrity during normal plant operation.	A test will be performed to demonstrate the squib valves are leak tight during normal plant conditions.	Testing concludes GDCS squib valves have zero leakage at normal plant operation pressure

Table 2.4.2-3

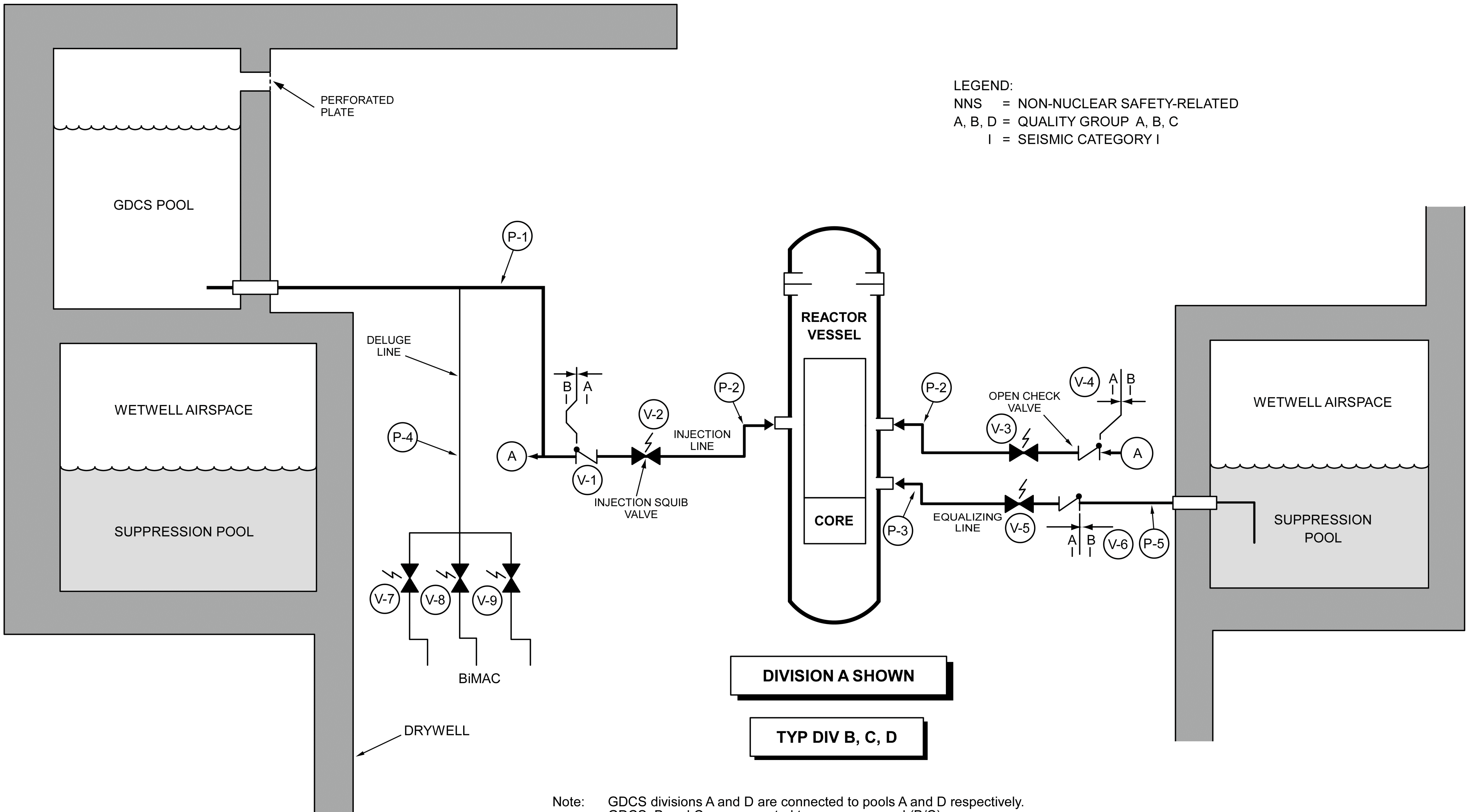
ITAAC For The Gravity-Driven Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. Each GDCS injection line includes a nozzle flow limiter to limit break size	Inspections of the as-built GDCS injection flow limiters will be taken	A calculation exists that confirms each GDCS injection nozzle flow limiter is less than or equal to 4.562E-3 m^2 (0.0491 ft^2).
14. Each GDCS equalizing line includes a nozzle flow limiter to limit break size.	Inspections of the as-built GDCS equalizing flow limiters will be taken	A calculation exists that confirms each GDCS equalizing line nozzle flow limiter is less than or equal to 2.027E-3 m^2 (0.0218 ft^2).
15. Each of the GDCS divisions is powered from their respective safety-related power divisions.	Tests will be performed on the GDCS by providing a test signal in only one safety-related power division at a time.	Testing confirms the signal exists only in the safety-related power division under test in the GDCS.
16. 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.	Inspections of the as-built GDCS will be performed.	Inspection confirms 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.
17. The GDCS pools A, B/C, and D are sized to hold a minimum drainable water volume.	An analysis of combined minimum drainable volume for GDCS pools A, B/C, and D will be performed.	Analysis confirms the combined minimum drainable water volume for GDCS pools A, B/C, and D is 1636 m^3 (57775 ft^3).
18. The GDCS pools A, B/C, and D are of sized for holding a specified minimum water level.	An analysis of minimum water level in GDCS pools A, B/C, and D will be performed.	Analysis confirms the minimum water level in GDCS pools A, B/C, and D is 6.5 m (21.33 ft).

Table 2.4.2-3

ITAAC For The Gravity-Driven Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
19. The minimum elevation change between minimum water level of GDCS pools and the centerline of GDCS injection line nozzles is sufficient to provide gravity-driven flow.	An analysis of minimum elevation change between minimum water level of GDCS pools and the centerline of GDCS injection line nozzles will be performed.	Analysis confirms 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).
20. The minimum drainable volume from the suppression pool to the RPV is sufficient to meet long-term post-LOCA core cooling requirements.	An analysis of minimum drainable volume from the suppression pool to the RPV will be performed.	Analysis confirms the minimum drainable volume from the suppression pool to the RPV is 799 m^3 (28,216 ft^3).
21. The long-term GDCS minimum equalizing driving head is based on RPV Level 0.5.	An analysis of the minimum equalizing driving head will be performed.	Analysis confirms the minimum equalizing driving head is 1 meter (3.28 ft).
22. The GDCS Deluge squib valves open as designed.	A vendor type test will be performed on a squib valve to open as designed.	Records of vendor type test concludes GDCS Deluge squib valves used open as designed.



Note: GDCS divisions A and D are connected to pools A and D respectively.
GDCS B and C are connected to a common pool (B/C).

Figure 2.4.2-1. Gravity-Driven Cooling System

2.5 REACTOR SERVICING EQUIPMENT

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

2.5.1 Fuel Servicing Equipment

No entry for this equipment.

2.5.2 Miscellaneous Servicing Equipment

No entry for this equipment.

2.5.3 Reactor Pressure Vessel Servicing Equipment

No entry for this equipment.

2.5.4 RPV Internals Servicing Equipment

No entry for this equipment.

2.5.5 Refueling Equipment

The ESBWR is supplied with a Reactor Building (RB) refueling machine for fuel movement and a fuel handling machine used for fuel servicing and transporting tasks in the Fuel Building (FB).

Design Description

The functional arrangement of the RB refueling machine is that it 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. The machine is a rigid structure built to ensure accurate and repeatable positioning during the refueling process.

The functional arrangement of the FB fuel handling machine is that it is equipped with a traversing trolley on which is mounted a telescoping mast and integral fuel grapple. The machine is a rigid structure built to ensure accurate and repeatable positioning while handling fuel.

(1) The functional arrangement of the RB refueling machine is as described in the Design Description of this Subsection 2.5.5.

(2) The RB refueling machine is classified as nonsafety-related, but is designed as seismic Category II.

(3) The RB refueling machine has an auxiliary hoist with sufficient load capability.

(4) The RB refueling machine is provided with controls interlocks.

(5) The functional arrangement of the FB fuel handling machine is as described in the Design Description of this Subsection 2.5.5.

(6) The FB fuel handling machine is classified as nonsafety-related, but is designed as seismic Category II.

(7) The FB fuel handling machine has an auxiliary hoist with sufficient load capability.

(8) The FB fuel handling machine is provided with controls interlocks.

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 Refueling Machine

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the RB refueling machine is as described in the Design Description of this Subsection 2.5.5.	Inspections of the as-built system will be performed.	Report(s) document that the as-built RB refueling machine conforms to the functional arrangement as described in the Design Description of the Subsection 2.5.5.
2. The RB refueling machine is classified as nonsafety-related, but is designed as seismic Category II.	Inspections and/or analyses of the as-built system will be performed.	Report(s) document that the as-built RB refueling machine can withstand seismic dynamic loads without loss of load carrying or structural integrity functions.
3. The RB refueling machine has an auxiliary hoist with sufficient load capability.	Load tests on the as-built auxiliary hoists will be conducted.	Report(s) document that a successful load test of each as-built auxiliary hoist has been performed.

Table 2.5.5-1

ITAAC for Refueling Machine

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The RB refueling machine is provided with controls interlocks	Test shall be performed with actual or simulated signals to demonstrate that the as-built interlocks function as required.	Report(s) document that the tests have been completed and results demonstrate that the as-built interlocks function as follows:a. Prevent hoisting a fuel assembly over the vessel with a control rod removed; b. Prevent collision with fuel pool walls or other structures; c. Limit travel of the fuel grapple; d. Interlock grapple hook engagement with hoist load and hoist up power; and e. Ensure correct sequencing of the transfer operation in the automatic or manual mode.
5. The functional arrangement of the FB fuel handling machine is as described in the Design Description of this Subsection 2.5.5.	Inspections and/or analyses of the as-built system will be performed.	Report(s) document that the as-built FB fuel handling machine conforms with the functional arrangement as described in the Design Description of the Subsection 2.5.5.
	Inspections and/or analyses of the as-built system will be performed.	Report(s) document that the as-built FB fuel handling machine can withstand seismic dynamic loads without loss of load carrying or structural integrity functions.

Table 2.5.5-1

ITAAC for Refueling Machine

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The FB fuel handling machine has an auxiliary hoist with sufficient load capability.	Load tests on the as-built auxiliary hoists will be conducted.	Report(s) document that a successful load test of the as-built auxiliary hoist has been performed.
8. The FB fuel handling machine is provided with controls interlocks.	Test will be performed with actual or simulated signals to demonstrate that the interlocks function as required.	Report(s) document that the tests have been completed and results demonstrate that the required interlocks function as follows: a. Prevent collision with fuel pool walls or other structures; b. Limit travel of the fuel grapple; c. Interlock grapple hook engagement with hoist load and hoist up power; and d. Ensure correct sequencing of the transfer operation in the automatic or manual mode.

2.5.6 Fuel Storage Facility

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

Design Description

(1) New fuel storage racks are designed to withstand a design bases seismic event.

(2) Spent fuel storage racks are designed to withstand a design bases seismic event.

(3) A full new fuel rack will remain subcritical by at least 5% Δk, i.e. $k_{eff} \leq 0.95$.

(4) A full spent fuel rack will remain subcritical by at least 5% Δk, i.e. $k_{eff} \leq 0.95$.

(5) The maximum spent fuel rack water coolant flow temperature at the rack exit shall be ≤ 100°C (212°F).

(6) The maximum stresses in the spent fuel racks do not exceed ASME Code, Section III, design allowable during accident conditions.

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. New fuel storage racks are designed to withstand a design bases seismic event.	An inspection of the new fuel storage racks configuration will be performed to ensure the design conforms to the seismic analyses.	Report(s) document that an analysis exists and concludes that the new fuel racks can withstand seismic design basis dynamic loads, and that the as-built configuration conforms to the analyses.
2. Spent fuel storage racks are designed to withstand a design bases seismic event.	An inspection of the spent fuel storage racks configuration will be performed to ensure the design conforms to the seismic analyses.	Report(s) document that an analysis exists and concludes that the spent fuel racks can withstand seismic design basis dynamic and that the as-built configuration conforms to the analyses.
3. A full new fuel rack will remain subcritical by at least 5% Δk, i.e. keff $\leq$ 0.95.	Analyses will be performed to determine k_{eff} for fully loaded new fuel pool storage racks.	Analysis records confirm that the maximum calculated $k_{eff} \leq 0.95$.
4. A full spent fuel rack will remain subcritical by at least 5% Δk, i.e. keff $\leq$ 0.95.	Analyses will be performed to determine k_{eff} for fully loaded spent fuel pool storage racks.	Analysis records confirm that the maximum calculated $k_{eff} \leq 0.95$.
5. The maximum spent fuel rack water coolant flow temperature at the rack exit shall be ≤ 100°C (212°F).	Analyses will be performed to determine the maximum temperature of the spent fuel racks.	Analysis records confirm that the maximum temperature in the spent fuel racks is < 100°C (212°F) at rack exit under normal operating conditions.
6. The maximum stresses in the spent fuel racks do not exceed ASME Code, Section III, design allowable during accident conditions.	Analyses will be performed to determine allowable stress under maximum rack temperature.	Report(s) document that analysis records confirm that the maximum stresses in the racks will not exceed ASME Code, Section III, design allowable during accident conditions.

2.5.7 Under-Vessel Servicing Equipment

No entry for this equipment.

2.5.8 FMCRD Maintenance Area

No entry for this area.

2.5.9 Fuel Cask Cleaning

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). The functional 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 effect 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 control fuel transfer. There is a means to seal off the upper and lower end of the tube while allowing filling and venting of the tube. 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. To ensure that 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 is vented to the building through the hoist cable piping that originates at the top of the transfer tube which extend above the level of the water in the RB with no valves or obstructions.

(1) The functional arrangement of the IFTS is as described in this Section 2.5.10.

(2) The seismic Category I, nonsafety-related IFTS tubes and supporting structure can withstand seismic design basis loads without failure of the basic structure or compromising the integrity of adjacent equipment and structures. The seismic portion of the IFTS tubes and supporting structure includes 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 (e.g., tube, valve, support structure, and bellows).

(3) The IFTS is functionally capable of moving fuel.

(4) 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 valves that could cause draining of water from the upper pool in an uncontrolled manner.

(5) The IFTS can be maintained filled with water for cooling in the event the fuel transport cart with fuel loaded within the IFTS cannot be moved.

(6) For personnel radiation protection, access (ingress and egress) to areas adjacent to the transfer tube is controlled through a system of physical barriers, interlocks and alarms.

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 functional arrangement of the IFTS is as described in this Section 2.5.10.	Inspections of the as-built system will be performed.	Report(s) document that the as-built IFTS conforms to the functional arrangement as described in this Section 2.5.10.
2. The seismic Category I, nonsafety-related IFTS tubes and supporting structure can withstand seismic design basis loads without failure of the basic structure or compromising the integrity of adjacent equipment and structures. The seismic portion of the IFTS tubes and supporting structure includes 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 (e.g., tube, valve, support structure, and bellows).	i) Inspection will be performed to verify that the Seismic Category I equipment is located on a seismic structure. ii) Type tests, analyses, or a combination of type tests and analyses of Seismic Category I equipment will be performed.	Report(s) document that: i) Inspection results verify that the Seismic Category I equipment is located on a seismic structure. ii) A report exists and concludes that the Seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
3. The IFTS is functionality capable of moving fuel.	Tests will be performed using installed controls and power supplies utilizing dummy fuel bundles for successful demonstration of fuel movement from the refuel pool to the spent fuel pool and return.	Report(s) document that tests conclude that the as-built IFTS passes functional testing.

Table 2.5.10-1

ITAAC For The Inclined Fuel Transfer System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. 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 valves that could cause draining of water from the upper pool in an uncontrolled manner.	Tests and inspection will be performed on the as-built IFTS to confirm it prevents loss of load and draining water from the upper pool in an uncontrolled manner.	Report(s) document that tests and inspections confirm the as-built IFTS prevents loss of load and draining water from the upper pool in an uncontrolled manner.
5. The IFTS can be maintained filled with water for cooling in the event the fuel transport cart with fuel loaded within the IFTS cannot be moved.	Tests and inspection will be performed on the as-built IFTS that confirm the as-built IFTS can be maintained filled with water in the event the fuel transport cart with fuel loaded within the IFTS cannot be moved.	Report(s) document tests and inspections that confirm the as-built IFTS can be maintained filled with water in the event the fuel transport cart with fuel loaded within the IFTS cannot be moved.
6. For personnel radiation protection, access (ingress and egress) to areas adjacent to the transfer tube is controlled through a system of physical barriers, interlocks and alarms.	Tests and inspections will be performed as follows: a. Inspections will be performed to verify that physical barriers exist between the transfer tube and adjacent areas. b. Tests or inspections will confirm that the as-built interlocks and alarms exist for controlling access to the transfer tube area and indicating operation of the IFTS.	Report(s) document results as follows: a. Inspection report(s) document that physical barriers exist between the transfer tube and adjacent areas. b. Test or inspection report(s) conclude that the as-built interlocks and alarms exist for controlling access to the transfer tube area and indicating operation of the IFTS.

2.5.11 (Deleted)

2.5.12 (Deleted)

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, 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. The RWCU/SDC system is as shown in Figure 2.6.1-1.

(1) The functional arrangement of the RWCU/SDC system is as described in the Design Description of Section 2.6.1, Table 2.6.1-1, and Figure 2.6.1-1.

(2) The containment isolation portions of the RWCU/SDC System are addressed in Tier 1, Subsection 2.15.1.

(3) The components identified in Table 2.6.1-1 as ASME Code Section III retain their pressure boundary integrity under internal pressures that will be experienced during service.

(4) Control room features provided for the RWCU/SDC System are defined in Table 2.6.1-1.

(5) Manual closure of the RPV bottom head isolation valve can be accomplished remotely.

(6) Each of the RWCU/SDC System safety-related components with safety-related power identified in Table 2.6.1-1 is powered from its respective safety-related division.

(7) The Seismic Category I equipment identified in Table 2.6.1-1 can withstand seismic design basis loads without loss of safety function.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.1-2 provides the inspections, tests, and/or analyses that will be undertaken for the RWCU/SDC system.

Table 2.6.1-1
Reactor Water Cleanup/Shutdown Cooling System Mechanical Equipment

Equipment Name (Description)	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
Demineralizers	Yes	Yes	No	No	No	---	Yes
Higher Capacity Pumps	Yes	Yes	No	No	Yes	Off	Yes
Lower Capacity Pumps	Yes	Yes	No	No	Yes	Off	Yes
Adjustable Speed Motor Drives	Yes	Yes	No	No	Yes	Off	Yes
Regenerative Heat Exchangers (RHXs)	Yes	Yes	No	No	No	---	No
Non-Regenerative Heat Exchangers (NRHXs)	Yes	Yes	No	No	No	---	No
Containment Inboard Isolation Valves - Pneumatic	Yes	Yes	Yes	Yes	Yes	Close	Yes
Containment Outboard Isolation Valves – Pneumatic	Yes	Yes	Yes	Yes	Yes	Close	Yes
Bottom Suction Outboard Valve – Motor Operated	Yes	Yes	No	No	Yes	As-Is	Yes
Pump Suction Valves	Yes	Yes	No	No	Yes	As-Is	No

Table 2.6.1-1
Reactor Water Cleanup/Shutdown Cooling System Mechanical Equipment

Equipment Name (Description)	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
Return Line Isolation Valve	Yes	Yes	No	No	Yes	As-Is	No
Demineralizer Inlet and Outlet Valves	Yes	Yes	No	No	No	As-Is	Yes
Demineralizer Bypass Flow Control Valve	Yes	Yes	No	No	Yes	Open	Yes
Overboard Flow Control Valve	Yes	Yes	No	No	Yes	Closed	Yes
Suction Line from RPV to the Outboard Containment Isolation Valves	Yes	Yes	Yes	--	--	--	--
Reactor Bottom Flow Sample Line to the Outboard Containment Isolation Valve	Yes	Yes	Yes	--	--	--	--
RWCU/SDC Return Line from the Isolation Valve up to and including the connection to the Feedwater Line	Yes	Yes	No	--	--	--	--

Table 2.6.1-1
Reactor Water Cleanup/Shutdown Cooling System Mechanical Equipment

Equipment Name (Description)	ASME Code Section III	Seismic Cat. I	RCPB Component	Containment Isolation Valve	Remotely Operated	Loss of Motive Power Position	MCR Alarms
RWCU/SDC Overboarding Line	Yes	Yes	No	--	--	--	--
Pump Suction Line from the Outboard Containment Isolation Valve Up to the Return Line Isolation Valve	Yes	Yes	No	--	--	--	--

Note: A dash means not applicable.

Table 2.6.1-2

ITAAC For The Reactor Water Cleanup/Shutdown Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the RWCU/SDC system is as described in the Design Description of Section 2.6.1, Table 2.6.1-1, and Figure 2.6.1-1.	Inspection of the as-built system will be performed.	Report(s) document that the as-built RWCU/SDC system conforms to the functional arrangement described in the Design Description of Section 2.6.1, Table 2.6.1-1, and Figure 2.6.1-1.
2. The containment isolation portions of the RWCU/SDC System are addressed in Tier 1, Subsection 2.15.1.	See Tier 1, Subsection 2.15.1.	See Tier 1, Subsection 2.15.1.
3. The components identified in Table 2.6.1-1 as ASME Code Section III retain their pressure boundary integrity under internal pressures that will be experienced during service.	A hydrostatic test will be conducted on those code components of the RWCU/SDC system required to be hydrostatically tested by the ASME code.	Report(s) document that the results of the hydrostatic test of the ASME Code components of the RWCU/SDC System conform to the requirements in the ASME Code, Section III.
4. Control room features provided for the RWCU/SDC System are defined in Table 2.6.1-1.	Inspections will be performed on the Control Room features for the RWCU/SDC system.	Report(s) document that features exist or can be retrieved in the Control Room as defined in Table 2.6.1-1. .
5. Manual closure of the RPV bottom head isolation valve can be accomplished remotely.	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.	Report(s) document that the RPV bottom head isolation valve can be manually closed remotely.

Table 2.6.1-2

ITAAC For The Reactor Water Cleanup/Shutdown Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Each of the RWCU/SDC System safety-related components with safety-related power identified in Table 2.6.1-1 is powered from its respective safety-related division.	Testing will be performed on the RWCU/SDC System by providing a test signal in only one safety-related division at a time.	Report(s) document that a test signal exists in the safety-related division (or at the equipment identified in Table 2.6.1-1 powered from the safety-related division) under test in the RWCU/SDC System.
7. The Seismic Category I equipment identified in Table 2.6.1-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the Seismic Category I equipment and valves identified in Table 2.6.1-1 are located in the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of Seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	Report(s) document that: i) The Seismic Category I equipment identified in Table 2.6.1-1 is located on a seismic structure. ii) A report exists and concludes that the Seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.

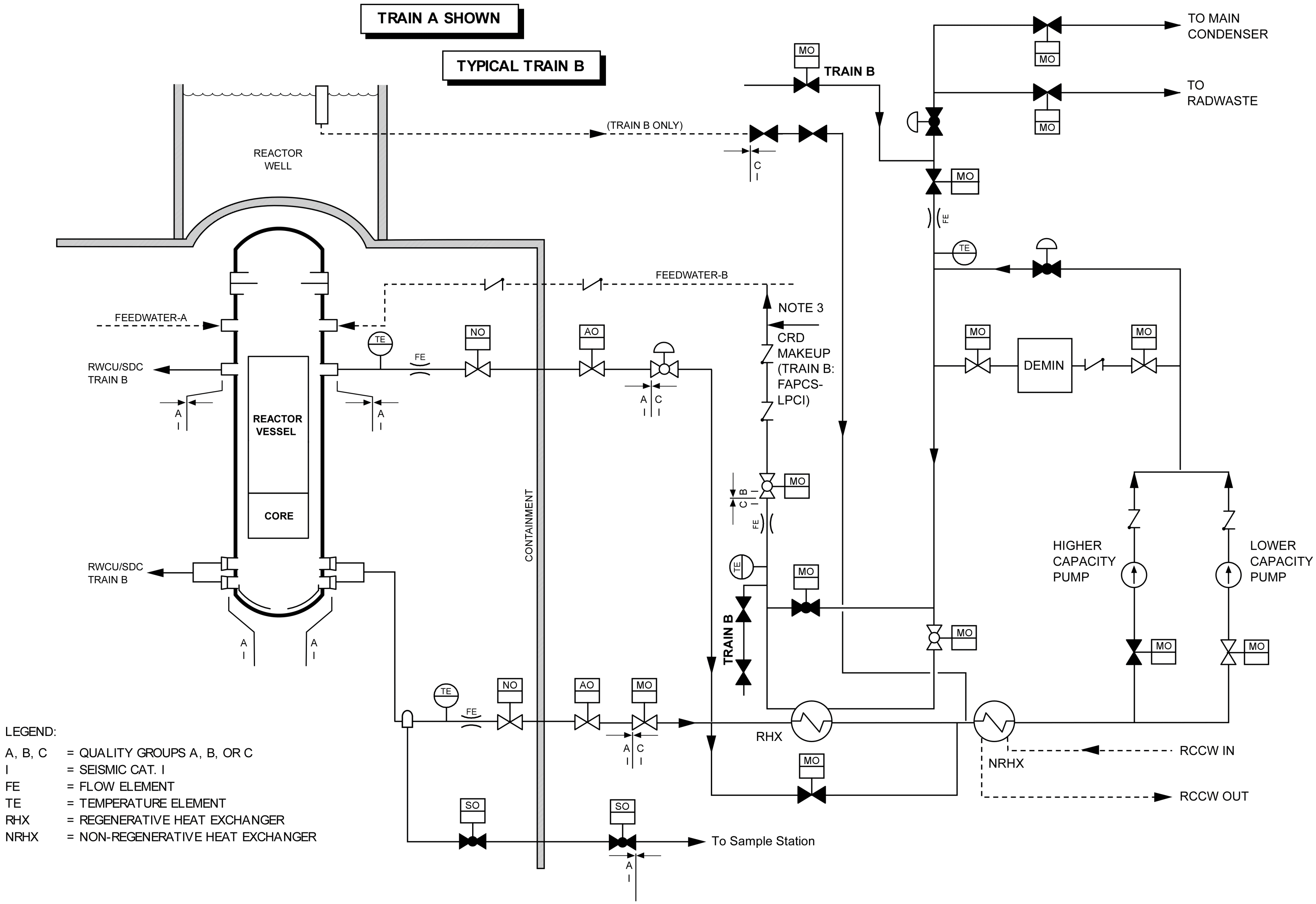


Figure 2.6.1-1. Reactor Water Cleanup/Shutdown Cooling System

2.6.2 Fuel And Auxiliary Pools Cooling System

Design Description

The Fuel and Auxiliary Pools Cooling System (FAPCS) provides cooling and cleaning of pools located in the containment, reactor building and fuel building, during normal plant operation. The FAPCS provides flow paths for filling and makeup of these pools during normal plant operation and under post-accident condition. The FAPCS provides suppression pool cooling and LPCI as active backup of the passive containment heat removal systems.

The FAPCS is as shown in Figure 2.6.2-1.

(1) The functional arrangement of the FAPCS is as described in the Design Description of this Subsection 2.6.2 and as shown in Figure 2.6.2-1.

(2) The components and piping identified in Table 2.6.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

(3) Pressure boundary welds in components and piping identified in Table 2.6.2-1 as ASME Code Section III meet ASME Code Section III requirements.

(4) The components and piping identified in Table 2.6.2-1 as ASME Code Section III retain their pressure boundary integrity under internal pressure that will be experienced during service.

(5) The Seismic Category I equipment and piping identified in Table 2.6.2-1 can withstand seismic design basis load without loss of structural integrity and safety function.

(6) The containment isolation portions of the FAPCS are addressed in Tier 1, Subsection 2.15.1.

(7) The FAPCS performs the following nonsafety-related functions:

   a. Suppression pool cooling mode

   b. Low-pressure coolant injection mode.

   c. External connection for emergency water to IC/PCC pool and Spent Fuel Pool.

(8) FAPCS minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.

(9) Level instruments with adequate operating ranges are provided for the Spent Fuel Pool and IC/PCC pools.

(10) Equipment qualification for the FAPCS is addressed in Tier 1 Section 3.8.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.2-2 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria for the FAPCS.

Table 2.6.2-1

Fuel and Auxiliary Pools Cooling System Equipment and Piping

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
Makeup Water From FPS to IC/PCC Pool	Yes	Yes	-	Yes	No	Closed	-	-
Post LOCA Fill Up Connection to IC/PCC Pool	Yes	Yes	-	Yes	No	Closed	-	-
Makeup Water From FPS to Spent Fuel Pool	Yes	Yes	-	Yes	No	Closed	-	-
Post LOCA Fill Up Connection to Spent Fuel Pool	Yes	Yes	-	Yes	No	Closed	-	-
FAPCS Low Pressure Cooling Injection Valves F335 A and B to RWCU/SDC Train B Connection	Yes	Yes	-	Yes	No	Closed	-	-
Fuel and Auxiliary Pools Cooling System Containment Isolation Valves	See Tier 1 Subsection 2.15.1							

[1] Entries on this table apply only to the flowpaths and components shown on Figure 2.6.2-1.

Table 2.6.2-2

ITAAC For The Fuel and Auxiliary Pools Cooling Cleanup System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The FAPCS functional arrangement is as described in Design Description of Subsection 2.6.2 and Figure 2.6.2-1.	Inspections of the as-built system will be conducted.	Inspection report(s) document that the as-built FAPCS configuration is as described in Subsection 2.6.2 and as shown on Figure 2.6.2-1.
2. The components and piping identified in Table 2.6.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection(s) will be conducted of the as-built FAPCS as documented in the ASME design reports.	Report(s) document that an ASME Code Section III design report(s) exist for the as-built components identified in Table 2.6.2-1 as ASME Code Section III.
3. Pressure boundary welds in components and piping identified in Table 2.6.2-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection(s) of the as-built pressure boundary welds will be performed in accordance with ASME Code Section III.	A report exists and documents that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.

Table 2.6.2-2

ITAAC For The Fuel and Auxiliary Pools Cooling Cleanup System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The components and piping identified in Table 2.6.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	i) A hydrostatic or pressure test will be performed on the components required by the ASME Code Section III to be tested.	i) A report exists and documents that the results of the pressure test of the components identified in Table 2.6.2-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
	ii) Impact testing will be performed on the containment and pressure-retaining materials in accordance with the ASME Code Section III to confirm the fracture toughness of the materials.	ii) A report exists and documents that the containment and pressure-retaining penetration materials conform with fracture toughness requirements of the ASME Code section III.
5. The seismic Category I equipment and piping identified in Table 2.6.2-1 can withstand seismic design basis loads without loss of structural integrity and safety function.	i) Type tests and/or analyses of seismic Category I equipment will be performed.	i) A report exists and documents that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of structural integrity and safety function.
	ii) Inspections will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	ii) The as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
6. The containment isolation portions of the FAPCS are addressed in Tier 1, Subsection 2.15.1.	See Tier 1 Subsection 2.15.1.	See Tier 1 Subsection 2.15.1

Table 2.6.2-2

ITAAC For The Fuel and Auxiliary Pools Cooling Cleanup System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The FAPCS performs the following nonsafety-related functions: a. Suppression pool cooling mode	Perform a test to confirm the flow path between the FAPCS to the suppression pools.	Test report(s) document that the cooling flow path is demonstrated and confirmed by operation of the function
b. Low-pressure coolant injection mode.	Perform a test to confirm the flow path from the FAPCS to the RWCU/SDC system.	Test report(s) document that the injection flow path is demonstrated and confirmed by operation of the function. The flowrate is ≥340 m^3/hr (1500 gpm) at a differential pressure of 1.03 MPa (150 psi).
c. External connection for emergency water to IC/PCC pool and Spent Fuel Pool from the Fire Protection System and offsite water supplies	Perform a test to confirm flow path and flow capacity from the Fire Protection System and offsite water sources to the pools.	Test report(s) document that the makeup water flow path is demonstrated and confirmed by operation of the function.
8. FAPCS minimum inventory of alarms, displays, and status indications in the main control room (MCR) are addressed in Section 3.3.	See Tier 1 Section 3.3.	See Tier 1 Section 3.3.
9. Level instruments with adequate operating ranges are provided for the Spent Fuel Pool and IC/PCC pools.	Inspections of the FAPCS will be conducted to verify that level instruments with adequate operating ranges are provided for the Spent Fuel Pool and IC/PCC pools.	Inspection report(s) document that the as-built FAPCS provides Spent Fuel Pool and IC/PCC pool level instrumentation with adequate operating ranges.
10. Equipment qualification for the FAPCS is addressed in Tier 1 Section 3.8.	See Tier 1 Section 3.8.	See Tier 1 Section 3.8.

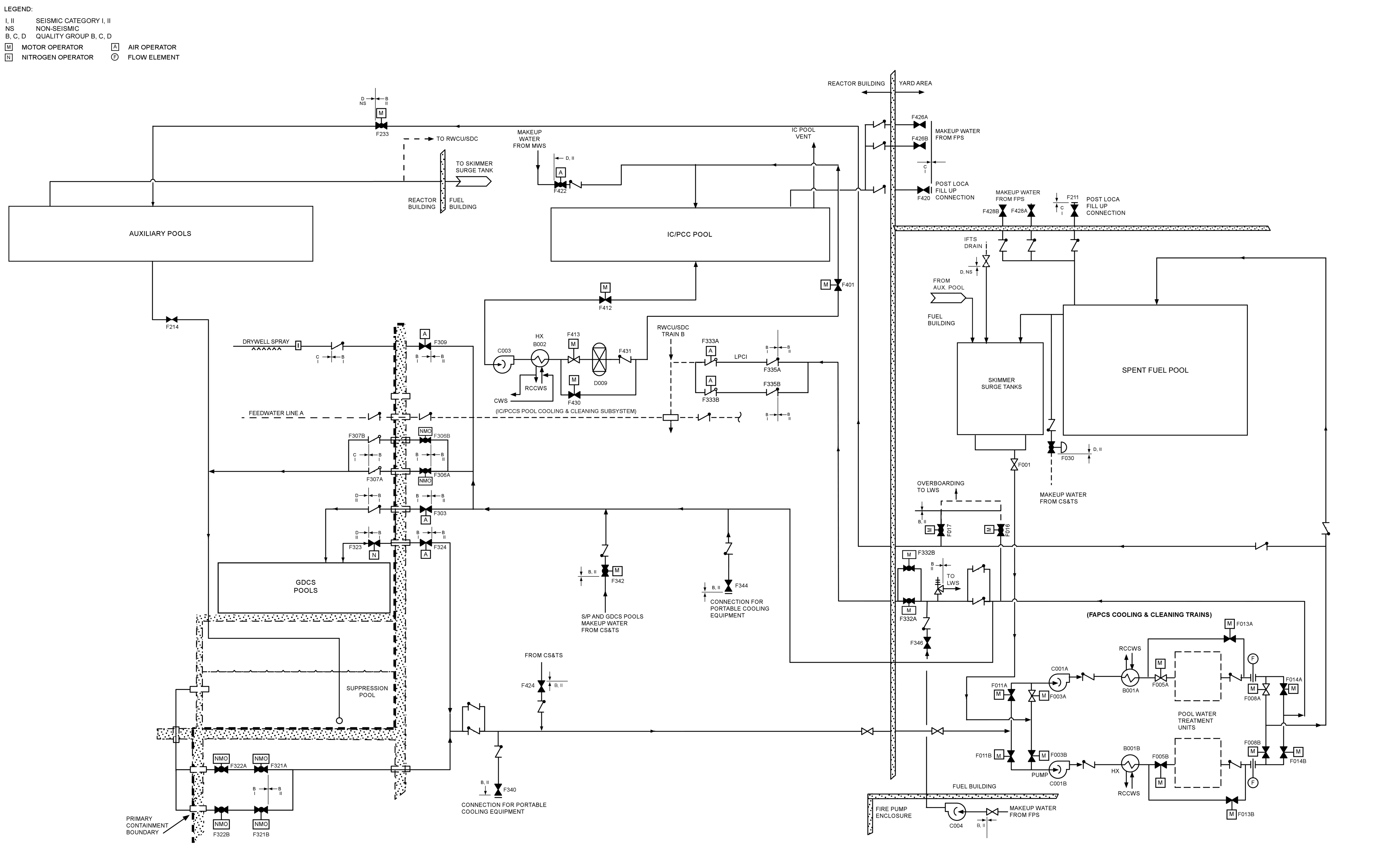


Figure 2.6.2-1. Fuel and Auxiliary Pools Cooling Cleanup System

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.

(1) The functional arrangement of the MCR Panels is as described in this Section 2.7.1.

(2) The safety-related MCR Panels conform to seismic Category I requirements and are housed in a seismic Category I structure.a.Independence is provided between safety-related divisions.

b. Separation is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.

(4) 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. The functional arrangement of the MCR Panels is as described in this Section 2.7.1.	Inspections of the as-built MCR Panels will be conducted.	Inspection report(s) document that the as-built MCR Panels conform with the functional arrangement as described in Section 2.7.1.
2. The safety-related MCR Panels • Conform to Seismic Category I requirements, and • Are housed in a seismic Category I structure.	i) Type tests and/or analyses of the safety-related MCR Panels will be performed.	i) A report exists and concludes that the safety-related MCR Panels conform to seismic Category I requirements.
	ii) Inspections of the as-built safety-related MCR Panels will be performed to verify that the equipment is installed in accordance with the configurations specified in the type tests and/or analyses.	ii) Inspection report(s) document that the as-built safety-related MCR Panels are installed in accordance with the configurations specified by the type tests and/or analyses.
	iii) Inspections of the as-built safety-related MCR Panels will be performed to verify that the equipment is housed in a seismic Category I structure.	iii) Inspection report(s) document that the as-built safety-related MCR Panels are housed in a seismic Category I structure.

Table 2.7.1-1

ITAAC For Main Control Room Panels

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3a) Independence is provided between safety-related divisions.	Tests will be performed on the as-built safety-related MCR Panels by providing a test signal in only one safety-related division at a time and checking for voltage in all divisions.	Test report(s) document that a test signal exists only in the as-built safety-related division under test in the MCR Panels.
b) Separation or electrical isolation is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	Inspection of the as-built safety-related MRC Panels will be performed.	Inspection report(s) document that, for the as-built safety-related MCR Panels, physical separation or electrical isolation exists between safety -related divisions. Physical separation or electrical isolation exists between safety-related divisions and nonsafety-related equipment.
4. Human factors engineering principles are incorporated into the MCR Panel design	See Tier 1 Section 3.3.	See Tier 1 Section 3.3.

2.7.2 Radioactive Waste Control Panels

No entry for this system.

2.7.3 Local Control Panels And Racks

Design Description

Local Control Panels 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. Because of the Reactor Protection System's fail-safe design, no potential sources of missiles or pipe breaks prevent modules from performing their safety-related reactor shutdown function.

(1) The functional arrangement of the Local Control Panels and Instrument Racks is as described in this Section 2.7.3.

(2) The safety-related Local Control Panels and Instrument Racks conform to seismic Category I requirements.a.Independence is provided between safety-related divisions.

  b. Separation is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.

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 Control Panels and Instrument Racks.

Table 2.7.3-1

ITAAC For Local Control Panels and Instrument Racks

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the Local Control Panels and Instrument Racks is described in this Section 2.7.3.	Inspections of the as-built Local Control Panels and Instrument Racks will be conducted.	Inspection report(s) document that the as-built Local Control Panels and Instrument Racks conform with the functional arrangement as described in Section 2.7.3.
2. The safety-related Local Control Panels and Instrument Racks conform to Seismic Category I requirements.	i) Type tests and/or analyses of the safety-related Local Control Panels and Instrument Racks will be performed.	i) A report exists and concludes that the safety-related Local Control Panels and Instrument Racks conform to seismic Category I requirements.
	ii) Inspections of the as-built safety-related Local Control Panels and Instrument Racks will be performed to verify that the equipment is installed in accordance with the configurations specified in the type tests and/or analyses.	ii) Inspection report(s) document that the as-built safety-related Local Control Panels and Instrument Racks are installed in accordance with the configurations specified by the type tests and/or analyses.

Table 2.7.3-1

ITAAC For Local Control Panels and Instrument Racks

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3a. Independence is provided between safety-related divisions.	Tests will be performed on the as-built safety-related Local Control Panels and Instrument Racks by providing a test signal in only one safety-related division at a time.	Test report(s) document that a test signal exists only in the as-built safety-related division under test in the Local Control Panels and Instrument Racks.
b. Separation or electrical isolation is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	Inspection of the as-built safety-related Local Control Panels and Instrument Racks will be performed.	Inspection report(s) document that, for the as-built safety-related Local Control Panels and Instrument Racks, physical separation or electrical isolation exists between safety -related divisions. Physical separation or electrical isolation exists between safety-related divisions and nonsafety-related equipment.

2.8 NUCLEAR FUEL

2.8.1 Fuel Rods and Bundles

No entry for this system.

2.8.2 Fuel Channel

No entry for this system.

2.9 CONTROL RODS

No entry 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 neither performs nor ensures any safety-related function, and is not required to achieve or maintain safe shutdown.

The functional arrangement of the LWMS is that it has components in four subsystems which receive and store radioactive or potentially radioactive liquid waste. The four LWMS subsystems are as follows:

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

Table 2.10.1-1 describes the major components in each of these four subsystems. Other equipment includes piping, pumps, and valves for transferring the process flow. The LWMS processing equipment is located in the Radwaste Building. The permanent LWMS will connect to non-permanent mobile systems that process radioactive waste (actual mobile system unit operations and chemical reactors may differ based on improvements in radwaste technology). The LWMS is operated and monitored from the Radwaste Building Control Room. Main control room alarms are provided for key parameters of the LWMS. The LWMS either returns processed water to the condensate system or discharges to the circulating water system.

(1) The functional arrangement of the LWMS is as described in Subsection 2.10.1.

(2) The LWMS piping systems retain their pressure boundary integrity under internal pressures that will be experienced during service.

(3) LWMS discharge flow to circulating water is monitored for high radiation.A radiation monitor provides an automatic closure signal to the discharge line isolation valve.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10.1-2 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

Major Equipment in LWMS

Equipment	Number of Equipment Items
Equipment (Low Conductivity Drain Subsystem	
Collection tanks	3
Collection pumps	3
Mobile processing system	1
Sample tanks	2
Sample pumps	2
Floor (High Conductivity) Drain Subsystem	
Collection tanks	2
Collection pumps	2
Mobile processing system	1
Sample tanks	2
Sample pumps	2
Chemical Drain Subsystem	
Collection tank	1
Collection pump	2
Detergent Drain Subsystem	
Collection tanks	2
Collection pumps	2
Mobile processing system	1
Sample tanks	2
Sample pumps	2

Table 2.10.1-2

ITAAC For The Liquid Waste Management System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the LWMS is as described in Subsection 2.10.1.	Inspections of the as-built system will be performed.	Reports document that the as-built LWMS conforms to the functional arrangement description in the Design Description of this Subsection 2.10.1.
2. The LWMS piping systems retain their pressure boundary integrity under internal pressures that will be experienced during service.	A hydrostatic test in accordance with ASME/ANSI B31.3 will be conducted on the LWMS piping systems, except (1) at atmospheric tanks where no isolation valves exist, (2) when such testing would damage equipment, and (3) when such testing could seriously interfere with other system or component.	Reports document that the results of the hydrostatic test of the LWMS piping systems in accordance with ASME/ANSI B31.3 indicate no unacceptable pressure boundary leakage.
3. LWMS discharge flow to circulating water is monitored for high radiation.A radiation monitor provides an automatic closure signal to the discharge line isolation valve.	Tests will be conducted on the as-built LWMS using a simulated high radiation signal.	Reports document that the discharge flow terminates upon receipt of a simulated high radiation signal.

2.10.2 Solid Waste Management System

No entry for this system

2.10.3 Gaseous Waste Management System

Design Description

The gaseous waste management system processes and controls the release of gaseous radioactive effluents to the environs. The Offgas System (OGS) is the principal gaseous waste management subsystem. The various building HVAC systems perform other gaseous waste functions.

The functional arrangement of the OGS is that the 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 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 includes redundant hydrogen/oxygen catalytic recombiners and ambient temperature charcoal beds to provide for process gas volume reduction and radionuclide retention/decay. The OGS processes 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 of the OGS process equipment is performed both locally and remotely from the main control room.

(1) The functional arrangement of the OGS is as described in Subsection 2.10.3.

(2) The OGS is designed to withstand internal hydrogen explosions.

(3) Leakage from the process through purge or tap lines to external atmospheric pressure is sufficiently low so it is undetectable by “soap bubble” test.

(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.

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 functional arrangement of the OGS is as described in Subsection 2.10.3.	Inspections of the as-built OGS will be performed.	Reports document that the functional arrangement 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.	A pressure test of the as-built OGS will be conducted in the plant in accordance ASME/ANSI B31.3 requirements.	Reports document that the pressure testing results conform to the requirements in ASME/ANSI B31.3.
3. Leakage from the process through purge or tap lines to external atmospheric pressure is sufficiently low so it is undetectable by “soap bubble” test.	“Soap bubble” tests will be performed on the OGS mechanical joints on purge or tap lines at normal system operating pressure.	Reports document that the “soap bubble” test results show no detectable leakage.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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.	Tests will be performed as follows: a. A simulated high charcoal gas discharge radioactivity signal will give a Main Control Room (MCR) alarm. b. When 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. When a simulated OGS gas discharge radioactivity signal reaches a high-high-high level, the off-gas system discharge valve will close.	Test reports document that: 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 consists of four lines from the seismic interface restraint to the main turbine stop valves. The TMSS is nonsafety-related. Regulatory Guide 1.26 Quality Group B portions of the TMSS are designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Class 2 requirements. The TMSS is located in the Reactor Building steam tunnel and Turbine Building.

(1) The functional arrangement of the TMSS is as described in 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.

(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).

(4) The SAIV(s) fail(s) closed and required MSIV fission product leakage path TMSS drain valve(s) fail(s) open 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 classified as Seismic Category II.

(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.

(7) The TMSS piping provides a nominal turbine inlet (throttle) pressure that is consistent with assumptions in Abnormal Event analyses.

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 TMSS functional arrangement is as described in Subsection 2.11.1.	Inspections of the as-built system will be conducted.	A report exists and confirms that the as-built TMSS conforms to the functional arrangement description in 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.	A hydrostatic test will be conducted on those Code components of the TMSS that are required to be hydrostatically tested by the ASME Code Section III.	A report exists and concludes that the results of the hydrostatic test of the TMSS ASME Code components 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).	Tests will be performed on the SAIV(s) and required MSIV fission product leakage path TMSS drain valve(s) using simulated MSIV closure signals.	A test report exists and concludes that the SAIV(s) 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) fail(s) open on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure.	A functional test will be performed on SAIV(s) and required MSIV fission product leakage path TMSS drain valve(s).	A test report exists and concludes that the SAIV(s) fail(s) closed and required MSIV fission product leakage path TMSS drain valve(s) fail(s) open on loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure.

Table 2.11.1-1

Turbine Main Steam System ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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 classified as Seismic Category II.	An inspection will be performed to verify that a seismic analysis has been completed for the as-built TMSS piping and SAIV(s) and required MSIV fission product leakage path.	A report exists and demonstrates that the as-built TMSS piping, SAIV(s), and required MSIV fission product leakage path meet Seismic Category II requirements.
6. The integrity of the as-built MSIV 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.	Inspections (e.g., walk down) of non-seismically designed systems, structures and components overhead, adjacent to, and attached to the MSIV leakage path (i.e., the main steam piping, bypass piping, required drain piping and main condenser) will be performed.	A report exists and concludes that the non-seismically designed systems, structures and components overhead, adjacent to, and attached to the MSIV leakage path to the condenser will not compromise the integrity of the main steam piping, bypass piping, required drain piping and main condenser.
7. The TMSS piping provides a nominal turbine inlet (throttle) pressure that is consistent with assumptions in Abnormal Event analyses.	Inspection and/or analysis of the as-built TMSS piping will be performed to confirm the nominal turbine inlet (throttle) pressure.	A report exists and concludes that the TMSS piping provides a nominal turbine inlet (throttle) pressure of [6.57 MPaG (953 psig)].

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. The C&FS is classified as nonsafety-related.

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 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 located in the Reactor Building steam tunnel and Turbine Building.

(1) The functional arrangement for the C&FS is as described in Subsection 2.11.2.

(2) The C&FS provides sufficient feedwater flow to mitigate Abnormal Events.

(3) The C&FS limits maximum feedwater flow to mitigate Abnormal Events.

(4) The C&FS, in conjunction with the feedwater control system, provides sufficient feedwater flow after MSIV isolation to mitigate Abnormal Events.

(5) The C&FS, in conjunction with the feedwater control system, limits the maximum feedwater flow for a single pump following a single active component failure or operator error to mitigate Abnormal Events.

(6) The C&FS, in conjunction with the feedwater control system, is designed so that the loss of feedwater heating is limited in the event of a single operator error or equipment failure.

(7) The C&FS, in conjunction with other Power Cycle Systems, provides a nominal full load final feedwater temperature that is consistent with assumptions in Abnormal Event analyses.

(8) The C&FS has a nominal feedwater flow rate at rated conditions that is consistent with inputs and assumptions in Abnormal Event analyses.

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 functional arrangement of the C&FS is as described in Subsection 2.11.2.	Inspections of the as-built system will be conducted to confirm the functional arrangement.	A report exists and documents that the as-built C&FS conforms to the functional arrangement described in Subsection 2.11.2
2. The C&FS provides sufficient feedwater flow and volume to mitigate Abnormal Events.	An analysis and/or type testing of the as-built C&FS and feedwater pumps will be performed to confirm the minimum capacity of three feedwater pumps.	A report exists and concludes that three operating feedwater pumps are capable of supplying [135%] of the rated feedwater flow at [7.34 MPa gauge (1065 psig)].
3. The C&FS limits maximum feedwater flow to mitigate Abnormal Events.	Analysis and/or type testing of the as-built C&FS and feedwater pumps will be performed to confirm that the C&FS limits maximum feedwater flow.	A report exists and concludes that the maximum capacity of three feedwater pumps at [7.34 MPa gauge (1065 psig)] is less than or equal to [155%] of rated feedwater flow.
4. The C&FS, in conjunction with the feedwater control system, provides sufficient feedwater flow after MSIV isolation to mitigate Abnormal Events.	Inspection and/or analysis of the as-built feedwater system will be performed to confirm that the C&FS provides sufficient feedwater flow after MSIV isolation.	A report exists and concludes that the C&FS, in conjunction with the feedwater control system, provides feedwater flow greater than or equal to [240 seconds] of rated feedwater flow after MSIV isolation.

Table 2.11.2-1

Condensate and Feedwater System ITAAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The C&FS, in conjunction with the feedwater control system, limits the maximum feedwater flow for a single pump following a single active component failure or operator error to mitigate Abnormal Events.	Testing and/or analysis of the as-built C&FS and feedwater pumps and/or type testing of a single feedwater pump will be performed to confirm that the C&FS limits the maximum feedwater flow from a single pump.	A report exists and concludes that the C&FS, in conjunction with the feedwater control system, limits the maximum feedwater flow for a single pump to [75%] of rated flow following a single active component failure or operator error.
6. The C&FS, in conjunction with the feedwater control system, is designed so that the loss of feedwater heating is limited in the event of a single operator error or equipment failure.	Inspection and/or analysis of the as-built feedwater system will be performed to confirm that the C&FS, in conjunction with the feedwater control system, limits the loss of feedwater heating in the event of a single operator error or equipment failure.	A report exists and documents that the C&FS, in conjunction with the feedwater control system, is designed so that the loss of feedwater heating is limited to a final feedwater temperature reduction less than or equal to [55.6°C (100°F)] in the event of a single operator error or equipment failure.
7. The C&FS, in conjunction with other Power Cycle Systems, provides a nominal full load final feedwater temperature that is consistent with assumptions in Abnormal Event analyses.	Inspection and/or analysis of the as-built feedwater system and other Power Cycle Systems will be performed to confirm the final nominal feedwater temperature.	A report exists and concludes that the C&FS, in conjunction with other Power Cycle Systems, provides a nominal full load final feedwater temperature of [216°C (420°F)].
8. The C&FS has a nominal feedwater flow rate at rated conditions that is consistent with inputs and assumptions in Abnormal Event analyses.	Testing and/or analysis of the as-built C&FS and feedwater pumps and/or type testing of a single feedwater pump will be performed to confirm the nominal feedwater flow rate at rated conditions.	A report exists and concludes that the C&FS has a nominal feedwater flow rate at rated conditions of [2429 kg/s (19.3 x 10^6 lb/hr)].

2.11.3 Condensate Purification System

No entry for this system.

2.11.4 Main Turbine System

Design Description

The Main Turbine System (MT) is nonsafety-related. The ESBWR standard plant design has a favorably oriented turbine to minimize any potential impact on safety-related structures and equipment.

(1) 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 MT system piping failures or failure of the connection(s) from the low pressure turbine exhaust hood(s) to the condenser. Essential systems and components are defined in BTP SPLB 3-1 as systems and components required to shut down the reactor and mitigate the consquences of a postulated piping failure, without offsite power.

(2) The MT has a favorable orientation to minimize the potential effects of turbine missiles. Favorably oriented turbine generators are located such that the containment and most safety-related Systems, Structures and Components outside containment are excluded from the low-trajectory hazard zone described in RG 1.115.

(3) The MT control valve closing times are limited to mitigate Abnormal Events.

(4) The MT stop valve nominal closing times are limited to mitigate Abnormal Events.

(5) The MT can accommodate sufficient steam flow through three control valves to mitigate Abnormal Events.

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 Main Turbine System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The physical layout of the system assures that protection is provided to essential systems and components from the effects of high and moderate energy MT system piping failures or failure of the connection(s) from the low pressure turbine exhaust hood(s) to the condenser. Essential systems and components are defined in BTP SPLB 3-1 as systems and components required to shut down the reactor and mitigate the consquences of a postulated piping failure, without offsite power.	Inspections of the as-built Turbine Building and plant arrangements will be conducted.	An inspection report exists and concludes that the physical layout protects essential systems and components from the effects of high and moderate energy MT system piping failures or failure of the connection(s) from the low pressure turbine exhaust hood to the condenser. Essential systems and components are defined in BTP SPLB 3-1 as systems and components required to shut down the reactor and mitigate the consquences of a postulated piping failure, without offsite power.
2. The MT has a favorable orientation to minimize the potential effects of turbine missiles. Favorably oriented turbine generators are located such that the containment and most safety-related Systems, Structures and Components outside containment are excluded from the low-trajectory hazard zone described in RG 1.115.	Inspections will be performed to verify the orientation of the MT.	An inspection report exists and concludes that the MT has a favorable orientation to minimize the potential effects of turbine missiles. Favorably oriented turbine generators are located such that the containment and most safety-related Systems, Structures and Components outside containment are excluded from the low-trajectory hazard zone described in RG 1.115.

Table 2.11.4-1

ITAAC For The Main Turbine System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The MT control valve closing times are limited to mitigate Abnormal Events.	Testing and/or analysis of the as-built MT and/or type testing of a single turbine control valve will be performed to confirm control valve closing times.	A report exists and documents that the MT control valve closing time characteristic is limited to a minimum greater than or equal to [0.08 seconds] and a maximum less than or equal to [2.5 seconds].
4. The MT stop valve nominal closing time is limited to mitigate Abnormal Events.	Testing and/or analysis of the as-built MT and/or type testing of a single turbine stop valve will be performed to confirm stop valve nominal closing time.	A report exists and documents that the MT stop valve nominal closing time is [0.100 seconds].
5. The MT can accommodate sufficient steam flow through three control valves to mitigate Abnormal Events.	An inspection of the analysis of the as-built MT will be performed to confirm that the MT can accommodate sufficient steam flow through three control valves.	A report exists and concludes that the MT can accommodate a flow greater than or equal to [85%] of rated steam flow through three control valves.

2.11.5 Turbine Gland Seal System

Design Description

The TGSS minimizes the escape of radioactive steam from the turbine shaft/casing penetrations and valve stems.

(1) The TGSS functional arrangement is as described in Subsection 2.11.5 and as shown on Figure 2.11.5-1.

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 TGSS functional arrangement is as described in Subsection 2.11.5 and as shown on Figure 2.11.5-1.	Inspections of the as-built system will be performed.	A report exists and documents that the as-built TGSS conforms to the functional arrangement as described in Subsection 2.11.5 and as shown on Figure 2.11.5-1.

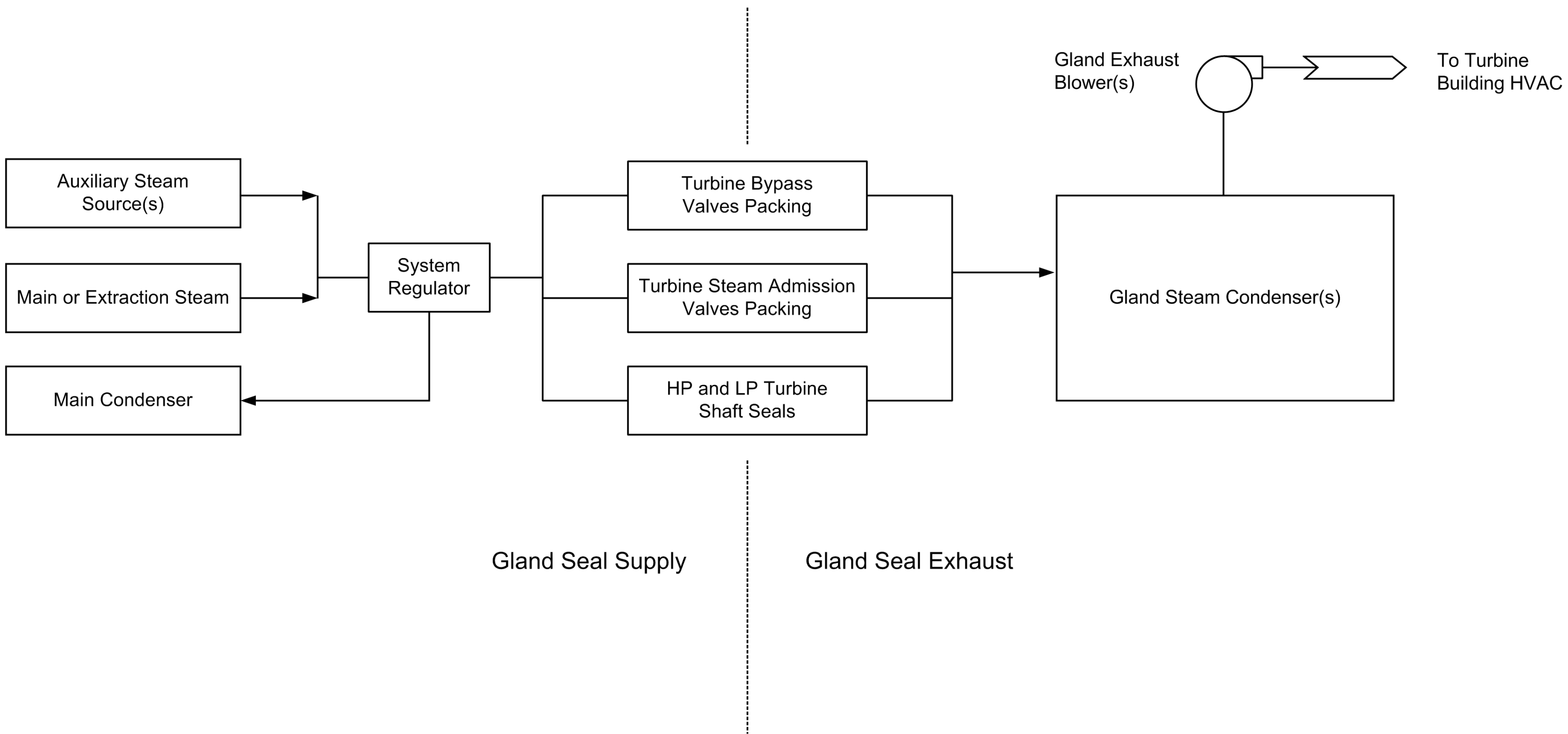


Figure 2.11.5-1. Turbine Gland Seal System Arrangement

2.11.6 Turbine Bypass System

Design Description

The TBS consists of hydraulically operated TBVs that are connected to the main steam header via TMSS piping. The TBS also includes the piping down stream of the TBVs to the main condenser. The Turbine Bypass System (TBS) passes steam to the main condenser in conjunction with the Turbine Main Steam System (TMSS) under the control of the Steam Bypass and Pressure Control (SB&PC) system. The TBS is classified as nonsafety-related. The TBS is used to mitigate abnormal events. The TBS is located in the Turbine Building.

(1) The functional arrangement for the TBS is as described in Subsection 2.11.6.

(2) The TBVs are controlled by signal(s) from the SB&PC System.

(3) The TBS steam pressure retaining and structural components are analyzed to demonstrate structural integrity under SSE loading conditions.

(4) The TBS accommodates steam flow to mitigate Abnormal Events.

(5) The TBS maintains sufficient capacity to mitigate Abnormal Events with a single active failure.

(6) The TBS design limits the capacity of individual TBVs.

(7) The TBS design allows the TBVs to open rapidly to support Abnormal Event mitigation.

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 functional arrangement for the TBS is described in Subsection 2.11.6.	Inspections of the as-built TBS will be conducted.	A report exists and concludes that the as-built TBS conforms to the functional arrangement described in Subsection 2.11.6.
2. The TBVs are controlled by signal(s) from the SB&PC System.	Tests will be conducted using a simulated signal(s).	A test report exists and confirms that the TBVs operate upon receipt of simulated signal(s) from the SB&PC System.
3. The TBS steam pressure retaining and structural components are analyzed to demonstrate structural integrity under SSE loading conditions.	An inspection of the as-built TBS will be performed to verify that it conforms with the seismic analysis.	An inspection report exists and concludes that the as-built TBS can withstand a SSE without loss of structural integrity.
4. The TBS accommodates steam flow to mitigate Abnormal Events.	An inspection will be performed to confirm that the as-built TBS accommodates steam flow to mitigate Abnormal Events.	An inspection report exists and concludes that the TBS accommodates at least [110%] of rated main steam flow.
5. The TBS maintains sufficient capacity to mitigate Abnormal Events with a single active failure.	An inspection will be performed to confirm that the as-built TBS maintains sufficient capacity to mitigate Abnormal Events with a single active failure.	An inspection report exists and confirms that the TBS maintains capacity greater than or equal to [50%] of the maximum capacity for a period greater than or equal to [6 seconds] with a single active failure.
6. The TBS design limits the capacity of individual TBVs.	A design analysis of the TBS will be performed to confirm that the TBS design limits the capacity of individual TBVs.	An analysis report exists and concludes that no single TBV has a capacity greater than [15%] of rated steam flow.

Table 2.11.6-1

ITAAC For The Turbine Bypass System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. The TBS design allows the TBVs to open rapidly to support Abnormal Event mitigation.	Testing and/or analyses of the TBS will be performed to confirm that the as-built TBS design allows the TBVs to open rapidly to support Abnormal Event mitigation.	A test and/or analysis report exists and concludes that the TBS can achieve a flow greater than or equal to [80%] of total bypass capacity in a time period less than or equal to [0.17] seconds after initiation of TBV fast opening function.

2.11.7 Main Condenser

Design Description

The MC is classified as nonsafety-related. However, the MC shell provides a hold-up volume for Main Steam Isolation Valve (MSIV) fission product leakage and accommodates the TBS steam flow to mitigate Abnormal Events.

(1) The MC structural members, supports, and anchors are designed to maintain condenser integrity following a safe shutdown earthquake (SSE).

(2) The MC can accommodate TBS steam flow to mitigate Abnormal Events.

Safety-related condenser pressure instruments are described in Tier 1, Section 2.2.7. MCES effluent radiation monitoring is described in Tier 1, Subsection 2.3.1.

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 structural members, supports, and anchors are designed to maintain condenser integrity following a safe shutdown earthquake.	An inspection will be performed to verify the ability of the as-built condenser structural members, supports, and anchors to maintain condenser integrity following a safe shutdown earthquake.	An inspection report exists and confirms that the as-built main condenser structural members, supports, and anchors are able to maintain condenser integrity following a safe shutdown earthquake.
2. The MC can accommodate TBS steam flow to mitigate Abnormal Events.	An inspection of the as-built condenser will be performed to confirm the capability of the as-built condenser to accommodate TBS steam flow to mitigate Abnormal Events.	An inspection report exists and confirms that the as-built main condenser has the capability to accommodate TBS steam flow for at least [6 seconds] following a loss of preferred power without the main condenser pressure exceeding the TBV isolation setpoint.

2.11.8 Circulating Water System

No entry for this system.

2.11.9 Power Cycle Auxiliary Systems

Design Description

The Power Cycle includes a number of auxiliary systems. The Power Cycle Auxiliary Systems include the Heater Drain and Vent System, Turbine Generator Control System, Turbine Lubricating Oil System, Moisture Separator Reheater System, Extraction System, Turbine Hydraulics System, Turbine Auxiliary Steam System, Generator System, Hydrogen Gas Control System, Stator Cooling Water System, Generator Lubricating and Seal Oil System, Hydrogen and Carbon Dioxide Bulk Storage System, and Generator Excitation System.

There are no entries for these systems.

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.

(1) The MWS has safety-related containment penetrations and isolation valves.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.12.1-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria 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.	See Tier 1, Subsection 2.15.1.	See Tier 1, Subsection 2.15.1.

2.12.2 Condensate Storage and Transfer System

No entry for this system.

2.12.3 Reactor Component Cooling Water System

Design Description

The Reactor Component Cooling Water System (RCCWS) does not perform any safety-related function. Therefore, the RCCWS has no safety design basis. The RCCWS is subject to additional regulatory oversight for its nonsafety-related functions to provide post 72-hour cooling to the nuclear island chillers and diesel generators and to provide cooling support to FAPCS under LOPP conditions.

The functional arrangement of the RCCWS is that it consists of two 100% capacity independent and redundant trains. Both trains share a chemical addition tank. The pumps in each train discharge to a common header leading to the RCCWS heat exchangers header. RCCWS cooling water is supplied to the following major users:

- Chilled Water System (CWS) Nuclear Island chiller-condenser
- RWCU/SDC non-regenerative heat exchanger
- FAPCS heat exchanger
- Standby On Site AC Power Supply Diesel Generators

(1) The RCCWS functional arrangement is described in the Design Description of Section 2.12.3.

(2) The RCCWS provides the nonsafety-related function to support post-72 hour cooling for nuclear island chillers and diesel generators and to provide cooling support for FAPCS under LOPP conditions.

(3) The RCCWS can be operated and controlled from the MCR.

(4) RCCWS flow indication is provided in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.12.3-1 provides definitions of the inspections, tests, and/or analyses, together with associated acceptance criteria for the RCCWS.

Table 2.12.3-1

ITAAC For The Reactor Component Cooling Water System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The RCCWS functional arrangement is described in the Design Description of Section 2.12.3.	Inspection of the as-built system will be performed.	Report(s) document that the as-built RCCWS System conforms to the functional arrangement described in the Design Description of this Section 2.12.3.
2. The RCCWS provides the nonsafety-related function to support post-72 hour cooling for nuclear island chillers and diesel generators and to provide cooling support for FAPCS under LOPP conditions.	Testing of the RCCWS will be performed to demonstrate flow to the nuclear island chillers, diesel generators and FAPCS.	A report documents that the RCCWS test demonstrates flow to the nuclear island chillers, diesel generators, and FAPCS.
3. The RCCWS can be operated and controlled from the MCR.	Testing to demonstrate flow capability will be performed on the RCCWS components using controls in the MCR.	A report documents that the MCR controls caused the RCCWS components to operate during the flow test.
4. RCCWS flow indication is provided in the MCR.	Inspection will verify that RCCWS flow indication can be retrieved in the MCR.	A report documents that the inspection verifies that RCCWS flow indication can be retrieved in the MCR.

2.12.4 Turbine Component Cooling Water System

No entry for this system.

2.12.5 Chilled Water System

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. In addition, the NICWS is subject to additional regulatory oversight for providing cooling support for certain safety-related HVAC systems.

The functional arrangement of the NICWS is shown on Figure 2.12.5-1.

(1) The NICWS functional arrangement is described in the Design Description of Subsection 2.12.5.

(2) The NICWS provides the nonsafety-related function to support post-72 hour cooling for HVAC.

(3) The NICWS can be operated and controlled from the MCR.

(4) NICWS flow indication is provided in the MCR.

(5) The CWS has safety-related containment penetrations and isolation valves.

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 NICWS functional arrangement is described in the Design Description of Subsection 2.12.5 with the system layout as shown on Figure 2.12.5-1.	Inspection of the as-built system will be performed.	Report(s) document that the as-built NICWS System conforms to the functional arrangement described in the Design Description of this Section 2.12.5 with the system layout as shown on Figure 2.12.5-1.
2. The NICWS provides the nonsafety-related function to support post-72 hour cooling for HVAC.	Testing of the NICWS will be performed to demonstrate flow to the HVAC systems.	A report documents that the NICWS test demonstrates flow to the HVAC systems.
3. The NICWS can be operated and controlled from the MCR.	Testing to demonstrate flow capability will be performed on the NICWS components using controls in the MCR.	A report documents that the MCR controls caused the NICWS components to operate during the flow test.
4. NICWS flow indication is provided in the MCR.	Inspection will verify that NICWS flow indication can be retrieved in the MCR.	A report documents that the inspection verifies that NICWS flow indication can be retrieved in the MCR.
5. The CWS has safety-related containment penetrations and isolation valves.	See Tier 1, Subsection 2.15.1.	See Tier 1, Subsection 2.15.1.

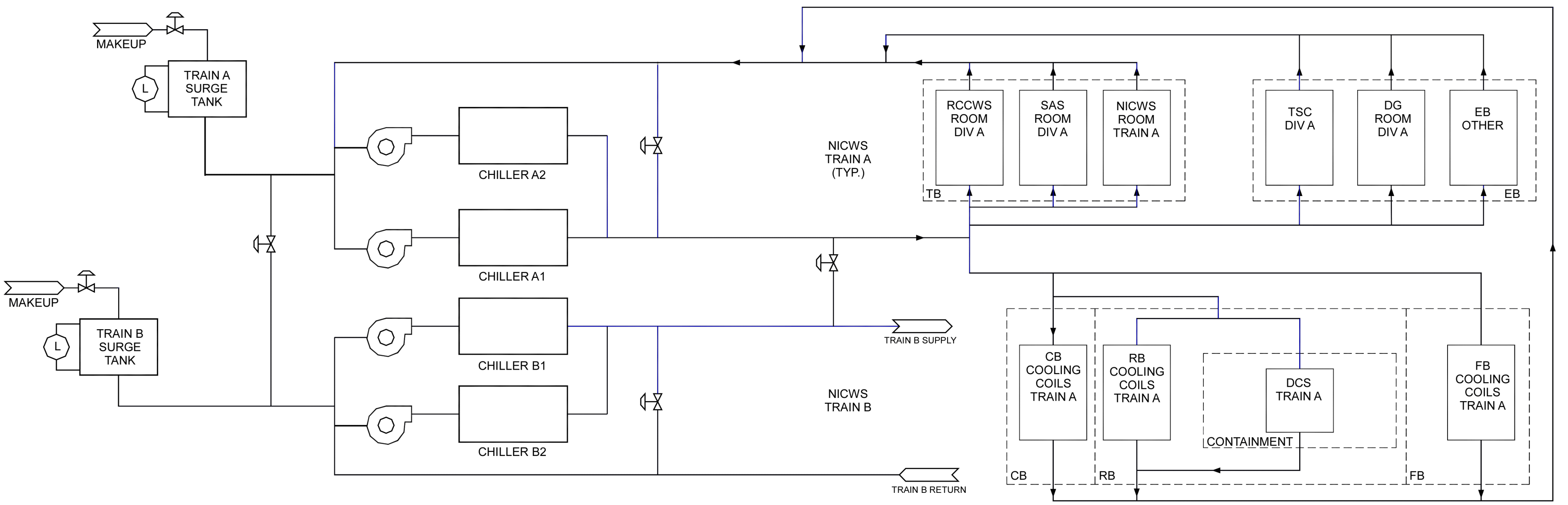


Figure 2.12.5-1. NICWS Layout

2.12.6 Oxygen Injection System

No entry for this system.

2.12.7 Plant Service Water System

Design Description

The PSWS does not perform or ensure any safety-related function, is not required to achieve or maintain safe shutdown, and has no interface with any safety-related component.

The functional arrangement of the PSWS is shown on Figure 2.12.7-1.

(1) The PSWS functional arrangement is described in the Design Description of Section 2.12.7 and is shown on Figure 2.12.7-1.

(2) The PSWS provides the nonsafety-related functions to support post-72 hour cooling for RCCWS.

(3) The PSWS can be operated and controlled from the MCR.

(4) PSWS flow indication is provided in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.12.7-1 provides definitions of the inspections, tests, and/or analyses, together with associated acceptance criteria for the PSWS.

Table 2.12.7-1

ITAAC For The Plant Service Water System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The PSWS functional arrangement is described in the Design Description of Section 2.12.7 and is shown on Figure 2.12.7-1.	Inspection of the as-built system will be performed.	Report(s) document that the as-built PSWS System conforms to the functional arrangement described in the Design Description of this Section 2.12.7 and as shown on Figure 2.12.7-1.
2. The PSWS provides the nonsafety-related function to support post-72 hour cooling for RCCWS.	Testing of PSWS will be performed to demonstrate flow to the RCCWS.	A report documents that the test of PSWS demonstrates flow to the RCCWS.
3.	Testing to demonstrate flow capability will be performed on the PSWS components using controls in the MCR.	A report documents that the MCR controls caused the PSWS components to operate during the flow test.
4. PSWS flow indication is provided in the MCR.	Inspection will verify that PSWS flow indication can be retrieved in the MCR.	A report documents that the inspection verifies that PSWS flow indication can be retrieved in the MCR.

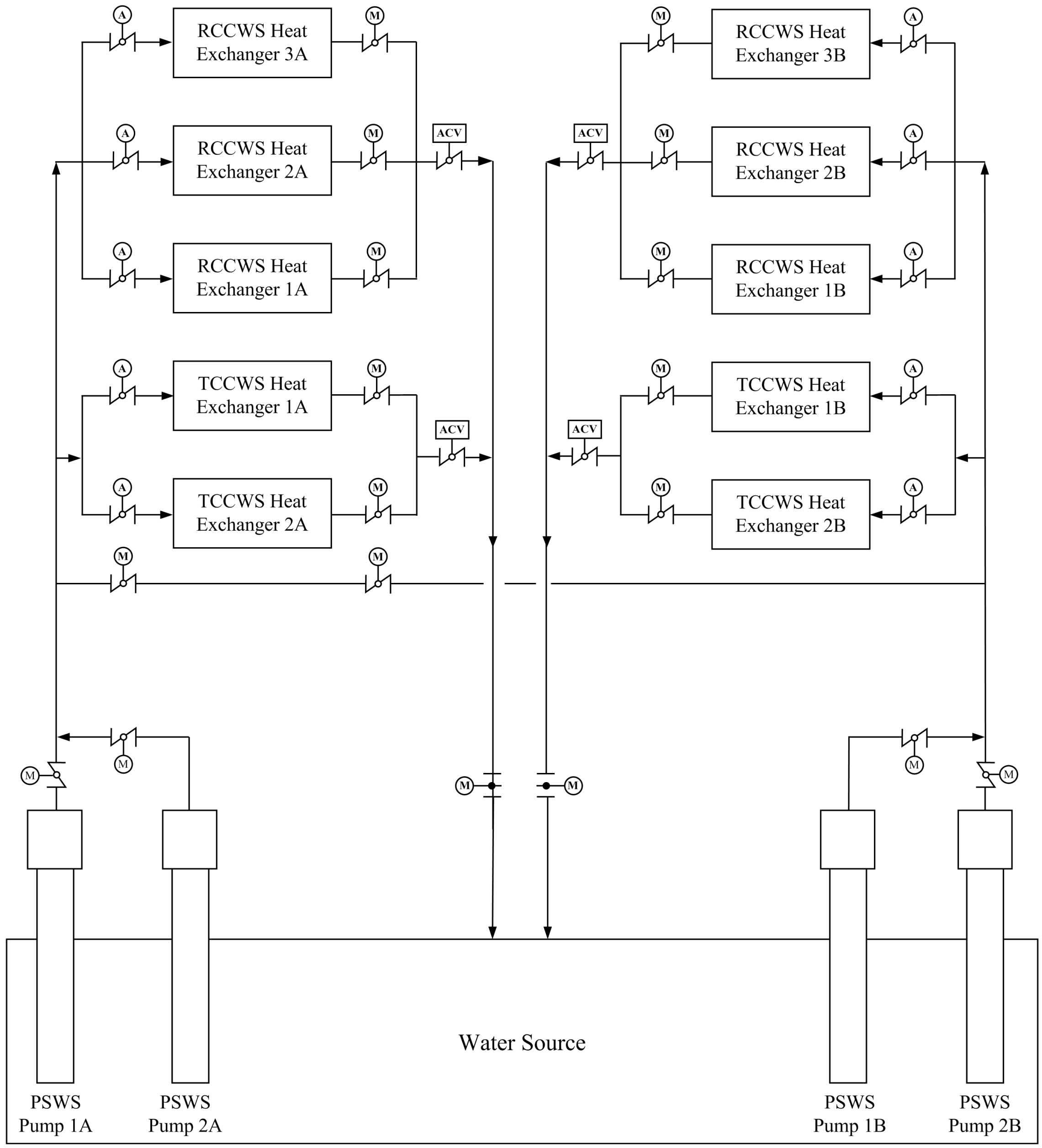


Figure 2.12.7-1. Plant Service Water System Arrangement Diagram

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 provisions for safety-related containment penetrations and isolation valves.

(1) The SAS has safety-related containment penetrations and isolation valves.

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.	See Tier 1, Subsection 2.15.1.	See Tier 1, Subsection 2.15.1.

2.12.9 Instrument Air System

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.

(1) The HPNSS has safety-related containment penetrations and isolation valves.

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.	See Tier 1, Subsection 2.15.1.	See Tier 1, Subsection 2.15.1.

2.12.11 Auxiliary Boiler System

No entry for this system.

2.12.12 Hot Water System

No entry for this system.

2.12.13 Hydrogen Water Chemistry System

This system is optional. If the optional system is implemented in a specific plant, there is no entry for this system.

2.12.14 Process Sampling System

No entry for this system.

2.12.15 Zinc Injection System

This system is optional. If the optional system is implemented in a specific plant, there is no entry for this system.

2.12.16 Freeze Protection

This system is optional. If the optional system is implemented in a specific plant, there is no entry for this system.

2.12.17 Station Water System

No entry for this system.

2.13 ELECTRICAL SYSTEMS

2.13.1 Onsite AC Power System

Design Description

The purpose of the Onsite AC Power System is to provide power to the power generation (PG) nonsafety-related loads and the plant's investment protection (PIP) nonsafety-related loads. The PIP buses supply power to the four (4) safety-related, 480VAC, Isolation Power Center buses. The nonsafety-related PIP buses have a Regulatory Treatment of Non-Safety Systems (RTNSS) function to supply power to RTNSS credited loads.

(1) The functional arrangement of Onsite AC Power System is as shown on Figure 2.13.1-1 and the component locations are shown in Table 2.13.1-1.

(2) The safety-related 480 VAC Isolation Power Center equipment identified in Table 2.13.1-1 conforms to Seismic Category I requirements and is housed in Seismic Category I structures.

(3) a. Independence is provided between safety-related divisions.

b. Separation is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.

c. Each safety-related Isolation Power Center supplies power to safety-related loads in its respective division.

(5) Isolation Power Centers and their associated loads are protected against undervoltage, degraded voltage and under- frequency conditions.

(6) The Onsite AC Power System provides the following nonsafety-related functions:

a. The Onsite AC Power System provides the capability for distributing nonsafety-related AC power from onsite sources to nonsafety-related RTNSS loads.

b. The Onsite AC Power System provides a PIP bus undervoltage signal to trip the PIP bus normal and alternate preferred power supply breakers, and start the standby diesel generator.

c. The standby power supply breaker closes when the standby diesel generator is ready to load.

(7) The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations and emergency procedure guidelines, is available in the main control room.

(8) Equipment qualification of safety-related 480 VAC Isolation Power Center equipment is addressed in DCD Tier 1 Section 3.8.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.13.1-2 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

System Equipment

Equipment Description	Location	Seismic Cat. I	Safety Related
Isolation Power Center Bus A Normal Main Circuit Breaker	Reactor Building	Yes	Yes
Isolation Power Center Bus A Alternate Main Circuit Breaker	Reactor Building	Yes	Yes
Isolation Power Center Bus B Normal Main Circuit Breaker	Reactor Building	Yes	Yes
Isolation Power Center Bus B Alternate Main Circuit Breaker	Reactor Building	Yes	Yes
Isolation Power Center Bus C Normal Main Circuit Breaker	Reactor Building	Yes	Yes
Isolation Power Center Bus C Alternate Main Circuit Breaker	Reactor Building	Yes	Yes
Isolation Power Center Bus D Normal Main Circuit Breaker	Reactor Building	Yes	Yes
Isolation Power Center Bus D Alternate Main Circuit Breaker	Reactor Building	Yes	Yes
Isolation Power Center Bus A1 Supply Breaker to Division 1 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Isolation Power Center Bus A2 Supply Breaker to Division 1 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Isolation Power Center Bus A3 Supply Breaker to Division 1 250 VDC Standby Battery Charger	Reactor Building	Yes	Yes
Isolation Power Center Bus B1 Supply Breaker to Division 2 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes

Table 2.13.1-1

System Equipment

Equipment Description	Location	Seismic Cat. I	Safety Related
Isolation Power Center Bus B2 Supply Breaker to Division 2 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Isolation Power Center Bus B3 Supply Breaker to Division 1 250 VDC Standby Battery Charger	Reactor Building	Yes	Yes
Isolation Power Center Bus C1 Supply Breaker to Division 3 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Isolation Power Center Bus C2 Supply Breaker to Division 3 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Isolation Power Center Bus C3 Supply Breaker to Division 3 250 VDC Standby Battery Charger	Reactor Building	Yes	Yes
Isolation Power Center Bus D1 Supply Breaker to Division 4 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Isolation Power Center Bus D2 Supply Breaker to Division 4 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Isolation Power Center Bus D3 Supply Breaker to Division 4 250 VDC Standby Battery Charger	Reactor Building	Yes	Yes

Table 2.13.1-2

ITAAC For The Onsite AC Power System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the Onsite AC Power System is as shown on Figure 2.13.1-1 and as described in section 2.13.1.	Inspections of the as-built Onsite AC Power System will be performed.	Inspection report(s) document that the as-built Onsite AC Power System conforms with the functional arrangement as shown in Figure 2.13.1-1 and as described in section 2.13.1.
2. The safety-related 480 VAC Isolation Power Center equipment identified in Table 2.13.1-1 - Conforms to Seismic Category I requirements, and - Is housed in Seismic Category I structures.	i) Type tests and/or analyses of the safety-related 480 VAC Isolation Power Center equipment will be performed. ii) Inspections of the as-built safety-related 480 VAC Isolation Power Centers will be performed to verify that the equipment is installed in accordance with the configurations specified in the type tests and/or analyses. iii) Inspections of the as-built safety-related 480 VAC Isolation Power Centers will be performed to verify that the equipment is housed in Seismic Category I structures.	i) A report exists and concludes that the as-built safety-related 480 VAC Isolation Power Center equipment conforms to Seismic Category I requirements. ii) Inspection report(s) document that the as-built safety-related 480 VAC Isolation Power Center equipment is installed in accordance with the configurations specified by the type tests and/or analyses. iii) Inspection report(s) document that the as-built safety-related 480 VAC Isolation Power Center equipment is housed in Seismic Category I structures.

Table 2.13.1-2

ITAAC For The Onsite AC Power System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3a. Independence is provided between safety-related divisions.	Tests will be performed on the as-built safety-related 480 VAC Isolation Power Centers by providing a test signal in only one safety-related division at a time.	Test report(s) document that a test signal exists only in the as-built safety-related division under test in the 480 VAC Isolation Power Center.
b. Separation is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	Inspection of the as-built safety-related 480 VAC Isolation Power Centers will be performed.	Inspection report(s) document that, for the as-built safety-related 480 VAC Isolation Power Centers, physical separation and electrical isolation exists between safety - related divisions. Physical separation and electrical isolation exists between safety-related divisions and nonsafety-related equipment.
4. Each safety-related Isolation Power Center supplies power to safety-related loads of their respective division.	Tests will be performed using a test signal to confirm that an electrical path exists from the as-built safety-related Isolation Power Center to its divisional safety-related loads. Each test may be a single test or a series of over-lapping tests.	Test report(s) demonstrate that a test signal originating from the as-built divisional Isolation Power Center exists at the terminals of its divisional safety-related load.
5. Isolation Power Centers and their associated loads are protected against undervoltage, degraded voltage and under- frequency conditions.	Testing will be performed using real or simulated signals.	Test report(s) document that the Isolation Power Centers are protected against undervoltage, degraded voltage and under- frequency conditions by applying a real or simulated signal and verifying that the as-built Isolation Power Center bus isolates from the nonsafety-related system.

Table 2.13.1-2

ITAAC For The Onsite AC Power System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6a. The Onsite AC Power System provides the capability for distributing nonsafety-related ac power from onsite sources to nonsafety-related RTNSS loads.	Tests will be performed using a test signal to confirm that an electrical path exists for each RTNSS load from its associated as-built PIP A or PIP B bus. Each test may be a single test or a series of over-lapping tests.	Test report(s) will demonstrate that a test signal originating from the as-built PIP A or PIP B bus exists at the terminals of each associated RTNSS load.
b. The Onsite AC Power System provides a PIP bus undervoltage signal to trip the PIP bus normal and alternate preferred power supply breakers and start the standby diesel generator.	Testing will be performed using real or simulated PIP bus undervoltage signals.	Test report(s) demonstrate that the as-built PIP bus normal and alternate preferred power supply breakers trip and the as-built standby onsite AC power source starts after receiving a real or simulated PIP bus undervoltage signal.
c. The standby power supply breaker closes when the standby diesel generator is ready to load.	Testing will be performed using real or simulated signals.	Test report(s) demonstrate that the as-built standby power supply breaker closes after receiving a real or simulated ready to load signal from the standby AC power system.
7. The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations and emergency procedure guidelines, is available in the main control room.	Inspection of the as-built main control room will verify that the minimum set of displays, alarms and controls for the Onsite AC Power System is available.	Inspection report(s) document that the minimum set of displays, alarms and controls for the Onsite AC Power System, as defined by the applicable codes and standards, including HFE evaluations and emergency operating procedures, exist in the as-built main control room.

Table 2.13.1-2

ITAAC For The Onsite AC Power System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Equipment qualification of safety-related 480 VAC Isolation Power Center equipment is addressed in DCD Tier 1 Section 3.8.	See Tier 1 Section 3.8.	See Tier 1 Section 3.8.

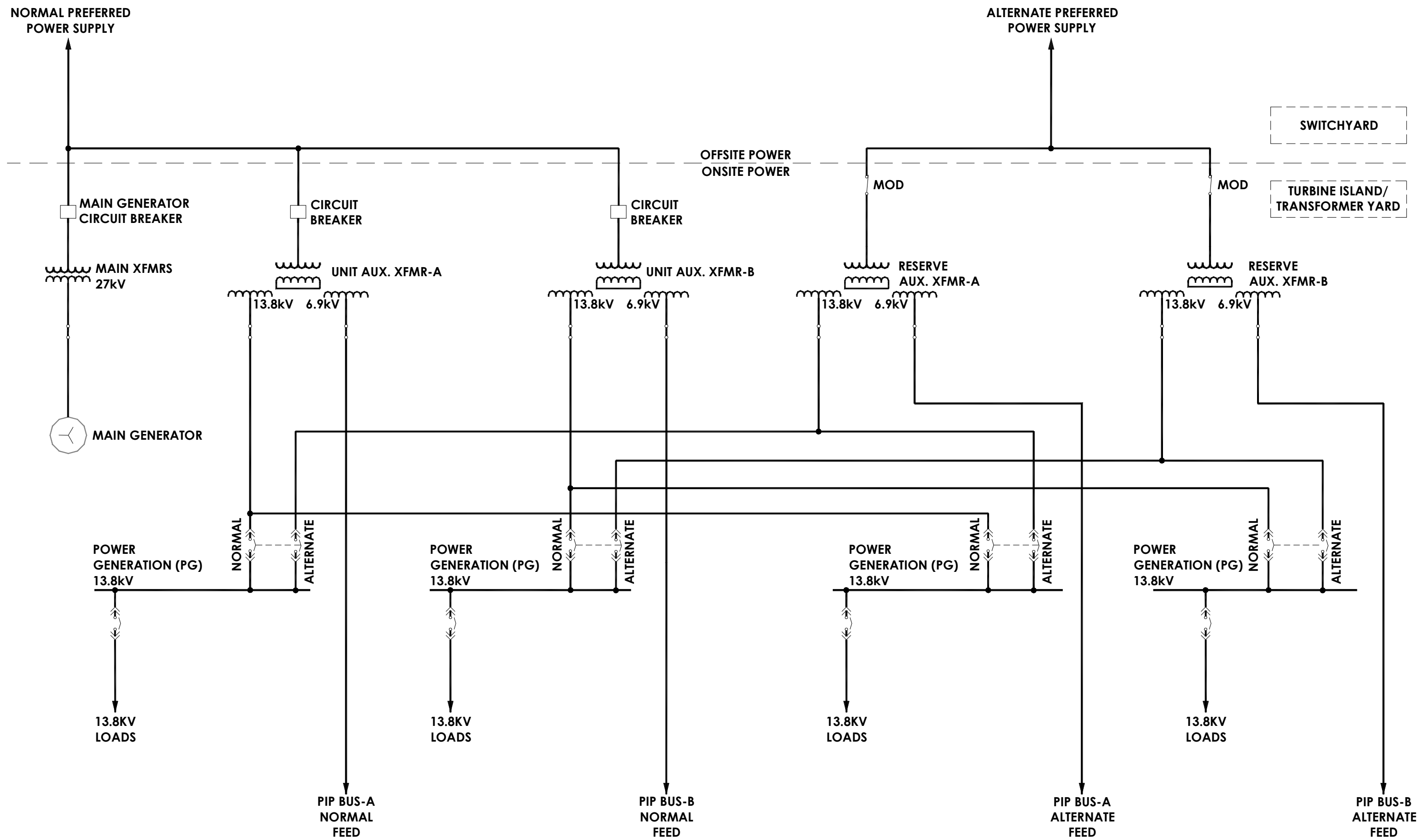


Figure 2.13.1-1 Sh 1.
Onsite AC Power System

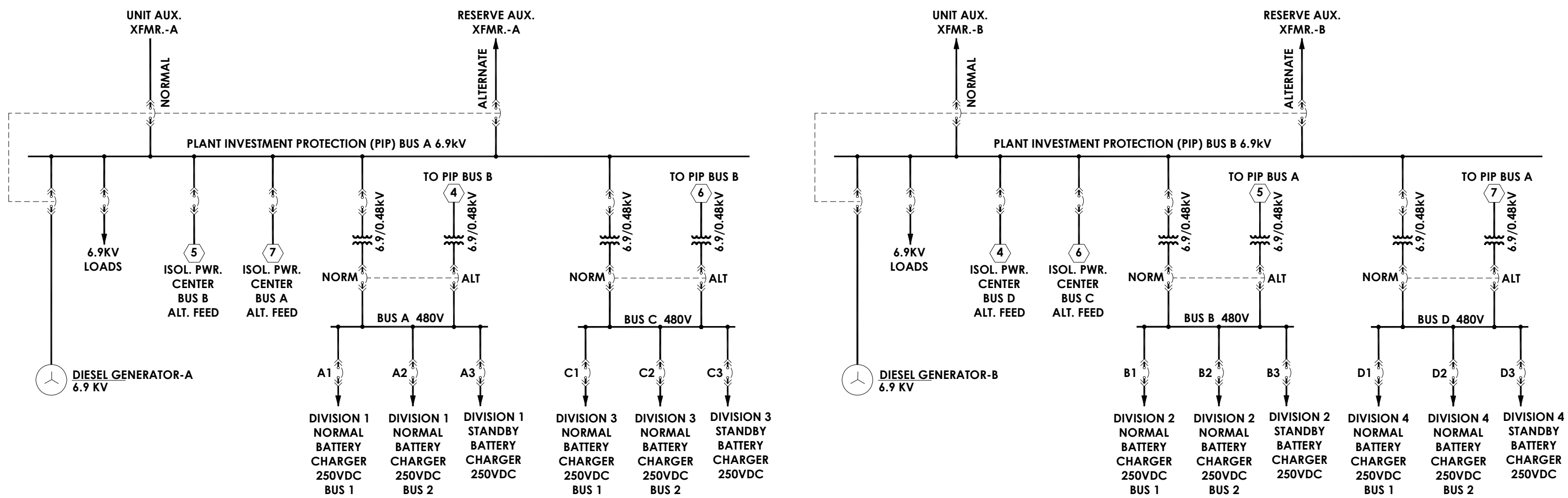


Figure 2.13.1-1 Sh 2.
Onsite AC Power System

2.13.2 Electrical Wiring Penetrations

Design Description – Primary Containment Electrical Wiring Penetrations

Moved to Subsection 2.15.1, "Containment".

Fire Barrier Penetrations

Moved to Subsection 2.16.3.1, "Fire Barriers".

Inspections, Tests, Analyses and Acceptance Criteria

Moved to Subsection 2.15.1, "Containment".

2.13.3 Direct Current Power Supply

Design Description

Completely independent safety-related and nonsafety-related DC power systems are provided.

Nonsafety-related DC power systems are not part of the plant safety design basis, and are independent and separated from the safety-related DC power supplies.

The 250 V Safety-Related DC systems provide four divisions of power to operate safety–related loads for at least 72 hours following a design basis accident. The 250V Safety-Related DC systems are also adequately sized for the station blackout loads.

(1) The functional arrangement of the 250V Safety-Related DC systems is as shown on Figure 2.13.3-1 and the component locations are shown in Table 2.13.3-1.

(2) The functional arrangement of the 125 V and 250 V Nonsafety-Related DC systems is as shown on Figure 2.13.3-2.

(3) The 250V Safety-Related DC systems identified in Table 2.13.3-1 conform to Seismic Category I requirements and are housed in Seismic Category I structures.

(4) The 250 V Safety-Related DC systems provide four independent and redundant safety-related divisions.

(5) Separation is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.

(6) The two 250 VDC safety-related batteries in each division are each capable of supplying power to their safety-related loads for at least 72 hours following a design basis accident.

(7) Each battery charger associated with each 250 VDC safety-related battery has sufficient capacity to meet the largest combined demands of the various continuous steady-state loads plus the charging capacity to restore the battery from the design minimum charge state to the fully charged state within the time stated in the design basis, consistent with the requirement given in IEEE 308.

(8) The 250 V Safety-Related DC battery and battery charger circuit breakers, and DC distribution panels and their circuit breakers and fuses, are sized to supply their load requirements.

(9) The battery chargers are designed to prevent their AC source from becoming a load on the 250 VDC safety-related batteries because of power feedback from loss of AC power.

(10) The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations and emergency procedure guidelines, is available in the main control room..

(11) Equipment qualification of the 250 V Safety-Related DC systems is addressed in DCD Tier 1 Section 3.8.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.13.3-3 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

System Equipment

Equipment Description	Location	Seismic Cat. I	Safety Related
Division 1 Bus 1 250 VDC Battery	Reactor Building	Yes	Yes
Division 1 Bus 2 250 VDC Battery	Reactor Building	Yes	Yes
Division 2 Bus 1 250 VDC Battery	Reactor Building	Yes	Yes
Division 2 Bus 2 250 VDC Battery	Reactor Building	Yes	Yes
Division 3 Bus 1 250 VDC Battery	Reactor Building	Yes	Yes
Division 3 Bus 2 250 VDC Battery	Reactor Building	Yes	Yes
Division 4 Bus 1 250 VDC Battery	Reactor Building	Yes	Yes
Division 4 Bus 2 250 VDC Battery	Reactor Building	Yes	Yes
Division 1 Bus 1 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Division 1 Bus 2 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Division 1 250 VDC Standby Battery Charger	Reactor Building	Yes	Yes
Division 2 Bus 1 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Division 2 Bus 2 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Division 2 250 VDC Standby Battery Charger	Reactor Building	Yes	Yes
Division 3 Bus 1 250 VDC Normal Battery	Reactor	Yes	Yes

Table 2.13.3-1

System Equipment

Equipment Description	Location	Seismic Cat. I	Safety Related
Charger	Building		
Division 3 Bus 2 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Division 3 250 VDC Standby Battery Charger	Reactor Building	Yes	Yes
Division 4 Bus 1 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Division 4 Bus 2 250 VDC Normal Battery Charger	Reactor Building	Yes	Yes
Division 4 250 VDC Standby Battery Charger	Reactor Building	Yes	Yes
Division 1 Bus 1 250 VDC Power Center	Reactor Building	Yes	Yes
Division 1 Bus 2 250 VDC Power Center	Reactor Building	Yes	Yes
Division 2 Bus 1 250 VDC Power Center	Reactor Building	Yes	Yes
Division 2 Bus 2 250 VDC Power Center	Reactor Building	Yes	Yes
Division 3 Bus 1 250 VDC Power Center	Reactor Building	Yes	Yes
Division 3 Bus 2 250 VDC Power Center	Reactor Building	Yes	Yes
Division 4 Bus 1 250 VDC Power Center	Reactor Building	Yes	Yes
Division 4 Bus 2 250 VDC Power Center	Reactor Building	Yes	Yes
Division 1 Bus 1 250 VDC Transfer Switch Box	Reactor Building	Yes	Yes
Division 1 Bus 2 250 VDC Transfer Switch	Reactor	Yes	Yes

Table 2.13.3-1

System Equipment

Equipment Description	Location	Seismic Cat. I	Safety Related
Box	Building		
Division 2 Bus 1 250 VDC Transfer Switch Box	Reactor Building	Yes	Yes
Division 2 Bus 2 250 VDC Transfer Switch Box	Reactor Building	Yes	Yes
Division 3 Bus 1 250 VDC Transfer Switch Box	Reactor Building	Yes	Yes
Division 3 Bus 2 250 VDC Transfer Switch Box	Reactor Building	Yes	Yes
Division 4 Bus 1 250 VDC Transfer Switch Box	Reactor Building	Yes	Yes
Division 4 Bus 2 250 VDC Transfer Switch Box	Reactor Building	Yes	Yes

Table 2.13.3-2

Equipment Displays/Status Indication

Equipment Description	Display/Status Indication
Division 1 Bus 1 250 VDC Battery	Yes
Division 1 Bus 2 250 VDC Battery	Yes
Division 2 Bus 1 250 VDC Battery	Yes
Division 2 Bus 2 250 VDC Battery	Yes
Division 3 Bus 1 250 VDC Battery	Yes
Division 3 Bus 2 250 VDC Battery	Yes
Division 4 Bus 1 250 VDC Battery	Yes
Division 4 bus 2 250 VDC Battery	Yes
Division 1 Bus 1 250 VDC Normal Battery Charger	Yes
Division 1 Bus 2 250 VDC Normal Battery Charger	Yes
Division 1 250 VDC Standby Battery Charger	Yes
Division 2 Bus 1 250 VDC Normal Battery Charger	Yes
Division 2 Bus 2 250 VDC Normal Battery Charger	Yes
Division 2 250 VDC Standby Battery Charger	Yes
Division 3 Bus 1 250 VDC Normal Battery Charger	Yes
Division 3 Bus 2 250 VDC Normal Battery Charger	Yes
Division 3 250 VDC Standby Battery Charger	Yes
Division 4 Bus 1 250 VDC Normal Battery Charger	Yes
Division 4 Bus 2 250 VDC Normal Battery Charger	Yes
Division 4 250 VDC Standby Battery Charger	Yes

Table 2.13.3-3

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the 250V Safety-Related DC systems is as shown on Figure 2.13.3-1 and the component locations are shown in Table 2.13.3-1.	Inspections of the as-built 250 V Safety-Related DC systems will be performed.	Inspection report(s) document that the as-built 250 V Safety-Related DC systems conform with the functional arrangement as shown in Figure 2.13.3-1 and as described in section 2.13.3
2. The functional arrangement of the 125 V and 250V Nonsafety-Related DC systems is as shown on Figure 2.13.3-2 and as described in section 2.13.3.	Inspections of the as-built 125 V and 250 V Nonsafety-Related DC systems will be performed.	Inspection report(s) document that the as-built 125 V and 250 V Nonsafety-Related DC systems conform with the functional arrangement as shown in Figure 2.13.3-2 and as described in section 2.13.3

Table 2.13.3-3

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The 250V Safety-Related DC systems equipment identified in Table 2.13.3-1 - Conform to Seismic Category I requirements, and - Are housed in Seismic Category I structures.	i) Type tests and/or analyses of the 250V Safety-Related DC system equipment will be performed. ii) Inspections of the as-built 250V Safety-Related DC systems will be performed to verify that the equipment is installed in accordance with the configurations specified in the type tests and/or analyses. iii) Inspections of the as-built 250V Safety-Related DC systems will be performed to verify that the equipment is housed in Seismic Category I structures.	i) A report exists and concludes that the as-built 250V Safety-Related DC system equipment conforms to Seismic Category I requirements. ii) Inspection report(s) document that the as-built 250V Safety-Related DC system equipment is installed in accordance with the configurations specified by the type tests and/or analyses. iii) Inspection report(s) document that the as-built 250V Safety-Related DC system equipment is housed in Seismic Category I structures.
4. The 250 V Safety-Related DC systems provide four independent and redundant safety-related divisions.	Tests will be performed on the as-built 250 V Safety-Related DC systems by providing a test signal in only one safety-related division at a time.	i) Test report (s) demonstrate that a test signal exists only in the as-built safety-related division under test in the 250 V Safety-Related DC systems. ii) Test report(s) demonstrate that a test signal originating from the as-built divisional safety-related 250 VDC distribution panel exists at the terminals of its divisional safety-related loads.

Table 2.13.3-3

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Separation is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	Inspection of the as-built 250 V Safety-Related DC systems will be performed.	Inspection report(s) document that, in the as-built 250 V Safety-Related DC systems, physical separation or electrical isolation exists between safety–related divisions. Physical separation or electrical isolation exists between safety-related divisions and nonsafety-related equipment.
6. The two 250 VDC safety-related batteries in each division are each capable of supplying power to their safety-related loads for at least 72 hours following a design basis accident.	A service test will be performed on each divisional Safety-Related 250 VDC battery in accordance with the applicable IEEE standard 1188.	Test report(s) document that the results of the service test meet the design basis requirement capacity as defined in the applicable IEEE standard 1188.
7. Each battery charger associated with each 250 VDC safety-related battery has sufficient capacity to meet the largest combined demands of the various continuous steady-state loads plus the charging capacity to restore the battery from the design minimum charge state to the fully charged state within the time stated in the design basis, consistent with the requirement given in IEEE 308.	Testing of each 250 VDC safety-related battery charger will be performed.	Test report(s) document that following a battery discharge to the bounding design basis accident discharge state, the battery charger is capable of recharging its associated battery to the fully charged state while supplying the largest combined demands of the various continuous steady state simulated and /or real loads consistent with the requirement given in IEEE 308.

Table 2.13.3-3

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The safety-related DC battery and battery charger circuit breakers, and DC distribution panels and their circuit breakers and fuses, are sized to supply their load requirements.	Analyses of the as-built 250V Safety-Related DC electrical distribution system will be performed to determine the capacities of the battery and battery charger circuit breakers, and DC distribution panels and their circuit breakers and fuses.	Analyses for the as-built 250V Safety-Related DC electrical distribution system exist, and analysis report(s) conclude that the capacities of safety-related battery and battery charger circuit breakers, and DC distribution panels and their circuit breakers and fuses, as determined by their nameplate ratings, exceed their analyzed load and DC interrupting current requirements.
9. The battery chargers are designed to prevent their AC source from becoming a load on the 250 VDC safety-related batteries because of power feedback from loss of AC power.	Testing of each 250 VDC safety-related battery charger will be performed to demonstrate that there is no power feedback from a loss of AC input power.	Test report(s) document that the 250 VDC safety-related battery charger prevents the AC input source from becoming a load on the 250 VDC safety-related batteries during a loss of AC power condition.
10. The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations and emergency procedure guidelines, is available in the main control room.	Inspection of the as-built main control room will verify that the minimum set of displays, alarms and controls for the 250 V Safety-Related DC systems are available.	Inspection report(s) document that the minimum set of displays, alarms and controls for the 250 V Safety-Related DC systems, as defined by the applicable codes and standards, including HFE evaluations and emergency operating procedures, exist in the as-built main control room.

Table 2.13.3-3

ITAAC For The Direct Current Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. Equipment qualification of the 250 V Safety-Related DC System is addressed in DCD Tier 1 Section 3.8.	See Tier 1 Section 3.8.	See Tier 1 Section 3.8.

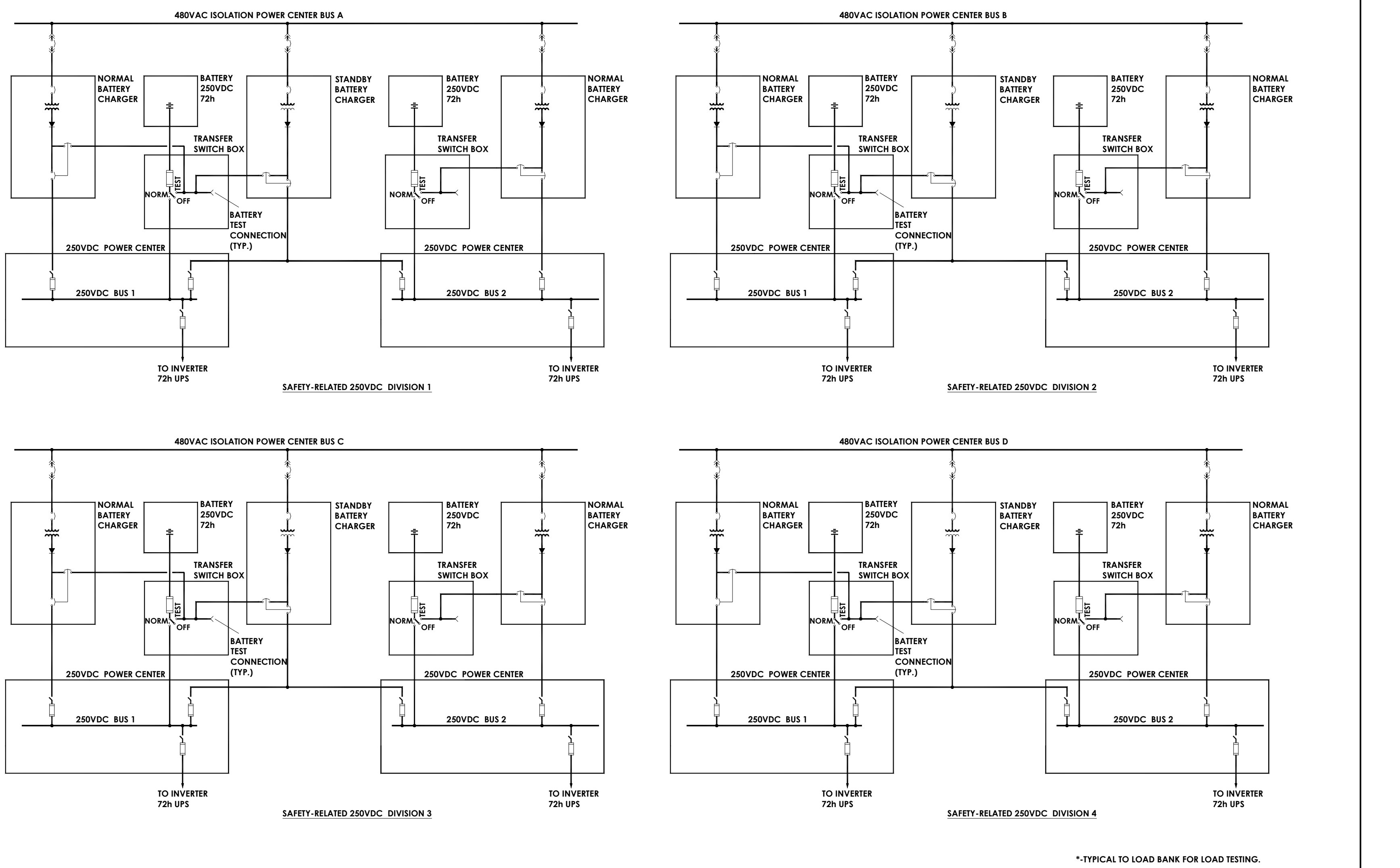


Figure 2.13.3-1.
Safety-Related 250 VDC System

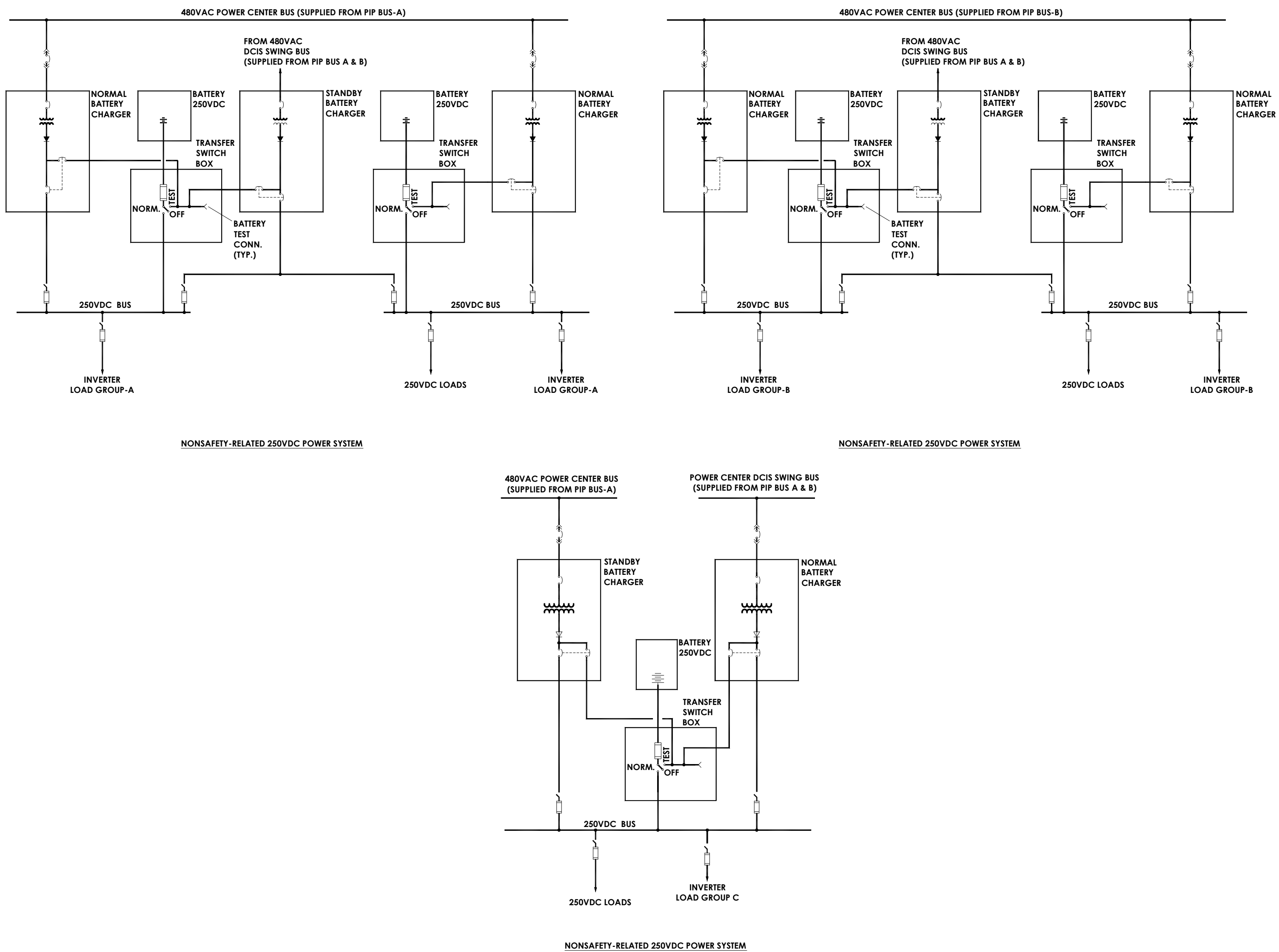


Figure 2.13.3-2 Sh 1.
Nonsafety-Related 250 VDC System

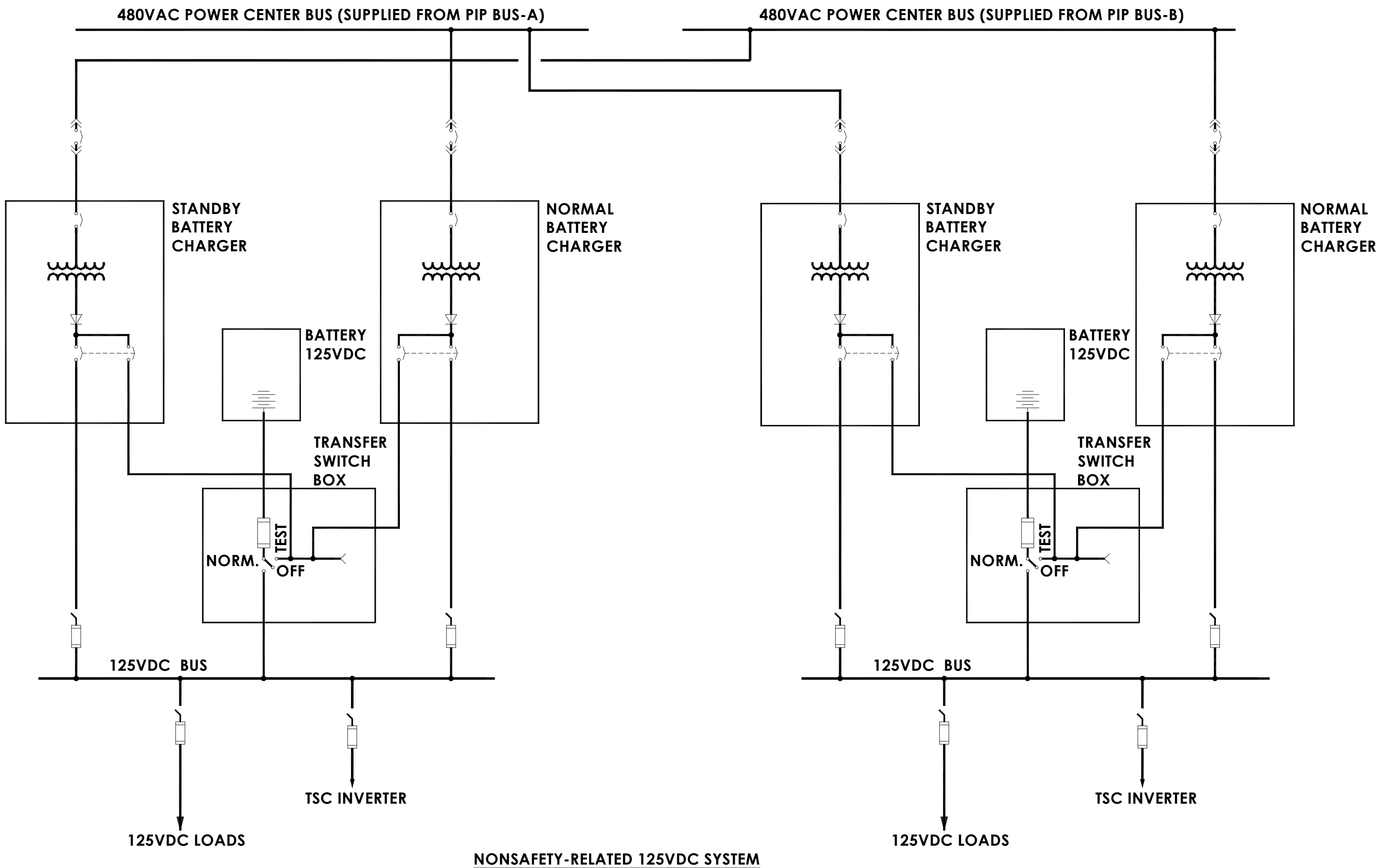


Figure 2.13.3-2 Sh 2.
Nonsafety-Related 125 VDC System

2.13.4 Standby On Site Power Supply

Design Description

Two independent nonsafety-related standby AC diesel generators, including their support systems, provide separate sources of on-site power for the nonsafety-related Plant Investment Protection (PIP) load groups when the normal and alternate preferred 6.9kV power supplies are not available. The nonsafety-related standby diesel generators have a Regulatory Treatment of Non-Safety Systems (RTNSS) function to provide power to the PIP buses that supply RTNSS credited loads.

(1) The functional arrangement of Standby On Site Power System is as described in the Design Description of this Section 2.13.4 and the component locations are shown in Table 2.13.4-1.

(2) The Standby On Site Power Supply System provides the following nonsafety-related functions:

   a. Upon receipt of an undervoltage signal from the On Site AC Power System, the standby diesel generator starts, achieves rated speed and voltage, and produces a ready to load signal.

   b. Each standby diesel generator is sized to accommodate its expected loads.

   c. Each standby diesel generator fuel oil storage tank contains adequate fuel oil capacity for 7 days of standby diesel generator operation.

   d. Each of the standby diesel generator fuel oil transfer pumps (two pumps per engine) start automatically and transfer fuel oil from the fuel oil storage tank to the standby diesel generator day tank at a rate greater than or equal to the usage rate of the standby diesel generator.

   e. Each of the standby diesel generator starting air receivers (two receivers per engine) is capable of three engine start attempts.

(3) The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations emergency procedure guidelines, is available in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.13.4-2 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria for the Standby On Site Power Supply System.

Table 2.13.4-1

Equipment Location

Equipment Description	Location
Standby Diesel Generator A	Electrical Building
Standby Diesel Generator B	Electrical Building

Table 2.13.4-2

ITAAC For The Standby On Site Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the Standby Onsite Power Supply is as described in Section 2.13.4	Inspections of the as-built system will be conducted.	Inspection report(s) document that the as-built Standby Onsite Power Supply system conform with the functional arrangement as described in section 2.13.4.
2a. Upon receipt of an undervoltage signal from the On Site AC Power System, the standby diesel generator starts, achieves rated speed and voltage, and produces a ready to load signal.	Tests of the as-built Standby Onsite Power Supply system will be conducted by providing a real or simulated undervoltage signal to start the standby diesel generators.	Test report(s) demonstrate that the as-built standby diesel generator starts upon receipt of a real or simulated undervoltage signal on its associated PIP bus, achieves rated speed and voltage within one minute of starting, and produces a ready to load signal that is available to close its associated standby power supply breaker to the PIP bus.
b. Each standby diesel generator is sized to accommodate its expected loads.	Each as-built standby diesel generator will be operated between rated and maximum nameplate load, and nameplate power factor for a time period required to reach engine temperature equilibrium.	Test report(s) demonstrate that each as-built standby diesel generator provides power at generator terminal rated voltage and frequency when fully loaded.
c. Each standby diesel generator fuel oil storage tank contains adequate fuel oil capacity for 7 days of standby diesel generator operation.	The as-built fuel oil storage tank capacity will be calculated.	Analysis demonstrates that the as-built fuel oil storage tank capacity is adequate to supply 7 days of fuel oil to the standby diesel generator that is operating at rated load.

Table 2.13.4-2

ITAAC For The Standby On Site Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
d. Each of the standby diesel generator fuel oil transfer pumps starts automatically and transfers fuel oil from the fuel oil storage tank to the standby diesel generator day tank at a rate greater than or equal to the usage rate of the standby diesel generator.	Testing will be performed to demonstrate that each as-built fuel oil transfer pump starts automatically and transfers fuel oil from the fuel oil storage tank to the standby diesel generator day tank at a rate greater than or equal to the usage rate of the standby diesel generator when operating between rated and maximum nameplate load.	Test report(s) demonstrate that that each as-built fuel oil transfer pump starts automatically and transfers fuel oil from the fuel oil storage tank to the standby diesel generator day tank at a rate greater than or equal to the usage rate of the standby diesel generator when running fully loaded.
e. Each of the standby diesel generator starting air receivers is capable of three engine start attempts.	Testing will be performed for each as-built starting air receiver.	Test report(s) demonstrate that each as-built starting air receiver is capable of three engine start attempts without recharging.
3. The minimum set of displays, alarms and controls, based on the applicable codes and standard, including HFE evaluations and emergency procedure guidelines, is available in the main control room.	Inspection of the as-built main control room will verify that the minimum set of displays, alarms and controls for the Standby Onsite Power Supply system is available.	Inspection report(s) document that the minimum set of displays, alarms and controls for the Standby Onsite Power Supply system, as defined by the applicable codes and standards, including HFE evaluations and emergency operating procedure guidelines, exists in the as-built main control room.

2.13.5 Uninterruptible AC Power Supply

Design Description

The Uninterruptible AC Power Supply (UPS) is divided into two subsystems, the safety-related UPS and the nonsafety-related UPS.

The nonsafety-related UPS system and the nonsafety-related Technical Support Center UPS system are not part of the plant safety design basis, and are independent and separated from the safety-related UPS system.

The safety-related UPS system provides four divisions of 120 VAC power to safety-related loads during normal, upset and accident conditions.

(1) The functional arrangement of the safety-related UPS system is as shown on Figure 2.13.5-1 and the component locations are shown in Table 2.13.5-1.

(2) The functional arrangement of the nonsafety-related UPS system is as shown on Figure 2.13.5-2, and as described in Section 2.13.5.

(3) The safety-related UPS system equipment identified in Table 2.13.5-1 conforms to Seismic Category I requirements and is housed in Seismic Category I structures.

(4) The safety-related UPS system provides four independent and redundant safety-related divisions.

(5) Separation is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.

(6) Each safety-related UPS inverter is capable of supplying its ac load.

(7) The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations and emergency procedure guidelines, is available in the main control room.

(8) Equipment qualification of the safety-related UPS system is addressed in DCD Tier 1 Section 3.8.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.13.5-2 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

System Equipment

Equipment Description	Location	Seismic Cat. I	Safety Related
Division 1 UPS Bus 1 480 VAC to UPS Rectifier	Reactor Building	Yes	Yes
Division 1 UPS Bus 2 480 VAC to UPS Rectifier	Reactor Building	Yes	Yes
Division 2 UPS Bus 1 480 VAC to UPS Rectifier	Reactor Building	Yes	Yes
Division 2 UPS Bus 2 480 VAC to UPS Rectifier	Reactor Building	Yes	Yes
Division 3 UPS Bus 1 480 VAC to UPS Rectifier	Reactor Building	Yes	Yes
Division 3 UPS Bus 2 480 VAC to UPS Rectifier	Reactor Building	Yes	Yes
Division 4 UPS Bus 1 480 VAC to UPS Rectifier	Reactor Building	Yes	Yes
Division 4 UPS Bus 2 480 VAC to UPS Rectifier	Reactor Building	Yes	Yes
Division 1 Bus 1 UPS	Reactor Building	Yes	Yes
Division 1 Bus 2 UPS	Reactor Building	Yes	Yes
Division 2 Bus 1 UPS	Reactor Building	Yes	Yes
Division 2 Bus 2 UPS	Reactor Building	Yes	Yes
Division 3 Bus 1 UPS	Reactor Building	Yes	Yes
Division 3 Bus 2 UPS	Reactor Building	Yes	Yes
Division 4 Bus 1 UPS	Reactor Building	Yes	Yes
Division 4 Bus 2 UPS	Reactor Building	Yes	Yes

Table 2.13.5-1

System Equipment

Equipment Description	Location	Seismic Cat. I	Safety Related
Division 1 UPS Bus 1 Power Distribution Panel	Reactor Building	Yes	Yes
Division 1 UPS Bus 2 Power Distribution Panel	Reactor Building	Yes	Yes
Division 2 UPS Bus 1 Power Distribution Panel	Reactor Building	Yes	Yes
Division 2 UPS Bus 2 Power Distribution Panel	Reactor Building	Yes	Yes
Division 3 UPS Bus 1 Power Distribution Panel	Reactor Building	Yes	Yes
Division 3 UPS Bus 2 Power Distribution Panel	Reactor Building	Yes	Yes
Division 4 UPS Bus 1 Power Distribution Panel	Reactor Building	Yes	Yes
Division 4 UPS Bus 2 Power Distribution Panel	Reactor Building	Yes	Yes

Table 2.13.5-2

ITAAC For The Uninterruptible AC Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the safety-related UPS system is as shown on Figure 2.13.5-1 and as described in section 2.13.5.	Inspections of the as-built safety-related UPS system will be performed.	Inspection report(s) document that the as-built safety-related UPS system conforms with the functional arrangement as shown in Figure 2.13.5-1 and as described in section 2.13.5.
2. The functional arrangement of the nonsafety-related UPS system is as shown on Figure 2.13.5-2 and as described in section 2.13.5.	Inspections of the as-built nonsafety-related UPS system will be performed.	Inspection report(s) document that the as-built nonsafety-related UPS system conforms with the functional arrangement as shown in Figure 2.13.5-2 and as described in section 2.13.5.

Table 2.13.5-2

ITAAC For The Uninterruptible AC Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The safety-related UPS system - conforms to Seismic Category I requirements, and - is housed in Seismic Category I structures.	i) Type tests and/or analyses of the safety-related UPS system seismic Category I equipment will be performed. ii) Inspections of the as-built safety-related UPS system will be performed to verify that the equipment is installed in accordance with the configurations specified in the type tests and/or analyses. iii) Inspections of the as-built safety-related UPS system will be performed to verify that the equipment is housed in Seismic Category I structures.	i) Test report(s)and/or analyses conclude that the as-built safety-related UPS system conforms to Seismic Category I requirements. ii) Inspection report(s) document that the as-built safety-related UPS system equipment is installed in accordance with the configurations specified by the type tests and/or analyses. iii) Inspection report(s) document that the seismic Category 1 equipment identified in Table 2.13.5-1 is housed in Seismic Category I structures.
4. Independence is provided between safety-related divisions.	Tests will be performed on the as-built safety-related UPS system by providing a test signal in only one safety-related division at a time.	i) Test report(s) demonstrate that a test signal exists only in the safety-related division under test in the as-built safety-related UPS system. ii) Test report(s) demonstrate that a test signal originating from the as-built divisional safety-related UPS distribution panel exists at the terminals of its divisional safety-related loads.

Table 2.13.5-2

ITAAC For The Uninterruptible AC Power Supply

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Separation is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	Inspection of the as-built safety-related UPS system will be performed.	Inspection report(s) document that, in the as-built safety-related UPS system, physical separation or electrical isolation exists between safety–related divisions. Physical separation or electrical isolation exists between safety-related divisions and nonsafety-related equipment.
6. Each safety-related UPS inverter is capable of supplying its ac load.	Testing of each as-built safety-related UPS inverter will be performed by applying a combination of simulated and/or real loads. The inverter input voltage at the inverter input terminals will be no more than 210 VDC during the test.	Test report(s) demonstrate that the as-built safety-related UPS inverter supplies its rated voltage at its rated frequency.
7. The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations and emergency procedure guidelines, is available in the main control room.	Inspection of the as-built main control room will verify that the minimum set of displays, alarms and controls for the safety-related UPS system are available.	Inspection report(s) document that the minimum set of displays, alarms and controls for the safety-related UPS system, as defined by the applicable codes and standards, including HFE evaluations and emergency operating procedures, exist in the as-built main control room.
8. Equipment qualification of the safety-related UPS system is addressed in DCD Tier 1 Section 3.8.	See Tier 1 Section 3.8.	See Tier 1 Section 3.8.

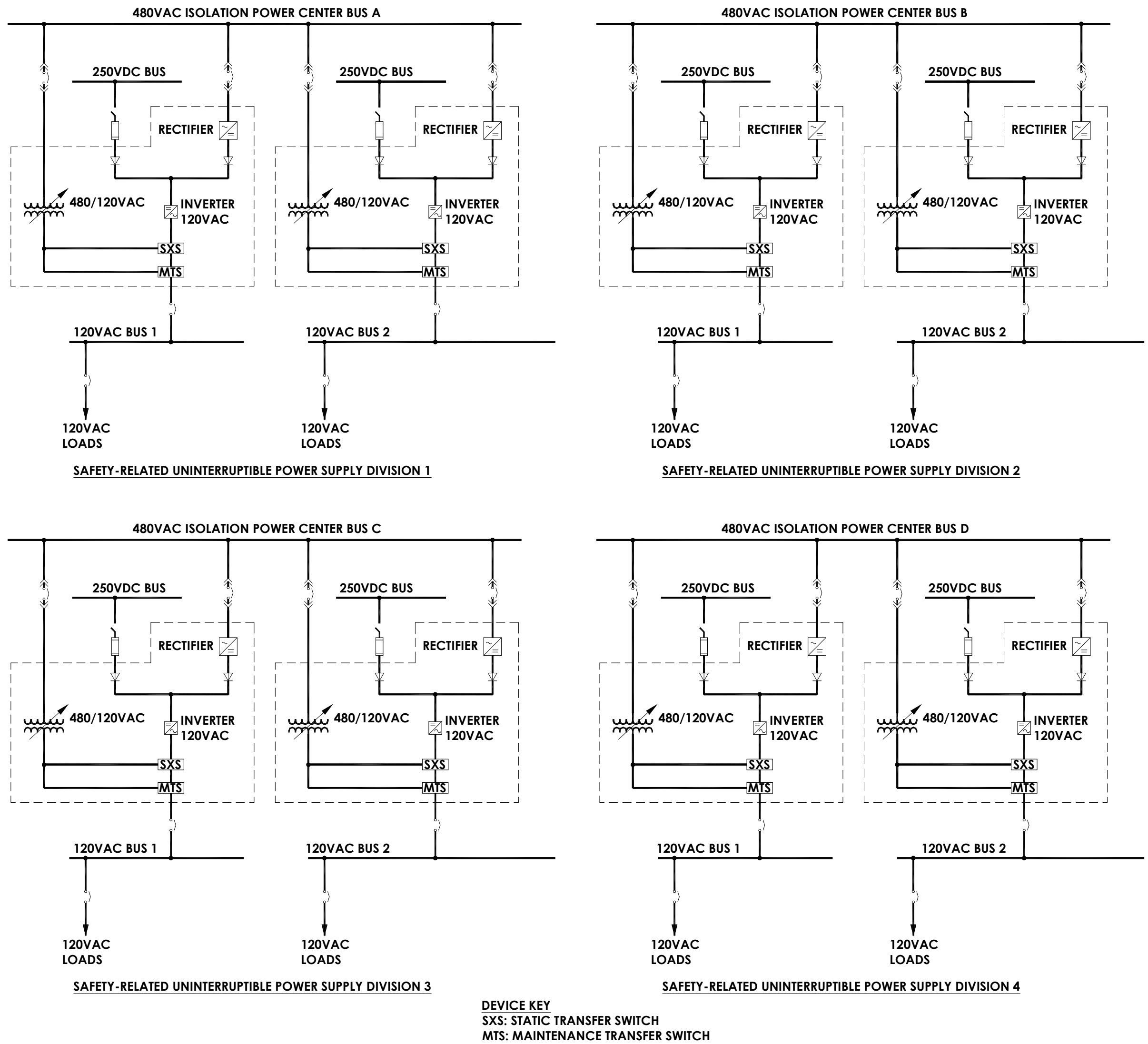


Figure 2.13.5-1.
Safety-Related UPS System

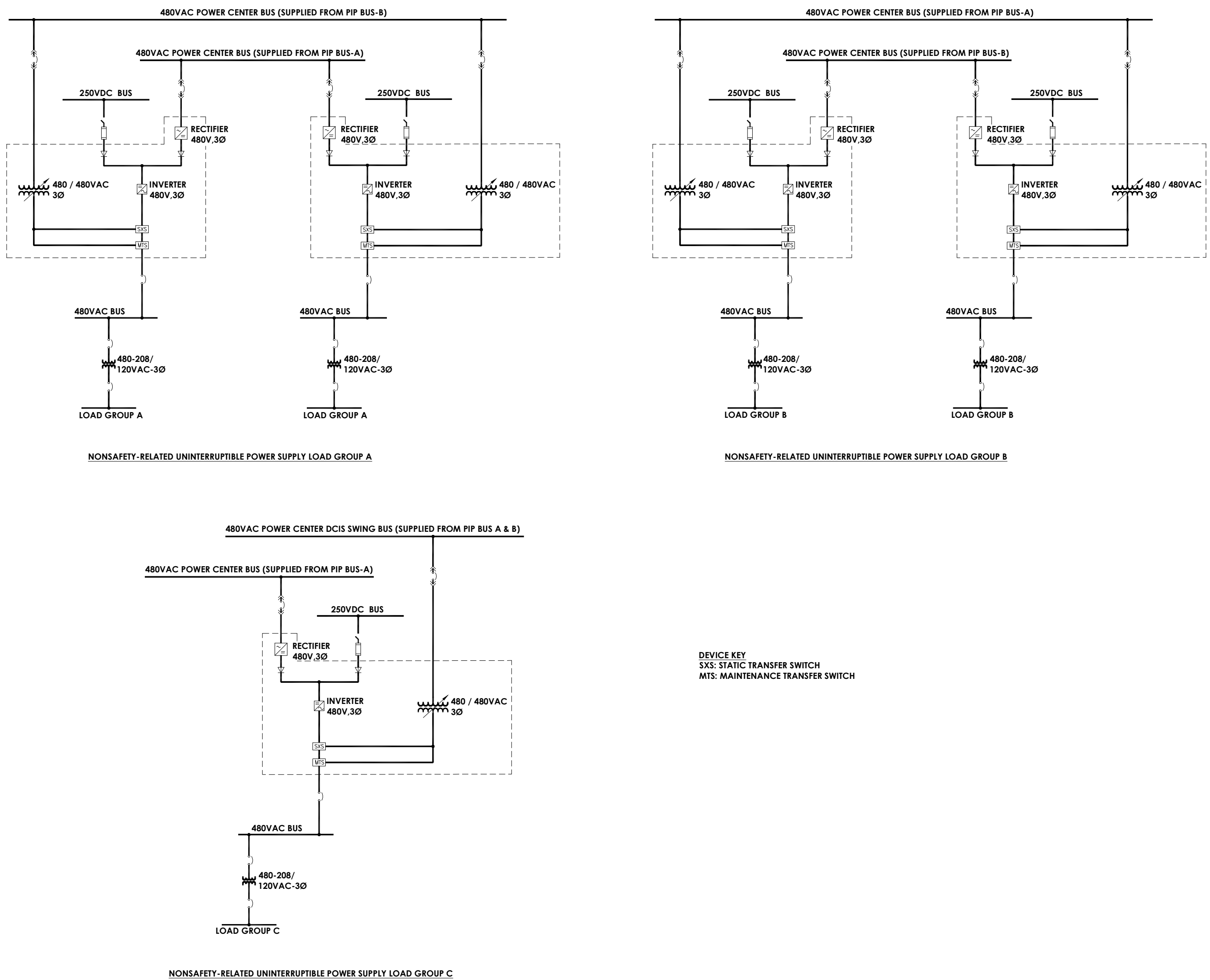


Figure 2.13.5-2. Sh 1
Nonsafety-Related UPS System

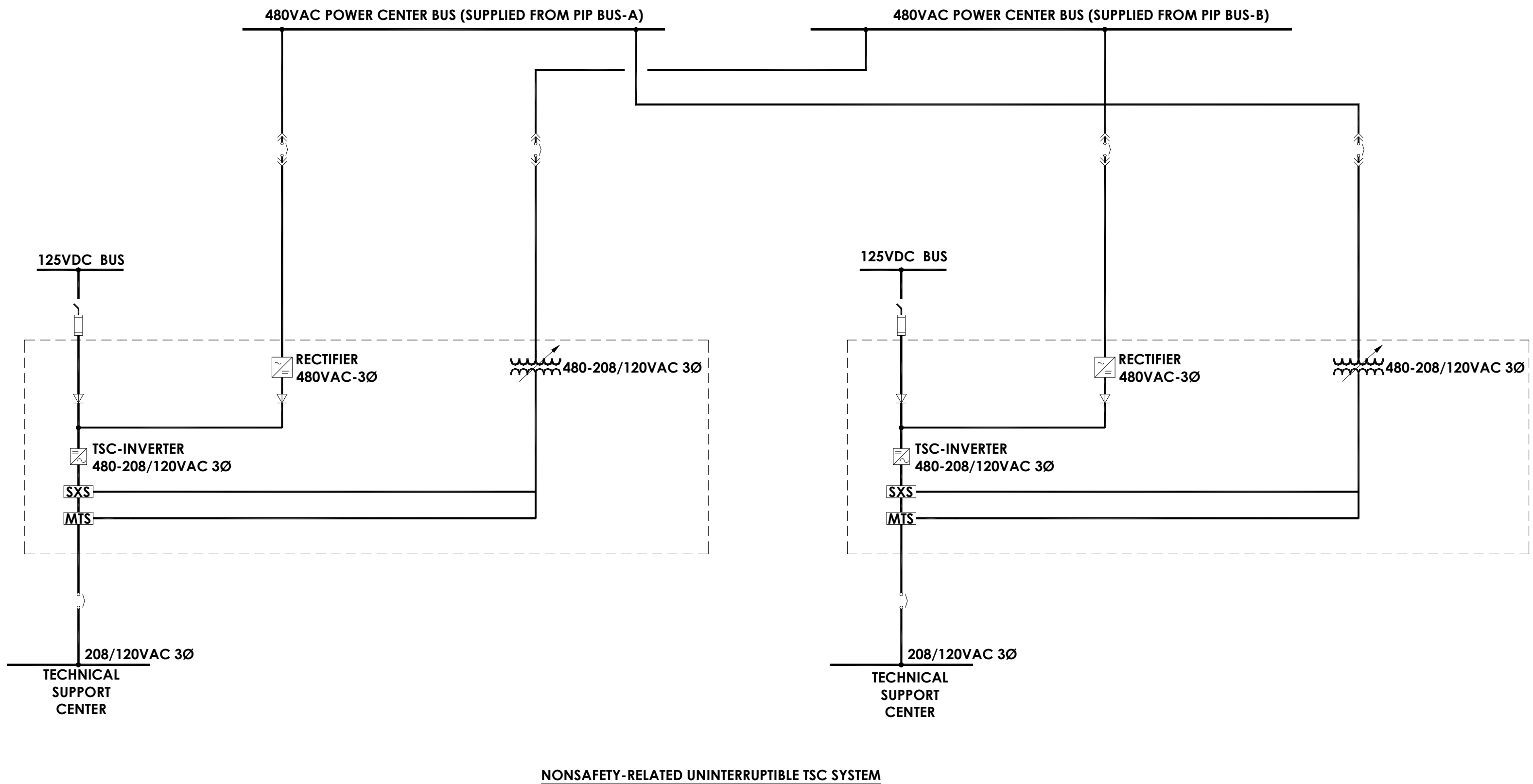


Figure 2.13.5-2 Sh 2
Nonsafety-Related UPS System

2.13.6 Instrument and Control Power Supply

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.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.13.7-1 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria for the Communication System.

Table 2.13.7-1

ITAAC For The Communication System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the Communication System is as described in Section 2.13.7.	Inspections of the as-built systems will be conducted.	Inspection report(s) demonstrate that the as-built communication system is as described in Section 2.13.7.

2.13.8 Lighting Power Supply

Design Description

The plant lighting systems furnish the illumination required for safe performance of plant operation, security, shutdown, and maintenance activities. The lighting systems include the Control Room Emergency Lighting System and normal, standby, and emergency lighting. The security lighting system is described in separate security documents.

(1) The functional arrangement of Control Room Emergency Lighting System is as described in this section 2.13.8.

(2) The Control Room Emergency Lighting System meets Seismic Category I requirements for mountings.

(3) The Control Room Emergency Lighting System is electrically independent and physically separated. Cables are routed in the respective divisional raceways.

(4) The Control Room Emergency Lighting system provides illumination levels in the main control room equal to or greater than those recommended by the Illuminating Engineering Society of North America (IESNA) for at least 72 hours following a design basis accident and a loss of all AC power sources.

(5) The DC Self-Contained Battery-Operated Lighting Units system provides illumination levels equal to or greater than those recommended by the IESNA in the remote shutdown rooms and in those areas of the plant required for power restoration and/or recovery from a fire, for at least 8 hours.

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 functional arrangement of Control Room Emergency Lighting System is as described in the Design Description of this Section 2.13.8.	Inspections of the as-built system will be conducted.	Inspection report(s) document that the as-built Control Room Emergency Lighting System conforms with the functional arrangement as described in the Design Description of this Section 2.13.8.
2. The Control Room Emergency Lighting System meets Seismic Category I requirements for mountings.	Analysis of the Control Room Emergency Lighting System mountings will be performed.	Analysis report(s) exist and document that the Control Room Emergency Lighting System mountings meet Seismic Category I requirements.
3. The Control Room Emergency Lighting System is electrically independent and physically separated. Cables are routed in the respective divisional raceways.	Inspection of the as-built Control Room Emergency Lighting System will be performed.	Inspection report(s) document that the as-built Control Room Emergency Lighting System equipment and cables are electrically independent and physically separated between safety divisions and between safety-related divisions and nonsafety-related equipment.
4. The Control Room Emergency Lighting system provides illumination levels in the main control room equal to or greater than those recommended by the IESNA for at least 72 hours following a design basis accident and a loss of all AC power sources.	Testing of the as-built Control Room Emergency Lighting System in the main control room will be performed.	Test report(s) demonstrate that the as-built Control Room Emergency Lighting System provides the illumination required by the IESNA at the main control room control stations for at least 72 hours.

Table 2.13.8-1

ITAAC For The Lighting Power Supply

Design Commitments	Inspections, Tests, Analyses	Acceptance Criteria
5. The DC Self-Contained Battery-Operated Lighting Units system provides illumination levels equal to or greater than those recommended by the IESNA in the remote shutdown rooms and in those areas of the plant required for power restoration and/or recovery from a fire, for at least 8 hours.	Testing of the as-built DC Self-Contained Battery-Operated Lighting Units System will be performed.	Test report(s) demonstrate that each division of the as-built DC Self-Contained Battery-Operated Lighting Units System provides the illumination required by the IESNA in areas required for power restoration / recovery to comply with the requirement of RG 1.189. Each unit will provide 8 hours of continuous illumination without battery recharge.

2.14 POWER TRANSMISSION

No entry for this system.

2.15 CONTAINMENT, COOLING AND ENVIRONMENTAL CONTROL SYSTEMS

2.15.1 Containment System

Design Description

The Containment System confines the potential release of radioactive material in the event of a design basis accident. The Containment System is comprised of a reinforced concrete containment vessel (RCCV), penetrations and drywell head.

The Containment System is as shown in Figure 2.15.1-1. The RCCV is located in the Reactor Building.

(1) The functional arrangement of the Containment System is described in the Design Description of this Section 2.15.1 and as shown in Figure 2.15.1-1.

(2) Components and piping identified in Table 2.15.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

  i. The RCCV and its liners are designed to meet the requirements in Article CC-3000 of ASME Code, Section III, Division 2.

  ii. The steel components of the RCCV are designed to meet the requirements in Article NE-3000 of ASME Code, Section III, Division 1.

(3) Pressure boundary welds in components and piping identified in Table 2.15.1-1 as ASME Code Section III meet ASME Code Section III requirements.

(4) The components and piping identified in Table 2.15.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

(5) The seismic Category I equipment identified in Table 2.15.1-1 can withstand seismic design basis load without loss of structural integrity and safety function.

(6) a. The equipment qualification of Containment Systems components is addressed in DCD Tier 1 Section 3.8.

  b. The safety-related components identified in Table 2.15.1-1 are powered from their respective safety-related division.

  c. Separate electrical penetrations are provided for circuits of each safety-related division and for nonsafety-related circuits.

  d. The circuits of each electrical penetration are of the same voltage class.

(7) The containment system provides a barrier against the release of fission products to the atmosphere.

(8) The containment system pressure boundary retains its integrity when subject to a design pressure of 310 kPa gauge (45 psig).

(9) The containment system provides the safety-related function of containment isolation for containment boundary integrity.

(10) Containment electrical penetration assemblies, whose maximum available fault current (including failure of upstream devices) is greater than the continuous rating of the penetration, are protected against currents that are greater than the continuous ratings.

(11) The minimum set of displays, alarms and controls, based on the emergency procedure guidelines and important operator actions, is available in the main control room

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.1-2 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria for the Containment System.

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
Penetration Identification: B21-MPEN-0001 (0002, 0003, 0004)								
Main Steam Line A (B, C, D) • F001A (B, C, D) Inboard • F002A (B, C, D) Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open	Closed	Closed
• F016A (B, C, D) Outboard	Yes	Yes	Yes	Yes	Yes	Open	Closed	As-is
Penetration Identification: B21-MPEN-0005								
Main Steam Line Drains • F010 Inboard • F011 Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open	Closed	Closed
Penetration Identification: B21-MPEN-0006 (0007)								
Feedwater Line A (B) • F102A (B) Inboard	Yes	Yes	N/A	Yes	Process Actuated	Open	N/A	N/A
• F101A (B) Outboard	Yes	Yes	Yes	Yes	Yes	Open	Closed	N/A
Penetration Identification: B32-MPEN-0001 (0002, 0003, 0004)								
IC Steam Supply • F001A (B, C, D) Inboard	Yes	Yes	Yes	Yes	Yes	Open	Open (Except on IC pipe or tube failure)	As-is

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
• F002A (B, C, D) Inboard	Yes	Yes	Yes	Yes	Yes	Open	Open (two in series valves)	As-is
Penetration Identification: B32-MPEN-0005 (0006, 0007, 0008)								
IC Condensate Return • F003A (B, C, D) Inboard • F004A (B, C, D) Inboard	Yes	Yes	Yes	Yes Yes	Yes	Open	Open (two in series valves)	As-is
Penetration Identification: B32-MPEN-0009 (0010, 0011, 0012)								
IC System Upper Header Vent • F007A (B, C, D) Inboard • F008A (B, C, D) Inboard	Yes	Yes	Yes	Yes Yes	No	Closed	Closed	Closed
Penetration Identification: B32-MPEN-0013 (0014, 0015, 0016)								
IC System Lower Header Vent • F009A (B, C, D) Inboard • F010A (B, C, D) Inboard IC System Lower Header Bypass Vent • F011A (B, C, D) Inboard • F012A (B, C, D) Inboard	Yes	Yes	Yes	Yes Yes Yes Yes	No	Closed	Closed	Closed
Penetration Identification: B32-MPEN-0017 (0018, 0019, 0020)								

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
IC System Purge Line • F013A (B, C, D) Inboard	Yes	Yes	Yes	Yes	Yes	Open	Open	Closed
IC System Excess Flow Purge • F014A (B, C, D) Inboard	Yes	Yes	N/A	Yes	Process Actuated	Open	Open	As-is
Penetration Identification: G31-MPEN-0001 (0002)								
RWCU/SDC system • F002A (B) Inboard • F003A (B) Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open/ Closed	Closed	Closed
Penetration Identification: G31-MPEN-0003 (0004)								
RWCU/SDC system • F007A (B) Inboard • F008A (B) Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open/ Closed	Closed	Closed
Penetration Identification: G31-MPEN-0005 (0006)								
RWCU/SDC system • F038A (B) Inboard • F039A (B) Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open/ Closed	Open/ Closed	Closed
Penetration Identification: C41-MPEN-0001 (0002)								

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
Standby Liquid Control • F005A (B) Inboard • F004A (B) Outboard	Yes	Yes	N/A	Yes Yes	Process Actuated	Closed		N/A
• F003A (B) Outboard • F003C (D) Outboard	Yes	Yes	N/A	Yes Yes	N/A	Closed	Open	As-is
Penetration Identification: G21-MPEN-0005								
Fuel and Auxiliary Pools Cooling System • F321A Outboard • F322A Outboard	Yes	Yes	Yes	Yes Yes	N/A	Closed	Closed	As-is
Penetration Identification: G21-MPEN-0002								
Fuel and Auxiliary Pools Cooling System • F306A Outboard	Yes	Yes	Yes	Yes	N/A	Closed	Closed	As-is
• F307A Inboard	Yes	Yes	N/A	Yes	Process Actuated	N/A	N/A	N/A
Penetration Identification: G21-MPEN-0007								

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
Fuel and Auxiliary Pools Cooling System • F321B Outboard • F322B Outboard	Yes	Yes	Yes	Yes Yes	N/A	Closed	Closed	As-is
Penetration Identification: G21-MPEN-0006								
Fuel and Auxiliary Pools Cooling System • F306B Outboard	Yes	Yes	Yes	Yes	N/A	Closed	Closed	As-is
• F307B Inboard	Yes	Yes	N/A	Yes	N/A	N/A	N/A	N/A
Penetration Identification: G21-MPEN-0004								
Fuel and Auxiliary Pools Cooling System • F323 Inboard	Yes	Yes	Yes	Yes	Yes	Closed	Closed	Closed
• F324 Outboard	Yes	Yes	Yes	Yes	Yes	Closed	Closed	Closed
Penetration Identification: G21-MPEN-0003								
Fuel and Auxiliary Pools Cooling System • F303 Outboard	Yes	Yes	Yes	Yes	N/A	Closed	Closed	Closed
• F304 Inboard	Yes	Yes	N/A	Yes	Process Actuated	N/A	N/A	N/A

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
Penetration Identification: G21-MPEN-0001								
Fuel and Auxiliary Pools Cooling System • F309 Outboard	Yes	Yes	Yes	Yes	N/A	Closed	N/A	Closed
• F310 Inboard	Yes	Yes	N/A	Yes	Process Actuated	Closed	N/A	As-is
Penetration Identification: T31-MPEN-0004								
Containment Inerting System • F012 Outboard • F011 Outboard	Yes	Yes	Yes	Yes Yes	Yes	Closed	Closed	Closed
Penetration Identification: T31-MPEN-0003								
Containment Inerting System • F010 Outboard • F011 Outboard • F014 Outboard • F015 Outboard	Yes	Yes	Yes	Yes Yes Yes Yes	Yes	Closed	Closed	Closed
Penetration Identification: T31-MPEN-0002								

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
Containment Inerting System • F008 Outboard • F007 Outboard	Yes	Yes	Yes	Yes Yes	Yes	Closed	Closed	Closed
• F024 Outboard • F023 Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open	Closed	Closed
Penetration Identification: T31-MPEN-0001								
Containment Inerting System • F008 Outboard • F009 Outboard	Yes	Yes	Yes	Yes Yes	Yes	Closed	Closed	Closed
• F025 Outboard • F023 Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open	Closed	Closed
Penetration Identification: T31-MPEN-0003 (0004)								
Containment Inerting System Main and Secondary Exhaust Line • {F010} • {F011} • {F012} • {F014} • {F015}	Yes	Yes	{Yes}	{Yes}	{Yes}	{Closed}	{Closed}	{Closed}

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
Penetration Identification: P25-MPEN-0001 (0003)								
Chilled Water System • F023A (B) Outboard • F024A (B) Inboard	Yes	Yes	Yes	Yes Yes	Yes	Open	Closed	Closed
Penetration Identification: P25-MPEN-0002 (0004)								
Chilled Water System • F025A (B) Inboard • F026A (B) Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open	Closed	Closed
Penetration Identification: P54-MPEN-0001								
High Pressure Nitrogen Gas Supply • F0026 Outboard • F009 Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open	Closed	Closed
• F027 Inboard • F010 Inboard	Yes	Yes	N/A	Yes Yes	Process Actuated	Open/ Closed	Closed	Closed
Penetration Identification: D11-MPEN-0001								
Process Radiation Monitoring System • F001 Outboard • F002 Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open	Closed	Closed
Penetration Identification: D11-MPEN-0002								

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
• F003 Outboard • F004 Outboard	Yes	Yes	Yes	Yes Yes	Yes	Open	Closed	Closed
Penetration Identification: T15-MPEN-0001 (0002, 0003, 0004, 0005, 0006 0007, 0008, 0009, 0010, 0011, 0012, 0013, 0014, 0015, 0016, 0017, 0018)								
Passive Containment Cooling System • Steam Inlet Line A (B, C, D, E, F) • Condenser Condensate + Vent Line A1, A2 (B1, B2, C1, C2, D1, D2, E1, E2, F1, F2)	Yes	Yes	-	-	-	-	-	-
Penetration Identification: T11-MPEN-TBD								
• Temporary Services During Outages and Spare Penetrations	Yes	Yes	-	-	-	-	-	-
Penetration Identification: B21-MPEN-TBD, B32-MPEN-TBD, E50-MPEN-TBD, T31-MPEN-TBD, T62-MPEN-TBD								
• Instrumentation and Monitoring	Yes	Yes	-	-	-	-	-	-
Penetration Identification: C21-MPEN-TBD								
• FMCRD Hydraulic Lines	Yes	Yes	-	-	-	-	-	-
Penetration Identification: G21-MPEN-TBD								
• Reactor Well Drain Line	Yes	Yes	-	-	-	-	-	-
Penetration Identification: P10-MPEN-0001								

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
• Demin Water Drywell Distribution	Yes	Yes	-	-	-	-	-	-
Penetration Identification: P52-MPEN-TBD								
• Service Air / Breathing Air Supply	Yes	Yes	-	-	-	-	-	-
Penetration Identification: U50-MPEN-TBD								
Equipment and Floor Drain System • Drywell LCW Sump Discharge Line {Inboard} • Drywell LCW Sump Discharge Line {Outboard} • Drywell HCW Sump Discharge Line {Inboard} • Drywell HCW Sump Discharge Line {Outboard}	Yes	Yes	{Yes}	Yes	{Yes}	{Closed}	{Closed}	{Closed}
Penetration Identification: R31-EPEN-TBD								
Electrical Penetrations	Yes	Yes	-	Yes	-	-	-	-
Penetration Identification: T11-SPEN-TBD								

Table 2.15.1-1

Containment System Penetrations and Equipment

Equipment Name	ASME Code Section III	Seismic Cat. 1	Remote Manual Operation	Safety-Related	Containment Isolation Signal	Normal Position	Post-Accident Position	Loss of Motive Power Position
• Lower Drywell Equipment Hatch • Lower Drywell Personnel Airlock • Wetwell Access Hatch • Upper Drywell Equipment Hatch • Upper Drywell Personnel Airlock	Yes	Yes	-	-	-	-	-	-

Table 2.15.1-2

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the Containment System (CS) is as described in Subsection 2.15.1 and as shown in Figure 2.15.1-1.	Inspections of the as-built system will be conducted.	Inspection report(s) document that the as-built Containment System conforms with the description in Subsection 2.15.1 and Figure 2.15.1-1.
2. The components and piping identified in Table 2.15.1-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements. The RCCV and its liners are designed to meet the requirements in Article CC-3000 of ASME Code, Section III, Division 2. The steel components of the RCCV are designed to meet the requirements in Article NE-3000 of ASME Code, Section III, Division 1.	Inspection(s) will be conducted of the as-built Containment System as documented in the Code Certified Stress Report.	Report(s) document that an ASME Code Section III stress report(s) exist for the as-built components identified in Table 2.1.51-1 as ASME Code Section III. 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-3350.
3. Pressure boundary welds in components and piping identified in Table 2.15.1-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection(s) of the as-built pressure boundary welds will be performed in accordance with ASME Code Section III.	A report exists and documents that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.

Table 2.15.1-2

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The components and piping identified in Table 2.15.1-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	i) A hydrostatic or pressure test will be performed on the components required by the ASME Code Section III to be tested.	i) A report exists and documents that the results of the pressure test of the components identified in Table 2.15.1-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.
	ii) Impact testing will be performed on the containment and pressure-retaining materials in accordance with the ASME Code Section III to confirm the fracture toughness of the materials.	ii) A report exists and documents that the containment and pressure-retaining penetration materials conform with fracture toughness requirements of the ASME Code section III.

Table 2.15.1-2

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The seismic Category I equipment identified in Table 2.15.1-1 can withstand seismic design basis loads without loss of structural integrity and safety function.	i) Type tests and/or analyses of seismic Category I equipment will be performed.	i) A report exists and documents that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of structural integrity and safety function.
	ii) Inspections will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	ii) The as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions.
	iii) Inspections of the equipment identified in Table 2.15.1-1 will be performed to verify that the equipment is housed in seismic Category I structures.	iii) The seismic category I equipment identified in Table 2.15.1-1 is housed in a seismic Category I structure.

Table 2.15.1-2

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6a. The equipment qualification of Containment Systems components is addressed in DCD Tier 1 Section 3.8.	See Tier 1 Section 3.8.	See Tier 1 Section 3.8.
b. The safety-related components identified in Table 2.15.1-1 are powered from their respective safety-related division.	See Tier 1 Section 2.13.	See Tier 1 Section 2.13.
c. Separate electrical penetrations are provided for circuits of each safety-related division and for nonsafety-related circuits.	Inspection of the as-built electrical containment penetrations will be performed.	Inspection report(s) document that each as-built electrical penetration contains cables of only one division or non-division.
d. The circuits of each electrical penetration are of the same voltage class.	Inspections of the as-built containment electrical pentrations will be performed.	Inspection report(s) document that each as-built circuit of each electrical penetration is of the same voltage class.
7. The containment system provides a barrier against the release of fission products to the atmosphere.	Perform Type A, B and C leakrate tests in accordance with 10 CFR 50 Appendix J.	Test report(s) conclude that leak rates are less than the acceptance criterion established per 10CFR 50 Appendix J.

Table 2.15.1-2

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. The containment system pressure boundary retains its integrity when subject to a design pressure of 310 kPa guage (45 psig).	A Structural Integrity Test (SIT) of the containment structure is performed in accordance with Article CC-6000 of ASME Code Section III, Division 2 and Regulatory Guide 1.136, after completion of the containment construction. The first prototype containment structure will be instrumented to measure strains per ASME Code Section III, Division 2, CC-6370.	Test report documents compliance with ASME Code Section III, Div. 2, CC-6000, Structural Integrity Test of Concrete Containments.
9. The containment system provides the safety-related function of containment isolation for containment boundary integrity.	i) Tests will be performed to demonstrate that remote manual operated containment isolation valves reposition to the required post-accident position within the required response times. ii) Tests will be performed to demonstrate that remote manual operated containment isolation valves reposition to the required post-accident position using real or simulated containment isolation signals.	i) Report(s) document that the remotely operated containment isolation valves identified in Table 2.1.51-1 reposition to the required post-accident state within the required response times. ii) Report(s) document that the remote manual operated valves identified in Table 2.15.1-1 as having a containment isolation signal reposition to the required post-accident state after receiving a containment isolation signal.

Table 2.15.1-2

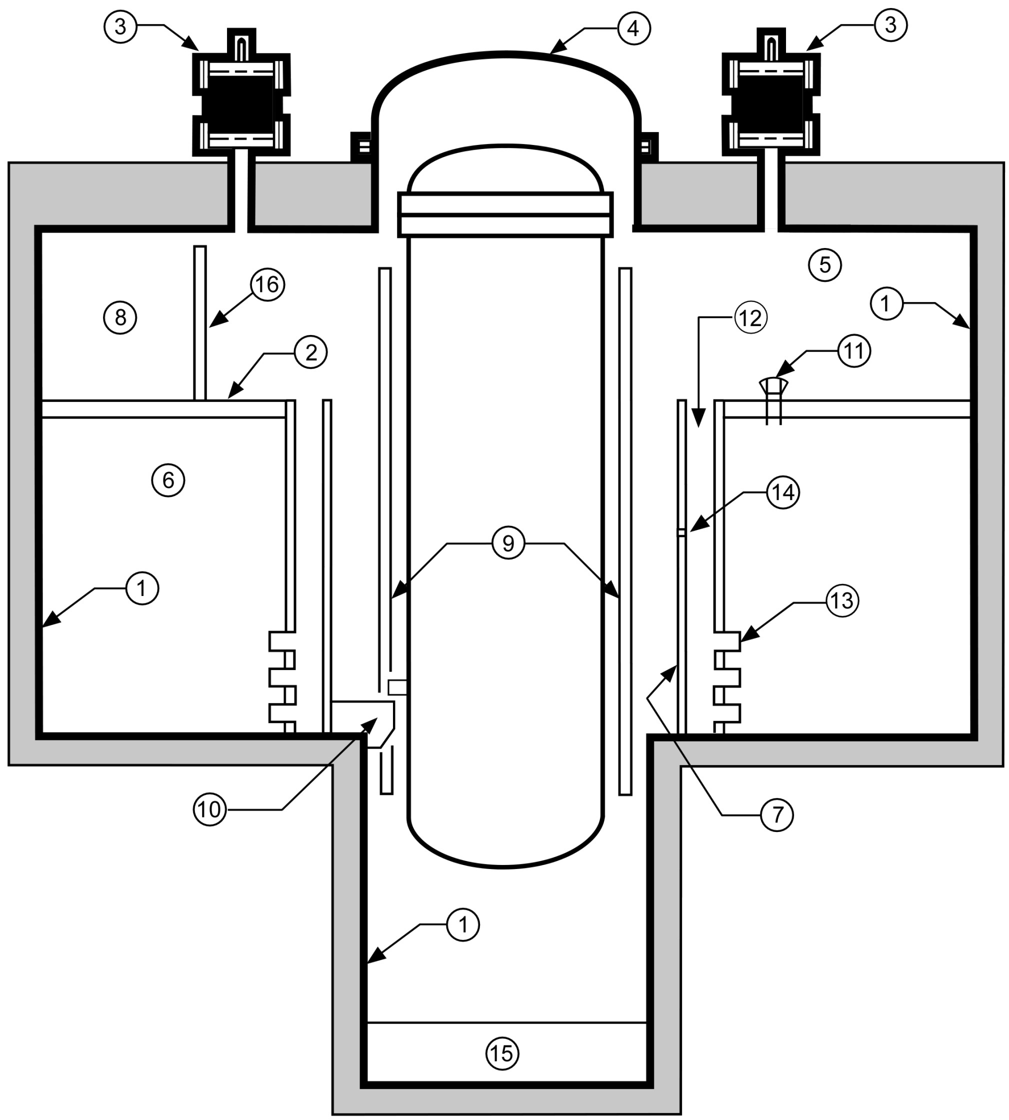
ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	iii) Exercise testing of the process actuated check valves identified in Table 2.15.1-1 will be performed under preoperational test pressure, temperature and fluid flow conditions. iv) Tests or type tests of motor-operated valves will be performed to demonstrate the capability of each valve to operate under design conditions.	iii) Test report(s) document that each as-built process actuated check valve changes position as indicated in Table 2.15.1-1. iv) Test report(s) exist and documents that each as-built motor-operated valve changes position as indicated in Table 2.15.1-1 under design conditions.
	v) Inspection will be performed for the existence of a report verifying that the as-built motor-operated valves are bounded by the tests or type tests. vi) Tests of the as-built motor-operated valves will be performed under preoperational flow, differential pressure, and temperature conditions.	v) Report(s) exist and documents that the as-built motor-operated valves are bounded by the tests or type tests. vi) Test report(s) document that each motor-operated valve changes position as indicated in Table 2.15.1-1 under pre-operational test conditions.

Table 2.15.1-2

ITAAC For The Containment System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	vii) Testing of the as-built valves will be performed under the conditions of loss of motive power.	vii) Test report(s) document that after a loss of motive power, each remote manual valve identified in Table 2.15.1-1 assumes the indicated loss of motive power position.
10. Containment electrical penetration assemblies, whose maximum available fault current (including failure of upstream devices) is greater than the continuous rating of the penetration, are protected against currents that are greater than the continuous ratings.	An analysis of the as-built containment electrical penetration assemblies will be performed to demonstrate either (1) the maximum overucrrent of the circuits does not exceed the continuous current rating of the penetration, or (2) circuits whose maximum available fault currents are greater than the continuous current rating of the penetration are provided with redundant overcurrent interrupting devices.	Report(s) document that analysis exists for the as-built containment electrical penetration assemblies and concludes that the penetrations, whose maximum available fault current (including failure of upstream devices) is greater than the continuous rating of the penetration, are protected against currents that are greater than their continuous ratings.
11. The minimum set of displays, alarms and controls, based on the emergency procedure guidelines and important operation actions, is available in the main control room.	Inspection of the as-built main control room will verify that the minimum set of displays, alarms and controls for the Containment System is available.	Inspection report(s) document that the minimum set of displays, alarms and controls for the Containment System, as defined by the emergency operating procedures and important operator actions exist in the as-built main control room.



LEGEND
1. Containment Vessel (RCCV)
2. Diaphram Floor Slab
3. Passive Containment Cooling System (PCCS), Total 6
4. Drywell Head
5. Drywell
6. Wetwell
7. Vent Wall
8. GDCS Pools, Total 3
9. Reactor Shield Wall
10. RPV Support Bracket (Typical 8)
11. Vacuum Breaker, Total 3
12. Vertical Vents, ≥ 13.6m^2 (146 ft^2) Total
13. Horizontal Vent, ≥ 0.7m (2.30 ft) I.D., 36 Total
14. Spillover Hole, 200mm (8 inch) Nominal Diameter, 12 Total
15. BiMAC
16. GDCS Pool Wall (Typical)

Figure 2.15.1-1. Containment System

2.15.2 Containment Vessel

Design Description and ITAAC are addressed in Subsection 2.15.1.

2.15.3 Containment Internal Structures

Design Description

The functions of the containment internal structures include (1) support of the reactor vessel radiation shielding, (2) support of piping and equipment, and (3) formation of the pressure suppression boundary. The containment internal structures consist of the diaphragm floor slab (DF) that separates the Drywell (DW) and the Wetwell (WW), vent wall, Gravity-Driven Cooling System (GDCS) pool walls, reactor shield wall, and the Reactor Pressure Vessel (RPV) support bracket.

The Containment Internal Structures are as shown in Figure 2.15.3-1 and the component locations of the Containment System are as shown in Table 2.15.3-1.

(1) The functional arrangement of the Containment Internal Structures is described in the Design Description of this Section 2.15.3.

(2) Containment Internal Structures are designed and constructed in accordance with ANSI/AISC N690 requirements.

(3) The Containment Internal Structures identified in Table 2.15.3-1 conform to Seismic Category I requirements and and can withstand seismic design basis loads and suppression pool hydrodynamic loads without loss of structural integrity and safety function.

(4) The DW and WW volumes are adequately sized to accommodate the calculated maximum DW temperature and absolute pressure that are postulated to occur as a result of a design basis accident.

(5) The diaphragm floor and vent wall structures that separate the DW and WW retain their integrity when subject to pressure at or above design pressure.

(6) The water volume of the WW suppression pool is adequately sized to condense the steam that is forced into the WW from the DW due to a postulated pipe break.

(7) Each vacuum breaker isolation valve automatically closes if the vacuum breaker does not fully close when required.

(8) Each vacuum breaker has proximity sensors to detect open/close position. This indication is available in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.3-2 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria for the Containment Internal Structures.

Table 2.15.3-1

Containment Internal Structures Locations

Component	Component Location
Diaphragm Floor Slab (DF)	Inside of Containment Boundary
Vent Wall	Inside of Containment Boundary
Gravity-Driven Cooling System (GDCS) Pool Walls	Inside of Containment Boundary
Reactor Shield Wall	Inside of Containment Boundary
RPV Support Bracket	Inside of Containment Boundary

Table 2.15.3-2

ITAAC For The Containment Internal Structure

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Containment Internal Structures is as described in Subsection 2.15.3.	Inspections of the as-built system will be conducted.	Inspection reports document that the as-built Containment Internal Structures conforms with the description in Subsection 2.15.3.
2. The Containment Internal Structures are designed and constructed in accordance with ANSI/AISC N690 requirements.	Inspections will be conducted of the as-built components of the Containment Internal Structures as documented in the design reports.	Inspection report(s) document that the design reports exist, and the as-built components of the Containment Internal Structures are constructed in accordance with ANSI/AISC N690 requirements.
3. The Containment Internal Structures identified in Table 2.15.3-1 conform to Seismic Category I requirements and can withstand seismic design basis loads and suppression pool hydrodynamic loads without loss of structural integrity and safety function.	i) Analyses of Seismic Category I requirements will be performed.	i) Analysis report(s) exists and demonstrates that the as-built Containment Internal Structures can withstand seismic design basis dynamic loads and suppression pool hydrodynamic loads without loss of structural integrity and safety function.
	ii) Inspections of the as-built Containment Internal Structures will be performed to verify that they are housed in a seismic Category I structure.	ii) A report documents that the as-built Containment Internal Structures are housed in a seismic Category I structure.

Table 2.15.3-2

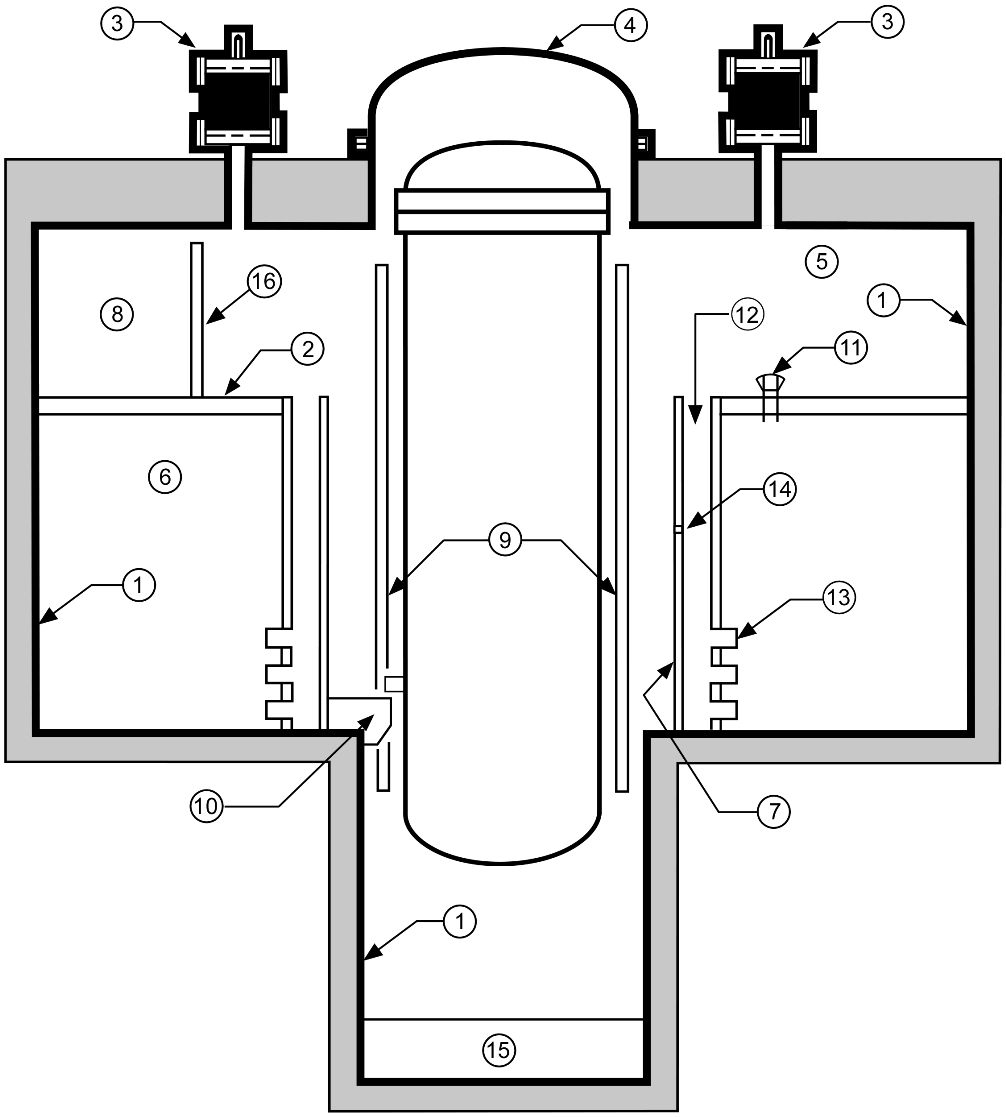
ITAAC For The Containment Internal Structure

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The DW and WW volumes are adequately sized to accommodate the calculated maximum DW temperature and absolute pressure that are postulated to occur as a result of a design basis accident.	Using as-built dimensions, the DW and WW volumes will be calculated.	A report documents that the calculated as-built DW and WW volumes are greater than or equal to the design basis values.
5. The diaphragm floor and vent wall structures that separate the DW and WW retain their integrity when subject to pressure at or above design pressure.	Part of the containment Structural Integrity Test specified in Tier 1 Table 2.15.1-2 ITAAC # 8 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 DW pressure greater than WW pressure.	A report of the SIT results demonstrates compliance with ASME Code requirements for the applied test pressure for the containment structures.
6. The water volume of the WW suppression pool is adequately sized to condense the steam that is forced into the WW from the DW due to a postulated design basis event.	Using as-built dimensions of the WW and a minimum measured suppression pool depth of 5.4 meters (213 inches), the volume of the suppression pool will be calculated.	A report demonstrates that the calculated suppression pool water volume is equal to or greater than the water volume assumed in the containment performance safety analysis.
7. Each vacuum breaker isolation valve automatically closes if the vacuum breaker does not fully close when required.	A test will be performed by providing a simulated or real not-fully closed vacuum breaker signal originating from the closed position proximity sensor to close the associated vacuum breaker isolation valve.	A report demonstrates that each as-built vacuum breaker isolation valve automatically closes when a simulated or real not-fully closed signal is provided from the closed position proximity sensor of its associated vacuum breaker.

Table 2.15.3-2

ITAAC For The Containment Internal Structure

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Each vacuum breaker has proximity sensors to detect open/close position. This indication is available in the main control room.	Testing will be performed with each as-built vacuum breaker to demonstrate that the proximity sensors indicate open and closed position.	Test report(s) demonstrate that each as-built vacuum breaker proximity sensor indicates an open position with the vacuum breaker fully open and indicates a closed position when the vacuum breaker is in the fully closed position.



LEGEND
1. Containment Vessel (RCCV)
2. Diaphram Floor Slab
3. Passive Containment Cooling System (PCCS), Total 6
4. Drywell Head
5. Drywell
6. Wetwell
7. Vent Wall
8. GDCS Pools, Total 3
9. Reactor Shield Wall
10. RPV Support Bracket (Typical 8)
11. Vacuum Breaker, Total 3
12. Vertical Vents, ≥ 13.6m^2 (146 ft^2) Total
13. Horizontal Vent, ≥ 0.7m (2.30 ft) I.D., 36 Total
14. Spillover Hole, 200mm (8 inch) Nominal Diameter, 12 Total
15. BiMAC
16. GDCS Pool Wall (Typical)

Figure 2.15.3-1. Containment Internal Structures

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.

The PCCS consists of six low pressure, independent sets of two steam condenser modules (passive containment cooling condensers) that condense steam on the tube side and transfer heat from the drywell 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 condenser.

Each condenser, which is an integral part of the containment, contains a drain line to one of the three GDCS pools, and a vent discharge line the end of which is submerged in the pressure suppression pool.

The PCCS condensers 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.

(1) The functional arrangement for the PCCS is as described in the Design Description in this Section 2.15.4, Table 2.15.4-1 and Figure 2.15.4-1.

(2) a. The components identified in Table 2.15.4-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

b. The piping identified in Table 2.15.4-1 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

(3) a. Pressure boundary welds in components identified in Table 2.15.4-1 as ASME Code Section III meet ASME Code Section III requirements.

b. Pressure boundary welds in piping identified in Table 2.15.4-1 as ASME Code Section III meet ASME Code Section III requirements.

(4) The pressure boundary of the PCCS retains its integrity under the design pressure of 310 kPa gauge (45 psig).

(5) a. The seismic Category I equipment identified in Table 2.15.4-1 can withstand seismic design basis loads without loss of safety function.

b. Each of the lines identified in Table 2.15.4-1 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.

(6) Each mechanical train of the PCCS (A, B, C, D, E & F)* is physically separated from the other trains. *As indicated on Figure 2.15.4-1. Physical separation is not required in the Primary Containment.

(7) 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.

(8) The equipment qualification of PCCS components is addressed in Tier 1 Section 3.8.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.4-2 provides a definition of the inspections, tests and/or analyses, together with associated acceptance criteria for the Passive Containment Cooling System.

Table 2.15.4-1

Passive Containment Cooling System Mechanical Equipment

Equipment Name (Description)	**Equipment Identifier see Figure 2.15.4-1**	**ASME Code Section III**	**Seismic Cat. I**	**RCPB Component**	**Containment Isolation Valve**	**Remotely Operated Valve**	**Loss of Motive Power Position**	**Functional Capability Required**
PCC Heat Condenser	PCC Condenser	Yes	Yes	No	-	-	-	Yes
PCC Inlet Line	P-1(A[1])	Yes	Yes	No	-	-	-	Yes
Condensate Drain Line	P-2(A[1])	Yes	Yes	No	-	-	-	Yes
Non-Condensables Vent Line	P-3(A[1])	Yes	Yes	No	-	-	-	Yes

Note: A dash means not applicable.
[1] Train A; Typical for Trains B, C, D, E & F.

Table 2.15.4-2

ITAAC For The Passive Containment Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement for the PCCS is as described in the Design Description in this Section 2.15.4, Table 2.15.4-1 and Figure 2.15.4-1.	Inspections of the as-built system will be conducted.	Report(s) document that the as-built PCCS conforms to the functional arrangement for the PCCS is as described in the Design Description in this Section 2.15.4, Table 2.15.4-1 and Figure 2.15.4 1.
2a. The components identified in Table 2.15.4-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	Report(s) document that the ASME Code Section III design reports exist for the as-built components identified in Table 2.15.4-1 as ASME Code Section III.
b. The piping identified in Table 2.15.4-1 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	Report(s) document that the ASME code Section III design reports exist for the as-built piping identified in Table 2.15.4-1 as ASME Code Section III.

Table 2.15.4-2

ITAAC For The Passive Containment Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3a. Pressure boundary welds in components identified in Table 2.15.4-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	Report(s) document that a report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
b. Pressure boundary welds in piping identified in Table 2.15.4-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	Report(s) document that a report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4. The pressure boundary of the PCCS retains its integrity under the design pressure of 310 kPa gauge (45 psig)	A containment Structural Integrity Test (SIT) will be conducted per ASME requirement 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 1, NE-6320.	Test results demonstrate compliance to ASME Code Section III, Div 1, NE-3226.

Table 2.15.4-2

ITAAC For The Passive Containment Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5a. The seismic Category I equipment identified in Table 2.15.4-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.15.4-1 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	Report(s) document that: i) The seismic Category I equipment identified in Table 2.15.4-1 is located on a seismic structure. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
b. Each of the lines identified in Table 2.15.4-1 for which functional capability is required is designed to withstand combined normal and seismic design basis loads without a loss of its functional capability.	Inspection will be performed for the existence of a report verifying that the as-built piping meets the requirements for functional capability.	Report(s) document that a report exists and concludes that each of the as-built lines identified in Table 2.15.4-1 for which functional capability is required meets the requirements for functional capability.

Table 2.15.4-2

ITAAC For The Passive Containment Cooling System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Each mechanical train of the PCCS (A, B, C, D, E & F)* is physically separated from the other trains. *As indicated on Figure 2.15.4-1. Physical separation is not required in the Primary Containment.	Inspections of the as-built PCCS will be performed.	Report(s) document that the each mechanical train of the PCCS is physically separated from other mechanical trains of the system by structural and/or fire barriers (with the exception of portions in Primary Containment).
7. 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.	An analysis will be performed using similar or more conservative performance characteristics than those of a test unit of established performance capability.	Analyzed containment pressure for 72 hours after a LOCA is less than containment design pressure.
8. The equipment qualification of PCCS components is addressed in Tier 1 Section 3.8.	See Tier 1 Section 3.8.	See Tier 1 Section 3.8.

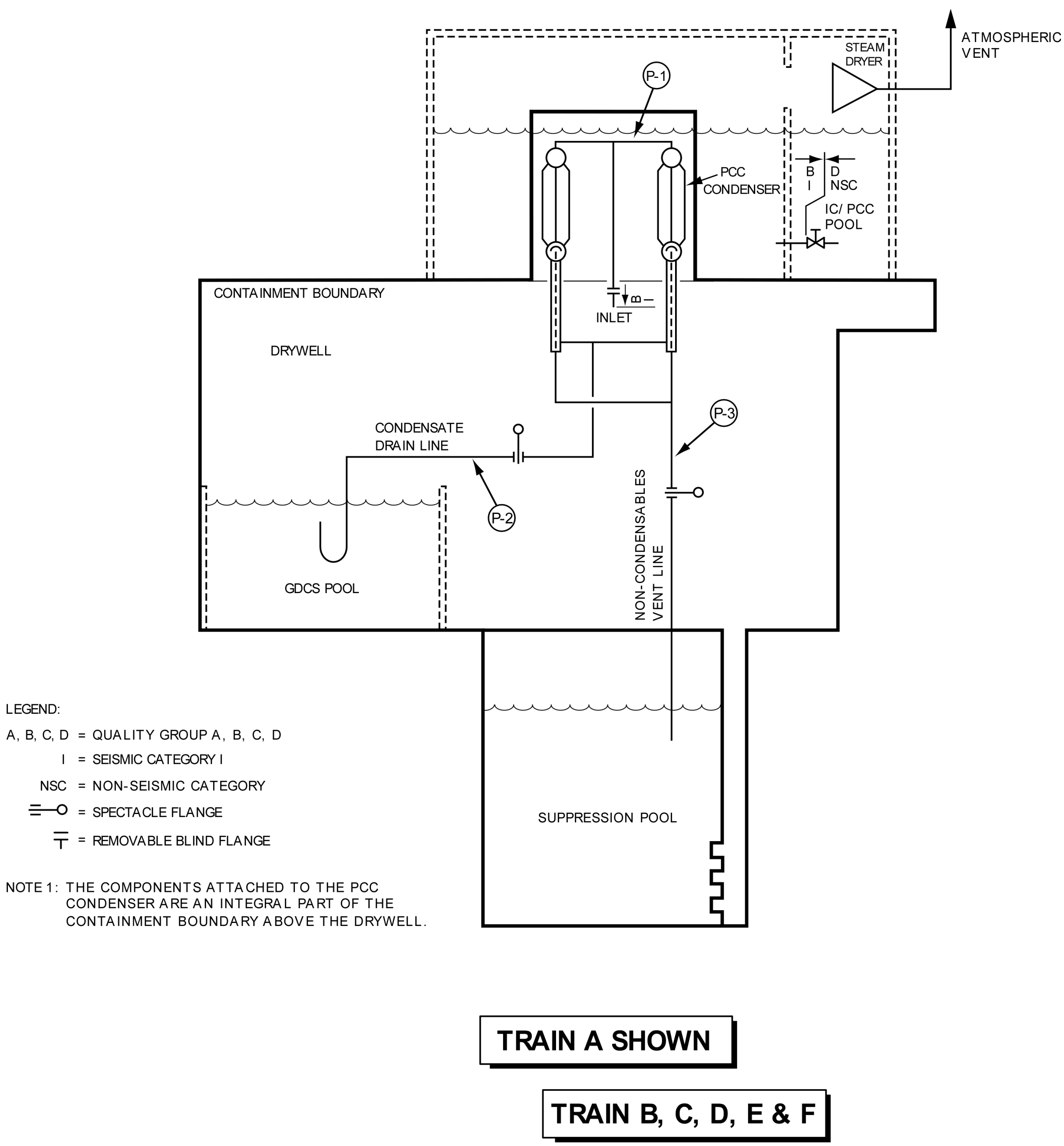


Figure 2.15.4-1. Passive Containment Cooling System Schematic

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 reduce oxygen concentration to levels that do not support post-accident hydrogen combustion.

The CIS does not perform any safety-related function except for its containment isolation function. Containment Isolation Valves and Penetrations are addressed in Tier 1 Subsection 2.15.1 for the Containment System.

(1) The containment can be inerted to less than or equal to 4% oxygen by volume.

(2) The containment isolation portions of the CIS are addressed in Tier 1, Subsection 2.15.1.

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 can be inerted to less than or equal to 4% oxygen by volume.	Test of the containment in an inerted state will be conducted to determine oxygen concentration by volume.	Test report concludes that the containment can be inerted to less than or equal to 4% oxygen by volume.
2. The containment isolation portions of the CIS are addressed in Tier 1, Subsection 2.15.1.	See Tier 1 Subsection 2.15.1.	See Tier 1 Subsection 2.15.1.

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. The DCS provides a drywell temperature measurement, described in Table 2.15.6-1, associated with the ITAAC below.

Instrumentation and Control

(1) The drywell temperature indications are retrievable in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.6-2 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

Drywell Cooling System Electrical Equipment

Equipment Name	Equipment ID	Control Q-DCIS/ DPS	Safety-Related Electrical Equipment	Safety-Related Display	Active Function	Loss of Motive Power Position	Remotely Operated Valve	Containment Isolation Valve
Drywell Temperature Transmitter(s)		No	No	No	No	-	-	-

Note: A dash means not applicable.

Table 2.15.6-2

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.	Inspections of main control room indications will be conducted and verified for retrievability of drywell temperature indications.	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) provides instrumentation listed in Table 2.15.7-1 to monitor the following parameters:

- Drywell and Wetwell Hydrogen and Oxygen concentrations
- Drywell and Wetwell Gross Gamma Radiation levels
- Drywell and Wetwell Pressures
- Drywell/Wetwell Differential Pressure
- Upper Drywell Level
- Lower Drywell Level
- Suppression Pool Water Level
- Suppression Pool Temperature

(1) The functional arrangement for the CMS is as described in the Design Description in this Section 2.15.7, Table 2.15.7-1 and Figure 2.15.7-1.

(2) Each of the safety-related components identified in Table 2.15.7-1 is powered from its respective safety-related division.

(3) Each CMS measured parameter in Table 2.15.7-1 will indicate the measured parameter and initiate separate alarms in the control room when levels exceed applicable setpoints.

(4) The Hydrogen/Oxygen (H_2/O_2) monitoring subsystem of CMS is active during normal operation and additional sampling capacity is automatically initiated by an ECCS initiation signal for post-accident monitoring of oxygen and hydrogen content in the containment.

(5) In each CMS Suppression Pool Temperature Monitoring (STPM) division, signals from the CMS STPM temperature and the CMS suppression pool water narrow range transmitters are provided for the divisional Safety System Logic and Control (SSLC)/Reactor Protection System (RPS) logic processors to calculate the suppression pool average temperature.

(6) The seismic Category I equipment identified in Table 2.15.7-1 can withstand seismic design basis loads without loss of safety function.

(7) The equipment qualification of CMS components is addressed in Tier 1 Section 3.8.

(8) The containment isolation portions of the CMS system are addressed in Tier 1, Subsection 2.15.1.

Refer to Subsection 2.2.15 for "Instrumentation & Controls Compliance With IEEE Std. 603."

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.7-2 provides 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

Containment Monitoring System Electrical Equipment

Equipment Name	Equipment Identifier See Figure 2.15.7-1	Control Q-DCIS/ DPS[1]	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated	Containment Isolation Valve Actuator
Drywell Hydrogen Sampling Transmitter (2 divisional channels)	(H_2 Skid)	Yes	Yes	Yes	Yes	No	-	-
Drywell Oxygen Sampling Transmitter (2 divisional channels)	(O_2 Skid)	Yes	Yes	Yes	Yes	No	-	-
Wetwell Hydrogen Sampling Transmitter (2 divisional channels)	(H_2 Skid)	Yes	Yes	Yes	Yes	No	-	-
Wetwell Oxygen Sampling Transmitter (2 divisional channels)	(O_2 Skid)	Yes	Yes	Yes	Yes	No	-	-
Upper Drywell Gamma Radiation Transmitter (2 divisional channels)	RDT[2] (Upper Drywell)	No	No	No	No	No	-	-
Lower Drywell Gamma Radiation Transmitter (2 divisional channels)	RDT[2] (Lower Drywell)	No	No	No	No	No	-	-

Table 2.15.7-1

Containment Monitoring System Electrical Equipment

Equipment Name	Equipment Identifier See Figure 2.15.7-1	Control Q-DCIS/ DPS[1]	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated	Containment Isolation Valve Actuator
Wetwell Gamma Radiation Transmitter (2 divisional channels)	RDT[2] (Wetwell)	No	No	No	No	No	-	-
Drywell Pressure Transmitter (Safety-related) (4 divisional channels)	PT[2] (Drywell)	Yes	Yes	Yes	Yes	No	-	-
Wetwell Pressure Transmitter (Safety-related) (4 divisional channels)	PT[2] (Wetwell)	Yes	Yes	Yes	Yes	No	-	-
Drywell Pressure Wide Range Transmitter (Safety-related) (2 divisional channels)	PT[2] (Drywell)	Yes	Yes	Yes	Yes	No	-	-
Wetwell Wide Range Pressure Transmitter (Safety-related)) (2 divisional channels)	PT[2] (Wetwell)	Yes	Yes	Yes	Yes	No	-	-
Drywell Pressure Transmitter (Nonsafety-related) (4)	PT[2] (Drywell)	Yes[1]	No	No	No	No	-	-
Wetwell Pressure Transmitter (Nonsafety-related)	PT[2] (Wetwell)	No	No	No	No	No	-	-

Table 2.15.7-1

Containment Monitoring System Electrical Equipment

Equipment Name	Equipment Identifier See Figure 2.15.7-1	Control Q-DCIS/ DPS[1]	Seismic Category I	Safety-Related	Safety-Related Display	Active Function	Remotely Operated	Containment Isolation Valve Actuator
Drywell/Wetwell Differential Pressure Transmitter (2 divisional channels)	PDT[2]	Yes	Yes	Yes	Yes	No	-	-
Upper Drywell Level Transmitter	LT[2] (Upper Drywell)	No	No	No	No	No	-	-
Lower Drywell Level Transmitter	LT[2] (Lower Drywell)	No	No	No	No	No	-	-
Suppression Pool Water Level Narrow Range Transmitter) (4 divisional channels)	LT[2] (Suppression Pool)	Yes	Yes	Yes	Yes	No	-	-
Suppression Pool Water Level Wide Range Transmitter (4 channels)	LT[2] (Suppression Pool)	No	No	No	No	No	-	-
Suppression Pool Temperature Transmitter (4 divisional channels; multiple sensors)	TE[2] (Suppression Pool)	Yes	Yes	Yes	Yes	No	-	-

Note: A dash means not applicable.

[1] DPS input; See Section 2.2.14.

[2] Shown as representative in Figure 2.15.7-1.

Table 2.15.7-2

ITAAC For The Containment Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement for the CMS is as described in the Design Description in this Section 2.15.7, Table 2.15.7-1 and Figure 2.15.7-1.	Inspections of the as-built system will be performed.	Report(s) document that the as-built CMS conforms with the functional arrangement described in the Design Description of this Section 2.15.7, Table 2.15.7-1 and Figure 2.15.7-1.
2. Each of the safety-related components identified in Table 2.15.7-1 is powered from its respective safety-related division..	Testing will be performed on the CMS by providing a test signal in only one safety-related division at a time.	Report(s) document that a test signal exists in the safety-related division (or at the equipment identified in Table 2.15.7-1 powered from the safety-related division) under test in the CMS.
3. Each CMS measured parameter in Table 2.15.7-1 will indicate the measured parameter and initiate separate alarms in the control room when levels exceed applicable setpoints.	Using simulated signal inputs, CMS testing will be performed.	Report(s) document that each simulated signal representing a measured parameter in Table 2.15.7-1 indicates the measured parameter and initiates separate alarms in the control room when levels exceed applicable setpoints.

Table 2.15.7-2

ITAAC For The Containment Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The Hydrogen/Oxygen (H2/O2) monitoring subsystem of CMS is active during normal operation and additional sampling capacity is automatically initiated by an ECCS initiation signal for post-accident monitoring of oxygen and hydrogen content in the containment.	Using simulated electrical signals, CMS testing will be performed.	Report(s) document that H2/O2 monitor to be in operation within [90] minutes, including warm-up time, after occurrence of an ECCS initiation signal, which requires the monitor to be functional.
5. In each CMS Suppression Pool Temperature Monitoring (STPM) division, signals from the CMS STPM temperature and the CMS suppression pool water narrow range transmitters are provided for the divisional Safety System Logic and Control (SSLC)/Reactor Protection System (RPS) logic processors to calculate the suppression pool average temperature	Tests will be conducted in each division of the SPTM using simulated temperature sensor signals.	Report(s) document that for each SPTM division, output signals from the CMS SPTM subsystem are received to generate a suppression pool average temperature by the SSLC/ RPS logic processors.

Table 2.15.7-2

ITAAC For The Containment Monitoring System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The seismic Category I equipment identified in Table 2.15.7-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment and valves identified in Table 2.15.7-1 are located on the Nuclear Island. ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	Report(s) document that: i) The seismic Category I equipment identified in Table 2.15.7-1 is located on a seismic structure. ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions. .
7. The equipment qualification of CMS components is addressed in Tier 1 Section 3.8.	See Tier 1 Section 3.8.	See Tier 1 Section 3.8.
8. The containment isolation portions of the CMS system are addressed in Tier 1, Subsection 2.15.1.	See Tier 1 Subsection 2.15.1.	See Tier 1 Subsection 2.15.1.

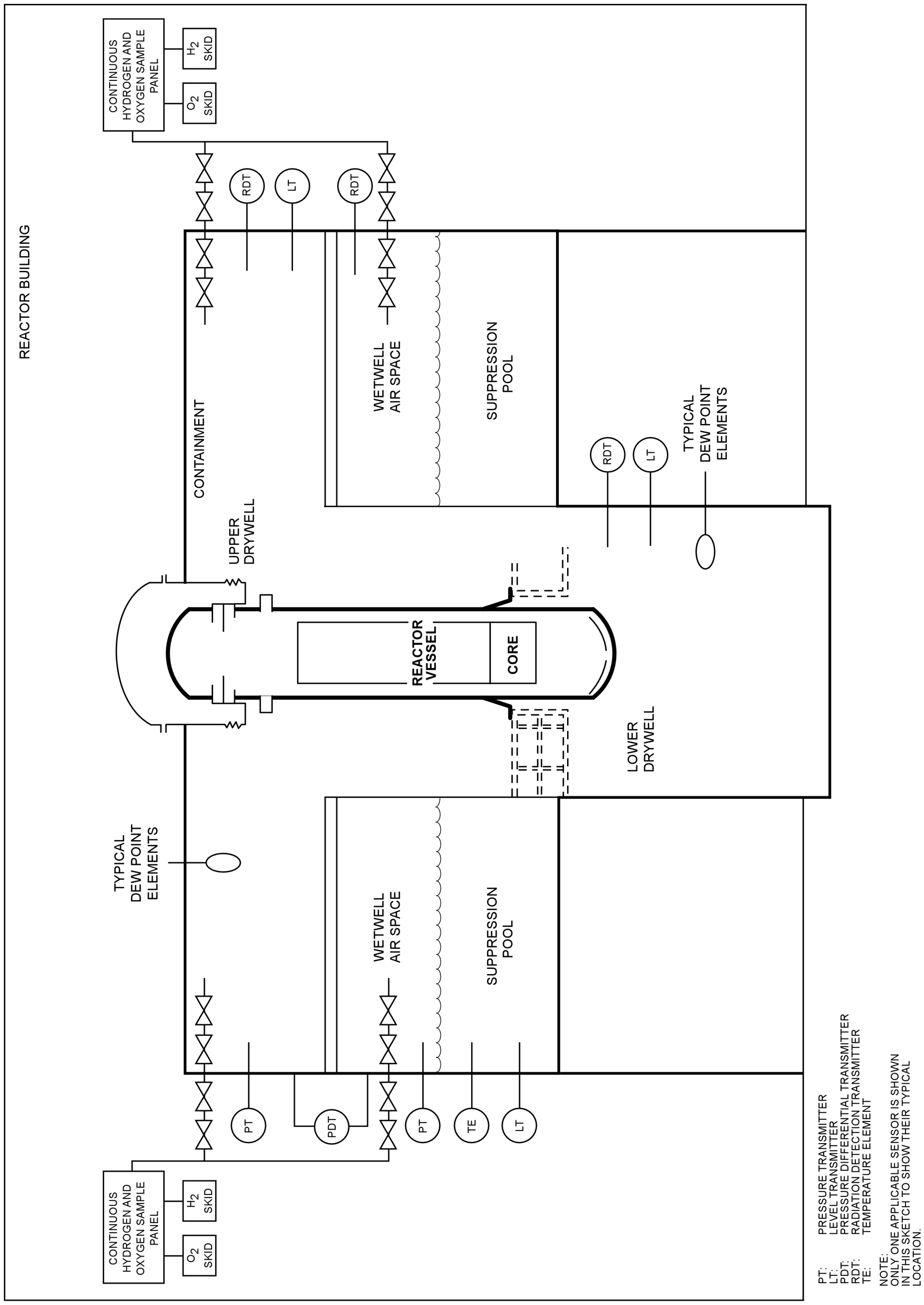


Figure 2.15.7-1. Containment Monitoring System Design

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 reactor building (RB) crane, fuel building (FB) crane and associated lifting devices, such as hoists, and elevators in various areas of the plant are nonsafety-related.

(1) The RB crane has a lifting capacity greater than its heaviest expected load.

(2) The FB crane has a lifting capacity greater than its heaviest expected load.

(3) The RB crane is interlocked to prevent movement of heavy loads over new or spent fuel in the RB.

(4) The FB crane is interlocked to prevent movement of heavy loads over spent fuel storage in the FB.

(5) The RB crane is classified as Seismic Category II to prevent structural failure with SSE loads, leading to degradation of the functioning of a Seismic Category I structure, system, or component to an unacceptable safety level.

(6) The FB crane is classified as Seismic Category II to prevent structural failure with SSE loads, leading to degradation of the functioning of a Seismic Category I structure, system, or component to an unacceptable safety level.

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 RB crane has a lifting capacity greater than its heaviest expected load.	A load test at 125% of the rated capacity will be performed.	Report documents that the RB crane is successfully load tested at 125% of its rated capacity.
2. The FB crane has a lifting capacity greater than its heaviest expected load.	A load test at 125% of the rated capacity will be performed.	Report documents that the FB crane is successfully load tested at 125% of its rated capacity.
3. The RB crane is interlocked to prevent movement of heavy loads over new or spent fuel in the RB.	Tests will be conducted of the as-built RB crane movement using a heavy load.	Report documents that the RB crane interlock prevents the carrying of a load greater than one fuel assembly and its associated handling device over new or spent fuel in the RB.
4. The FB crane is interlocked to prevent movement of heavy loads over spent fuel storage in the FB.	Tests will be conducted of the as-built FB crane movement using a heavy load.	Report documents that the 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.
5. The RB crane is classified as Seismic Category II to prevent structural failure with SSE loads, leading to degradation of the functioning of a Seismic Category I structure, system, or component to an unacceptable safety level.	Inspection of the as-built RB crane will be performed to verify that the design meets Seismic Category II requirements.	Report documents that the RB crane conforms to Seismic Category II requirements.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The FB crane is classified as Seismic Category II to prevent structural failure with SSE loads, leading to degradation of the functioning of a Seismic Category I structure, system, or component to an unacceptable safety level.	Inspection of the as-built FB crane will be performed to verify that the design meets Seismic Category II requirements.	Report documents that the FB crane conforms to Seismic Category II requirements.

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. The RBVS consists of three subsystems. The Reactor Building Clean Area HVAC Subsystem (CLAVS) serves the clean (non-radiologically controlled) areas of the Reactor Building and is shown in Figure 2.16.2-1. The Reactor Building Contaminated Area HVAC Subsystem (CONAVS) serves the potentially contaminated areas of the Reactor Building and is shown in Figure 2.16.2-2. The Reactor Building Refueling and Pool Area HVAC Subsystem (REPAVS) serves the refueling area of the Reactor Building and is shown in Figure 2.16.2-3.

The RBVS automatically isolates the Reactor Building boundary during accidents. The isolation dampers and ducting penetrating the Reactor Building boundary, and associated controls that provide the isolation signal are safety-related. Safety-related components for the RBVS are listed in Table 2.16.2-1.

The remaining portion of the RBVS is nonsafety-related.

(1) The basic configuration of the RBVS is as described in Subsection 2.16.2.1 and is as shown in Figures 2.16.2-1, 2.16.2-2 and 2.16.2-3.

(2) The RBVS isolation dampers automatically close upon receipt of a high radiation signal or loss of AC power.

(3) The safety-related components identified in Table 2.16.2-1 can withstand Seismic Category I loads without loss of safety-related function.

(4) The RBVS maintains the hydrogen concentration levels in the battery rooms below 2% by volume.

(5) CONAVS maintains served areas of the reactor building at a slightly negative pressure relative to surrounding clean areas to minimize the exfiltration of potentially contaminated air.

(6) REPAVS maintains served areas of the reactor building at a slightly negative pressure relative to surrounding clean areas to minimize the exfiltration of potentially contaminated air.

(7) The RBVS provides post 72-hour cooling for DCIS, CRD and RWCU pump rooms.Indications and controls for safety-related components of the RBVS as indicated in Table 2.16.2-1 are available in the main control room (MCR).

(9) Independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.

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 and is shown in Figure 2.16.2-4. The CBGAVS serves the areas outside the CRHA and is shown in Figures 2.16.2-5 and 2.16.2-6.

(1) The basic configuration of the CRHAVS is as described in Subsection 2.16.2.2 and is as shown in Figure 2.16.2-4.

(2) The CRHA isolation dampers automatically close upon receipt of any of the following signals:

   a. high radiation in the CRHAVS intake;

   b. high radiation downstream of an Emergency Filter Unit (EFU) during emergency operation;

   c. low airflow through an EFU during emergency operation; or

   d. loss of AC power.

(3) The safety-related components identified in Table 2.16.2-3 can withstand Seismic Category I loads without loss of safety-related function.

(4) The CRHAVS provides cooling to the CRHA.

   a. In the CRHA, temperature rise on a loss of normal cooling will not exceed 8.3^{o}C (15^{o}F) for 72 hours.

   b. The CRHA heat sink is maintained at or below 25.56°C (78°F).

(5) Independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.

(6) CRHA isolation damper and EFU operational status (Open/Closed) indication is provided 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.

(1) The basic configuration of the EFU is as described in Subsection 2.16.2.3 and is as shown in Figure 2.16.2-4.

(2) The selected redundant EFU dampers open upon receipt of a control room habitability envelope isolation signal.

(3) The safety-related EFU components identified in Table 2.16.2-5 can withstand Seismic Category I loads without loss of safety-related function.

(4) Independence for the EFU trains is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.

(5)

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 cfm).

b. The in-leakage does not exceed the unfiltered in-leakage assumed by control room operator dose analysis.

(6) The powered EFU dampers can be remotely operated from the MCR.

(7) EFUs meet the in-place leakage testing requirements of ASME AG-1 and RG 1.52.

(8) Indications and controls for the safety-related components of the EFU system as indicated in Table 2.16.2-5 are available in the MCR.

(9) The dedicated portable AC generator(s), available on site, is capable of providing post 72-hour power to the EFU fan system.

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 TBVS includes the Turbine Building supply air fans and associated AHUs, and the Turbine Building exhaust fans and associated filter trains.

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.

(1) The basic configuration of the Turbine Building Ventilation System (TBVS) is as described in Subsection 2.16.2.4.

(2) The TBVS provides post 72-hour cooling for DCIS in the Turbine Building and room cooling for the Nuclear Island Chilled Water System and RCCW pumps.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.2-7 provides the design commitments, inspections, tests, analyses and acceptance criteria for the Turbine Building HVAC 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 ventilation systems to mitigate the consequences of fuel handling accidents with significant radiological releases. The Fuel Building HVAC subsystems include the Fuel Building General Area HVAC Subsystem (FBGAVS) shown in Figure 2.16.2-7 and the Fuel Building Fuel Pool HVAC Subsystem (FBFPVS) shown in Figure 2.16.2-8.

(1) The basic configuration of the FBVS is as described in Subsection 2.16.2.5 and is as shown in Figures 2.16.2-7 and 2.16.2-8.

(2) The Fuel Building HVAC isolation dampers automatically close upon receipt of a high radiation signal.

(3) The safety-related components identified in Table 2.16.2-8 can withstand Seismic Category I loads without loss of safety-related function.

(4) The FBVS maintains the fuel building at a slightly negative pressure relative to surrounding areas.

(5) The FBVS provides post 72-hour cooling for FAPCS.

(6) Indications and controls for the safety-related components of the FBVS as indicated in Table 2.16.2-8 are available in the MCR.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.2-9 provides the design commitments, inspections, tests, analyses and acceptance criteria for the Fuel Building HVAC.

2.16.2.6 Radwaste Building HVAC System

No entry for this system.

2.16.2.7 Electrical Building HVAC System

Design Description

The Electrical Building Ventilation System (EBVS) is nonsafety-related and 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).

(1) The basic configuration of the Electrical Building Ventilation System (EBVS) is as described in Subsection 2.16.2.7.

(2) The EBVS provides post 72-hour cooling for Diesel Generators and safety-related Electrical Distribution and support for electrical power to FAPCS.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.2-10 provides the design commitments, inspections, tests, analyses and acceptance criteria for the Electrical Building HVAC System.

2.16.2.8 Other Building HVAC Systems

No entry for this system.

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	AG-1	Closed
CONAVS building exhaust air isolation dampers	I	AG-1	Closed
REPAVS building supply air isolation dampers	I	AG-1	Closed
REPAVS building exhaust air isolation dampers	I	AG-1	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 and is as shown in Figures 2.16.2-1, 2.16.2-2 and 2.16.2-3.	Inspections of the RBVS configuration will be conducted.	Inspection report(s) document that the as-built RBVS conforms to the description in Subsection 2.16.2.1 and is as shown in Figures 2.16.2-1, 2.16.2-2 and 2.16.2-3.
2. The RBVS isolation dampers automatically close upon receipt of a high radiation signal or loss of AC power.	Testing of the RBVS isolation dampers will be performed using simulated signals to close the RBVS isolation dampers.	Test report(s) document that upon receipt of a simulated high radiation signal or a simulated loss of AC power signal, the as-built RBVS isolation dampers automatically close.
3. The safety-related components identified in Table 2.16.2-1 can withstand Seismic Category I loads without loss of safety-related function.	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-built equipment identified in Table 2.16.2-1, including anchorage, is seismically bounded by the testing or analyzed conditions.	a) A report exists and concludes that the equipment identified in Table 2.16.2-1 can withstand seismic design basis without loss of safety-related function. b) A report exists and concluded that the as-built equipment identified in Table 2.16.2-1, including anchorage, conforms to tested or analyzed conditions necessary to ensure functioning following a SSE.

Table 2.16.2-2

ITAAC For The Reactor Building HVAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The RBVS maintains the hydrogen concentration levels in the battery rooms below 2% by volume.	Testing and analysis of the system will be performed to demonstrate the air flow capability of the RBVS is adequate to maintain the hydrogen concentration levels in the battery rooms below 2%.	Analysis and test report(s) demonstrate that the air flow capability of the as-built RBVS is adequate to maintain the hydrogen concentration levels in the battery rooms below 2%.
5. CONAVS maintains served areas of the reactor building at a slightly negative pressure relative to surrounding clean areas to minimize the exfiltration of potentially contaminated air.	a) Testing will be performed to confirm that the contaminated areas of the reactor building served by CONAVS maintain a minimum negative pressure of [62 Pa (-1/4 inch W.G.)] relative to surrounding clean areas when operating CONAVS supply and exhaust fans in the normal system fan lineup. b) 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.	a) Test report(s) document that the time average pressure differential in the as-built 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 as-built CONAVS supply flow rate.

Table 2.16.2-2

ITAAC For The Reactor Building HVAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. REPAVS maintains served areas of the reactor building at a slightly negative pressure relative to surrounding clean areas to minimize the exfiltration of potentially contaminated air.	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.)] relative to surrounding clean areas 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.	a) Test report(s) document that the time average pressure differential in the as-built 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 as-built REPAVS supply flow rate.
7. The RBVS provides post 72-hour cooling for DCIS, CRD and RWCU pump rooms	Testing of the integrated system will be performed to demonstrate the air flow capability of the RBVS to support post-72 hour cooling for DCIS, CRD and RWCU pump rooms.	A report documents that the integrated system test demonstrates the air flow capability to support post-72 hour cooling for DCIS, CRD and RWCU pump rooms.
8. Indications and controls for safety-related components of the RBVS as indicated in Table 2.16.2-1 are available in the main control room (MCR).	Inspection of the MCR will be performed to verify that the safety-related system functions of the RBVS are available.	Inspection report(s) document that indications and controls for the safety-related components of the RBVS as indicated in Table 2.16.2-1 are available in the MCR.

Table 2.16.2-2

ITAAC For The Reactor Building HVAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. Independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	a) Tests will be performed on the RBVS dampers by providing a test signal in only one safety-related division at a time. b) Inspection of the as-built safety-related divisions in the system will be performed.	a) Test reports document that the test signal exists only in the safety-related division under test in the as-built RBVS damper. b) Physical separation or electrical isolation exists between as-built RBVS dampers. Physical separation or electrical isolation exists between safety-related divisions and nonsafety-related equipment.

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	AG-1	Fail Closed
EFU downstream isolation dampers	I	AG-1	Fail-As Is
CRHA Restroom Exhaust isolation dampers	I	AG-1	Fail Closed
CRHA Smoke Exhaust intake isolation dampers	I	AG-1	Fail Closed
CRHA Smoke Exhaust output isolation dampers	I	AG-1	Fail Closed
CRHAVS EFUs	I	AG-1	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 and is as shown in Figure 2.16.2-4.	Inspections of the CRHAVS configuration will be conducted.	Inspection report(s) document that the as-built CRHAVS conforms to the description in Subsection 2.16.2.2 and is as shown in Figure 2.16.2-4.
2. The CRHA isolation dampers automatically close upon receipt of any of the following signals: a) high radiation in the CRHAVS intake; b) high radiation downstream of an Emergency Filter Unit (EFU) during emergency operation; c) low airflow through an EFU during emergency operation; or d) loss of AC power.	Testing of the CRHA isolation dampers will be performed using simulated signals to close the CRHA isolation dampers.	Test reports document that the as-built CRHA isolation dampers automatically close upon receipt of any of the following simulated signals: a) high radiation in the CRHAVS intake; b) a high radiation downstream of an Emergency Filter Unit (EFU) during emergency operation; c) low airflow through an EFU during emergency operation; or d) a loss of AC power signal

Table 2.16.2-4

ITAAC For The Control Building Habitability HVAC Subsystem

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The safety-related components identified in Table 2.16.2-3 can withstand Seismic Category I loads without loss of safety-related function.	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-built equipment identified in Table 2.16.2-3, including anchorage, is seismically bounded by the testing or analyzed conditions.	a) A report exists and concludes that the equipment identified in Table 2.16.2-3 can withstand seismic design basis without loss of safety-related function. b) A report exists and concludes that the as-built equipment identified in Table 2.16.2-3, including anchorage, conforms to tested or analyzed conditions necessary to ensure functioning following a SSE.
4. The CRHAVS provides cooling to the CRHA. a) In the CRHA, 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).	a) The temperature rise in the CRHA will be analyzed. The internal heat loads will be verified to be less than that assumed in the analysis. b) The CRHA air temperature will be calculated and shown to be maintained at or below the maximum assumed initial air and heat sink temperature.	Analysis demonstrates that: a) In the CRHA, the temperature rise on a loss of normal cooling will not exceed 8.3°C (15°F) for 72 hours. b) The average ambient air temperature in the CRHA is calculated to be ≤ 25.56°C (78°F).

Table 2.16.2-4

ITAAC For The Control Building Habitability HVAC Subsystem

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Independence is provided between safety-related divisions, and between safety-related divisions and nonsafety-related equipment.	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-built safety-related divisions in the system will be performed.	a. Test reports document that the test signal exists only in the safety-related division under test in the as-built CRHA isolation damper and EFU control. b. Physical separation or electrical isolation exists between as-built CRHA isolation dampers and EFU safety-related divisions. Physical separation or electrical isolation exists between safety-related divisions and nonsafety-related equipment.
6. CRHA isolation damper and EFU operational status (Open/Closed) indication is provided in the MCR.	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.	a. Inspection report(s) document that the as-built CRHA isolation damper and EFU operational status indication is provided in the MCR. b. A report exists and concludes that the operational status indication accurately depicts the operational status of the as-built CRHA isolation dampers and EFUs.

Table 2.16.2-5

Emergency Filter Units

Component	Seismic Category	ASME Code	Notes
EFU (fan, HEPA, and charcoal filters)	I	AG-1	Minimum flow rate of 200 l/s (424 cfm) [9.5 l/s (20 cfm) per person for up to 21 persons], Independent trains
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 and is as shown in Figure 2.16.2-4.	Inspections of the EFU configuration will be conducted.	Inspection report(s) document that the as-built EFU system conforms with the description in Subsection 2.16.2.3 and is as shown in Figure 2.16.2-4.
2. The selected redundant EFU dampers open upon receipt of a control room habitability envelope isolation signal.	Testing of the EFU dampers will be performed using simulated control room habitability envelope isolation signal to open the EFU dampers.	Test report(s) document that upon receipt of a simulated control room habitability envelope isolation signal, the as-built EFU dampers automatically open.
3. The safety-related EFU components identified in Table 2.16.2-5 can withstand Seismic Category I loads without loss of safety-related function.	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-built equipment identified in Table 2.16.2-5, including anchorage, is seismically bounded by the testing or analyzed conditions.	a) A report exists and concludes that the equipment identified in Table 2.16.2-5 can withstand seismic design basis without loss of safety-related function. b) A report exists and concluded that the as-built equipment identified in Table 2.16.2-5, including anchorage, conforms to tested or analyzed conditions necessary to ensure functioning following a SSE.

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.	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-built safety-related divisions in the EFU system will be performed.	a) Test report(s) document that the test signal exists only in the safety-related division under test for the EFU trains. b) Inspection report(s) document that, for the as-built 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. a) EFUs maintain the CRHA at a positive pressure of > 31 pascals (0.125 inch water gauge) with respect to the surrounding adjacent areas at the required air addition flow rate of 200 l/s (424 cfm). b) The in-leakage does not exceed the unfiltered in-leakage assumed by control room operator dose analysis.	a) Testing will be performed to measure the differential pressure between the CRHA and surrounding adjacent areas. b) Tracer gas testing in accordance with ASTM E741 will be performed to measure the unfiltered in-leakage into the CRHA with EFUs operating.	Test report(s) document a) The as-built EFUs maintain the CRHA at a positive pressure of > 31 pascals (0.125 inch water gauge) with respect to the surrounding adjacent areas is maintained with EFU operating. b) The unfiltered in-leakage measured by tracer gas testing does not exceed the unfiltered in-leakage assumed by control room operator dose analysis.
6. The powered EFU dampers can be remotely operated from the MCR.	EFU dampers will be opened and closed using manually initiated signals from the MCR.	Test report(s) document that the as-built EFU dampers open and close when manually imitated signals are sent from the MCR.

Table 2.16.2-6

ITAAC For Emergency Filter Units

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. EFUs meet the in-place leakage testing requirements of ASME AG-1 and RG 1.52.	EFUs will be in-place leak tested in accordance with ASME AG-1, Section TA, to meet the requirements of RG 1.52.	Test report(s) document that the as-built 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.
8. Indications and controls for the safety-related components of the EFU system as indicated in Table 2.16.2-5 are available in the MCR.	Inspection of the MCR will be performed to verify that the safety-related functions of the EFU system are available.	Inspection report(s) document that indications and controls for the safety-related components of the EFU system as indicated in Table 2.16.2-5 are available in the MCR.
9. The dedicated portable AC generator(s), available on site, is capable of providing post 72-hour power to the EFU fan system.	Inspection will be performed to verify that the dedicated portable AC generator(s) is available on site.	Inspection report(s) document that the dedicated portable AC generator(s), capable of providing post 72-hour power to the EFU fan system, is physically located on site.

Table 2.16.2-7

ITAAC For Turbine Building Ventilation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Turbine Building Ventilation System (TBVS) is as described in Subsection 2.16.2.4.	Inspections of the TBVS configuration will be conducted.	Inspection report(s) document that the as-built TBVS system conforms with the description in Subsection 2.16.2.4.
2. The TBVS provides post 72-hour cooling for DCIS in the Turbine Building and room cooling for the Nuclear Island Chilled Water System and RCCW pumps.	System testing will be performed and cooling air flow to the specified cubicles will be verified	Test report(s) document that the cooling air flow capability is adequate to support post-72 hour cooling for DCIS in the Turbine Building and room cooling for the Nuclear Island Chilled Water System and RCCW pumps.

Table 2.16.2-8

Fuel Building HVAC System Safety-Related Components

Component	Seismic Category	ASME Code Classification	Fail Safe Position
FBGAVS building supply air isolation dampers	I	AG-1	Closed
FBGAVS building exhaust air isolation dampers	I	AG-1	Closed
FBFPVS building supply air isolation dampers	I	AG-1	Closed
FBFPVS building exhaust air isolation dampers	I	AG-1	Closed

Table 2.16.2-9

ITAAC For The Fuel Building HVAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the FBVS is as described in Subsection 2.16.2.5 and is as shown in Figures 2.16.2-7 and 2.16.2-8.	Inspections of the Fuel Building HVAC configuration will be conducted.	Inspection report(s) document that the as-built FBVS system conforms to the description in Subsection 2.16.2.5 and is as shown in Figures 2.16.2-7 and 2.16.2-8.
2. The Fuel Building HVAC isolation dampers automatically close upon receipt of a high radiation signal.	Using a simulated high radiation signal, tests will be performed on the (Fuel Building HVAC isolation dampers) isolation logic.	Upon receipt of a simulated high radiation signal, the Fuel Building HVAC isolation dampers automatically close.
3. The safety-related components identified in Table 2.16.2-8 can withstand Seismic Category I loads without loss of safety-related function.	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-built equipment identified in Table 2.16.2-8, including anchorage, is seismically bounded by the testing or analyzed conditions.	a) A report exists and concludes that the equipment identified in Table 2.16.2-8 can withstand seismic design basis events without loss of safety-related function. b) A report exists and concluded that the as-built FBVS equipment, including anchorage, conforms to tested or analyzed conditions necessary to ensure functioning following a SSE.

Table 2.16.2-9

ITAAC For The Fuel Building HVAC

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The FBVS maintains the fuel building at a slightly negative pressure relative to surrounding areas.	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.	a) Test report(s) document that the average differential pressure 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(s) document that the exhaust flow rate is greater than or equal to the FBVS supply flow rate.
5. The FBVS provides post 72-hour cooling for FAPCS.	System testing will be performed and cooling air flow to the specified cubicles will be verified.	Test report(s) documents that the cooling air flow capability is adequate to support post-72 hour cooling for FAPCS.
6. Indications and controls for the safety-related components of the FBVS as indicated in Table 2.16.2-8 are available in the MCR.	Inspection of the MCR will be performed to verify that the safety-related indication and control functions of the FBVS are available.	Inspection report(s) document that indications and controls for the safety-related components of the FBVS as indicated in Table 2.16.2-8 are available in the MCR.

Table 2.16.2-10

ITAAC For Electrical Building Ventilation System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The basic configuration of the Electrical Building Ventilation System (EBVS) is as described in Subsection 2.16.2.7.	Inspections of the EBVS configuration will be conducted.	Inspection report(s) document that the as-built EBVS system conforms with the description in Subsection 2.16.2.7.
2. The EBVS provides post 72-hour cooling for Diesel Generators and Safety-Related Electrical Distribution, and support for electrical power to FAPCS.	System testing will be performed and cooling air flow to the specified cubicles will be verified.	Test report(s) documents that the cooling air flow capability is adequate to support post-72 hour cooling for Diesel Generators and safety-related Electrical Distribution, and support for electrical power to FAPCS.

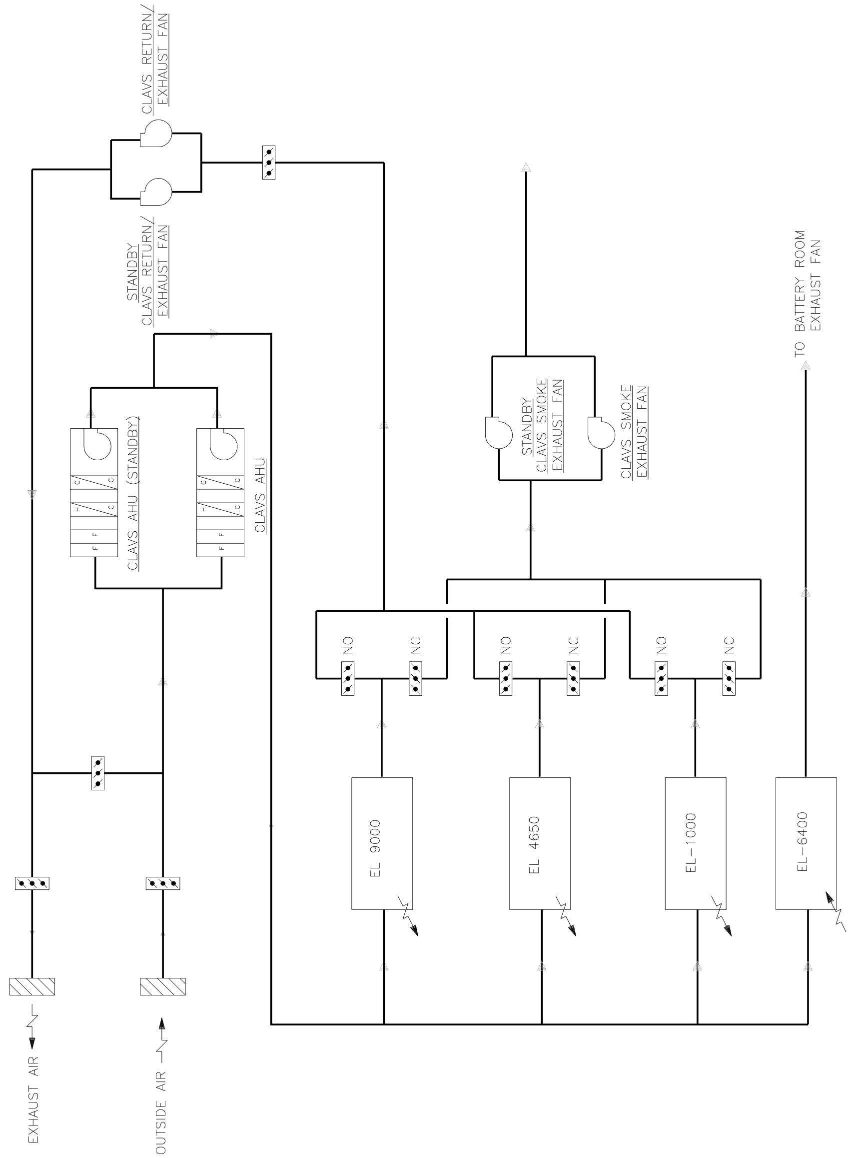


Figure 2.16.2-1. CLAVS Simplified System Diagram

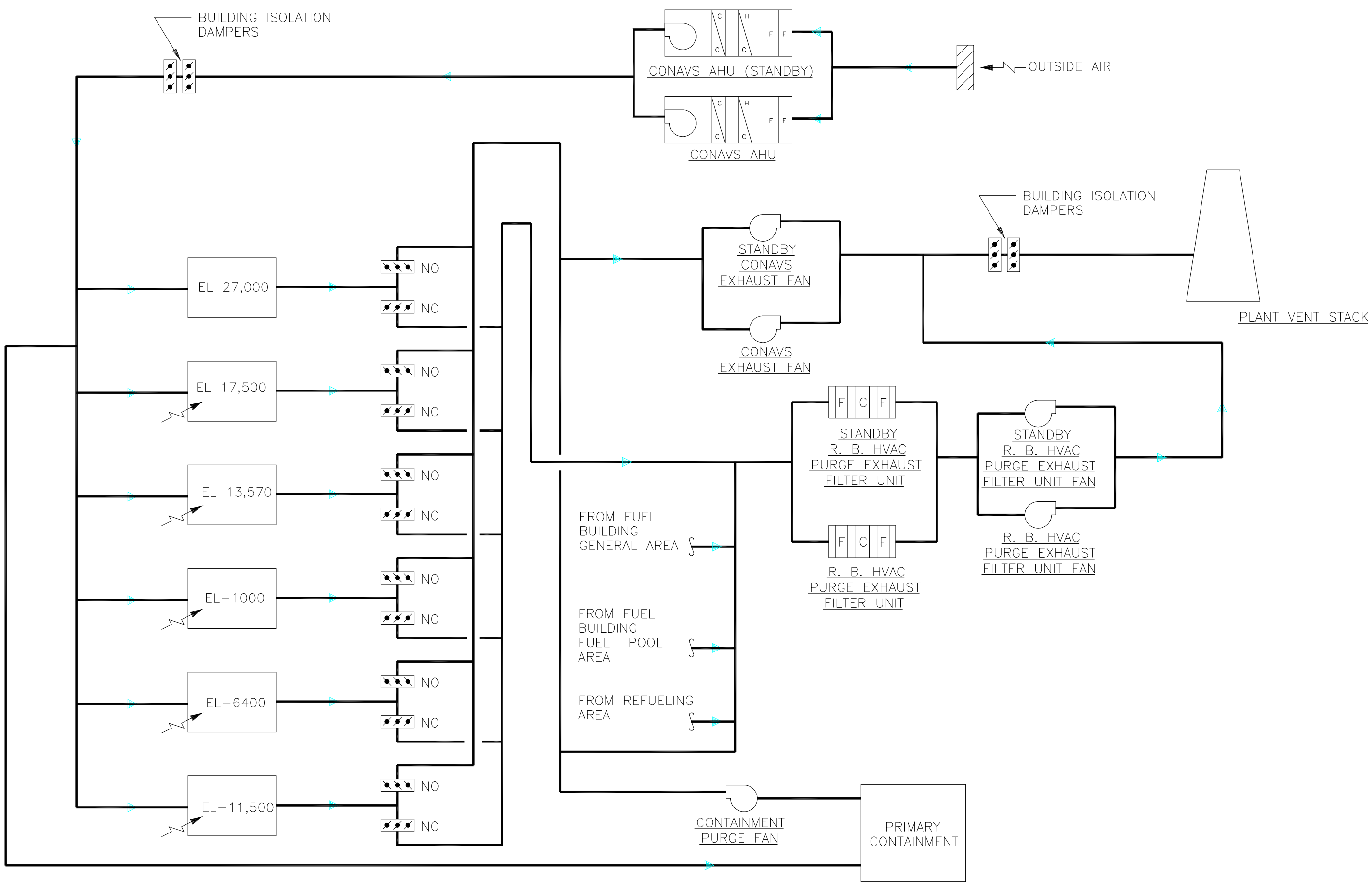


Figure 2.16.2-2. CONAVS Simplified System Diagram

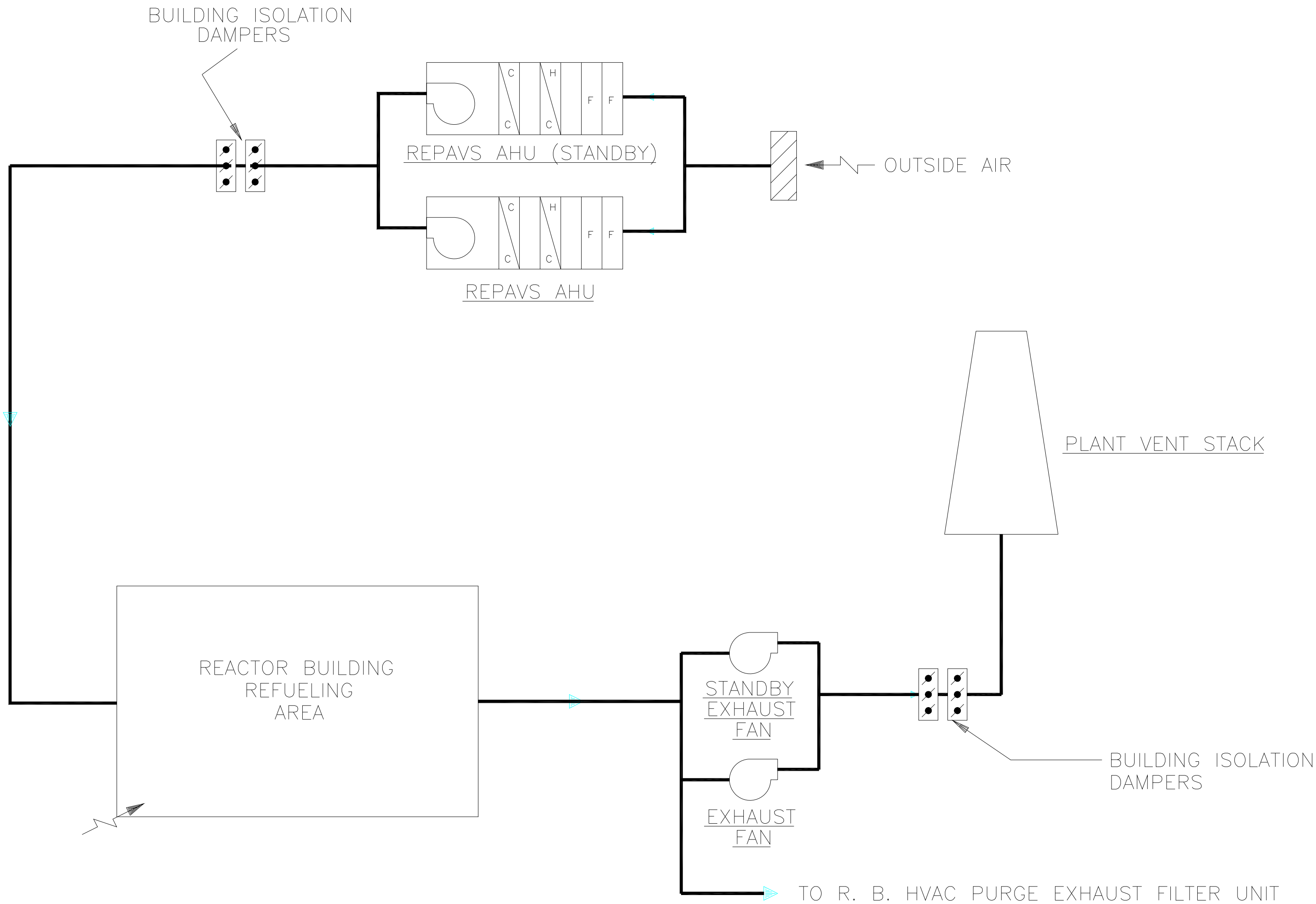


Figure 2.16.2-3. REPAVS Simplified System Diagram

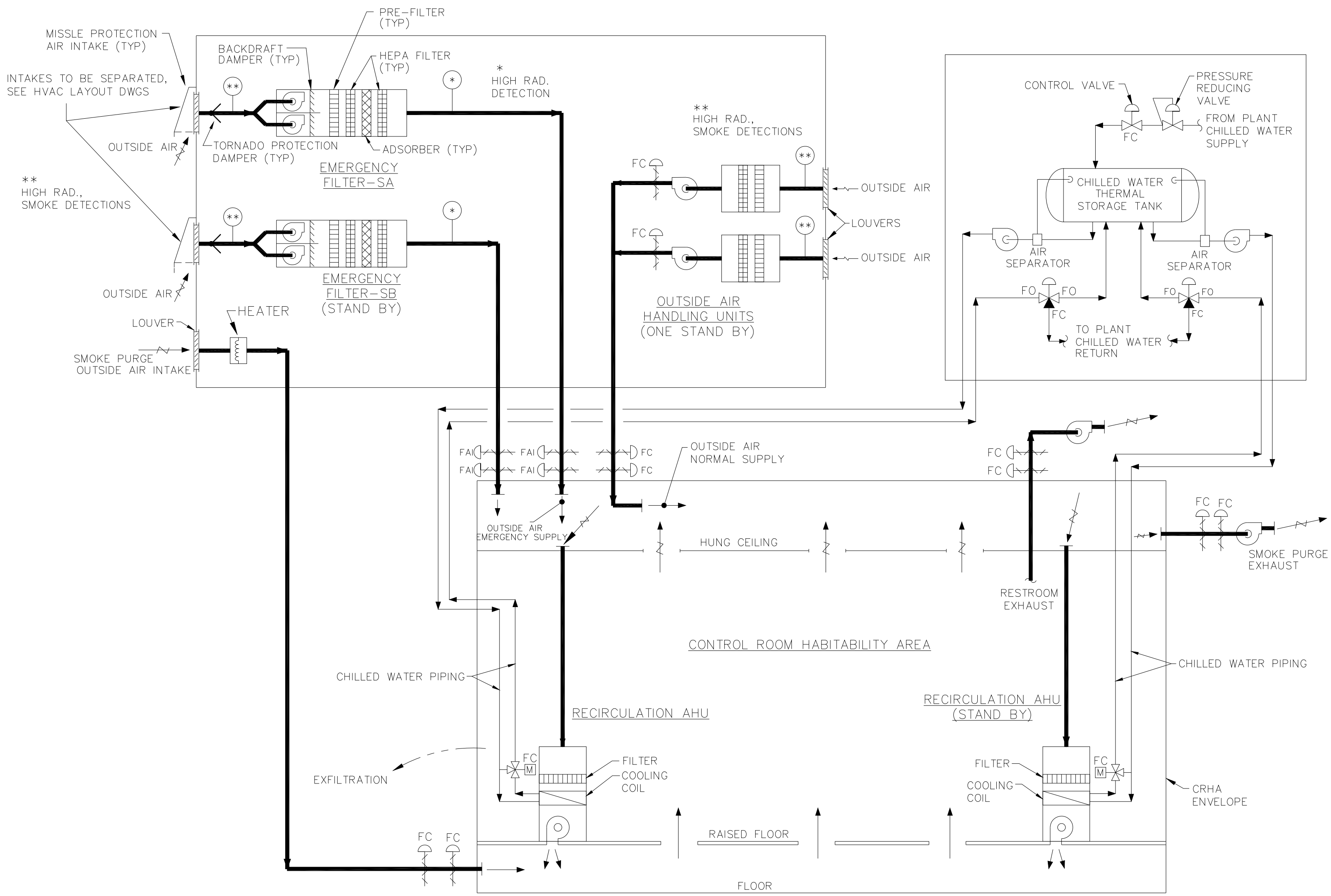


Figure 2.16.2-4. CRHAVS Simplified System Diagram

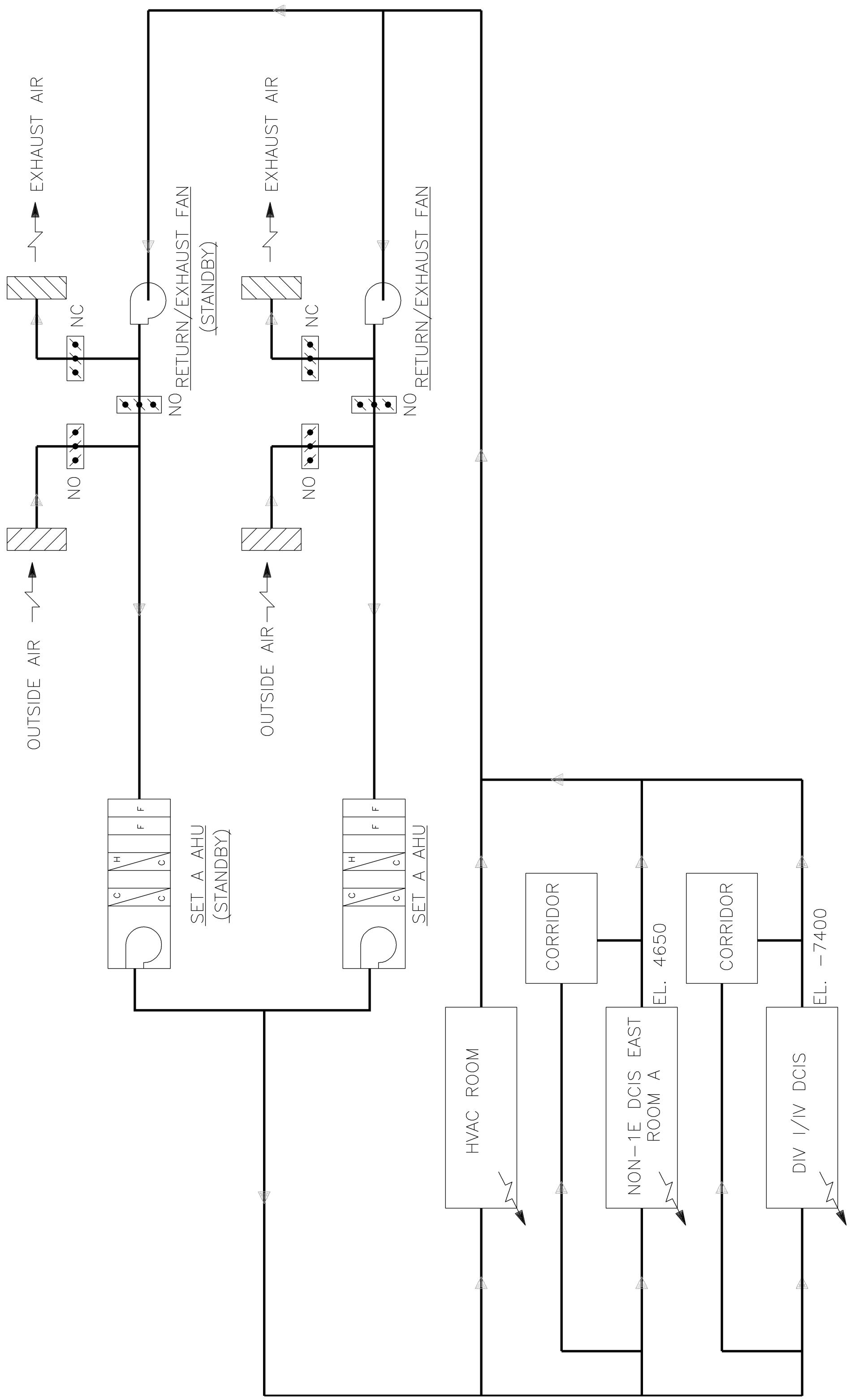


Figure 2.16.2-5. CBGAVS (Set A) Simplified System Diagram

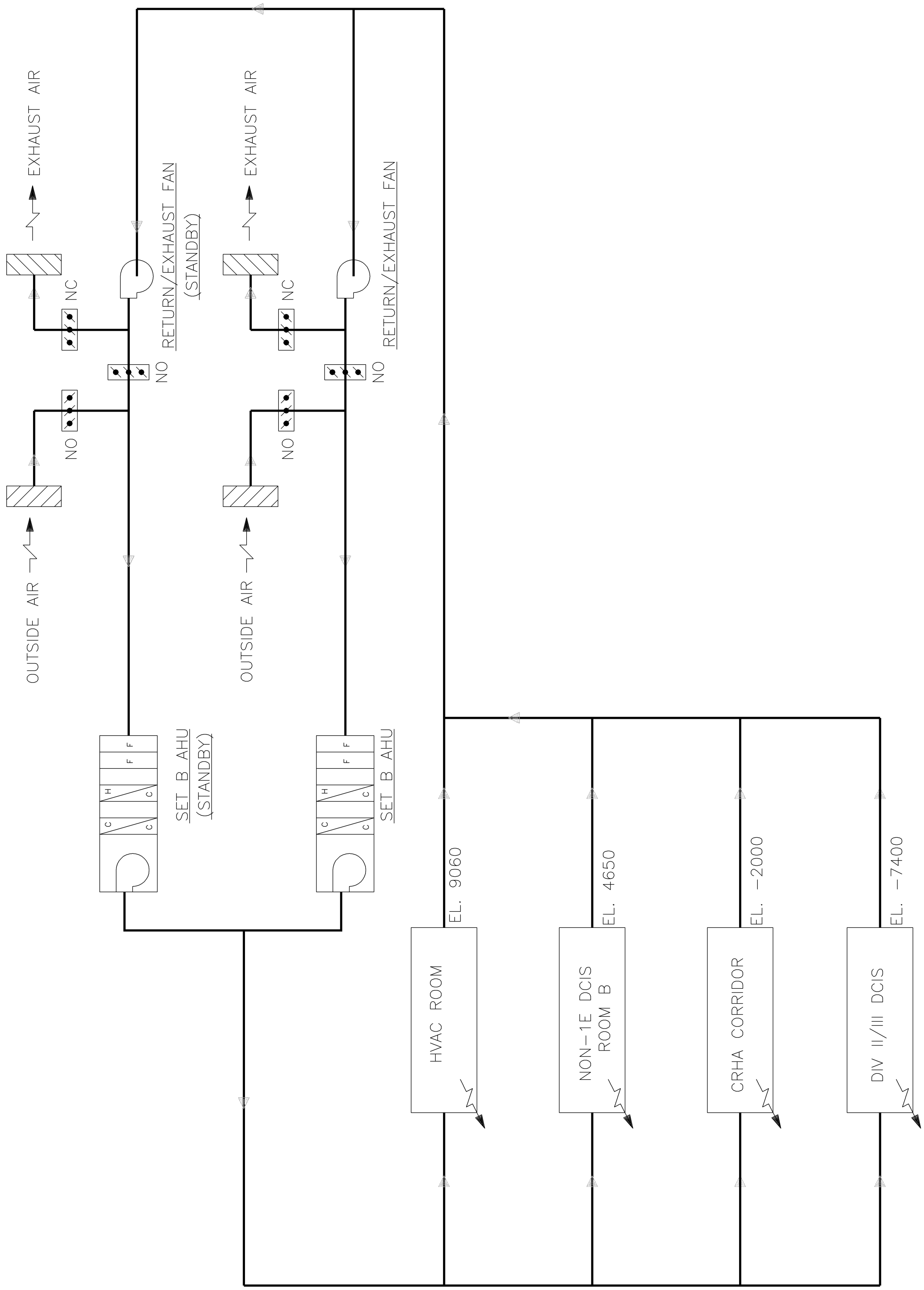


Figure 2.16.2-6. CBGAVS (Set B) Simplified System Diagram

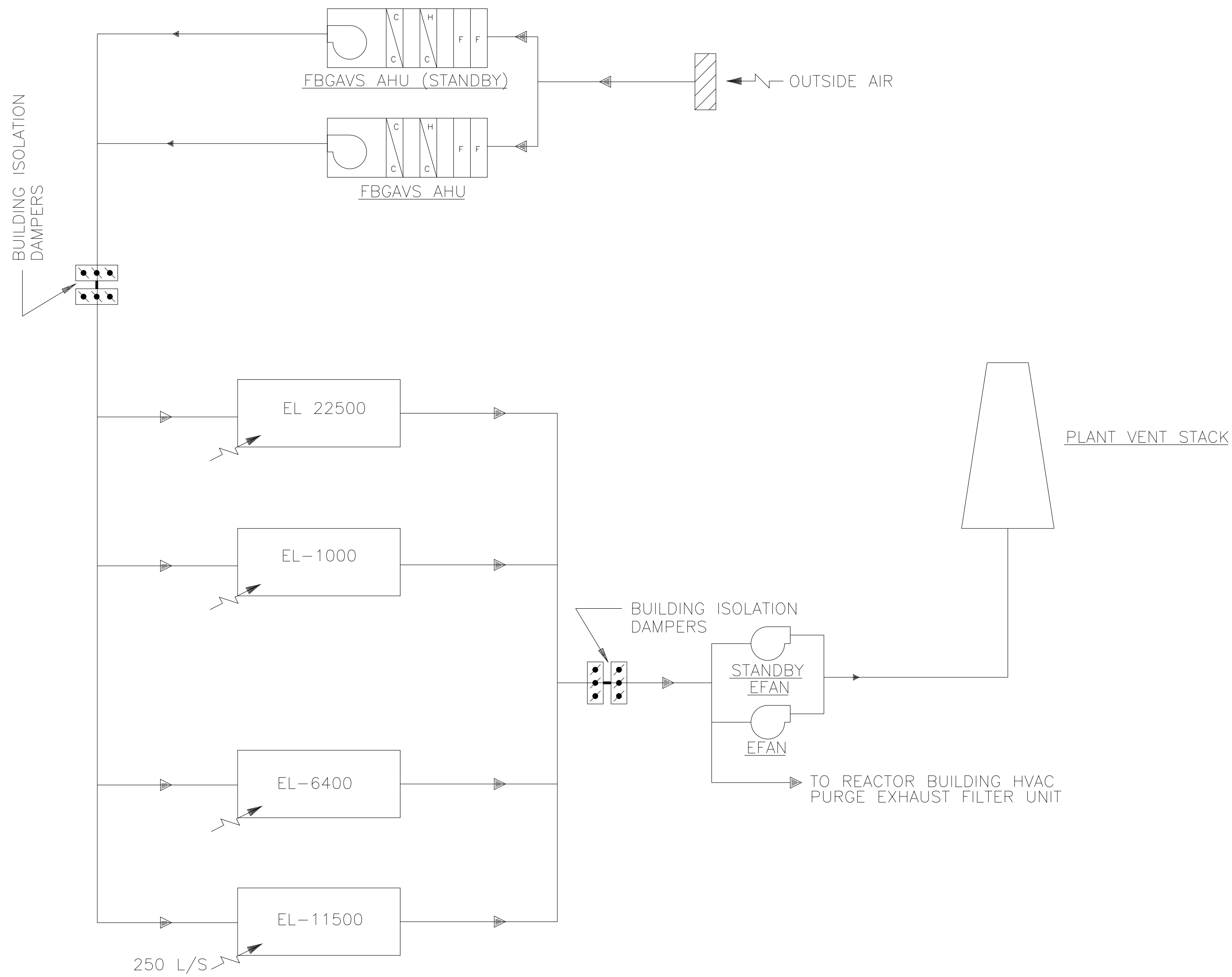


Figure 2.16.2-7. FBGAVS Simplified System Diagram

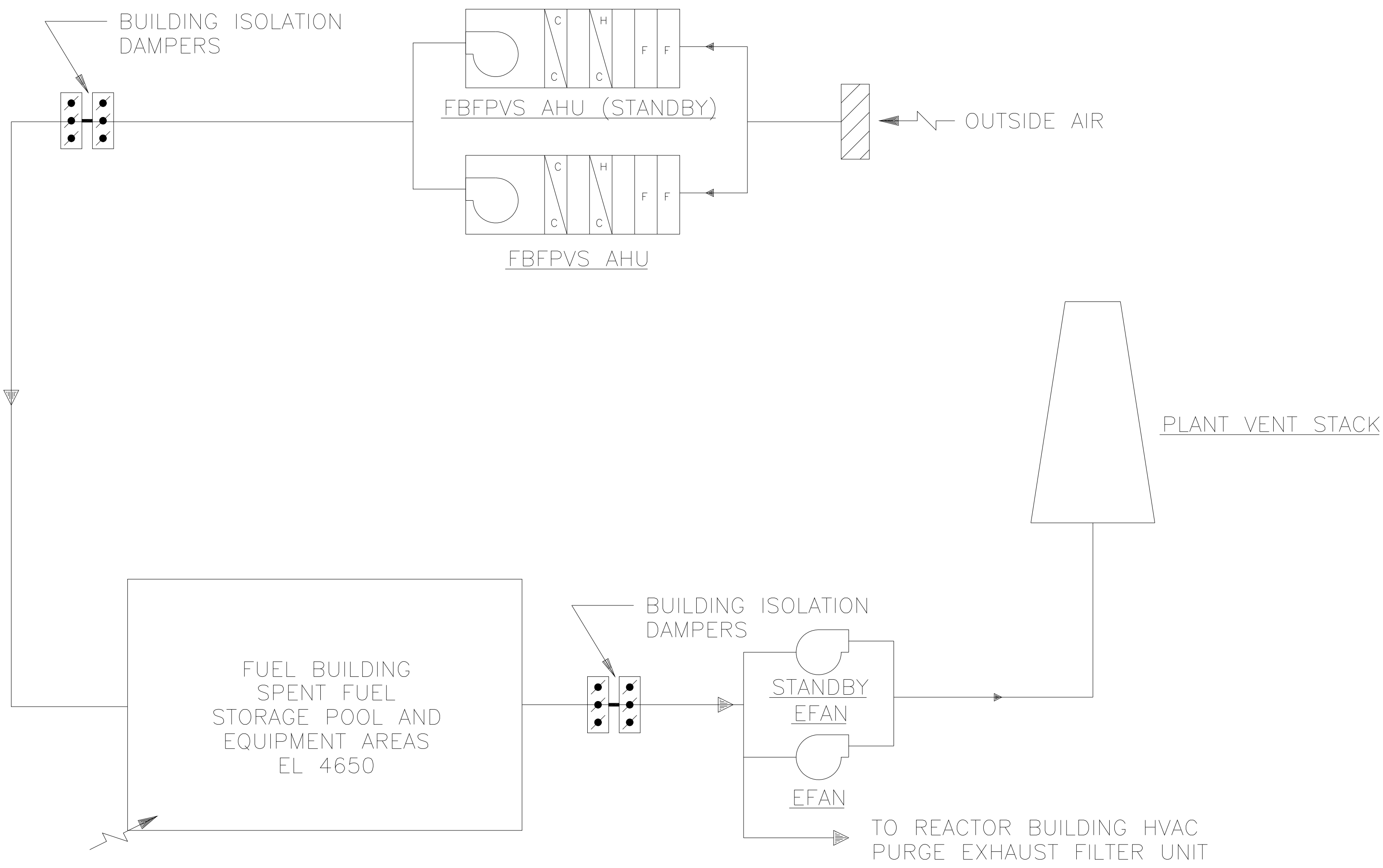


Figure 2.16.2-8. FBFPVS Simplified System Diagram

2.16.3 Fire Protection System

Design Description

The Fire Protection System (FPS) is a nonsafety-related, integrated complex of components and equipment that detects and suppresses fires in the plant.

The FPS is a as shown in Figure 2.16.3-1 and the component locations of the FPS are as shown in Table 2.16.3-1

(1) The functional arrangement of the FPS is as described in the Design Description of this Section 2.16.3 and as shown in Figure 2.16.3-1.

(2) The FPS components and piping identified in Table 2.16.3-1 remain functional during and after an SSE.

(3) The FPS provides for manual fire suppression capability to plant areas containing safety-related equipment, including those that have automatic fire suppression systems , excluding the containment vessel.

(4) The FPS provides two firewater storage sources.

   a. The Primary storage tanks contain a combined minimum usable firewater storage capacity of ≥3900 m^3 (1,030,000 gallons) of water.

   b. The designated site-specific Secondary firewater storage source contains a combined minimum usable firewater storage capacity ≥1135.6 m^3 (300,000 gallons) of water.

(5) Each fire pump provides at least 454.2 m^3/hr (2000 gpm) discharge flow with adequate pressure.

(6) Smoke detectors provide fire detection capability and can be used to initiate fire alarms in areas containing safety-related equipment.

(7) The FPS provides post 72-hour makeup to the IC/PCC pools and Spent Fuel Pool.

   a. The primary diesel-driven fire pump is available to provide post-72 hour makeup to the IC/PCC pools and Spent Fuel Pool.

   b. The fuel oil tank for the primary diesel-driven fire pump contains adequate fuel oil capacity for 96 hours of fire pump operation.

(8) The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations and emergency procedure guidelines, is available in the main control room..

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.3-2 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

Principal Components	Location(s)	Seismic Category
Seismic Category I piping loop and valves including supports	OO, RB, CB, FB	I
Fire water storage tank	OO	I
Fire pump enclosure	OO	I
Seismic Category I pump including diesel-engine drive	OO	I

Location codes:

CB = Control Building
RB = Reactor Building
OO = Outdoors Onsite
FB = Fuel Building

Table 2.16.3-2

ITAAC For The Fire Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the FPS is as described in Subsection 2.16.3 and Figure 2.16.3-1.	Inspection of the as-built system will be conducted.	Inspection report(s) document that the as-built FPS conforms with the basic configuration contained in the Design Description of Subsection 2.16.3 and Figure 2.16.3-1.
2. The FPS components and piping identified in Table 2.16.3-1 remain functional during and after an SSE.	Analysis of the FPS components and piping identified in Table 2.16.3-1 will be performed to demonstrate that the components and piping will remain functional during and after an SSE. Inspection of the as-built FPS components and piping identified in Table 2.16.3-1 will be performed to verify that the components and piping are installed in accordance with the configurations specified by the analyses.	Analyses demonstrate that the FPS components and piping identified in Table 2.16.3-1 will remain functional during and after an SSE. Inspection report(s) document that the as-built components and piping identified in Table 2.16.3-1 are installed in accordance with the configurations specified by the analyses.
3. The FPS provides for manual fire suppression capability to plant areas containing safety-related equipment, including those that have automatic fire suppression systems, excluding the containment vessel.	Inspection of the as-built manual fire suppression system will be performed to verify that any location that contains or could present a hazard to safety-related equipment can be reached by at least one effective hose stream with a maximum of 30.5 meters (100 feet) of hose.	Inspection report(s) of the as-built manual fire suppression system document that standpipes and manual hose stations exist. 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. Standpipe and hose rack stations are located such that no safe shutdown equipment is more than 30.5 m (100 ft) from two hose stations on separate standpipes.

Table 2.16.3-2

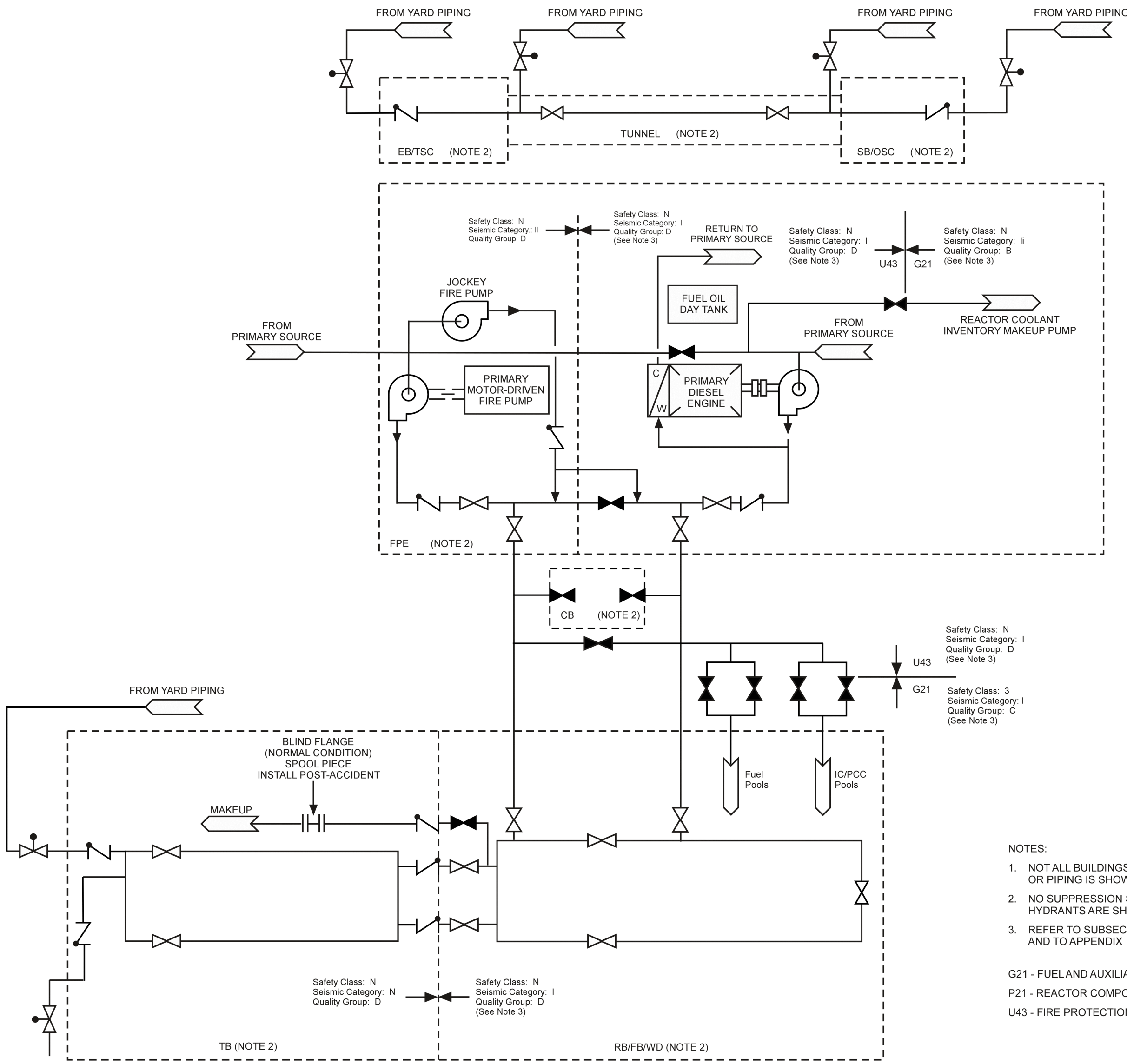
ITAAC For The Fire Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. The FPS provides two firewater storage sources. a. The Primary storage tanks contain a combined minimum usable firewater storage capacity of ≥3900 m^3 (1,030,000 gallons).	Inspection of the as-built water supply sources and volumetric calculations using as-built dimensions will be performed.	Inspection report(s) document that the as-built water supply sources meet the volumetric requirements specified in the Certified Design Commitment.
b. The designated site-specific Secondary firewater storage source contains a combined minimum usable firewater storage capacity ≥1135.6 m^3 (300,000) gallons.	Inspection of the as-built water supply sources and volumetric calculations using as-built dimensions will be performed.	Inspection report(s) document that the as-built water supply sources meet the volumetric requirements specified in the Certified Design Commitment.
5. Each fire pump provides at least 454.2 m^3/hr (2000 gpm) discharge flow with adequate pressure.	i) Testing and/or analysis of each fire pump will be performed to demonstrate that each fire pump provides a flow rate of at least 454.2 m^3/hr (2000 gpm). ii) Testing will be performed to demonstrate rated flow and rated water pressure at the most hydraulically remote standpipes in the Turbine Building and the Reactor Building.	i) Test report(s) and/or analysis demonstrate that each fire pump provides a flow rate of at least 454.4 m^3/hr (2000 gpm). ii) Test reports document acceptable flow and rated pressure at the most hydraulically remote Turbine Building and Reactor standpipe - a.) 40 mm (1.5 inch) hoses; total flow of 22.7 m^3/hr (100 gpm) at a minimum pressure of 448.2 kPaG (65 psig) and b.) 65 mm (2.5 inch) hoses; total flow of 113.5 m^3/hr (500 gpm) at a minimum pressure of 689 kPaG (100 psig).

Table 2.16.3-2

ITAAC For The Fire Protection System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Smoke detectors provide fire detection capability and can be used to initiate fire alarms in areas containing safety-related equipment.	Testing will be performed on the as-built individual fire detectors in areas containing safety-related equipment by providing a simulated fire condition.	Test report(s) document that the as-built individual smoke detectors respond to simulated fire conditions and initiate fire alarms in area containing safety-related equipment.
7. The FPS provides post 72-hour makeup to the IC/PCC pools and Spent Fuel Pool. a. The primary diesel-driven fire pump is available to provide post-72 hour makeup to the IC/PCC pools and Spent Fuel Pool.	Test will be performed to demonstrate that the primary diesel-driven fire pump starts on a manual signal and supplies a total of 45.4 m^3/hr (≥200 gpm) make up water to the IC/PCCS pool and the Spent Fuel Pool.	Test report(s) document that the primary diesel-driven fire pump starts on a manual signal and supplies a total of 45.4 m^3/hr (≥200 gpm) make-up water to the IC/PCCS pools and Spent Fuel Pool.
b. The fuel oil tank for the primary diesel-driven fire pump contains adequate fuel oil capacity for 96 hours of fire pump operation.	The as-built primary diesel-driven fire pump fuel oil tank capacity will be calculated.	Test report documents that the as-built fuel oil tanks for the diesel-driven fire pumps have greater than a 3.79 m^3 (1000 gallon) 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.
8. The minimum set of displays, alarms and controls, based on the applicable codes and standards, including HFE evaluations and emergency procedure guidelines, is available in the main control room.	Inspection of the as-built main control room will verify that the minimum set of displays, alarms and controls for the FPS is available.	Inspection report(s) document that the minimum set of displays, alarms and controls for the FPS, as defined by the applicable codes and standards, including HFE evaluations and emergency operating procedures, exist in the as-built main control room.



NOTES:

1. NOT ALL BUILDINGS, VALVES, PUMPS, OR PIPING IS SHOWN.
2. NO SUPPRESSION SYSTEMS, STANDPIPES OR HYDRANTS ARE SHOWN.
3. REFER TO SUBSECTION 3.2 FOR ADDITIONAL INFORMATION AND TO APPENDIX 19A FOR RTNSS REQUIREMENTS.

G21 - FUEL AND AUXILIARY POOLS COOLING SYSTEM

P21 - REACTOR COMPONET COOLING WATER

U43 - FIRE PROTECTION SYSTEM

Figure 2.16.3-1. Fire Protection System

2.16.3.1 Fire Barriers

Design Description

A Fire Barrier is a continuous vertical or horizontal fire-resistance rated construction assembly designed and constructed to limit the spread of heat and fire and to restrict the movement of smoke. Fire dampers protect ventilation duct openings in fire barriers.

(1) Fire barriers of 3-hour fire resistance rating are provided that separate:

- 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.

(2) Penetrations through fire barriers are sealed or closed to provide fire resistance ratings at least equal to that of the barriers.

(3) Fire dampers protect ventilation duct openings in fire barriers.

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.	Inspections will assure 3-hour fire barriers are installed.	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.	Inspections will be performed to confirm that penetrations through fire barriers are sealed or closed.	Inspection report(s) document that the as-built fire penetrations in the fire areas listed in Table 2.16.3.1-1 are sealed or closed.
3. Fire dampers protect ventilation duct openings in fire barriers.	Inspections will be performed to confirm the presence of fire dampers in ventilation duct openings.	Inspection report(s) document the presence of fire dampers in ventilation duct openings, consistent with the fire areas identified in Table 2.16.3.1-1.

2.16.4 Equipment and Floor Drain System

Design Description

The Equipment and Floor Drain System (EFDS) collects waste liquids from their point of origin and transfers them to a suitable processing or disposal system. The Reactor Coolant Pressure Boundary (RCPB) leakage detection systems utilize features of the EFDS to provide a means of detecting and, to the extent practical, identifying the source of the reactor coolant leakage.

The detection of small, unidentified leakage within the drywell is accomplished by monitoring the drywell floor drain high conductivity waste (HCW) sump pump activity and the drywell sump level changes;

The detection of small, identified leakage within the drywell is accomplished by monitoring the drywell equipment drain [Low Conductivity Waste (LCW)] sump pump activity and sump level increases.

(1) The functional arrangement of the EFDS is as described in this Section 2.16.4.

(2) The EFDS collects liquid wastes from the equipment and floor drainage in the drywell and directs these wastes to the drywell floor drain high conductivity waste (HCW) sump.

(3) The EFDS collects liquid wastes emanating from the large process valves' stem packing seals in the drywell and directs these wastes to the drywell equipment drain [Low Conductivity Waste (LCW)] sump.

(4) The containment isolation portion of the EFDS is addressed in Tier 1, Subsection 2.15.1.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.16.4-1 provides a definition of the inspections, test and/or analyses, together with associated acceptance criteria for the Equipment and Floor Drain System.

Table 2.16.4-1

ITAAC For The Equipment and Floor Drain System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The functional arrangement of the EFDS is as described in Section 2.16.4.	Inspections of the as-built EFDS will be performed.	Inspection report(s) document that the as-built EFDS conforms with the description in Subsection 2.16.4.
2. The EFDS collects liquid wastes from the equipment and floor drainage in the drywell and directs these wastes to the drywell floor drain high conductivity waste (HCW) sump.	A test will be performed by pouring water into the equipment and floor drains in the drywell inside the containment boundary.	Test report(s) demonstrate that the water poured into these drains is collected in the drywell floor drain high conductivity waste (HCW) sump.
3. The EFDS collects liquid wastes emanating from the large process valves' stem packing seals in the drywell and directs these wastes to the drywell equipment drain [Low Conductivity Waste (LCW)] sump.	A test will be performed by pouring water into the stem packing seal leak-off lines of large process valves in the drywell inside the containment boundary.	Test report(s) demonstrate that the water poured into these leak-off lines is collected in the drywell equipment drain [Low Conductivity Waste (LCW)] sump.
4. The containment isolation portion of the EFDS is addressed in Tier 1, Subsection 2.15.1.	See Tier 1, Subsection 2.15.1.	See Tier 1, Subsection 2.15.1.

2.16.5 Reactor Building

Design Description

The Reactor Building (RB) 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 Inclined Fuel Transfer System and the overhead crane. The Isolation Condenser/Passive Containment Cooling System pools are below the refueling floor.

The RB structure is integrated with a 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 key characteristics of the RB are as follows:

(1) The RB is designed and constructed to accommodate the dynamic, static and thermal 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 including LOCA and missiles.
- Normal plant operation—live loads, dead loads, temperature effects and building vibration loads.

(2) The physical arrangement of the RB is as shown in Figures 2.16.5-1 through 2.16.5-11.

(3) The critical dimensions used for seismic analyses and the acceptable tolerances are provided in Table 2.16.5-1.

(4) The RB offers some holdup and decay of fission products that may leak from the containment after an accident. Assuming a LOCA, the offsite dose limits and the control room dose limits are met based on a 50 wt% per day leakage rate from the RB.

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

(6) 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:

a. Wall thicknesses below flood level designed to withstand hydrostatic loads;

b. Water stops provided in all expansion and construction joints below flood and groundwater levels;

c. Waterproofing of below flood and groundwater levels external surfaces;

d. Water seals in external walls at pipe penetrations below flood and groundwater levels; and

e. Roofs designed to prevent pooling of large amounts of water in excess of the structural capacity of the roof for design loads.

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

a. Structural enclosures or barriers;

b. Curbs and sills;

c. Leakage detection components; and

d. Drainage systems.

(8) The internal flooding protection features prevent flood water in one division from propagating to other division(s) and ensure equipment necessary for safe shutdown is located above the maximum flood level for that location or is qualified for flood conditions by:

a. Divisional walls

b. Sills

c. Watertight doors.

(9) The RB is protected against pressurization effects associated with postulated rupture of pipes containing high-energy fluid that occur in subcompartments of the RB.

Inspections, Tests, Analyses and Acceptance Criteria

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

Table 2.16.5-1

Critical Dimensions of Reactor Building – Part 1

Label	Wall or Section Description	Column Line or Region	Floor Elevation or Elevation Range (EL: mm)	Concrete Thickness (mm)	Tolerance (mm)
	Concrete Containment				
101	RPV Pedestal Cylinder	Not Applicable	From -10400 to 4650	2500	+60/-0
102	RCCV Cylinder	Not Applicable	From 4650 to 24600	2000	+60/-0
103	Containment Basemat	Below RPV Pedestal Cylinder	-10400	5100	+50/-20
104	Suppression Pool Slab	Between RPV Pedestal Cylinder and RCCV Cylinder	4650	2000	+60/-0
105	Top Slab	From Drywell Head to RCCV Cylinder	27000	2400	+60/-0
	Outside Concrete Containment				
1	Wall at Column Line R1	From RA to RG	From -11500 to 4650	2000	+25/-20
2	Wall at Column Line R7	From RA to RG	From -11500 to 4650	2000	+25/-20
3	Wall at Column Line RA	From R1 to R7	From -11500 to 4650	2000	+25/-20
4	Wall at Column Line RG	From R1 to R7	From -11500 to 4650	2000	+25/-20
5	Wall between Column Lines R1 and R2	From between RA and RB to RC	From -11500 to -7400	1000	+25/-20
6	Wall between Column Lines R1 and R2	From RE to between RF and RG	From -11500 to -7400	1000	+25/-20
7	Wall between Column Lines RA and RB	From between R1 and R2 to R3	From -11500 to -7400	1000	+25/-20
8	Wall between Column Lines RF and RG	From between R1 and R2 to R3	From -11500 to -7400	1000	+25/-20
9	Cylinder below RCCV	Not Applicable	From -11500 to 4650	2000	+60/-0
10	Cylinder between RPV Pedestal and Cylinder below RCCV	Northeast Quadrant	From -11500 to -1700	Varies, Minimum 1400	N/A
11	Cylinder between RPV Pedestal and Cylinder below RCCV	Northwest Quadrant	From -11500 to -1700	Varies, Minimum 1400	N/A
12	Cylinder between RPV Pedestal and Cylinder below RCCV	Southwest Quadrant	From -11500 to -1700	Varies, Minimum 600	N/A
13	Cylinder between RPV Pedestal and Cylinder below RCCV	Southeast Quadrant	From -11500 to -6400	Varies, Minimum 600	N/A
14	Cylinder between RPV Pedestal and Cylinder below RCCV	Southeast Quadrant	From -6400 to -1900	Varies, Minimum 1350	N/A

Table 2.16.5-1

Critical Dimensions of Reactor Building – Part 1

Label	Wall or Section Description	Column Line or Region	Floor Elevation or Elevation Range (EL: mm)	Concrete Thickness (mm)	Tolerance (mm)
15	Wall at Column Line R1	From RA to RG	From 4650 to 17500	1500	+25/-20
16	Wall at Column Line R7	From RA to RG	From 4650 to 17500	1500	+25/-20
17	Wall at Column Line RA	From R1 to R7	From 4650 to 27000	1500	+25/-20
18	Wall at Column Line RG	From R1 to R7	From 4650 to 27000	1500	+25/-20
19	Wall at Column Line R1	From RA to RC	From 17500 to 27000	1500	+25/-20
20	Wall at Column Line R1	From RE to RG	From 17500 to 27000	1500	+25/-20
21	Wall at Column Line R7	From RA to RC	From 17500 to 27000	1500	+25/-20
22	Wall at Column Line R7	From between RD and RE to RG	From 17500 to 27000	1500	+25/-20
23	Main Steam Tunnel Wall	East side	From 17500 to 24600	1300	+25/-20
24	Main Steam Tunnel Wall	West side	From 17500 to 24600	1300	+25/-20
25	Wall at Column Line R1	From RA to RB	From 27000 to 34000	1000	+25/-20
26	Wall at Column Line R1	From RB to RC	From 27000 to 34000	1500	+25/-20
27	Wall at Column Line R1	From RC to RE	From 27000 to 34000	3500	+25/-20
28	Wall at Column Line R1	From RE to RF	From 27000 to 34000	1500	+25/-20
29	Wall at Column Line R1	From RF to RG	From 27000 to 34000	1000	+25/-20
30	Wall at Column Line R2	From RA to RC	From 27000 to 33000	1000	+25/-20
31	Wall at Column Line R2	From RE to RG	From 27000 to 33000	1000	+25/-20
32	Wall at Column Line R6	From RA to RC	From 27000 to 33000	1000	+25/-20
33	Wall at Column Line R6	From RE to RG	From 27000 to 33000	1000	+25/-20
34	Wall at Column Line R7	From RA to RB	From 27000 to 34000	1000	+25/-20
35	Wall at Column Line R7	From RB to between RC and RD	From 27000 to 34000	1500	+25/-20
36	Wall at Column Line R7	From between RC and RD to RE	From 27000 to 34000	2440	+25/-20
37	Wall at Column Line R7	From RE to RF	From 27000 to 34000	1500	+25/-20

Table 2.16.5-1

Critical Dimensions of Reactor Building – Part 1

Label	Wall or Section Description	Column Line or Region	Floor Elevation or Elevation Range (EL: mm)	Concrete Thickness (mm)	Tolerance (mm)
38	Wall at Column Line R7	From RF to RG	From 27000 to 34000	1000	+25/-20
39	Wall at Column Line RA	From R1 to R7	From 27000 to 34000	1000	+25/-20
40	Wall at Column Line RB	From R1 to R7	From 27000 to 34000	2000	+25/-20
41	Wall between Column Lines RB and RC	From R6 to R7	From 27000 to 33000	1000	+25/-20
42	Wall at Column Line RC (Pool Girder)	From R1 to R7	From 27000 to 34000	1600	+25/-20
43	Wall at Column Line RE (Pool Girder)	From R1 to R7	From 27000 to 34000	1600	+25/-20
44	Wall between Column Lines RE and RF	From R6 to R7	From 27000 to 33000	1000	+25/-20
45	Wall at Column Line RF	From R1 to R7	From 27000 to 34000	2000	+25/-20
46	Wall at Column Line RG	From R1 to R7	From 27000 to 34000	1000	+25/-20
47	Reactor Cavity Wall (Northeast side)	From RC to between RC and RD	From 27000 to 34000	1300	+25/-20
48	Reactor Cavity Wall (Northwest side)	From between RD and RE to RE	From 27000 to 34000	1300	+25/-20
49	Reactor Cavity Wall (Southeast side)	From RC to RD	From 27000 to 34000	1300	+25/-20
50	Reactor Cavity Wall (Southwest side)	From RD to RE	From 27000 to 34000	1300	+25/-20
51	IC/PCCS Pool Wall between Column Lines R2 and R3	From between RB and RC to RC	From 27000 to 33000	1000	+25/-20
52	IC/PCCS Pool Wall between Column Lines R2 and R3	From RE to between RE and RF	From 27000 to 33000	1000	+25/-20
53	IC/PCCS Pool Wall at Column Line R3	From between RB and RC to RC	From 27000 to 33000	400	+15/-10
54	IC/PCCS Pool Wall at Column Line R3	From RE to between RE and RF	From 27000 to 33000	400	+15/-10
55	IC/PCCS Pool Wall between Column Lines R3 and R4	From between RB and RC to RC	From 27000 to 33000	400	+15/-10
56	IC/PCCS Pool Wall between Column Lines R3 and R4	From RE to between RE and RF	From 27000 to 33000	400	+15/-10
57	IC/PCCS Pool Wall between Column Lines R4 and R5	From between RB and RC to RC	From 27000 to 33000	400	+15/-10
58	IC/PCCS Pool Wall between Column Lines R4 and R5	From RE to between RE and RF	From 27000 to 33000	400	+15/-10
59	IC/PCCS Pool Wall at Column Line R5	From between RB and RC to RC	From 27000 to 33000	400	+15/-10
60	IC/PCCS Pool Wall at Column Line R5	From RE to between RE and RF	From 27000 to 33000	400	+15/-10

Table 2.16.5-1

Critical Dimensions of Reactor Building – Part 1

Label	Wall or Section Description	Column Line or Region	Floor Elevation or Elevation Range (EL: mm)	Concrete Thickness (mm)	Tolerance (mm)
61	IC/PCCS Pool Wall between Column Lines R5 and R6	From between RB and RC to RC	From 27000 to 33000	470	+15/-10
62	IC/PCCS Pool Wall between Column Lines R5 and R6	From RE to between RE and RF	From 27000 to 33000	470	+15/-10
63	IC/PCCS Pool Wall between Column Lines RB and RC	From between R2 and R3 to between R5 and R6	From 27000 to 33000	600	+15/-10
64	IC/PCCS Pool Wall at Column Line RC	From R2 to between R2 and R3	From 27000 to 33000	1000	+25/-20
65	IC/PCCS Pool Wall at Column Line RE	From R2 to between R2 and R3	From 27000 to 33000	1000	+25/-20
66	IC/PCCS Pool Wall between Column Lines RE and RF	From between R2 and R3 to between R5 and R6	From 27000 to 33000	600	+15/-10
67	Wall at Column Line R1	From RB to RF	From 34000 to 52400	1000	+25/-20
68	Wall at Column Line R7	From RB to RF	From 34000 to 52400	1000	+25/-20
69	Wall at Column Line RB	From R1 to R7	From 34000 to 52400	1000	+25/-20
70	Wall at Column Line RF	From R1 to R7	From 34000 to 52400	1000	+25/-20
71	Basemat	From R1 to R7 and RA and RG	-11500	4000	+50/-20
72	Floor inside Cylinder below RCCV	Northeast Quadrant	-6400	600	+15/-10
73	Floor inside Cylinder below RCCV	Northwest Quadrant	-6400	600	+15/-10
74	Floor inside Cylinder below RCCV	Southeast Quadrant	-6400	1000	+25/-20
75	Floor inside Cylinder below RCCV	Southwest Quadrant	-6400	600	+15/-10
76	Floor outside Cylinder below RCCV	From R1 to R7 and RA and RG	-6400	1000	+25/-20
77	Floor inside Cylinder below RCCV	Northeast Quadrant	-1000	700	+15/-10
78	Floor inside Cylinder below RCCV	Northwest Quadrant	-1000	700	+15/-10
79	Floor inside Cylinder below RCCV	Southeast Quadrant	-1000	900	+15/-10
80	Floor inside Cylinder below RCCV	Southwest Quadrant	-1000	700	+15/-10
81	Floor outside Cylinder below RCCV	From R1 to R7 and RA to RG	-1000	1000	+25/-20

Table 2.16.5-1

Critical Dimensions of Reactor Building – Part 1

Label	Wall or Section Description	Column Line or Region	Floor Elevation or Elevation Range (EL: mm)	Concrete Thickness (mm)	Tolerance (mm)
82	Floor	From R1 to R7 and RA to RG	4650	1000	+25/-20
83	Floor	From R1 to R7 and RA to RG	9060	1000	+25/-20
84	Floor	From R1 to R7 and RA to RG	13570	1000	+25/-20
85	Main Steam Tunnel Floor	From RC to RE	17500	1600	+25/-20
86	Floor excluding Main Steam Tunnel Floor	From R1 to R7 and RA to RG	17500	1000	+25/-20
87	Main Steam Tunnel Roof	From RC to RE	27000	2400	+25/-20
88	Floor	From R1 to R7 and RA to RC	27000	1000	+25/-20
89	Floor	From R1 to R7 and RE to RG	27000	1000	+25/-20
90	Floor	From R1 to R7 and RA to RC	34000	1000	+25/-20
91	Floor	From R1 to R7 and RE to RG	34000	1000	+25/-20
92	Roof	From R1 to R7 and RB to RF	52700	700	+15/-10

Table 2.16.5-1

Critical Dimensions of Reactor Building – Part 2

Key Dimension	Reference Dimension	Nominal Dimension (mm)	Tolerance (mm)
Distance from RPV Centreline to Outside Surface of Wall at Column Line RA when Measured at Column Line R1	X1 (Figure 2.16.5-1)	24500	+/-300
Distance from RPV Centreline to Outside Surface of Wall at Column Line RG when Measured at Column Line R1	X2 (Figure 2.16.5-1)	24500	+/-300
Distance from RPV Centreline to Outside Surface of Wall at Column Line R1 when Measured at Column Line RA	X3 (Figure 2.16.5-1)	24500	+/-300
Distance from RPV Centreline to Outside Surface of Wall at Column Line R7 when Measured at Column Line RA	X4 (Figure 2.16.5-1)	24500	+/-300
Distance from Top of Basemat Outside Containment to Design Plant Grade	X5 (Figure 2.16.5-10)	16150	+/-300
Distance of Design Plant Grade Relative to Finished Ground Elevation	X6 (Figure 2.16.5-10)	150	+/-20
Distance from Design Plant Grade to Top Surface of Roof	X7 (Figure 2.16.5-10)	48050	+/-300

Table 2.16.5-2

ITAAC For The Reactor Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. 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 including LOCA and missiles. • Normal plant operation—live loads, dead loads, temperature effects and building vibration loads.	Analyses of the as-built RB will be conducted.	Report(s) document that the as-built RB conforms to the structural design basis loads specified in the Design Description of this subsection 2.16.5 associated with: • Natural phenomena—wind, floods, tornados (including tornado missiles), earthquakes, rain and snow. • Internal events—floods, pipe breaks including LOCA and missiles. • Normal plant operation—live loads, dead loads, temperature effects and building vibration loads.
2. The physical arrangement of the RB is as described in the Design Description of this subection 2.16.5 and Figures 2.16.5.1 through 2.16.5.11.	Inspections of the as-built RB will be conducted.	Report(s) document that the RB conforms to the physical arrangement described in the Design Description of this subsection 2.16.5.

Table 2.16.5-2

ITAAC For The Reactor Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The critical dimensions and acceptable tolerances for the RB are as described in Table 2.16.5-1.	Inspection of the as-built RB will be performed. Deviations from the design and as-built conditions will be analyzed for the design basis loads.	Report(s) exist which reconcile construction deviations from the critical dimensions and tolerances specified in Table 2.16.5-1 and conclude that the as-built RB will withstand the design basis loads specified in the Design Description of this subsection 2.16.5 without loss of structural integrity or the safety-related functions.
4. Assuming a LOCA, the offsite dose limits and the control room dose limits are met based on a 50 wt% per day leakage rate from the RB.	Leakage rate testing of the as-built RB under conditions expected to exist during a LOCA, will be conducted.	Test/analysis report(s) document that the RB leakage rate under the conditions expected to exist during a LOCA is ≤ 50 wt% per day.
5. The RB provides three-hour fire barriers for separation of the four independent safe shutdown divisions.	Inspections of the as-built RB will be conducted.	Inspection report(s) document that each division is separated by fire barriers having ≥ 3-hour fire ratings.

Table 2.16.5-2

ITAAC For The Reactor Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. For 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: a. Wall thicknesses below flood level designed to withstand hydrostatic loads; b. Water stops provided in all expansion and construction joints below flood and groundwater levels; c. Waterproofing of below flood and groundwater levels external surfaces; d. Water seals in external walls at pipe penetrations below flood and groundwater levels; and e. Roofs designed to prevent pooling of large amounts of water in excess of the structural capacity of the roof for design loads.	Inspection of the as-built flood control features will be conducted.	Inspection report(s) document that the as-built RB conforms with the following flood protection features specified in the Design Description of this subsection 2.16.5. a. Wall thicknesses below flood level are designed to withstand hydrostatic loads; b. Water stops are provided in all expansion and construction joints below flood and groundwater levels; c. Waterproofing exists below flood and groundwater levels external surfaces; d. Water seals exist in external walls at pipe penetrations below flood and groundwater levels; and e. Roofs are built to prevent pooling of large amounts of water in excess of the structural capacity of the roof for design loads.

Table 2.16.5-2

ITAAC For The Reactor Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7. Protective features used to mitigate or eliminate the consequences of internal flooding are: a. Structural enclosures or barriers; b. Curbs and sills; c. Leakage detection components; and d. Drainage systems	Inspections of the as-built RB flood protection features will be conducted.	Reports document that the following flood protection features specified in the Design Description 2.16.5 are in place in the as-built RB to mitigate or eliminate the consequences of internal flooding: a. Structural enclosures or barriers; b. Curbs and sills; c. Leakage detection components; and d. Drainage systems
8. The internal flooding protection features prevent flood water in one division from propagating to other division(s) and ensure equipment necessary for safe shutdown is located above the maximum flood level for that location or is qualified for flood conditions by: a. Divisional walls b. Sills c. Watertight doors.	Inspections of the as built RB flood protection features will be conducted.	Reports document that the following flood protection features specified in the Design Description 2.16.5 are in place in the as-built RB to prevent flood water in one division from propagating to other division(s) and to ensure equipment necessary for safe shutdown not located above the maximum flood level for that location is qualified for flood conditions: a. Divisional walls b. Sills c. Watertight doors.

Table 2.16.5-2

ITAAC For The Reactor Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. The RB is protected against pressurization effects associated with postulated rupture of pipes containing high-energy fluid that occur in subcompartments of the RB.	Inspections of the RB subcompartments that rely on overpressure protection devices will be conducted.	Reports document that as-built RB subcompartments which rely on overpressure protection devices are equipped with devices specified in the Design Description 2.16.5.

Figure 2.16.5-1. RB Concrete Outline Plan at EL -11500
{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.5-2. RB Concrete Outline Plan at EL -6400

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.5-3. RB Concrete Outline Plan at EL -1000

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.5-4. RB Concrete Outline Plan at EL 4650
{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.5-5. RB Concrete Outline Plan at EL 9060

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.5-6. RB Concrete Outline Plan at EL 13570

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.5-7. RB Concrete Outline Plan at EL 17500

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.5-8. RB Concrete Outline Plan at EL 27000

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.5-9. RB Concrete Outline Plan at EL 34000

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.5-10. RB Concrete Outline N-S Section

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.5-11. RB Concrete Outline E-W Section

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

2.16.6 Control Building

Design Description

The Control Building (CB) houses the essential electrical, control and instrumentation equipment, the Main Control Room(MCR), and the CB HVAC equipment. The CB is a reinforced concrete box type shear wall structure consisting of walls and slabs and is supported on a foundation mat. The CB structure is a Seismic Category I structure.

The key characteristics of the CB are as follows:

(1) The CB is designed and constructed to accommodate the dynamic, static, and thermal 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.

(2) The CB is as shown in Figures 2.16.6-1 through 2.16.6-5.

(3) The critical CB dimensions used for seismic analyses and the acceptable tolerances are provided in Table 2.16.6-1.

(4) The MCR envelope is separated from the rest of the CB by walls, floors, doors and penetrations, which have three-hour fire ratings.

(5) The lowest elevation in the CB is divided into separate divisional areas for instrumentation and control equipment. CB flooding resulting from component failures in any of the CB divisions does not prevent safe shutdown of the reactor.

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.

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 – Part 1

Label	Wall or Section Description	Column Line or Region	Floor Elevation or Elevation Range (EL: mm)	Concrete Thickness (mm)	Tolerance (mm)
1	Wall at Column Line C1	From CA to CD	From -7400 to 9060	900	+15/-10
2	Wall at Column Line C5	From CA to CD	From -7400 to 9060	900	+15/-10
3	Wall at Column Line CA	From C1 to C5	From -7400 to 9060	900	+15/-10
4	Wall at Column Line CD	From C1 to C5	From -7400 to 9060	900	+15/-10
5	Wall at Column Line C3	From CA to CB	From -7400 to -2500	1000	+25/-20
6	Wall at Column Line C3	From CC to CD	From -7400 to -2500	1000	+25/-20
7	Wall at Column Line C1	From CA to CD	From 9060 to 13800	700	+15/-10
8	Wall at Column Line C5	From CA to CD	From 9060 to 13800	700	+15/-10
9	Wall at Column Line CA	From C1 to C5	From 9060 to 13800	700	+15/-10
10	Wall at Column Line CD	From C1 to C5	From 9060 to 13800	700	+15/-10
11	Basemat	From C1 to C5 and CA to CD	-7400	3000	+50/-20
12	Floor	From C1 to C5 and CA to CD	-2000	500	+15/-10
13	Floor	From C1 to C5 and CA to CD	4650	500	+15/-10
14	Floor	From C1 to C5 and CA to CD	9060	500	+15/-10
15	Roof	From C1 to C5 and CA to CD	13800	700	+15/-10

Table 2.16.6-1

Critical Dimensions of Control Building – Part 2

Key Dimension	Reference Dimension	Nominal Dimension (mm)	Tolerance (mm)
Distance from Outside Surface of Wall at Column Line CA to Column Line CB when Measured at Column Line C1	X1 (Figure 2.16.6-1)	10400	+/-300
Distance from Outside Surface of Wall at Column Line CD to Column Line CB when Measured at Column Line C1	X2 (Figure 2.16.6-1)	13400	+/-300
Distance from Outside Surface of Wall at Column Line C1 to Column Line C3 when Measured at Column Line CA	X3 (Figure 2.16.6-1)	15150	+/-300
Distance from Outside Surface of Wall at Column Line C5 to Column Line C3 when Measured at Column Line CA	X4 (Figure 2.16.6-1)	15150	+/-300
Distance from Top of Basemat to Design Plant Grade	X5 (Figure 2.16.6-5)	12050	+/-300
Distance of Design Plant Grade Relative to Finished Ground Elevation	X6 (Figure 2.16.6-5)	150	+/-20
Distance from Design Plant Grade to Top Surface of Roof	X7 (Figure 2.16.6-5)	9150	+/-300

Table 2.16.6-2

ITAAC For The Control Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The CB is designed and constructed to accommodate the dynamic, static, and thermal 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.	Analyses of the as-built CB loads will be conducted.	Report(s) document that the as-built CB conforms to the structural design basis loads specified in the Design Description of this subsection 2.16.6 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
2. The physical arrangement of the CB is described in the Design Description of this Subsection 2.16.6 and Figures 2.16-6-1 through 2.16.6-5	Inspections of the as-built CB will be conducted.	Report(s) document that the CB conforms to the physical arrangement described in the Design Description of this subsection 2.16.6.

Table 2.16.6-2

ITAAC For The Control Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The critical CB dimensions and acceptable tolerance are provided in Table 2.16.6-1.	Inspection of the as-built CB will be performed. Deviations from the design and as-built conditions will be analyzed for the design basis loads.	Report(s) exist which reconcile construction deviations from the critical dimensions and tolerances specified in Table 2.16.6-1 and conclude that the as-built CB will withstand the design basis loads specified in the Design Description of this subsection 2.16.6 without loss of structural integrity or the safety-related functions.
4. The MCR envelope is separated from the rest of the CB by walls, floors, doors and penetrations, which have a three-hour fire rating.	Inspections of the as-built CB will be conducted.	Inspection report(s) document that the as-built CB has a MCR envelope separated from the rest of the CB by walls, floors, doors and penetrations with $\geq$3-hour fire rating.

Table 2.16.6-2

ITAAC For The Control Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The lowest elevation in the CB is divided into separate divisional areas for instrumentation and control equipment. CB flooding resulting from component failures in any of the CB divisions does not prevent safe shutdown of the reactor. 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. maximum flood level for that location or is qualified for flood conditions.	Inspections of the as-built CB flood control features will be conducted.	Inspection report(s) document that the as-built CB contains the following 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.

Table 2.16.6-2

ITAAC For The Control Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. (continued) 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.		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.

Figure 2.16.6-1. CB Concrete Outline Plan at EL -7400

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.6-2. CB Concrete Outline Plan at EL -2000
{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.6-3. CB Concrete Outline Plan at EL 4650

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.6-4. CB Concrete Outline Plan at EL 9060

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.6-5. CB Concrete Outline E-W Section

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

2.16.7 Fuel Building

Design Description

The Fuel Building (FB) contains the spent fuel pool, cask loading area, fuel handling systems and storage areas, lower connection to the inclined fuel transfer system, overhead crane, and other plant systems and equipment. The FB is a Seismic Category I structure except for the penthouse that houses HVAC equipment. The penthouse is a Seismic Category II structure. The FB is a rectangular reinforced concrete box type shear wall structure consisting of walls and slabs and is supported on a foundation mat. The FB is integrated with the RB, sharing a common wall between the RB and FB as well as a large common foundation mat. The building is partially embedded.

There is no safety-related component in the FB that could be affected by internal 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 key characteristics of the FB are as follows:

(1) The FB is designed and constructed to accommodate the dynamic, static, and thermal 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; and
- Normal plant operation—live loads, dead loads and temperature effects.

(2) The FB physical arrangement is shown in Figures 2.16.7-1 through 2.16.7-6

(3) The critical dimensions of the FB used for seismic analyses and the acceptable tolerance are provided in Table 2.16.7-1.

(4) The walls forming the boundaries of the FB and penetrations through these walls have three-hour fire ratings.

(5) The FB external flooding protection features are:

   (1) Exterior access openings are sealed in external walls below flood and groundwater levels;

   (2) Water seals at pipe penetrations are installed in external walls below flood and groundwater levels; and

   (3) Water stops are in expansion and construction joints below flood and groundwater levels.

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 – Part 1

Label	Wall or Section Description	Column Line or Region	Floor Elevation or Elevation Range (EL: mm)	Concrete Thickness (mm)	Tolerance (mm)
1	Wall at Column Line F3	From FA to between FB and FC	From -10000 to 4650	3640	+25/-20
2	Wall at Column Line F3	From between FB and FC to FF	From -11500 to 4650	2000	+25/-20
3	Wall at Column Line FA	From F1 to F3	From -10000 to 4650	2000	+25/-20
4	Wall at Column Line FF	From F1 to F3	From -11500 to 4650	2000	+25/-20
5	Wall between Column Lines F1 and F2	From FA to FB	From -10000 to 4650	4500	+25/-20
6	Wall between Column Lines F1 and F2	From FB to between FB and FC	From -10000 to 4650	1935	+25/-20
7	Wall between Column Lines F1 and F2	From between FB and FC to FC	From -10000 to -6400	2000	+25/-20
8	Wall between Column Lines F1 and F2 (Wall between Cask Pit and Incline Fuel Transfer Tube Pit)	From between FB and FC to FC	From -10000 to 4650	1000	+25/-20
9	Wall between Column Lines F1 and F2	From FE to FF	From -11500 to -7200	1000	+25/-20
10	Wall at Column Line F2	From between FE and FF to FF	From -11500 to -7200	1000	+25/-20
11	Wall between Column Lines F2 and F3	From between FB and FC to FC	From -10000 to -1300	1150	+25/-20
12	Wall between Column Lines F2 and F3	From FE to FF	From -11500 to -7200	1000	+25/-20
13	Wall between Column Lines FB and FC	From between F1 and F2 to F3	From -10000 to 4650	1500	+25/-20
14	Wall at Column Line FC	From F1 to between F1 and F2	From -11500 to 4650	1500	+25/-20
15	Wall at Column Line FC	From between F1 and F2 to F3	From -11500 to 4650	1000	+25/-20
16	Wall at Column Line FE	From between F1 and F2 to F2 and F3	From -11500 to -7200	1000	+25/-20
17	Wall at Column Line FE	From between F2 and F3 to F3	From -11500 to -7200	600	+15/-10
18	Wall between Column Lines FB and FC	From F1 to F1 and F2	From -6400 to 4650	2000	+25/-20
19	Wall between Column Lines F2 and F3	From between FB and FC to FC	From -1300 to 4650	1000	+25/-20
20	Wall at Column Line F3	From FA to FF	From 4650 to 22500	1000	+25/-20

Table 2.16.7-1

Critical Dimensions of Fuel Building – Part 1

Label	Wall or Section Description	Column Line or Region	Floor Elevation or Elevation Range (EL: mm)	Concrete Thickness (mm)	Tolerance (mm)
21	Wall at Column Line FA	From F1 to F3	From 4650 to 22500	1000	+25/-20
22	Wall at Column Line FF	From F1 to F3	From 4650 to 22500	1000	+25/-20
23	Basemat of Spent Fuel Pool, Cask Pit, and Incline Fuel Transfer Tube Pit	Not Applicable	-10000	5500	+50/-20
24	Basemat excluding Spent Fuel Pool, Cask Pit, and Incline Fuel Transfer Tube Pit	Not Applicable	-11500	4000	+50/-20
25	Floor	From F1 to F3 and FC to FF	-6400	800	+15/-10
26	Floor	From F1 to F3 and FC to FF	-1000	800	+15/-10
27	Floor (Cask Pit)	Not Applicable	-1300	1175	+25/-20
28	Floor	From F1 to F3 and FC to FF	4650	1300	+25/-20
29	Roof	From F1 to F3 and FA to FF	22500	700	+15/-10

Table 2.16.7-1

Critical Dimensions of Fuel Building – Part 2

Key Dimension	Reference Dimension	Nominal Dimension (mm)	Tolerance (mm)
Distance from Outside Surface of Wall at Column Line FA to Column Line FC when Measured at Column Line F1	X1 (Figure 2.16.7-1)	21700	+/-300
Distance from Outside Surface of Wall at Column Line FF to Column Line FC when Measured at Column Line F1	X2 (Figure 2.16.7-1)	27300	+/-300
Distance between Outside Surface of Walls at Column Lines R7 and F3 when Measured at Column Line FA	X3 (Figure 2.16.7-1)	21000	+/-300
Distance from Top of Basemat to Design Plant Grade (Basemat excluding Spent Fuel Pool, Cask Pit, and Incline Fuel Transfer Tube Pit)	X4 (Figure 2.16.7-6)	16150	+/-300
Distance of Design Plant Grade Relative to Finished Ground Elevation	X5 (Figure 2.16.7-6)	150	+/-20
Distance from Design Plant Grade to Top Surface of Roof (Excluding C-II Portion)	X6 (Figure 2.16.7-6)	17850	+/-300

Table 2.16.7-2
ITAAC For The Fuel Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The FB is designed and constructed to accommodate the dynamic, static, and thermal 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; and • Normal plant operation—live loads, dead loads and temperature effects.	Analyses of the as-built FB will be conducted.	Report(s) document that the as-built FB conforms to the structural design basis loads specified in the Design Description of this subsection 2.16.7 associated with: • Natural phenomena—wind, floods, tornadoes (including tornado missiles), earthquakes, rain and snow; • Internal events—floods and missiles; and • Normal plant operation—live loads, dead loads and temperature effects.
2. The physical arrangement of the FB is as described in the Design Description of this subsection 2.16.7 and Figures 2.16.7-1 through 2.16.7-6.	Inspections of the as-built FB will be conducted.	Report(s) document that the FB conforms to the physical arrangement described in the Design Description of this subsection 2.16.7.
3. The critical dimensions and acceptable tolerances for the FB are as described in Table 2.16.7-1.	Inspection of the as-built FB will be performed. Deviations from the design and as-built conditions will be analyzed for the design basis loads.	Report(s) exist which reconcile construction deviations from the critical dimensions and tolerances specified in Table 2.16.7-1 and conclude that the as-

Table 2.16.7-2

ITAAC For The Fuel Building

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		built FB will withstand the design basis loads specified in the Design Description of this subsection 2.16.7 without loss of structural integrity or the safety-related functions.
4. The walls forming the boundaries of the FB and penetrations through these walls have three-hour fire ratings.	Inspections of the as-built FB walls and penetrations will be conducted.	Inspection report(s) document that the as-built walls forming the boundaries of the FB and penetrations through these walls have ≥ 3-hour fire ratings.
5. The FB is protected against an external flooding. Protection features are: (1) Exterior access openings are sealed in external walls below flood and groundwater levels. (2) Water seals at pipe penetrations are installed in external walls below flood and groundwater levels. (3) Water stops are in expansion and construction joints below flood and groundwater levels.	Inspection of the as-built FB flood control features will be conducted.	Inspection report(s) document that the following as-built FB flood protection features exist: Protection features are: (1) Exterior access openings are sealed in external walls below flood and groundwater levels. (2) Water seals at pipe penetrations are installed in external walls below flood and groundwater levels. (3) Water stops are provided in expansion and construction joints below flood and groundwater levels.

Figure 2.16.7-1. FB Concrete Outline Plan at EL -11500

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.7-2. FB Concrete Outline Plan at EL -6400

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.7-3. FB Concrete Outline Plan at EL -1000

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.7-4. FB Concrete Outline Plan at EL 4650

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.7-5. FB Concrete Outline Plan at EL 22500
{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

Figure 2.16.7-6. FB Concrete Outline N-S Section

{{{Security-Related Information – Withheld Under 10 CFR 2.390}}}

2.16.8 Turbine Building

Design Description

No entry for the Turbine Building.

2.16.9 Radwaste Building

Design Description

No entry for this building.

2.16.10 Other Buildings and Structures

Design Description

No entry for this building.

2.17 INTAKE STRUCTURE AND SERVICING EQUIPMENT

2.17.1 Intake and Discharge Structure

No entry for this system.

2.18 YARD STRUCTURES AND EQUIPMENT

2.18.1 Oil Storage and Transfer Systems

No entry for this system.

2.18.2 Site Security

No entry for this system.

2.19 PLANT SECURITY SYSTEM

Design Description

The physical security system provides physical features to detect, delay, assist response to, and defend against the design basis threat (DBT) for radiological sabotage. The physical security system consists of physical barriers and an intrusion detection system. The details of the physical security system are categorized as Safeguards Information. The physical security system provides protection for vital equipment and plant personnel.

(1) The walls, doors, ceiling, and floors in the main control room and central alarm station are bullet-resistant to a UL level 4 round.

(2) Central Alarm Station is a vital area.

(3) Secondary Security power supply system for alarm annunciator equipment and non-portable communications equipment is located within a vital area.

(4) Unoccupied vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion.

(5) The locks used for the protection of the vital areas are manipulative-resistant.

(6) The Vehicle Barrier System is installed and located at the necessary stand-off distance to protect against the DBT vehicle bomb.

(7) Vital equipment is located within a defined vital area, which is protected by at least two physical barriers.

(8) Isolation zones and all exterior areas within the Protected Area are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.

(9) Emergency exits in the protected area perimeter and the vital area are alarmed.

(10) An intrusion detection system is installed to detect penetration or attempted penetration of the protected area barrier and the vital area portals.

(11) An access control system is installed to identify and authorize personnel entering the protected area.

(12) One or more access control points are established to monitor all vehicle and personnel access into the protected area.

(13) The central and secondary alarm stations have communication capabilities with local law enforcement authorities.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.19-1 provides a definition of the inspections, tests and/or analysis, together with associated acceptance criteria for physical security system.

Table 2.19-1

ITAAC For Plant Security System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The walls, doors, ceiling, and floors in the main control room and the central alarm station are bullet resistant to a UL level 4 round.	Type test, analysis, or a combination of type test and analysis will be performed for the walls, doors, ceilings, and floors in the main control room and the central alarm station.	A report exists and concludes that the walls, doors, ceilings, floors in the main control room and the central alarm station are bullet resistant to a UL level 4 round.
2. Central Alarm Station is a vital area.	An inspection of the central alarm station will be performed.	A report exists and concludes that access to the Central Alarm Station is though an activated intrusion detection system and at least two security hardened barriers.
3. Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is located within the vital area.	An inspection of the secondary security power supply system for alarm annunciator equipment and non-portable communications equipment will be performed.	A report exists and concludes that access to the Secondary security power supply system for alarm annunciator equipment and non-portable communications equipment is through an activated intrusion detection system and at least two security hardened barriers.
4. Unoccupied vital areas are locked and alarmed with active intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion.	An inspection of vital areas will be performed.	A report exists and concludes that the unoccupied vital areas are locked and alarmed with intrusion detection systems that annunciate in the central and secondary alarm stations upon intrusion.
5. The locks used for the protection of the vital areas are manipulative-resistant.	Type test, analysis, or a combination of type test and analysis will be performed for locks used in the protection of the vital areas.	A report exists and concludes that the locks used for protection of the vital areas are manipulative-resistant.

Table 2.19-1

ITAAC For Plant Security System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The Vehicle Barrier System is installed and located at the necessary stand-off distance to protect against the DBT vehicle bombs.	Type test, analysis, or a combination of type test and analysis will be performed for the vehicle barrier system used to protect against the DBT vehicle bombs.	A report exists and concludes that the vehicle barrier system will protect against the DBT vehicle bombs based upon the stand-off distance of the system.
7. Vital equipment is located within a defined vital area, which is protected by at least two physical barriers.		A report exists and concludes that:
	i. Inspection of the equipment on the vital equipment list confirms that they are within a vital area boundary.	i. All vital equipment is located within a vital area.
	ii. Inspection is performed to confirm that there are two physical barriers between the inside of a vital area and the outside of the protected area boundary.	ii. There are two physical barriers installed between the inside of a vital area and the outside of the protected area boundary.
8. Isolation zones and all exterior areas within the Protected Area are provided with illumination to permit observation of abnormal presence or activity of persons or vehicles.	Inspection of the exterior area within the protected area confirms illumination to permit observation and detection per design.	A report exists and concludes that the illumination in the exterior portion of the protected area is sufficient to permit observation and detection per design.
9. Emergency exits in the protected area perimeter and the vital area are alarmed.	A test is performed to verify that the emergency exits from the protected area perimeter and the vital area are alarmed.	A report exists and concludes that the emergency exits from the protected area perimeter and the vital area are alarmed.

Table 2.19-1

ITAAC For Plant Security System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. An intrusion detection system is installed to detect penetration or attempted penetration of the protected area barrier and the vital area portals.	Tests are performed for the as-built intrusion detection system used to detect penetration or attempted penetration of the protected area barrier and the vital area portals.	A report exists and concludes that the intrusion detection system annunciates in the Central and Secondary Alarm Stations upon penetration or attempted penetration into the protected area barrier or the vital area portals.
11. An access control system is installed to identify and authorize personnel entering the protected area.	A test of the access control system is performed.	A report exists and concludes that the access control system can identify and authorize personnel entering the protected area.
12. One or more access control points are established to monitor all vehicle and personnel access into the protected area.	An inspection is performed to verify access control point(s) to the protected area exits.	A report exists and concludes that one or more access control points to the protected area are established.
13. The central and secondary alarm stations have communication capabilities with local law enforcement authorities.	Inspection of the central and secondary alarm stations confirms that each is equipped with the capability to communicate with local law enforcement authorities.	A report exists and concludes that the central and secondary alarm stations have communication capabilities with local law enforcement authorities.

3. NON-SYSTEM BASED MATERIAL

3.1 DESIGN OF PIPING SYSTEMS AND COMPONENTS

Design Description

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

(1) Safety-related piping systems are designed to ASME Code Section III requirements.

(2) Safety-related piping systems are designed to Seismic Category I requirements.

(3) 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.

(4) ASME Code Class 1 piping systems are designed to remain within allowable stress limits and fatigue limits to prevent fatigue failure.ASME Code Class 2 and 3 piping systems are designed to remain within allowable stress limits, including those piping systems which may be subjected to thermal transients.On an individual system basis, the as-built piping shall be reconciled with the piping design requirements in Section 3.1.

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. Similar ITAAC are included in the system-based ITAAC. As appropriate, each of the ITAAC in Section 3.1 may be closed on a system-by-system basis throughout construction, in order that systems may be placed in service.

Table 3.1-1

ITAAC For The Generic Piping Design

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Safety-related piping systems are designed to ASME Code Section III requirements.	Inspections of ASME Code required documents will be conducted.	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. Safety-related piping systems are designed to Seismic Category I requirements.	i) Inspection will be performed to verify that the seismic Category I piping systems are located on a seismic structure. ii) Type tests, analyses, or a combination of type tests and analyses of as-built seismic Category I piping will be performed. iii) Inspection will be performed for the existence of a report verifying that the as-installed piping including anchorage is seismically bounded by the tested and/or analyzed conditions.	Report(s) document that: i) The seismic Category I piping systems are located on a seismic structure. ii) A report exists and concludes that the as-built seismic Category I piping can withstand seismic design basis loads without loss of safety function. iii) A report exists and concludes that the as-installed piping including anchorage is seismically bounded by the tested and/or analyzed conditions.

Table 3.1-1

ITAAC For The Generic Piping Design

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. 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.	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. i) Pipe break events involving high-energy fluid systems are analyzed for the effects of pipe whip, jet impingement, flooding, room pressurization, and temperature effects. Pipe break events involving moderate-energy fluid systems are analyzed for wetting from spray, flooding, and other environmental effects, as appropriate. ii) Based on the analyses results, inspection of the as-built piping systems and equipment will be performed to identify features that protect against dynamic effects of pipe failures, such as whip restraints, equipment shields, drainage systems, and physical separation of piping, equipment, and instrumentation.	A report documents that the as-built pipe analysis concludes that for each postulated piping failure, the reactor can be shut down safely. i) Reports document the results of the analyses to determine where protection features are necessary to mitigate the consequences of a pipe break. ii) Reports document that the protective features are installed in the as-built plant.

Table 3.1-1

ITAAC For The Generic Piping Design

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. ASME Code Class 1 piping systems are designed to remain within allowable stress limits and fatigue limits to prevent fatigue failure.	Fatigue analyses shall be performed in accordance with the ASME Code Class 1 piping requirements. Environmental effects shall be included in the fatigue analysis.	Reports document that the Class 1 piping stress and fatigue analyses demonstrate that the ASME Code Class 1 piping analyses requirements are met, considering environmental effects in the analyses.
5. ASME Code Class 2 and 3 piping systems are designed to remain within allowable stress limits, including those piping systems which may be subjected to thermal transients.	Analyses shall be performed on the as-built piping systems to calculate piping stress ranges due to thermal expansion in accordance with the ASME Code Class 2 and 3 piping requirements. For the ASME Code Class 2 and 3 piping systems and their associated components that may be subjected to thermal transients, the effects of these transients shall be included in the design.	Reports document that the piping stress analyses demonstrate that the ASME Code Class 2 and 3 piping analyses requirements, including thermal expansion stress limits, are met, and verify that, where appropriate, thermal transients are considered in the analyses.
6. On an individual system basis, the as-built piping shall be reconciled with the piping design requirements in Section 3.1.	A reconciliation analysis using the as-designed and as-built information will be performed.	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

Inspections, Tests, Analyses, and Acceptance Criteria Summary

Design Description

NUREG-0800, Branch Technical Position HICB-14 (BTP-14) Revision 4, outlines activities to be considered when establishing a software development program for software-based Instrumentation and Control (I&C) 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 the software plans address and document the elements necessary to ensure the production and delivery of High Quality Software.

The ESBWR Software Management and Software Quality Assurance Plans, based in part on Section 2.1 of BTP-14, have been developed and submitted to the NRC for review in support of DCD Certification. During development of the ESBWR Software Lifecycle process, Regulatory Guidelines (RG) 1.173–1997, "Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants" and RG 1.152–2006, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants", were referred to extensively to ensure compliance with BTP 7-14.

GEH has completed a detailed analysis of the guidelines and standards and incorporated information from that study into the ESBWR Software Plans. In certain cases, deviation is taken from the guidelines and standards, in which case the GEH software plans will be followed. Compliance with this process will provide a sound base for development of High Quality Software.

The ESBWR Cyber Security Program Plan is discussed in Item 11 of the ITAAC. While it is not required to be discussed by BTP 7-14 Revision 4, it is appropriate to include it in the Software ITAAC.

Software Plans and Programs

The software plans are identified in the ESBWR Man-Machine Interface (MMI) System and Human Factor & Engineering (HFE) Implementation Plan (MMIS/HFE IP). The software plans included in the GEH ESBWR Software Management Plan document, referred to as the Software Management Plan (SMP), are:

(1) Software Development Plan (SDP)

(2) Software Integration Plan (SIntP)

(3) Software Installation Plan (SIP)

(4) Software Operation and Maintenance Plan (SOMP)

(5) Software Training Plan (STrngP)

The ESBWR I&C Software Quality Assurance Plan (SQAP), herein referred to as SQAP, includes the software plans used by the Quality Assurance (QA) and the Software Project Engineering (SPE) organizations, governing the same I&C software scope identified in the MMIS/HFE IP.

(1) Software Verification & Validation Plan (SVVP)

(2) Software Safety Plan (SSP)

(3) Software Configuration Management Plan (SCMP)

The ESBWR Cyber Security Program Plan is further defined by a separate Licensing Topical Report, ESBWR Cyber Security Program Plan (CSPP).

The applicable Software Products (software and firmware) covered in the SMP encompass all I&C systems, as specifically defined in the MMIS/HFE IP, which perform the monitoring, control, alarming, and protection functions associated with all modes of ESBWR plant normal operation (i.e., startup, shutdown, standby, power operation, and refueling) as well as off-normal, emergency, and accident conditions.

Software Management Plan

The purpose of the Software Management Plan (SMP) is to establish the managerial processes and the technical direction for the design and development activities of Digital Computer-Based I&C Software within the scope of the ESBWR.

The Software Management Plan (SMP) includes the key planning documents for the Instrumentation and Controls (I&C) design team and governs the design and development activities for the Digital Computer-Based I&C software for the ESBWR.

As outlined within the SMP, an organization has been established to address the control of software management and to ensure that independence is maintained between the design organization and the quality assurance, software safety, and Verification and Validation (V&V) organizations.

Software Development Plan

The Software Development Plan (SDP) describes the software engineering development process for each phase of the software products life cycle process. The phases include Planning, Requirements, Design, Implementation, Test, Installation, Operations & Maintenance (O&M), and Retirement. The SDP also addresses the preparation, execution, and documentation of software testing for software products. The SDP conforms to RG 1.173-1997 and IEEE Std. 1074-1995, except as specified in Appendix A of the Software Management Plan (SMP).

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 Std. 1012-1998 "IEEE Standard for Software Verification and Validation." This plan, in conjunction with other plans described in this section, addresses the various elements described in the related guidance documents, including IEEE Std. 1012-1998 which is endorsed by RG 1.168-2004.

Software Integration Plan

The Software Integration Plan (SIntP) describes the software integration 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 RG 1.170-1997, "Software Integration 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 Operation and Maintenance Plan (SOMP) defines the software process and activities used to operate and maintain the software product during plant operation. The SOMP defines requirements, methods, and considerations for developing the system O&M manual. The SOMP also addresses maintenance procedures and activities to enhance, modify, and maintain software once the software is installed in the plant.

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

The Software Safety Plan (SSP) establishes the processes and activities intended to ensure that the safety concerns of the software products are properly considered during the software development and are consistent with the defined system safety analyses as defined by RG 1.173-1997, "Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants". The SSP meets the guidelines specified in Chapter 7 of NUREG 0800 Standard Review Plan and the requirements outlined in section IEEE Std. 1228-1994, "IEEE Standard for Software Safety Plans".

Software Verification and Validation Plan

This Software Verification and Validation Plan (SVVP) establishes the V&V tasks for the software designed and developed for software products. This SVVP satisfies the requirements of RG 1.168-2004, except where specified in Appendix A. RG 1.168-2004 endorses IEEE Std. 1012-1998, "IEEE Standard for Verification and Validation Plans" and IEEE Std. 1028-1997, "IEEE Standard for Software Reviews and Audits".

Software Configuration Management Plan (SCMP)

The Software Configuration Management Plan (SCMP) establishes the Software Configuration Management activities for the design and development of the software products. This SCMP satisfies the requirements of RG 1.169-1997, "Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants", except where specified in Appendix A of the Software Quality Assurance Plan. RG 1.169-1997 endorses IEEE Std. 828, "IEEE Standard for Software Configuration Management Plans".

Cyber Security Program Plan

The Cyber Security Program Plan (CSPP) 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:

- Planning Phase
- Requirement Phase
- Design Phase
- Implementation Phase
- Test Phase
- Installation Phase
- Operation and Maintenance 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 defines the management, implementation and resource software characteristics.	A Results Analysis of the Software Management Plan (SMP) will be performed.	A results summary report shall be developed and it shall verify that the Software Management Plan (SMP) satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the design and development of the ESBWR software based products. The report shall also verify that assessments of the quality of vendor efforts are acceptable.

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. The Software Development Plan (SDP) describes the management of the ESBWR software development and defines the management, implementation and resource software characteristics.	A Results Analysis of the Software Development Plan (SDP) will be performed.	A results summary report shall be developed and it shall verify that the Software Development Plan (SDP) satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the design and development of the ESBWR software based products. The report shall verify that the software plan defines which tasks are associated with each life cycle phase and that inputs and outputs are defined. The report shall also ensure that methods of review, verification and validation of outputs are defined in an acceptable manor.

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The Software Quality Assurance Plan (SQAP) describes a systematic approach to the development and use of ESBWR software. It also defines the management, implementation and resource software characteristics.	A Results Analysis of the Software Quality Assurance Plan (SQAP) will be performed.	A results summary report shall be developed and it shall verify that the Software Quality Assurance Plan (SQAP) satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the design and development of the ESBWR software based products. The report shall also verify that high quality software, which performs all intended safety functions, is produced as a result of plan execution.
4. The Software Integration Plan (SIntP) summarizes the management, implementation, and resource characteristics of the integration program. It also defines the management, implementation and resource software characteristics.	A Results Analysis of the Software Integration Plan (SIntP) will be performed.	A results summary report shall be developed and it shall verify that the Software Integration Plan (SIntP) satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the integration of the ESBWR software based products. This report shall also verify that methods of software integration between commercial off the shelf (COTS), as well as previously developed software (PDS) and newly developed software are satisfactory.

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The Software Installation Plan (SIP) summarizes the management, implementation of software operations maintenance, and resource characteristics of the installation program. It also defines the management, implementation and resource software characteristics.	A Results Analysis of the Software Integration Plan (SIP) will be performed.	A Results Summary Report shall be developed and it shall verify that the Software Installation Plan (SIP) satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the installation of the ESBWR software based products.

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. The Software Operations andMaintenance Plan (SOMP), which provides an acceptance approach for management and execution of the software operations and maintenance activities, will be established for software-based products. The SOMP also defines the management, implementation and resource software characteristics.	A Results Analysis of the Software Operations and Maintenance Plan (SOMP) will be performed.	A Results Summary Report shall be developed and it shall verify that the Software Operations and Maintenance Plan (SOMP) satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the operations and maintenance tasks associated with ESBWR software based products. This report shall specify the methods of performing software operations and maintenance functions following turn over to the COL holder. This report shall also provide an assessment of the system's operational security by verifying the existence of a means to ensure that no unauthorized changes to hardware, software, and system parameters can be made. The report shall demonstrate existence of a monitoring program to detect penetration (or attempted penetration) of the system.

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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. It also defines the management, implementation and resource software characteristics.	A Results Analysis of the Software Training Plan (STrngP) will be performed.	A Results Summary Report shall be developed and it shall verify that the Software Training Plan (STrngP) satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the training tasks for staff working in the design, development, peer review, and testing of the software based products. This includes requirements for the training program of 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. It also defines the management, implementation and resource software characteristics.	A Results Analysis of the Software Safety Plan (SSP) will be performed. A Safety Analysis shall be performed for all Safety Related Software at prescribed points in the Software Life Cycle as defined in the SMP.	A Results Summary Report shall be developed ant it shall verify that the Software Safety Plan (SSP) satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the design and development tasks associated with ESBWR software based products Safety Analysis Reports shall demonstrate that management, implementation, and resource characteristics are maintained throughout the SW Life Cycle process.

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. The Software Verification and Validation Plan (SVVP) describes the Independent V&V organization responsible for executing the V&V tasks to ensure that the design requirements of each life cycle phase are traceable to a relevant requirement defined in the previous phase, and that the developed software-based product meets its specified requirements, performs its intended functions correctly and performs no unintended functions. The SVVP also defines the management, implementation and resource software characteristics.	A Results Analysis of the Software Verification and Validation Plan (SVVP) will be performed. A Verification and Validation Analysis shall be performed for all Safety Related Software at prescribed points in the Software Life Cycle as defined in the SMP. A Requirements Traceability Analysis (RTA) shall be performed for all Q and N3 Software.	A Results Summary Report shall be developed and it shall verify that the Software Verification and Validation Plan (SVVP) satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the design and development tasks associated with ESBWR software based productsThe report shall verify that the organizational, scheduling, and financial independence is maintained throughout the development process. V & V Reports shall demonstrate that management, implementation, and resource characteristics are maintained throughout the SW Life Cycle process. The Requirements Traceability Matrix shall demonstrate that management, implementation, and resource characteristics are maintained throughout the SW Life Cycle process.

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
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. It also defines the management, implementation and resource software characteristics.	A Results Analysis of the Software Configuration Management Plan (SCMP) will be performed. A Configuration Management analysis shall be performed for all Safety Related Software during the Baseline Review for each Software Life Cycle phase.	A Results Summary Report shall be developed and it shall verify that the Software Configuration Management Plan (SCMP) satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the design and development tasks associated with ESBWR software based products. In addition, the report shall verify that the following items are being maintained under the control of an organization that is responsible for processing and archiving the various versions of the software as well as supporting documentation. 1. Software requirements, designs, and code 2. Support software used in development 3. Libraries of software components essential to safety 4. Software plans that could affect quality

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		5. Test software requirements, designs, or code used in testing 6. Test results and analyses used to qualify software 7. Software documentation 8. Databases and software configuration data 9. Pre-developed software items that are safety system software 10. Software change documentation 11. Tools used in the software project for management, development, or assurance CM Reports shall demonstrate that management, implementation, and resource characteristics are maintained throughout the SW Life Cycle process.

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
11. The Cyber Security Program Plan (CSPP) developed using a structured design process to protect digital assets from cyber attack. The CSPP provides specific documentation and reviews during the following lifecycle phases: 1. Planning Phase 2. Requirements Phase 3. Design Phase 4. Implementation Phase 5. Test Phase 6. Installation Phase 7. Operation and Maintenance Phase	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 factory acceptance.	Inspection and test reports exist and conclude(s) that the Cyber Security Program Plan (CSPP) is developed using a structured design process to protect digital assets from cyber attack. The CSPP provides specific documentation and reviews during the following lifecycle phases: 1. Planning Phase 2. Requirements Phase 3. Design Phase 4. Implementation Phase 5. Test Phase 6. Installation Phase 7. Operation and Maintenance Phase

Table 3.2-1

ITAAC For Software Development

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
12. Software Design Documentation (SDD) is information recorded about a specific life cycle activity. Documentation includes software life-cycle design outputs and software life cycle process documentation.	A Baseline Review Analysis, which includes a configuration management report, shall be performed at each SW lifecycle phase.	A Results Summary Report shall be developed and it shall verify that that the SW Design Documentation satisfactorily addresses all of the managerial, implementation and resource characteristics necessary to accomplish the design and development tasks associated with ESBWR software based products.

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 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.	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.	Summary report(s) document that: 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.	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.	Summary report(s) document that: a. The FRA and AOF team members and backgrounds. b. Plant safety functional requirements. c. Safety 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.	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.	Summary report(s) document that: 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.	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.	Summary report(s) document that: 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.	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.	Summary report(s) document that: 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.

Table 3.3-1

ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6. Human System Interface (HSI) Design is performed in accordance with the MMIS and HFE Implementation Plan and its requirements.	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. Summary report(s) document that: 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 comprising the minimum inventory of HSI and 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 the HSI Design 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
7. Procedure Development is performed in accordance with the MMIS and HFE Implementation Plan and its requirements.	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.	Summary report(s) document that: a. Effective plant procedures derived from ESBWR EPGs are approved. 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.	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 the Training Development outcomes and results.	Summary report(s) document that: a. The Training Development team members and backgrounds. b. The purpose and scope of the Training Development. c. The roles of organizations involved and the facilities and resources needed to

Table 3.3-1

ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		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.

Table 3.3-1

ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9. Human Factors Verification and Validation (HF V&V) is performed in accordance with the MMIS and HFE Implementation Plan and its requirements.	HF V&V 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 HF V&V outcomes and results.	Summary report(s) document that: a. The HF V&V team members and 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.

Table 3.3-1

ITAAC For Human Factors Engineering

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10. Design Implementation is performed in accordance with the MMIS and HFE Implementation Plan and its requirements.	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.	Summary report(s) document that: 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 corrected or justified. f. The methodology and implementation for the Design Implementation 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
11. Human Performance Monitoring (HPM) is performed in accordance with the MMIS and HFE Implementation Plan and its requirements.	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.	Summary report(s) document that: 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 to maintain radiation exposures to plant personnel as low as reasonably achievable (ALARA). Radiation protection is provided by application of the design and radiation control principles:

(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.

(2) Area radiation monitoring provides local alarms (visual alarms in high noise areas) with variable alarm setpoints and readout/alarm capability.

(3) The plant design provides radiation shielding for rooms, corridors and operating areas commensurate with their occupancy requirements.

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 and shielding.

Table 3.4-1

ITAAC For Ventilation and Airborne Monitoring and Shielding

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.	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 • 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.	Calculation of radioactive airborne concentration demonstrates that: • 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 radionuclides will be a small fraction (10% or less) of the occupational concentration limits listed in 10 CFR 20 Appendix 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 or below) the occupational concentration limits listed in 10 CFR 20 Appendix B during the periods that occupancy is required. • For rooms that seldom require access, plant design provides containment and ventilation to reduce airborne contamination spread to other areas of lower contamination.

Table 3.4-1

ITAAC For Ventilation and Airborne Monitoring and Shielding

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. Area radiation monitoring provides local alarms (visual alarms in high noise areas) with variable alarm setpoints, and readout/alarm capability.	See Tier 1, Subsection 2.3.2.	See Tier 1, Subsection 2.3.2.
3. The plant design provides radiation shielding for rooms, corridors and operating areas commensurate with their occupancy requirements.	Analyses (with inspections) of the expected radiation levels in each plant area will verify the adequacy of the shielding designs.	Analysis/inspection report(s) demonstrate that the maximum expected radiation dose rates in each plant area (deep dose equivalent measured at 30 cm from the source of the radiation, not contact dose rates) are no greater than the dose rates specified for the following zones, based on the access requirements of that area for plant operation and maintenance. <u>Zone</u> / <u>Dose Rate (mSv/hr)</u> / <u>Access Requirements</u> A / ≤ 0.006 / Uncontrolled, unlimited access B / 0.006 ≤ 0.01 / Controlled and unlimited access C / 0.01 ≤ 0.05 / Controlled and limited access (20 hr/week) D / 0.05 ≤ 0.25 / Controlled and limited access (4 hr/week)

Table 3.4-1

ITAAC For Ventilation and Airborne Monitoring and Shielding

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
		Zone / Dose Rate (mSv/hr) / Access Requirements E / $0.25 \leq 1$ / Controlled and limited access (1 hr/week) F / $1 \leq 10$ / Limited and controlled access with special authorization permit required G / $10 \leq 100$ / Same as Zone F H / $100 \leq 1000$ / Same as Zone F I / $1000 \leq 5000$ / Same as Zone F J / > 5000 / Inaccessible during power and shutdown operations

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 GEH 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 of 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.

(1) 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.	Inspection will be performed for the existence of a report which establishes the estimated reliability of as-built risk –significant SSCs.	Reports and/or specifications exist that contain the list of risk-significant SSCs and conclude that the reliability of each as-built risk-significant SSC is consistent with the reliability assumed in the ESBWR Design 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. 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:

<u>Performance criteria</u>

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

<u>Design criteria</u>

- 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

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.	Inspections tests and/or analysis will be performed to verify that the post accident monitoring instrumentation is designed in conformance with the requirements as described in Section 3.7. {{DAC}}	Report(s) exists and conclude(s) that the post accident monitoring instrumentation is designed in conformance with the requirements as described in Section 3.7.

3.8 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

Design Description

(1) Safety-related electrical equipment located in a harsh environment can perform its safety-related function under normal, abnormal and design bases accident environmental conditions.

(2) Safety-related mechanical equipment located in a harsh environment can perform its safety-related function under normal, abnormal and design bases accident environmental conditions.

(3) Safety-related digital I&C equipment located in a mild environment is designed to perform its safety-related function under normal and AOO environmental conditions.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.8-1 specifies the environmental qualification inspections, test, analyses, and associated acceptance criteria for safety-related mechanical and electrical equipment.

Table 3.8-1

ITAAC for Environmental Qualification of Mechanical and Electrical Equipment

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. Safety-related electrical equipment located in a harsh environment can perform its safety-related function under normal, abnormal and design bases accident environmental conditions.	Safety-related electrical equipment located in a harsh environment is identified and:	
	i. Analysis will be performed to identify the environmental design bases including the definition of anticipated operational occurrences and normal, accident, and post-accident environments.	i. Reports document the analyses results identifying the environmental design bases including the definition of anticipated operational occurrences and normal, accident, and post-accident environments for safety-related electrical equipment located in a harsh environment.
	ii. Type tests, analyses, or a combination of type tests and analyses will be performed on as-built safety-related equipment identified as located in a harsh environment.	ii. Reports exist and conclude that the as-built safety-related equipment identified in the analyses as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

Table 3.8-1

ITAAC for Environmental Qualification of Mechanical and Electrical Equipment

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	iii. Inspection will be performed of the as-installed safety-related equipment and the associated wiring, cables, and terminations located in a harsh environment.	iii. Reports exist and conclude that the as-installed safety-related equipment and the associated wiring, cables, and terminations identified in the analyses as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.

Table 3.8-1

ITAAC for Environmental Qualification of Mechanical and Electrical Equipment

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. Safety-related mechanical equipment located in a harsh environment can perform its safety-related function under normal, abnormal and design bases accident environmental conditions.	Safety-related mechanical equipment located in a harsh environment is identified and:	
	i. Analysis will be performed to identify the environmental design bases including the definition of anticipated operational occurrences and normal, accident, and post-accident environments.	i. Reports document the analyses results identifying the environmental design bases including the definition of anticipated operational occurrences and normal, accident, and post-accident environments for the safety-related mechanical equipment located in a harsh environment.
	ii. Type tests and/or analyses of material data will be performed on safety-related mechanical equipment identified as located in a harsh environment.	ii. Report exists and concludes that the material data ensure that the as-built safety-related mechanical equipment identified in the type tests and/or analyses as qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
	iii. Inspection will be performed to verify proper non-metallic materials of the as-installed safety-related mechanical equipment located in a harsh environment.	iii. Reports exist and conclude that all non-metallic materials of safety-related mechanical equipment has been installed per the qualification requirements.

Table 3.8-1

ITAAC for Environmental Qualification of Mechanical and Electrical Equipment

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. Safety-related digital I&C equipment located in a mild environment is designed to perform its safety-related function under normal and AOO environmental conditions.	Safety-related digital I&C equipment located in a mild environment is identified and:	
	i. Analysis will be performed to identify the environmental design bases including the definition of anticipated operational occurrences and normal environments.	i. Report(s) document the analyses results identifying the environmental design bases including the definition of anticipated operational occurrences and normal environments for safety-related digital I&C equipment located in a mild environment:
	ii. Type tests, analyses, or a combination of type tests and analyses will be performed on safety-related digital I&C equipment located in a mild environment.	ii. Report(s) exist and conclude that all safety-related digital I&C equipment located in a mild environment is qualified to perform its safety function during the applicable normal and abnormal environmental conditions by type tests, analyses, or a combination of type tests and analyses for the time required to perform the safety function.
	iii. Inspection will be performed to verify the as-installed safety-related digital I&C equipment located in a mild environment	iii. Reports exist and conclude that all safety-related digital I&C equipment located in a mild environment has been installed per the qualification requirements.

4. INTERFACE MATERIAL

An applicant for a combined license (COL) that references the ESBWR certified design must provide design features or characteristics that comply with the interface requirements for the plant design and inspections, tests, analyses, and acceptance criteria (ITAAC) for the site-specific portion of the facility design, in accordance with 10 CFR 52.79 (c).

Tier 1 interfaces were identified for the conceptual design portion of the Plant Service Water System for the certified design.

4.1 PLANT SERVICE WATER SYSTEM

Design Description

The Plant Service Water System (PSWS) is the heat sink for the Reactor Component Cooling Water System. 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 interface requirements are necessary for supporting the post-72-hour cooling function of the PSWS. The PSWS is required to remove 2.02×10^{7} MJ (1.92×10^{10} BTU) over a period of 7 days without active makeup. 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. The combined license applicant referencing the certified design shall develop these inspections, tests, and analyses, together with their associated acceptance criteria.

5. SITE PARAMETERS

5.1 SCOPE AND PURPOSE

The intent of this section is to provide Tier 1 material that complies with the 10 CFR 52 requirements to define the site parameters postulated for the ESBWR certified design.

Assuming the certified design will be referenced for a wide range of sites, it is necessary to specify a set of site parameters enveloping the conditions that could occur at most potential power plant sites in the United States. These parameters are provided in Table 5.1-1. It is intended that any facility that references the certified design will utilize a site where the actual site-specific conditions are within the defined envelope.

In the case of seismic design parameters, deviations from the defined conditions may be justified by site-specific soil-structure interaction analyses. The results may be used to confirm the seismic design adequacy of the certified design using approved methods and acceptance criteria.

Table 5.1-1

Envelope of ESBWR Standard Plant Site Design Parameters

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** - 50-year Wind Speed (3-sec gust) 58.1 m/s (130 mph)
Maximum Flood (or Tsunami) Level:	0.3 m (1 ft) below plant grade
Tornado:	- Maximum Tornado Wind Speed: 147.5 m/s (330 mph) - Maximum Rotational Speed: 116.2 m/s (260 mph) - Translational Speed: 31.3 m/s (70 mph) - Radius: 45.7 m (150 ft) - Pressure Drop: 16.6 kPa (2.4 psi) - Rate of Pressure Drop: 11.7 kPa/s (1.7 psi/s) - Missile Spectrum: Spectrum I of SRP 3.5.1.4, Rev 2 applied to full building height.
Precipitation (for Roof Design):	- Maximum Rainfall Rate: 49.3 cm/hr (19.4 in/hr) - Maximum Short Term Rate: 15.7 cm (6.2 in) in 5 min. - Maximum Roof Load: 2873 Pa (60 lbf/ft^2) - Maximum Ground Snow Load (100-year recurrence interval): 2394 Pa (50 lb/ft^2) - Maximum 48-hr Winter Rainfall: 91.4 cm (36 in)
Ambient Design Temperature:	**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: 47.2°C (117°F) dry bulb 26.7°C (80°F) wet bulb (coincident) 31.1°C (88°F) wet bulb (non-coincident) - Minimum: -40°C (-40°F)

Table 5.1-1

Envelope of ESBWR Standard Plant Site Design Parameters

Soil Properties:	- Minimum Static Bearing Capacity: Reactor/Fuel Building: 699 kPa (14,600 lbf/ft^2) Control Building: 292 kPa (6,100 lbf/ft^2) Fire Water Service Complex: 165 kPa (3,450 lbf/ft^2) - Minimum Dynamic Bearing Capacity: **Reactor/Fuel Building:** Soft: 2700 kPa (56,400 lbf/ft^2) Medium: 7300 kPa (152,500 lbf/ft^2) Hard: 5400 kPa (112,800 lbf/ft^2) **Control Building:** Soft: 2800 kPa (58,500 lbf/ft^2) Medium: 2500 kPa (52,300 lbf/ft^2) Hard: 2400 kPa (50,200 lbf/ft^2) **Fire Water Service Complex (FWSC):** Soft: 440 kPa (9,200 lbf/ft^2) Medium: 540 kPa (11,300 lbf/ft^2) Hard: 670 kPa (14,000 lbf/ft^2) - Minimum Shear Wave Velocity: 300 m/s (1000 ft/s) - Liquefaction Potential: Seismic Category I Structures: None under footprint of Seismic Category I structures resulting from site-specific SSE - Angle of Internal Friction: ≥ 30 degrees
Seismology:	- SSE Horizontal Ground Response Spectra: See Figure 5.1-1 - SSE Vertical Ground Response Spectra: See Figure 5.1-2
Hazards in Site Vicinity:	- Site Proximity Missiles and Aircraft: $\leq 10^{-7}$ per year - Toxic Gases: None * - Volcanic Activity: None
* Maximum toxic gas concentrations at the Main Control Room (MCR) HVAC intakes	< toxicity limits
Required Stability of Slopes:	- Factor of safety for static (non-seismic) loading 1.5 - Factor of safety for dynamic (seismic) loading 1.1

Table 5.1-1

Envelope of ESBWR Standard Plant Site Design Parameters

Meteorological Dispersion (X/Q):	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 air intakes (emergency and normal)	Control Room X/Q: *		
** Due to symmetry, Turbine Building X/Q values are identical for unfiltered inleakage and air intakes.	Reactor Building – Diffuse Source		
NA Values are not required for any dose analysis.	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 **		
	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
	Fuel Building – Diffuse Source		
	0-2 hours:	NA	2.80E-03 s/m3
	2-8 hours:	NA	2.50E-03 s/m3
	8-24 hours:	NA	1.25E-03 s/m3
	1-4 days:	NA	1.10E-03 s/m3
	4-30 days:	NA	1.00E-03 s/m3

Table 5.1-1

Envelope of ESBWR Standard Plant Site Design Parameters

Meteorological Dispersion (X/Q): (continued)	Fuel Building Cask Doors		
	0-2 hours:	NA	1.50E-03 s/m3
	2-8 hours:	NA	1.30E-03 s/m3
	8-24 hours:	NA	6.80E-04 s/m3
	1-4 days:	NA	5.60E-04 s/m3
	4-30 days:	NA	4.30E-04 s/m3
	Radwaste Building		
	0-2 hours:	NA	1.50E-03 s/m^3
	2-8 hours:	NA	1.30E-03 s/m^3
	8-24 hours:	NA	6.80E-04 s/m^3
	1-4 days:	NA	5.60E-04 s/m^3
	4-30 days:	NA	4.30E-04 s/m^3

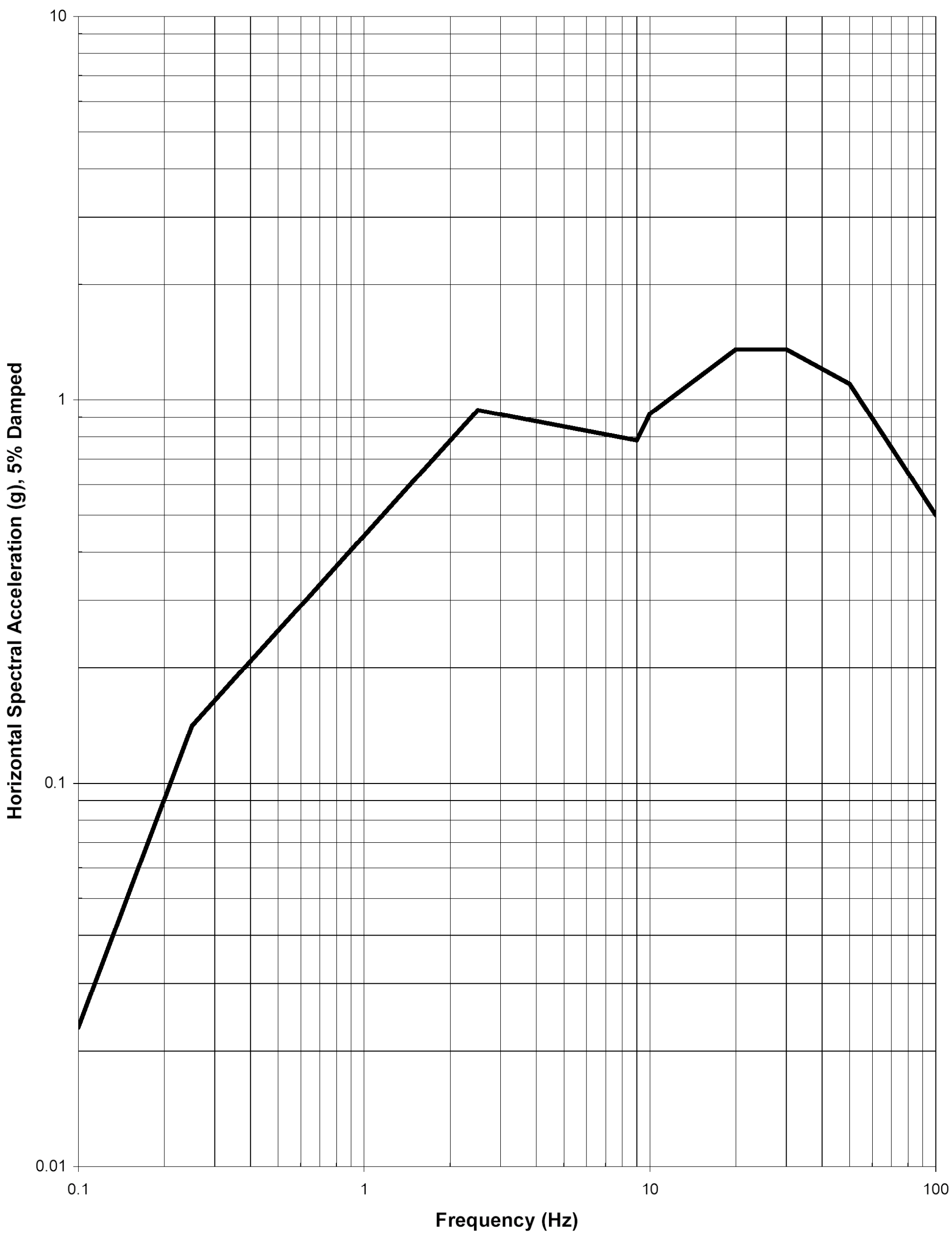


Figure 5.1-1. ESBWR Horizontal SSE Design Ground Spectra at Foundation Level

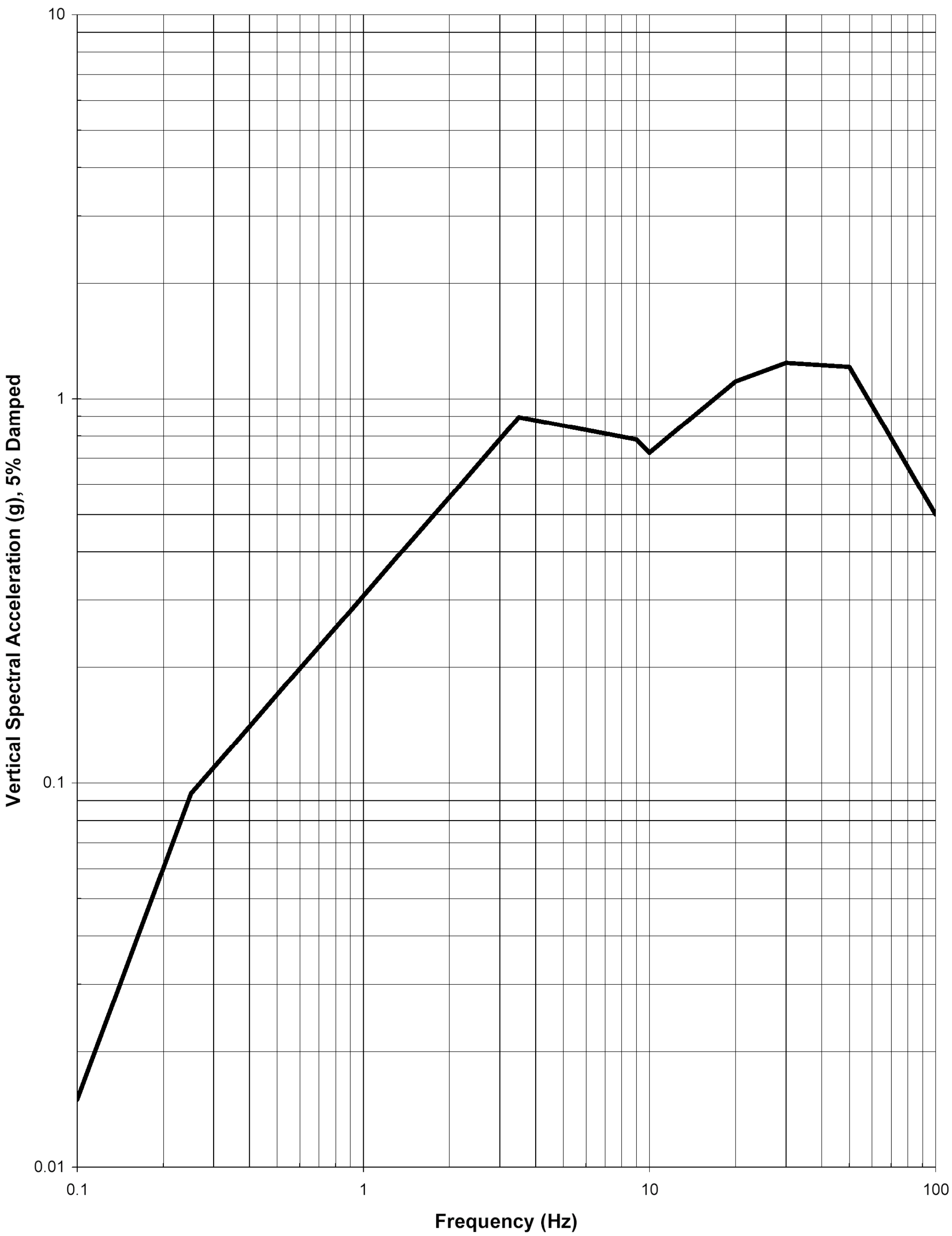


Figure 5.1-2. ESBWR Vertical SSE Design Ground Response Spectra at Foundation Level