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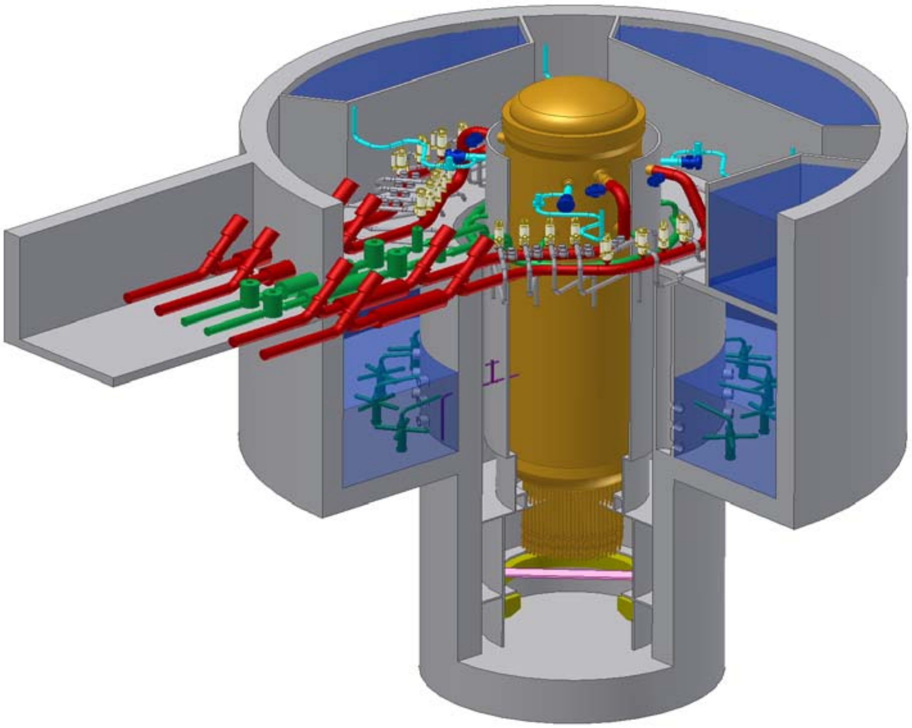

ESBWR Design Control Document
Tier 2
Chapter 6
Engineered Safety Features

(Conditional Release - pending closure of design verifications)

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Global Abbreviations And Acronyms List

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

Global Abbreviations And Acronyms List

Term	**Definition**
ASTM	American Society of Testing Methods
AT	Unit Auxiliary Transformer
ATLM	Automated Thermal Limit Monitor
ATWS	Anticipated Transients Without Scram
AV	Allowable Value
AWS	American Welding Society
AWWA	American Water Works Association
B&PV	Boiler and Pressure Vessel
BAF	Bottom of Active Fuel
BHP	Brake Horse Power
BiMAC	Basemat-Internal Melt Arrest Coolability
BOC	Beginning of Cycle
BOP	Balance of Plant
BPU	Bypass Unit
BPV	Bypass Valve
BPWS	Banked Position Withdrawal Sequence
BRE	Battery Room Exhaust
BRL	Background Radiation Level
BTP	NRC Branch Technical Position
BTU	British Thermal Unit
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAV	Cumulative Absolute Velocity
C&FS	Condensate and Feedwater System
C&I	Control and Instrumentation
C/C	Cooling and Cleanup
CB	Control Building
CBGAHVS	Control Building General Area
CBHVAC	Control Building HVAC
CBHVS	Control Building Heating, Ventilation and Air Conditioning System
CCI	Core-Concrete Interaction
CDF	Core Damage Frequency
CDU	Condensing Unit
CFR	Code of Federal Regulations
CH	Chugging
CIRC	Circulating Water System
CIS	Containment Inerting System
CIV	Combined Intermediate Valve
CLAVS	Clean Area Ventilation Subsystem of Reactor Building HVAC

Global Abbreviations And Acronyms List

Term	Definition
CM	Cold Machine Shop
CMS	Containment Monitoring System
CMU	Control Room Multiplexing Unit
CO	Condensate Oscillation
COL	Combined Operating License
COLR	Core Operating Limits Report
CONAVS	Controlled Area Ventilation Subsystem of Reactor Building HVAC
CPR	Critical Power Ratio
CPS	Condensate Purification System
CPU	Central Processing Unit
CR	Control Rod
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDH	Control Rod Drive Housing
CRDHS	Control Rod Drive Hydraulic System
CRGT	Control Rod Guide Tube
CRHA	Control Room Habitability Area
CRHAHVS	Control Room Habitability Area HVAC Sub-system
CRT	Cathode Ray Tube
CS&TS	Condensate Storage and Transfer System
CSDM	Cold Shutdown Margin
CS / CST	Condensate Storage Tank
CT	Main Cooling Tower
CTVCF	Constant Voltage Constant Frequency
CUF	Cumulative usage factor
CWS	Chilled Water System
D-RAP	Design Reliability Assurance Program
DAC	Design Acceptance Criteria
DAW	Dry Active Waste
DBA	Design Basis Accident
DBE	Design Basis Event
DB%	Dry-Basis-Percent
dc / DC	Direct Current
DCD	Design Control Document
DCS	Drywell Cooling System
DCIS	Distributed Control and Information System
DEPSS	Drywell Equipment and Pipe Support Structure
DF	Decontamination Factor
D/F	Diaphragm Floor

Global Abbreviations And Acronyms List

Term	Definition
DG	Diesel-Generator
DHR	Decay Heat Removal
DM&C	Digital Measurement and Control
DOF	Degree of Freedom
DOI	Dedicated Operators Interface
DORT	Discrete Ordinates Techniques
DOT	Department of Transportation
dPT	Differential Pressure Transmitter
DPS	Diverse Protection System
DPV	Depressurization Valve
DR&T	Design Review and Testing
DTM	Digital Trip Module
DW	Drywell
EAB	Exclusion Area Boundary
EB	Electrical Building
EBAS	Emergency Breathing Air System
EBHV	Electrical Building HVAC
ECCS	Emergency Core Cooling System
E-DCIS	Essential DCIS (Distributed Control and Information System)
EDO	Environmental Qualification Document
EFDS	Equipment and Floor Drainage System
EFPY	Effective Full Power Years
EFU	Emergency Filter Unit
EHC	Electro-Hydraulic Control (Pressure Regulator)
ENS	Emergency Notification System
EOC	Emergency Operations Center
EOC	End of Cycle
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EPDS	Electric Power Distribution System
EPG	Emergency Procedure Guidelines
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ERICP	Emergency Rod Insertion Control Panel
ERIP	Emergency Rod Insertion Panel
ESF	Engineered Safety Feature
ESP	Early Site Permit
ETS	Emergency Trip System
FAC	Flow-Accelerated Corrosion

Global Abbreviations And Acronyms List

Term	Definition
FAPCS	Fuel and Auxiliary Pools Cooling System
FATT	Fracture Appearance Transition Temperature
FB	Fuel Building
FBHV	Fuel Building HVAC
FCI	Fuel-Coolant Interaction
FCISL	Fuel Cladding Integrity Safety Limit
FCM	File Control Module
FCS	Flammability Control System
FCU	Fan Cooling Unit
FDDI	Fiber Distributed Data Interface
FEBAVS	Fuel Building Ventilation System
FFT	Fast Fourier Transform
FFWTR	Final Feedwater Temperature Reduction
FHA	Fire Hazards Analysis
FHA	Fuel Handling Accident
FIV	Flow-Induced Vibration
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analysis
FPS	Fire Protection System
FO	Diesel Fuel Oil Storage Tank
FOAKE	First-of-a-Kind Engineering
FPC	Fuel Pool Cleanup
FPE	Fire Pump Enclosure
FS	Partial Full Scale
FSI	Fluid Structure Interaction
FTDC	Fault-Tolerant Digital Controller
FW	Feedwater
FWCS	Feedwater Control System
FWLB	Feedwater Line Break
FWS	Fire Water Storage Tank
GCS	Generator Cooling System
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GENE	GE Nuclear Energy
GEN	Main Generator System
GETAB	General Electric Thermal Analysis Basis
GL	Generic Letter
GM	Geiger-Mueller Counter

Global Abbreviations And Acronyms List

Term	**Definition**
GM-B	Beta-Sensitive GM (Geiger-Mueller Counter) Detector
GENE	General Electric Nuclear Energy
GNF	Global Nuclear Fuel
GSIC	Gamma-Sensitive Ion Chamber
GSOS	Generator Sealing Oil System
GWSR	Ganged Withdrawal Sequence Restriction
HAZ	Heat-Affected Zone
HCU	Hydraulic Control Unit
HCW	High Conductivity Waste
HDVS	Heater Drain and Vent System
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HELSA	High Energy Line Separation Analysis
HEP	Human Error Probability
HEPA	High Efficiency Particulate Air/Absolute
HFE	Human Factors Engineering
HFF	Hollow Fiber Filter
HGCS	Hydrogen Gas Cooling System
HIC	High Integrity Container
HID	High Intensity Discharge
HIS	Hydraulic Institute Standards
HM	Hot Machine Shop & Storage
HP	High Pressure
HPNSS	High Pressure Nitrogen Supply System
HPT	High-Pressure Turbine
HRA	Human Reliability Assessment
HSI	Human-System Interface
HSSS	Hardware/Software System Specification
HVAC	Heating, Ventilation and Air Conditioning
HVS	High Velocity Separator
HVT	Horizontal Vent Test
HWC	Hydrogen Water Chemistry
HWCS	Hydrogen Water Chemistry System
HWS	Hot Water System
HX	Heat Exchanger
I&C	Instrumentation and Control
I/O	Input/Output
IAS	Instrument Air System
IASCC	Irradiation Assisted Stress Corrosion Cracking

Global Abbreviations And Acronyms List

Term	Definition
IBA	Intermediate Break Accident
IBC	International Building Code
IC	Ion Chamber
IC	Isolation Condenser
ICD	Interface Control Diagram
ICPR	Initial Critical Power Ratio
ICS	Isolation Condenser System
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IED	Instrument and Electrical Diagram
IEEE	Institute of Electrical and Electronic Engineers
IFTS	Inclined Fuel Transfer System
IGSCC	Intergranular Stress Corrosion Cracking
IIS	Iron Injection System
ILRT	Integrated Leak Rate Test
IOP	Integrated Operating Procedure
IMC	Induction Motor Controller
IMCC	Induction Motor Controller Cabinet
IRM	Intermediate Range Monitor
ISA	Instrument Society of America
ISI	In-Service Inspection
ISLT	In-Service Leak Test
ISM	Independent Support Motion
ISMA	Independent Support Motion Response Spectrum Analysis
ISO	International Standards Organization
ITA	Inspections, Tests or Analyses
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
ITA	Initial Test Program
LAPP	Loss of Alternate Preferred Power
LBB	Leak Before Break
LCO	Limiting Conditions for Operation
LCW	Low Conductivity Waste
LD	Logic Diagram
LDA	Lay down Area
LDW	Lower Drywell
LD&IS	Leak Detection and Isolation System
LERF	Large Early Release Frequency
LFCV	Low Flow Control Valve
LHGR	Linear Heat Generation Rate

Global Abbreviations And Acronyms List

Term	**Definition**
LLRT	Local Leak Rate Test
LMU	Local Multiplexer Unit
LO	Dirty/Clean Lube Oil Storage Tank
LOCA	Loss-of-Coolant-Accident
LOFW	Loss-of-feedwater
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPCRD	Locking Piston Control Rod Drive
LPMS	Loose Parts Monitoring System
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LUA	Lead Use Assembly
LWMS	Liquid Waste Management System
MAAP	Modular Accident Analysis Program
MAPLHGR	Maximum Average Planar Linear Head Generation Rate
MAPRAT	Maximum Average Planar Ratio
MBB	Motor Built-In Brake
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MCOP	Manual containment overpressure protection (function)
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MCRP	Main Control Room Panel
MELB	Moderate Energy Line Break
MLHGR	Maximum Linear Heat Generation Rate
MMI	Man-Machine Interface
MMIS	Man-Machine Interface Systems
MOV	Motor-Operated Valve
MPC	Maximum Permissible Concentration
MPL	Master Parts List
MRBM	Multi-Channel Rod Block Monitor
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSL	Main Steamline
MSLB	Main Steamline Break
MSLBA	Main Steamline Break Accident
MSR	Moisture Separator Reheater

Global Abbreviations And Acronyms List

Term	**Definition**
MSV	Mean Square Voltage
MT	Main Transformer
MTTR	Mean Time To Repair
MWS	Makeup Water System
NBR	Nuclear Boiler Rated
NBS	Nuclear Boiler System
NCIG	Nuclear Construction Issues Group
NDE	Nondestructive Examination
NE-DCIS	Non-Essential Distributed Control and Information System
NDRC	National Defense Research Committee
NDT	Nil Ductility Temperature
NFPA	National Fire Protection Association
NIST	National Institute of Standard Technology
NICWS	Nuclear Island Chilled Water Subsystem
NMS	Neutron Monitoring System
NOV	Nitrogen Operated Valve
NPHS	Normal Power Heat Sink
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NS	Non-seismic (non-seismic Category I)
NSSS	Nuclear Steam Supply System
NT	Nitrogen Storage Tank
NTSP	Nominal Trip Setpoint
O&M	Operation and Maintenance
O-RAP	Operational Reliability Assurance Program
OBCV	Overboard Control Valve
OBE	Operating Basis Earthquake
OGS	Offgas System
OHLHS	Overhead Heavy Load Handling System
OIS	Oxygen Injection System
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLU	Output Logic Unit
OOS	Out-of-Service
ORNL	Oak Ridge National Laboratory
OSC	Operational Support Center
OSHA	Occupational Safety and Health Administration
OSI	Open Systems Interconnect
P&ID	Piping and Instrumentation Diagram

Global Abbreviations And Acronyms List

Term	Definition
PA/PL	Page/Party-Line
PABX	Private Automatic Branch (Telephone) Exchange
PAM	Post Accident Monitoring
PAR	Passive Autocatalytic Recombiner
PAS	Plant Automation System
PASS	Post Accident Sampling Subsystem of Containment Monitoring System
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PCV	Primary Containment Vessel
PDA	Piping Design Analysis
PFD	Process Flow Diagram
PGA	Peak Ground Acceleration
PGCS	Power Generation and Control Subsystem of Plant Automation System
PH	Pump House
PL	Parking Lot
PM	Preventive Maintenance
PMCS	Performance Monitoring and Control Subsystem of NE-DCIS
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PQCL	Product Quality Check List
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitoring
PS	Plant Stack or Pool Swell
PSD	Power Spectral Density
PSS	Process Sampling System
PSTF	Pressure Suppression Test Facility
PSWS	Plant Service Water System
PT	Pressure Transmitter
PWR	Pressurized Water Reactor
QA	Quality Assurance
RACS	Rod Action Control Subsystem
RAM	Reliability, Availability and Maintainability
RAPI	Rod Action and Position Information
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBC	Rod Brake Controller
RBCC	Rod Brake Controller Cabinet

Global Abbreviations And Acronyms List

Term	Definition
RBCWS	Reactor Building Chilled Water Subsystem
RBHV	Reactor Building HVAC (Heating, Ventilation and Air Conditioning)
RBS	Rod Block Setpoint
RBV	Reactor Building Vibration
RC&IS	Rod Control and Information System
RCC	Remote Communication Cabinet
RCCV	Reinforced Concrete Containment Vessel
RCCWS	Reactor Component Cooling Water System
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RDA	Rod Drop Accident
RDC	Resolver-to-Digital Converter
REPAVS	Refueling and Pool Area Ventilation Subsystem of Fuel Building HVAC (Heating, Ventilation and Air Conditioning)
RFP	Reactor Feed Pump
RG	Regulatory Guide
RHR	Residual Heat Removal (function)
RHX	Regenerative Heat Exchanger
RMS	Root Mean Square
RMS	Radiation Monitoring Subsystem
RLP	Reference Loading Pattern
RMU	Remote Multiplexer Unit
RO	Reverse Osmosis
ROM	Read-only Memory
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRPS	Reference Rod Pull Sequence
RSM	Rod Server Module
RSPC	Rod Server Processing Channel
RSS	Remote Shutdown System
RSSM	Reed Switch Sensor Module
RSW	Reactor Shield Wall
RTIF	Reactor Trip and Isolation Function(s)
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RTP	Reactor Thermal Power
RW	Radwaste Building
RWBCR	Radwaste Building Control Room
RWBGA	Radwaste Building General Area
RWBHVAC	Radwaste Building HVAC (Heating, Ventilation and Air Conditioning)

Global Abbreviations And Acronyms List

Term	Definition
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
SA	Severe Accident
SAM	Severe Accident Management
SAR	Safety Analysis Report
SB	Service Building
SBA	Small Break Accident
S/C	Digital Gamma-Sensitive GM (Geiger-Mueller Counter) Detector
SC	Suppression Chamber
S/D	Scintillation Detector
S/DRSRO	Single/Dual Rod Sequence Restriction Override
S/N	Signal-to-Noise
S/P	Suppression Pool
SAS	Service Air System
SB&PC	Steam Bypass and Pressure Control System
SBO	Station Blackout
SBWR	Simplified Boiling Water Reactor
SCEW	System Component Evaluation Work
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling
SDM	Shutdown Margin
SDS	System Design Specification
SEOA	Sealed Emergency Operating Area
SER	Safety Evaluation Report
SF	Service Water Building
SFA	Spent Fuel Assembly
SFP	Spent fuel pool
SIL	Service Information Letter
SIT	Structural Integrity Test
SIU	Signal Interface Unit
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SMU	SSLC (Safety System Logic and Control) Multiplexing Unit
SOV	Solenoid Operated Valve
SP	Setpoint
SPC	Suppression Pool Cooling

Global Abbreviations And Acronyms List

Term	Definition
SPDS	Safety Parameter Display System
SPTMS	Suppression Pool Temperature Monitoring Subsystem of Containment Monitoring System
SR	Surveillance Requirement
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRS	Software Requirements Specification
SRSRO	Single Rod Sequence Restriction Override
SRSS	Square Root Sum of Squares
SRV	Safety Relief Valve
SRVDL	Safety Relief Valve Discharge Line
SSAR	Standard Safety Analysis Report
SS	Sub-scale
SST	Sub-scale Test
SSC(s)	Structure, System and Component(s)
SSE	Safe Shutdown Earthquake
SSI	Soil Structure Interaction
SSLC	Safety System Logic and Control
SSPC	Steel Structures Painting Council
ST	Spare Transformer
STI	Startup Test Instruction
STP	Sewage Treatment Plant
STRAP	Scram Time Recording and Analysis Panel
STRP	Scram Time Recording Panel
SV	Safety Valve
SWH	Static Water Head
SWMS	Solid Waste Management System
SY	Switch Yard
TAF	Top of Active Fuel
TASS	Turbine Auxiliary Steam System
TB	Turbine Building
TBCE	Turbine Building Compartment Exhaust
TBAS	Turbine Building Air Supply
TBE	Turbine Building Exhaust
TBLOE	Turbine Building Lube Oil Area Exhaust
TBS	Turbine Bypass System
TBHV	Turbine Building HVAC (Heating, Ventilation and Air Conditioning)
TBV	Turbine Bypass Valve

Global Abbreviations And Acronyms List

Term	Definition
TC	Training Center
TCCWS	Turbine Component Cooling Water System
TCS	Turbine Control System
TCV	Turbine Control Valve
TDH	Total Developed Head
TEDE	Total Effective Dose Equivalent
TEMA	Tubular Exchanger Manufacturers' Association
TFSP	Turbine First Stage Pressure
TG	Turbine Generator
TGSS	Turbine Gland Seal System
THA	Time-History Accelerograph
TIP	Traversing In-core Probe
TLOS	Turbine Lubricating Oil System
TLU	Trip Logic Unit
TMI	Three Mile Island
TMSS	Turbine Main Steam System
TRAC	Transient Reactor Analysis Code
TRM	Technical Requirements Manual
TS	Technical Specification(s)
TSC	Technical Support Center
TSI	Turbine Supervisory Instrument
TSV	Turbine Stop Valve
TTWFATBV	Turbine trip with failure of all bypass valves
UBC	Uniform Building Code
UHS	Ultimate Heat Sink
UL	Underwriter's Laboratories Inc.
UPS	Uninterruptible Power Supply
URD	Utilities Requirements Document
USE	Upper Shelf Energy
USM	Uniform Support Motion
USMA	Uniform Support Motion Response Spectrum Analysis
USNRC	United States Nuclear Regulatory Commission
USS	United States Standard
UV	Ultraviolet
V&V	Verification and Validation
Vac / VAC	Volts Alternating Current
Vdc / VDC	Volts Direct Current
VDU	Video Display Unit
VW	Vent Wall

Global Abbreviations And Acronyms List

Term	Definition
VWO	Valves Wide Open
WD	Wash Down Bays
WH	Warehouse
WS	Water Storage
WT	Water Treatment
WW	Wetwell
XMFR	Transformer
ZPA	Zero Period Acceleration

6. ENGINEERED SAFETY FEATURES

6.0 GENERAL

The engineered safety features (ESF) of this plant are those systems provided to mitigate the consequences of postulated accidents. The features can be divided into three general groups: (1) containment and fission product removal systems; (2) emergency core cooling systems; and (3) control room habitability systems. The systems and major features in each general group are:

(1) Containment and Fission Product Removal Systems
   - a. Containment System (CS)
   - b. Passive Containment Cooling System (PCCS)

(2) Emergency Core Cooling Systems (ECCS)
   - a. Gravity-Driven Cooling System (GDCS)
   - b. Automatic Depressurization System (ADS)
   - c. Isolation Condenser System (ICS)
   - d. Standby Liquid Control (SLC) system

(3) Control Room Habitability Systems
   - a. Sealed Emergency Operating Area (SEOA)
   - b. Emergency Breathing Air System (EBAS)

6.1 ENGINEERED SAFETY FEATURE MATERIALS

Materials used in the engineered safety features (ESF) components have been evaluated to ensure that material interactions do not occur that can potentially impair operation of the ESF. Materials have been selected to withstand the environmental conditions encountered during normal operation and postulated accidents. Their compatibility with core and containment spray water has been considered, and the effects of radiolytic decomposition products have also been evaluated.

Coatings used on exterior surfaces within the primary containment are suitable for the environmental conditions expected. Primarily metallic and metal-encapsulated insulation are used inside containment, except antisweat insulation used on cooling water lines. All nonmetallic thermal insulation employed is required to have the proper ratio of leachable sodium plus silicate ions to leachable chloride plus fluoride (Regulatory Guide 1.36), in order to minimize the possible contribution to stress corrosion cracking of austenitic stainless steel.

6.1.1 Metallic Materials

This subsection addresses or references to other DCD locations that address the relevant requirements of General Design Criteria (GDC) 1, 4, 14, 31, 35, 41, Appendix B and 10 CFR 50.55a discussed in SRP 6.1.1 Draft R2. The plant meets the requirements of

(1) GDC 1 and 10 CFR 50.55a as they relate to quality standards being used for design, fabrication, erection and testing of ESF components and the identification of applicable codes and standards;

(2) GDC 4 as it relates to compatibility of ESF components with environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents;

(3) GDC 14 as it relates to design, fabrication, erection, and testing of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture;

(4) GDC 31 as it relates to extremely low probability of rapidly propagating fracture and gross rupture of the reactor coolant pressure boundary;

(5) GDC 35 as it relates to assurance that core cooling is provided following a LOCA at such a rate that fuel and clad damage that could inhibit core cooling is prevented and that the clad metal-water reaction is limited to negligible amounts;

(6) GDC 41 as it relates to control of the concentration of hydrogen in the containment atmosphere following postulated accidents to assure that containment integrity is maintained; and

(7) Appendix B to 10 CFR Part 50, Criteria IX and XIII "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" - as they relate to control of special processes and to the requirement that measures be established to control the cleaning of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.

6.1.1.1 Materials Selection and Fabrication

The evaluation of reactor coolant pressure boundary (RCPB) materials is provided within Subsection 5.2.3, and Table 5.2-4 lists the principal pressure-retaining materials and the appropriate materials specifications for the RCPB components. Table 6.1-1 lists the principal pressure-retaining materials and the appropriate material specifications of the Containment System, and the emergency core cooling systems.

6.1.1.2 Compatibility of Construction Materials with Core Cooling Water and Containment Sprays

All materials of construction used in essential portions of ESF systems are resistant to corrosion, both in the medium contained and the external environment. General corrosion of all materials, except carbon and low-alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low-alloy steel.

Demineralized water, with no additives, is employed in ESBWR core cooling water and containment sprays. (See Subsection 9.2.3 for a description of the water quality requirements.) Leaching of chlorides from concrete and other substances is not significant. No detrimental effects occur on any of the ESF construction materials from allowable containment levels in the high-purity water. Thus, the materials are compatible with the post-LOCA environment.

6.1.1.3 Controls for Austenitic Stainless Steel

6.1.1.3.1 Control of the Use of Sensitized Stainless Steel

Controls to avoid severe sensitization are discussed within Subsection 5.2.3.

6.1.1.3.2 Process Controls to Minimize Exposure to Contaminants

Process controls for austenitic stainless steel are discussed within Subsection 5.2.3.

6.1.1.3.3 Use of Cold Worked Austenitic Stainless Steel

Austenitic stainless steels (300 series) are generally used in the solution heat treated condition. During bending and fabrication, the bend radius, the material hardness, and the surface finish of ground surfaces are controlled. Where the controls are not met, the material is required to be re-solution heat treated.

6.1.1.3.4 Thermal Insulation Requirements

Nonmetallic thermal insulation materials used on ESF systems are selected, procured, tested and stored in accordance with Regulatory Guide 1.36. Insulation is required to have the proper ratio of leachable sodium plus silicate ions to leachable chloride plus fluoride ions as described in Regulatory Guide 1.36.

6.1.1.3.5 Avoidance of Hot Cracking of Stainless Steel

Process controls to avoid hot cracking are discussed in Subsection 5.2.3.4.2.

6.1.1.4 Composition, Compatibility and Stability of Containment and Core Coolants

Demineralized water from the condensate storage tank or the suppression pool, with no additives is employed in the core cooling water and containments sprays (see Subsections 9.2.3 and 9.2.6). One exception is that the sodium pentaborate liquid control solution if used, enters through the Standby Liquid Control system sparger system.

The post-LOCA ESF coolant, which is high-purity water, comes from one of two sources. Water in the 304L stainless steel-lined Gravity-Driven Cooling System (GDCS) pools and suppression pool is maintained at high purity (low corrosion attack) by the Fuel and Auxiliary Pools Cooling System (FAPCS). See Subsection 9.1.3 for further details. Since the design pH range (5.6-8.6) is maintained, corrosive attack on the pool liner (304L SS) will be insignificant over the life of the plant (Subsection 3.8.1.4).

Because of the methods described above (coolant storage provisions, insulation materials requirements, and the like), as well as the fact that the containment has no significant stored quantities of acidic or basic materials, the post-LOCA aqueous phase pH in all areas of containment will have a flat time history. In other words, the liquid coolant will remain at its design basis pH throughout the event. As a result, post-LOCA hydrogen generation due to corrosion is considered negligible.

6.1.2 Organic Materials

Relevant to organic materials, this subsection addresses or references to other DCD locations that address the relevant requirements of Appendix B to 10 CFR Part 50 as it relates to the quality assurance requirements for the design, fabrication and construction of safety-related structures, systems and components. The coating systems applied inside the containment meet the regulatory positions of Regulatory Guide 1.54 and the standards of ASTM D 5144, as applicable.

6.1.2.1 Protective Coatings

The use of organic protective coatings within the containment has been kept to a minimum. The major use of such coatings is on the carbon steel containment liner, internal steel structures and equipment inside the drywell and wetwell.

The epoxy coatings are specified to meet the requirements of Regulatory Guide 1.54 and are qualified using the standard ANSI tests. However, because of the impracticability of using these special coatings on all equipment, certain exemptions are allowed. The exemptions are restricted to small-size equipment where, in case of a LOCA, the paint debris is not a safety hazard. Exemptions include such items as electronic/electrical trim, covers, face plates and valve handles. Other than these minor exemptions, all coatings within the containment are qualified to Regulatory Guide 1.54. (See Subsection 6.1.3.1 for COL items.)

6.1.2.2 Other Organic Materials

Materials used in or on the ESF equipment have been reviewed with respect to radiolytic and pyrolytic decomposition and attendant effects on safe operation of the system. For example, fluorocarbon plastic (Teflon) is not permitted in environments that reach temperatures greater than 149°C (300°F), or radiation exposures above 10^4 rads.

Other organic materials in the containment are qualified to environmental conditions in the containment. (See Subsection 6.1.3.1 for COL items.)

6.1.2.3 Evaluation

For each application the materials have been specified to withstand an appropriate radiation dose for their design life, without suffering any significant radiation-induced damage. The specified integrated radiation doses are consistent with those listed in Section 3.11.

In addition-since the containment post-accident environment consists of mostly hot water, nitrogen, and steam no significant chemical degradation of these materials is expected, nor should be because of strict application of inspection and testing. Solid debris from the organic materials discussed is not expected to be generated to any significant extent.

6.1.3 COL Information

6.1.3.1 Protective Coatings and Organic Materials

The COL applicant shall:

(1) Indicate the total amount of protective coatings and organic materials used inside the containment that do not meet the requirements of ASTM D 5144 and Regulatory Guide 1.54.

(2) Evaluate the generation rate, as a function of time, of combustible gases that can be formed from these unqualified organic materials under DBA conditions.

(3) Provide the technical basis and assumptions used for this evaluation.

6.1.4 References

None.

Table 6.1-1

Engineered Safety Features Component Materials

Component	Form	Material	Specification (ASTM/ASME)
Containment			
Containment Vessel Liner [1]			
	Plate ≤ 64 mm	Carbon Steel	SA-285 Gr ASA-516 Gr 60 or Gr 70
	Plate > 64 mm	Carbon Steel	SA-516 Gr 60 or Gr 70
	Plate	Stainless Steel	SA-240 Type 304L
Penetrations	Plate	Carbon Steel	SA-516 Gr 60 or Gr 70 SA-537 Class 1
	Pipe	Carbon Steel	SA-333 Gr 6
Pool Liner	Sheet	Stainless Steel	A 240 Type 304L or A 167 Type 304L
Drywell Head, Personnel Lock, Equipment Hatch			
	Plate	Carbon Steel	SA-516 Gr 70 or SA-537 Class 1
Structural Steel	Shapes	Carbon Steel	A 36, A 572 Gr 50
Vent Pipe	Plate	Stainless Steel	SA-240 Gr 304L
PCCS			
Condenser	Forging	Stainless Steel	SA-182 Gr F304L
	Tube	Stainless Steel	SA-213 Gr TP304L
	Pipe	Stainless Steel	SA-312 Gr TP304L
Piping	Pipe	Stainless Steel	SA-312 Gr TP304L
Flanges	Forging	Stainless Steel	SA-182 Gr F304L
Nuts and Bolts	Bar	Stainless Steel	SA-194 Gr 8, SA-193 Gr B8
ADS Depressurization Valves			
Body	Forging	Stainless Steel	SA-182 Gr F304L or F316L [2]
	Casting	Stainless Steel	SA-351 Gr CF3 or CF3M
GDCS			

Table 6.1-1

Engineered Safety Features Component Materials

Component		Form	Material	Specification (ASTM/ASME)
	Piping	Pipe	Stainless Steel	SA-376 Gr TP304L or TP316L [2] SA-312 Gr TP304L or TP316L [2] SA-358 Gr TP304L or TP316L [2]
	Fittings	Forging	Stainless Steel	SA-182 Gr F304L or F316L [2] SA-403 WP 304L or WP 316L [2]
	Flanges	Forging	Stainless Steel	SA-182 Gr F304L or F316L [2]
	Valves (Gate, Squib, Check)			
	Body	Forging	Stainless Steel	SA-182 Gr F304L or F316L [2]
		Casting	Stainless Steel	SA-351 Gr CF3 or CF3M
	Bolts	Bar	Low Alloy Steel	SA-193 Gr B7 or B7M
	Nuts	Bar	Low Alloy Steel	SA-194 Gr 7 or 7M
ICS				
	Condenser	Tube	Alloy Steel	SB-163 (Inconel 600)
		Header	Alloy Steel	SB-566 (Inconel 600)
	Steam Piping	Pipe	Carbon Steel	SA-333 Gr 6
	Condensate Piping	Pipe	Stainless Steel	SA-376 Gr TP316L [2] SA-312 Gr TP316L [2] SA-358 Gr TP316L [2]
SLC				
	Accumulator	Plate Forging	Low Alloy Steel	SA-533 Gr B Cl 2 SA-508 Gr 3 Cl 1
	Piping	Pipe	Stainless Steel	SA-376 Gr TP316L [2] SA-312 Gr TP316L [2] SA-358 Gr TP316L [2]

1. All carbon plate is Gr 60 or Gr 70 regardless of thickness.
2. Carbon content not to exceed 0.020% for components exposed to reactor water that exceeds 93°C (200°F) during normal plant operation.

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

6.2.1.1 Pressure Suppression Containment

Relevant to ESBWR pressure suppression containment system, this subsection addresses or references to other DCD locations that address the applicable requirements of General Design Criteria (GDC) 4, 16, 50, and 53 discussed in Standard Review Plan (SRP) 6.2.1.1.C R2. The plant meets the requirements of

(1) GDC 4, as it relates to the environmental and missile protection design, requires that safety-related structures, systems, and components be designed to accommodate the dynamic effects (e.g., effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures) that may occur during normal plant operation or following a loss-of-coolant accident;

(2) GDC 16 and 50, as they relate to the containment being designed with sufficient margin, require that the containment and its associated systems can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident; and

(3) GDC 53 as it relates to the containment design capabilities provided to ensure that the containment design permits periodic inspection, an appropriate surveillance program, and periodic testing at containment design pressure.

6.2.1.1.1 Design Bases

The pressure suppression containment system, which comprises the Drywell (DW) and Wetwell (WW) and supporting systems, is designed to meet the following Safety Design Bases:

- The containment structure shall maintain its functional integrity during and following the peak transient pressures and temperatures, which would occur following any postulated loss-of-coolant accident (LOCA). A design basis accident (DBA) is defined as the worst pipe break, which leads to maximum DW and WW pressure and/or temperature, and is postulated to occur simultaneously with loss of preferred power. For structural integrity evaluation, safe shutdown earthquake (SSE) loads are combined with LOCA loads.

- The containment structure design shall accommodate the full range of loading conditions consistent with normal plant operation, safety/relief valve (SRV) discharge and accident conditions including the LOCA related design loads.

- The containment structure is designed to accommodate the maximum internal negative pressure difference between DW and WW, and the maximum external negative pressure difference relative to the reactor building surrounding the containment.

- The containment structure and reactor building, with concurrent operation of containment isolation function (isolates all pipes or ducts which penetrate the containment boundary) and other accident mitigation systems, shall limit fission product leakage during and following the postulated DBA to values less than leakage rates which would result in off-site doses greater than those set forth in 10 CFR 50.34(a).

- The containment structure shall withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.
- The containment structure shall accommodate flooding to a sufficient depth above the active fuel to maintain core cooling and to permit safe removal of the fuel assemblies from the reactor core after the postulated DBA.
- The containment structure shall be protected from or designed to withstand hypothetical missiles from internal sources and uncontrolled motion of broken pipes, which could endanger the integrity of the containment.
- The containment structure shall direct the high energy blowdown fluids from postulated LOCA pipe ruptures in the DW to the pressure suppression pool and to the Passive Containment Cooling System (PCCS).
- The containment system shall allow for periodic tests at the calculated peak or reduced test pressure to measure the leakage from individual penetrations, isolation valves and the integrated leakage rate from the containment structure to confirm the leak-tight integrity of the containment.
- The Containment Inerting System establishes and maintains the containment atmosphere to $\leq 3\%$ by volume oxygen during normal operating conditions to ensure inert atmosphere operation.
- PCCS shall remove post-LOCA decay heat from the containment for a minimum of 72 hours, without operator action, to maintain containment pressure and temperature within design limits.

6.2.1.1.2 Design Features

The containment structure, is a reinforced concrete cylindrical structure, which encloses the reactor pressure vessel (RPV) and its related systems and components. Key containment components and design features are exhibited in Figures 6.2-1 through 6.2-5. The containment structure has an internal steel liner providing the leak-tight containment boundary. The containment is divided into a DW region and a WW region with interconnecting vent system. The functions of these regions are as follows:

- The DW region is a leak-tight gas space, surrounding the reactor pressure vessel and reactor coolant pressure boundary, which provides containment of radioactive fission products, steam and water released by a LOCA, prior to directing them to the suppression pool, via the DW/WW Vent System (a relatively small quantity of DW steam is also directed to the PCCS during the LOCA blowdown).
- The WW region consists of the suppression pool and the gas space above it. The suppression pool is a large body of water to absorb energy by condensing steam from SRV discharges and pipe break accidents. The pool is an additional source of reactor water makeup and serves as a reactor heat sink. The flow path to the WW is designed to entrain radioactive materials by routing fluids through the suppression pool during and following a LOCA. The gas space above the suppression pool is leak-tight and sized to collect and retain the DW gases following a pipe break in the DW, without exceeding the containment design pressure.

The DW/WW Vent System directs LOCA blowdown flow from the DW into the suppression pool.

The containment structure consists of the following major structural components: RPV support structure (pedestal), diaphragm floor separating DW and WW, suppression pool floor slab, containment cylindrical outer wall, cylindrical vent wall, containment top slab, and DW head. The containment cylindrical outer wall extends below the suppression pool floor slab to the common basemat. This extension is not part of containment boundary, however, it supports the upper containment cylinder. The reinforced concrete basemat foundation supports the entire containment system and extends to support the reactor building surrounding the containment.

The design parameters of the containment and the major components of the containment system are given in Tables 6.2-1 through 6.2-4. A detailed discussion of their structural design bases is given in Section 3.8.

Drywell

The DW (Figure 6.2-1) comprises two volumes: (1) an upper DW volume surrounding the upper portion of the RPV and housing the main steam and feedwater piping, Gravity Driven Cooling System (GDCS) pools and piping, PCCS piping, Isolation Condenser System (ICS) piping, SRVs and piping, depressurization valves (DPVs) and piping, DW coolers and piping, and other miscellaneous systems; and (2) a lower DW volume below the RPV support skirt housing the lower portion of the RPV, fine motion control rod drives, other miscellaneous systems and equipment below the RPV, and vessel bottom drain piping.

The upper DW is a cylindrical, reinforced concrete structure with a removable steel head and a diaphragm floor constructed of steel girders with concrete fill. The cylindrical RPV support skirt, which is connected rigidly to the RPV support structure separates the lower DW from the upper DW. There is an open communication path between the two DW volumes via upper DW to lower DW connecting vents, built into the RPV support structure. Penetrations through the liner for the DW head, equipment hatches, personnel locks, piping, electrical and instrumentation lines are provided with seals and leak-tight connections.

The DW is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the DW, and also the negative differential pressures associated with containment depressurization events, when the steam in the DW is condensed by the PCCS, the GDCS, the Fuel and Auxiliary Pools Cooling System (FAPCS), and cold water cascading from the break following post-LOCA flooding of the RPV.

For a postulated DBA, the calculated maximum DW temperature and absolute pressure remain below their design values, shown in Table 6.2-1.

A vacuum breaker system has been provided between the DW and WW. The purpose of the DW-to-WW vacuum breaker system is to protect the integrity of the diaphragm floor slab and vent wall between the DW and the WW, and the DW structure and liner, and to prevent back-flooding of the suppression pool water into the DW. The vacuum breaker is provided with redundant proximity sensors to detect its closed position. On the upstream side of the vacuum breaker, a DC solenoid operated butterfly valve designed to fail-close is provided. During a LOCA, when the vacuum breaker opens to equalize the DW and WW pressure and subsequently does not completely close as detected by the proximity sensors, a control signal will close the upstream butterfly valve to prevent extra bypass leakage due to the opening created by the vacuum breaker and therefore maintain the pressure suppression capability of the containment. Redundant vacuum breaker systems are provided to protect against a single failure. The design DW-to-WW pressure difference is given in Table 6.2-1.

A safety-related PCCS is incorporated into the design of the containment to remove decay heat from DW following a LOCA. The PCCS uses six elevated heat exchangers (condensers) located outside the containment in large pools of water at atmospheric pressure to condense steam that has been released to the DW following a LOCA. This steam is channeled to each of the condenser tube-side heat transfer surfaces where it condenses and the condensate returns by gravity flow to the GDCS pools. Noncondensable gases are purged to the suppression pool via vent lines. The PCCS condensers are an extension of the containment boundary, do not have isolation valves, and start operating immediately following a LOCA. These low pressure PCCS condensers provide a thermally efficient heat removal mechanism. No forced circulation equipment is required for operation of the PCCS. Steam produced, due to boil-off in the pools surrounding the PCCS condensers, is vented to the atmosphere. There is sufficient inventory in these pools to handle at least 72 hours of decay heat removal. The PCCS is described and discussed in detail in Subsection 6.2.2.

The containment design includes a Drywell Cooling System (DCS) to maintain DW temperatures during normal operation within acceptable limits for equipment operation as described in Subsection 9.1.8.

For the containment design leak-before-break (LBB) is considered with respect to the protection against dynamic effects associated with a postulation of rupture in high energy piping. Section 3.6.3 and Appendix 3C describe the methodology of the LBB approach for excluding design against the dynamic effects from postulation of breaks in high energy piping. Protection against the dynamic effects from the piping systems not qualified by the LBB approach is provided by the DW structure. The DW structure provides protection against the dynamic effects of plant-generated missiles (Section 3.5).

An equipment hatch for removal of equipment during maintenance and an air lock for entry of personnel are provided in both the lower and upper DW. These access openings are sealed under normal plant operation and are opened when the plant is shut down for refueling and/or maintenance.

During normal operation, the Containment Inerting System has a nitrogen makeup subsystem, which automatically supplies nitrogen to the WW and the DW to maintain a slightly positive pressure to preclude air in-leakage from the surrounding reactor building region. Before personnel can enter the DW, it is necessary to de-inert the DW atmosphere. The Containment Inerting System provides the purge supply and exhaust subsystems for de-inerting, and is discussed in Subsection 9.4.8.

Wetwell

The WW is comprised of a gas volume and suppression pool water volume. The WW is connected to the DW by a vent system comprising twelve (12) vertical/horizontal vent modules. Each module consists of a vertical flow steel pipe, with three horizontal vent pipes extending into the suppression pool water (Figures 6.2-4 and 6.2-5). Each vent module is built into the vent wall, which separates the DW from the WW (Figure 6.2-1). The WW boundary is the annular region between the vent wall and the cylindrical containment wall and is bounded above by the DW diaphragm floor. All normally wetted surfaces of the liner in the WW are stainless steel and the rest are carbon steel.

The suppression pool water is located inside the WW region. The vertical/horizontal vent system (Figures 6.2-4 and 6.2-5) connects the DW to the suppression pool.

In the event of a pipe break within the DW, the increased pressure inside the DW forces a mixture of noncondensable gases, steam and water through either the PCCS or the vertical/horizontal vent pipes and into the suppression pool where the steam is rapidly condensed. The noncondensable gases transported with the steam and water are contained in the free gas space volume of the WW.

Performance of the pressure suppression concept in condensing steam under water (during LOCA blowdown and SRV discharge) has been demonstrated by a large number of tests, as described in Appendix 3B.

The SRVs discharge steam through their discharge piping (equipped with quencher discharge device) into the suppression pool. Operation of the SRVs is intermittent, and closure of the valves with subsequent condensation of steam in the discharge piping can produce a partial vacuum, thereby drawing suppression pool water into the exhaust pipes. Vacuum relief valves are provided on the discharge piping to limit reflood water levels in the SRV discharge pipes, thus controlling the maximum SRV discharge bubble pressure resulting from a subsequent valve actuation and water clearing transient.

The WW design absolute pressure and design temperature are shown in Table 6.2-1. Table 6.2-2 shows the normal plant operating conditions for the allowed suppression pool water and WW airspace temperature.

After an accident, the nonsafety-related Fuel and Auxiliary Pools Cooling System (FAPCS) may be available in the suppression pool cooling mode and/or containment spray mode to control the containment pressure and temperature conditions. Heat is removed via the FAPCS heat exchanger(s) to the Reactor Component Cooling Water System (RCCWS) and finally to the Plant Service Water System (PSWS). The FAPCS is described in Subsection 9.1.3.

There is sufficient water volume in the suppression pool to provide adequate submergence over the top of the upper row of horizontal vents, as well as the PCCS return vent, when water level in RPV reaches at one meter above the top of active fuel and water is removed from the pool during post-LOCA equalization of pressure between RPV and the WW. Water inventory, including the GDCS, is sufficient to flood the RPV to at least one meter above the top of active fuel.

6.2.1.1.3 Design Evaluation

Summary Evaluation

The key design parameters for the containment and their calculated values under the DBA conditions are shown in Tables 6.2-1 and 6.2-5, respectively.

The evaluation of the containment design is based on the analysis of a postulated instantaneous guillotine rupture of a feedwater line. Table 6.2-6 provides the nominal and bounding values for the plant initial and operating conditions for this evaluation. This evaluation utilizes the GE computer code TRACG (Reference 6.2-1). NRC has reviewed and approved the application of

TRACG to ESBWR loss-of-coolant (LOCA) analyses, per the application methodology outlined in the report. TRACG is applicable to LOCAs covering the complete spectrum of pipe break sizes, from a small break accident to a design basis accident (DBA), and covering the entire LOCA transient including the blowdown period, the GDCS period and the long-term cooling PCCS period.

Containment Design Parameters

Tables 6.2-1 through 6.2-4 provide a listing of key design and operating parameters of the containment system, including the design characteristics of the DW, WW and the pressure suppression vent system and key assumptions used for the design basis accident analysis.

Tables 6.3-1 through 6.3-4 provide the performance parameters of the related ESF systems, which supplement the design conditions of Table 6.2-1, for containment performance evaluation. Performance parameters given in these tables include those applicable to full capacity operation and reduced capacities assumed for containment analyses.

Accident Response Analysis

The containment functional evaluation is based upon the consideration of a representative spectrum of postulated accidents, which would result in the release of reactor coolant to the containment. These accidents include:

- Liquid Breaks
  - An instantaneous guillotine rupture of a feedwater line;
  - An instantaneous guillotine rupture of a GDCS line; and
  - An instantaneous guillotine rupture of a vessel bottom drain line.
- Steam Breaks
  - An instantaneous guillotine rupture of a main steamline.

Results of these analyses show that an instantaneous guillotine rupture of a feedwater line produces the most limiting responses for the containment evaluation.

6.2.1.1.3.1 Feedwater Line Break – Nominal Analysis

This analysis initializes the RPV and containment at the base conditions shown in the Nominal Value column of Table 6.2-6. Figure 6.2-6 and 6.2-7 show the TRACG nodalization of the RPV and the containment. Its fundamental structure is an axisymmetric “VSSL” component with 42 axial levels and eight radial rings. The inner 4 rings in the first 21 axial levels represent the RPV; the outer 4 rings in these levels are not used. Axial levels 22 to 35 represent the DW, suppression pool, WW, and GDCS pools (Figure 6.2-7). Axial levels 36 to 42 represent the IC/PCC pool, expansion pools, and the Dryer/Separator Storage pool. Figure 6.2-8 shows the nodalization for the steam line system, including the SRVs and DPVs. This analysis follows the application methodology outlined in Reference 6.2-1.

Table 6.2-7 shows the sequence of events for this analysis. Figures 6.2-9 through 6.2-11 show the pressure, temperature and PCCS responses for this analysis. Table 6.2-5 summarizes the results of this calculation. The peak drywell pressure for the nominal case is below the containment design pressure.

6.2.1.1.3.2 Feedwater Line Break – Bounding Analysis

This analysis initializes the RPV and containment at the base conditions shown in the Bounding Value column of Table 6.2-2. In addition, this bounding analysis sets the other TRACG model parameters in the conservative direction as described in Reference 6.2-1. Table 6.2-8 summarizes the specific bounding values for these model parameters. This analysis follows the application methodology outlined in Reference 6.2-1.

Figures 6.2-12 through 6.2-14 show the pressure, temperature and PCCS responses for this analysis. Table 6.2-5 summarizes the results of this calculation. The peak drywell pressure for the bounding case is below the containment design pressure.

6.2.1.1.4 Negative Pressure Design Evaluation

During normal plant operation, the inerted WW and the DW volumes remain at a pressure slightly above atmospheric conditions. However, certain events could lead to a depressurization transient that can produce a negative pressure differential in the containment. A DW depressurization results in a negative pressure differential across the DW walls, vent wall, and diaphragm floor. A negative pressure differential across the DW and WW walls means that the reactor building pressure is greater than the DW and WW pressures, and a negative pressure differential across the diaphragm floor and vent wall means that the WW pressure is greater than the DW pressure. If not mitigated, the negative pressure differential can damage the containment steel liner. The ESBWR design provides the vacuum relief function necessary to limit these negative pressure differentials within design values. The events that may cause containment depressurization are:

- Post-LOCA DW depressurization caused by the ECCS (GDCS, CRD, etc.) flooding of the RPV and cold water spilling out of the broken pipe or cold water spilling out of broken GDCS line directly into DW;
- The DW sprays are inadvertently actuated during normal operation or during post-LOCA recovery period.
- The combined heat removal of the ICS and PCCS exceeds the rate of decay heat steam production.

Drywell depressurization following a LOCA is expected to produce the most severe negative pressure transient condition in the DW. The results of the Main Steam Line break analysis show that the containment does not reach negative pressure relative to the reactor building, and the maximum Wetwell-Drywell differential pressure is within the design capability.

6.2.1.1.5 Steam Bypass of Suppression Pool

6.2.1.1.5.1 Bypass leakage area in Design Basis Accident

The concept of the pressure suppression reactor containment is that any steam released from a pipe rupture in the primary system is condensed by the suppression pool, and thus, does not produce a significant pressurization effect on the containment. This is accomplished by channeling the steam into the suppression pool through a vent system. If a leakage path were to exist between the drywell and the suppression pool (wetwell) gas space, the leaking steam would produce undesirable pressurization of the containment. The bounding design basis accident calculation assumes a bypass leakage of 1 cm^2 (A/√K). Table 6.2-5 shows this results in acceptable containment pressures.

6.2.1.1.5.2 Suppression Pool Bypass During Severe Accidents

Suppression pool bypass is defined as the transport of fission products through the pathways, which do not include the suppression pool. In such cases, the scrubbing action for fission product retention is lost and the potential consequences for the release are higher. ESBWR containment design requires that the potential risk of suppression pool bypass paths, except for the WW vacuum breakers, present no significant risk following severe accidents. The vacuum breakers are considered in the ESBWR PRA. Potential bypass through the WW-DW vacuum breakers are included in the containment event trees. Isolation capability of the failed open vacuum breaker is provided to minimize the effects of the failed vacuum breakers on the suppression pool bypass.

6.2.1.1.5.3 Justification for Deviation From SRP Acceptance Criteria

6.2.1.1.5.3.1 Actuation of PCCS

The provision of automatic PCCS design meet the intent of the SRP (Appendix A to SRP Section 6.2.1.1.C) for automatic actuation of sprays, without the use of a containment spray system. The SRP states that the wetwell spray should be automatically actuated 10 minutes following a LOCA signal and an indication of pressurization of the wetwell to quench steam bypassing the suppression pool. However, in determining maximum allowable steam bypass leakage area for ESBWR design, analyses take credit for PCCS operation immediately following LOCA initiation.

The PCCS is considered adequate to provide mitigation for consequences due to steam bypass leakage during a LOCA event. There is no technical merit in changing ESBWR design to provide wetwell sprays.

6.2.1.1.5.3.2 Vacuum Valve Operability Tests

Section B.3.b of Appendix A to SRP Section 6.2.1.1.C specifies that all vacuum valves should be operability tested at monthly intervals to assure free movement of the valves. Operability tests are conducted at plants of earlier BWR designs using an air actuated cylinder attached to the valve disk. The air actuated cylinders have been found to be one of the root causes of vacuum breakers failing to close. Free movement of the vacuum breakers in the ESBWR design has been enhanced by eliminating this potential actuator failure mode, improving the valve hinge design and selecting materials which are resistant to wear and galling. Therefore, this requirement for monthly testing is deemed unnecessary for the ESBWR. However, the vacuum breakers will be tested for free movement during each outage.

6.2.1.1.5.4 Bypass Leakage Tests and Surveillance

There are a provision for leakage tests and surveillance to provide assurance that suppression pool bypass leakage is not substantially increased over the plant life. This includes both periodic low-pressure leak tests, a pre-operational high-pressure leak test, and a periodic visual inspection.

6.2.1.1.5.4.1 High-Pressure Leak Test

A single pre-operational high-pressure bypass leakage test will be performed. The purpose of this test is to detect leakage from the drywell to wetwell. This test will be performed at approximately the peak drywell to wetwell differential pressure, and will follow the high-pressure structural test of the diaphragm floor. The acceptance criteria are specified in Subsection 6.2.1.1.5.7.3.

6.2.1.1.5.4.2 Low-Pressure Leak Test

A post-operational low-pressure leakage test is performed to detect leakage from the drywell to wetwell. This test is performed at each refueling outage at a differential pressure corresponding to approximately, but less than, the submergence of the top horizontal vents. The acceptance criteria are specified in Subsection 6.2.1.1.5.7.3.

6.2.1.1.5.4.3 Acceptance Criteria for Leakage Tests

The acceptance criteria for both high- and low-pressure leakage tests shall be a measured bypass leakage area, which is less than 10% of the suppression pool steam bypass capability discussed in Subsection 6.2.1.1.5.4.

6.2.1.1.5.4.4 Surveillance Test

A visual inspection will be conducted to detect possible leak paths at each refueling outage. Each vacuum relief valve and associated piping will be checked to determine that it is clear of foreign matter. Also, at this time each vacuum breaker will be tested for free disk movement.

6.2.1.1.5.5 Vacuum Relief Valve Instrumentation and Tests

6.2.1.1.5.5.1 Position Indicators and Alarms

Redundant position indicators are placed on all vacuum breakers with redundant indication and an alarm in the control room. The vacuum breaker position indicator system is designed to provide the plant operators with continuous surveillance of the vacuum breaker position. The vacuum relief valve position indicator system has adequate sensitivity to detect a total valve opening, for all valves, that is less than the design bypass capability, discussed in Subsection 6.2.1.1.5.4. The detectable valve opening is determined by the actual value of the pressure loss coefficient, K, and is based on the assumption that the valve opening is evenly divided among all the vacuum breakers.

6.2.1.1.5.5.2 Vacuum Valves Operability Tests

The vacuum relief valves will be tested for free movement during each refueling outage.

6.2.1.1.6 Suppression Pool Dynamic Loads

During a postulated LOCA, DW-to-WW flow of gas and steam/water mixture produces hydrodynamic loading conditions on the suppression pool boundary. Also, SRV flow discharging into the suppression pool during SRV actuation produces hydrodynamic loading conditions on the pool boundary.

The containment and its internal structures are designed to withstand all suppression pool dynamic loads, due to LOCA and SRV actuation events in combination with those from the postulated seismic events. The load combinations are described and specified in Section 3.8.

A complete description of and diagrammatic representation of these loads is provided in Appendix 3B.

6.2.1.1.7 Asymmetric Loading Conditions

Asymmetric loads are included in the load combination specified in Subsection 3.8. The containment and internal structures are designed for these loads within the acceptance criteria specified in Subsection 3.8.

Localized pipe forces and SRV actuation would lead to asymmetric pressure loads on the containment and internal structures. For magnitudes of these loads, see Appendix 3B.

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

6.2.1.1.8 Containment Environment Control

The Drywell Cooling System (DCS) function, which is to maintain the thermal conditions in the containment and subcompartments during the normal operation, is not a safety-related function. Also the loss of the DCS does not result in environmental conditions that exceed the expected design basis accident conditions for the safety-related equipment inside containment. Therefore, the DCS is not classified as safety-related. The safety-related containment heat removal systems, described in Section 6.2.2, maintain the required containment atmosphere conditions following a LOCA.

6.2.1.1.9 Post-Accident Monitoring

Subsection 6.2.1.7 identifies instrumentation provided for post-accident monitoring of containment parameters. For discussion of instrumentation inside the containment, which may be used for monitoring various containment parameters during post-accident conditions, refer to Section 7.5.

6.2.1.1.10 Severe Accident Conditions

Severe accident (SA) considerations are in the design of the ESBWR. The ESBWR design philosophy is to continue to maintain design flexibility in order to allow for potential modifications.

This section reviews the design approach and proposed ESBWR design features for the prevention and mitigation of SAs.

6.2.1.1.10.1 Layered Defense-in-Depth Approach

The ESBWR utilizes the concept of defense-in-depth as a basic design philosophy. This is an approach that relies on providing numerous barriers. These barriers include both physical barriers (e.g., fuel pellet, fuel cladding, reactor vessel and ultimately the containment), as well as layers that emphasize accident prevention and accident mitigation. The ESBWR considers beyond design basis events in its design approach. It provides for additional defense-in-depth by considering a broad range of events, including those with very low estimated frequency of occurrence (< 1.0E-5 per reactor year) and by incorporating design features to mitigate significant containment challenges.

Using this layered defense-in-depth approach, the following are the main elements in the design against severe accidents:

- Accident prevention
- Accident mitigation
- Containment performance including design features to address containment challenges during a severe accident

6.2.1.1.10.2 ESBWR Design Features for Severe Accident Control

Several features are designed into the ESBWR that serve either to prevent or mitigate the consequences of a severe accident. Key ESBWR features, their design intent, and the corresponding issues are summarized in Table 6.2-9. For each feature listed in Table 6.2-9, brief discussion is made below.

(1) Isolation Condenser System (ICS)

The Isolation Condensers (ICs) are the first defense against a SA. The ESBWR is equipped with four ICs, which conserve RPV inventory in the event of RPV isolation. Basically, the ICs take steam from the RPV and return condensate back to the RPV. The ICs begin operation when the condensate lines open automatically on diverse signals including RPV level dropping to Level 2. After operation begins, the ICs are capable of keeping the RPV level above Level 1, the setpoint for Automatic Depressurization System (ADS) actuation. The design mitigates noncondensable buildup in the ICs (that can impair heat removal capacity) by temporarily opening a small vent line connecting the ICs to the suppression pool. The vent line is operated automatically when high RPV pressure is maintained for more than a set time. The vent line valves re-close automatically when RPV pressure is decreased below the setpoint pressure.

In the event of a break in the primary system or ADS actuation, the RPV depressurizes. The ICs are not required to prevent containment pressurization. That function is served by the Passive Containment Cooling System (PCCS).

(2) Automatic Depression System

The ESBWR reactor vessel is designed with a highly reliable depressurization system. This system plays a major role in preventing core damage. Furthermore, even in the event of core damage, the depressurization system can minimize the potential for high-pressure melt ejection and lessen the resulting challenges to containment integrity. If the reactor vessel fails at elevated pressure, fragmented core debris could be transported into the upper drywell. The resulting heatup of the upper drywell atmosphere could overpressurize the containment or cause over temperature failure of the drywell head seals. The RPV depressurization system decreases the uncertainties associated with this failure mechanism by minimizing the occurrences of high pressure melt ejection.

(3) Compact Containment Design

The reactor building volume is reduced by relocating selected equipment and systems to areas outside of the reactor building. The major portion of this relocation is to remove non-safety items from the Seismic Class 1 structure to other structures that are classified as Non-Seismic. Along with other simplified system design and relocation of non-safety items, a compact containment design is achieved with minimum penetrations. This reduces the leakage from the containment.

(4) PCC Heat Exchangers

The basic design of the ESBWR ensures that any fission products that are generated following an accident are not released outside the plant. One such removal mechanism is the PCC heat exchanger tubes. These tubes act like a filter for the aerosols. They essentially ‘filter out’ any aerosols that are transported into the PCC units along with the steam and non-condensable gas flow. Aerosols that are not retained, in the drywell or the PCC heat exchangers, get transported via the PCCS vent line to the suppression pool where they are efficiently scrubbed.

The PCC heat exchanger not only cools the containment by removing decay heat during accident, but also provides fission product retention within the containment.

(5) Lower Drywell Configuration

The floor area of the lower drywell has been maximized to improve the potential for ex-vessel debris cooling.

(6) Manual Containment Overpressure Protection Subsystem (MCOPS)

The ESBWR design includes vents from the suppression chamber air space consisting of the containment over-pressurization protection system and connected to the rooms directly below the suppression pool.

In the event that containment heat removal fails or core-concrete interaction continues unabated, the Containment Inerting System lines are used to manually vent the containment to control pressure, preventing the overpressure failure of containment.

(7) Deluge Lines Flooder System

The lower drywell deluged lines flooder system has been included in the ESBWR to provide automatic cavity flooding in the event of core debris discharge from the reactor vessel. This system is actuated on high lower drywell floor temperature. The system consists of three lines that connect each of the GDCS water pools to the drywell connecting vents. The volume of water in the GDCS pools is capable of flooding the RPV and lower drywell to the top of active fuel.

The two deluged flooder lines from the GDCS pools provide sufficient water to quench all core debris. The deluged lines originating from the GDCS provide water to the BiMAC device embedded into the lower drywell floor to cool the ex-vessel core-melt debris from top and bottom sides. By flooding the lower drywell after the introduction of core material, the potential for energetic fuel-coolant interaction is minimized. Additionally, covering core debris provides for debris cooling and scrubbing of fission products released from the debris due to core-concrete interaction. From an overall containment performance point of view, the flooder provides a significant benefit for accident mitigation.

(8) Passive Containment Cooling System (PCCS)

The PCCS system is designed to remove decay heat from the containment. The PCC heat exchangers receive a steam-gas mixture from the drywell atmosphere, condense the steam and return the condensate to the RPV via an intermediate holding tank. The non-condensable gas is drawn to the suppression pool through a submerged vent line driven by the differential pressure between the drywell and wetwell.

(9) Suppression Pool and Airspace

The suppression chamber is a large chamber with communication to the drywell through the horizontal vents, the PCCS vents and the vacuum breakers. Approximately one-half of the suppression chamber volume is filled with a large body of water, the suppression pool. The gas space in the suppression chamber acts as a receiver for noncondensable gases during a severe accident. The suppression pool plays a large role in containment performance because it provides:

- A large containment heat sink.
- Quenching of steam, which flows through the horizontal vents during rapid increases in drywell pressure.
- Effective scrubbing of fission products, which flow through the horizontal vents and the PCCS vents.

(10) GDCS in Wetwell Configuration

The GDCS pools are placed above the reactor pressure vessel with their air space connected to the wetwell. This connection effectively increases the wetwell air space and provides a larger volume for noncondensable gases produced during a severe accident. Once the GDCS pools are drained, the total volume of the GDCS pools are added to the volume of the wetwell airspace.

A line with normally closed valves connects the GDCS pool to the vessel downcomer for low pressure injection. After the GDCS pools are exhausted following LOCA injection, coolant flow to keep the core covered is supplied from the suppression pool through an equalizing line, which branches from the GDCS line.

(11) Inerted Containment

During a severe accident, gases are generated that could form a combustible mixture if oxygen were present. Combustion of these gases would increase the containment temperature and pressure, possibly resulting in structural damage. To avoid this potential challenge to containment integrity, the ESBWR containment is inerted during operation.

Figure 6.2-15 summarizes all of the above systems in the framework of the ESBWR containment. From top down:

(1) PCCS pool and heat exchangers provide passive containment cooling;

(2) ICS pool and heat exchangers provide natural circulation decay heat removal from RPV;

(3) GDCS (three pools, four divisions) with ADS (DPV, SRV) makes up the ECCS; GDCS deluge line supplies BiMAC for long-term coolability;

(4) MCOPS provides manual venting from the wetwell in a controlled manner; and

(5) Basemat-Internal Melt Arrest and Coolability (BiMAC) device, commonly call a core catcher, (shown by the insert to Figure 6.2-15) is initially fed by water flow from squib-valve-operated GDCS deluge lines into a distributor channel, and through a pipe jacket (with inclined and vertical portions) into the LDW cavity. The cooling in a later phase is provided by natural circulation of water in the LDW feeding into the distributor channel through downcomers (at the end of LDW, not shown in the insert).

6.2.1.2 Containment Subcompartments

This subsection addresses or references to other DCD locations that address the applicable requirements of GDC 4 and 50 discussed in SRP 6.2.1.2 R2 relevent to ESBWR containment subcompartment design. The plant meets the requirements of

- GDC 4, as it relates to the environmental and missile protection provided to ensure that safety-related structures, systems and components be designed to accommodate the dynamic effects (e.g., effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures) that may occur during plant normal operations or during an accident; and
- GDC 50, as it relates to the subcompartments being designed with sufficient margin to prevent fracture of the structure due to pressure differential across the walls of the subcompartment. In meeting the requirements of GDC 50, the following specific criterion or criteria that pertain to the design and functional capability of containment subcompartments are used as indicated below.
  - The initial atmospheric conditions within a subcompartment are selected to maximize the resultant differential pressure. The model assumes air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity. For a restricted class of subcompartments, another model is used that involves simplifying the air model outlined above. For this model, the initial atmosphere within the subcompartment is modeled as a homogeneous water-steam mixture with an average density equivalent to the dry air model. This approach is limited to subcompartments that have choked flow within the vents. This simplified model is not used for subcompartments having primarily subsonic flow through the vents.
  - Subcompartment nodalization schemes are chosen such that there is no substantial pressure gradient within a node, i.e., the nodalization scheme is verified by a sensitivity study that includes increasing the number of nodes until the peak calculated pressures converge to small resultant changes. The guidelines of Section 3.2 of NUREG-0609 are followed, and a nodalization sensitivity study is performed which includes consideration of spatial pressure variation, e.g., pressure variations circumferentially, axially and radially within the subcompartment, for use in calculating the transient forces and moments acting on components.
  - When vent flow paths are used which are not immediately available at the time of pipe rupture, the following criteria apply:
    - The vent area and resistance as a function of time after the break are based on a dynamic analysis of the subcompartment pressure response to pipe ruptures.

- The validity of the analysis is supported by experimental data or a testing program that supports this analysis.
- In meeting the requirements of GDC 4, the effects of missiles that may be generated during the transient are considered in the safety analysis.

– The vent flow behavior through all flow paths within the nodalized compartment model is based on a homogeneous mixture in thermal equilibrium, with the assumption of 100% water entrainment. In addition, the selected vent critical flow correlation is conservative with respect to available experimental data. An acceptable vent critical flow correlations are the “frictionless Moody” with a multiplier of 0.6 for water-steam mixtures, and the thermal homogeneous equilibrium model for air-steam-water mixtures.

– During the construction permit stage, a factor of 1.4 is applied to the peak differential pressure calculated for the subcompartment, structure and the enclosed components, for use in the design of the structure and the component supports. At the operating license (OL) stage, the peak calculated differential pressure shall not exceed the design pressure. It is expected that the post-OL peak calculated differential pressure would not be substantially different from that of the construction permit stage. However, improvements in the analytical models or changes in the as-built subcompartment may affect the available margin.

6.2.1.2.1 Design Bases

The design of the containment subcompartments is based upon a postulated DBA occurring in each subcompartment.

For each containment subcompartment in which high energy lines are routed, mass and energy release data corresponding to a postulated double ended line break are calculated. The mass and energy release data, subcompartment free volumes, vent path geometry and vent loss coefficients are used as input to an analysis to obtain the pressure/temperature transient response for each subcompartment.

6.2.1.2.2 Design Features

The DW and WW subcompartments are described in Subsection 6.2.1.1. The remaining containment subcompartments as follows.

Drywell Head Region

The DW head region is covered with a removable steel head, which forms part of the containment boundary. The DW bulkhead connects the containment vessel flange to the containment and represents the interface between the DW head region and the DW. There are no high energy lines in the DW head region.

Reactor Shield Annulus

The reactor shield annulus exists between the reactor shield wall (RSW) and the RPV. The RSW is a steel cylinder surrounding the RPV and extending up close to the DW top slab, as shown in Figure 6.2-2. The opening between the RSW and the DW top slab provides the vent pathway necessary to limit pressurization of the annulus due to a high energy pipe rupture inside the annulus region. The shield wall is supported by the reactor support structure.

Several high energy lines extend from the RPV through the reactor shield wall. There are also penetrations in the RSW for other piping, vents, and instrumentation lines. The reactor shield wall is designed for transient pressure loading conditions from the worst high energy line rupture inside the annulus region.

6.2.1.2.3 Design Evaluation

The peak subcompartment pressure responses were found to be below the design pressure for all postulated pipe break accidents.

6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents

Relevant to mass and energy analyses, this subsection addresses or references to other DCD locations that address the applicable requirements of GDC 50 and 10 CFR Part 50, Appendix K, paragraph I.A discussed in SRP 6.2.1.3 R1. The plant meets the requirements of

- GDC 50, as it relates to the containment being designed with sufficient margin, requires that the containment and its associated systems can accommodate, without exceeding the design leakage rate, and the containment design can withstand the calculated pressure and temperature conditions resulting from any loss-of-coolant accident; and
- 10 CFR 50, Appendix K, as it relates to sources of energy during the LOCA, provides requirements to assure that all the energy sources have been considered.

In meeting the requirements of GDC 50 the following criteria, which pertain to the mass and energy analyses, are used.

- Sources of Energy
  - The sources of stored and generated energy that are considered in analyses of LOCAs include reactor power, decay heat, stored energy in the core and stored energy in the reactor coolant system metal, including the reactor vessel and reactor vessel internals and metal-water reaction energy.
  - Calculations of the energy available for release from the above sources are done in general accordance with the requirements of 10 CFR 50, Appendix K, paragraph I.A. However, additional conservatism is included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA.
  - The requirements of paragraph I.B in Appendix K, concerning the prediction of fuel cladding swelling and rupture are not considered, to maximize the energy available for release from the core.
- Break Size and Location
  - The choice of break locations and types is discussed in Subsection 6.2.
  - Of several breaks postulated on the basis of a., above, the break selected as the reference case yields the highest containment pressure consistent with the criteria for establishing the break location and area.

– Containment design basis calculations are performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified.

- Calculations

In general, calculations of the mass and energy release rates for a LOCA are performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure).

A spectrum of breaks was considered and analyzed using GE-developed and approved computer codes described in Reference 6.2-1.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment (PWR)

Not Applicable to the ESBWR.

6.2.1.5 Maximum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System (PWR)

Not Applicable to the ESBWR.

6.2.1.6 Testing and Inspection

Preoperational Testing

Preoperational testing and inspection programs for the containment and associated structures, systems and components are described in Chapter 14. These programs demonstrate the structural integrity and desired leak-tightness of the containment and associated structures, systems, and components.

Post-Operational Leakage Rate Test

For descriptions of the containment integrated leak rate test (ILRT) and other post-operational leakage rate tests (10 CFR 50, Appendix J, Test Types A and B); see Subsection 6.2.6.

Accessible portions of the vacuum breaker system will be visually inspected at each refueling outage to determine and assure that they are free of foreign debris and the valve disk will be manually tested for its freedom to move and functionality.

Design Provisions for Periodic Pressurization

In order to assure the structural capability of the containment to withstand the application of peak accident pressure at any time during plant life, and to pass periodic integrated leakage rate tests, close attention is given to certain design and maintenance provisions. Specifically, the effects of corrosion on the structural integrity of the containment have been minimized by the use of stainless steel liner in the suppression pool area. Other design features, which have the potential to deteriorate with age, such as flexible seals, will be inspected and tested. In this manner, the structural and leak integrity of the containment remains essentially the same as originally accepted.

6.2.1.7 Instrumentation Requirements

Instrumentation is provided to monitor the following containment parameters:

- Drywell (DW) temperature;
- Drywell pressure;
- Differential pressure from DW-to-Wetwell and DW-to-Reactor Building;
- Drywell oxygen and hydrogen concentrations;
- Drywell radiation levels;
- Wetwell (WW) temperature;
- Wetwell pressure;
- Differential pressure between the Wetwell and Reactor Building;
- Wetwell oxygen and hydrogen concentrations;
- Wetwell radiation levels;
- Suppression pool temperature;
- Suppression pool level;
- GDCS pools water level;
- Water level in Drywell;
- Drywell and Wetwell nitrogen makeup flow; and
- Open/close position indicators for WW-to-DW vacuum breakers.

DW pressure is an input signal to containment isolation and Reactor Protection System (RPS). Suppression pool temperature is an input to RPS and suppression pool cooling initiation logic. Pressure indicators are also provided to monitor both the Drywell and Wetwell as part of the Containment Monitoring System that maintains containment pressure above the reactor building pressure.

DW-to-WW differential pressure is monitored to assure proper functioning of the WW-to- DW vacuum breaker system.

DW spatial temperatures are input signals to the Leak Detection and Isolation System (LD&IS). Thermocouples are mounted at appropriate elevations of the DW for monitoring the DW temperatures. Temperature, pressure and radiation are monitored for environmental conditions of equipment in the containment during normal, abnormal and accident conditions.

Suppression pool-level sensors are provided in the suppression pool water for hi-lo level alarms. Suppression pool temperature readouts from the immersed temperature sensors are located and alarmed in the control room. The sensors are used for normal indications, scram signal, and for post-LOCA pool monitoring.

Oxygen and hydrogen analyzers are provided for the Drywell and Wetwell. Each analyzer draws a sample from an appropriate area of the Drywell or Wetwell. High oxygen and hydrogen concentration levels are recorded and alarmed in the control room.

Radiation detectors in the Drywell and Wetwell areas provide inputs to radiation monitors, and radiation levels are recorded and alarmed on high level.

Refer to Section 7.2 for a description of Drywell pressure as an input to the RPS, and Section 7.3 for a description of containment parameters as input signals to the ESF systems. The display instrumentation for all containment parameters, including the number of channels, recording of parameters, instrument range and accuracy and post-accident monitoring equipment is discussed in Section 7.5.

6.2.2 Passive Containment Cooling System

Relevant to containment heat removal, this subsection addresses (or references to other DCD locations that address) the applicable requirements of GDC 38, 39 and 40 discussed in SRP 6.2.2, R4. The plant meets the following containment cooling requirements.

- GDC 38 as it relates to:
  - The Passive Containment Cooling System (PCCS) being capable of reducing the containment pressure and temperature following a LOCA, and maintaining them at acceptably low levels;
  - The PCCS performance being consistent with the function of other systems;
  - The PCCS being a safety-grade design; i.e., having suitable redundancy of components and features, and interconnections, that ensures that for a loss of onsite power, the system function can be accomplished assuming a single failure; and
  - Leak detection, isolation and containment capabilities being incorporated in the design of the PCCS.
- GDC 39, as the PCCS is designed to permit periodic inspection of components.
- GDC 40, as the PCCS is designed to permit periodic testing to assure system integrity, and operability of the system and its active components.[AS66]

6.2.2.1 Design Basis

Functions

The Passive Containment Cooling System (PCCS) removes the core decay heat rejected to the containment after a LOCA. It provides containment cooling for a minimum of 72 hours post-LOCA, with containment pressure never exceeding its design pressure limit, and with the Isolation Condenser/Passive Containment Cooling (IC/PCC) pool inventory not being replenished.

The PCCS is an engineered safety feature (ESF), and therefore a safety-related system.

General System Level Requirements

The PCCS condenser is sized to maintain the containment within its pressure limits for design basis accidents (DBAs). The PCCS is designed as a passive system without power actuated valves or other components that must actively function. Also, it is constructed of stainless steel to design pressure, temperature and environmental conditions that equal or exceed the upper limits of containment system reference severe accident capability.

Performance Requirements

The PCCS consists of six PCCS condensers. Each PCCS condenser is made of two identical modules and each entire PCCS condenser two-module assembly is designed for 11 MWt capacity, nominal, at the following conditions:

- Pure saturated steam in the tubes at 308 kPa absolute (45 psia) and 134°C (273°F); and
- Pool water temperature at atmospheric pressure and 101°C (214°F).

Design Pressure and Temperature

The PCCS design pressure and temperature are provided in Table 6.2-10.

The PCCS condenser is in a closed loop extension of the containment pressure boundary. Therefore, ASME Code Section III Class 2, Seismic Category I, and TEMA Class R apply. Material is nuclear grade stainless steel or other material, which is not susceptible to intergranular stress corrosion cracking (IGSCC).

6.2.2.2 System Description

6.2.2.2.1 Summary Description

The PCCS consists of six totally independent closed loop extensions of the containment. Each loop contains a heat exchanger (PCCS condenser) that condenses steam on the tube side and transfers heat to water in a large pool, which is vented to atmosphere.

The PCCS operates by natural circulation. Its operation is initiated by the difference in pressure between the Drywell and the Wetwell, which are parts of the ESBWR pressure suppression type containment system.

The PCCS condenser, which is open to the containment, receives a steam-gas mixture supply directly from the Drywell. The condensed steam is drained to a GDCS pool and the gas is vented through the vent line, which is submerged in the pressure suppression pool.

The PCCS loop does not have valves, so the system is always available.

6.2.2.2.2 Detailed System Description

The PCCS maintains the containment within its pressure limits for DBAs. The system is designed as a passive system with no components that must actively function, and it is also designed for conditions that equal or exceed the upper limits of containment reference severe accident capability.

The PCCS consists of six, low-pressure, totally independent loops, each containing a steam condenser (Passive Containment Cooling Condenser), as shown Figure 6.2-16. Each PCCS condenser loop is designed for 11 MWt capacity and is made of two identical modules. Together with the pressure suppression containment (Subsection 6.2.1.1), the PCCS condensers limit containment pressure to less than its design pressure for at least 72 hours after a LOCA without makeup to the IC/PCC pool, and beyond 72 hours with pool makeup.

The PCCS condensers are located in a large pool (IC/PCC pool) positioned above, and outside, the ESBWR containment (DW).

Each PCCS condenser is configured (see Figure 6.2-16) as follows.

A central steam supply pipe is provided which is open to the containment at its lower end, and it feeds two horizontal headers through two branch pipes at its upper end. Steam is condensed inside vertical tubes and the condensate is collected in two lower headers.

The vent and drain lines from each lower header are routed to the DW through a single containment penetration per condenser module as shown on the diagram.

The condensate drains into an annular duct around the vent pipe and then flows in a line that connects to a large common drain line, which also receives flow from the other header.

The PCCS loops receive a steam-gas mixture supply directly from the DW. The PCCS loops are initially driven by the pressure difference created between the containment DW and the suppression pool during a LOCA and then by gravity drainage of steam condensed in the tubes, so they require no sensing, control, logic or power-actuated devices to function. The PCCS loops are an extension of the safety-related containment and do not have isolation valves.

Spectacle flanges are included in the drain line and in the vent line to conduct post-maintenance leakage tests separately from Type A containment leakage tests.

Located on the drain line and submerged in the GDCS pool, just upstream of the discharge point, is a loop seal: it prevents back-flow of steam and gas mixture from the DW to the vent line, which would otherwise short circuit the flow through the PCCS condenser to the vent line. It also provides long-term operational assurance that the PCCS condenser is fed via the steam supply line.

Each PCCS condenser is located in a subcompartment of the IC/PCC pool, and all pool subcompartments communicate at their lower ends to enable full use of the collective water inventory independent of the operational status of any given IC/PCCS sub-loop.

A valve is provided at the bottom of each PCC subcompartment that can be closed so the subcompartment can be emptied of water to allow PCCS condenser maintenance.

Pool water can heat up to about 101°C (214°F); steam formed, being non-radioactive and having a slight positive pressure relative to station ambient, vents from the steam space above each PCCS condenser where it is released to the atmosphere through large-diameter discharge vents.

A moisture separator is installed at the entrance to the discharge vent lines to preclude excessive moisture carryover and loss of IC/PCC pool water.

IC/PCC pool makeup clean water supply for replenishing level is normally provided from the Makeup Water System (Subsection 9.2.3).

Level control is accomplished by using an air-operated valve in the make-up water supply line. The valve opening and closing is controlled by water level signal sent by a level transmitter sensing water level in the IC/PCC pool.

Cooling and cleanup of IC/PCC pool water is performed by the Fuel and Auxiliary Pools Cooling System (FAPCS) (Section 9.1.3).

The FAPCS provides safety-related dedicated makeup piping, independent of any other piping, which provides an attachment connection at grade elevation in the station yard outside the reactor building, whereby a post-LOCA water supply can be connected.

6.2.2.2.3 System Operation

Normal Plant Operation

During normal plant operation, the PCCS loops are in "ready standby."

Plant Shutdown Operation

During refueling, the PCCS condenser maintenance can be performed, after closing the locked open valve, which connects the PCCS pool subcompartment to the common parts of the IC/PCC pool, and drying the individual partitioned PCCS pool subcompartment.

Passive Containment Cooling Operation

The PCCS receive a steam-gas mixture supply directly from the DW; it does not have any valve, so it immediately starts into operation, following a LOCA event. Noncondensables, together with steam vapor, enter the PCCS condenser; steam is condensed inside PCCS condenser vertical tubes, and the condensate, which is collected in the lower headers, is discharged to the GDCS pool. The noncondensables are purged to the Wetwell through the vent line.

6.2.2.3 Design Evaluation

The PCCS condenser is an extension of the containment (DW) pressure boundary and it is used to mitigate the consequences of an accident. This function classifies it as a safety-related ESF. ASME Code Section III, Class 2 and Section XI requirements for design and accessibility of welds for inservice inspection apply to meet 10 CFR 50, Appendix A, Criterion 16. Quality Group B requirements apply per RG 1.26. The system is designed to Seismic Category I per RG 1.29. The common cooling pool that PCCS condensers share with the ICs of the Isolation Condenser System is a safety-related ESF, and it is designed such that no locally generated force (such as an IC system rupture) can destroy its function. Protection requirements against mechanical damage, fire and flood apply to the common IC/PCC pool.

As protection from missile, tornado and wind, the PCCS parts outside the containment are located in a subcompartment of the safety-related IC/PCC pool to comply with 10 CFR 50, Appendix A, Criteria 2 & 4.

The PCCS condenser can not fail in a manner that damages the safety-related ICS/PCC pool because it is designed to withstand induced dynamic loads, which are caused by combined seismic, DPV/SRV or LOCA conditions in addition to PCCS operating loads.

In conjunction with the pressure suppression containment (Subsection 6.2.1.1), the PCCS is designed to remove heat from the containment to comply with 10 CFR 50, Appendix A, Criterion 38. Provisions for inspection and testing of the PCCS are in accordance with Criteria 39, 52 & 53. Criterion 51 is satisfied by using nonferritic stainless steel in the design of the PCCS.

The intent of Criterion 40, testing of containment heat removal system is satisfied as follows:

- The structural and leak-tight integrity can be tested by periodic pressure testing.
- Functional and operability testing is not needed because there are no active components of the system.

- Performance testing during in-plant service is not feasible; however, the performance capability of the PCCS was proven by full-scale PCCS condenser prototype tests at a test facility before their application to the plant containment system design. Performance is established for the range of in-containment environmental conditions following a LOCA. Integrated containment cooling tests have been completed on a full-height reduced-section test facility, and the results have been correlated with TRACG computer program analytical predictions; this computer program is used to show acceptable containment performance, which is reported in Subsection 6.2.1.1 and Chapter 15.

6.2.2.4 Testing and Inspection Requirements

The PCCS is an extension of the containment, and it will be periodically pressure tested as part of overall containment pressure testing (Section 6.2.6). Also, the PCCS loops can be isolated for individual pressure testing during maintenance.

If additional inservice inspection becomes necessary, it is unnecessary to remove the PCCS condenser because ultrasonic testing of tube-to-header welds and eddy current testing of tubes can be done with the PCCS condensers in place during refueling outages.

6.2.2.5 Instrumentation Requirements

The PCCS does not have instrumentation that is separate from the Containment System. Control logic is not needed for its functioning. There are no sensing and power actuated devices. Containment System instrumentation is described in Subsection 6.2.1.7.

6.2.3 Reactor Building Functional Design

Relevant to the function of a secondary containment design, this subsection addresses (or references to other DCD locations that address) the applicable requirements of GDC 4, 16, and 43 and Appendix J to 10 CFR 50 discussed in SRP 6.2.3 R2. The plant meets the relevant and applicable requirements of:

- GDC 4 as it relates to safety-related structures, systems and components being designed to accommodate the effects of normal operation, maintenance, testing and postulated accidents, and being protected against dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures.
- GDC 16 as it relates to reactor containment and associated systems being provided to establish essentially leak-tight barriers against the uncontrolled release of radioactive material to the environment.
- GCD 43 as it relates to atmosphere cleanup systems having the design capability to permit periodic functional testing to ensure system integrity, the operability of active components, and the operability of the system as a whole and the performance of the operational sequence that brings the system into operation.
- 10 CFR 50, Appendix J as it relates to the secondary containment being designed to permit preoperational and periodic leakage rate testing so that bypass leakage paths are identified.

This subsection applies to the ESBWR Reactor Building (RB) design. The RB structure encloses all penetrations through the containment (except for those of the main steam tunnel and IC/PCC pools). The RB:

- provides an added barrier to fission product released from the containment in case of an accident;
- contains, dilutes, and holds up any leakage from the containment; and
- houses safety-related systems.

The RB under accident conditions is automatically isolated to provide a hold up and plate out barrier. When isolated, the RB can be serviced by the Reactor Building HVAC system through a HEPA filtration system (Refer to Subsection 9.4.6). With low leakage and stagnant conditions, hold up and plate out mechanisms perform the basic mitigating functions. The ESBWR design does not include a secondary containment and minimal credit is taken for the existence of the RB surrounding the primary containment vessel in any radiological analyses. The radiological dose consequences for LOCAs, based on an assumed containment leak rate of 0.5% per day and RB bypass leakage, equal to 100% of the containment leak rate, show that off-site and control room doses after an accident are less than allowable limits, as discussed in Chapter 15. The RB envelope is not intended to provide a leak-tight barrier against radiological releases. Therefore, the design criterion of GDC 16 does not apply.

During normal plant operation, potentially contaminated areas within the Reactor Building are kept at a negative pressure with respect to the environment while clean areas are maintained at positive pressure. The ESBWR does not need, and thus has no filter system that performs a safety-related function following a design basis accident, as discussed in Subsection 6.5.1. Therefore the design criterion of GDC 43 is not applicable.

Personnel and equipment entrances to the RB consist of vestibules with interlocked doors and hatches. Large equipment access is by means of a dedicated, external access tower that provides the necessary interlocks.

6.2.3.1 Design Bases

The Reactor Building is designed to meet the following safety design bases:

- The RB maintains its integrity during the environmental conditions postulated for a DBA.
- The RB HVAC system automatically isolates upon detection of high radiation levels in the ventilation exhaust system.
- All openings through the RB boundary, such as personnel and equipment doors, are closed during normal operation and after a DBA by interlocks or administrative control. These doors are provided with position indicators and alarms which are monitored in the control room.
- Detection and isolation capability for high-energy pipe breaks within the RB is provided.
- The compartments within the RB are designed to withstand the maximum pressure due to a high-energy line break (HELB). Each line break analyzed is a double-ended break. In this analysis, the rupture producing the greatest blowdown of mass and enthalpy in

conjunction with worst-case single active component failure is considered. Blowout panels between compartments provide flow paths to relieve pressure.

- The RB is capable of periodic testing to assure that the leakage rates assumed in the radiological analyses are met.

6.2.3.2 Design Description

The Reactor Building is a reinforced concrete structure that forms an envelope completely surrounding the containment (except the basemat). The boundary of the clean areas and the Reactor Building are shown in Figure 6.2-17.

During normal operation, the Reactor Building potentially contaminated areas are maintained at a slightly negative pressure relative to adjoining areas by the Contaminated Area HVAC Sub-system (CONAVS) portion of the Reactor Building HVAC system (Section 9.4.6). This assures that any leakage from these areas is collected and treated before release. Airflow is from clean to potentially contaminated areas. Reactor Building effluents are monitored for radioactivity by stack radiation monitors. If the radioactivity level rises above set levels, the discharge can be routed through CONAVS purge system for treatment before further release.

Penetrations through the RB envelope are designed to minimize leakage. All piping and electrical penetrations are sealed for leakage. Access to the Reactor Building is through interlocked doors. The Reactor Building HVAC system is designed and tested for isolation under accident conditions.

High-energy pipe breaks (HELB) in any of the Reactor Building compartments do not require the building to be isolated. These breaks are detected and the broken pipe is isolated by the closure of system isolation valves (Section 7.3.3). There is no significant release of radioactivity postulated from these types of accidents because reactor fuel is not damaged.

The following paragraphs are brief descriptions of the major compartments in the ESBWR design.

Reactor Water Cleanup (RWCU) Equipment and Valve Rooms

The two independent RWCU divisions are located in the 0–90° and 270-0° quadrants of the Reactor Building. The RWCU equipment (pumps, heat exchangers, and filter/demineralizers) is located on floor elevations -11500 mm and -6400 mm with separate rooms for equipment and valves. The RWCU piping originates at the reactor pressure vessel. High energy piping leads to the RWCU divisions through a dedicated, enclosed, pipe chase. The steam/air mixture resulting from a high energy line break in any RWCU compartment is directed through adjoining compartments and pipe chase to the Reactor Building operating floor. Figure 6.2-18 shows the model of the Reactor Building compartments with the interconnecting flow paths for a typical analysis. The design basis break for the RWCU system compartment network is a double-ended break. The selected break cases are identified in Table 6.2-11.

Isolation Condenser (IC) System

The isolation condensers are located in the Reactor Building at the 27000 mm elevation. The IC steam supply line is connected directly to the reactor pressure vessel. The supply line leads to a steam distribution header, which feeds four pipes. Each pipe has a flow limiter to mitigate the consequences of an IC line break. The IC design basis break is a double-ended break in the piping after the steam header and flow restrictors. The IC/PCC pool is vented to atmosphere to remove steam generated in the IC pools by the condenser operation. In the event of an IC break, the steam/air mixture is expected to preferentially exhaust through hatches in the refueling floor (see Figure 1.2-9) and into the RB operating area with portions of the steam directed through the pool compartments to the stack, which is vented to the atmosphere. Because the vent path through the hatches leads to the refueling floor area, which is a large open space with no safety implications, this event was excluded from the pressurization analysis.

Main Steam (MS) Tunnel

The Reactor Building main steam tunnel is located between the primary containment vessel and the turbine building. The limiting break is a double-ended main steam line break. The main steam lines originate at the reactor pressure vessel and are routed through the steam tunnel to the turbine building. The steam/air mixture resulting from a main steam line break is directed to the turbine building through the steam tunnel. The pressure capability of the steam tunnel compartment is discussed in Subsection 3G.1.5.2.1.10. No blowout panels are required in the steam tunnel because the flow path between the steam tunnel and the turbine building is open. The main steam line break is excluded from pressurization analysis given the ability of the steam to blow down into the turbine building.

6.2.3.3 Design Evaluation

Fission Product Containment

There is sufficient water stored within the containment to cover the core during both the blowdown phase of a LOCA and during the long-term post-blowdown condition. Because of this continuous core cooling, fuel damage and fission product release is a very low probability event. If there is a release from the fuel, most fission products are readily trapped in water. Consequently, the large volume of water in the containment is expected to be an effective fission product scrubbing and retention mechanism. Also, because the containment is located entirely within the Reactor Building, multiple structural barriers exist between the containment and the environment. Therefore, fission product leakage from the RB is mitigated.

Compartment Pressurization Analysis

RWCU pipe breaks in the Reactor Building and outside the containment were postulated and analyzed. For compartment pressurization analyses, HELB accidents are postulated due to piping failures in the RWCU system where locations and size of breaks result in maximum pressure values. Calculated pressure responses have been considered in order to define the peak pressure, of the RB compartments, for structural design purposes. The calculated peak compartment pressure is 3.26 kPag which is below the RB compartment pressurization design requirements as discussed in Subsection 3G.1.5.2.1.11.

Values of the mass and energy releases produced by each break are in accordance with ANSI/ANS-56.4. The break fluid enthalpy for energy release considerations is equal to the stagnation enthalpy of the fluid in the rupture pipe. The mass and energy blowdown from the postulated broken pipe terminates when system isolation valves are fully closed after receiving the pertinent isolation closure signal.

Subcompartment pressurization effects resulting from the postulated breaks of high-energy piping have been performed according to ANSI/ANS-56.10. In order to calculate the pressure response in the Reactor Building and outside the containment due to high-energy line break accidents, CONTAIN 2.0 code was used according to the nodalization schemes shown in Figure 6.2-18. The nodalization contains the rooms where breaks occur, and all interconnected rooms/regions through flow paths such as doors, hatches, etc. Flow path and blow out panel characteristics are given in Table 6.2-12.

6.2.3.4 Tests and Inspections

Position status indication and alarms for doors, which are part of the RB envelope, are tested periodically. Leakage testing and inspection of all other architectural openings are also performed on a regular basis.

6.2.3.5 Instrumentation Requirements

Details of the initiating signals for isolation are given in Subsection 7.3.3.

Doors that form part of the RB boundary are fitted with position status indication and alarms.

6.2.4 Containment Isolation Function

The primary objective of the containment isolation function is to provide protection against releases of radioactive materials to the environment as a result of an accident. The objective is accomplished by isolation of lines or ducts that penetrate the containment vessel. Actuation of the containment isolation function is automatically initiated at specific limits defined for reactor plant operation. After the isolation function is initiated, it goes through to completion.

Relevant to the containment isolation function, this subsection addresses or references to other DCD locations that address the applicable requirements of General Design Criteria (GDC) 1, 2, 4, 16, 54, 55, 56, and 57 and Appendix K to 10 CFR Part 50 discussed in SRP 6.2.4 R2. The plant meets the relevant requirements of the following.

- GDC 1, 2, and 4 as they relate to safety-related systems being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed; systems being designed to withstand the effects of natural phenomena (e.g., earthquakes) without loss of capability to perform their safety functions; and systems being designed to accommodate postulated environmental conditions and protected against dynamic effects (e.g., missiles, pipe whip, and jet impingement), respectively.
- GDC 16 as it relates to a system, in concert with the reactor containment, being provided to establish an essentially leak tight barrier against the uncontrolled release of radioactive material to the environment.
- GDC 54, as it relates to piping systems penetrating the containment being provided with leak detection, isolation, and containment capabilities having redundant and reliable performance capabilities, and as it relates to design function incorporated to permit periodic operability testing of the containment isolation function, and leak rate testing of isolation valves.

- GDC 55 and 56 as it relates to lines that penetrate the primary containment boundary and either are part of the reactor coolant pressure boundary or connect directly to the containment atmosphere being provided with isolation valves as follows:
  - One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
  - One automatic isolation valve inside and one locked closed isolation valve outside containment; or
  - One locked closed isolation valve inside and one automatic isolation valve outside containment; or
  - One automatic isolation valve inside and one automatic isolation valve outside containment.
- GDC 57 as it relates to lines that penetrate the primary containment boundary and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere being provided with at least one locked closed, remote-manual, or automatic isolation valve outside containment.
- Appendix K to 10 CFR 50 as it relates to the determination of the extent of fuel failure (source term) used in the radiological calculations.

6.2.4.1 Design Bases

Safety Design Bases

- Containment isolation valves provide the necessary isolation of the containment in the event of accidents or other conditions and prevent the unfiltered release of containment contents that cannot be permitted by 10 CFR 50 or 10 CFR 100 limits. Leak-tightness of the valves shall be verified by Type C test.
- Capability for rapid closure or isolation of all pipes or ducts that penetrate the containment is performed means or devices to limit leakage within permissible limits.
- The design of isolation valves for lines penetrating the containment follows the requirements of General Design Criteria 54 through 57 to the greatest extent practicable consistent with safety and reliability.
- Isolation valves for instrument lines that penetrate the DW/containment conform to the requirements of Regulatory Guide 1.11.
- Isolation valves, actuators and controls are protected against loss of their safety-related function from missiles and postulated effects of high and moderate energy line ruptures.
- Design of the containment isolation valves and associated piping and penetrations meets the requirements for Seismic Category I components.
- Containment isolation valves and associated piping and penetrations meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1, 2, or MC, in accordance with their quality group classification.

- The design of the control systems for automatic containment isolation valves ensures that resetting the isolation signal shall not result in the automatic reopening of containment isolation valves.

Design Requirements

The containment isolation function, automatically closes fluid penetrations of fluid systems not required for emergency operation. Fluid penetrations supporting ESF systems have remote manual isolation valves that can be closed from the control room, if required.

The isolation criteria for the determination of the quantity and respective locations of isolation valves for a particular system conform to General Design Criteria 54, 55, 56, 57, and Regulatory Guide 1.11. Redundancy and physical separation are required in the electrical and mechanical design to ensure that no single failure in the containment isolation function prevents the system from performing its intended functions.

Protection of containment isolation function components from missiles is considered in the design, as well as the integrity of the components to withstand seismic occurrences without loss of operability. For power-operated valves used in series, no single event can interrupt motive power to both closure devices. Pneumatic-operated containment isolation valves are designed to fail to the closed position for containment isolation upon loss of the operator gas supply or electrical power.

The containment isolation function is designed to Seismic Category I. Safety and quality group classifications of equipment and systems are found in Table 3.2-1. Containment isolation valve functions are identified in Subsection 6.2.4.2.

The criteria for the design of the Leak Detection and Isolation System (LD&IS), which provides containment and reactor vessel isolation control, are listed in Subsection 7.1.2. The bases for assigning certain signals for containment isolation are listed and explained in Subsection 7.3.3.

6.2.4.2 System Design

The containment isolation function is accomplished by valves and control signals, required for the isolation of lines penetrating the containment. The reactor coolant pressure boundary (RCPB) influent lines are identified in Table 6.2-13, and the RCPB effluent lines are identified in Table 6.2-14. Using the legend in Table 6.2-15, Tables 6.2-16 through 6.2-45 show the pertinent data for the containment isolation valves. A detailed discussion of the LD&IS controls associated with the containment isolation function is included in Subsection 7.3.3.

Power-operated containment isolation valves have position switches in the control room to show whether the valve is open or closed. Loss of power to each motor-operated valve is detected and annunciated. Power for valves used in series originates from physically independent sources without cross ties to assure that no single event can interrupt motive power to both closure devices.

All motor-operated isolation valves remain in their last position upon failure of valve power. All pneumatic-operated valves (not applicable to air-testable check valves) close on loss of gas supply.

The design of the containment isolation function includes consideration for possible adverse effects of sudden isolation valve closure when the plant systems are functioning under normal operation.

General compliance or alternate approach assessment for Regulatory Guide 1.26 may be found in Subsection 3.2.2. General compliance or alternate approach assessment for Regulatory Guide 1.29 may be found in Subsection 3.2.1.

Containment isolation valves are generally automatically actuated by the various signals in primary actuation mode or are remote-manually operated in secondary actuation mode. Other appropriate actuation modes, such as self-actuated check valves, are identified in the containment isolation valve information Tables 6.2-16 through 6.2-45.

6.2.4.2.1 Containment Isolation Valve Closure Times

Containment isolation valve closure times are established by determining the isolation requirements necessary to keep radiological effects from exceeding guidelines in 10 CFR 100. For system lines, which can provide an open path from the containment to the environment, a discussion of valve closure time bases is provided in Chapter 15.

6.2.4.2.2 Instrument Lines Penetrating Containment

Sensing instrument lines penetrating the containment follow all the recommendations of Regulatory Guide 1.11. Each line has a 6-mm (1/4-inch) orifice inside the DW, as close to the beginning of the instrument line as possible, and a manually-operated isolation valve just outside the containment.

6.2.4.2.3 Compliance with General Design Criteria and Regulatory Guides

In general, all requirements of General Design Criteria 54, 55, 56, and Regulatory Guide 1.11 are met in the design of the containment isolation function. A case-by-case analysis of all such penetrations is given in Subsection 6.2.4.3.

6.2.4.2.4 Operability Assurance, Codes and Standards, and Valve Qualification and Testing

Protection is provided for isolation valves, actuators and controls against damage from missiles. All potential sources of missiles are evaluated. Where possible hazards exist, protection is afforded by separation, missile shields or by location outside the containment. Tornado missile protection is afforded by the fact that all containment isolation valves are inside the missile-proof reactor building. Internally-generated missiles are discussed in Subsection 3.5.1, and the conclusion is reached that there are no potentially damaging missiles generated. Dynamic effects from pipe break (jet impingement and pipe whip) are discussed in Section 3.6. The arrangement of containment isolation valves inside and outside the containment affords sufficient physical separation such that a high energy pipe break would not preclude containment isolation. The containment isolation function piping and valves are designed in accordance with Seismic Category I requirements as defined in Section 3.7 using the techniques of Subsection 3.9.3.2.

Section 3.11 presents a discussion of the environmental conditions, both normal and accidental, for which the containment isolation valves and pipe are designed. The section discusses the qualification tests required to ensure the performance of the isolation valves under particular environmental conditions.

Containment isolation valves are designed in accordance with the requirements of ASME Code, Section III. Where necessary, a dynamic system analysis which covers the impact effect of rapid valve closures under operating conditions is included in the design specifications of piping systems involving containment isolation valves. Valve operability assurance testing is discussed in Subsection 3.9.3.2. The power-operated and automatic isolation valves will be cycled during normal operation to assure their operability.

Subsection 6.2.6 describes leakage rate testing of containment isolation barriers.

6.2.4.2.5 Redundancy and Modes of Valve Actuations

The main objective of the containment isolation function is to provide environmental protection by preventing releases of radioactive materials. This is accomplished by complete isolation of system lines penetrating the containment. Redundancy is provided in all design aspects to satisfy the requirement that no active failure of a single valve or component prevents containment isolation.

Mechanical components are redundant, in that isolation valve arrangements provide backup in the event of accident conditions. Isolation valve arrangements satisfy all requirements specified in General Design Criteria 54, 55, 56 and 57, and Regulatory Guide 1.11.

Isolation valve arrangements with appropriate instrumentation are shown in the P&IDs. The isolation valves generally have redundancy in the mode of actuation, with the primary mode being automatic and the secondary mode being remote manual.

A program of testing (Subsection 6.2.4.4) is maintained to ensure valve operability and leak-tightness. The design specifications require each isolation valve to be operable under the most severe operating conditions that it may experience. Each isolation valve is afforded protection by separation and/or adequate barriers from the consequences of potential missiles.

Electrical redundancy is provided for each set of isolation valves, eliminating dependency on one power source to attain isolation. Electrical cables for isolation valves in the same line are routed separately. Cables are selected and based on the specific environment to which they may be subjected (e.g., magnetic fields, high radiation, high temperature and high humidity).

Functions for administrative controls and/or locks ensure that the position of all nonpowered isolation valves is maintained and known. The position of all power-operated isolation valves is indicated in the control room. Discussion of instrumentation and controls for the isolation valves is included in Subsection 7.3.3.

6.2.4.3 Design Evaluation

A discussion of the main objectives of the containment, the arrangements, the redundancies and the position control of all non-powered isolation valves and all power operated isolation valves is included in Subsection 6.2.4.2.5.

6.2.4.3.1 Evaluation Against General Design Criterion 55

The reactor coolant pressure boundary (RCPB), as defined in 10 CFR 50, Section 50.2, consists of the reactor pressure vessel, pressure-retaining appurtenances attached to the vessel, valves and pipes which extend from the reactor pressure vessel up to and including the outermost isolation valves. The lines of the RCPB, which penetrate the containment, include functions for isolation of the containment, thereby precluding any significant release of radioactivity. Similarly, for lines which do not penetrate the containment but which form a portion of the RCPB, the design ensures that isolation of the RCPB can be achieved.

The following paragraphs summarize the basis for ESBWR compliance with the requirements imposed by General Design Criterion 55.

6.2.4.3.1.1 Influent Lines

GDC 55 states that each influent line, which penetrate the containment directly to the RCPB, be equipped with at least two isolation valves, one inside the containment and the other as close to the external side of the containment as practical. Table 6.2-13 lists the influent pipes that comprise the RCPB and penetrate the containment. The table summarizes the design of each line as it satisfies the requirements imposed by General Design Criterion 55.

Feedwater Line

The feedwater line is part of the reactor coolant pressure boundary as it penetrates the containment to connect with the reactor pressure vessel. It has two automatically closing isolation valves. The isolation valve inside the containment is a check valve, located as close as practicable to the containment wall. Outside the containment is a spring-check valve located as close as practicable to the containment wall. The spring-check valve outside containment is provided with an air-opening, spring-closing operator, which, upon remote manual signal from the main control room, provides additional seating force on the valve disk to assist in long-term leakage protection. Should a break occur in the feedwater line, the check valves prevent significant loss of reactor coolant inventory and offer immediate isolation.

Isolation Condenser Condensate and Venting Lines

The isolation condenser condensate lines penetrate the containment and connect directly to the reactor pressure vessel. The isolation condenser venting lines extend from the isolation condenser through the containment and connect together downstream of two normally closed control valves in series. The venting line terminates below the minimum drawdown level in the suppression pool. An isolation condenser purge line also penetrates the containment and it contains an excess flow check valve and a normally open shutoff valve. Each IC condensate line has two open isolation gate-valves (F003 and F004) located in the containment where they are protected from outside environmental conditions, which may be caused by a failure outside the containment. In case of the venting lines there are two normally closed control globe-valves in series in each branch of the vent line. The condensate lines are automatically isolated when leakage is detected.

The IC condensate line isolation valves and the pipes penetrating the containment are designed in accordance to ASME Code Section III, Class 1 Quality Group A, Seismic Category I. Penetration sleeves used at the locations where the condensate return pipes exit the pool at the containment pressure boundary are designed and constructed in accordance with the requirements specified within Subsection 3.6.2.1. In addition, the IC System outside the containment consists of a closed loop designed to ASME Code Section III, Class 2, Quality Group B, Seismic Category I, which is a "passive" substitute for an open "active" valve outside the containment. This closed-loop substitute for an open isolation valve outside the containment implicitly provides greater safety. The combination of an already isolated loop outside the containment plus the two series automatic isolation valves inside the containment comply with the intent of isolation functions of US NRC Code of Federal Regulations 10 CFR 50, Appendix A, Criteria 55 and 56.

Standby Liquid Control System Line

The Standby Liquid Control (SLC) system line penetrates the containment to inject directly into the reactor pressure vessel. In addition to a simple check valve inside the containment, a check valve, together with two parallel squib-valves are located outside the DW. Because the SLC line is normally closed, rupture of this non-flowing line is extremely improbable. However, should a break occur subsequent to the opening of the squib-valves, the check valves ensure isolation.

All mechanical components required for boron injection are at least Quality Group B. Those portions which are part of the reactor coolant pressure boundary are classified Quality Group A.

6.2.4.3.1.2 Effluent Lines

GDC 55 states that each effluent line, which form part of the reactor coolant pressure boundary and penetrate the containment, be equipped with two isolation valves; one inside the containment and one outside, located as close to the containment wall as practicable.

Table 6.2-14 lists those effluent lines that comprise the reactor coolant pressure boundary and which penetrate the containment.

Main Steam and Drain Lines

The main steam lines, which extend from the reactor pressure vessel to the main turbine and condenser system, penetrate the containment. The main steam drain lines connect the low points of the steam lines, penetrate the containment and are routed to the condenser hotwell. For these lines, isolation is provided by automatically actuated globe-valves, one inside and one just outside the containment.

The main steamline isolation valves (MSIVs) are spring loaded, pneumatically-operated globe valves designed to close on loss of gas pressure or loss of power to the solenoid-operated pilot valves. Each valve has two pilot valves supplied from independent power sources, both of which must be de-energized to close the MSIV. Two MSIVs are used in series to assure isolation when needed. Each MSIV uses gas pressure for closure upon interruption of electrical power to the pilot valves. A spring closes the valve when there is no gas pressure. The separate and independent action of either gas pressure or spring force is capable of closing an isolation valve. Refer to Subsection 5.4.5 for Main Steamline Isolation System description.

Isolation Condenser Steam Supply Lines

The isolation condenser steam supply lines penetrate the containment and connect directly to the reactor pressure vessel. Two isolation gate-valves are located in the containment where they are protected from outside environmental conditions, which may be caused by a failure outside the containment. The isolation valves in each IC loop are signaled to close automatically on excessive flow. The flow is sensed by four differential flow transmitters in either the steam supply line or the condensate drain line. The isolation valves are also automatically closed on high radiation in the steam leaving an IC-pool compartment. The isolation functions are based on any 2-out-of-4 channel trips.

The IC isolation valves and the pipe penetrating the containment are designed in accordance to ASME Code Section III, Class 1 Quality Group A, Seismic Category I. Penetration sleeves used at the locations where the IC steam supply lines enter the pool at the containment pressure boundary are designed and constructed in accordance with the requirements specified within Subsection 3.6.2.1. In addition to the IC isolation valves, the IC system outside the containment consists of a closed loop designed to ASME Code Section III, Class 2, Quality Group B, Seismic Category I, which is a “passive” substitute for an open “active” valve outside the containment. This closed-loop substitute for an open isolation valve outside the containment implicitly provides greater safety. The combination of an already isolated loop outside the containment plus the series automatic isolation valves inside the containment comply with the intent of isolation funtions of US NRC Code of Federal Regulations 10 CFR 50, Appendix A, Criteria 55 and 56.

Reactor Water Cleanup System /Shutdown Cooling System

The Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System takes its suction from the reactor pressure vessel. The RWCU/SDC suction lines of each loop are isolated by one automatic nitrogen-operated gate valve inside and two parallel motor-operated gate valves outside the containment. During normal operation, the larger of these parallel valves (used for shutdown cooling) is closed.

RWCU/SDC pumps, heat exchangers and demineralizers are located outside the containment.

6.2.4.3.1.3 Conclusion on Criterion 55

In order to assure protection against the consequences of accidents involving the release of radioactive material, pipes which form the reactor coolant pressure boundary are shown to provide adequate isolation capabilities on a case-by-case basis. In all cases, two isolation barriers were shown to protect against the release of radioactive materials.

In addition to meeting the isolation requirements stated in Criterion 55, the pressure-retaining components which comprise the reactor coolant pressure boundary are designed to meet other appropriate requirements which minimize the probability or consequences of an accidental pipe rupture. The quality requirements for these components ensure that they are designed, fabricated, and tested to the highest quality standards of all reactor plant components. The classification of components which comprise the reactor coolant pressure boundary are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 1.

It is therefore concluded that the design of piping systems which comprise the reactor coolant pressure boundary and which penetrate the containment satisfies Criterion 55.

6.2.4.3.2 Evaluation Against Criterion 56

Criterion 56 requires that lines, which penetrate the containment and communicate with the containment atmosphere, must have two isolation valves; one inside the containment, and one outside, unless it can be demonstrated that the containment isolation functions for a specific class of lines are acceptable on some other basis.

The following paragraphs summarize the basis for ESBWR compliance with the requirements imposed by Criterion 56.

6.2.4.3.2.1 Influent Lines to Containment

On a system basis, Tables 6.2-16 through 6.2-45 address the isolation valve functions in the influent lines to the containment.

Fuel and Auxiliary Pool Cooling System

The lines from the Fuel and Auxiliary Pool Cooling System penetrate the containment separately and are connected to the drywell spray, the suppression pool and to the Gravity-Driven Cooling System (GDCS) pools. In each of these lines there are one motor-operated isolation -valve outside and one check valve inside the containment. Only the GDCS pool return line motor-operated isolation valve is automatically closed on a containment isolation signal.

Subsection 9.1.3.1.2 contains additional information about the containment isolation design for FAPCS including any justifications for deviation from the GDC 56 requirements.

Chilled Water System

Isolation is provided for the Chilled Water System (CWS) cooling lines penetrating containment. It is assumed that the nonsafety-related Seismic Category II coolant boundary of the CWS or Drywell Cooling System heat exchanger may fail, opening to the containment atmosphere. Therefore, Criterion 56 is applied to the design of the CWS containment penetration. The CWS containment influent lines have a motor-operated gate valve outside and a motor-operated gate inside the containment.

Containment Inerting System

The penetration of the Containment Inerting System consists of two in-series butterfly isolation valves (normally closed) in parallel with two in-series globe isolation valves. All isolation valves on these lines are outside of the containment to provide accessibility to the valves. The first valve is located as close as practical to the containment.

High Pressure Nitrogen Supply System

The High Pressure Nitrogen Supply System penetrates the containment with one globe valve outside and one check valve inside each containment penetration. Because the pressure in this system is higher than the containment pressure, it is only isolated on low pressure signal inside the High Pressure Nitrogen Supply System.

6.2.4.3.2.2 Effluent Lines from Containment

On a system basis, Tables 6.2-16 through 6.2-45 address the isolation functions in the effluent lines from the containment.

Fuel and Auxiliary Pools Cooling System Suction Lines

The FAPCS suction line from the GDCS pool is provided with two power-assisted isolation valves, one pneumatic-operated inside and one motor-operated outside the containment.

The FAPCS suction line from the suppression pool has one isolation valve outside the containment as the first barrier and the FAPCS piping outside containment as the second barrier. Because the penetration can be under water under certain accident conditions, there can be no isolation valve located inside the containment. The valve is located as close as possible to the containment.

Subsection 9.1.3.1.2 contains additional information about the containment isolation design for FAPCS including any justifications for deviation from the GDC 56 requirements.

Chilled Water System

The CWS effluent lines penetrating the containment each have a motor-operated gate valve outside containment and a motor-operated gate valve inside the containment.

Containment Inerting System

The penetration of the Containment Inerting System consists of two in-series butterfly isolation valves (normally closed) in parallel with two in-series globe isolation valves. All isolation valves on these lines are outside of the containment to provide accessibility to the valves. The first valve is located as close as practical to the containment. The piping to both valves is an extension of the containment boundary.

Post Accident Sampling System

The penetrations for this system consist of eight lines, four sampling lines and four return lines. Four lines each penetrate the DW and wetwell. Each line uses two valves in series, one normally open mechanical globe valve positioned close to the containment and the other a solenoid operated gate valve. This system is primarily used for post-accident sampling of the atmosphere in the containment. The solenoid-operated valves are normally closed and are sequenced open and close for extraction of air samples. All valves are located outside the containment for easy access. The piping to these valves is considered an extension of the containment boundary.

Process Radiation Monitoring System

The penetrations for the fission products monitor sampling lines consist of one sampling line and one return line. Each line uses three globe valves in series. One valve is a mechanical globe valve used for maintenance and is located close to the containment. The other two valves are air-operated solenoid valves and are used for isolation. All three valves are located outside the containment for easy access. The piping to these valves is considered an extension of the containment boundary.

6.2.4.3.2.3 Conclusion on Criterion 56

In order to ensure protection against the consequences of an accident involving release of significant amounts of radioactive materials, pipes that penetrate the containment have been demonstrated to provide isolation capabilities on a case-by-case basis in accordance with Criterion 56.

In addition to meeting isolation requirements, the pressure-retaining components of these systems are designed to the quality standards commensurate with their importance to safety.

6.2.4.3.2.4 Evaluation Against General Design Criterion 57

The ESBWR has no closed system lines penetrating the containment that require automatic isolation.

6.2.4.3.2.5 Evaluation Against Regulatory Guide 1.11

Instrument lines that connect to the RCPB and penetrate the containment have 1/4-inch orifices and manual isolation valves, in compliance with Regulatory Guide 1.11 requirements.

6.2.4.3.3 Evaluation of Single Failure

A single failure can be defined as a failure of a component (e.g., a pump, valve, or a utility such as offsite power) to perform its intended safety-related functions as a part of a safety-related system. The purpose of the evaluation is to demonstrate that the safety-related function of the system would be completed even with that single failure. Appendix A to 10 CFR 50 requires that electrical systems be designed specifically against a single passive or active failure. Section 3.1 describes the implementation of these standards, as well as General Design Criteria 17, 21, 35, 38, 41, 44, 54, 55 and 56.

Electrical and mechanical systems are designed to meet the single-failure criterion, regardless of whether the component is required to perform a safety-related action or function. Even though a component, such as an electrically-operated valve, is not designed to receive a signal to change state (open or closed) in a safety scheme, it is assumed as a single failure if the system component changes state or fails. Electrically-operated valves include valves that are electrically piloted but air/nitrogen-operated, as well as valves that are directly operated by an electrical device. In addition, all electrically-operated valves that are automatically actuated can also be manually actuated from the main control room. Therefore, a single failure in any electrical system is analyzed, regardless of whether the loss of a safety-related function is caused by a component failing to perform a requisite mechanical motion or a component performing an unnecessary mechanical motion.

6.2.4.4 Test and Inspections

The containment isolation function is scheduled to undergo periodic testing during reactor operation. The functional capabilities of power-operated isolation valves are tested remote-manually from the control room. By observing position indicators and changes in the affected system operation, the closing ability of a particular isolation valve is demonstrated.

A discussion of leak rate testing of isolation valves is provided in Subsection 6.2.6. Leakage integrity tests shall be performed on the containment isolation valves with resilient material seals at least once every three months.

6.2.5 Combustible Gas Control in Containment

According to 10 CFR 50.44(c)(2), which provides the combustible gas control requirements for future water-cooled reactor applicants and licensees, containments with an inerted atmosphere do not require a method to control the potential buildup of post accident hydrogen.

In SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) And Recommendations on Risk-informed Changes to 10 CFR 50.44 (Combustible Gas Control)," dated September 14, 2000, the NRC staff recommended changes to 10 CFR 50.44 that reflect the position that only combustible gas generated by a beyond-design-basis accident is a risk-significant threat to containment integrity.

Based on those recommendations, 10 CFR 50.44 eliminates requirements that pertain to only design-basis LOCAs.

During severe accident conditions with a significant amount of fission product gases and hydrogen release to the containment, the containment will remain inerted without any additional action because radiolytic oxygen production remains below the concentration that could pose a risk of hydrogen burning for a significant period of time following the event. Accumulation of combustible gases that may develop in the period after about 24 hours can be managed by implementation of the severe accident management guidelines. For a severe accident with a substantial release of hydrogen, the oxygen concentration in containment from radiolysis is not expected to reach 5% for significantly longer than 24 hours.

6.2.5.1 Design Bases

The specific requirements in 10 CFR 50.44, "Combustible gas control for nuclear power reactors, Section (c)(2), establishes for future water-cooled reactor applicants and licensees that "all containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features". The design of the ESBWR provides for an inerted containment and, as a result, no system to limit hydrogen concentration is required.

In the ESBWR, the Containment Inerting System is provided to establish and maintain an inert atmosphere within the containment. The Containment Inerting System design is discussed in Section 9.4.8 and summarized later in this subsection.

Relevant to combustible gas control, this subsection addresses or references other DCD locations that address the applicable requirements of 10 CFR 50.44 and General Design Criteria (GDC) 5, 41, 42 and 43 as discussed in SRP 6.2.5 Revision 2 and Regulatory Guide 1.7 Revision 3. The plant meets the relevant requirements of the following:

- 10 CFR 50.44 and 50.46 as they relate to BWR plants being designed to have containments with an inerted atmosphere.
- GDC 5 does not apply to the inerting function because there is no sharing of structures, systems and components between different units.
- GDC 41, as it relates to systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained, does not apply to the ESBWR because the safety function is accomplished by keeping the containment inerted. Thus, no redundancy or single failure criteria shall be considered as the inerted containment is intrinsically safe and passive.
- GDC 42 & 43, related to the design of the systems to permit appropriate periodic inspection and periodic testing of components to ensure the integrity and capability of the systems, do not apply to the inerting function; periodic monitoring of oxygen concentration is adequate to confirm the safety function.

- Regulatory Guide 1.7 Revision 3 as it relates to the systems being designed to limit the oxygen gas concentrations within the containment.

6.2.5.1.1 Containment Purging Under Accident Conditions

In accordance with 10 CFR 50.34(2)(xv), (NUREG-0933 Item II.E.4.4), the capability for containment purging/venting is designed to minimize the purging time consistent with ALARA principles for occupational exposure. The piping, valves and controls in the Containment Inerting System can be used to control containment pressure (i.e., purge the containment), and can reliably be isolated under accident conditions.

6.2.5.2 Containment Inerting System

The Containment Inerting System design and functions are described in-detail within Section 9.4, and its instrumentation and controls are described in-detail within Section 7.7.

The objective of the Containment Inerting System is to preclude combustion of hydrogen and prevent damage to essential equipment and structures by providing an inerted containment environment. This is the method of combustible gas control for the ESBWR, as required by 10 CFR 50.44.

The Containment Inerting System establishes and maintains an inert atmosphere within the primary 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 system is designed to permit de-inerting the containment for safe operator access. In addition, the Containment Inerting System maintains a positive pressure in the primary containment to prevent air (oxygen) in-leakage into the inerted spaces from the reactor building.

The Containment Inerting System is comprised of a pressurized liquid storage tank, a steam heated main vaporizer for large nitrogen flow, electric heater for vaporizing makeup flow, two supply injection lines (a makeup line and an inerting line), two exhaust lines, a bleed line, a containment overpressure protection line, and associated valves, controls and instrumentation.

During plant startup, an inert atmosphere throughout the primary containment system is established by a large flow of nitrogen from the liquid nitrogen storage tank which is vaporized by the steam heated vaporizer and injected into the primary containment. The exhaust line is kept open to displace the containment resident atmosphere with nitrogen. Once the desired concentration of nitrogen is reached, the exhaust line is allowed to close. When the required inerted containment operating pressure is attained, the inerting process is terminated by the closure of the nitrogen supply shutoff valve and inerting isolation valves.

The containment can be inerted to ≤ 3 % dry-basis-percent (DB%) oxygen by volume within four hours. The Containment Inerting System is designed to establish a more completely inert atmosphere, equal to or less than 2 DB % in both the drywell and in the suppression pool space, within the next eight hours after reaching the 3 DB % condition.

The Containment Inerting System can be used under post accident conditions for containment atmosphere dilution to maintain the containment in an inerted condition by a controlled purge of the containment atmosphere with nitrogen, to prevent reaching a combustible gas condition.

6.2.5.3 Containment Atmosphere Monitoring

The Containment Monitoring System (CMS) provides the function that is necessary to meet or exceed the requirements of 10 CFR 50.44 (c)(4) with regard to oxygen and hydrogen monitoring.

The CMS is a safety-related, Seismic Category 1 system consisting of two redundant, physically and electrically independent post-accident monitoring divisions. Each division is capable of measuring and recording the radiation levels and the oxygen and hydrogen concentration levels in the drywell and suppression chamber. The functions of the CMS are:

- To monitor hydrogen and oxygen concentrations and gross gamma radiation levels in the drywell and suppression chamber under post-accident conditions;
- To provide main control room display and alarms; and
- To provide alarm enunciating signals if alarm levels are reached or if the system is in an inoperative state.

6.2.5.3.1 Hydrogen Monitoring

Hydrogen monitoring consists of two hydrogen monitoring channels containing hydrogen sensors, sample lines to bring a sample from the drywell or suppression chamber to the sensor, hydrogen monitor electronics assemblies, visual displays and a calibration gas supply. Each hydrogen channel determines the hydrogen content of a sample from the containment. The data is transmitted to the main control room where the data is continuously displayed. High hydrogen concentration alarms are provided. The channels are equipped with an inoperative alarm to indicate malfunctions. The channels are divided into two redundant divisions.

6.2.5.3.2 Oxygen Monitoring

Oxygen monitoring consists of two oxygen monitoring channels containing oxygen sensors, sample lines to bring a sample form the drywell or suppression chamber, oxygen monitor electronics assemblies in the control room, visual displays and a calibration gas supply. Each oxygen channel determines the oxygen content of a sample from the containment. The data is transmitted to the main control room where the data is continuously displayed. Trips are provided to indicate unacceptable oxygen levels. The channels are equipped with an inoperative alarm to indicate malfunctions. The channels are divided into two redundant divisions.

6.2.5.3.3 Radiation Monitoring

Radiation monitoring consists of two channels per division (1 and 2) of radiation detector assemblies, radiation electronic assemblies and visual displays. The channels measure gross gamma radiation in the drywell and suppression chamber. The signals are carried back to the main control room where the signals are continuously displayed. The channels are equipped with an inoperative alarm to indicate channel malfunction. The radiation monitoring channels are divided into two redundant measurement divisions.

6.2.5.3.4 Containment Atmosphere Mixing

The ESBWR design provides protection from localized combustible gas deflagrations including the capability to mix the steam and non-condensable gases throughout the containment atmosphere and minimize the accumulation of high concentrations of combustible gases in local areas. The following are containment design features that will reduce the likelihood of combustible gas deflagrations resulting from localized buildup of combustible gases during degraded core accidents:

- The inerted containment precludes the possibility of a combustible gas deflagration for a significant period of time following a severe accident. The radiolytic generation of oxygen in the post accident period is a very slow process and provides sufficient time for implementation of severe accident management procedures.
- The PCCS piping conveys a continuous flow of post-accident containment steam/non-condensable mixture to the upper drums of each HX module. Drywell pressure will be slightly higher than suppression chamber gas-space pressure and provide a continuous purging of any non-condensable gases accumulating within the HX tube bundle regions to the suppression pool. These natural circulation flow processes will provide a continuous mixing flow throughout the containment atmosphere and prevent localized buildup within the drywell and the suppression pool.
- The drywell sprays will be utilized during a severe accident and will provide a significant amount of mixing in the drywell. Communication between the drywell and the suppression pool via the vacuum breakers between the compartments will also promote mixing.
- Natural circulation is also promoted by steam flowing into the drywell through the open DPVs and condensing on the containment walls.

6.2.5.4 Containment Overpressure Protection

6.2.5.4.1 Design Evaluation

The pressure capability of the ESBWR containment vessel is such that it will not be exceeded by any design basis or special event.

The pressure capability of the ESBWR containment limiting component is higher than the pressure which results from assuming 100% fuel clad-coolant reaction. There is sufficient margin to the containment pressure capability such that there is no need for an automatic containment overpressure protection system. In the hypothetical situation where containment depressurization is required, this depressurization can be performed by manual operator action.

The containment can be manually vented through the Containment Inerting System. The Containment Inerting System is equipped with containment penetrations, valves and pipes that may be used for containment depressurization. This system is provided with two normal de-inerting lines with double containment isolation valves in series. One of the de-inerting lines takes suction from the suppression pool airspace. For containment overpressure protection, only this second line will be used. Having the release point in the suppression pool airspace forces the escaping fission products through the suppression pool.

6.2.5.4.2 Containment Structural Integrity

In accordance with Regulatory Guide (RG) 1.7 Revision 3, an acceptable method for demonstration of containment structural integrity is to meet the ASME Section III acceptance criteria as follows:

- That steel containments meet the requirements of the ASME Boiler and Pressure Vessel Code (Edition and Addenda as incorporated by reference in 10 CFR 50.55a(b)(1)), Section III, Division 1, Subarticle NE - 3220, Service Level C Limits, considering pressure and dead load alone (evaluation of instability is not required); and
- That concrete containments meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subarticle CC - 3720, Factored Load Category, considering pressure and dead load alone.

This evaluation is termed Level C pressure capability and is presented below.

Level C pressure capability of the steel components of major penetrations (Drywell Head, Equipment Hatch, Personnel Airlock and Wetwell Hatch) in the ESBWR concrete containment is evaluated based on the primary membrane stress Pm applying ASME Section III NE-3324, in which the maximum allowable stress S is taken to be Sy (material yield strength) as Level C stress limit in accordance with NE-3220. The local membrane stress PL and local membrane plus primary bending stress PL + Pb are non-controlling.

The basic equation for Level C pressure capability is:

$$P_c = (S_c - \sigma_d) / \sigma_{up} \tag{6.2-1}$$

where:

P_c = Level C pressure

S_c = Level C allowable stress

σ_d = Primary membrane stress due to dead load, this is negligible small for these penetrations

σ_{up} = Primary membrane stress due to unit pressure (1MPa)

The σ_{up} of each part of the penetrations are derived from following article of ASME Sec III, Subsection NE -

- Penetration sleeve: NE-3324.3(a)
- Torispherical drywell head: NE-3324.8(b)
- Other penetration head: NE-3324.4(a)

The yield strength Sy of the material (SA-516 Gr.70) at ambient temperature is applied as Sc. Under internal pressures the knuckle region of the torispherical drywell head is under compression in the hoop direction. Although evaluation of instability is not required by RG 1.7 Revision 3, a separate check for compressive stress is conducted to determine the buckling strength.

Level C internal pressure buckling capability of the drywell head is evaluated using a design equation derived by Galletly from test data (Reference 6.2-2) and adjusted for factors of safety required by ASME Section III, NE-3222 and Code Case N-284 provisions. The design equation proposed by Galletly for preventing buckling in fabricated torispherical shell under internal pressure is:

$$P_d = \frac{80\, S_y \left(\frac{r}{D}\right)^{0.825}}{\left(\frac{D}{t}\right)^{1.5}\left(\frac{L}{D}\right)^{1.15}} = 1.317\text{MPa} \tag{6.2-2}$$

where:

S_y	=	yield strength of the material	= 262MPa
t	=	uniform thickness of the head	= 40mm
r	=	radius of the knuckle shell	= 1796mm
D	=	diameter of the cylindrical shell	= 10400mm
L	=	radius of the spherical cap	= 9407mm

This equation is formulated for design use with knock-down (capacity reduction) factors included. As compared to all known test results (43 in total), the ratios of the actual buckling pressure to the allowable buckling pressure predicted by this equation were found to range from 1.51 to 4.01. Hence, a minimum factor of safety of 1.5 is ensured by this equation. In accordance with the statistical analysis performed in the ABWR DCD of the test data on which Eq. 6.2-2 is based, the lower bound and best estimate buckling pressures can be obtained as:

Lower Bound $P_{lb} = 1.5*P_d = 1.975\text{MPa}$ (6.2-3)

Best Estimate $P_{be} = 2.27*P_d = 2.989\text{MPa}$ (6.2-4)

For Level C evaluation ASME NE-3222 requires a 2.5 factor of safety applied to the best estimate buckling stress. The factor of safety required by Code Case N-284 for Level C is 1.67 and it is applied to the lower bound value. Hence, the Level C internal pressure capability of the drywell head is determined to be the smaller value predicted from the two equations below.

$P_c = P_{be}/2.5$ =1.195MPa per NE-3222 (6.2-5)

$P_c = P_{lb}/1.67$ =1.182MPa per N-284 (6.2-6)

The Level C pressure capabilities of the steel components of major penetrations are summarized in Table 6.2-46. The governing pressure is 1.182 MPa, which is controlled by buckling strength of the drywell head.

The PCCS heat exchangers are also part of containment boundary. The Level C pressure capacity of the most critical component in the PCCS heat exchangers is found to be 1.33 MPa.

Level C pressure capability of the concrete containment is evaluated to meet the liner strain limits stipulated in ASME Section III, Division 2, CC-3720. A nonlinear finite element analysis of the containment concrete structure including liner plates is performed for over-pressurization. The analysis results show that when the internal pressure reaches as high as 1.468 MPa, the maximum liner strain is only 0.165% tension, which is well within the 0.3% limit for Factored Load Category specified in ASME Table CC-3720-1. Thus, Level C pressure capacity of the concrete containment is at least 1.468 MPa and it is higher than the 1.182 MPa controlling pressure for the steel components.

In summary, the Level C pressure capability of the ESBWR containment structure is 1.182 MPa under pressure and dead load alone.

6.2.5.5 Post Accident Radiolytic Oxygen Generation

For a design basis loss of coolant accident (LOCA) in the ESBWR, the Automatic Depressurization System (ADS) would depressurize the reactor vessel and the Gravity Driven Cooling System (GDCS) would provide gravity driven flow into the vessel for emergency core cooling. The safety analyses show that the core does not uncover during this event and as a result, there is no fuel damage or fuel clad-coolant interaction that would result in the release of fission products or hydrogen. Thus, for design basis LOCA, the generation of post accident oxygen would not result in a combustible gas condition and a design basis LOCA does not have to be considered in this regard.

For the purposes of post accident radiolytic oxygen generation for the ESBWR, a severe accident with a significant release of iodine and hydrogen is more appropriate to consider.

Because the ESBWR containment is inerted, the prevention of a combustible gas deflagration is assured in the short term following a severe accident. In the longer term there would be an increase in the oxygen concentration resulting from the continued radiolytic decomposition of the water in the containment. Because the possibility of a combustible gas condition is oxygen limited for an inerted containment, it is important to evaluate the containment oxygen concentration versus time following a severe accident to assure that there will be sufficient time to implement severe accident management (SAM) actions. It is desirable to have at least a 24-hour period following an accident to allow for SAM implementation. This section discusses the rate at which post accident oxygen will be generated by radiolysis in the ESBWR containment following a severe accident, and establishes the period of time that would be required for the oxygen concentration in containment to increase to a value that would constitute a combustible gas condition (5% oxygen by volume) in the presence of a large hydrogen release, thus de-inerting the containment in the absence of mitigating SAM actions.

6.2.5.5.1 Background

The rate of gas production from radiolysis depends upon the power decay profile and the amount of fission products released to the coolant. Appendix A of Standard Review Plan (SRP) Section 6.2.5 provides a methodology for calculation of radiolytic hydrogen and oxygen generation. The analysis results discussed herein were developed in a manner that is consistent with the guidance provided in SRP 6.2.5 and RG 1.7.

There are unique design features of the ESBWR that are important with respect to the determination of post accident radiolytic gas concentrations. In the post accident period, the ESBWR does not utilize active systems for core cooling and decay heat removal. As indicated earlier, for a design basis LOCA, the ADS would depressurize the reactor vessel and the GDCS would provide gravity driven flow into the vessel for emergency core cooling. The core would be subcooled initially and then it would saturate resulting in steam flow out of the vessel and into the containment. The Passive Containment Cooling System (PCCS) heat exchangers would remove the energy by condensing the steam. This would be the post accident mode and the core coolant would be boiling throughout this period.

A similar situation would exist for a severe accident that results in a core melt followed by reactor vessel failure. In this case, the GDCS liquid would be covering the melted core material in the lower drywell, with an initial period of subcooling followed by steaming. The PCCS heat exchangers would be removing the energy in the same manner as described above for a design basis LOCA.

In order to prevent non-condensable related termination of steam condensation, the PCCS heat exchangers are provided with a vent which will transfer any non-condensable gases which accumulate in the heat exchanger tubes to the suppression pool vapor space, driven by the drywell to suppression pool pressure differential. In this way, the majority of the non-condensable gases will be in the suppression pool. The calculation of post accident radiolytic oxygen generation accounts for this movement of non-condensable gases to the suppression pool after they are formed in the drywell.

The effect of the core coolant boiling is to strip dissolved gases out of the liquid phase resulting in a higher level of radiolytic decomposition. This effect was accounted for in the analysis.

6.2.5.5.2 Analysis Assumptions

The analysis of the radiolytic oxygen concentration in containment was performed consistent with the methodology of Appendix A to SRP 6.2.5 and RG 1.7. Some of the key assumptions are as follows:

- Reactor power was 102% of rated
- $G(O_2)$ = 0.25 molecules/100eV
- Initial containment O_2 concentration = 4%
- Allowed containment O_2 concentration = 5%
- Stripping of drywell non-condensable gases to wetwell vapor space
- Fuel clad-coolant reaction up to 100%
- Iodine release up 100%

6.2.5.5.3 Analysis Results

The analysis results show that the time required for the oxygen concentration to increase to the de-inerting value of 5% is significantly greater than 24 hours for a wide range of fuel clad-coolant interaction and iodine release assumptions up to and including 100%. The results support the conclusion that there will be sufficient time available to activate the emergency response organization and implement the SAM actions necessary to preclude a combustible gas deflagration.

6.2.6 Containment Leakage Testing

This subsection describes the testing program for determining the containment integrated leakage rate (Type A tests), containment penetration leakage rates (Type B tests), and containment isolation valve leakage rates (Type C tests) that complies with Appendix J and GDC 52, 53 and 54. Type A, B, and C tests are performed prior to operations and periodically thereafter to assure that leakage rates through the containment and through systems or components that penetrate containment do not exceed their maximum allowable rates. Maintenance of the containment, including repairs on systems and components penetrating the containment, is performed as necessary to maintain leakage rates at or below acceptable values.

The ESBWR conformance with Appendix J satisfies the requirements of the following GDC.

- GDC 52 as it relates to the reactor containment and exposed equipment being designed to accommodate the test conditions for the containment integrated leak rate test (up to the containment design pressure).
- GDC 53 as it relates to the reactor containment being designed to permit appropriate inspection of important areas (such as penetrations), an appropriate surveillance program, and leak testing at the containment design pressure of penetrations having resilient seals and expansion below.
- GDC 54 as it relates to piping systems penetrating primary reactor containment being designed with a capability to determine if valve leakage is within acceptable limits.

6.2.6.1 Containment Integrated Leakage Rate Test (Type A)

6.2.6.1.1 Initial Integrated Leak Rate Test

After construction of the reactor containment, including installation of all portions of mechanical, electrical, and instrumentation systems penetrating the containment pressure boundary, and upon satisfactory completion of all structural integrity tests described in Subsections 3.8.1 and 3.8.3, the initial (preoperational) Type A integrated leakage rate test (ILRT) is performed to verify that the actual containment leakage rate does not exceed the design limit.

The ILRT is performed by pressurizing the containment with air. The air shall be dry, clean, and free of contaminants. Pressurization shall be conducted preferably when there is relatively low humidity in the outside atmosphere to avoid moisture condensation within the containment structure. To provide low humidity and improve pumping efficiency, cool night air is also preferred. The containment ILRT consists of three phases, namely:

- Pressurization Phase: Portable air compressors shall be used to pressurize the containment at a calculated accidental peak containment internal pressure, P_{ac}. Pressurization takes approximately 8 hours.
- Pressure Stabilization Phase: After the required test pressure has been achieved, the containment pressure shall be allowed to stabilize for at least 4 hours before leakage measurements may be performed. Pressure stability shall be considered achieved when a condition of essential temperature equilibrium has been attained.
- Integrated Leakage Rate Test Phase: After the containment atmosphere has stabilized, the ILRT test begins. The test duration shall extend to 24 hours of retained internal pressure. A shorter test period may be acceptable if it can be demonstrated that the leakage rate can be accurately determined during the shorter test period.

The absolute method, as described in ANSI N45.4, shall be used to determine the mass of air in the containment. This method calculates air mass at a stated time by means of direct pressure, temperature, and humidity measurements. The contained mass is calculated using the ideal gas law. The calculated mass shall be plotted against time during the test period, and the mass point method, as described in ANSI/ANS 56.8, shall be used to determine the leakage rate. Instrumentation and monitors used in the ILRT shall be designed, calibrated, and tested so that containment parameters can be precisely measured. A computer shall be used for data acquisition and computation of the leakage rate.

Acceptance Criteria

- A standard statistical analysis of the data is conducted by a linear regression analysis using the method of least squares to determine the leakage rate and associated 95% Upper Confidence Limit (UCL). ILRT results are satisfactory if the UCL is less than 75% of the maximum allowable leakage rate, L_a. The maximum allowable leakage rate (L_a) is 0.5% by weight of the contained atmosphere in a 24-hour period (excluding MSIV leakage). The calculated leakage rate and upper 95% confidence limit are reported to the NRC.

- After completing the initial ILRT, a verification test is conducted to confirm the ability of the ILRT method and equipment to satisfactorily determine the containment leakage rate (L_{am}). The accuracy of the leakage rate tests is verified by superimposing a calibrated leak on the normal containment leakage rate or by other methods of demonstrated equivalency. The difference between the total leakage and the superimposed known leakage is the actual leakage rate. This method confirms the test accuracy. The measurements are acceptable if the correlation between the verification test data and ILRT data demonstrates an agreement within ± 25%. Appendix C of ANSI/ANS 56.8 includes more descriptive information on verification methods.

- During the ILRT (including the verification test), if excessive leakage occurs through locally testable penetrations or isolation valves to the extent that it would interfere with satisfactory completion of the test, these leakage paths may be isolated and the Type A test continued until completion. A local test shall be performed before and after the repair of each isolated path. The sum of the local leakage rates and the UCL shall be less than 75% of the maximum allowable leakage, L_a. Local leakage rates shall not be subtracted from the Type A test results to determine acceptability of the test. The test results shall be reported with both pre- and post-repair local leakage rates, as if two Type A tests had been conducted. Record of corrective actions shall be documented and included in the report to be submitted to the NRC.

Prerequisites

The following prerequisites are completed before starting an ILRT:

- A visual examination of critical areas and general inspection of the accessible interior and exterior surfaces of the containment structure and components are performed to uncover any evidence of structural deterioration that may affect either the structural integrity or leak-tightness of the containment. If there is evidence of significant structural deterioration, corrective action is taken in accordance with approved repair procedures before the ILRT is performed. The structural deterioration and corrective action are reported to the NRC in accordance with Appendix J of 10 CFR 50. Except for the inspections and actions described above, during the period between the initiation of the inspection and the initiation of the ILRT, no preliminary leak detection surveys and repairs are performed before conducting the Type A test.

- Closure of containment isolation valves is accomplished by the normal mode of actuation and without preliminary exercises or adjustments (e.g., no tightening of the valves by manual handwheel after closure by valve motor). All malfunctions and subsequent corrective actions are reported in conjunction with the ILRT results.
- The Type B and Type C leakage rate tests (Subsections 6.2.6.2 and 6.2.6.3) are completed before the Type A test is performed.

6.2.6.1.2 Periodic Integrated Leakage Rate Tests

Following the initial preoperational tests, ILRTs (Type A tests) are conducted periodically according to 10 CFR 50 Appendix J to ensure that the containment integrity is maintained and to determine if the leakage rate has increased since the previous ILRT. The tests are performed at regular intervals (described below), after major repairs, and upon indication of excessive leakage. The periodic ILRTs follow the same method as the initial ILRT, and the same test prerequisites and acceptance criteria also apply to the periodic ILRTs. Verification tests are also performed after each ILRT.

After the initial ILRT, ILRTs shall be performed at approximately equal intervals during each 10-year service period. In addition, any major modification or replacement of components of the reactor containment performed after the initial ILRT are followed by either a Type A or a Type B test of the area affected by the modification, with the affected area meeting the applicable acceptance criteria. This frequency of testing is established on the basis of 10 CFR 50 Appendix J.

If any ILRT fails to meet the acceptance criteria prior to corrective action, the test schedule applicable to subsequent ILRTs shall be subject to review and approval by the NRC. If two consecutive periodic ILRTs fail to meet the acceptance criteria prior to corrective action, an ILRT is performed at each plant shutdown for major refueling or approximately every 24 months (whichever occurs first), until two consecutive ILRTs meet the acceptance criteria, after which time the previously established periodic retest schedule may be resumed.

Additional Criteria for Integrated Leakage Rate Tests

- The following portions of systems are kept open or vented to the containment atmosphere during the ILRT:
    - portions of fluid systems that are part of the reactor coolant pressure boundary that are open directly to the reactor containment atmosphere under post-accident conditions and that become an extension of the boundary of the reactor containment; and
    - portions of closed systems inside containment that penetrate containment and that are not relied upon for containment isolation purposes following a LOCA.
- All systems not designed to remain filled with fluid (e.g., vented) after a LOCA are drained of water to the extent necessary to ensure exposure of the system containment isolation valves to the containment air test pressure.

- Those portions of fluid systems penetrating containment that are external to the containment and that are not designed to provide a containment isolation barrier are vented to the outside atmosphere, as applicable, to ensure that full post-accident differential pressure is maintained across the containment isolation barrier.
- Systems that are required to maintain the plant in a safe condition during the ILRT are operable in their normal mode and are not vented. Also, systems that are normally filled with water and operating under post-LOCA conditions are not vented. Results of local leakage rate tests of penetrations associated with these systems are added to the ILRT results.

6.2.6.2 Containment Penetration Leakage Rate Test (Type B)

Containment penetrations whose designs incorporate resilient seals, bellows, gaskets, or sealant compounds; air-locks and air-lock door seals; equipment and access hatch seals; and electrical penetration canisters receive preoperational and periodic Type B leakage rate tests in accordance with 10 CFR 50 Appendix J. Containment penetrations subject to Type B tests are listed in Table 6.2-47. The local leak detection tests of Type B and Type C (Subsection 6.2.6.3) are completed prior to the preoperational or periodic Type A tests.

Type B tests are performed at containment peak accident pressure, P_{ac}, by local pressurization using either the pressure-decay or flowmeter method. For the pressure-decay method, a test volume is pressurized with air or nitrogen to at least P_{ac}. The rate of decay of pressure of the know test volume is monitored to calculate the leakage rate. For the flowmeter method, the required test pressure is maintained in the test volume by making up air, nitrogen, or water (if applicable) through a calibrated flowmeter. The flowmeter fluid flow rate is the leakage rate from the test volume.

The acceptance criteria for Type B tests are given in the plant-specific Technical Specifications. The combined leakage rate of all components subject to Type B and Type C tests do not exceed 60% of L_a. If repairs are required to meet this limit, the results are reported in a separate summary to the NRC. The summary includes a description of the structural conditions of the components that contributed to failure.

In accordance with 10 CFR 50 Appendix J, Type B tests (except for air-locks) are performed during each reactor shutdown for major fuel reloading, or other convenient intervals, but in no case at intervals greater than two years. Air-locks opened when containment integrity is required are tested in manual mode within 3 days of being opened. If the air-lock is to be opened more frequently than once every 3 days, the air-lock is tested at least once every 3 days during the period of frequent openings. Air-locks are tested at initial fuel loading, and at least once every 6 months thereafter. Air-locks may be tested at full power so as to avoid shutting down. These air-locks contain no inflatable seals.

Personnel air-locks through the containment include provisions for testing the door seals and the overall air lock leakage rates. Each door includes test connections that allow the annulus between the seals to be pressurized and the pressure decay monitored to determine the leak-tight integrity of the seals. Test connections are also provided on the outer face of each bulkhead so that the entire lock interior can be pressurized and the pressure decay monitored to determine the overall lock leakage. Clamps or tiedowns are installed to keep the doors sealed during the overall lock test, because normal locking mechanisms are not designed for the full differential pressure across the door in the reverse direction. Because the restraining force on the door is not critical for the performance of the overall lock pressure test on a lock with inflatable seals, no mechanism for monitoring the force is provided. (See Subsection 6.2.8.3 for COL items.)

6.2.6.3 Containment Isolation Valve Leakage Rate Test (Type C)

Type C tests are performed on all containment isolation valves required to be tested per 10 CFR 50 Appendix J. Containment isolation valves subject to Type C tests are listed within Tables 6.2-16 through 6.2-45.

Type C tests (like Type B tests) are performed by local pressurization using either the pressure-decay or flowmeter method. The test pressure is applied in the same direction as when the valve is required to perform its safety function, unless it can be shown that results from tests with pressure applied in a different direction are equivalent or conservative. For the pressure-decay method, test volume is pressurized with air or nitrogen to at least P_{ac}. The rate of decay of pressure of the know test volume is monitored to calculate the leakage rate. For the flowmeter method, the required test pressure is maintained in the test volume by making up air, nitrogen, or water (if applicable) through a calibrated flowmeter. The flowmeter fluid flow rate is the isolation valve leakage rate.

All isolation valve seats that are exposed to containment atmosphere subsequent to a LOCA are tested with air or nitrogen at containment peak accident pressure P_{ac}.

Valves that are in lines designed to be, or remain, filled with a liquid for at least 30 days subsequent to a LOCA are leakage rate tested with that liquid at a pressure not less than 1.1 P_{ac}. The liquid leakage measured is neither converted to equivalent air leakage nor added to the Type B and C test totals, but is compared to acceptable leakage rate values identified in the plant-specific Technical Specifications. The testing media used for testing containment isolation valves is identified in Section 6.2.4.

All test connections, vent lines, or drain lines consisting of double or multiple barriers (e.g., two valves in series, one valve and a cap, or one valve and a flange) that are connected between isolation valves and form a part of the containment boundary may not be Type-C tested due to their infrequent use, because the multiple barrier configurations are maintained using an administrative control program.

Type C testing shall be performed in the correct direction of the leakage path unless it can be demonstrated that testing in the reverse direction is equivalent or more conservative. The correct direction of the leakage path is from inside the containment to outside containment.

Instrument lines that penetrate containment conform to Regulatory Guide 1.11 and may not be Type-C tested. The lines that connect to the reactor coolant pressure boundary include a restricting orifice inside containment, are Seismic Category I, and terminate in Seismic Category I instruments. The instrument lines also include manual isolation valves and excess flow check valves or equivalent. These valves are normally open and are considered extensions of the containment, whose integrity is continuously demonstrated during normal operation. In addition, these lines are subject to the periodic Type A test, because they are open (up to the pressure boundary instruments) during the ILRT. Leak-tight integrity is also verified during functional and surveillance activities as well as visual observations during operator tours.

The combined leakage rate of all components subject to Type B (Subsection 6.2.6.2) and Type C tests shall not exceed 60% of L_a. If repairs are required to meet this limit, the results are reported in a separate summary to the NRC, including description of the structural conditions of the components that contributed to the failure.

6.2.6.4 Scheduling and Reporting of Periodic Tests

The periodic leakage rate test schedule requirements for Types A, B, and C tests are specified in the plant-specific Technical Specifications.

Type B and C tests are conducted at any time during normal plant operations or during shutdown periods, as long as the time interval between tests for any individual Type B or C tests does not exceed 30 months. Each time a Type B or C test is completed, the overall total leakage rate for all required Type B and C tests is updated to reflect the most recent test results. In addition to the periodic tests, any major modification or replacement of a component that is part of the primary reactor containment boundary performed after the preoperational leakage rate test will be followed by either a Type A, B, or C test (as applicable) for the area affected by the modification. Type A, B, and C test results are submitted to the NRC in the summary report approximately three months after each test.

The leakage test summary report will include descriptions of the containment inspection method, any repairs necessary to meet the acceptance criteria, and the test results.

6.2.6.5 Special Testing Requirements

Following the DW structural integrity test, a preoperational DW-to-WW leakage rate test is performed at the peak DW-to-WW differential pressure. Also, DW-to-WW leakage rate tests are conducted at a reduced differential pressure corresponding approximately to the submergence of the vents. These tests are performed following the preoperational ILRT and periodically thereafter. They verify that no paths exist for gross leakage from the DW to the WW air space that bypass the pressure suppression pool. The combination of the peak pressure and reduced pressure leakage tests also verifies adequate performance of the DW over the full range of postulated primary system break sizes.

DW-to-WW leakage rate tests are performed with the DW isolated from the WW. Valves and system lineups are the same as for the ILRT, except for paths that equalize DW and WW pressure, which are open during the ILRT and are isolated during the DW leakage test. The DW atmosphere is allowed to stabilize for a period of one hour after attaining the test pressure. Leakage rate test calculations, using the WW pressure rise method, commence after the stabilization period.

The pressure rise method is based on containment atmosphere pressure and temperature observations and the known WW volume. The leakage rate is calculated from the pressure and temperature data, WW free air volume, and elapsed time.

The plant-specific Technical Specifications specify the periodic DW-to-WW leakage rate test pressure, duration, frequency, and acceptance criteria.

6.2.7 Fracture Prevention of Containment Pressure Boundary

The reactor containment system includes the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products attendant postulated accidents.

Fracture prevention of the containment pressure boundary is assured. The ESBWR meets the relevant requirements of the following regulations:

- General Design Criterion 1 (as it relates to the quality standards for design and fabrication) - See Subsection 3.1.1.1.
- General Design Criterion 16 (as it relates to the prevention of the release of radioactivity to the environment) - See Subsection 3.1.2.7.
- General Design Criterion 51 (as it relates to the reactor containment pressure boundary design) - See Subsection 3.1.5.2.

To meet the requirements of GDC 1, 16 and 51, the ferritic containment pressure boundary materials meet the fracture toughness criteria for ASME Section III Class 2 components. These criteria provide for a uniform review, consistent with the safety function of the containment pressure boundary within the context of Regulatory Guide 1.26, which assigns correspondence of Group B Quality Standard to ASME Code Section III Class 2.

6.2.8 COL Information

6.2.8.1 Administrative Control Maintaining Containment Isolation

The COL licensee shall maintain the primary containment boundary by administrative controls in accordance with Subsection 6.2.6.3.

6.2.8.2 Wetwell-to-Drywell Vacuum Breaker Protection

The COL applicant/licensee shall propose for NRC staff review, appropriate design features providing complete structural shielding of vacuum breaker valves from pool swell loads. The structural shielding features shall be designed for pool swell loads determined based on the methodology approved for Mark II/III designs.

6.2.8.3 Containment Penetration Leakage Rate Test (Type B)

The COL licensee shall perform Type B leakage rate tests in conformance with 10 CFR 50 Appendix J for containment penetrations whose designs incorporate resilient seals, bellows, gaskets, or sealant compounds, airlocks and lock door seals, equipment and access hatch seals, and electrical canisters, and other such penetrations. (See Subsection 6.2.6.2.)

6.2.9 References

6.2-1 GE Nuclear Energy, "TRACG Application for ESBWR," NEDC-33083P-A, Class III, (Proprietary), March 2005.

6.2-2 Galletly, G.D., "A Simple Design Equation for Preventing Buckling in Fabricated Torispherical Shells under Internal Pressure," ASME Journal of Pressure Vessel Technology, Vol.108, November 1986.

Table 6.2-1

Containment Design Parameters

<u>Design Conditions</u>:	
Upper and Lower Drywell	
Design Pressure	310 kPa(g) [45 psig] / 414 kPa [60 psia]
Design Temperature	171°C (340°F)
Internal/External Differential Pressure	-20.7 kPa(d) [-3.0 psid]
Drywell to Wetwell Differential Pressure	241 kPa(d) [35 psid]/ -20.7 kPa(d) [-3.0 psid]
Inerting Gas	Nitrogen (with ≤ 3% Oxygen by Volume)
Wetwell	
Design Pressure	310 kPa(g) [45 psig] / 414 kPa [60 psia]
Design Temperature	121°C (250°F)
Inerting Gas	Nitrogen (with ≤ 3% Oxygen by Volume)
Horizontal Vent System	
Design Pressure	310 kPa(g) [45 psig] / 414 kPa [60 psia]
Design Temperature	171°C (340°F)
Containment Leak Rates	
Maximum Containment Leakage Excluding MSIV Leakage)	0.5% of Containment Volume per 24 hours Pressure 310 kPa(g) [45 psig]

Table 6.2-2

Containment Conditions During Normal Operation

Upper and Lower Drywell	
Pressure during Normal Operation	
Nominal	106.5 kPa (15.45 psia)
Maximum	110.3 kPa (16.0 psia)
Minimum	101.4 kPa (14.7 psia)
Temperature during Normal Operation	
Upper Drywell (Average)	57.2°C (135°F)
Lower Drywell (Average)	57.2°C (135°F)
Relative Humidity during Normal Operation	
Nominal	50%
Drywell Pressure SCRAM Initiation Setpoint	13.8 kPa gauge (2.0 psig)
Wetwell	
Pressure during Normal Operation	
Nominal	106.6 kPa (15.45 psia)
Maximum	110.3 kPa (16.0 psia)
Minimum	101.4 kPa (14.7 psia)
Suppression Pool Temperature during Normal Operation	
Maximum	43.3°C (110°F)
Hot Standby Maximum	54.4°C (130°F)
Gas Space Conditions, during Normal Operation	
Temperature	43.3°C (110°F)
Humidity	100%

Table 6.2-3

Containment Major Configuration Data

Drywell	
Upper Drywell Free Gas Volume	6016 m^3 (~212500 ft^3)
Lower Drywell Free Gas Volume	1190 m^3 (~42020 ft^3)
Wetwell	
Free gas space volume Normal water level	5432 m^3 (~191800 ft^3)
Suppression Pool Volume (includes vents) at normal water level	4424 m^3 (~156200 ft^3)
Suppression Pool surface area	
Pool surface only	799 m^2 (86000 ft^2)
Vertical vents (Total of 12 vents)	13.6 m^2 (146 ft^2)
Suppression Pool Depth at High Water Level	5.5 m (18 ft)
Suppression Pool Depth at Nominal Water Level	5.45 m (17.9 ft)
Suppression Pool Depth at Low Water Level	5.4 m (17.7)
GDCS Pools	
Total Water Volume (per pool for pools at 90 and 270 degrees) at Normal water level	560 m^3 (~19800 ft^3)
Total Water Volume (for pool at 180 degrees) at Normal water level	739 m^3 (~26100 ft^3)
Non-Drainable Water Volume (per pool for pools at 90 and 270 degrees)	25 m^3 (883 ft^3)
Non-Drainable Water Volume (for pool at 180 degrees)	34 m^3 (~1200 ft^3)
Pool Surface Area (per pool for pools at 90 and 270 degrees)	84.8 m^2 (913 ft^2)
Pool Surface Area (for pool at 180 degrees)	112 m^2 (1206 ft^2)

Table 6.2-4

Major Design Parameters of Vent System

Vertical Vents (Flow Channels)		
Total Number of Vertical Flow Channels	12	
Inside Diameter	1.2 m (3.9 ft)	
Height	12.54 m (41.1 ft)	
Horizontal Vents		
Number of Vents per Vertical Vent	3	
Total Number	36	
Inside Diameter	0.700 m (2.30 ft)	
	Submergence at NWL	Height above Pool Floor
Top Row (centerline)	1.95 m (6.4 ft)	3.5 m (11.48 ft)
Middle Row (centerline)	3.32 m (10.9 ft)	2.13 m (6.99 ft)
Bottom Row (centerline)	4.69 m (15.4 ft)	0.76 m (2.49 ft)

Table 6.2-5

Summary of Calculated Results for A Feedwater Line Break with Failure of One DPV

	Peak DW Pressure (a)	Peak DW Pressure (g)	Margin to Design Pressure of 45 psig (%)
Nominal Case	322 kPa (46.7 psia)	221 kPa (gauge) (32.0 psig)	29
Bounding Case	342 kPa (49.6 psia)	241 kPa (gauge) (34.9 psig)	22

	Short-term DW Temperature (C)	Long-term DW Temperature (C)	Long-term WW Temperature (C)
Nominal Case	168	141	90
Bounding Case	170	144	87

Table 6.2-6

Plant Initial Conditions Considered in the Containment DBA Cases

No.	Plant Parameter	Nominal Value	Bounding Value
1	RPV Power	100%	102%
2	WW relative humidity	100%	100%
3	PCC pool level	4.8 m	4.8 m
4	PCC pool temperature	110°F (316.5°K)	110°F (316.5°K)
5	DW Pressure	14.7 psia (101.3 kPa)	16.0 psia (110.3 kPa)
6	DW Temperature	115°F (319.3°K)	115°F (319.3°K)
7	WW Pressure	14.7 psia (101.3 kPa)	16.0 psia (110.3kPa)
8	WW Temperature	110°F (316.5°K)	110°F (316.5°K)
9	Suppression pool Temp.	110°F (316.5°K)	110°F (316.5°K)
10	GDCS pool temperature	115°F (319.3°K)	115°F (319.3°K)
11	Suppression pool level	5.45 m	5.50 m
12	GDCS pool level	6.60 m	6.60 m
13	DW relative humidity	20%	20%
14	RPV pressure	1040 psia (7.17 MPa)	1055 psia (7.274 MPa)
15	RPV Water Level	NWL	NWL+0.3m

Table 6.2-7
Operational Sequence of ECCS For A Feedwater Line Break with Failure of One DPV (Nominal Case)

Time (sec)	**Events**
~0	Guillotine break of feedwater line inside containment; normal auxiliary power assumed to be lost; feedwater is lost; Scram signal initiated.
~1	High drywell pressure setpoint for ADS is reached.
~2	Loss of normal auxiliary power confirmed; reactor scram initiated; IC initiated.
~5	Level 3 is reached; Reactor receives second signal to scram.
~6	Maximum (first peak) DW pressure of 322 kPa (46.7 psia) is reached.
~9	Level 2 is reached; Reactor isolation timer initiated.
~14	Level 1.5 is reached; Reactor isolation initiated; ADS/GDCS timer initiated.
~17	IC drain valve begins to open.
~24	Level 1.5 signal confirmed; ADS-SRV actuation begins.
~33	IC drain valve fully open.
~74	DPV actuation begins; SLC system signaled to start.
~100	Minimum chimney water level is reached.
~164	GDCS timer timed out. GDCS injection valves open.
~250	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
from ~250 to 259000 (72 hrs)	RPV water level remains higher than Level 0.5. Therefore equalizing line valves are not expected to open for this event.
~ 140000 (39 hrs)	PCC pool level drops below the elevation of 29.6 m; top ¼ portion of the PCC tube length becomes uncover; Connection valves open to allow the water from the Dryer/Separator storage pool to flow into the PCC pools.
~ 250000 (70 hrs)	Long-term DW pressure (2nd peak) of 290 kPa (42 psia) is reached.

Table 6.2-8

Model Parameters for Containment Bounding Calculation

No.	Model Parameter	Base value	Distribution	Uncertainty (1 sigma)	Bounding case	Bounding value used
1	Crit Flow (PIRT84)	1.0	Normal	9.5%	- 2 sigma	0.81
2	Decay Heat Mult.	1.0	Normal	~0.05	+ 2 sigma	D.H. + 2 sigma
3	Surf. HT (PIRT07)	100	Uniform	1 to 200	Lower bound	1
4	PCC inlet Loss (k/A^2)	$1065m^{-4}$	Normal	$260.0m^{-4}$	+ 2 sigma	$1585m^{-4}$
5	PCC HT (PIRT78)	1.0	Normal	7.9% (bias – 6.0%)	- 2 sigma	0.902
6	VB Loss (k/A^2)	$169.0m^{-4}$	Normal	$21.18m^{-4}$	+ 2 sigma	$211.4m^{-4}$

Table 6.2-9

ESBWR Design Feature for Severe Accident Control

Design Feature	**Function**: Prevention / Mitigation	**Purpose/Description**
Isolation Condenser System (IC)	Prevention	Controls reactor pressure. First line of defense against accidents.
Automatic Depression System (ADS)	Prevention	Depressurizes reactor pressure vessel and prevents high-pressure core-melt accident. Minimizes probability of direct containment heating.
Compact containment design with minimum penetrations. Lower drywell kept dry.	Mitigation	Containment isolation with minimum leakage. High retention of aerosols. Fuel Coolant Interactions and Ex-Vessel Steam Explosions minimized.
Lower drywell configuration	Mitigation	Lower drywell floor provides spreading area for cooling of molten core.
Containment overpressure protection system	Mitigation	A system that provides additional defense in depth.
Deluge Lines Flooder system supplying water to BiMAC Device.	Mitigation	Provides additional cooling for corium on the floor from top and bottom that minimizes Ex-Vessel Core-Concrete Interactions and provides long term cooling of debris.
PCC heat exchangers	Mitigation	Filter aerosols - minimize offsite dose.
Passive Containment Cooling System (PCCS)	Prevention /Mitigation	Provides long term containment cooling. Keeps pressure within design limits.
Suppression pool and Airspace	Prevention /Mitigation	Suppression pool is heat sink. Scrubs aerosols. Airspace volume is sized for 100% metal water reaction.
GDCS in wetwell configuration	Prevention /Mitigation	Increases airspace volume to handle non-condensable gas release in SA.
Inerted containment with nitrogen.	Prevention /Mitigation	Prevents hydrogen detonation

Table 6.2-10

Passive Containment Cooling Design Parameters

Number of PCCS Loops-	Six (6)
Heat Removal Capacity for Each Loop-	11 MWt Nominal for pure saturated steam at a pressure of 308 kPa (absolute) (45 psia) and temperature of 134°C (273.2 °F) condensing inside tubes with an outside pool water temperature of 102°C.
System Design Pressure-	758.5 kPa(g) (110 psig)
System Design Temperature-	171°C (340°F)

Table 6.2-11

RWCU/SDC Break Locations

Break Case	Description
1	Break in RWCU/SDC Non-Regenerative Heat Exchanger (NRHX) Room
2	Break in NRHX Valve Room
3	Break in Regenerative Heat Exchanger Room
4	Break in RWCU/SDC Pump Rooms

Table 6.2-12

Subcompartment Vent Path Designation

FIGURE	FLOW PATH NO.	CELL FROM	CELL TO	TYPE	FLOW PATH		LOSS COEFFICIENTS				FLOW DIRECTION	BLOW-OUT PRESSURE (Pa g)	COMMENTS
					LENGTH (m)	AREA (m^2)	t	FORWARD DIRECTION	REVERSE DIRECTION	ANALYSIS LOSS COEFFICIENT			
6.2-24	1	1	2	DOOR	2.0	4.0	0.24	1.56	1.61	0.79	NO BLOW-OUT PANEL		two-way path
6.2-24	2	2	3	DOOR	1.0	4.0	0.97	1.51	1.24	0.69	FORWARD	1.034E4	--
6.2-24	3	2	3	DOOR	1,0	4.0	0.97	1.52	1.26	0.7	FORWARD	1.034E4	--
6.2-24	4	3	4	DOOR	0.7	4.0	1.13	1.25	1.24	0.62	FORWARD	1.034E4	--
6.2-24	5	3	5	DOOR	0.5	4.0	1.19	1.31	1.32	0.66	FORWARD	1.034E4	--
6.2-24	6	6	7	DOOR	2.0	4.0	0.24	1.56	1.61	0.79	NO BLOW-OUT PANEL		two-way path
6.2-24	7	7	5	DOOR	1.0	4.0	0.97	1.52	1.26	0.7	FORWARD	1.034E4	--
6.2-24	8	7	5	DOOR	1.0	4.0	0.97	1.51	1.24	0.69	FORWARD	1.034E4	--
6.2-24	9	8	4	DOOR	2.0	4.0	0.24	1.43	1.47	0.72	FORWARD	1.034E4	--
6.2-24	10	9	10	DOOR	2.0	4.0	0.24	1.49	1.48	0.74	FORWARD	1.034E4	--
6.2-24	11	10	5	DOOR	0.7	4.0	1.13	1.25	1.24	0.62	FORWARD	1.034E4	--
6.2-24	12	10	4	DOOR	0.5	4.0	1.19	1.24	1.24	0.62	FORWARD	1.034E4	--
6.2-24	13	11	10	DOOR	2.0	4.0	0.24	1.48	1.51	0.75	FORWARD	1.034E4	--
6.2-24	14	12	16	OPEN SPACE	1.0	5.0	0.97	0.90	0.47	0.34	NO BLOW-OUT PANEL		two-way path
6.2-24	15	13	16	OPEN SPACE	1.0	5.0	0.97	0.90	0.47	0.34	NO BLOW-OUT PANEL		two-way path
6.2-24	16	14	16	OPEN SPACE	1.0	5.0	0.97	0.93	0.48	0.35	NO BLOW-OUT PANEL		two-way path
6.2-24	17	15	16	OPEN SPACE	1.0	5.0	0.97	0.93	0.48	0.35	NO BLOW-OUT PANEL		two-way path
6.2-24	18	3	12	OPEN SPACE	1.0	5.0	0.97	0.47	0.90	0.34	NO BLOW-OUT PANEL		two-way path
6.2-24	19	5	14	OPEN SPACE	1.0	5.0	0.97	0.48	0.93	0.35	NO BLOW-OUT PANEL		two-way path
6.2-24	20	10	15	OPEN SPACE	1.0	5.0	0.97	0.48	0.93	0.35	NO BLOW-OUT PANEL		two-way path
6.2-24	21	4	13	OPEN SPACE	1.0	5.0	0.97	0.47	0.90	0.34	NO BLOW-OUT PANEL		two-way path
6.2-24	22	16	17	BLOW-OUT PANEL	1.0	16.0	0.97	2.46	2.44	1.23	FORWARD	1.034E4	Assumed blow-out to atmosphere

Table 6.2-13

Reactor Coolant Pressure Boundary Influent Lines Penetrating Drywell

Influent Line		Inside Drywell	Outside Drywell
1	Feedwater	CV	SPCV
2	IC Condensate	2 x NMOV 2 x MOV	None (closed loop outside containment)
3	Standby liquid control	CV	CV & SQUIB
4	IC Purge Line	1 x CV 1 x MOV	None (closed loop outside containment)

CV = Check valve
SPCV = Check valve - spring closed with air operator to open
MOV = Motor-operated valve
SQUIB = Squib valve - normally closed with solid metal isolation barrier
NMOV = Nitrogen motor operated valve

Table 6.2-14

Reactor Coolant Pressure Boundary Effluent Lines Penetrating Drywell

Effluent Line		Inside Drywell	Outside Drywell
1	Main steam	NOV	AOV
2	IC steam supply	1 x MOV 1 x NMO	None (closed loop outside containment)
3	RWCU/SDC system	NOV	MOV

ADV = Air-operated valve
MOV = Motor-operated valve
NOV = Nitrogen-operated valve
NMO = Nitrogen rotary motor operated

Table 6.2-15

Legend For Tables 6.2-16 through 6.2-43

(a) Termination Region of the leakage through packing/stem only for outboard valves:

a1	=	Reactor Building
a2	=	Main Steam Tunnel

(b) Termination Region outside containment of the leakage past seat:

b1	=	Pool open to reactor building
b2	=	External environment
b3	=	Main Condenser
b4	=	Isolation Condenser pool
b5	=	Reactor building
b6	=	Close loop outside containment
b7	=	Radwaste System

(c) Value Operator Types:

AO/Ac	=	Air-operated valve with accumulator
AO	=	Air-operated valve without accumulator
NO/Acc	=	Nitrogen-operated valve with accumulator
NO	=	Nitrogen-operated valve without accumulator
NMO/Acc	=	Nitrogen-motor operated valve with accumulator
NMO	=	Nitrogen-motor operated valve without accumulator
MO	=	Motor-operated valve
SO	=	Solenoid-operated valve

(d) Isolation Signal Codes:

B	Reactor vessel low water level - Level 2
C	Reactor vessel low water level - Level 1
D	Main steamline high flow rate
E	Turbine inlet low pressure
F	Main steamline tunnel high ambient temperature
G	Turbine area steamline high ambient temperature
H	High DW pressure
I	IC/PCC pool high radiation
K	IC lines high flow
L	Low main condenser vacuum
M	High flow in the RWCU/SDC loop
N	Standby Liquid Control System operating
P	Remote manual
Q	Self actuating
R	Local manual (By Hand)
S	High radiation in DW sump line
T	High HVAC radiation exhaust from refueling area or from Reactor Building.

Table 6.2-16

Containment Isolation Valve Information for the Nuclear Boiler System

Main Steam Line A

Penetration Identification			
Valve No.	F001A	F002A	F016A
Applicable Basis	GDC 55	GDC 55	GDC 55
Fluid	Steam	Steam	Steam/Water
Line Size	700 mm	700 mm.	50 mm.
Leakage Through Packing[a]	N/A	(a_2)	(a_2)
Leakage Past Seat[b]	(b_3)	(b_3)	(b_3)
Location	Inboard	Outboard	Outboard
Valve Type	Globe	Globe	Globe
Operator[c]	NO/Acc	AO/Acc	MO
Normal Position	Open	Open	Open
Shutdown Position	Closed	Closed	Open
Post-Acc Position	Closed	Closed	Closed
Power Fail Position	Closed	Closed	As is
Cont. Iso. Signal[d]	B,C,D,E,F,G,L	B,C,D,E,F,G,L	B,C,D,E,F,G,L
Primary Actuation	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual
Closure Time (sec)	3.0-4.5	3.0-4.5	15
Power Source	Div. 1/2	Div. 1/2	Div. 1

Note: For explanation of codes, see legend in Table 6.2-15.

Table 6.2-17

Containment Isolation Valve Information for the Nuclear Boiler System

Main Steam Line B

Penetration Identification			
Valve No.	F001B	F002B	F016B
Applicable Basis	GDC 55	GDC 55	GDC 55
Fluid	Steam	Steam	Steam/Water
Line Size	700 mm.	700 mm	50 mm
Leakage Through Packing[(a)]	N/A	(a_2)	(a_2)
Leakage Past Seat[(b)]	(b_3)	(b_3)	(b_3)
Location	Inboard	Outboard	Outboard
Valve Type	Globe	Globe	Gate
Operator[(c)]	NO/Acc	AO/Acc	MO
Normal Position	Open	Open	Open
Shutdown Position	Closed	Closed	Open
Post-Acc Position	Closed	Closed	Closed
Pwr Fail Position	Closed	Closed	As is
Cont. Iso. Signal[(d)]	B,C,D,E,F,G,L	B,C,D,E, F,G,L	B,C,D,E,F,G,L
Primary Actuation	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual
Closure Time (sec)	3.0-4.5	3.0-4.5	15
Power Source	Div. 1/2	Div. 1/2	Div. 1

Note: For explanation of codes, see legend in Table 6.2-15.

Table 6.2-18

Containment Isolation Valve Information for the Nuclear Boiler System

Main Steam Line C

Penetration Identification			
Valve No.	F001C	F002C	F016C
Applicable Basis	GDC 55	GDC 55	GDC 55
Fluid	Steam	Steam	Steam/Water
Line Size	700 mm	700 mm	50 mm
Leakage Through Packing[(a)]	N/A	(a_2)	(a_2)
Leakage Past Seat[(b)]	(b_3)	(b_3)	(b_3)
Location	Inboard	Outboard	Outboard
Valve Type	Globe	Globe	Globe
Operator[(c)]	NO/Acc	AO/Acc	MO
Normal Position	Open	Open	Open
Shutdown Position	Closed	Closed	Open
Post-Acc Position	Closed	Closed	Closed
Power Fail Position	Closed	Closed	As is
Cont. Iso. Signal[(d)]	B,C,D,E,F,G,L	B,C,D,E,F,G,L	B,C,D,E,F,G,L
Primary Actuation	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual
Closure Time (sec)	3.0-4.5	3.0-4.5	15
Power Source	Div. 1/2	Div. 1/2	Div. 1

Note: For explanation of codes, see legend in Table 6.2-15.

Table 6.2-19

Containment Isolation Valve Information for the Nuclear Boiler System

Main Steam Line D

Penetration Identification			
Valve No.	F001D	F002D	F016D
Applicable Basis	GDC 55	GDC 55	GDC 55
Fluid	Steam	Steam	Steam/Water
Line Size	700 mm	700 mm	50 mm
Leakage Through Packing[a]	N/A	(a_2)	(a_2)
Leakage Past Seat[b]	(b_3)	(b_3)	(b_3)
Location	Inboard	Outboard	Outboard
Valve Type	Globe	Globe	Globe
Operator[c]	NO/Acc	AO/Acc	MO
Normal Position	Open	Open	Open
Shutdown Position	Closed	Closed	Open
Post-Acc Position	Closed	Closed	Closed
Power Fail Position	Closed	Closed	As is
Cont. Iso. Signal[d]	B,C,D,E,F,G,L	B,C,D,E,F,G,L	B,C,D,E,F,G,L
Primary Actuation	Automatic	Automatic	Automatic
Secary Actuation	Remote manual	Remote manual	Remote manual
Closure Time (sec)	3.0-4.5	3.0-4.5	15
Power Source	Div. 1/2	Div. 1/2	Div. 1

Note: For explanation of codes, see legend in Table 6.2-15.

Table 6.2-20

Containment Isolation Valve Information

for the Nuclear Boiler System Main Steam Line Drains

Penetration Identification	**MSL # 3**	
Valve No.	F010	F011
Applicable Basis	GDC 55	GDC 55
Fluid	Steam/water	Steam/water
Line Size	80 mm	80 mm
Leakage Through Packing[(a)]	N/A	(a_2)
Leakage Past Seat[(b)]	(b_3)	(b_3)
Location	Inboard	Outboard
Valve Type	Globe	Gate
Operator[(c)]	NO	MO
Normal Position	Open	Open
Shutdown Position	Open	Open
Post-Acc Position	Closed	Closed
Power Fail Position	Closed	As is
Cont. Iso. Signal[(d)]	B,C,D,E,F,G,L	B,C,D,E,F,G,L
Primary Actuation	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual
Closure Time (sec)	15	15
Power Source	Div.2	Div.1

Note: For explanation of codes, see legend in Table 6.2-15.

Table 6.2-21

Containment Isolation Valve Information for the Nuclear Boiler System

Feedwater Line A

Penetration Identification	**MFWL # 1**	
Valve No.	F103A	F102A
Applicable Basis	GDC 55	GDC 55
Fluid	Water	Water
Line Size	550 mm	550 mm
Leakage Through Packing[a]	N/A	(a_2)
Leakage Past Seat[b]	N/A	(b_3)
Location	Inboard	Outboard
Valve Type	Check	Stop Check
Operator[c]	N/A	AO
Normal Position	Open	Open
Shutdown Position	N/A	Open
Post-Acc Position	N/A	Closed
Power Fail Position	N/A	N/A
Cont. Iso. Signal[d]	Q	Q
Primary Actuation	Self	Air to open
Secondary Actuation	N/A	Remote manual
Closure Time (sec)	N/A	N/A
Power Source	N/A	Spring

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-22

Containment Isolation Valve Information for the Nuclear Boiler System

Feedwater Line B

Penetration Identification	**MFWL # 2**	
Valve No.	F103B	F102B
Applicable Basis	GDC 55	GDC 55
Fluid	Water	Water
Line Size	550 mm	550 mm
Leakage Through Packing[a]	N/A	(a_2)
Leakage Past Seat[b]	N/A	(b_3)
Location	Inboard	Outboard
Valve Type	Check	Stop Check
Operator[c]	N/A	AO
Normal Position	Open	Open
Shutdown Position	N/A	Open
Post-Acc Position	N/A	Closed
Power Fail Position	N/A	N/A
Cont. Iso. Signal[d]	Q	Q
Primary Actuation	Self	Air to open
Secondary Actuation	N/A	Remote manual
Closure Time (sec)	N/A	N/A
Power Source	N/A	Spring

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-23

Containment Isolation Valve Information for Isolation Condenser Loop A

Penetration Identification	**IC # 1A**[1]		**IC # 2A1**	
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Fluid	Steam	Steam	Condensate	Condensate
Line Size	350mm	350mm	200mm	200mm
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[2] b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard
Valve Type	Gate	Gate	Gate	Gate
Operator[(c)]	MO	NMO/Acc	MO	NMO/Acc
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Acc Position	Open[3]	Open[3]	Open[3]	Open[3]
Power Fail Position	As is	As is	As is	As is
Cont. Iso. Signal[(d)]	I,K	I,K	I,K	I,K
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)**	< 60	< 60	< 35	< 35
Power Source	Div. 1	Div. 2	Div. 1	Div. 2

* With respect to meeting the intent of US NRC 10CFR 50, Appendix A, General Design Criteria 55, the closed loop safety-related IC loop outside the containment is a "passive" substitute for an open "active" valve outside the containment. This closed loop substitute for an open isolation valve outside the containment implicitly provides greater safety. The

[1]Two in series valves
[2]Piping of I.C. Quality Group B Design
[3]Except on IC pipe or tube failure

combination of an already isolated loop outside the containment plus the two series automatic isolation valves inside the containment comply with the intent of the isolation guidelines of 10 CFR50, App.A, Criterion 55 and 56.

** Closing Times are estimates and will be confirmed during detailed design stage

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-24
Containment Isolation Valve Information for the Isolation Condenser Loop A

Penetration Identification	**IC # 3A**[4]		**IC # 4A**[5]				**IC # 5A**[4]	
Valve Location	Upper Header Vent	Upper Header Vent	Lower Header Vent	Lower Header Vent	Lower Header Bypass Vent	Lower Header Bypass Vent	Purge line	Excess Flow Purge
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Fluid	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases
Line Size	20mm	20mm	20mm	20mm	20mm	20mm	20mm	20mm
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[6]b6	b6	b6	b6	b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard
Valve Type	Globe	Globe	Globe	Globe	Globe	Globe	Globe	Excess
Operator[(c)]	SO	SO	SO	SO	MO	MO	MO	Flow CV
Normal Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Shutdown Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open

[4]Two in series valves
[5]Two in series valves (F009/F010) in parallel with two in series valves (F011/F012)
[6]Closed barrier outside containment

Table 6.2-24

Containment Isolation Valve Information for the Isolation Condenser Loop A

Penetration Identification	**IC # 3A[4]**		**IC # 4A[5]**				**IC # 5A[4]**	
Valve Location	Upper Header Vent	Upper Header Vent	Lower Header Vent	Lower Header Vent	Lower Header Bypass Vent	Lower Header Bypass Vent	Purge line	Excess Flow Purge
Post-Acc Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Power Fail Position	Closed	Closed	Closed	Closed	As is	As is	As is	As is
Cont. Iso. Signal[(d)]	P	P	P	P	P	P	P	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Diff Pressure
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Closure Time (sec)**	< 30	< 30	< 30	< 30	< 30	< 30	< 30	< 30
Power Source	Div. 1	Div. 1	Div. 4	Div. 4	Div. 1	Div. 1	Div. 1	N/A

* The piping and valve arrangement for these lines meet the intent of 10CFR50, App. A, GDC 55 because there are two normally closed valves in series in the line that leads from the suppression chamber back to the closed IC loop outside the containment.

** Closing Times are estimates and will be confirmed during detailed design stage

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-25

Containment Isolation Valve Information for the Isolation Condenser System Loop B

Penetration Identification	**IC # 1B**[7]		**IC # 2B**[7]	
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Fluid	Steam	Steam	Condensate	Condensate
Line Size	350mm	350mm	200mm	200mm
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[8] b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard
Valve Type	Gate	Gate	Gate	Gate
Operator[(c)]	MO	NMO/Acc	MO	NMO/Acc
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Acc Position	Open[9]	Open[3]	Open[3]	Open[3]
Power Fail Position	As is	As is	As is	As is
Cont. Iso. Signal[(d)]	I,K	I,K	I,K	I,K
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)**	< 60	< 60	< 35	< 35
Power Source	Div. 1	Div. 2	Div. 1	Div. 2

* With respect to meeting the intent of US NRC 10CFR 50, Appendix A, General Design Criteria 55, the closed loop safety-related IC loop outside the containment is a "passive" substitute for an open "active" valve outside the containment. This closed loop substitute for an open isolation valve outside the containment implicitly provides greater safety. The

7 Two in series valves
8 Closed barrier outside containment (Piping of I.C. Quality Group B Design)
9 Except on IC pipe or tube failure

combination of an already isolated loop outside the containment plus the two series automatic isolation valves inside the containment comply with the intent of the isolation guidelines of 10 CFR50, App.A, Criterion 55 and 56.

** Closing Times are estimates and will be confirmed during detailed design stage

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-26

Containment Isolation Valve Information for the Isolation Condenser Loop B

Penetration Identification	**IC # 3B**[10]		**IC # 4B**[11]				**IC # 3B**[10]	
Valve Location	Upper Header Vent	Upper Header Vent	Lower Header Vent	Lower Header Vent	Lower Header Bypass Vent	Lower Header Bypass Vent	Purge line	Excess Flow Purge
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Fluid	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases
Line Size	20mm	20mm	20mm	20mm	20mm	20mm	20mm	20mm
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[12]b6	b6	b6	b6	b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard
Valve Type	Globe	Globe	Globe	Globe	Globe	Globe	Globe	Excess
Operator[(c)]	SO	SO	SO	SO	MO	MO	MO	Flow CV
Normal Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Shutdown Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open

10Two in series valves
11Two in series valves (F009/F010) in parallel with two in series valves (F011/F012)
12Closed barrier outside containment

Table 6.2-26

Containment Isolation Valve Information for the Isolation Condenser Loop B

Penetration Identification	**IC # 3B[10]**		**IC # 4B[11]**				**IC # 3B[10]**	
Valve Location	Upper Header Vent	Upper Header Vent	Lower Header Vent	Lower Header Vent	Lower Header Bypass Vent	Lower Header Bypass Vent	Purge line	Excess Flow Purge
Post-Acc Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Power Fail Position	Closed	Closed	Closed	Closed	As is	As is	As is	As is
Cont. Iso. Signal[(d)]	P	P	P	P	P	P	P	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Diff Pressure
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Closure Time (sec)**	< 30	< 30	< 30	< 30	< 30	< 30	< 30	< 30
Power Source	Div. 2	Div. 2	Div. 1	Div. 1	Div. 2	Div. 2	Div. 2	N/A

* The piping and valve arrangement for these lines meet the intent of 10CFR50, App. A, GDC 55 because there are two normally closed valves in series in the line that leads from the suppression chamber back to the closed IC loop outside the containment.

** Closing Times are estimates and will be confirmed during detailed design stage

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-27

Containment Isolation Valve Information for the Isolation Condenser Loop C

Penetration Identification	**IC # 1C**[13]		**IC # 2C1**[13]*	
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Fluid	Steam	Steam	Condensate	Condensate
Line Size	350 mm.	350 mm.	350 mm	350 mm
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[14] b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard
Valve Type	Gate	Gate	Gate	Gate
Operator[(c)]	MO	NMO/Acc	MO	NMO/Acc
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Acc Position	Open[15]	Open[15]	Open[15]	Open[15]
Power Fail Position	As is	As is	As is	As is
Cont. Iso. Signal[(d)]	I,K	I,K	I,K	I,K
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)**	< 60	< 60	< 35	< 35
Power Source	Div. 1	Div. 2	Div. 1	Div. 2

* With respect to meeting the intent of US NRC 10CFR 50, Appendix A, General Design Criteria 55, the closed loop safety-related IC loop outside the containment is a "passive" substitute for an open "active" valve outside the containment. This closed loop substitute for an open isolation valve outside the containment implicitly provides greater safety. The combination of an already isolated loop outside the containment plus the two series automatic isolation valves inside the containment comply with the intent of the isolation guidelines of 10 CFR50, App.A, Criterion 55 and 56.

** Closing Times are estimates and will be confirmed during detailed design stage

Note: For explanation of codes, see legend on Table 6.2-15.

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[13] Two in series valves
[14] Closed barrier outside containment (Piping of I.C. Quality Group B Design)
[15] Except on IC pipe or tube failure

Table 6.2-28

Containment Isolation Valve Information for the Isolation Condenser Loop C

Penetration Identification	**IC # 3C[16]**		**IC # 4C[17]**				**IC # 5C[16]**	
Valve Location	Upper Header Vent	Upper Header Vent	Lower Header Vent	Lower Header Vent	Lower Header Bypass Vent	Lower Header Bypass Vent	Purge line	Excess Flow Purge
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Fluid	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases
Line Size	20mm	20mm	20mm	20mm	20mm	20mm	20mm	20mm
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[18]	b6	b6	b6	b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard
Valve Type	Globe	Globe	Globe	Globe	Globe	Globe	Globe	Excess
Operator[(c)]	SO	SO	SO	SO	MO	MO	MO	Flow CV
Normal Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Shutdown Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Post-Acc Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open

[16]Two in series valves
[17]Two in series valves (F009/F010) in parallel with two in series valves (F011/F012)
[18]Closed barrier outside containment

Table 6.2-28

Containment Isolation Valve Information for the Isolation Condenser Loop C

Penetration Identification	**IC # 3C[16]**		**IC # 4C[17]**				**IC # 5C[16]**	
Valve Location	Upper Header Vent	Upper Header Vent	Lower Header Vent	Lower Header Vent	Lower Header Bypass Vent	Lower Header Bypass Vent	Purge line	Excess Flow Purge
Power Fail Position	Closed	Closed	Closed	Closed	As is	As is	As is	As is
Cont. Iso. Signal[(d)]	P	P	P	P	P	P	P	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Diff Pressure
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Closure Time (sec)**	< 30	< 30	< 30	< 30	< 30	< 30	< 30	< 30
Power Source)	Div. 3	Div. 3	Div. 2	Div. 2	Div. 3	Div. 3	Div. 3	N/A

* The piping and valve arrangement for these lines meet the intent of 10CFR50, App. A, GDC 55 because there are two normally closed valves in series in the line that leads from the suppression chamber back to the closed IC loop outside the containment.

** Closing Times are estimates and will be confirmed during detailed design stage.

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-29

Containment Isolation Valve Information for the Isolation Condenser Loop D

Penetration Identification	IC # 1D[19]		IC # 2D[19]	
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Fluid	Steam	Steam	Condensate	Condensate
Line Size	350 mm	350 mm	350 mm	350 mm
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[20] b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard
Valve Type	Gate	Gate	Gate	Gate
Operator[(c)]	MO	NMO/Acc	MO	NMO/Acc
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Acc Position	Open[21]	Open[21]	Open[21]	Open[21]
Power Fail Position	As is	As is	As is	As is
Cont. Iso. Signal[(d)]	I,K	I,K	I,K	I,K
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)**	< 60	< 60	< 35	< 35
Power Source	Div. 1	Div. 2	Div. 1	Div. 2

* With respect to meeting the intent of US NRC 10CFR 50, Appendix A, General Design Criteria 55, the closed loop safety-related IC loop outside the containment is a "passive" substitute for an open "active" valve outside the containment. This closed loop substitute for an open isolation valve outside the containment implicitly provides greater safety. The combination of an already isolated loop outside the containment plus the two series automatic isolation valves inside the containment comply with the intent of the isolation guidelines of 10 CFR50, App.A, Criterion 55 and 56.

** Closing Times are estimates and will be confirmed during detailed design stage

Note: For explanation of codes, see legend on Table 6.2-15.

[19] Two in series valves
[20] Closed barrier outside containment (Piping of I.C. Quality Group B Design)
[21] Except on IC pipe or tube failure

Table 6.2-30

Containment Isolation Valve Information for the Isolation Condenser Loop D

Penetration Identification	**IC # 3B[22]**		**IC # 4B[23]**				**IC # 5D[22]**	
Valve Location	Upper Header Vent	Upper Header Vent	Lower Header Vent	Lower Header Vent	Lower Header Bypass Vent	Lower Header Bypass Vent	Purge line	Excess Flow Purge
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Fluid	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/steam /Non Cond Gases	Cond/Steam /Non Cond Gases	Cond/Steam /Non Cond Gases
Line Size	20mm	20mm	20mm	20mm	20mm	20mm	20mm	20mm
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[24] b6	b6	b6	b6	b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard
Valve Type	Globe	Globe	Globe	Globe	Globe	Globe	Globe	Excess
Operator[(c)]	SO	SO	SO	SO	MO	MO	MO	Flow CV
Normal Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Shutdown Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Post-Acc Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open

[22]Two in series valves
[23]Two in series valves (F009/F010) in parallel with two in series valves (F011/F012)
[24]Closed barrier outside containment

Table 6.2-30

Containment Isolation Valve Information for the Isolation Condenser Loop D

Penetration Identification	IC # 3B[22]		IC # 4B[23]				IC # 5D[22]	
Valve Location	Upper Header Vent	Upper Header Vent	Lower Header Vent	Lower Header Vent	Lower Header Bypass Vent	Lower Header Bypass Vent	Purge line	Excess Flow Purge
Power Fail Position	Closed	Closed	Closed	Closed	As is	As is	As is	As is
Cont. Iso. Signal[(d)]	P	P	P	P	P	P	P	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Diff Pressure
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Closure Time (sec)**	< 30	< 30	< 30	< 30	< 30	< 30	< 30	< 30
Power Source	Div. 4	Div. 4	Div. 3	Div. 3	Div. 4	Div. 4	Div. 4	N/A

* The piping and valve arrangement for these lines meet the intent of 10CFR50, App. A, GDC 55 because there are two normally closed valves in series in the line that leads from the suppression chamber back to the closed IC loop outside the containment.

** Closing Times are estimates and will be confirmed during detailed design stage

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-31
Containment Isolation Valve Information for the Reactor Water Cleanup/Shutdown Cooling System

Penetration Identification	RWCU # 1				RWCU # 2			
Valve No.	F002A	F003A	F007A	F008A	F002B	F003B	F007B	F008B
Applicable Basis	GDC 55	GDC 55	GDC 55	GDC 55	GDC 55	GDC 55	GDC 55	GDC 55
Fluid	Water	Water	Water	Water	Water	Water	Water	Water
Line Size	300 mm	300 mm.	150 mm	150 mm	300 mm	300 mm.	150 mm	150 mm
Leakage Through Packing(a)	N/A	(a_1)	(a_1)	(a_1)	(a_1)	(a_1)	(a_1)	(a_1)
Leakage Past Seat(b)1	(b_3)	(b_3)	(b_3)	(b_3)	(b_3)	(b_3)	(b_3)	(b_3)
Location	Inboard	Outboard	Inboard	Outboard	Inboard	Outboard	Inboard	Outboard
Valve Type	Gate	Gate	Gate	Gate	Gate	Gate	Gate	Gate
Operator(c)	NO	AO	NO	AO	NO	AO	NO	AO
Normal Position 1	O/C	O/C	O/C	O/C	O/C	O/C	C	C
Shutdown Position	O/C	O/C	O/C	O/C	O/C	O/C	O/C	O/C
Post-Acc Position	C	C	C	C	C	C	C	C
Power Fail Position	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed
Cont. Iso. Signal(d)1	B,C,F,M,N	B,C,F,M,N	B,C,F,M,N	B,C,F,M,N	B,C,F,M,N	B,C,F,M,N	B,C,F,M,N	B,C,F,M,N
Primary Actuation	Automatic	Automatic	Automatic	Automatic	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)	30	30	15	15	30	30	15	15
Power Source	Div. 2	Div. 1	Div. 2	Div 1	Div. 2	Div. 1	Div. 2	Div. 1

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-32

Containment Isolation Valve Information for the Standby Liquid Control System

Penetration Identification	**SLC # 1**	
Valve No.	F005	F004
Applicable Basis	GDC 55	GDC 55
Fluid	Boron/Water	Boron/Water
Line Size	80 mm	80 mm
Leakage Through Packing(a)	N/A	(a1)
Leakage Past Seat(b)	(b5)	(b5)
Location	Inboard	Outboard
Valve Type	Check	Check
Operator(c)	N/A	N/A
Normal Position	Closed	Closed
Shutdown Position	Closed	Closed
Post-Acc Position	Operable	Operable
Power Fail Position	N/A	N/A
Cont. Iso. Signal(d)	Q	Q
Primary Actuation	Self	Self
Secondary Actuation	N/A	N/A
Closure Time (sec)	N/A	N/A
Power Source	N/A	N/A

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-33

Containment Isolation Valve Information for the Fuel and Auxiliary Pools Cooling System, Part 1

Penetration Identification	**FAPC # 1**	**FAPC # 2**	
Valve No.	F321	F306	F307
Applicable Basis	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water
Line Size	250 mm.	250 mm.	250 mm.
Leakage Through Packing[(a)]	(a_1)	(a_1)	N/A
Leakage Past Seat[(b)]	b6	b6	b6
Location	Outboard	Outboard	Inboard
Valve Type	Gate	Gate	Check
Operator[(c)]	AO	MO	N/A
Normal Position	Closed[25]	Closed[25]	N/A
Shutdown Position	Closed[25]	Closed[25]	N/A
Post-Acc Position	Closed[26]	Closed[27]	N/A
Power Fail Position	Closed	As is	N/A
Cont. Iso. Signal[(d)]	N/A	N/A	Q
Primary Actuation	Remote manual	Remote manual	Self
Secondary Actuation	Remote manual	Remote manual	N/A
Closure Time (sec)	*	*	N/A
Power Source	Div. 2	Div. 1	N/A

* Closing Times will be confirmed during detailed design stage

Note: For explanation of codes, see legend on Table 6.2-15.

[25]The valve is open occasionally for the suppression pool cooling and cleanup function.
[26]The valve is opened remote manually for performing LPCI, Drywell Spray, or Suppression Pool Cooling function if required.
[27]The valve is opened remote manually for performing Suppression Pool Cooling function if required.

Table 6.2-34

Containment Isolation Valve Information for the Fuel and Auxiliary Pools Cooling System Part 2

Penetration Identification	FAPC # 3		FAPC # 4	
Valve No.	F323	F324	F303	F304
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water	Water
Line Size	250 mm.	250 mm	250 mm.	250 mm.
Leakage Through Packing[(a)]	N/A	(a_1)	(a_1)	N/A
Leakage Past Seat[(b)]	b6	b6	b6	b6
Location	Inboard	Outboard	Outboard	Inboard
Valve Type	Gate	Gate	Gate	Check
Operator[(c)]	NO	MO	MO	N/A
Normal Position	Closed[28]	Closed[28]	Closed[28]	N/A
Shutdown Position	Closed	Closed	Closed	N/A
Post-Acc Position	Closed	Closed	Closed	N/A
Power Fail Position	Closed	As is	As is	N/A
Cont. Iso. Signal[(d)]	B,C,H	B,C,H	B,C,H	Q
Primary Actuation	Automatic	Automatic	Automatic	Self
Secondary Actuation	Remote manual	Remote manual	Remote manual	N/A
Closure Time (sec)	*	*	*	N/A
Power Source	Div. 2	Div. 1	Div. 1	N/A

* Closing Times will be confirmed during detailed design stage.

Note: For explanation of codes, see legend on Table 6.2-15.

[28]The valve is open occationally for GDCS pools cooling and cleanup function.

Table 6.2-35

Containment Isolation Valve Information for the Fuel and Auxiliary Pools Cooling System

Penetration Identification	**FAPC # 5**	
Valve No.	F309	F310
Applicable Basis	GDC 56	GDC 56
Fluid	Water	Water
Line Size	250 mm.	250 mm.
Leakage Through Packing[(a)]	(a_1)	N/A
Leakage Past Seat[(b)]	b6	b6
Location	Outboard	Inboard
Valve Type	Globe	Check
Operator[(c)]	MO	N/A
Normal Position	Closed	N/A
Shutdown Position	Closed	N/A
Post-Acc. Position	Closed[29]	N/A
Power Fail Position	As is	N/A
Cont. Iso. Signal[(d)]	B,C,H	N/A
Primary Actuation	Electrical	Self
Secondary Actuation	Remote manual	N/A
Closure Time(sec)	*	N/A
Power Source	Div. 1	N/A

* Closing Times will be confirmed during detailed design stage.

Note: For explanation of codes, see legend on Table 6.2-15.

[29]The valve would be opened remote manually to perform the drywell spray function if required.

Table 6.2-36

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	**CAC # 1**[30]			
Valve No.	F010	F011	F014	F015
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Air/N_2	Air/N_2	Air/N_2	Air/N_2
Line Size	400 mm	500 mm	25 mm.	25mm
Leakage Through Packing[(a)]	(a_1)	(a_1)	(a_1)	(a_1)
Leakage Past Seat[(b)]	(b_2/b_5)	(b_2/b_5)	(b_2/b_5)	(b_2/b_5)
Location	Outboard	Outboard	Outboard	Outboard
Valve Type	Butterfly	Butterfly	Globe	Globe
Operator[(c)]	AO	AO	AO	AO
Normal Position	Closed	Closed	Closed[31]	Closed[31]
Shutdown Position	Closed[32]	Closed[32]	Closed	Closed
Post-Acc Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Cont. Iso. Signal[(d)]	B,C,H,T	B,C,H,T	B,C,H,T	B,C,H,T
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)	< 30	< 30	< 5	< 5
Power Source	Div. 2	Div. 1	Div. 2	Div. 1

Note: For explanation of codes, see legend on Table 6.2-15.

[30] Two valves in series (F011/F010) in parallel with two in series valves (F015/F014).
[31] Open to purge excess pressure to prevent inadvertent reactor scram after which are closed.
[32] Open during the early stage of Ineriting/De-inerting modes to purge resident air/N2 after which are closed.

Table 6.2-37

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	**CAC # 2**[33]		
Valve No.	F006	F024	F023
Applicable Basis	GDC 56	GDC 56	GDC 56
Fluid	Air/N_2	Air/N_2	Air/N_2
Line Size	500 mm	25 mm	25 mm.
Leakage Through Packing[(a)]	(a_1)	(a_1)	(a_1)
Leakage Past Seat[(b)]	(b_2)	(b_2)	(b_2)
Location	Outboard	Outboard	Outboard
Valve Type	Butterfly	Globe	Globe
Operator[(c)]	AO	AO	AO
Normal Position	Closed	Open	Open
Shutdown Position	Open	Closed	Closed
Post-Acc Position	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed
Cont. Iso. Signal[(d)]	B,C,H,T	B,C,H,T	B,C,H,T
Primary Actuation	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual
Closure Time (sec.)	< 30	< 5	< 5
Power Source	Div. 1	Div. 2	Div. 1

Note: For explanation of codes, see legend on Table 6.2-15.

[33]Valve F006 in series with F007, valve F024 in series with F023.

Table 6.2-38

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	**CAC # 3[34]**			
Valve No.	F025	F023	F008	F009
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Air/N_2	Air/N_2	Air/N_2	Air/N_2
Line Size	25 mm	25 mm	25 mm	25 mm
Leakage Through Packing[(a)]	(a_1)	(a_1)	(a_1)	(a_1)
Leakage Past Seat[(b)]	(b_2)	(b_2)	(b_2)	(b_2)
Location	Outboard	Outboard	Outboard	Outboard
Valve Type	Globe	Globe	Butterfly	Butterfly
Operator[(c)]	AO	AO	AO	AO
Normal Position	Open	Open	Closed	Closed
Shutdown Position	Closed	Closed	Open	Open
Post-Acc Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Cont. Iso. Signal[(d)]	B,C,H,T	B,C,H,T	B,C,H,T	B,C,H,T
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)	< 5	< 5	< 30	< 30
Power Source	Div. 2	Div. 1	Div. 1	Div. 2

Note: For explanation of codes, see legend on Table 6.2-15.

[34]Valve F008 in series with F009, valve F025 in series with F023.

Table 6.2-39

Containment Isolation Valve Information for the Chilled Cooling Water System

Penetration Identification	**CWS # 1 (A+B)**		**CWS # 2 (A+B)**	
Valve No.	F001* (A+B)	F002 *(A+B)	F003* (A+B)	F004 *(A+B)
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water	Water
Line Size	3 in.	3 in.	3 in.	3 in.
Leakage Through Packing[a]	(a_1)	N/A	N/A	(a_1)
Leakage Past Seat[b]	(b_5)	(b_6)	(b_6)	(b_5)
Location	Outboard	Inboard	Inboard	Outboard
Valve Type	Gate	Gate	Gate	Gate
Operator[c]	MO	NO	NO	MO
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Acc Position	Closed	Closed	Closed	Closed
Power Fail Position	As is	Closed	Closed	As is
Cont. Iso. Signal[d]	B,C,H	B,C,H	B,C,H	B,C,H
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)	30	30	30	30
Power Source	Div. 1	Div. 2	Div. 2	Div. 1

* Temporary assigned valve numbers

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-40

Containment Isolation Valve Information for the High Pressure Nitrogen Supply System

Penetration Identification	**HPNSS#1**	
Valve No.	F011A	F010A
Applicable Basis	GDC 56	GDC 56
Fluid	N_2	N_2
Line Size	1 in.	1 in.
Leakage Through Packing[(a)]	N/A	(a_1)
Leakage Past Seat[(b)]	(b_6)	(b_5)
Location	Inboard	Outboard
Valve Type	Check	Globe
Operator[(c)]	N/A	MO
Normal Position	Open	Open
Shutdown Position	Open	Open
Post-Acc Position	Open	Open
Power Fail Position	N/A	As-Is
Cont. Iso. Signal[(d)]	Q	P
Primary Actuation	Self	Remote manual
Secondary Actuation	N/A	N/A
Closure Time (sec)	N/A	15
Power Source	N/A	Div.1

Note: For explanation of codes, see legend on Table 6.2-15

Table 6.2-41

Containment Isolation Valve Information for the High Pressure Nitrogen Gas Supply System

Penetration Identification	**HPNSS#3**	
Valve No.	F004	F003
Applicable Basis	GDC 56	GDC 56
Fluid	N_2	N_2
Line Size	1 in.	1 in.
Leakage Through Packing[a]	N/A	(a_1)
Leakage Past Seat[b]	(b_6)	(b_5)
Location	Inboard	Outboard
Valve Type	Check	Globe
Operator[c]	N/A	MO
Normal Position	Open	Open
Shutdown Position	Open	Open
Post-Acc Position	N/A	Open
Power Fail Position	N/A	As-Is
Cont. Iso. Signal[d]	Q	P
Primary Actuation	Self	Remote manual
Secondary Actuation	N/A	N/A
Closure Time (sec)	N/A	15
Power Source	N/A	Div.1

Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-42

Containment Isolation Valve Information for the Process Radiation Monitoring System

COL applicant to provide.

Table 6.2-43

Containment Isolation Valve Information for the Post Accident Sampling System

COL applicant to provide.

Table 6.2-44

Isolation Valve Provisions In The Influent Lines To The Containment

COL applicant to provide.

Table 6.2-45

Isolation Valve Provisions In The Effluent Lines From The Containment

COL applicant to provide.

Table 6.2-46

Level C Pressure Capability of Containment Steel Components

Component	**Part**	**Inner Radius** (mm)	**Thickness** (mm)	**Calculated Pressure Capability** (MPa gauge)
Drywell Head	Sleeve	5200	50	2.504
	Torispherical head	9407	40	1.182
Equipment Hatch	Sleeve	1200	16	3.465
	Head	2400	20	4.359
Personnel Airlock	Sleeve	1200	16	3.465
Wetwell Hatch	Sleeve	1000	16	4.152
	Head	2000	20	5.229

Table 6.2-47

Containment Penetrations Subject To Type B Testing

COL applicant to provide.

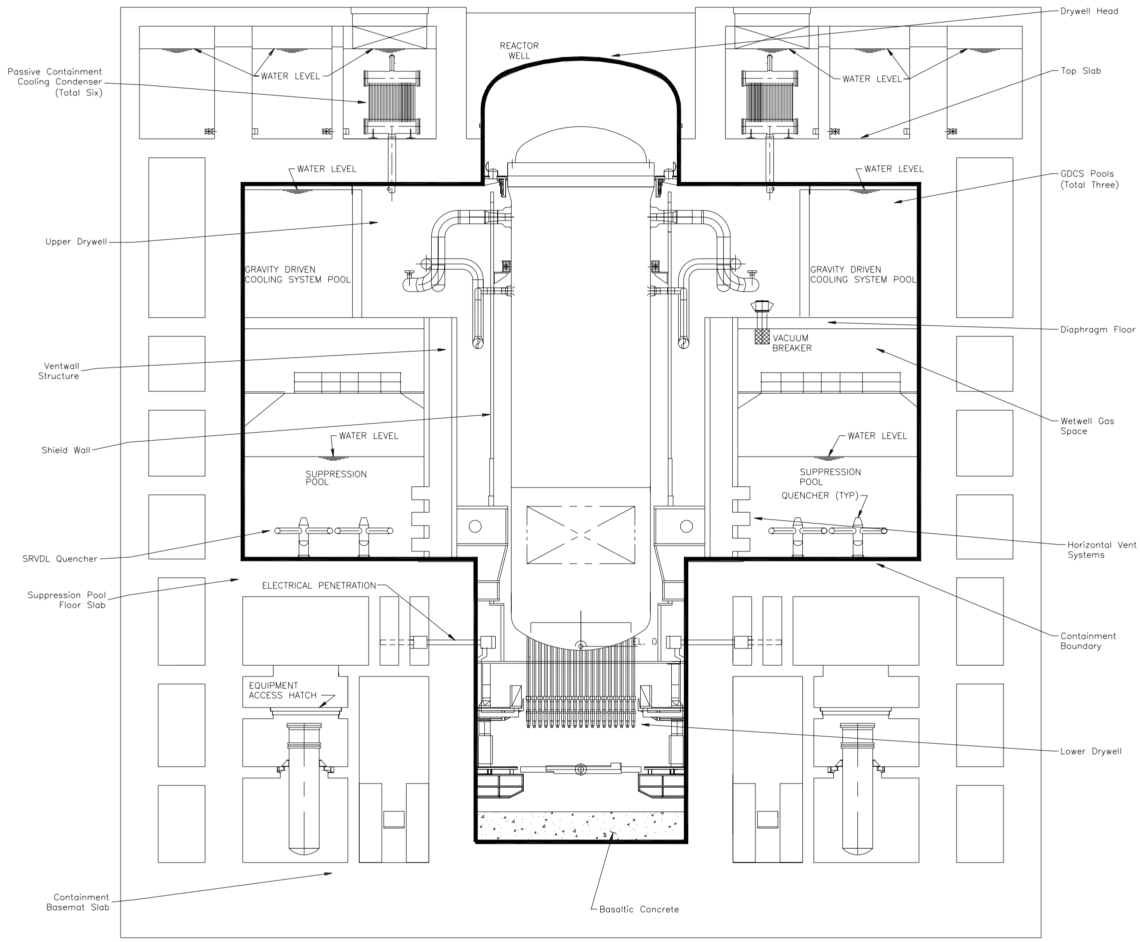


Figure 6.2-1. Containment System

Figure 6.2-2. IC/PCC Pools Configuration

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390.}}}

Figure 6.2-3. GDCS Pools Configuration

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390.}}}

Figure 6.2-4. Suppression Pool and Vent System Configuration

{{{Sensitive unclassified information provided under separate submittal per 10 CFR 2.390.}}}

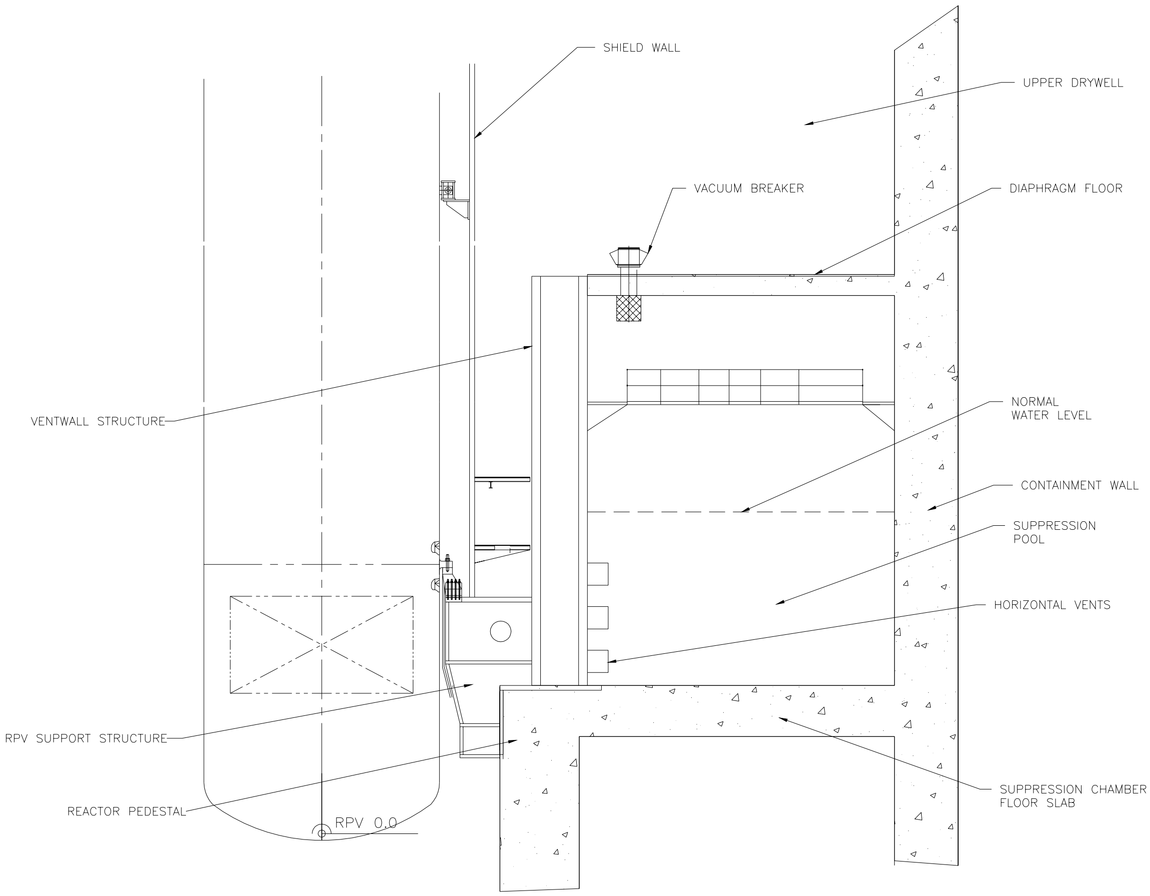


Figure 6.2-5. Horizontal Vent System Configuration

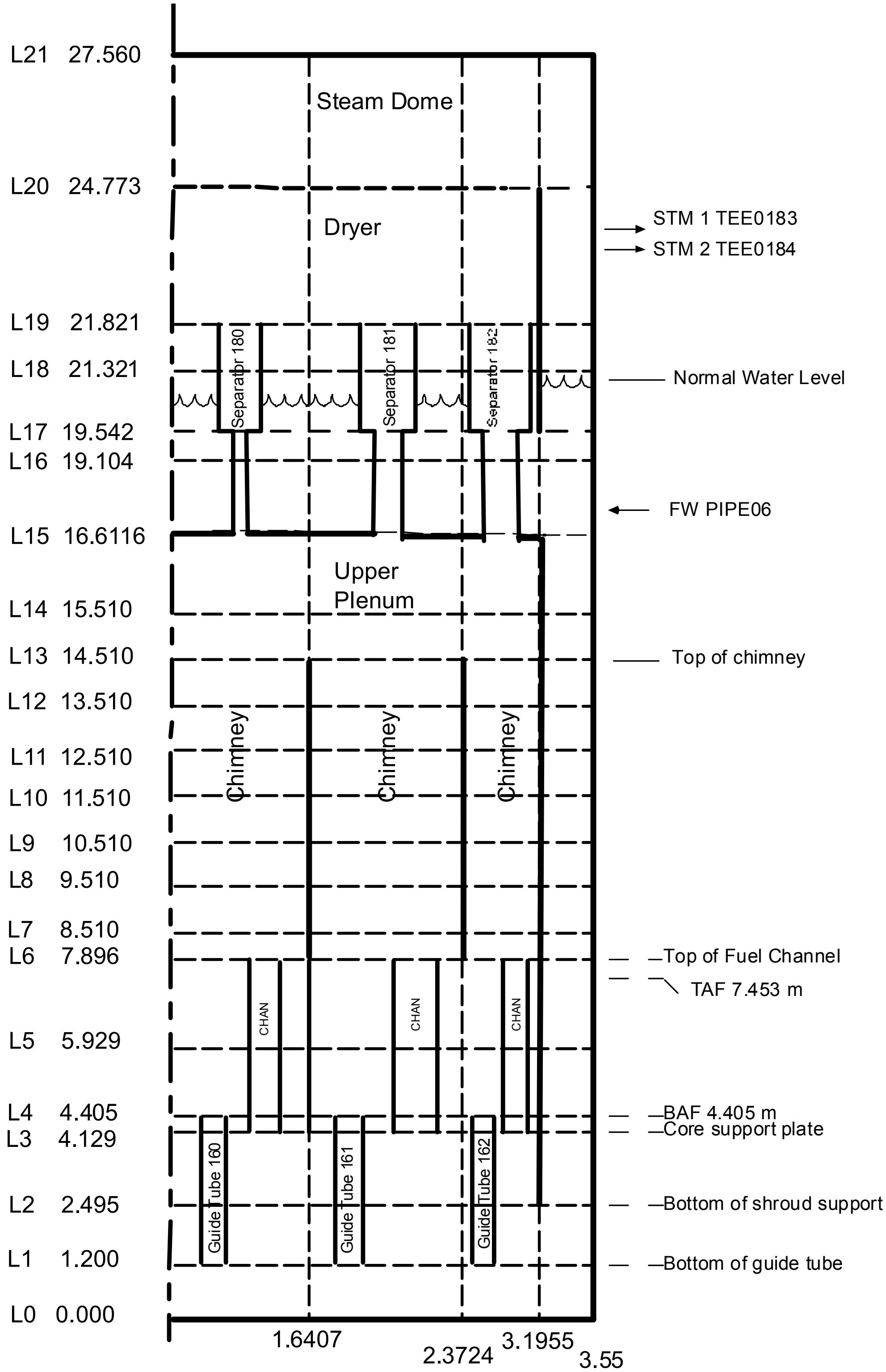


Figure 6.2-6. TRACG Nodalization of the ESBWR RPV

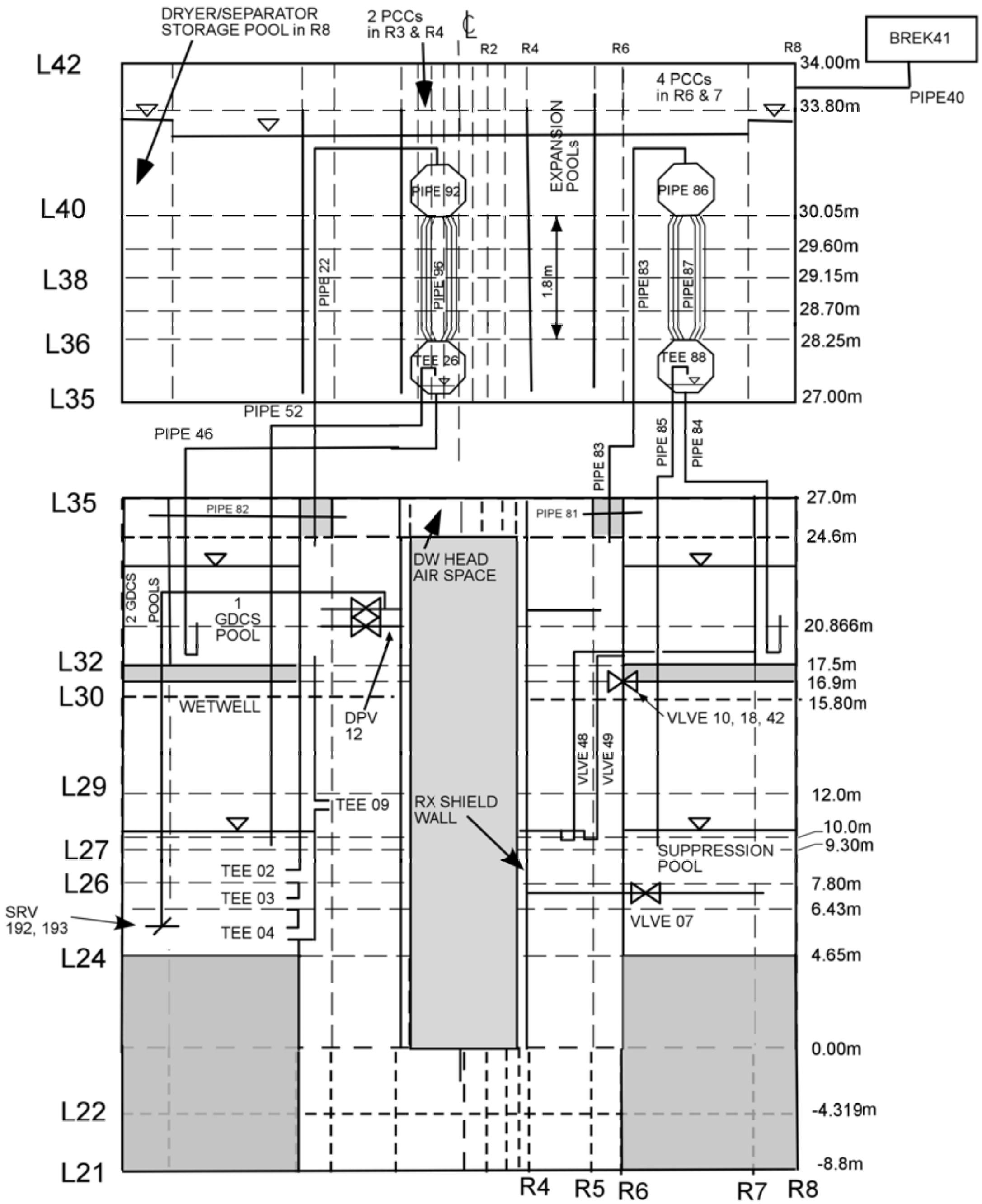


Figure 6.2-7. TRACG Nodalization of the ESBWR Containment

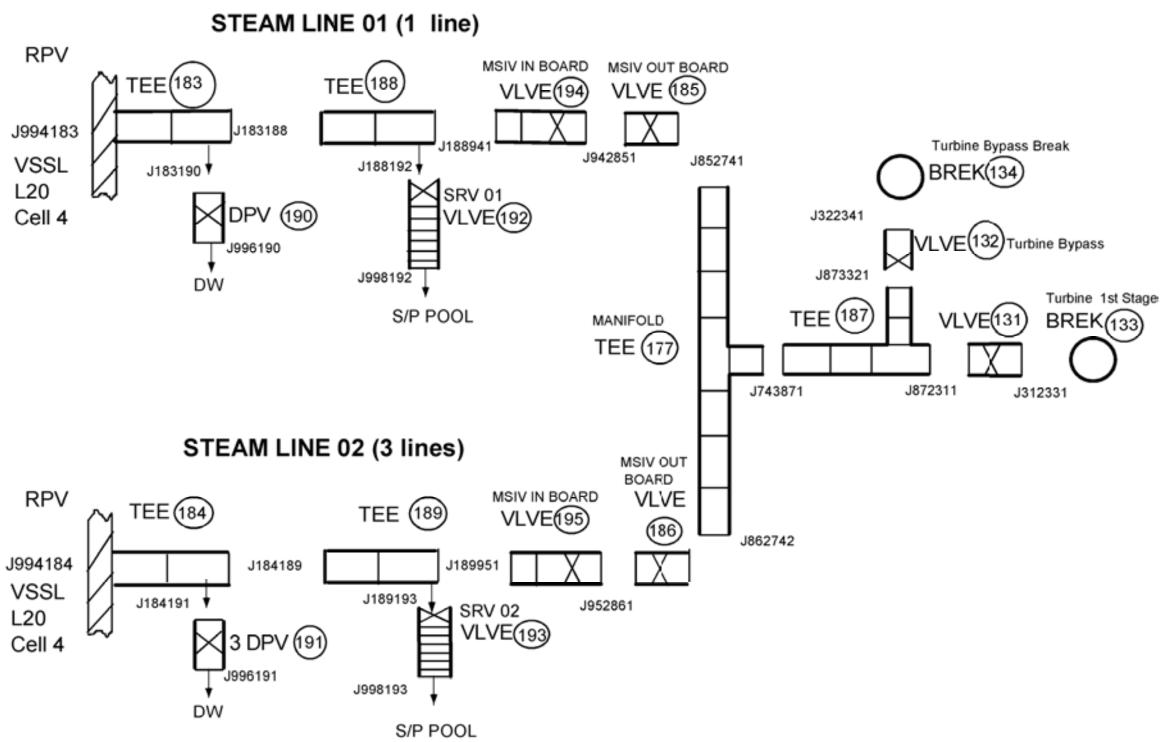


Figure 6.2-8. TRACG Nodalization of the ESBWR Main Steam Lines

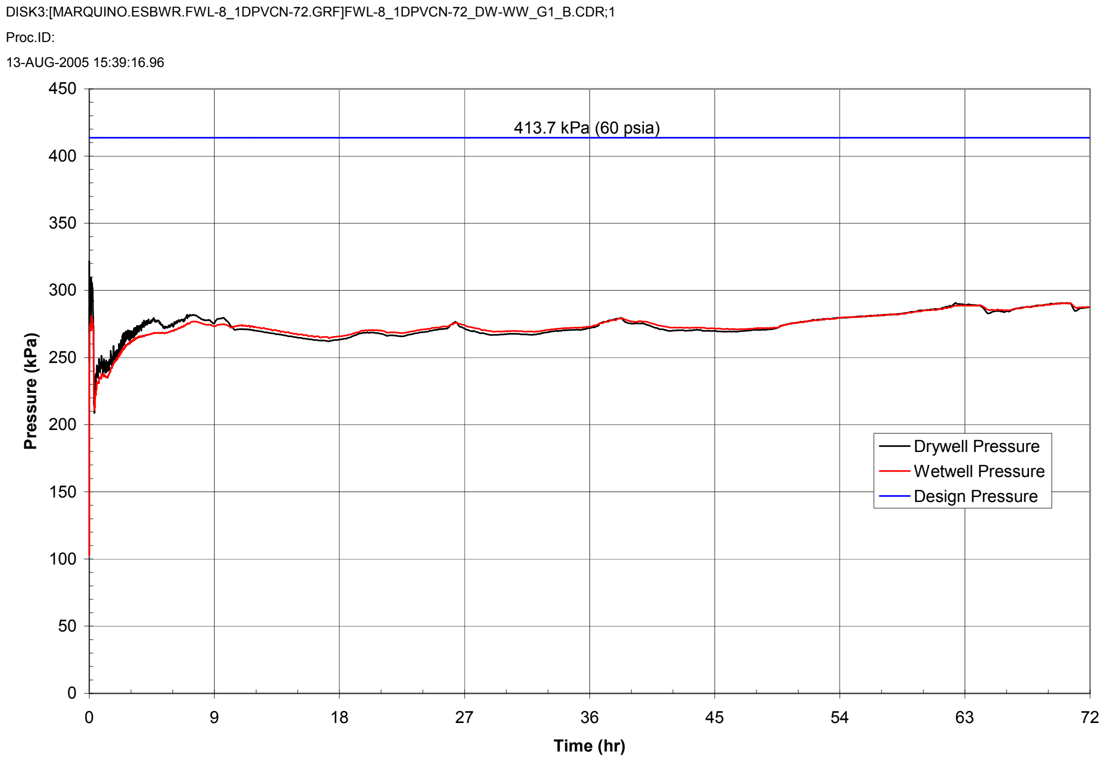


Figure 6.2-9. Feedwater Line Break (Nominal Case) — Containment Pressures

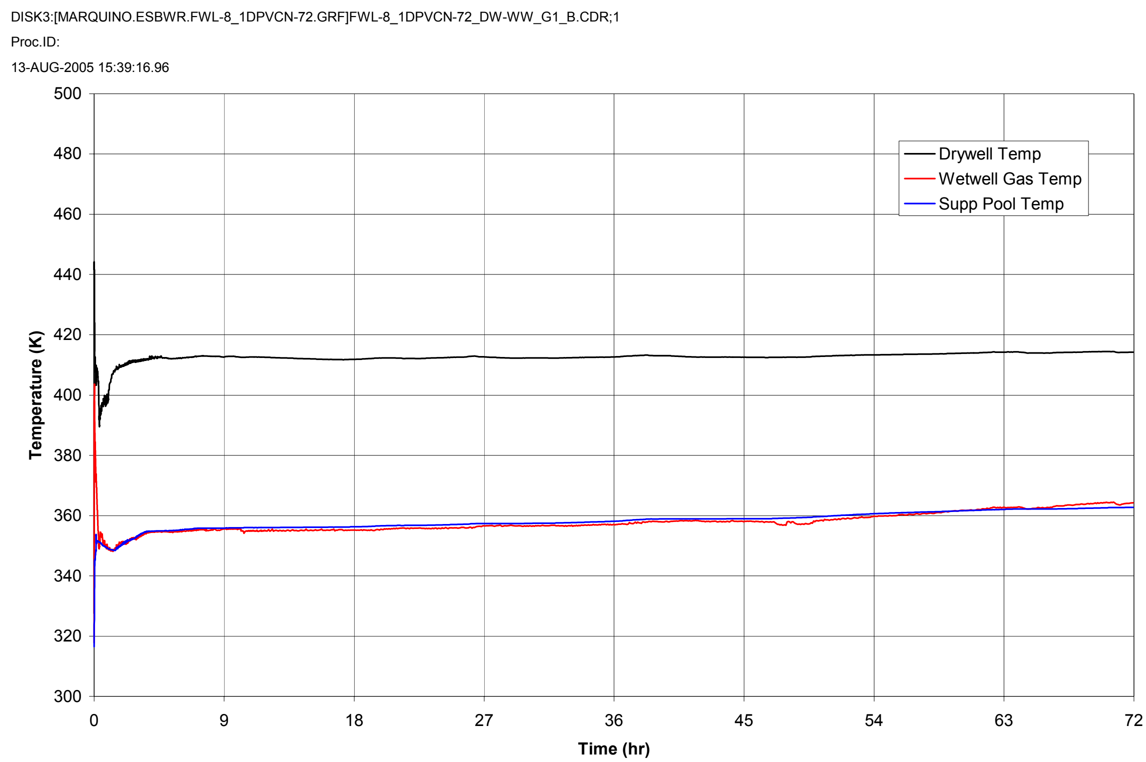


Figure 6.2-10. Feedwater Line Break (Nominal Case) — Containment Temperatures

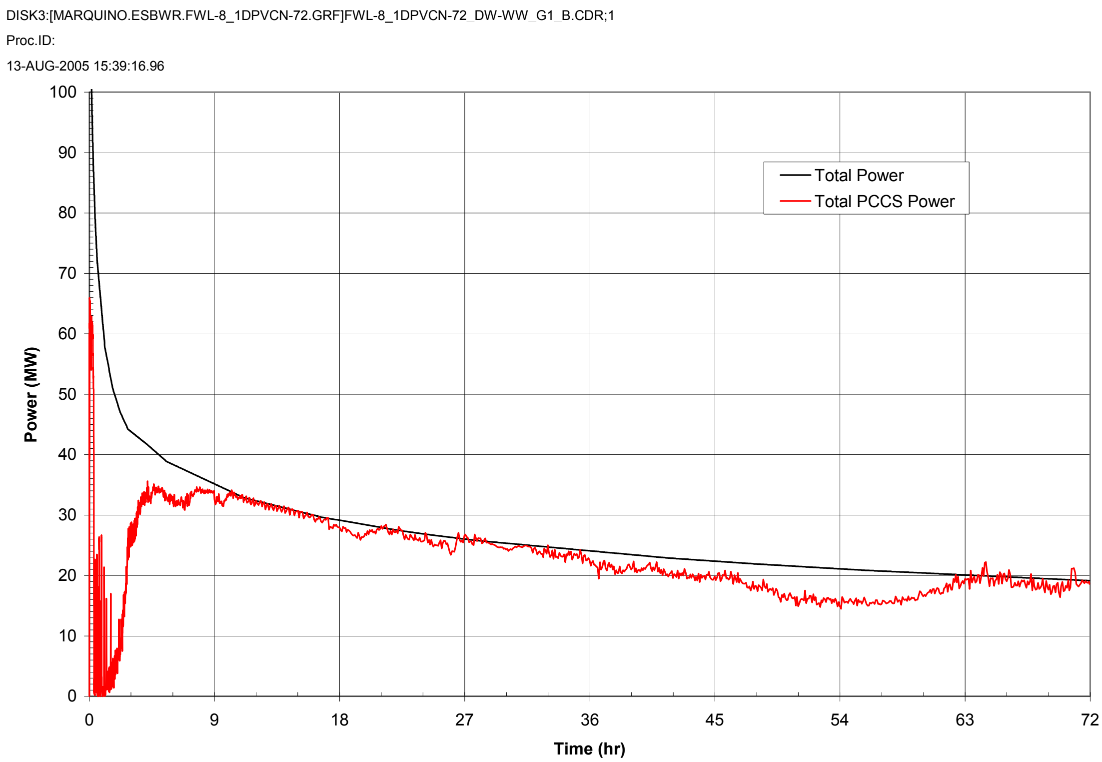


Figure 6.2-11. Feedwater Line Break (Nominal Case) — PCCS Heat Removal versus Decay Heat

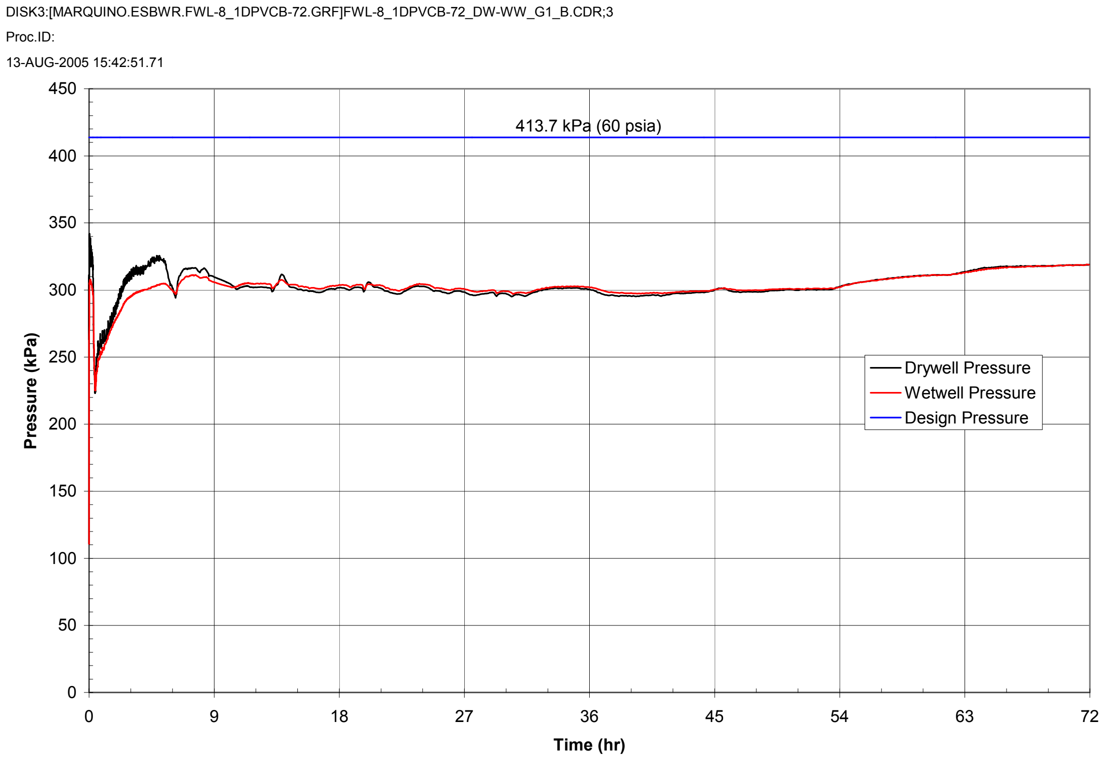


Figure 6.2-12. Feedwater Line Break (Bounding Case) — Containment Pressures

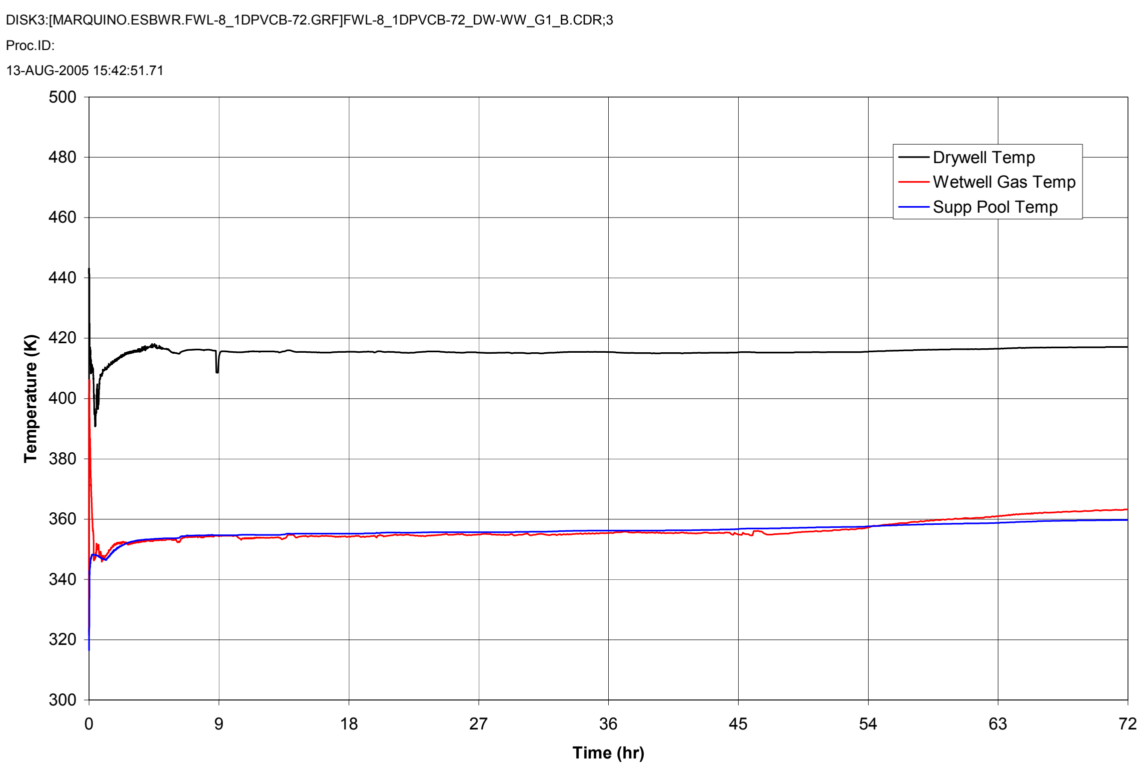


Figure 6.2-13. Feedwater Line Break (Bounding Case) — Containment Temperatures

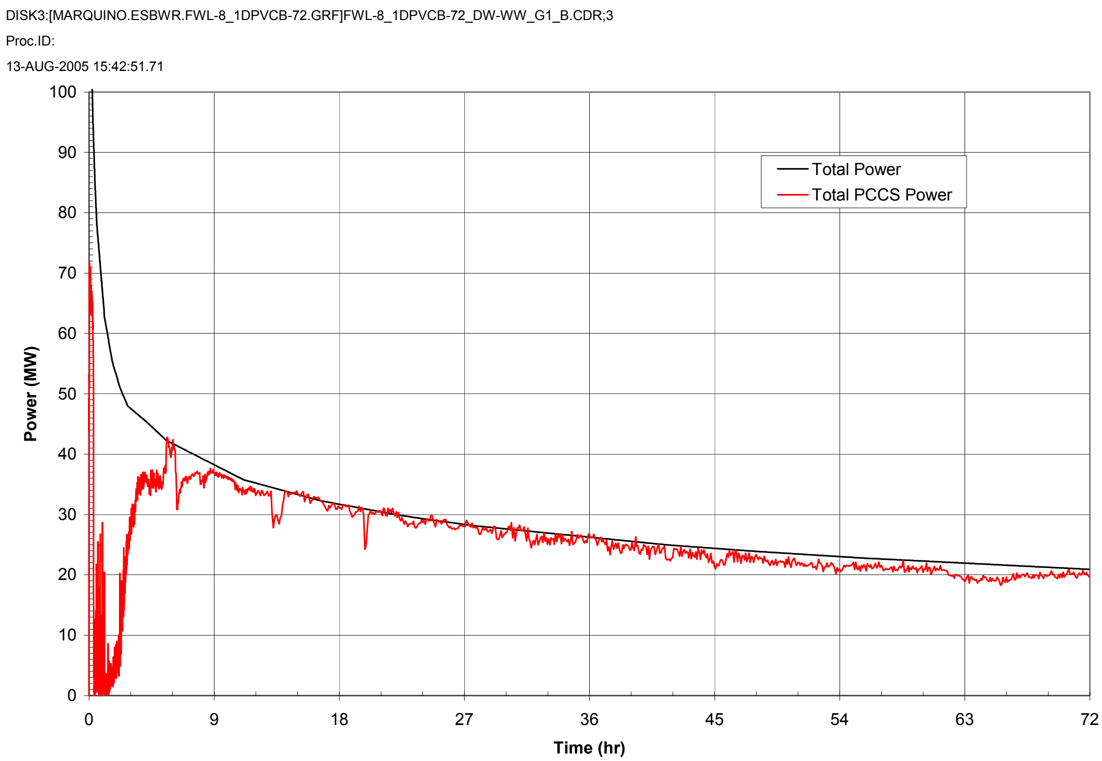


Figure 6.2-14. Feedwater Line Break (Bounding Case) — PCCS Heat Removal versus Decay Heat

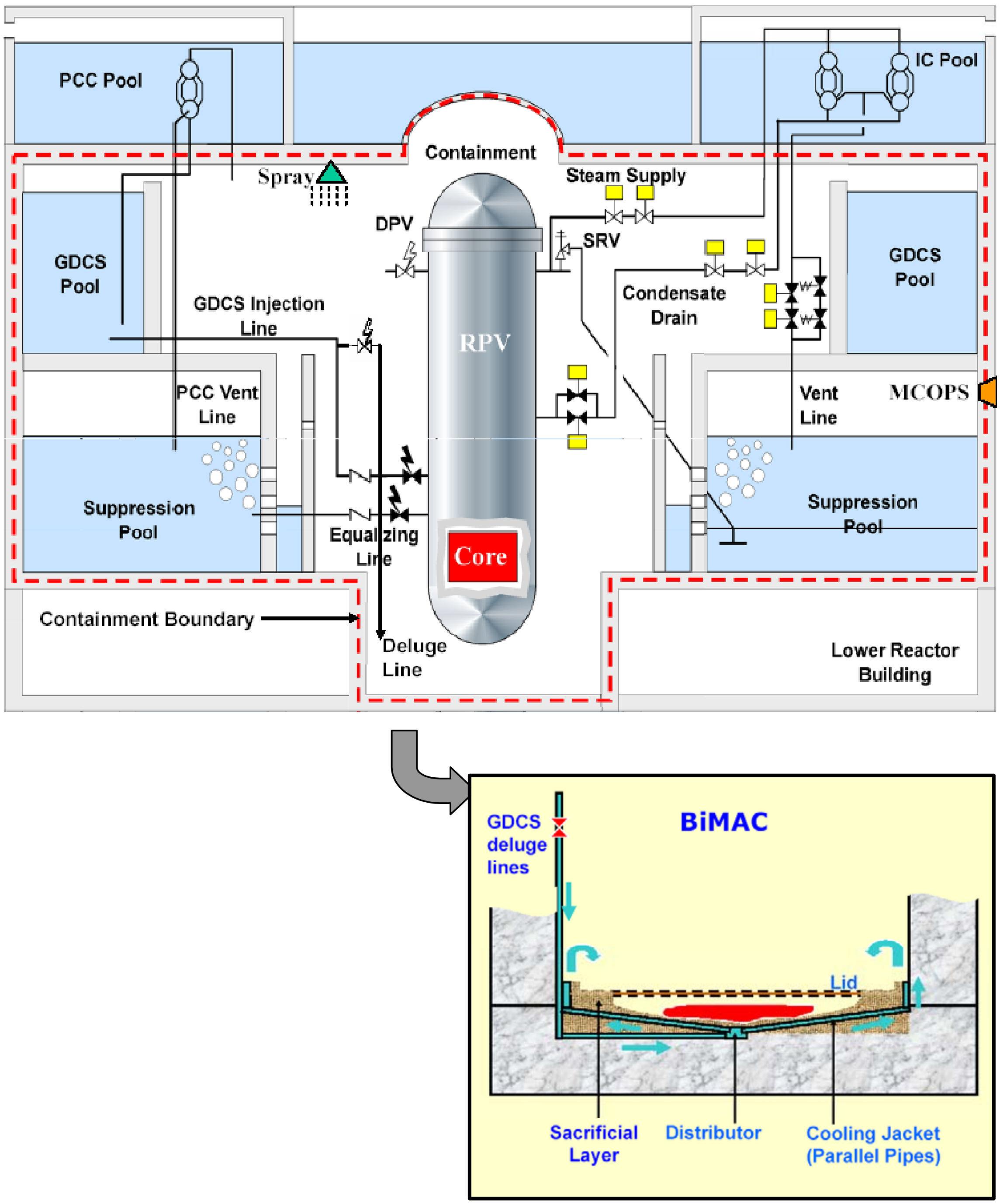


Figure 6.2-15. Summary of Severe Accident Design Features

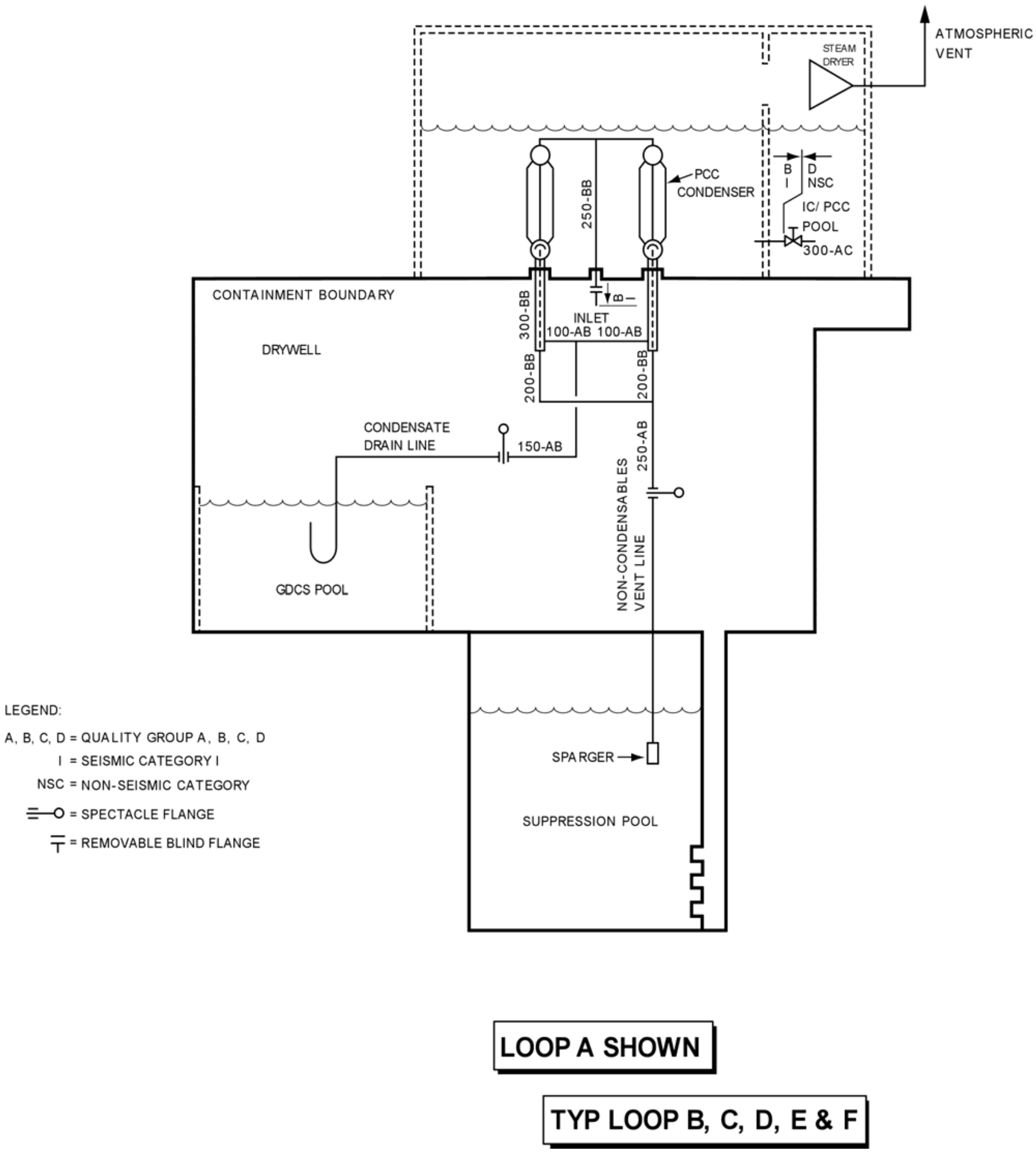


Figure 6.2-16. PCCS Schematic Diagram

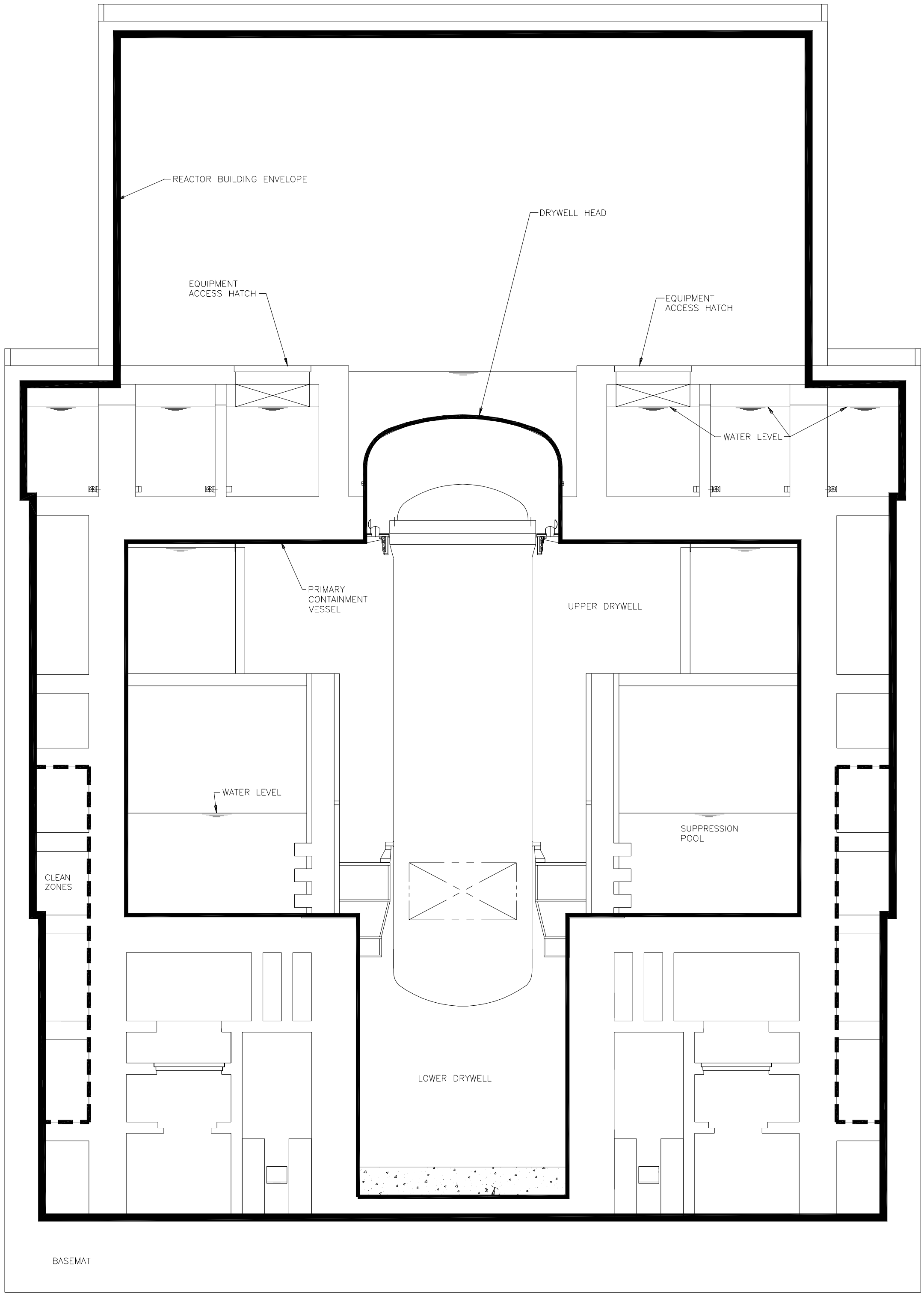


Figure 6.2-17. Reactor Building Envelope

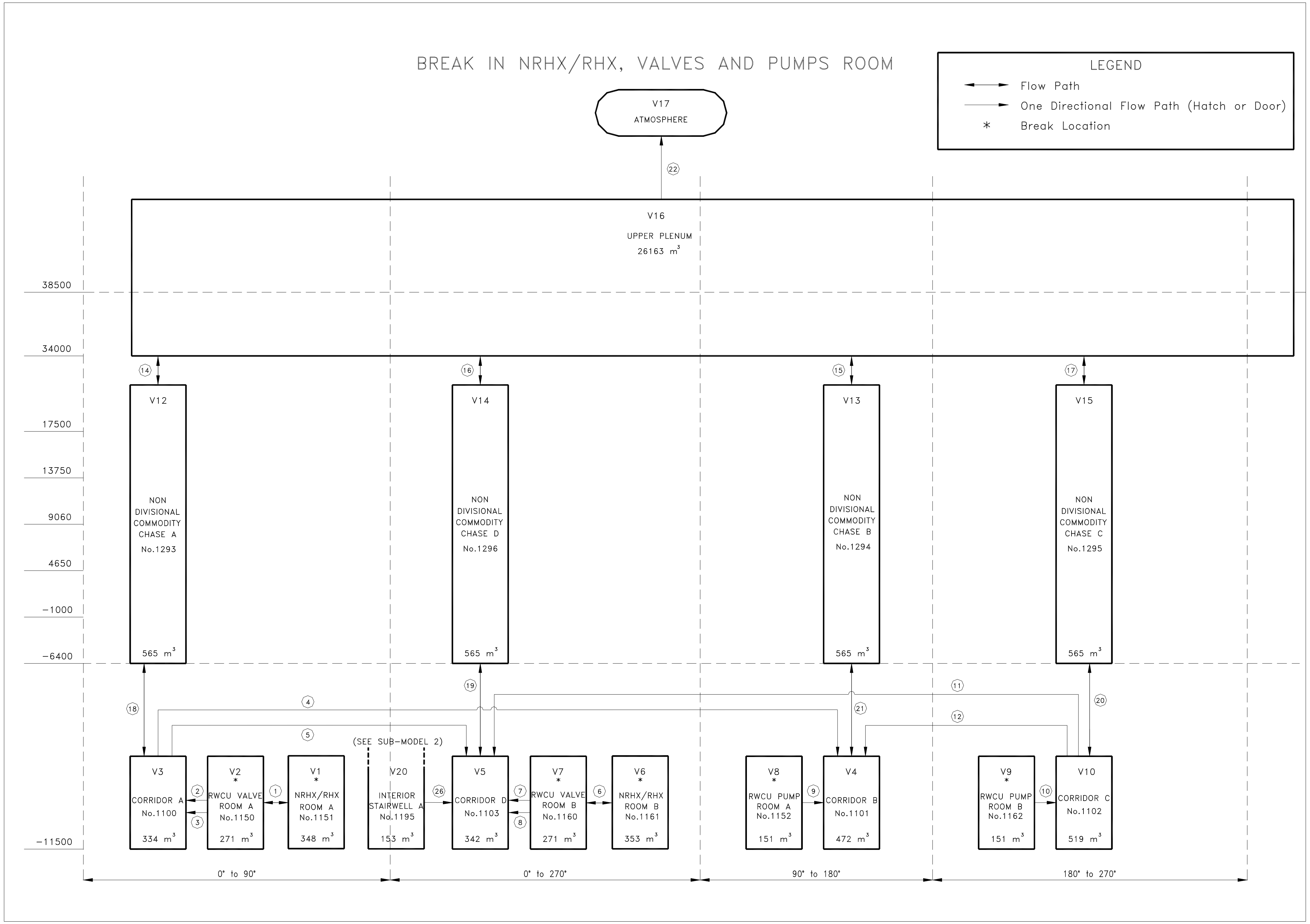


Figure 6.2-18. RWCU System Subcompartment Pressurization Analysis

6.3 EMERGENCY CORE COOLING SYSTEMS

Relevant to ESBWR emergency core cooling systems (ECCS), this subsection addresses or references to other DCD locations that address the applicable requirements of General Design Criteria (GDC) 2, 4, 5, 13, 17, 19, 20, 21, 22, 23, 24, 25, 27, 29, 35, 36 and 37, 10 CFR 50.46, TMI Action Plan items in 10 CFR 50.34(f), discussed in Standard Review Plan (SRP) 6.3 draft R3.

The ESBWR ECCS meets the requirements of GDC 2 as it relates to the seismic design of structures, systems, and components (SSCs) whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function.

The ESBWR meets the intent of GDC 4 as related to dynamic effects associated with flow instabilities and loads (e.g., water hammer), because its gravity-driven ECCS is not subject to flow instabilities.

The ESBWR ECCS meets the requirements of GDC 5 as it relates to safety-related SSCs not being shared among nuclear power units, because the design of the ESBWR ECCS precludes the possibility of sharing any ECCS between units.

The top of ESBWR core remains covered during all anticipated operational occurrences (AOOs) and accident conditions. Therefore, the ESBWR ECCS meets the requirements of GDC 17 as it relates to the design of the ECCS having sufficient capacity and capability to ensure that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during AOOs, and that the core is cooled during AOOs and accident conditions.

Regardless if the core has stuck control rods or not, for all abnormal events, the ECCS maintains the vessel water level above the top of the core. Therefore, the ECCS meets GDC 27 as it relates to the ECCS design having the capability to ensure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

As a result of the fact that the reactor scrams with sufficient water level above the top of the core, and the ECCS ensures that the core remains covered during all abnormal events, there is no fuel heat up. The containment and ECCS are designed to allow for periodic inspection of important components, and periodic pressure and functional testing. Therefore, the ESBWR meets the requirements of GDC 35, 36, and 37 as they relate to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and cladding damage does not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing.

As discussed in Subsection 6.3.3, the loss-of-coolant-accident (LOCA) modeling code has been reviewed and approved by the NRC, and the ECCS performance analysis results demonstrate the that ECCS meets all of the 10 CFR 50.46 acceptance criteria. Therefore, ESBWR complies with 10 CFR 50.46, in regard to the ECCS being designed so that its cooling performance is in accordance with an acceptable evaluation model.

The ECCS meets the intent of 10 CFR 50.34(f)(1)(vii) (equivalent to TMI Action Plan item II.K.3.18 of NUREG-0737), because no manual actuation of the ADS is needed to assure adequate core cooling for any design basis event.

The ECCS is initiated via the use of squib values that cannot be closed after initiation, no operator action is needed to assure core cooling, and the ECCS has no pump that can be stopped or restarted. Therefore the concern addressed in 10 CFR 50.34(f)(1)(viii) (equivalent to TMI Action Plan item II.K.3.21 of NUREG-0737) with respect to BWR core spray and low pressure coolant injection systems automatically restarting on loss of water level, after having been manually stopped, is not applicable.

The ESBWR ADS complies with 10 CFR 50.34(f)(1)(x) (equivalent to TMI Action Plan item II.K.3.28 of NUREG-0737), the ADS-associated equipment and instrumentation are capable of performing their intended functions during and following an accident, while taking no credit for nonsafety-related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves.

Without the use of ADS, safety/relief valves, depressurization valves, isolation condensers and turbine bypass valves can depressurize the reactor vessel without exceeding any vessel integrity limit. Therefore, ESBWR meets the intent 10 CFR 50.34(f)(1)(xi) (equivalent to TMI Action Plan item II.K.3.45 of NUREG-0737) with regard to providing depressurization, other than full actuation of the ADS, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs.

6.3.1 Design Bases and Summary Description

The ESBWR ECCS are the Gravity-Driven Cooling System (GDCS), Isolation Condenser System (ICS), Standby Liquid Control (SLC) system, and the Automatic Depressurization System (ADS) function of the Nuclear Boiler System.

This subsection provides the design bases and summary description for the ECCS as an introduction to the more detailed design descriptions provided in Subsection 6.3.2 and 6.3.3, and the performance analysis provided in Subsection 6.3.3.

6.3.1.1 Design Bases

6.3.1.1.1 Performance and Functional Requirements

The ECCS is designed to provide protection against postulated loss-of-coolant accidents (LOCAs) caused by ruptures in primary system piping. The functional requirements (e.g., coolant delivery rates) are such that the system performance under all LOCA conditions postulated in the design satisfies the requirements of 10 CFR 50.46, (Acceptance Criteria for Emergency Core Cooling System for Light Water Nuclear Power Reactors). These requirements are summarized in Subsection 6.3.3.2. In-addition, the ECCS is designed to meet the following requirements:

- Protection is provided for any primary system line break up to and including the double-ended break of the largest line;
- No operator action is required until 72 hours after an accident; and

- A sufficient water source and the necessary piping, and other hardware are provided so that the containment and reactor core can be flooded for core heat removal following a LOCA.

6.3.1.1.2 Reliability Requirements

The following reliability requirements apply:

- The ECCS network has built-in redundancy so that adequate cooling can be provided, even in the event of specified failures. As a minimum, the following equipment make up the ECCS:
  - GDCS;
  - ICS;
  - SLC; and
  - ADS function of the Nuclear Boiler System.
- The system is designed so that no single failure, including power buses, electrical and mechanical parts, cabinets and wiring prevents the ECCS from performing its function.
- In the event of a break in a pipe that is not part of the ECCS, no single active component failure in the ECCS prevents automatic initiation and successful operation of less than the combinations of ECCS equipment shown in Table 6.3-1.
- In the event of a break in a pipe that is a part of ECCS, no single active component failure in the ECCS prevents automatic initiation and successful operation of less than the combination of ECCS equipment as identified above, minus the ECCS in which the break is assumed. A break in a GDCS injection line eliminates flow through 2 RPV nozzles.
- Long-term cooling requirements call for the removal of decay heat from drywell via the Passive Containment Cooling System (See Subsection 6.2.2).
- Systems that interface with, but are not part of, the ECCS are designed and operated such that failure(s) in the interfacing systems do not propagate to and/or affect the performance of the ECCS.
- The logic required to automatically initiate component action of each system of the ECCS is capable of being tested during plant operation.
- Provisions for testing the ECCS network components (electronic, mechanical, hydraulic and pneumatic, as applicable) are installed in such a manner that they are an integral part of the design.

6.3.1.1.3 ECCS Requirements for Protection from Physical Damage

The ECCS piping and components are protected against damage from:

- movement;
- thermal stresses;
- effects of the LOCA; and

- effects of the safe shutdown earthquake.

The ECCS is protected against the effects of pipe whip, which might result from piping failures up to and including the design basis event LOCA. This protection is provided by separation, pipe whip restraints, or energy-absorbing materials if required. One or more of these three methods is applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level.

6.3.1.1.4 ECCS Environmental Design Basis

ECCS safety-related valves (located within the drywell) and the ECCS equipment located outside the drywell and within the Reactor Building are qualified for the environmental conditions defined in Section 3.11.

6.3.1.2 Summary Descriptions of ECCS

Gravity-Driven Cooling System

The GDCS provides flow to the annulus region of the reactor through dedicated nozzles. It provides gravity-driven flow from three separate water pools located within the drywell at an elevation above the active core region. It also provides water flow from the suppression pool to meet long-term post-LOCA core cooling requirements. The system provides these flows by gravity forces alone (without reliance on active pumps) once the reactor pressure is reduced to near containment pressure.

Automatic Depressurization System

The ADS provides reactor depressurization capability in the event of a pipe break. The ADS is a function of the Nuclear Boiler System (NBS). The depressurization function is accomplished through the use of safety/relief valves (SRVs) and depressurization valves (DPVs).

Isolation Condenser System

The ICS provides additional liquid inventory upon opening of the condensate return valves to initiate the system. The IC system also provides reactor with initial depressurization of the reactor before ADS in event of loss of feed water, such that the ADS can take place from a lower water level. (See section 5.4.5 for the detailed description of the ICS.)

Standby Liquid Control System

The SLC system provides reactor additional liquid inventory in the event of DPV actuation. This function is accomplished by firing squib type injection valves to initiate the SLC system. (See section 9.3.5 for the detailed description of the SLC system.)

6.3.2 System Design

Subsections 6.3.2.1 through 6.3.2.6 provide details of those design features and characteristics that are common to all subsystems. More detailed descriptions of the individual systems, including individual design characteristics of the systems, are provided in Subsections 6.3.2.7 and 6.3.2.8.

6.3.2.1 Equipment and Component Descriptions

The starting signal for the ECCS comes from independent and redundant sensors of low reactor water level. The ECCS is actuated automatically and requires no operator action during the first 72 hours following the accident.

Electric power for operation of the ECCS is from redundant onsite safety-related power sources. Emergency sources have sufficient capacity so that all ECCS requirements are satisfied. Each ECCS division has its own independent power source. Section 8.3 contains a more detailed description of the power supplies for the ECCS.

Because ECCS flow is gravity driven, NPSH is not a concern.

6.3.2.2 Applicable Codes and Classifications

The applicable codes and classification of the ECCS are specified in Section 3.2. The edition of the codes applicable to the design are provided in Table 1.9-22. The ECCS piping and components within containment are designed as Seismic Category I. This seismic designation applies to all structures and equipment essential to the core cooling function. IEEE codes applicable to the controls and power supply are specified in Section 7.1.

6.3.2.3 Materials Specifications and Compatibility

Materials specifications and compatibility for the ECCS are presented in Section 6.1.

Nonmetallic materials such as lubricants, seals, packings, paints and primers, insulation, as well as metallic materials, etc., are selected as a result of an engineering review and evaluation for compatibility with other materials in the system and the surroundings with concern for chemical, radiolytic, mechanical and nuclear effects. Materials used are reviewed and evaluated with regard to radiolytic and pyrolytic decomposition and attendant effects on safe operation of the ECCS.

6.3.2.4 System Reliability

No single failure prevents the initiation of the ECCS, when required, or the delivery of coolant to the reactor vessel. Each individual system of the ECCS is single-failure proof. The most severe effects of single failures with respect to loss of equipment and the consequences of the most severe single failures are discussed within Subsection 6.3.3.

6.3.2.5 Protection Provisions

Protection provisions are included in the design of the ECCS. Protection is afforded against missiles, pipe whip and flooding. Also accounted for in the design are thermal stresses, loadings from a LOCA, and seismic effects.

The ECCS is protected against the effects of missiles, pipe whip, etc… which might result from piping failures up to and including the design basis event LOCA. This protection is provided by separation, pipe whip restraints, and energy absorbing materials. One of these three methods is applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level (see Section 3.6 for criteria on pipe whip).

Subsection 5.4.14 discusses the component supports that protect against damage from movement and from seismic events. Subsection 3.9.3 describes the methods used to provide assurance that thermal stresses do not cause damage to the ECCS.

6.3.2.6 Manual Actions

The operator cannot override or interrupt an ECCS action once it has been sealed-in to the plant's Safety System Logic and Control (SSLC) System. Also, the operator cannot close any valves in the GDCS system. The initiation scheme for the ADS and GDCS is designed such that no single failure in the initiation circuitry can prevent the GDCS from providing the core with adequate cooling. Furthermore, the GDCS has no protective interlocks that could interrupt automatic system operation. While all of the detection and signaling functions that cause ECCS operation are automatic and require no operator action or intervention over the 72-hour period following a DBA, the operator can manually initiate any of the systems in any of the divisions. To initiate the GDCS short-term injection and long-term injection systems manually, a low pressure signal must be present in the RPV, thus preventing inadvertent manual initiation of the system during normal reactor operation. To initiate the deluge system manually, a high drywell pressure signal must be present.

6.3.2.7 Gravity-Driven Cooling System

6.3.2.7.1 Design Bases

Safety Design Bases

The GDCS provides emergency core cooling after any event that threatens the reactor coolant inventory. Once the reactor has been depressurized the GDCS is capable of injecting large volumes of water into the depressurized reactor pressure vessel (RPV) to keep the core covered for at least 72 hours following LOCA.

The system also drains the GDCS pools to the lower drywell in the event of a core melt sequence that causes failure of the lower vessel head and allows the molten fuel to reach the lower drywell cavity floor. This action is accomplished by detection of elevated temperatures registered by thermocouples penetrating the protective layer in the lower drywell cavity, and by logic circuits that actuate squib-type valves on independent pipelines draining GDCS pool water to the lower drywell region.

The GDCS requires no external AC electrical power source or operator intervention. The GDCS initiation signal is the receipt of a confirmed ECCS initiation signal from the NBS (see Subsection 7.3.1.3). This signal initiates ADS and GDCS injection valve timers as well as longer equalization valve timers in the GDCS logic. After injection valve timer duration, squib valves are activated in each of the injection lines leading from the GDCS pools to the RPV, thus making GDCS flow possible. The actual GDCS flow delivered to the RPV is a function of the differential pressure between the reactor and the GDCS injection nozzles, as well as the loss of head due to inventory drained from GDCS pool. The timer delay allows the RPV to be substantially depressurized prior to squib valve actuation.

After a longer equalization valve time delay and when the RPV coolant level decreases to 1 m (3.28 ft.) above the top of the active fuel (TAF), squib valves are actuated in each of four GDCS equalizing lines. The open equalizing lines leading from the suppression pool to the RPV make long-term coolant makeup possible. The longer equalization valve delay ensures that the GDCS pools have had time to drain to the RPV and that the initial RPV level collapse as a result of the blowdown does not open the equalizing line. The long-term flow requirements for the GDCS equalizing lines are as follows: with the suppression pool water at saturation temperature, with vessel water level below equalizing line nozzles, the flow delivered inside the RPV through the GDCS equalizing lines is as shown in Table 6.3-2. This flow is required assuming a double-ended-guillotine-break in one GDCS equalizing line, and the worst single failure in a second equalizing line.

In the event of a core melt accident in which molten fuel reaches the lower drywell, the flow through the deluge lines is required to flood the lower drywell region with a required deluge network flow rate as shown in Table 6.3-2. The system design is such that with a single active failure in one of the deluge valves does not prevent any of the pools from draining into the drywell.

All piping connected with the RPV is classified as Safety-Related, Seismic Category I. The electrical design of the GDCS is classified as Class 1E. The GDCS piping and components are protected against damage from:

- movement;
- thermal stresses;
- effects of the LOCA; and
- effects of the safe shutdown earthquake.

The GDCS is protected against the effects of pipe whip, which might result from piping failures up to and including the design basis event LOCA. This protection is provided by separation, pipe wipe restraints, energy-absorbing materials (if required) or by providing structural barriers.

The GDCS is mechanically separated into four identical divisions. Each GDCS division takes inventory from the GDCS pools (one division each from two pools and two divisions from the third pool) and the suppression pool. The equipment in each division is separated from that of the other two divisions.

6.3.2.7.2 System Description

Summary Description

The GDCS provides short-term post-LOCA water makeup to the annulus region of the reactor through eight injection line nozzles, by gravity-driven flow from three separate water pools located within the drywell at an elevation above the active core region. The system provides long-term post-LOCA water makeup to the annulus region of the reactor through four equalization nozzles and lines connecting the suppression pool to the RPV. During severe accidents the GDCS floods the lower drywell region directly via four GDCS injection drain lines (one each from two pools and two from the third pool) through deluge system, if the core melts through the RPV.

Detailed System Description

The GDCS is composed of four divisions. Electrical and mechanical separation between the divisions is complete. Physical separation is ensured between divisions by locating each division in a different area of the reactor building. A single division of the GDCS consists of three independent subsystems: a short-term cooling (injection) system, a long-term cooling (equalizing) system, and a deluge line. The short-term and long-term systems provide cooling water under force of gravity to replace RPV water inventory lost during a LOCA and subsequent decay heat boil-off. The deluge line connects the GDCS pool to the lower drywell.

Table 6.3-2 provides the design basis parameters for the GDCS, and includes:

- For GDCS pools, the minimum total drainable inventory;
- The minimum surface elevation of the GDCS pools above the RPV nozzle elevation;
- The minimum suppression pool available water inventory 1 meter above TAF; and
- The minimum GDCS equalizing line driving head, which is determined by the elevation differential between the top inside diameter of the first S/P horizontal vent and the centerline of the GDCS equalizing line RPV nozzle.

The GDCS deluge lines provide a means of flooding the lower drywell region with GDCS pool water in the event of a core melt sequence which causes failure of the lower vessel head and allows the molten fuel to reach the lower drywell floor.

The core melt sequence results from a common mode failure of the short-term and long-term systems, which prevents them from performing their intended function. Deluge line flow is initiated by thermocouples, which sense high lower drywell region basemat temperature indicative of molten fuel on the lower drywell floor. Logic circuits actuate squib-type valves in the deluge lines upon detection of basemat temperatures exceeding setpoint values. The deluge lines do not require the actuation of squib-actuated valves on the injection lines of the GDCS piping to perform their function.

Each division of the GDCS injection system consists of one 200-mm (8-inch) pipe (with a temporary strainer and a block valve) exiting from the GDCS pool. Just after the 200-mm (8-inch) block valve a 100-mm (4-inch) deluge line branches off and is terminated with three 50-mm (2-inch) squib valves and deluge line tailpipe to flood lower drywell. The 200-mm (8-inch) injection line continues after the 100-mm (4-inch) deluge line connection from the upper drywell region through the drywell annulus where the 200-mm (8-inch) line branches into two 150-mm (6-inch) branch lines each containing a biased-open check valve, squib valve, and block valve. Each division of the long-term system consists of one 150-mm (6-inch) equalizing line with two block valves, a check valve and a squib valve. All piping is stainless steel and rated for reactor pressure and temperature. Figure 6.3-1 illustrates the arrangement of GDCS piping configuration.

The RPV injection line nozzles and the equalizing line nozzles all contain integral flow limiters with a venturi shape for pressure recovery. The minimum throat diameter of the nozzles in the short-term system is 76.2 mm (3 in) and the minimum throat diameter of the nozzles in the long-term system is 50.8 mm (2 in.). The nozzle throat length is long enough to ensure that the homogeneous flow model can be used in LOCA analyses. Each injection line and equalizing line contains a locked open, manually-operated maintenance valve located near the vessel nozzle and another such valve located near the water source.

In the injection lines and the equalizing lines, there exists a biased-open check valve located upstream of the squib-actuated valve. Downstream of the squib-actuated injection valve is a test line which can be used to back-flush. This operation is conducted during refueling and maintenance outages for the region of piping between the reactor and the squib valve of crud buildup or debris.

The GDCS squib valves are gas propellant type shear valves that are normally closed and which open when a pyrotechnic booster charge is ignited. The squib valve is designed to withstand the drywell LOCA environment sufficiently long to perform its intended function. During normal reactor operation, the squib valve is designed to provide zero leakage. Once the squib valve is actuated it provides a permanent open flow path to the vessel.

The check valves mitigate the consequences of spurious GDCS squib valve operation and minimize the loss of RPV inventory after the squib valves are actuated and the vessel pressure is still higher than the GDCS pool pressure plus its gravity head. Once the vessel has depressurized below GDCS pool surface pressure plus its gravity head, the differential pressure opens the check valve and allow water to begin flowing into the vessel.

The deluge valve is a squib-actuated valve that is initiated by a high temperature in the lower drywell region. This temperature is sensed by thermocouples penetrating in the basemat protective layer. The deluge valve is designed to survive the severe accident environment of a core melt and still perform its intended function. The pyrotechnic material of the squib charge used in the deluge valve is different than what is used in the other GDCS squib valves to prevent common mode failure. The deluge valve is designed to withstand the water hammer expected as a result of an inadvertent GDCS squib valve opening while the reactor is at normal operating pressure and temperature. Once the deluge valve is actuated it provides a permanent open flow path from the GDCS pools to the lower drywell region. Flow then drains to the lower drywell via permanently open drywell lines.

The GDCS check valves remain partially open when zero differential pressure exists across the valve. This is to minimize the potential for sticking in the closed position during long periods of non-use. A test connection line downstream of the check valve allows the check valve to be tested during refueling outages. This provides a means for testing the operation of the check valve.

All system block valves are normally locked open and are used for maintenance during a plant refueling or maintenance outage.

Suppression pool equalization lines have an intake strainer to prevent the entry of debris material into the system that might be carried into the pool during a large break LOCA. The GDCS pool airspace opening to DW will be covered by a mesh screen or equivalent to prevent debris from entering pool and potentially blocking the coolant flow through the fuel. A splash guard is added to the opening to minimize any sloshing of GDCS pool water into the drywell following dynamic event.

The GDCS is designed to operate from safety-related power. The system instrumentation is powered by divisionally separated safety-related 120 V AC, and the squib valve and deluge valve initiation circuitry is powered by divisionally separated, safety-related, 250 V DC.

System Operation

During normal plant operation, GDCS is in a standby condition. It can be actuated simply by transmitting a firing signal to the squib valves. The firing signal can be initiated automatically or manually from switches in the main control room. The design basis for the system during normal plant operation is to maintain RPV backflow leak-tight. Each GDCS injection line positively prevents unnecessary heating of the GDCS pools and transport of radioactive contamination to the GDCS pools and/or suppression pool.

When the reactor is shutdown, the GDCS is normally in a standby condition. Deactivating and isolating GDCS divisions are governed by plant Technical Specifications.

During a LOCA, GDCS is initiated following a confirmed ECCS initiation signal from NBS. The signal starts two sets of timers in each division; two injection valve timers for initiation of the short-term water injection lines and two longer equalization timers which create a permissive signal (in combination with RPV water level below Level 0.5 or 1 meter above TAF) for initiation of the long-term injection lines. After the injection valve timer expires after a confirmed ECCS initiation signal, the short-term injection squib valves open to allow water to flow from the GDCS pools to the RPV. Once the reactor becomes adequately depressurized the water flow refills the RPV thereby ensuring core coverage and decay heat removal.

The long-term portion of GDCS can begin operation following a longer equalization valve time delay initiated by a confirmed ECCS initiation signal and when RPV level reaches Level 0.5, which is 1 m (3.28 ft.) above the TAF. Flow is initiated with the opening of the squib valve on each GDCS equalizing line. The GDCS equalizing lines perform the RPV inventory control function in the long term and makeup for the following inventory losses:

- For any LOCA above the core the equalizing lines provide for coolant boiloff losses to the drywell (Most coolant boil-off is returned to the RPV as condensate from the isolation condensers or the Passive Containment Cooling System heat exchangers).
- For a vessel bottom line break, the equalizing line provides inventory for coolant boiloff losses to the drywell and break flow losses in the mid-term. In the long term the equalizing lines provide for evaporation losses to the drywell.

The GDCS is designed to mitigate the consequences of a hypothetical severe accident with molten core material on the lower drywell floor. The lower drywell basemat has a grid of thermocouples, assigned to 4 divisions, which sense the presence of molten fuel on the lower drywell floor. Deluge line flow is initiated when the lower drywell basemat temperature exceeds a preset setpoint, as sensed by a preset number of thermocouple in two separate divisions. The initiation signal opens the deluge valve on each separate deluge line to allow GDCS pool water to drain to the lower drywell. This water aids in cooling the molten core.

Equipment and Component Description

The following describes the GDCS squib valve, deluge valve and biased-open check valve, which are unique system components that are not used in previous BWR designs.

Squib Valve

The function of the squib valve is to open upon an externally applied signal and to remain in its full open position without any continuing external power source in order to admit reactor coolant makeup into the reactor pressure vessel in the event of a LOCA. The valves also functions in the close position to maintain RPV backflow leaktight and maintaining reactor coolant pressure boundary during normal plant operation. The GDCS squib valves have a C_v that will permit development of full GDCS flow. The valve is a horizontally mounted, straight through, long duration submersible, pyrotechnic actuated, non-reclosing valve with metal diaphragm seals and flanged ends. The valve design is such that no leakage is possible across the diaphragm seals throughout the 60-year life of the valve. The squib valve is classified as Quality Group A, Seismic Category I, and ASME Section III Class 1. The valve diaphragm forms part of the reactor pressure boundary and as such is designed for RPV service level conditions.

Illustrated in Figures 6.3-2 is a typical squib valve design that satisfies GDCS system requirements. This valve has similar design features to the ADS depressurization valve.

Valve actuation initiates upon the actuation of either of two squib valve initiators, a pyrotechnic booster charge is ignited, and hot gasses are produced. When these gasses reach a designed pressure, a tension bolt holding a piston breaks allowing the piston to travel downward until it impacts the ram and nipple shear caps. Once the piston impacts the ram and nipple shear caps, the nipples are sheared. The ram and shear caps are then driven forward and are locked in place at the end of stroke by an interference fit with the nipple retainer. This lock ensures that the nipples cannot block the flow stream and provides a simple means of refurbishment by simply unthreading the plug. A switch located on the bottom of the valve provides a method of indication to the control room of an actuated valve. The shear nipple sections are designed to produce clean shear planes.

The piston is allowed to backup after shearing the nipples. Standard metal seals are installed on the piston to reduce the potential of ballistic products from entering the flow stream.

The squib valve can be completely refurbished once fired. The squib valve housing, nipples, adapter flanges, actuator housing, indicator switch body, indicator plunger, head cap, coupling, collar and adapter are machined. The piston, ram, and tension bolt is made from heat treated material for necessary strength.

Biased-Open Check Valve

The GDCS biased-open check valves are designed such that the check valve active mechanism is off the seat when zero differential pressure is applied across the check valve. The check valve fully closes to prevent excess backflow when a low reverse differential pressure is applied across it. The biased-open check valve is a straight through, horizontally mounted, long duration submersible valve. The valve meets the minimum flow requirements for a valve stuck in the "valve biased" open position. The biased-open check valve is classified as Quality Group A, Seismic Category I, and ASME Section III Class 1.

The biased-open check valve illustrated in Figure 6.3-3 is a tilting disc check valve, biased open by gravity forces on the disc and counterweight. No torsion spring is used on the hinge pin and the hinge pin does not extend through any packing. The check valve is fabricated out of low carbon stainless steel.

Remote check valve position indication is provided in the main control room by an eccentric ferromagnetic target on an extension from the tilting disc, which rotates within a non-magnetic (stainless steel) pressure shell. A micro switch proximity sensor locates the magnetic target when it is in the proper position. Two switches are required to separate full-closed and biased-open indication. This indicator requires no pressure boundary penetration.

Deluge Valve

The deluge valve is a 50 mm (2 inch) squib valve similar in design to the SLC squib valves or ADS depressurization valves. To minimize the probability of common mode failure, the deluge valve pyrotechnic booster material is different from the booster material in the other GDCS squib valves. The pyrotechnic charge for the deluge valve is qualified for the severe accident environment in which it must operate.

6.3.2.7.3 Safety Evaluation

GDCS performance evaluation during a LOCA is covered in Subsection 6.3.3.

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

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

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

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

The electrical design shall be classified Class 1E.

6.3.2.7.4 Testing and Inspection Requirements

Performance Tests

During fabrication, the GDCS components are subjected to various tests and examinations as required by the ASME Code, including hydrostatic testing and operability testing.

The GDCS will be tested for its operational ECCS function during the preoperational test program. Each component is tested for power source, range, setpoint, position indication, etc.

All GDCS logic elements are tested individually and then as a system to verify complete system response to an ECCS initiation signal.

See Chapter 14 for a thorough discussion of preoperational testing on the GDCS.

Reliability Test and Inspections

No system component tests are conducted during plant operation. The trip logic units of each logic division and the time delay units for squib actuation may be tested during plant operation. The trip logic units are continually self-tested every 30 minutes. See Table 6.3-3 for the components to be tested, the type of test to be conducted, and component alignment. The only valves directly operated for testing are the normally-closed isolation valves on the test lines. Flow through the system test lines is used to open and close the GDCS check valves and to show that there is no obstruction of the RPV nozzles. Valve realignment following test is controlled administratively.

During scheduled plant shutdowns, a minimum of 20% of the pyrotechnic charges are replaced in the squib valves. Replacement is done without any opening of the reactor coolant pressure boundary (RCPB). Subsequently in the laboratory, the removed charges are tested to confirm end of life capability to function upon demand.

6.3.2.7.5 Instrumentation Requirements

GDCS control logic for the system and design details including redundancy and logic are covered in Subsection 7.3.1.2. The following paragraphs give a brief description of the system instrumentation and control logic.

Level Instrumentation

Level instrumentation is provided in each GDCS pool and the suppression pool to monitor and record water level. The level instrumentation for each pool consists of two instrument lines that penetrate the drywell and connect to the high and low pressure sides of two level transmitters. Each line penetrating the drywell contains a series of isolation valves. The output of the level transmitter is sent to the Safety System Logic and Control (SSLC) and the Control Rod Drive (CRD) system for processing. The SSLC uses a microprocessor based design to compare detection levels with preset trip settings. If the trip settings are exceeded then an alarm is sounded in the main control room. The operator must take manual action to restore the water level to the proper elevation. The level signal sent to the CRD system trips the CRD pumps following RPV Level 2 CRD pump injection. This prevents excessive makeup water being pumped in to the containment during a LOCA.The GDCS pool high and low water level signal is also sent to FAPCS pump for GDCS pool cooling and cleanup.

Controls

Controls for the GDCS are gathered in a single area of the control room to facilitate system monitoring and operation.

Status Indication

Switches in the control room enable the reactor operator to manually actuate the GDCS squib-actuated valves and deluge valves as backup action if safety logic should fail to develop the automatic initiation signals. Manual initiation for the GDCS squib valves is interlocked with a low RPV pressure signal to prevent inadvertent system initiation. Manual initiation of the deluge valves is interlocked with a high drywell pressure signal to prevent inadvertent initiation. Refer to Subsection 7.6.1 for a more detailed discussion of this interlock.

During operation the assessment of GDCS status is determined by monitoring key component status indications and GDCS pool water level measurement indications.

The following GDCS indications are reported in the control room:

- Status of the locked-open maintenance valves;
- The status of the squib-actuated valves;

- GDCS pools and suppression pool level indication;
- Position of each GDCS check valve;
- Suppression pool high and low level alarm;
- GDCS pools high and low level alarms; and
- Squib valve open alarms.

6.3.2.8 Automatic Depressurization System

6.3.2.8.1 Design Bases

Safety (10 CFR 50.2) Design Bases

The Automatic Depressurization System (ADS) shall:

- Quickly depressurize the RPV in a time sufficient to allow the GDCS injection flow to replenish core coolant to maintain core temperature below design limits in the event of a LOCA;
- Maintain the reactor depressurized for continued operation of GDCS after an accident without the need for external power;
- Accomplish its safety-related functions assuming the single failure of an active component;
- Be designed so a single failure does not render more than one ADS valve inoperative;
- Withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, or faulted conditions and the associated ambient environmental conditions;
- Employ valves designed so a safe shutdown earthquake (SSE) cannot open a closed valve or cause an open valve to close; and
- Be capable of opening over the full range of reactor vessel pressures and reactor vessel-to-drywell differential pressures down to and including a differential pressure of zero.

Non-Safety, Power Generation Design Bases

The ADS shall be designed to minimize the potential for interruption of normal plant operation as a result of excessive component leakage or inadvertent actuation without diminishing the safety of the system.

There are no other power generation design bases for the ADS. Power generation design bases for the SRV components of ADS, when functioning as part of the overpressure protection system, are discussed in Subsection 5.2.2.

6.3.2.8.2 System Description

Summary Description

The ADS is a part of the ECCS and operates to depressurize the reactor for the low pressure GDCS to be able to make-up coolant to the reactor. The ADS is composed of the SRVs and depressurization valves (DPVs) and their associated instrumentation and controls.

The ADS system components are shown on Figure 6.3-4.

Detailed System Description

The ADS consists of the 10 SRVs and 8 DPVs. The SRVs are mounted on top of the main steamlines in the drywell and discharge through lines routed to quenchers in the suppression pool as described in Section 5.2.2. Four DPVs are horizontally mounted on horizontal stub tubes connected to the RPV at about the elevation of the main steamlines. The other four DPVs are horizontally mounted on horizontal lines branching from each main steamline.

The use of a combination of SRVs and DPVs to accomplish the ADS function minimizes components and maintenance as compared to using only SRVs or only DPVs for this function. By using the SRVs for two different purposes, the number of DPVs required is minimized. By using DPVs for the additional depressurization capability needed beyond what the SRVs can provide, the total number of SRVs, SRV discharge lines, and quenchers in the suppression pool are minimized. The need for SRV maintenance, periodic calibration and testing, and the potential for simmering are all minimized with this arrangement.

The SRVs and DPVs and associated controls and actuation circuits are located or protected so that their function cannot be impaired by consequential effects of the accidents. The ADS is designed to withstand the effects of flooding, pipe whip and jet impingement. ADS components are also qualified to withstand the harsh environments postulated for design basis accidents inside containment, including temperature, pressure, and radiation. Further details are provided in Section 3.11.

The SRVs and DPVs are designed with flange connections to allow easy removal for maintenance, testing, or rebuilding. They are designed, however, so that routine maintenance and inspection can be accomplished at their installed locations.

The SRVs are described in detail in Subsection 5.2.2.

Figure 6.3-5 depicts a DPV assembly in the closed and open positions. The DPVs are of a non-leak/non-simmer/non-maintenance design. They are straight-through, squib-actuated, non-reclosing valves with a metal diaphragm seal. The valve size provides about twice the depressurization capacity as a SRV. The DPV is closed with a cap covering the inlet chamber. The cap shears off when pushed by a valve plunger that is actuated by the explosive initiator-booster. This opens the inlet hole through the plug. The sheared cap is hinged such that it drops out of the flow path and does not block the valve. The DPVs are designed so that there is no leakage across the cap throughout the life of the valve.

Two initiators (squibs), singly or jointly, actuate a booster, which actuates the shearing plunger. The squibs are initiated by either one or both of, two battery-powered, independent firing circuits. The firing of one initiator-booster is adequate to activate the plunger. Nominal firing voltage is 250 V DC, however the initiator-boosters are designed to function with any applied voltage between 185 and 310 V DC. The valve design and initiator-booster design is such that there is substantial thermal margin between operating temperature and the self-ignition point of the initiator-booster.

The DPVs form a part of the reactor coolant pressure boundary (RCPB), and therefore, are Quality Group A, ASME Section III, Class 1, and Seismic Category I.

The DPV inlet side design pressure, outlet-side design pressure, valve body materials and design life are provided in Table 6.3-4. The design life includes remaining functional after being subjected to a variety of normal and abnormal pressure-temperature transients, including two cycles of full ADS depressurization of the reactor. Certain components, such as the initiator-boosters, require periodic replacement.

The DPV operating fluid conditions, rated flow capacity of each DPV, specified response times (opening time to full rated capacity) of the DPVs are provided in Table 6.3-4.

The DPV has undergone engineering development testing using a prototype to demonstrate the proper operability, reliability, and flow capability of the design. This testing is documented in the test program final report, Reference 6.3-1.

Functional tests were performed to assure proper operability and the adequacy (amount and chemical compound) of the initiator-booster to operate the valve assembly. Heat transfer tests were also performed to determine the temperature of the initiator-booster based on the valve inlet temperature and a range of ambient temperatures.

Reliability testing was conducted on a sufficient number of initiator-boosters to demonstrate the reliability of the chemical to fire and properly actuate the valve while at the same time avoiding accidental, unwanted firing. These tests involved irradiating, thermally aging, and subjecting the initiator-booster to LOCA environmental conditions before firing.

Flow capacity tests of the prototype were also conducted to assure the flow rate requirements noted above were satisfied.

System Operation

The ADS automatically actuates on a reactor low-low level (Level 1) signal that persists for at least 10 seconds. A two-out-of-four Level 1 logic is used to activate the SRVs and DPVs. The 10-second persistence requirement for the Level 1 signal ensures that momentary system perturbations do not actuate ADS when it is not required. The two-out-of-four logic ensures that a single failure does not cause spurious system actuation while also assuring that a single failure cannot prevent initiation.

The SRVs and DPVs are actuated in groups of two or five valves at staggered times as the reactor undergoes a relatively slow depressurization. This minimizes reactor level swell and unnecessary loss of reactor coolant through the SRVs and DPVs during the depressurization. The staggered opening of the valves is achieved by delay timers set per the sequence shown in Table 6.3-1.

The ADS may also be manually initiated from the main control room. Further details on the ADS control and actuation design is provided in Subsection 7.3.1.1.

6.3.2.8.3 Safety Evaluation

The performance of the ADS in conjunction with the other elements of the ECCS is discussed in the overall ECCS evaluation in Subsection 6.3.3.

In all cases core temperature limits are not exceeded for the spectrum of break sizes postulated, indicating that the sizing and actuation logic of the ADS, assuming the failure of one valve to actuate, is adequate. The limiting case for ADS operation is the small break LOCA, where the break itself contributes little to the depressurization of the vessel.

The ability of the SRVs and DPVs to withstand the various loads and forces from normal plant operations, expected transients (such as turbine stop valve closure and SRV operation), the SSE, blowdown and hydrodynamic loads associated with design basis accidents is discussed in Section 3.9.

6.3.2.8.4 Testing and Inspection Requirements

During fabrication, the SRV and DPV valves are subjected to various tests and examinations as required by the ASME Code, including hydrostatic testing and operability testing.

After installation, the valves and their controls are functionally tested to ensure they operate properly. The valve connections to the main steam lines and stub tubes are tested in the hydrostatic testing of the RPV. Further details on this preoperational testing are given in Chapter 14.

During plant operation, periodic tests and inspections are required as indicated in the plant-specific Technical Specifications. The SRVs can be exercised while the plant is operating. Required tests on the DPVs can be performed during planned shutdown periods, such as refueling outages.

6.3.2.8.5 Instrumentation Requirements

The position of the SRVs and DPVs are indicated in the main control room.

Continuity of the actuation circuitry for both the SRVs and DPVs is monitored in the main control room, and an alarm is actuated if continuity is lost. Continuity of the DPV is established by use of a continuous very low amperage bridge current.

Further description of the ADS instrumentation is provided in Subsection 7.3.1.1.

6.3.2.9 Isolation Condenser System

6.3.2.9.1 Design Bases

Refer to Subsection 5.4.6.1.

6.3.2.9.2 System Description

Refer to Subsection 5.4.6.2.

6.3.2.9.3 Safety Evaluation

ICS performance evaluation during a LOCA is covered in Subsection 6.3.3.

6.3.2.9.4 Testing and Inspection Requirements

Refer to Subsection 5.4.6.4.

6.3.2.9.5 Instrumentation Requirements

Refer to Subsection 5.4.6.5 and 7.4.4.

6.3.2.10 Standby Liquid Control System

6.3.2.10.1 Design Bases

Refer to Subsection 9.3.5.1.

6.3.2.10.2 System Description

Refer to Subsection 9.3.5.2.

6.3.2.10.3 Safety Evaluation

SLC performance evaluation during a LOCA is covered in Subsection 6.3.3.

6.3.2.10.4 Testing and Inspection Requirements

Refer to Subsection 9.3.5.3.

6.3.2.10.5 Instrumentation Requirements

Refer to Subsection 9.3.5.4.

6.3.3 ECCS Performance Evaluation

Performance of the ECCS network is determined by evaluating the system response to an instantaneous break of a pipe. The analyses included in this subsection demonstrates the adequacy of ESBWR ECCS network performance for the entire spectrum of postulated break sizes.

The analyses are based upon the core loading shown within Section 4.3 and were performed with the TRACG model. The Maximum Linear Heat Generation Rate (MLHGR) used in the ECCS-LOCA analysis for each bundle design ensures that the criteria documented in Appendix 4B is met. These results will be provided by the utility referencing the ESBWR design to the USNRC for information. See Subsection 6.3.5.

The Chapter 15 accidents for which ECCS operation is required are:

- Feedwater Line Break;
- Spectrum of BWR Steam System Piping Failures Outside Containment; and
- Loss-of-Coolant Accidents (inside containment)

Chapter 15 provides the radiological consequences of the above listed events.

6.3.3.1 ECCS Bases for Technical Specifications

The MLHGR operating limits, used in the ECCS performance analysis, are documented in each cycle-specific Core Operating Limits Report (COLR), which is referenced to by the Technical Specifications. Minimum ECCS functional requirements are specified in Subsections 6.3.3.4 and 6.3.3.5, and testing requirements are discussed within Subsections 6.3.2.7 and 6.3.3.8. Limits on minimum suppression pool water level are discussed in Section 6.2.1.

6.3.3.2 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria, extracted from 10 CFR 50.46, are evaluated below.

Criterion 1: Peak Cladding Temperature (PCT)

"The calculated maximum fuel element cladding temperature shall not exceed 2200°F," which is equivalent to 1204°C. Conformance to Criterion 1 is shown for the system response analyses within Subsection 6.3.3.7 and specifically in Table 6.3-5 (Summary of LOCA Analysis Results). For each plant-specific application, conformance will be re-confirmed for the limiting break. See Subsection 6.3.6 for Combined Operating License (COL) information requirements.

Criterion 2: Maximum Cladding Oxidation

"The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation." Conformation to Criterion 2 is shown in Table 6.3-5. For each plant-specific application, conformance will be re-confirmed for the limiting break. See Subsection 6.3.6 for COL information requirements.

Criterion 3: Maximum Hydrogen Generation

"The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react." Conformance to Criterion 3 is addressed in Table 6.3-5.

Criterion 4: Coolable Geometry

"Calculated changes in core geometry shall be such that the core remains amenable to cooling." As described in Reference 6.3-2, Section III.A, conformance to Criterion 4 is demonstrated by conformance to Criteria 1 and 2.

Criterion 5: Long-Term Cooling

"After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." Conformance to Criterion 5 is assured for any LOCA where the water level can be restored and maintained at a level above the top of the core. For ESBWR, the core never uncovers during a design basis LOCA event due to flow from the GDCS pools. Flow from the suppression pool via the GDCS equalizing lines maintains the water level in the vessel above the core for 72 hours (refer to section 6.2).

6.3.3.3 Single-Failure Considerations

Subsections 6.3.2 and 6.3.3 discuss the functional consequences of potential operator errors and single failures (including those that might cause any manually controlled electrically operated valve in the ECCS to move to a position which could adversely affect the ECCS). Because the Isolation Condensers and the Standby Liquid Control system are single failure proof, it was shown that all potential single failures are no more severe than one of the single failures identified in Table 6.3-6.

It is therefore only necessary to consider each of these single failures in the ECCS performance analyses.

As shown in Table 6.3-6, the worst single failure following a LOCA is the failure of either 1 DPV (or SRV) or 1 GDCS injection valve. The failure of a DPV or SRV results in the greatest reduction in the depressurization rate from ADS actuation and results in a delay in GDCS injection. The failure of one GDCS injection valve results in the greatest reduction in the GDCS reflooding rate. Each break location is evaluated assuming each failure to determine the most limiting single failure for that LOCA event.

6.3.3.4 System Performance During the Accident

In general, the system response to an accident can be described as:

- receiving an initiation signal;
- a small lag time (to open all valves and depressurize the vessel); and
- finally, the GDCS flow entering the vessel.

Key ECCS actuation setpoints and time delays for all the ECCS systems are provided in Table 6.3-1.

The ADS actuation logic includes a 10 second delay to confirm the presence of a low water level (Level 1.5) initiation signal.

The GDCS flow delivery rates are addressed within Subsection 6.3.3.7 for the various breaks analyzed. Piping and instrumentation for the GDCS and ADS are addressed within Subsection 6.3.2. The operational sequence of ECCS for the limiting case is shown in Table 6.3-7.

Operator action is not required, except as a monitoring function, following any LOCA.

6.3.3.5 Use of Dual Function Components for ECCS

The systems of the ECCS (specifically, the ADS and the GDCS) are designed to accomplish only one function; to cool the reactor core following a LOCA. To this extent, components or portions of these systems (except for the pressure relief function of SRVs) are not required for operation of other systems. Because the SRV opens either on ADS initiating signal or by spring-actuated pressure relief in response to an overpressure condition, no conflict exists.

6.3.3.6 Limits on ECCS Parameters

Subsections 6.3.3.1 and 6.3.3.7.1 and the tables referenced in those sections provide limits on ECCS parameters. Any number of components in any given system may be out of service, up to the entire system. The maximum allowable out-of-service time is a function of the level of redundancy and the specified test intervals.

6.3.3.7 ECCS Performance Analysis for LOCA

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

For the system response analysis, the TRACG model was used. The input variables are based on nominal values. A conservative assumption made in the analysis is that all preferred power is lost simultaneously with the initiation of the LOCA. The significant input variables used for the response analysis are listed in Table 6.3-1 and the MLHGRs as defined in Chapter 4.

6.3.3.7.2 Accident Description

The sequence of events for the 4 limiting break locations are shown in Tables 6.3-7 through 6.3-10.

6.3.3.7.3 Break Spectrum Calculations

A representative set of cases was analyzed to evaluate the spectrum of postulated break sizes and locations to demonstrate ECCS system performance. A summary of results of these calculations is shown in Table 6.3-5 and graphically in Figure 6.3-5.

Conformance to the acceptance criteria [PCT ≤ 1204°C (2200°F), local oxidation ≤ 17% and core-wide metal-water reaction ≤ 1%] is demonstrated for the core loading as shown within Section 4.3. Results for the limiting break for each bundle design in a plant will be supplied by the COL applicant/licensee (see Subsection 6.3.5). Details of calculations for specific breaks are included in subsequent paragraphs.

6.3.3.7.4 Large Line Breaks Inside Containment

Because the ESBWR design has no recirculation lines, the maximum DPV stub tube break, the maximum inside steam line break, the maximum feedwater line break, and the maximum RWCU/SDC suction line break are the largest area break locations. The total stub tube break flow includes back flow from the IC through the IC return line. Similarly, the total RWCU/SDC suction line break flow includes flow through the bottom head drain line. The maximum inside steam line break and the maximum feedwater line break were analyzed as representative cases for this group of breaks. Important output variables from these cases are shown in Table 6.3-5 and Figures 6.3-7 through 6.3-22.

The variables are:

- minimum critical power ratio (MCPR) as function of time;
- chimney water level as a function of time;
- downcomer water level as a function of time;
- system pressures as a function of time;
- steamline and break flow as a function of time;
- ADS flow as a function of time;
- flow into vessel as a function of time; and
- PCT as a function of time.

6.3.3.7.5 Intermediate Line Breaks Inside Containment

The only case in this group of breaks is the IC return line break. Since the ESBWR response to this LOCA event is rapid depressurization through the ADS values, the results for this case are similar to the large steam line break case previously discussed.

6.3.3.7.6 Small Line Breaks Inside Containment

For these cases the GDCS injection line break and the bottom head drain line break were analyzed. RWCU/SDC line. Important variables from these analyses are shown in Table 6.3-5 and Figures 6.3-23 through 6.3-38.

6.3.3.7.7 Line Breaks Outside Containment

This group of breaks is characterized by a rapid isolation of the break. Since for ESBWR the isolation condenser system is part of the ECCS network, once the break is isolated the isolation condensers will control the vessel pressure and level thereby terminating the transient.

6.3.3.7.8 Summary of ECCS-LOCA Performance Analysis Results

From the results presented in the above subsections it is concluded that for the ESBWR there is no core uncovery or heatup for any design basis LOCA. Also, the system response to both large and small break LOCAs is similar i.e. rapid vessel depressurization followed by GDCS injection to maintain the vessel water level. Thus the key LOCA result of minimum chimney static head above vessel zero is similar for all LOCA events (see Table 6.3-5) with the results for maximum feedwater line break with 1 GDCS valve or 1DPV failure) being slightly more limiting than the other LOCA cases.

The results of the limiting case for each bundle design will be provided by the COL applicant to the USNRC for information.

6.3.3.7.9 Bounding LOCA Evaluations

Consistent with previous LOCA model application methodology, LOCA evaluations in the previous sections are compared to a bounding result. Table 6.3-11 presents the significant plant variables that were considered in the determination of the bounding LOCA result. Because the ESBWR LOCA results have large margins to the acceptance criteria, a conservative LOCA evaluation was performed which bounds the 95% probability LOCA results. This bounding LOCA result was calculated by varying all plant variables in the conservative direction simultaneously. The limiting feedwater line break cases (refer to Subsection 6.3.3.7.8) and the bottom drain line break (the second most limiting break location, refer to Table 6.3-5) were evaluated. The results of these calculation are given in Table 6.3-5 with the feedwater line break with a GDCS injection valve failure result in the lowest minimum chimney static head level above vessel zero. Because the ESBWR results have large margins to the 10 CFR 50.46 licensing acceptance criteria, the ESBWR licensing LOCA results can be based on this bounding LOCA case.

6.3.3.8 ECCS-LOCA Performance Analysis Conclusions

Having shown compliance with the applicable acceptance criteria, it is concluded that the ECCS would perform its function in an acceptable manner. For a reference core loading is described in Section 4.3.

6.3.3.9 ECCS Performance Tests

All systems of the ECCS are tested for their operational ECCS function during the preoperational and/or startup test program. As applicable, each component is tested for power source, range, setpoint, limit switch setting, torque switch setting, etc. Subsection 6.3.2.7.4 contains additional details on GDCS testing, and Subsection 6.3.2.8.4 contains additional details on ADS testing. See Chapter 14 for a thorough discussion of preoperational testing for these systems.

6.3.3.10 Reliability Tests and Inspections

The average reliability of a standby (non-operating) safety system is a function of the duration of the interval between periodic functional tests. The factors considered in determining the periodic test interval of the ECCS are:

- The desired system availability (average reliability);
- The number of redundant functional system success paths;
- The failure rates of the individual components in the system; and
- The schedule of periodic tests (simultaneous versus uniformly staggered versus randomly staggered)

The SRVs and DPVs are all tested during plant initial power ascension per Regulatory Guide 1.68, Appendix A.

An ADS logic system functional test and simulated automatic operation of all ADS logic channels are to be performed at least once per plant operating interval between reactor refuelings. Instrumentation channels are demonstrated operable by the performance of a channel functional test and a trip unit calibration at least once per month and a transmitter calibration at least once per operating interval.

All SRVs, which include those used for ADS, and DPVs are bench tested to establish lift settings in compliance with ASME Code Section XI.

Testing of the initiating instrumentation and controls portion of the ECCS is discussed in Subsection 7.3.1. The emergency power system, which supplies electrical power to the ECCS is tested as described in Subsection 8.3.1. The frequency of testing is specified in the plant-specific Technical Specifications. Components inside the drywell can be visually inspected only during periods of access to the drywell.

6.3.4 Instrumentation Requirements

Design details including redundancy and logic of the ECCS instrumentation are discussed in Section 7.3.

All instrumentation required for automatic and manual initiation of the GDCS and ADS is discussed in Subsection 7.3.1, and is designed to meet the requirements of IEEE-279 and other applicable regulatory requirements. The GDCS and ADS can be manually initiated from the control room.

The GDCS is automatically initiated on low reactor water level. The ADS is automatically actuated by sensed variables for reactor vessel low water level. The GDCS flow into the RPV begins when reactor pressure decreases below the GDCS pool pressure plus its gravity head.

6.3.5 Combined Operating License Information

6.3.5.1 ECCS Performance Results

The initial core specific peak cladding temperature and oxidation fraction for each fuel bundle design, based on the limiting break size will be provided by the COL licensee to the USNRC for information.

6.3.5.2 ECCS Testing Requirements

In accordance with the plant-specific Technical Specifications, the COL licensee will perform a test every refueling in which each ECCS subsystem is actuated through the emergency operating sequence.

6.3.5.3 Limiting Break Results

For the initial core, the results for the limiting break for each bundle design will be provided to the USNRC by the COL licensee.

6.3.6 References

6.3-1 GE Nuclear Energy, "Depressurization Valve Development Test Program Final Report," GEFR-00879, October 1990.

6.3-2 GE Nuclear Energy, "TRACG Application for ESBWR," NEDC-33083P-A, Class III (Proprietary), March 2005.

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

A. Plant Parameters		
Variable	**Units**	**Value**
Core thermal Power	MWt	4500
Vessel Steam Output	kg/hr [lbm/hr]	8.76×10^6 19.31×10^6
Vessel Steam Dome Pressure	MPa (absolute) [psia]	7.17 1040

B. Emergency Core Cooling System Parameters		
B.1 ECCS Initiation Signals		
Variable	**Units**	**Value**
Initiating Signals:		
Level 1.5	meters (above TAF) [ft] (above TAF)	5.547 [18.20]
plus High Drywell Pressure	kPa (absolute) [psia]	34.457 [5]
or		
Level 1.5	meters (above TAF) [ft] (above TAF)	5.547 [18.20]
plus Delay Timer Timed Out	min	15
or Level 1	meters (above TAF) [ft] (above TAF)	3.547 [11.64]
Maximum Allowable Time Delay to Confirm ECCS-LOCA Signal	sec	10

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

B.2 Gravity-Driven Core Cooling System

Variable	Units	Value
Initiating Signal	—	ECCS-LOCA confirmed initiating signal (See B.1)
GDCS Injection valve timer delay	sec	150
Minimum drainable inventory per GDCS pool		See Table 6.3-2
Minimum elevation of GDCS pool surfaces above the RPV nozzles		See Table 6.3-2
GDCS drain line loss coefficient (k/A2)	$1/m^4$ [$1/ft^4$]	$12.587*10^3$ [$1.458*10^6$]

B.3 Isolation Condenser System

Variable	Units	Value
Initiating Signal	—	Loss of feedwater
Maximum Sensor Response Time	sec	2
Heat Removal Capacity per Unit	MW	33.75
Minimum Drainable Liquid Volume per System	m^3 [ft^3]	4.98

B.4 Standby Liquid Control System

Variable	Units	Value
Initiating Signal	—	DPV actuation (See B.5)
Liquid Volume per Tank	m3 [ft3]	7.8

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

B.5 Automatic Depressurization Subsystem

Variable	**Units**	**Value**
Initiating Signal	—	ECCS-LOCA confirmed initiating signal (See B.1)
Valve Actuation Sequence:		
5 ADS	sec	0
5 ADS	sec	10
3 DPVs	sec	50
2 DPVs	sec	100
2 DPVs	sec	150
1 DPVs	sec	200
Total Number of Safety/Relief Valves With ADS Function	—	10
Total Min. ADS Flow Capacity at Vessel Pressure	kg/hr MPa (gauge) [lbm/hr] [psig]	4.356×10^6 8.618 [9.61×10^6] [1250]
Total Number of Depressurization Valves	—	8
Total min. DPV flow capacity at vessel pressure	kg/hr MPa (gauge) [lbm/hr] [psig]	6.89×10^6 7.481 [15.2×10^6] [1085]

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

Total max. DPV flow capacity at vessel pressure	kg/hr MPa (gage) [lbm/hr] [psig]	8.47×10^6 7.481 [18.7×10^6] [1085]
C. Fuel Parameters *		
Variable	**Units**	**Value**
Fuel type	—	See Chapter 4
Peak Linear Heat Generation Rate (Bounding))	kW/m [kW/ft]	44 13.4
Initial Minimum Critical Power Ratio (Bounding)	—	1.12

Table 6.3-2

GDCS Design Basis Parameters

Parameter	Value
Number of separate/independent GDCS divisions	4
Per division, number of (short-term core cooling injection) lines from its GDCS pool	1
Per division, number of injection line RPV nozzles	2
Per division, number of equalizing line RPV nozzles	1
Minimum total drainable inventory (for 3 GDCS pools)	1760 m^3 (62,150 ft^3)
Minimum elevation of GDCS pool surfaces above the RPV nozzles	13.3 m (43.6 ft)
Minimum long-term core cooling flow delivered by the GDCS equalizing lines for a ΔP of 9.12 kPa (1.32 psid) across the equalizing lines	22.7 m^3/s (100 gpm)
Minimum flow through the deluge lines required to flood the lower drywell region	70 kg/sec (1541 m/sec)*
Minimum available suppression pool water inventory 1 meter above TAF	334m^3 (11,800 ft^3)
Minimum GDCS equalizing line driving head	1.0 m (3.3 ft)

* Core melt scenario instead of ECCS performance evaluation scenarios.

Table 6.3-3

GDCS Surveillance Testing

Component	Type of Test	Component to be aligned
Check Valves	Functional tests: flushing the line from dedicated test connection	Opening of test line isolation valves
Squib Valve Initiators	Explosive tests	Each initiator is tested in laboratory after replacement
Flushing of injection line to remove any possible plugging	Flushing during refueling outage	Alignment of test connection lines
Venturi within GDCS-RPV injection nozzles	Flushing during refueling outage	Alignment of test connection lines
Deluge Line Flushing	Flushing lines from dedicated connection to prevent crud build up during refueling outage	Alignment of flushing connection lines

Table 6.3-4

DPV Design and Performance Parameters

Parameter	Value/Description
Inlet side design pressure	10.34 MPa gauge (1500 psig) at a design temperature of 313°C (595°F)
Outlet-side design pressure	4.97 MPa gauge (720 psig) at a design temperature of 264°C (508°F)
Material of valve bodies	304 or 316 stainless steel (304L or 316L stainless steel where welding is employed)
Design life	60 yrs
Design operating fluid conditions	Saturated steam flow ranging from 95% quality to 2.8°C (5°F) superheat
Rated flow capacity of each DPV (based on dry saturated steam conditions and a flow-induced backpressure of up to 50% of the inlet pressure)	Between 8.62×10^5 and 1.06×10^6 kg/hr (1.90×10^6 to 2.33×10^6 lbm/hr) at an inlet pressure of 7.48 MPa gauge (1085 psig)
Specified response times (opening time to full rated capacity) of the DPVs, with a static backpressure of up to 50% of the inlet pressure	• 0.45 second or less with an inlet pressure of 6.89 MPa gauge (1000 psig) or greater • 5 seconds or less with an inlet pressure between 6.89 MPa gauge (1000 psig) and 0.69 MPa gauge (100 psig). • 30 seconds or less with an inlet pressure below 0.69 MPa gauge (100 psig).

Table 6.3-5

Summary of ECCS-LOCA Performance Analyses

Break Location	Break Size m² (ft²)	Minimum Chimney Static Head* Level Above Vessel Zero Per Active Single Failure m (ft)			PCT **	Maximum Local and Core Wide Oxidations (%) ***	Minimum Downcomer Collapsed Water Level Above Vessel Zero Per Active Single Failure m (ft)			Change in MCPR From Start of Event	Change in RPV Press. From Start of Event
		1 SRV	1 GDCS	1 DPV			1 SRV	1 GDCS	1 DPV		
Based on standard TRACG evaluation model:											
Feedwater Line	0.08387 (0.9028)	8.86 (29.07)	8.89 (29.17)	8.84 (29.00)	No heatup	<1.0	6.45 (21.16)	6.39 (20.96)	6.19 (20.31)	Increases	Decreases
Steam Line Inside Containment	0.09832 (1.058)	9.04 (29.66)	9.10 (29.86)	9.10 (29.86)	No heatup	<1.0	7.48 (24.54)	7.28 (23.88)	7.37 (24.18)	Increases	Decreases
Bottom Head Drain Line	0.004052 (0.04361)	8.94 (29.33)	8.91 (29.23)	9.04 (29.66)	No heatup	<1.0	6.88 (22.57)	6.61 (21.69)	6.71 (22.01)	Increases	Decreases
GDCS Injection Line	0.004561 (0.04910)	9.08 (29.79)	9.03 (29.63)	8.98 (29.46)	No heatup	<1.0	6.98 (22.90)	6.9 (22.64)	6.85 (22.47)	Increases	Decreases
Based on bounding values:											
Feedwater Line	0.08387 (0.9028)	—	8.17 (26.80)	8.23 (27.00)	No heatup	<1.0	—	5.22 (17.13)	5.19 (17.03)	Increases	Decreases
Bottom Head Drain Line	0.004052 (0.04361)	—	8.42 (27.62)	—	No heatup	<1.0	—	5.57 (18.27)	—	Increases	Decreases

* Chimney static head is calculated by adding the static head in the chimney to the elevation of the top of the active fuel.

** No break results in core uncovery, and thus, there is no cladding heatup and PCT remains < 316°C (600°F).

*** Maximum local oxidation values are provided. The local oxidation values are calculated using TRACG. This results in a fraction of total cladding volume of fueled rods and water rods of <1.0%. The core-wide metal-water reaction is also <0.1%.

Table 6.3-6

Single Failure Evaluation

Assumed Failure*	Systems Remaining**
One Depressurization Valve	10 SRVs, 7 DPVs, 4 Ics***, 2 SLC system and 4 GDCS with 8 Injection Lines
One Safety/Relief Valve	9 SRVs, 8 DPVs 4 Ics***, 2 SLC system and 4 GDCS with 8 Injection Lines
One GDCS Injection Valve	10 SRVs, 8 DPVs 4 Ics***, 2 SLC system and 4 GDCS with 7 Injection Lines

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

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

*** A sensitivity study was performed to determine the impact of assuming only 3 ICs available at the start of the LOCA event. The results of this studied showed that the minimum water levels identified in Table 6.3-5 and Figure 6.3-6 would be affected by <10cm. Because a change of this magnitude is minimal compared to the level margins above the top of the active fuel identified in Figure 6.3-6, it is concluded that the assumption of only 3 ICs available for the ECCS-LOCA performance analysis would not invalidate the ECCS-LOCA performance analysis results presented in Section 6.3.3.

Table 6.3-7

Operational Sequence of ECCS For A Feedwater Line Break with Failure of One GDCS Injection Valve (nominal calculation)

Time (sec)	**Events**
~0	Guillotine break of feedwater line inside containment; normal auxiliary power assumed to be lost; Feedwater is lost; Scram signal initiated.
~1	High drywell pressure setpoint for ADS is reached.
~2	Loss of normal auxiliary power confirmed; reactor scram initiated; IC initiated.
~5	Level 3 is reached; Reactor receives second signal to scram.
~9	Level 2 is reached; Reactor isolation timer initiated.
~14	Level 1.5 is reached; Reactor isolation initiated; ADS/GDCS timer initiated.
~17	IC drain valve begins to open.
~24	Level 1.5 signal confirmed; ADS-SRV actuation begins.
~33	IC drain valve fully open.
~74	DPV actuation begins; SLC system signaled to start.
~101	Minimum chimney water level is reached.
~174	GDCS timer expired. GDCS injection valves open.
~250	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
from ~250 to 2000	RPV water level remains higher than Level 0.5. Therefore equalizing line valves are not expected to open for this event.

Table 6.3-8

Operational Sequence of ECCS For a Bottom Drain Line Break

Time (sec)	**Events**
~0	Guillotine break of feedwater line inside containment; normal auxiliary power assumed to be lost; Feedwater tripped; Scram signal initiated.
~2	Loss of normal auxiliary power confirmed; reactor scram initiated; IC initiated.
~5	High drywell pressure setpoint for ADS is reached.
~7	Level 3 is reached; Reactor receives second signal to scram.
~11	Level 2 is reached; Reactor isolation timer initiated.
~16	Reactor isolation initiated.
~17	IC drain valve begins to open.
~19	Level 1.5 is reached; ADS/GDCS timer initiated.
~29	Level 1.5 signal confirmed; SRV actuation begins.
~33	IC drain valve fully open.
~79	DPV actuation begins; SLC system signaled to start.
~169	GDCS timer timed out. GDCS injection valves open.
~338	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins.
~987	Minimum chimney water level is reached.
	Level 0.5 is not reached and equalizing line valves are not open.

Table 6.3-9

Operational Sequence of ECCS For a GDCS Line Break

Time (sec)	**Events**
~0	Guillotine break of feedwater line inside containment; normal auxiliary power assumed to be lost; Feedwater tripped; Scram signal initiated.
~2	Loss of normal auxiliary power confirmed; reactor scram initiated; IC initiated.
~4	High drywell pressure setpoint for ADS is reached.
~7	Level 3 is reached; Reactor receives second signal to scram.
~11	Level 2 is reached; Reactor isolation timer initiated.
~16	Reactor isolation initiated.
~17	IC drain valve begins to open.
~19	Level 1.5 is reached; ADS/GDCS timer initiated.
~29	Level 1.5 signal confirmed; SRV actuation begins.
~33	IC drain valve fully open.
~79	DPV actuation begins; SLC system signaled to start.
~169	GDCS timer timed out. GDCS injection valves open.
~332	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins.
~1281	Minimum chimney water level is reached.
	Level 0.5 is not reached and equalizing line valves are not open.

Table 6.3-10

Operational Sequence of ECCS For a Main Steam Line Break

Time (sec)	**Events**
~0	Guillotine break of feedwater line inside containment; normal auxiliary power assumed to be lost; Feedwater tripped; Scram signal initiated.
~1	High drywell pressure setpoint for ADS is reached.
~2	Loss of normal auxiliary power confirmed; reactor scram initiated; IC initiated.
~7	Level 3 is reached; Reactor receives second signal to scram.
~11	Level 2 is reached; Reactor isolation timer initiated.
~16	Reactor isolation initiated.
~17	IC drain valve begins to open.
~33	IC drain valve fully open.
~190	Level 1.5 is reached; ADS/GDCS timer initiated.
~200	Level 1.5 signal confirmed; SRV actuation begins.
~250	DPV actuation begins; SLC system signaled to start.
~340	GDCS timer timed out. GDCS injection valves open.
~430	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins.
~478	Minimum chimney water level is reached.
	Level 0.5 is not reached and equalizing line valves are not open.

Table 6.3-11

Plant Variables with Nominal and Bounding Calculation Values

Plant Variable	Nominal Value	Bounding Calculation Value*
1. Vessel Steam Dome Pressure	7.17 MPa (1040 psia)	7.274 MPa (1055 psia)
2. Decay Heat	1979 ANS (Figure 6.3-39)	+ 2σ
3. Core Power	Rated	+ 2%
4. PLHGR	44.0 kW/m (13.4 kW/ft)	44.8 kW/m (13.7 kW/ft)
5. Initial MCPR	1.12	1.10
6. Initial Downcomer Level	NWL	NWL – 0.3m
7. Significant TRACG Modeling Parameters**	Nominal	Bounding

* Represents upper 95% or higher probability value.

** Reference 6.3-2.

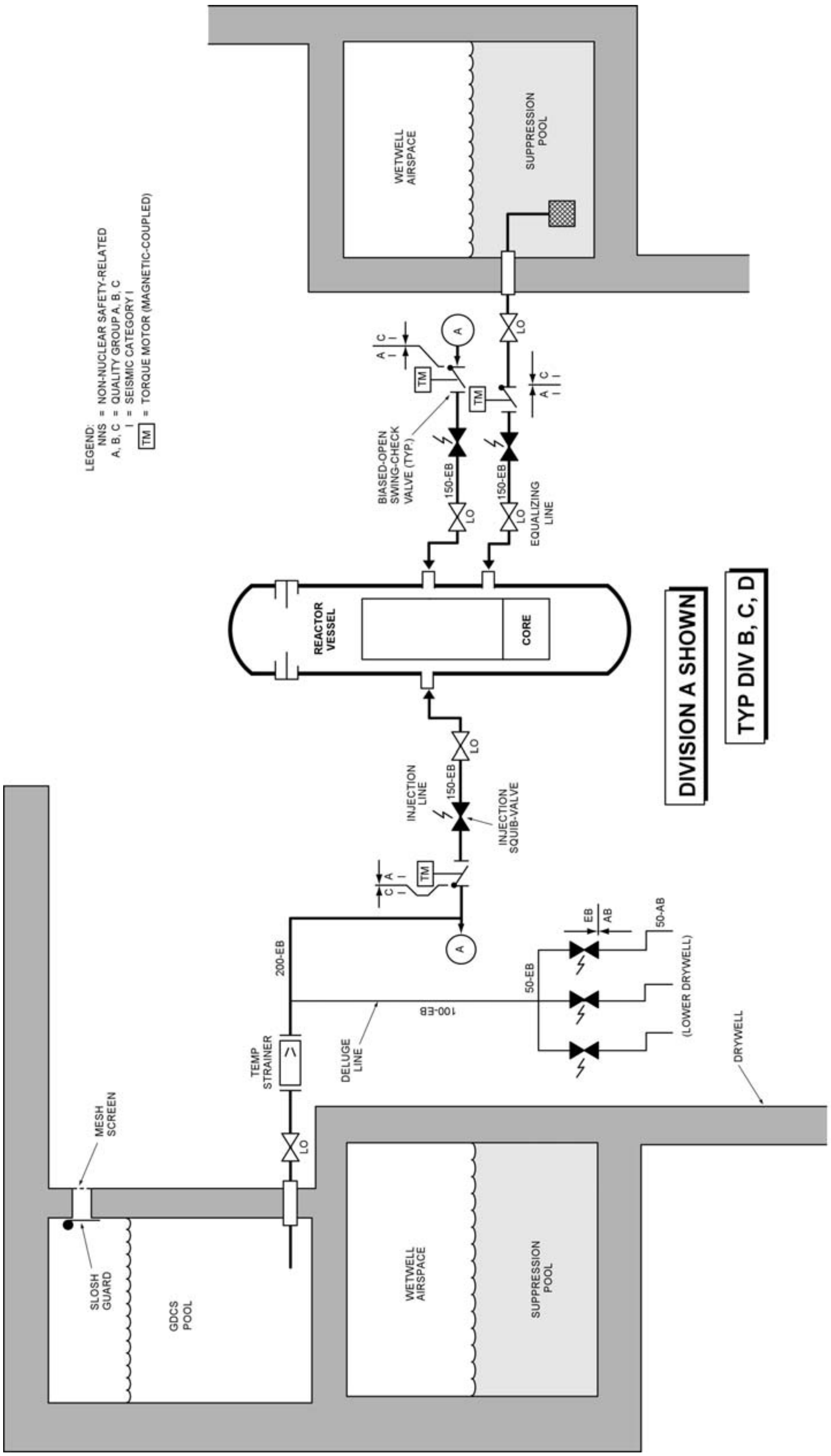


Figure 6.3-1. GDCS Configuration

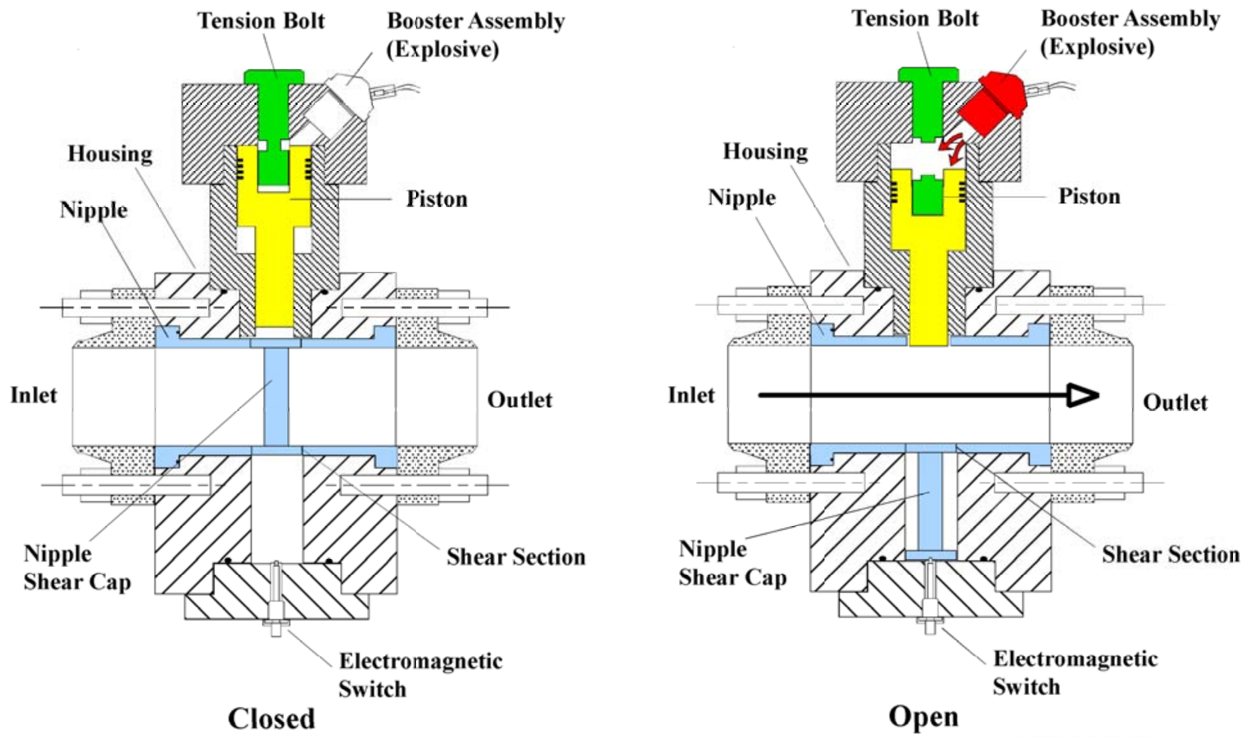
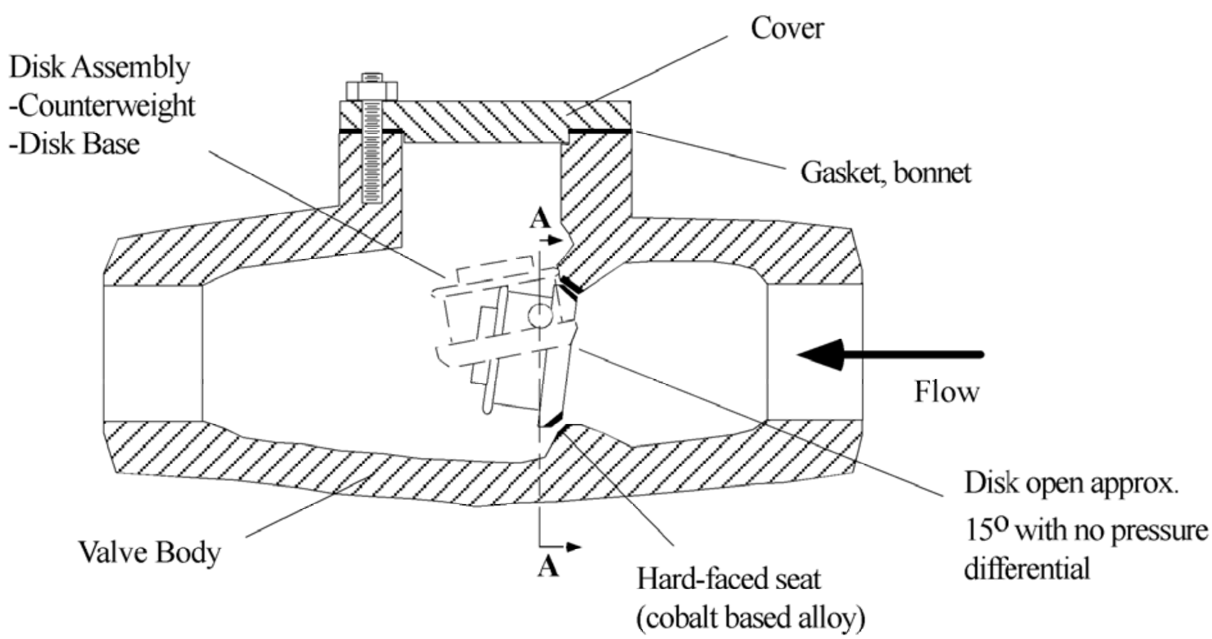


Figure 6.3-2. Typical GDCS Squib Valve



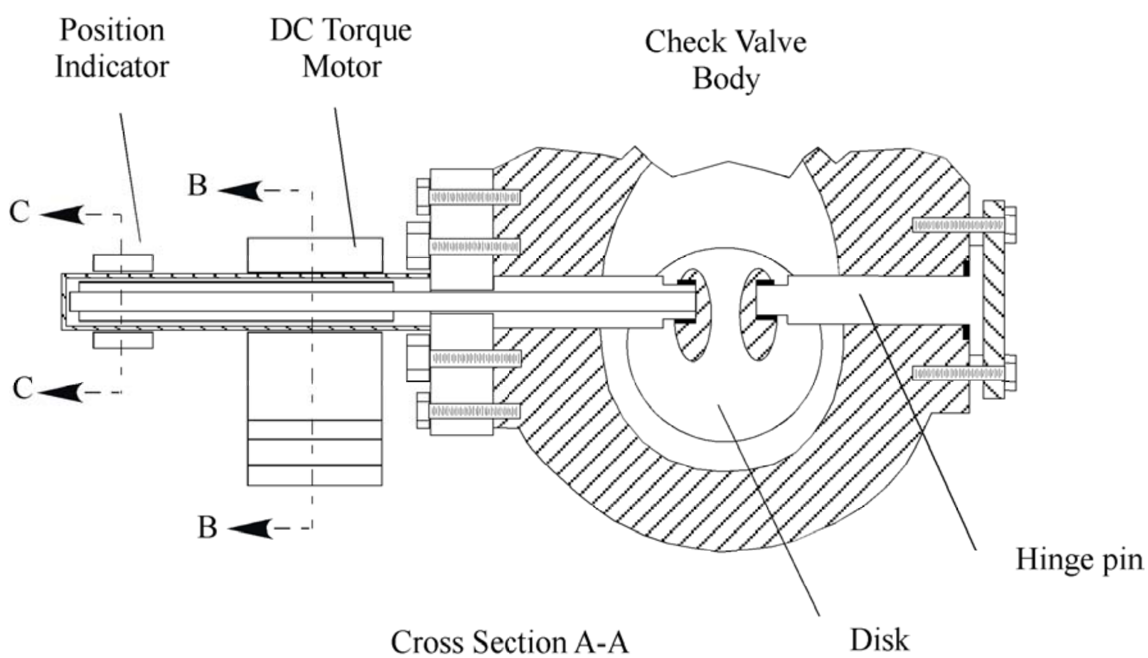


Figure 6.3-3. GDCS Biased Open Check Valve

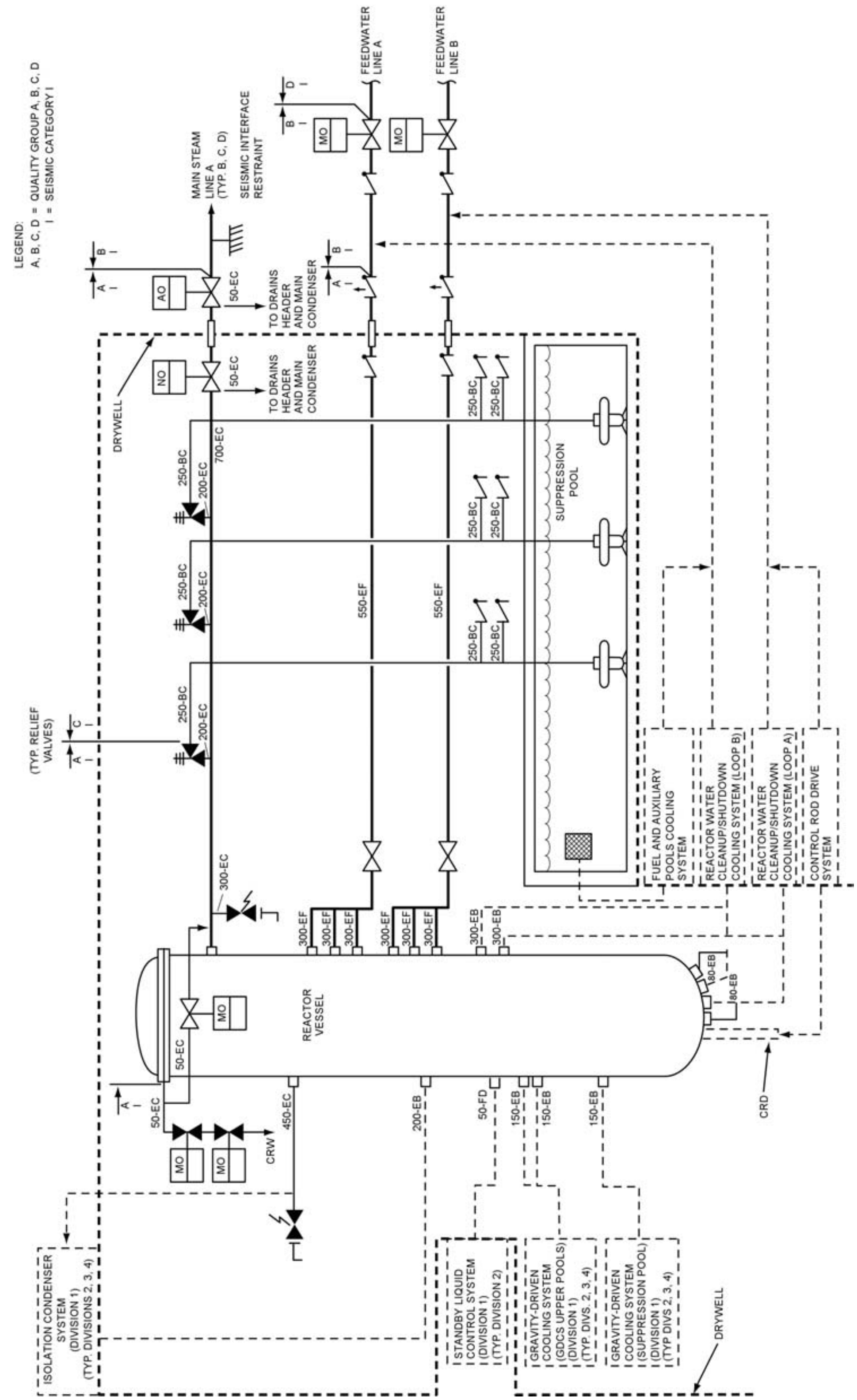


Figure 6.3-4. ADS Component Schematic Diagram

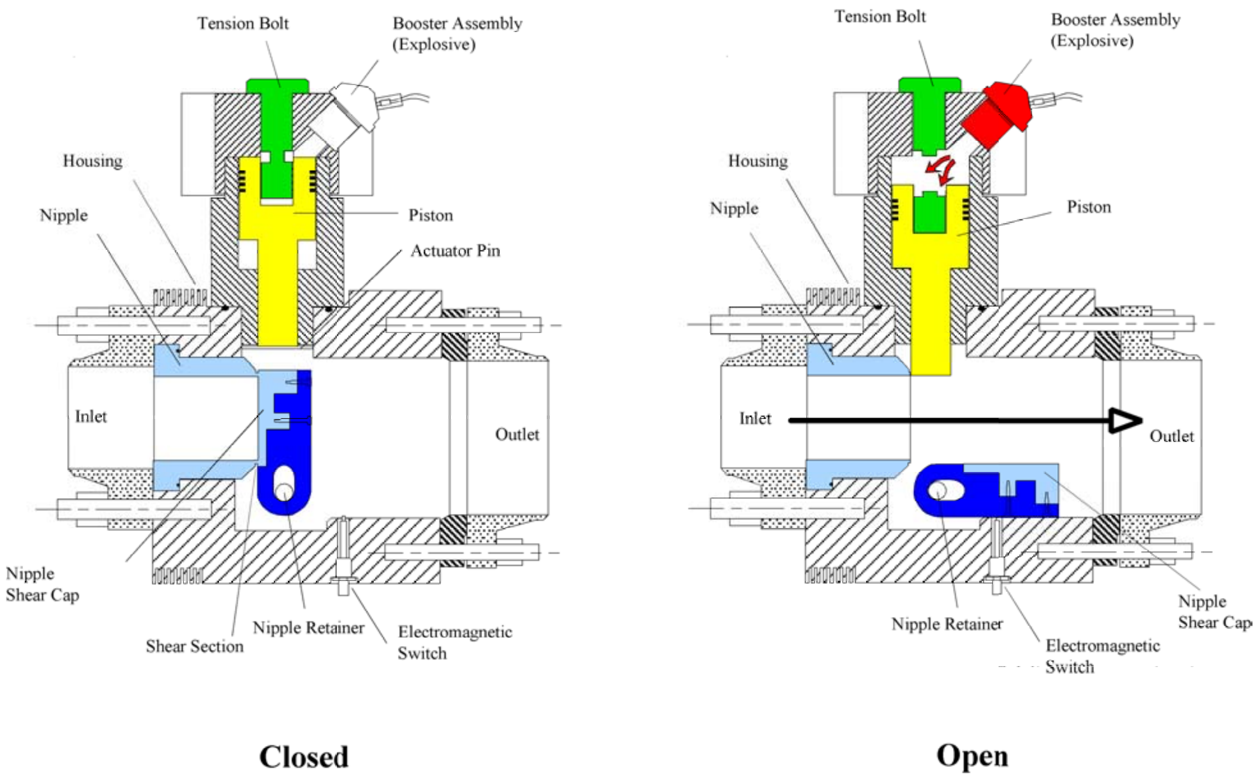


Figure 6.3-5. ADS Depressurization Valve

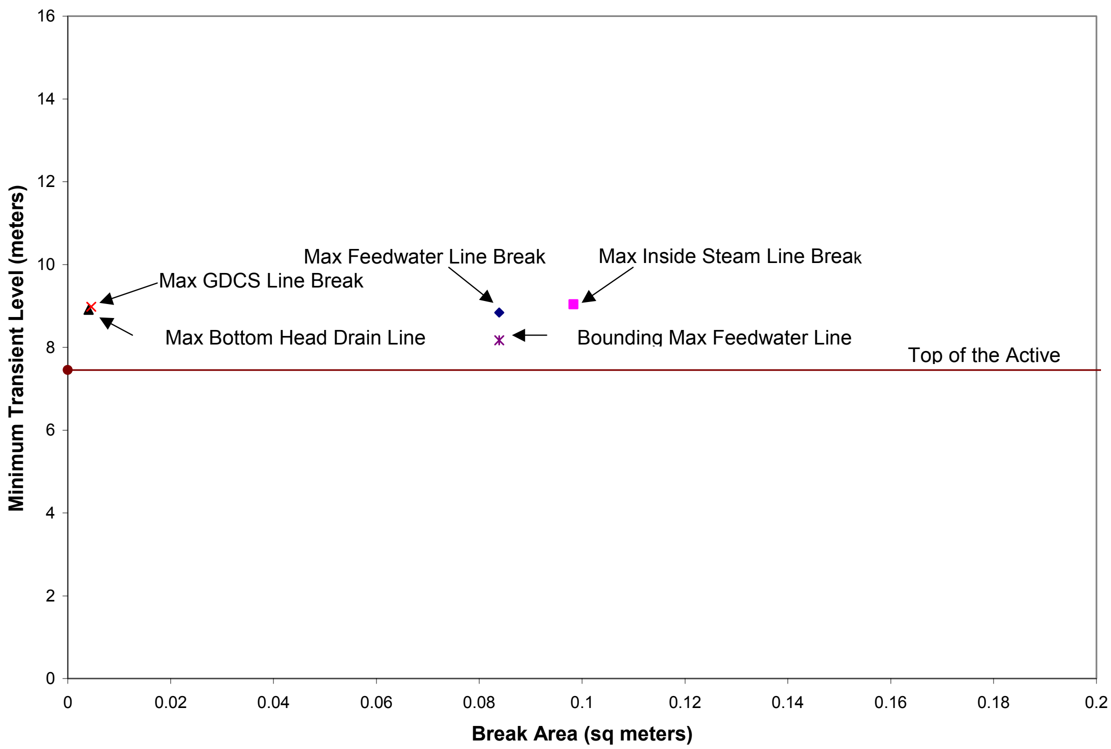


Figure 6.3-6. Minimum Chimney Water Level vs. Break Area

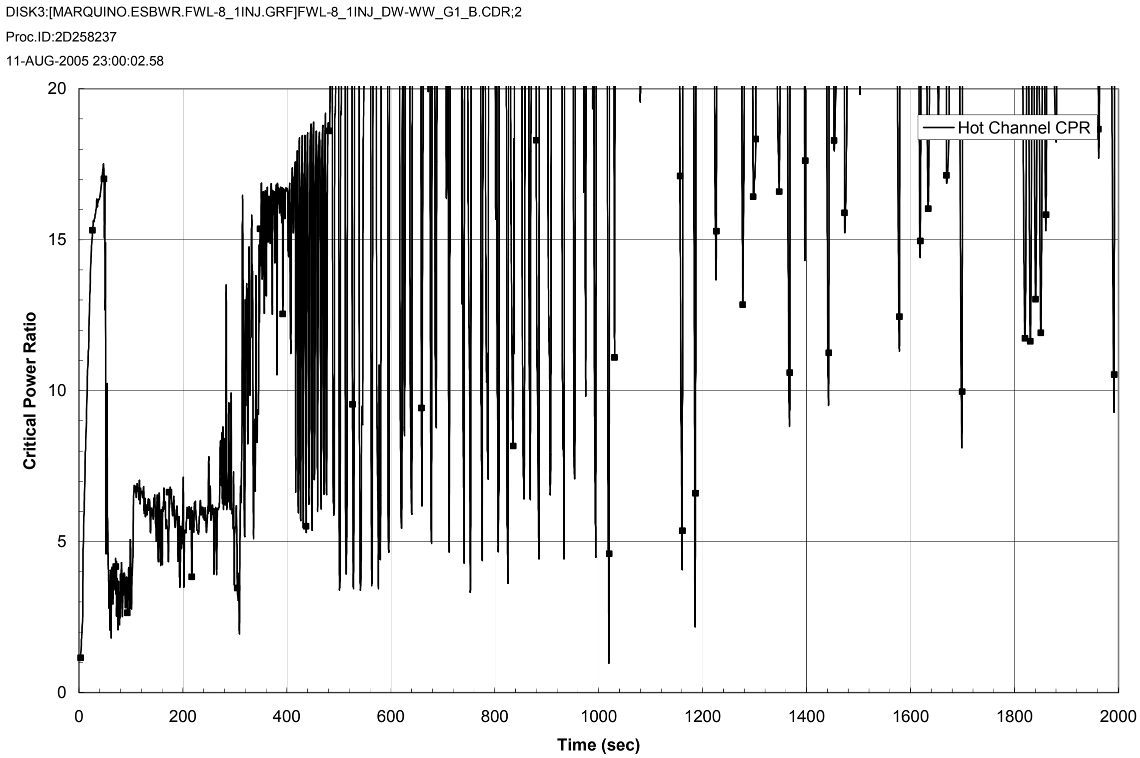


Figure 6.3-7. MCPR, Feedwater Line Break, 1 GDCS Valve Failure

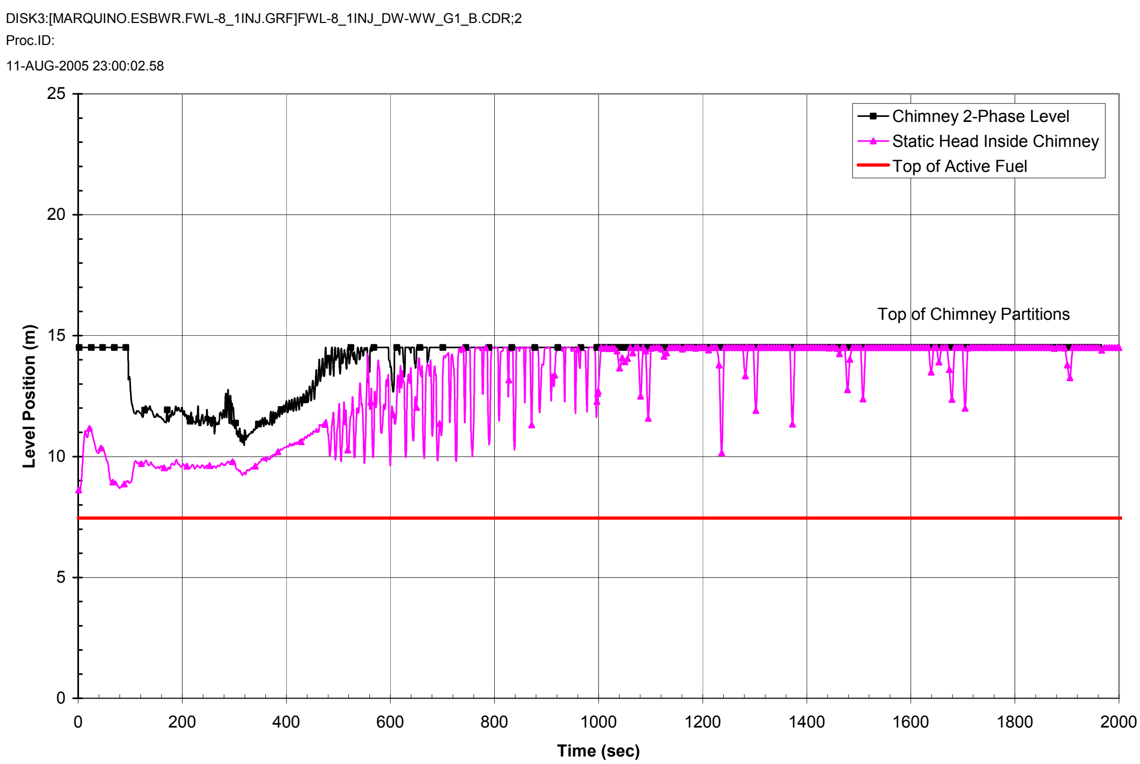


Figure 6.3-8. Chimney Water Level, Feedwater Line Break, 1 GDCS Valve Failure

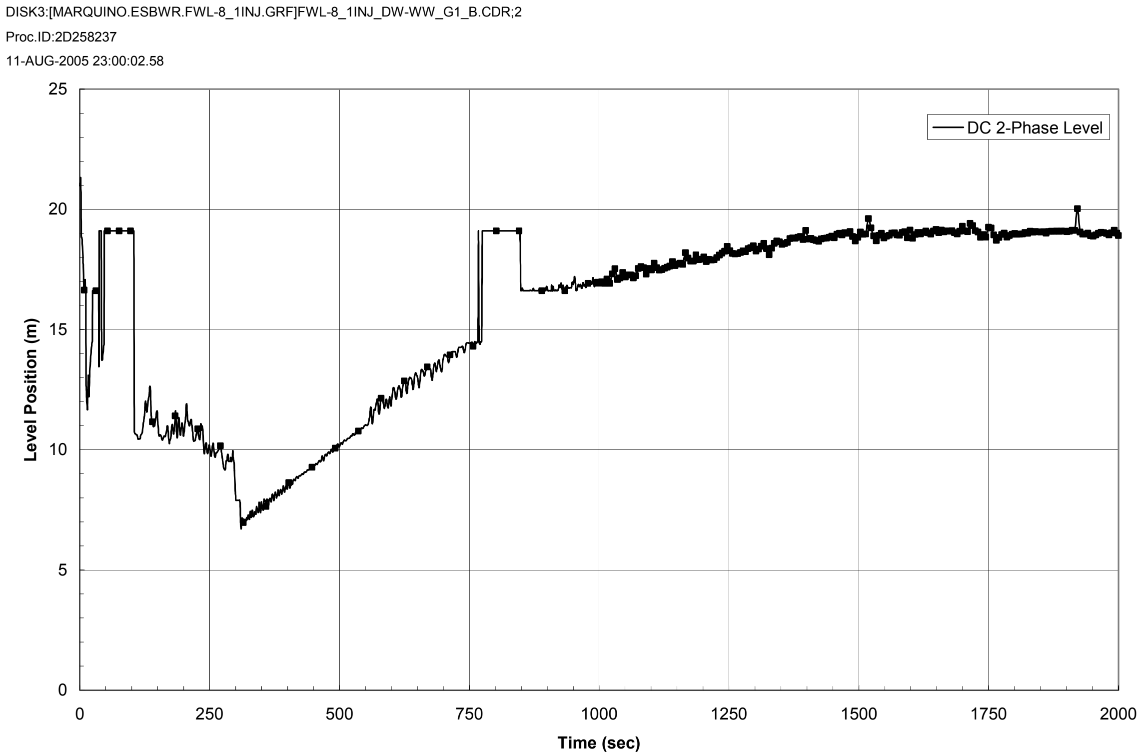


Figure 6.3-9. Downcomer Water Level, Feedwater Line Break, 1 GDCS Valve Failure

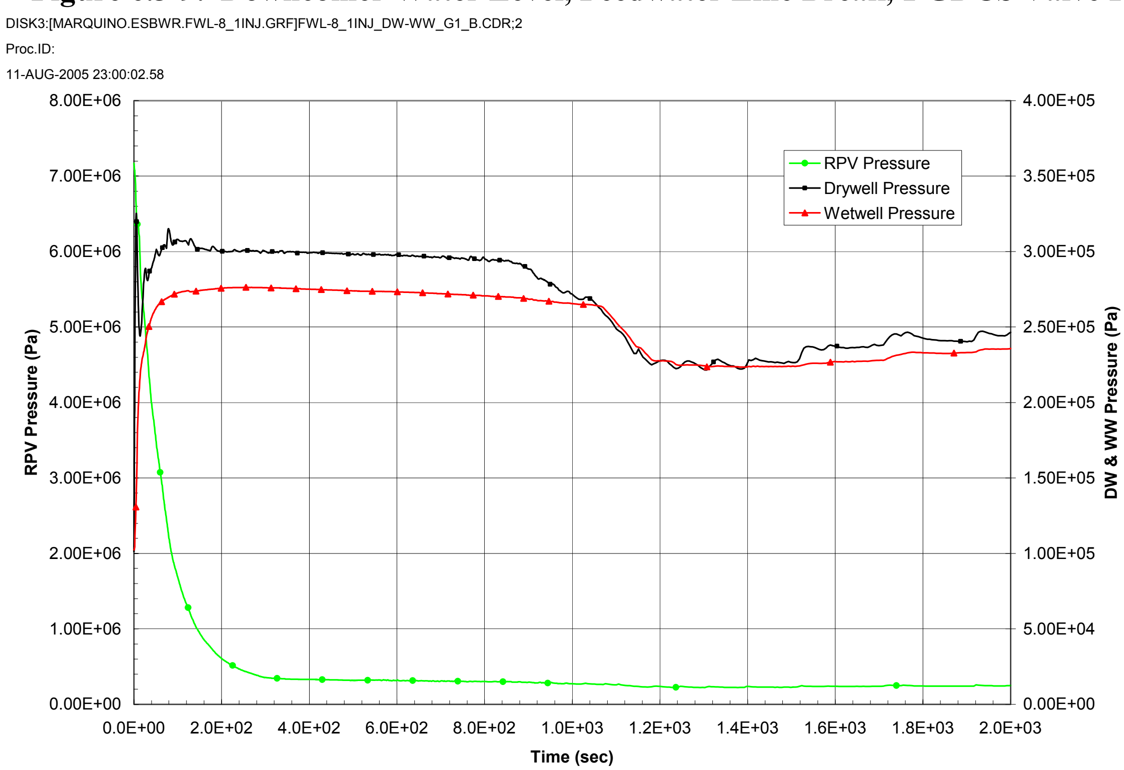


Figure 6.3-10. System Pressures, Feedwater Line Break, 1 GDCS Valve Failure

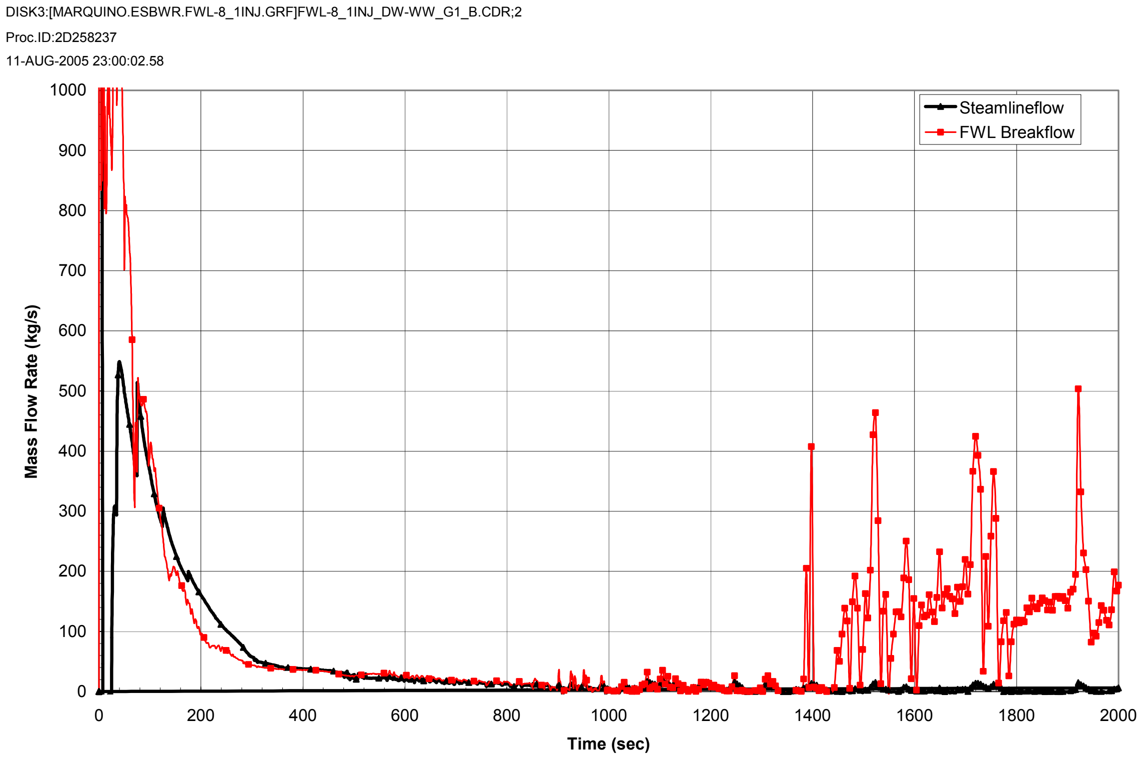


Figure 6.3-11. Steam Line and Break Flow, Feedwater Line Break, 1 GDCS Valve Failure

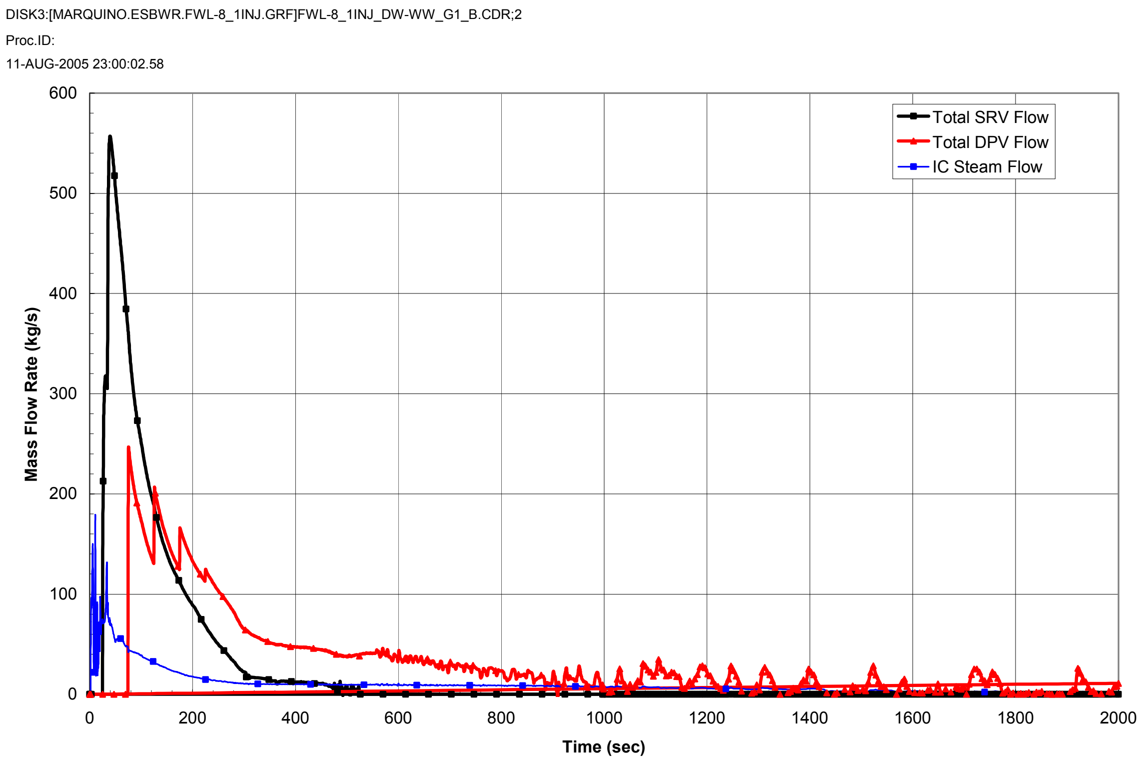


Figure 6.3-12. ADS Flow, Feedwater Line Break, 1 GDCS Valve Failure

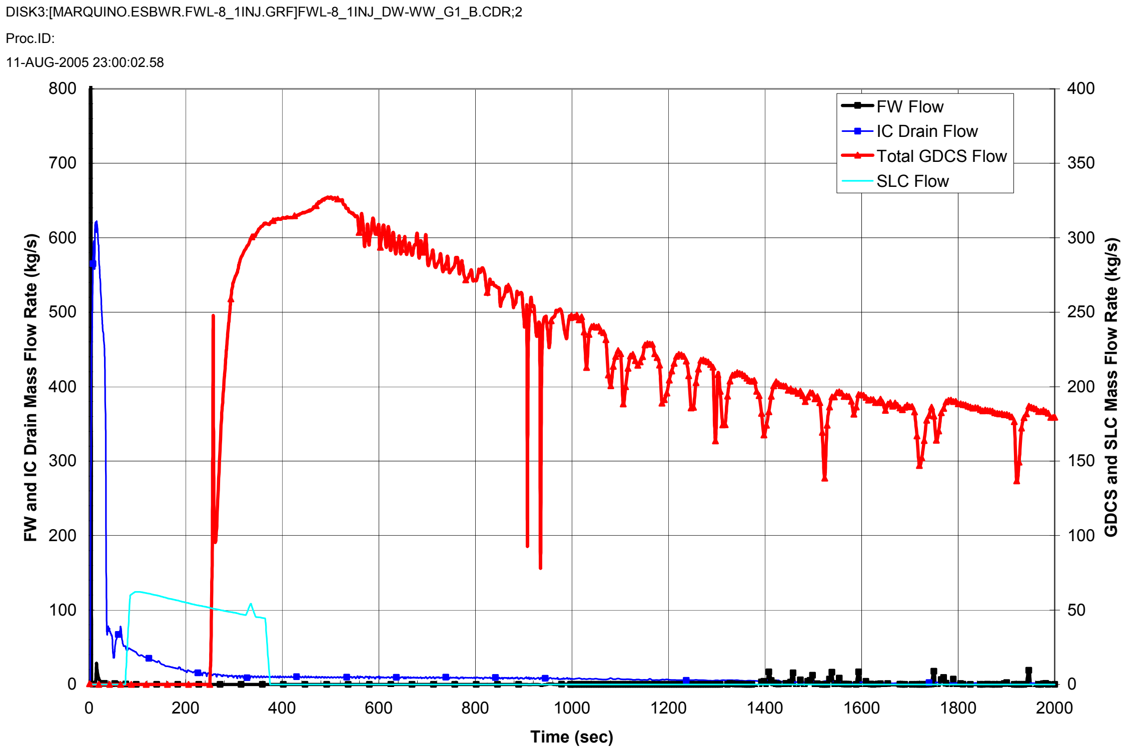


Figure 6.3-13. Flows Into Vessel, Feedwater Line Break, 1 GDCS Valve Failure

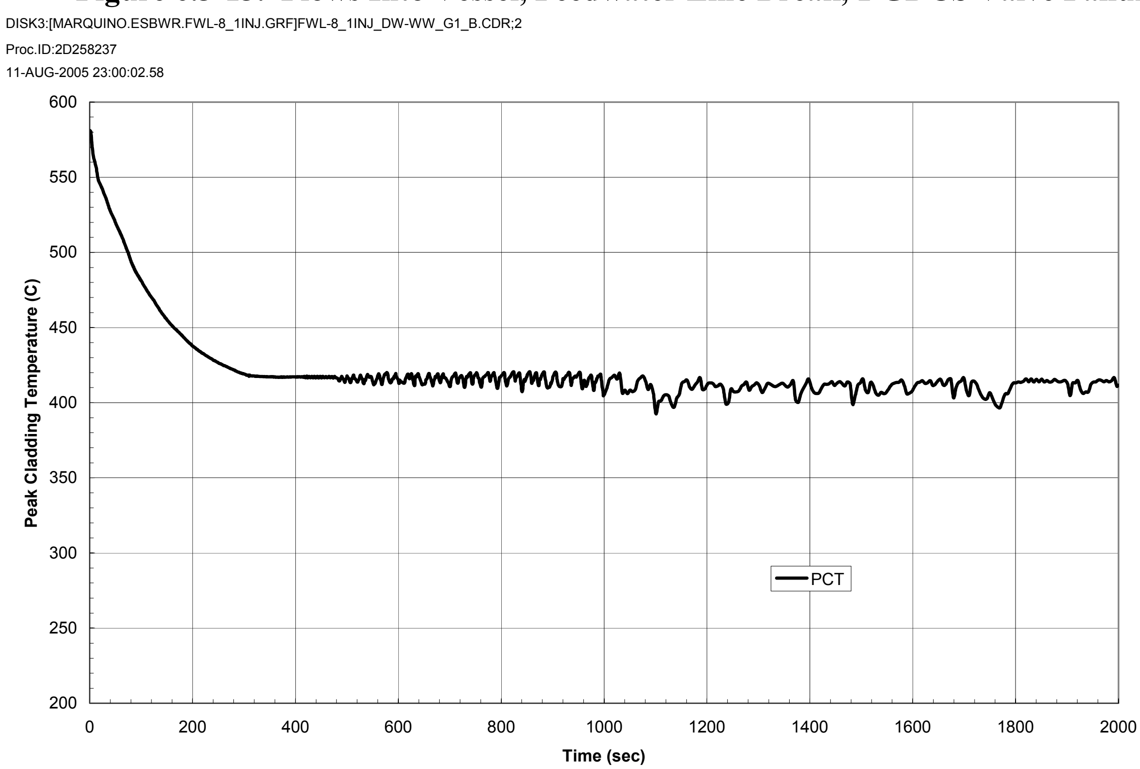


Figure 6.3-14. PCT, Feedwater Line Break, 1 GDCS Valve Failure

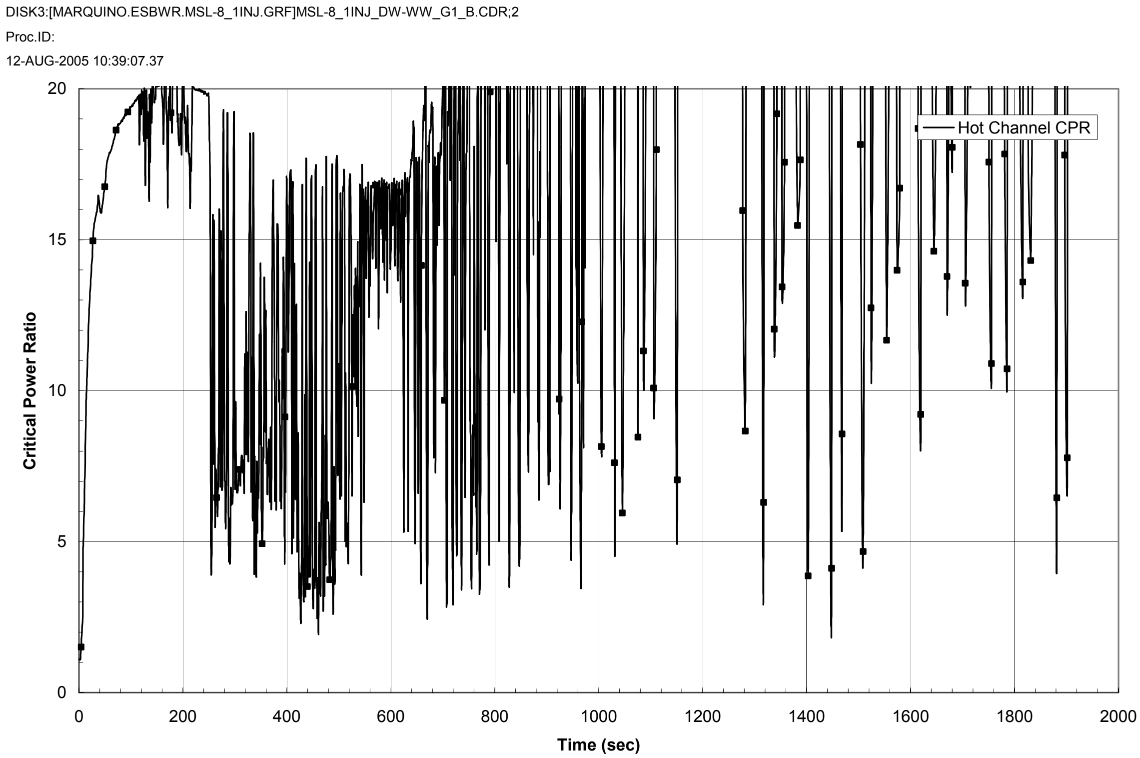


Figure 6.3-15. MCPR, Inside Steam Line Break, 1 GDCS Valve Failure

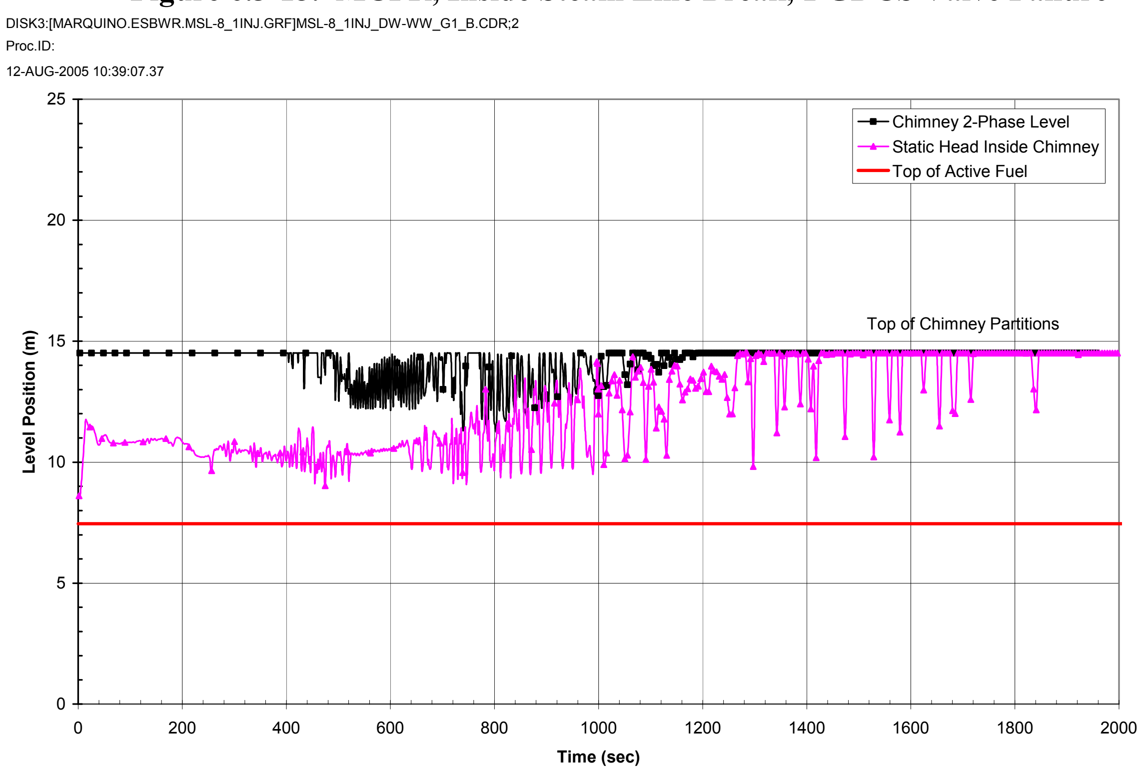


Figure 6.3-16. Chimney Water Level, Inside Steam Line Break, 1 GDCS Valve Failure

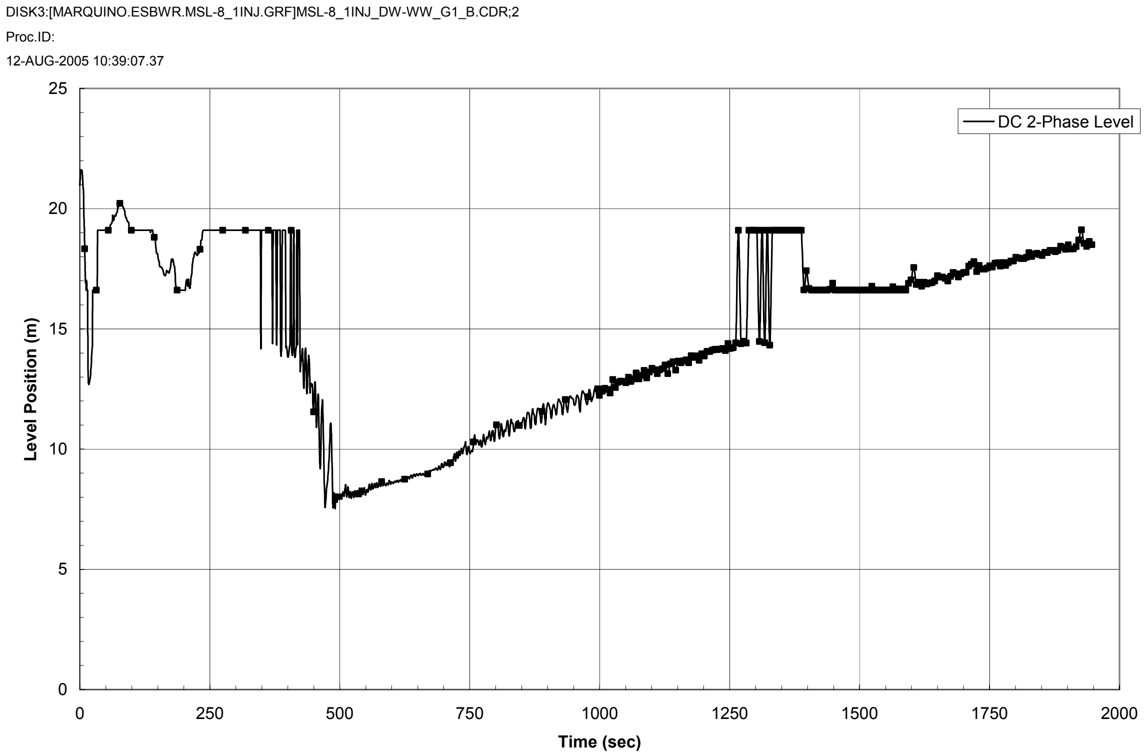


Figure 6.3-17. Downcomer Water Level, Inside Steam Line Break, 1 GDCS Valve Failure

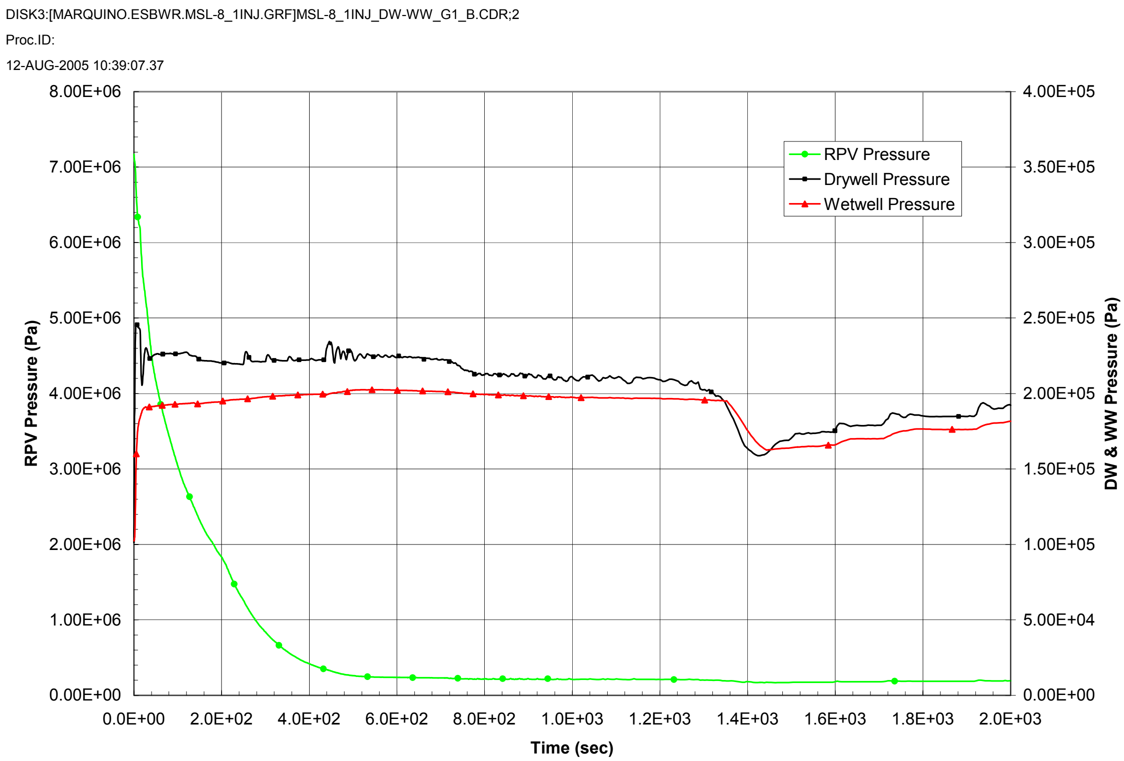


Figure 6.3-18. System Pressures, Inside Steam Line Break, 1 GDCS Valve Failure

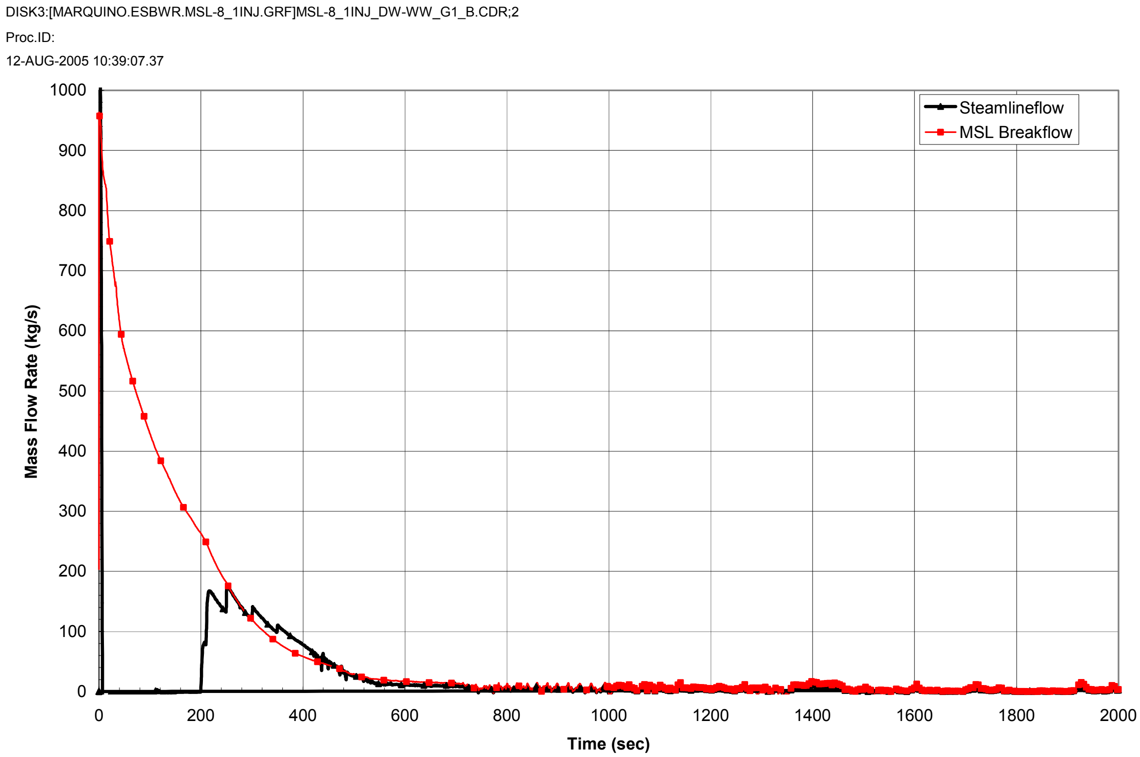


Figure 6.3-19. Steam Line and Break Flow, Inside Steam Line Break, 1 GDCS Valve Failure

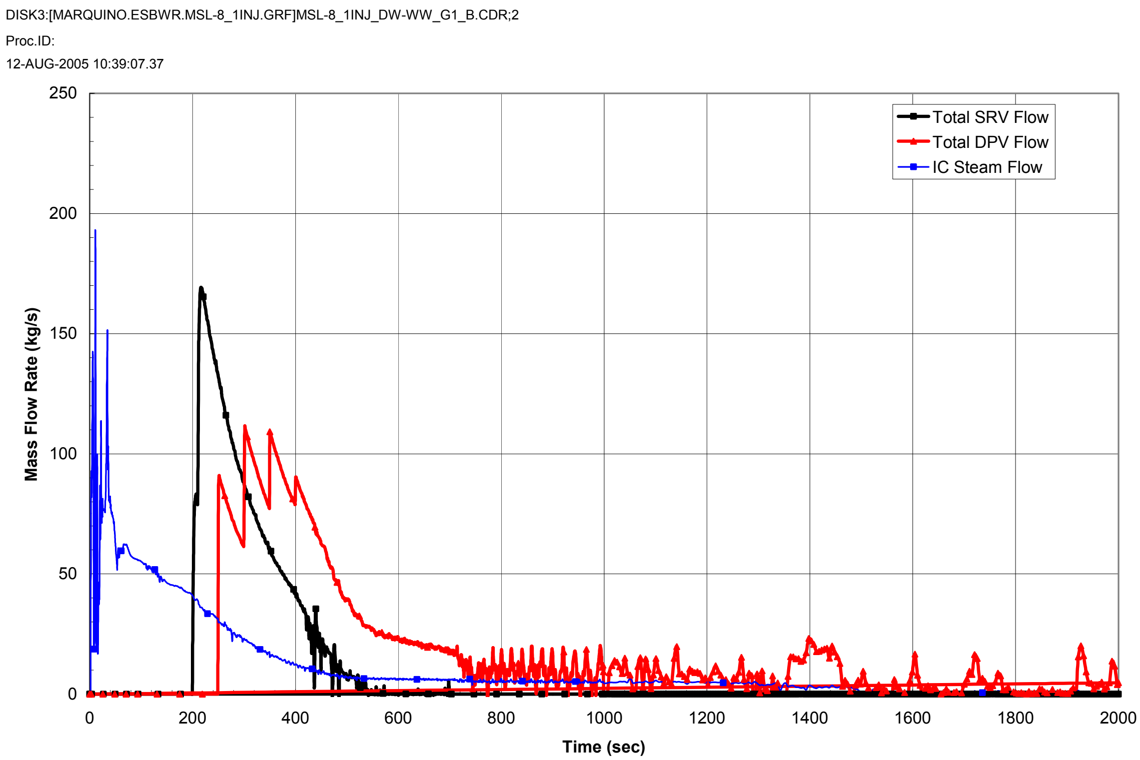


Figure 6.3-20. ADS Flow, Inside Steam Line Break, 1 GDCS Valve Failure

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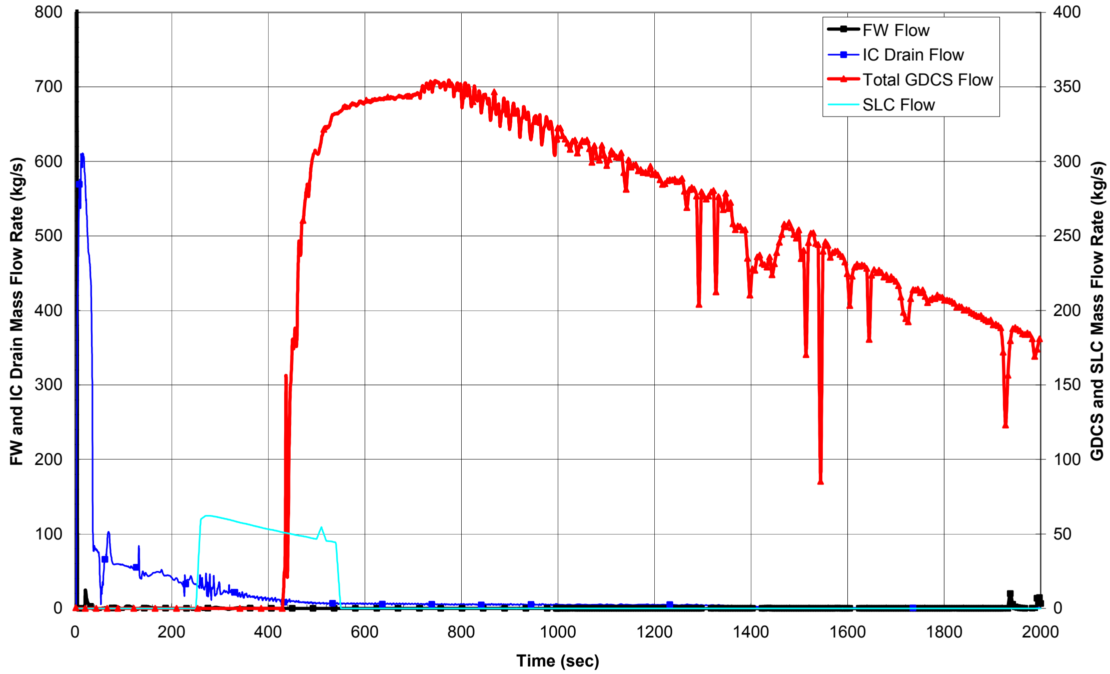

Figure 6.3-21. Flows Into Vessel, Inside Steam Line Break, 1 GDCS Valve Failure

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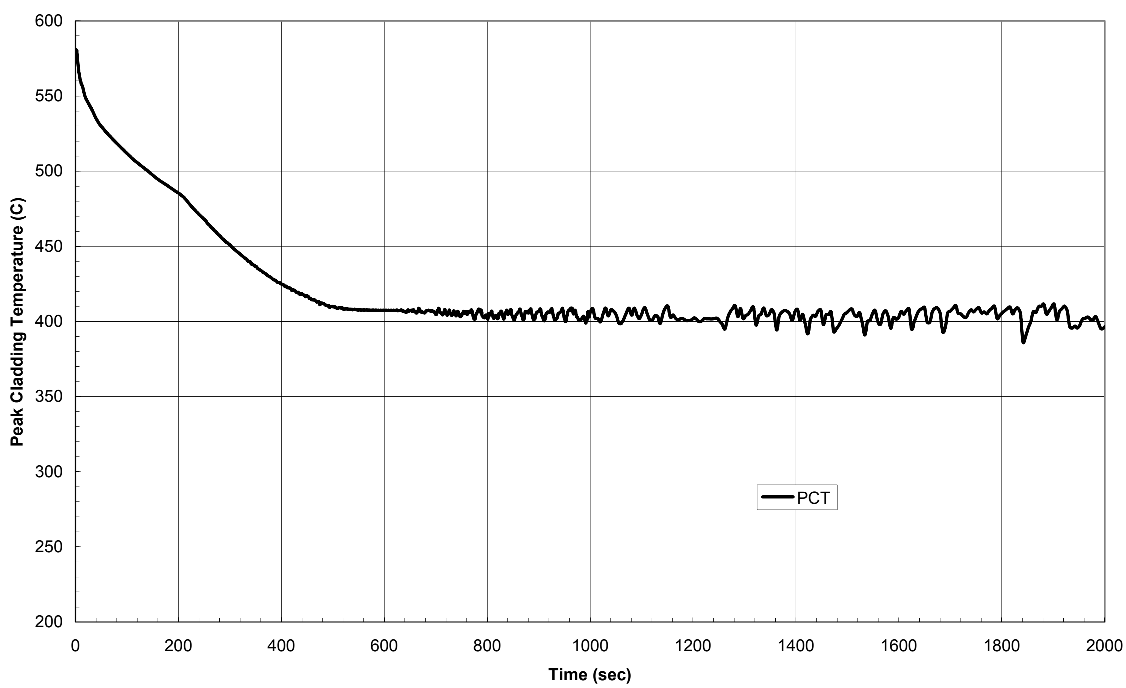

Figure 6.3-22. PCT, Inside Steam Line Break, 1 GDCS Valve Failure

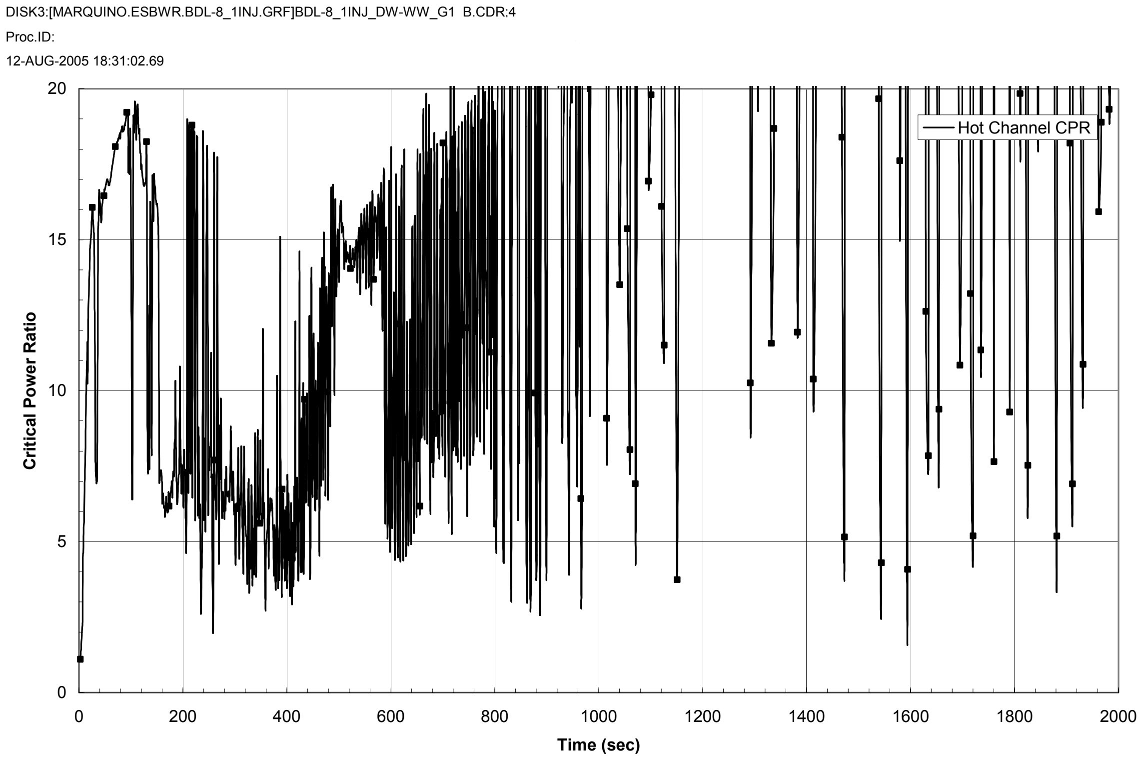

Figure 6.3-23. MCPR, Bottom Drain Line Break, 1 GDCS Valve Failure

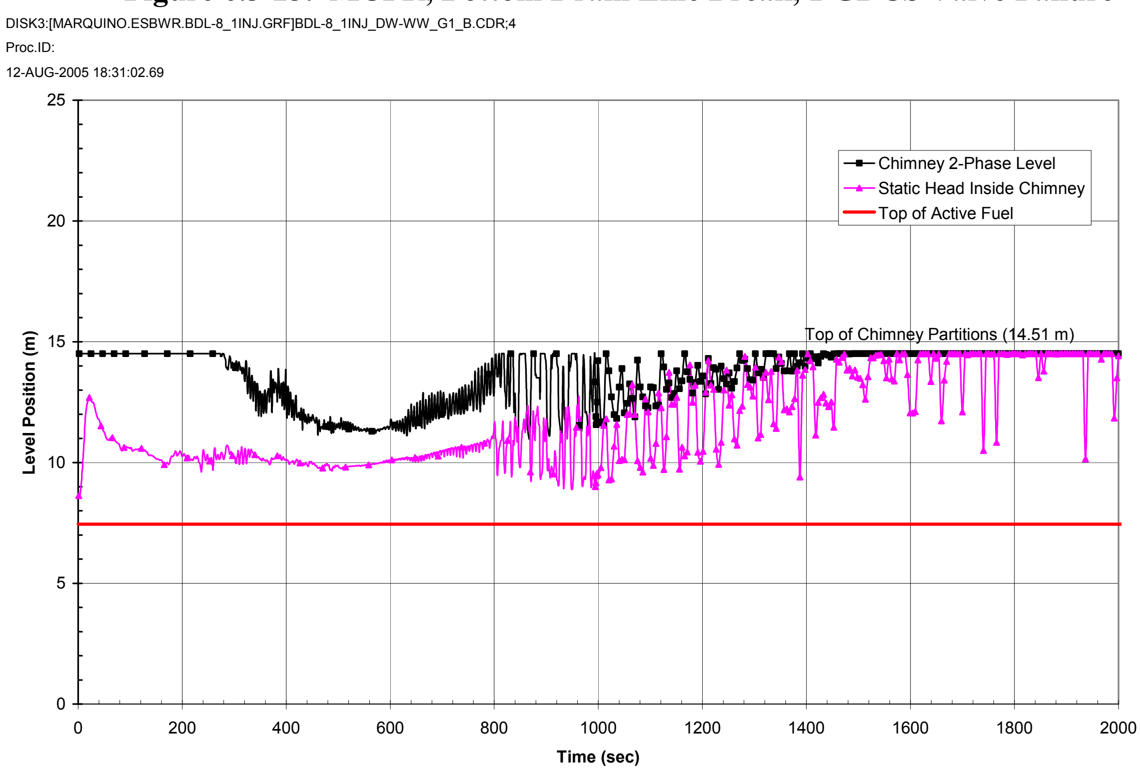

Figure 6.3-24. Chimney Water Level, Bottom Drain Line Break, 1 GDCS Valve Failure

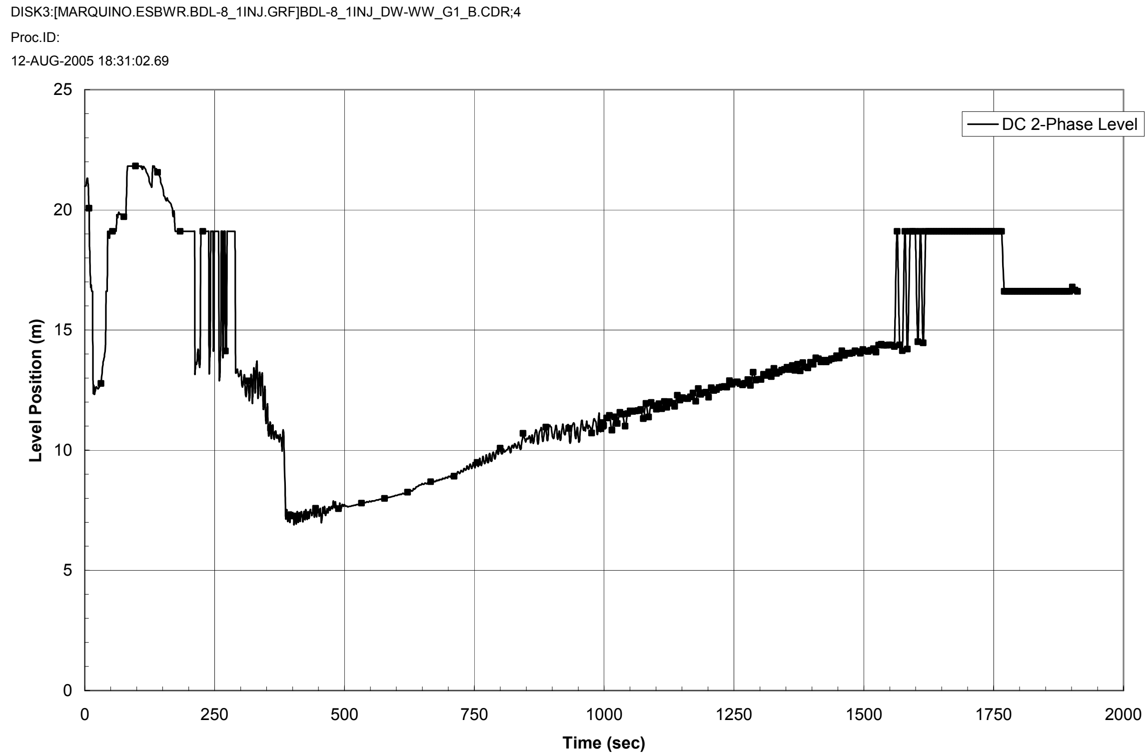

Figure 6.3-25. Downcomer Water Level, Bottom Drain Line Break, 1 GDCS Valve Failure

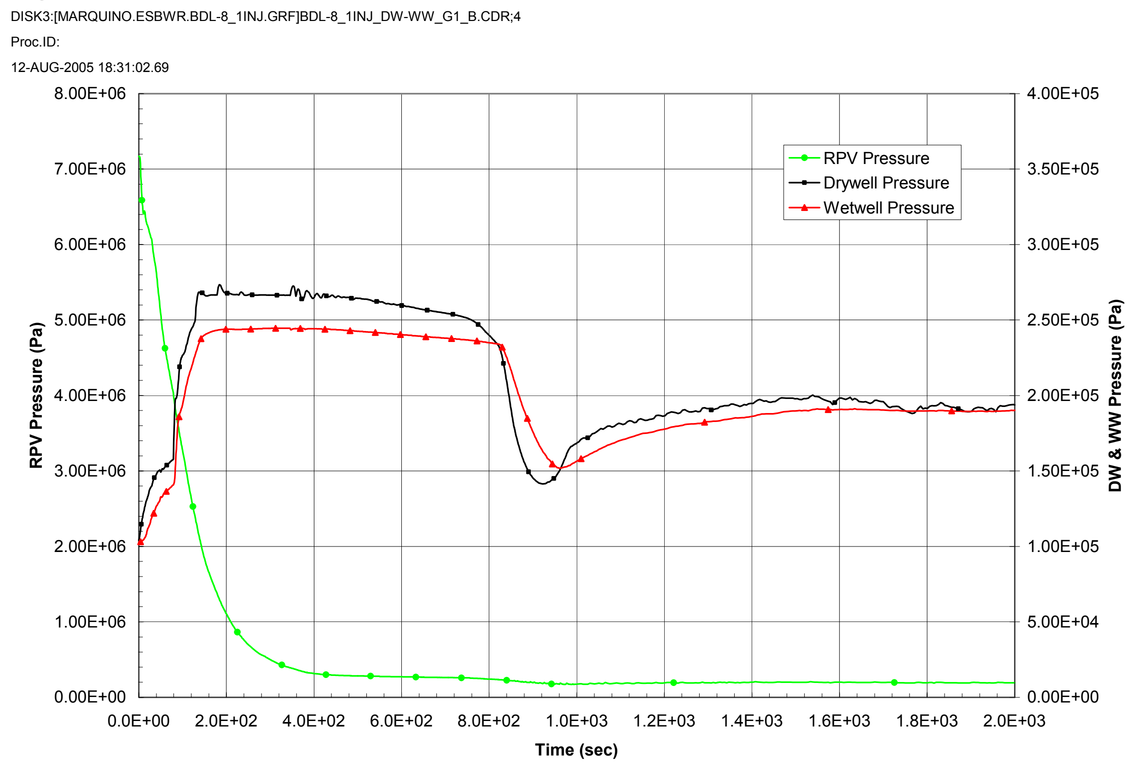

Figure 6.3-26. System Pressures, Bottom Drain Line Break, 1 GDCS Valve Failure

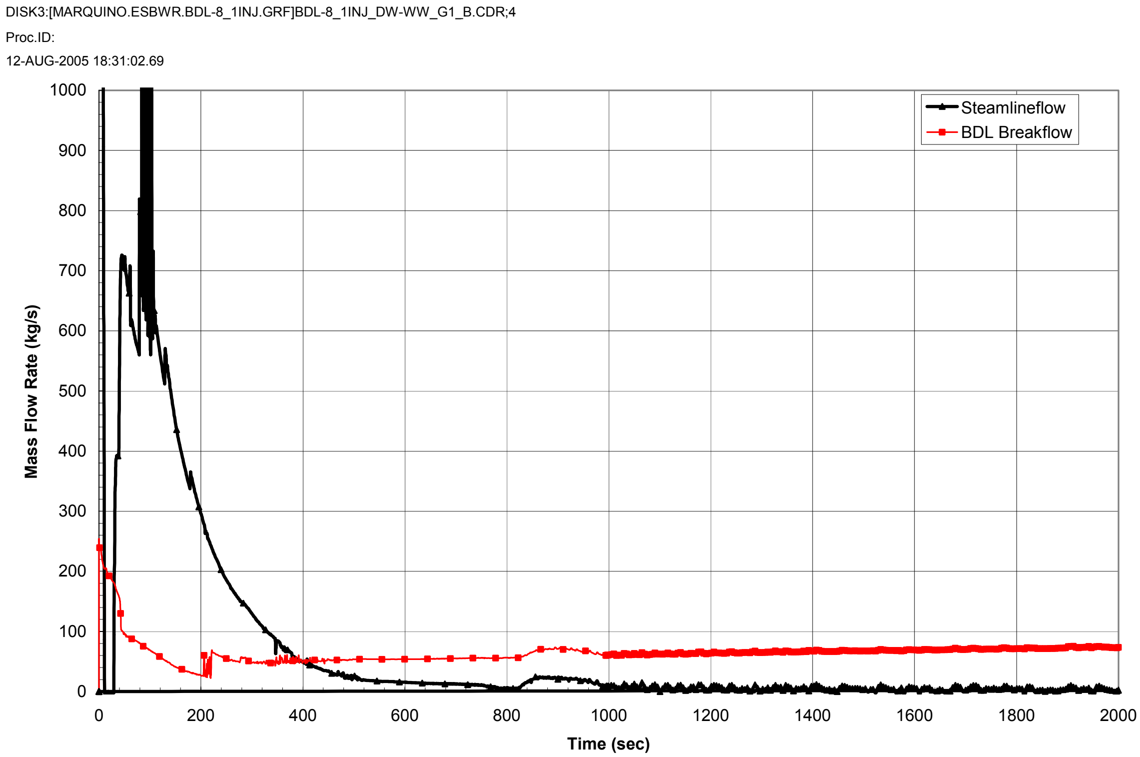

Figure 6.3-27. Steam Line and Break Flow, Bottom Drain Line Break, 1 GDCS Valve Failure

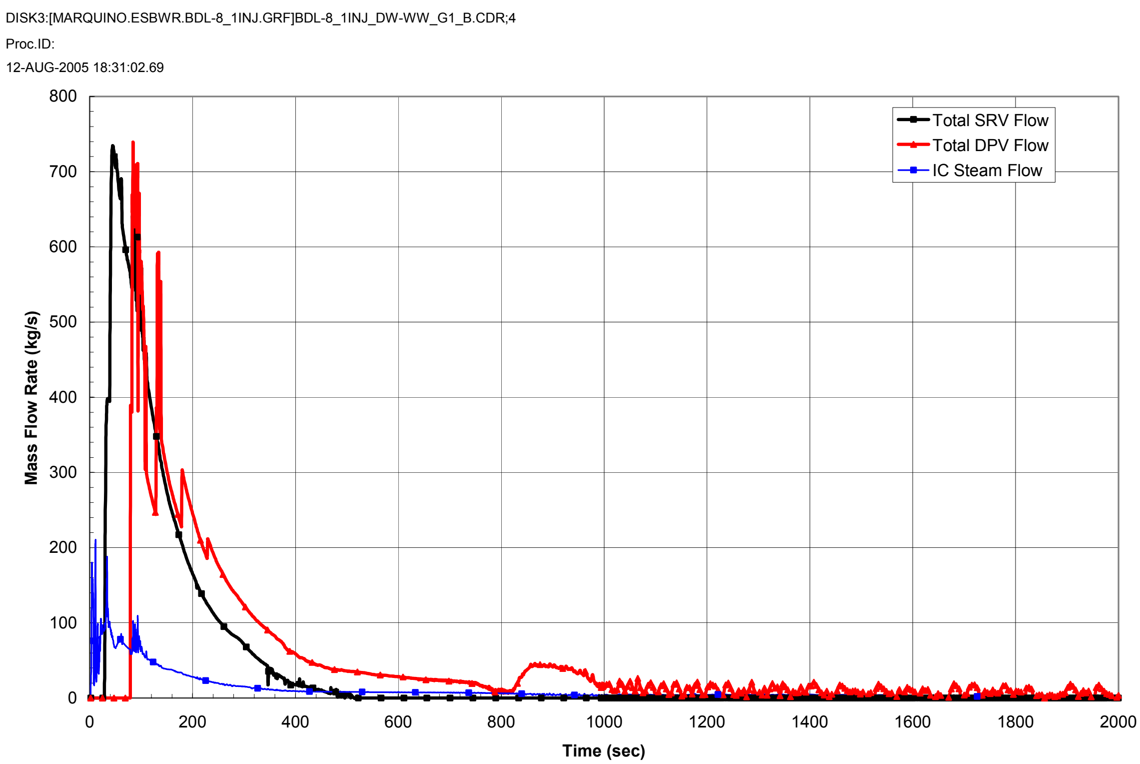

Figure 6.3-28. ADS Flow, Bottom Drain Line Break, 1 GDCS Valve Failure

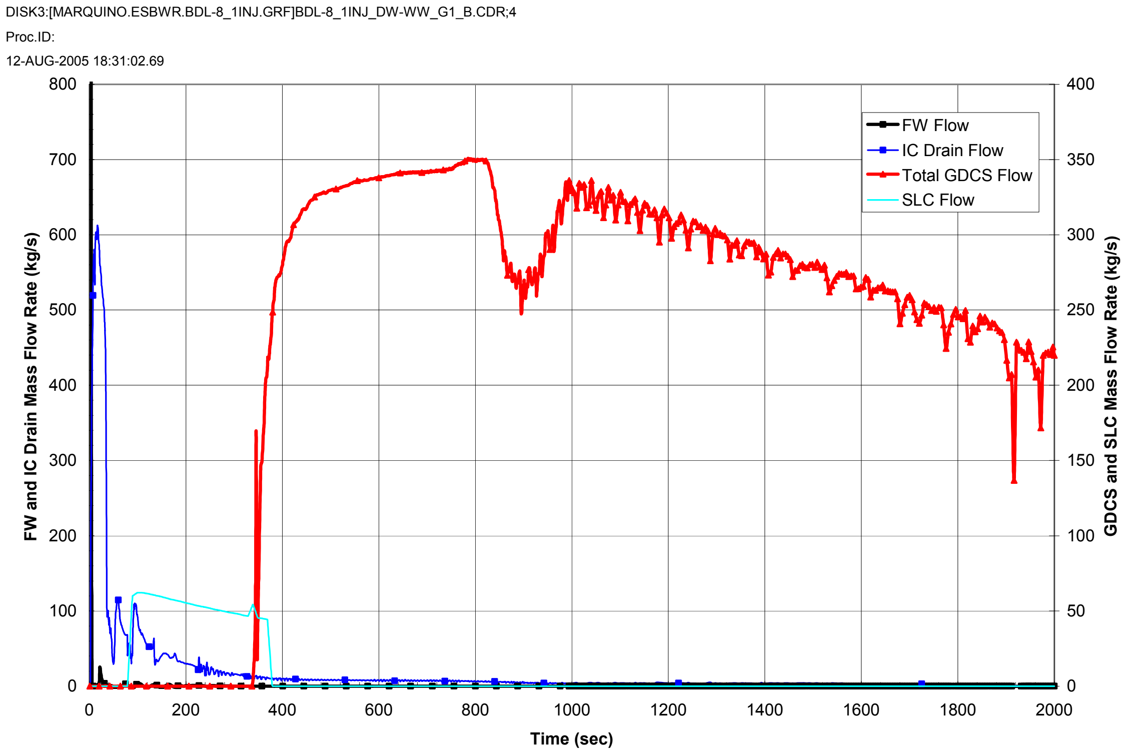

Figure 6.3-29. Flows Into Vessel, Bottom Drain Line Break, 1 GDCS Valve Failure

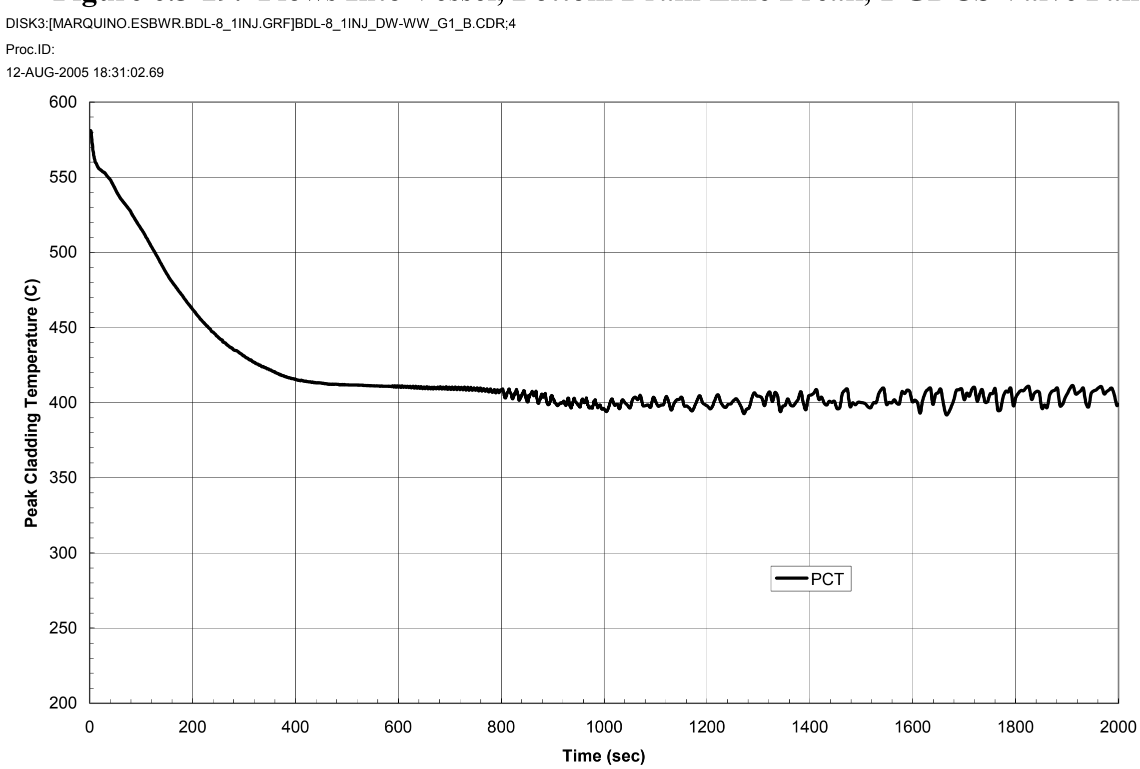

Figure 6.3-30. PCT, Bottom Drain Line Break, 1 GDCS Valve Failure

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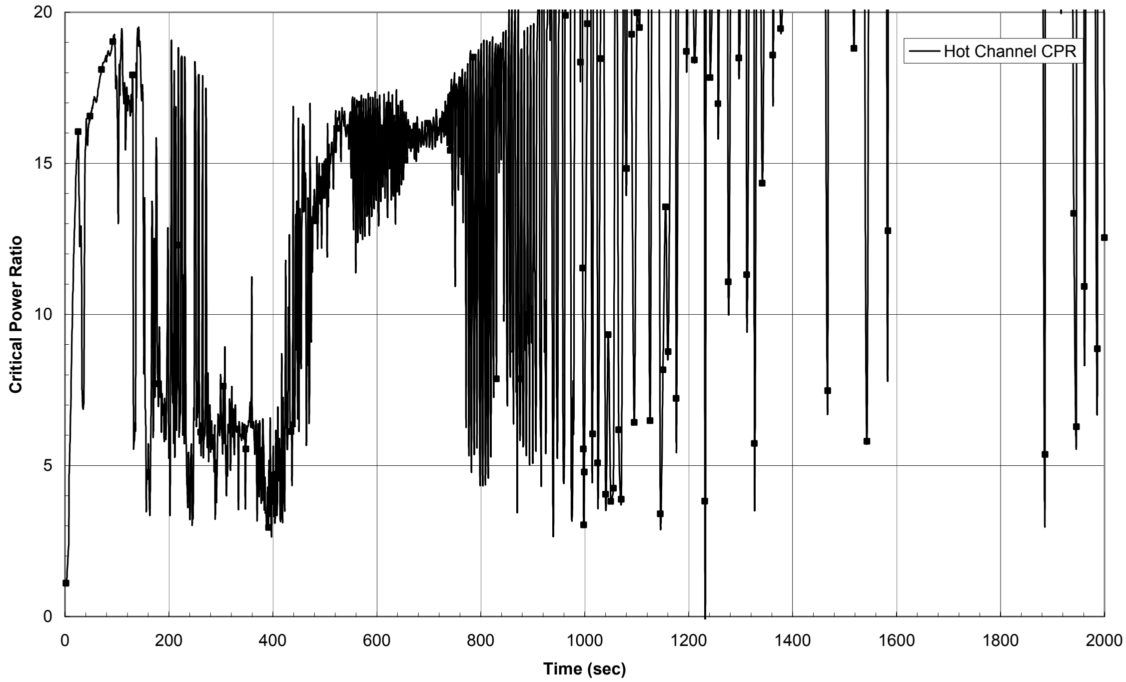

Figure 6.3-31. MCPR, GDCS Injection Line Break, 1 GDCS Valve Failure

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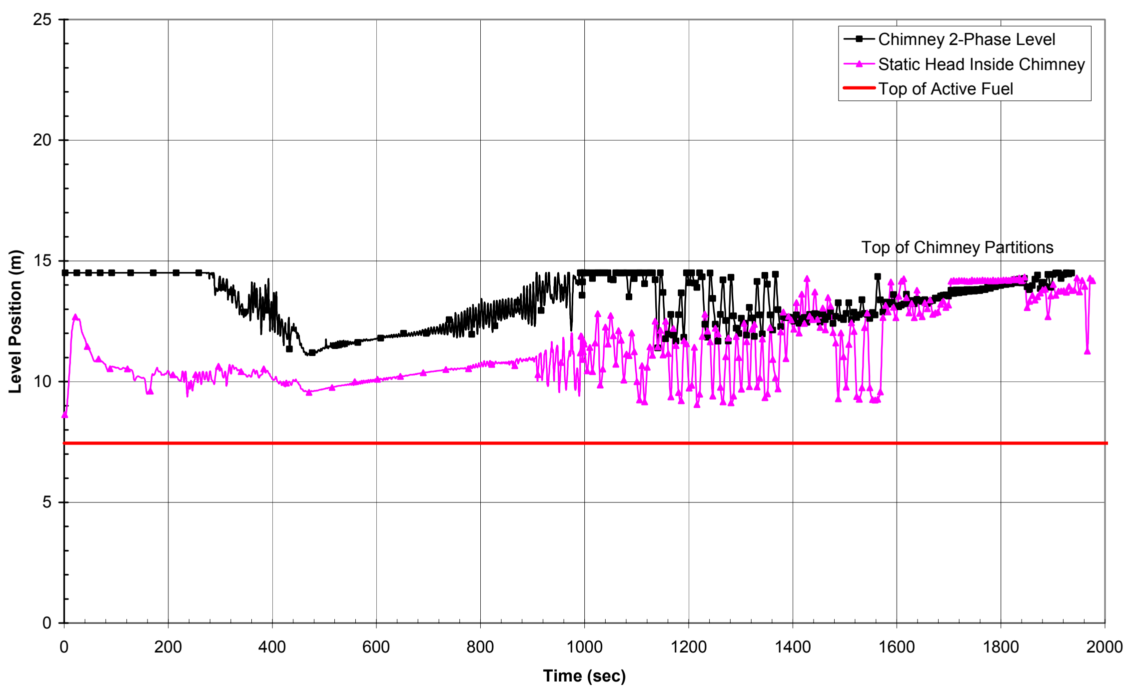

Figure 6.3-32. Chimney Water Level, GDCS Injection Line Break, 1 GDCS Valve Failure

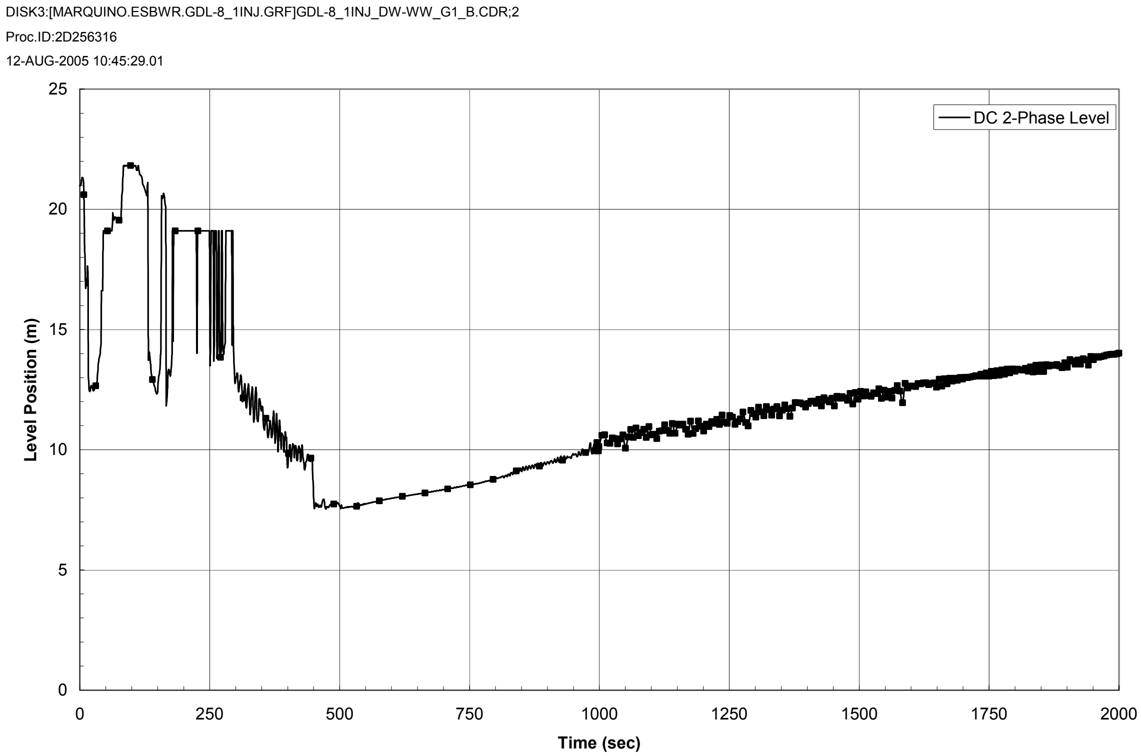

Figure 6.3-33. Downcomer Water Level, GDCS Injection Line Break, 1 GDCS Valve Failure

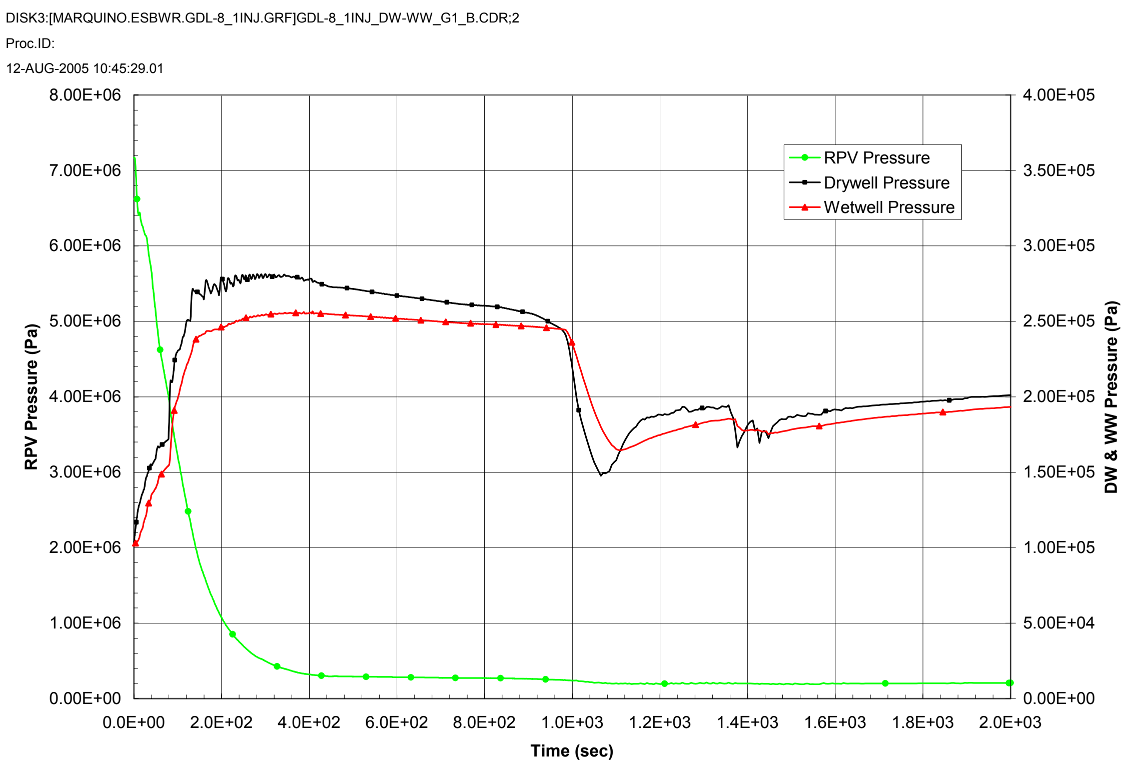

Figure 6.3-34. System Pressures, GDCS Injection Line Break, 1 GDCS Valve Failure

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12-AUG-2005 10:45:29.01

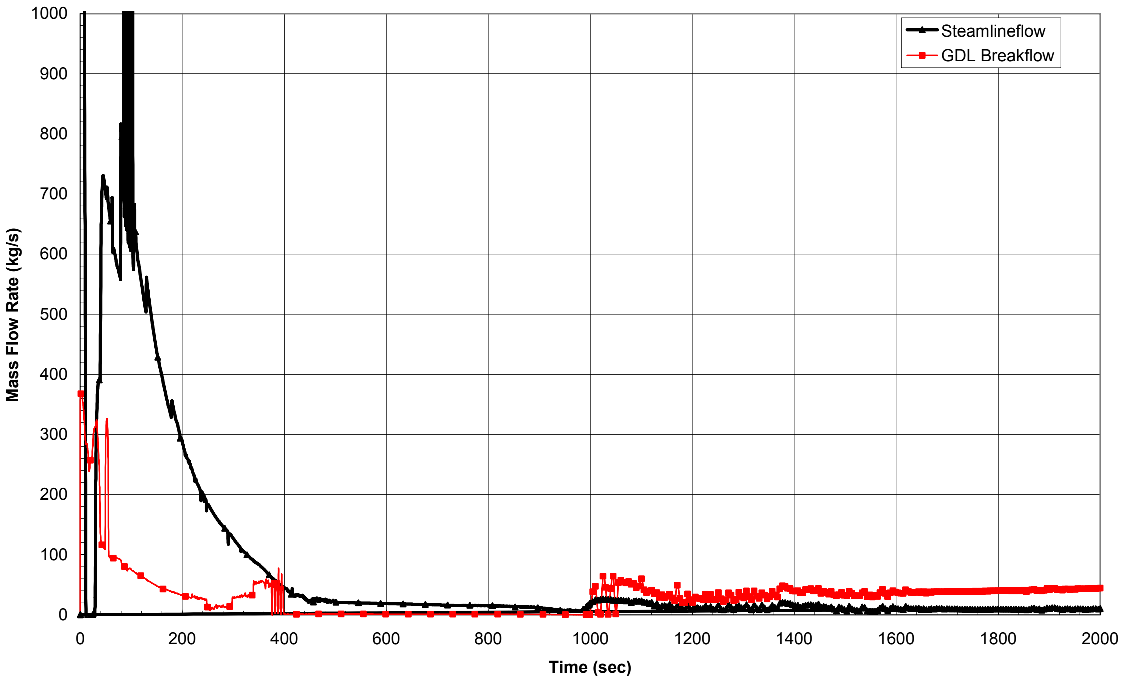

Figure 6.3-35. Steam Line and Break Flow, GDCS Injection Line Break, 1 GDCS Valve Failure

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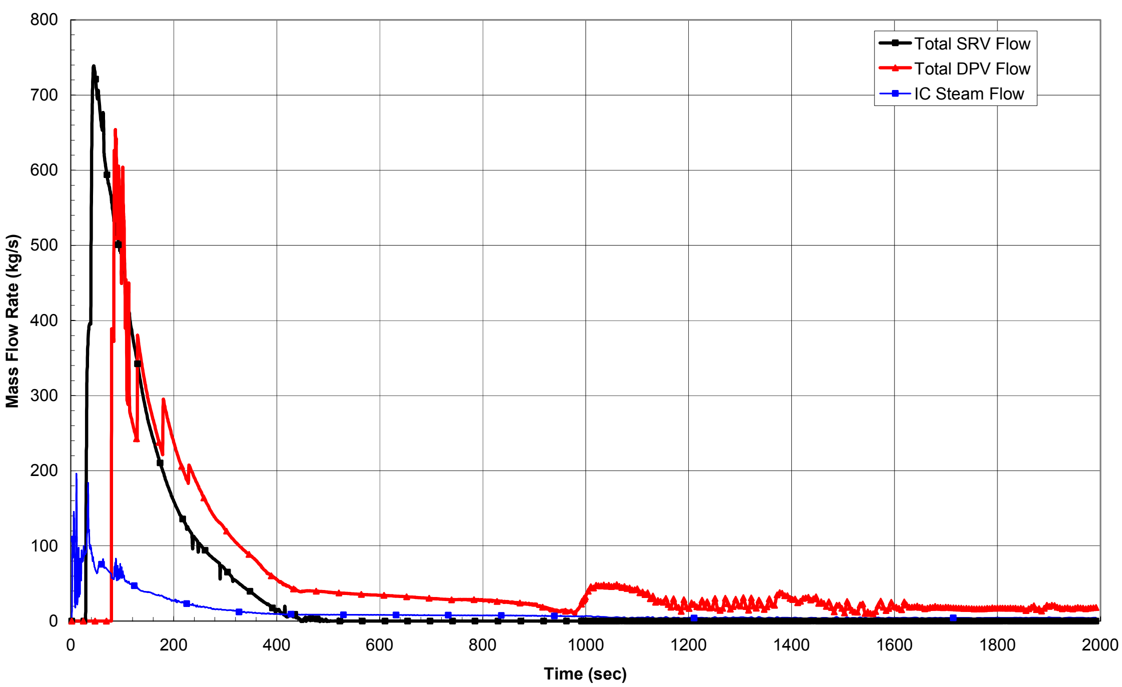

Figure 6.3-36. ADS Flow, GDCS Injection Line Break, 1 GDCS Valve Failure

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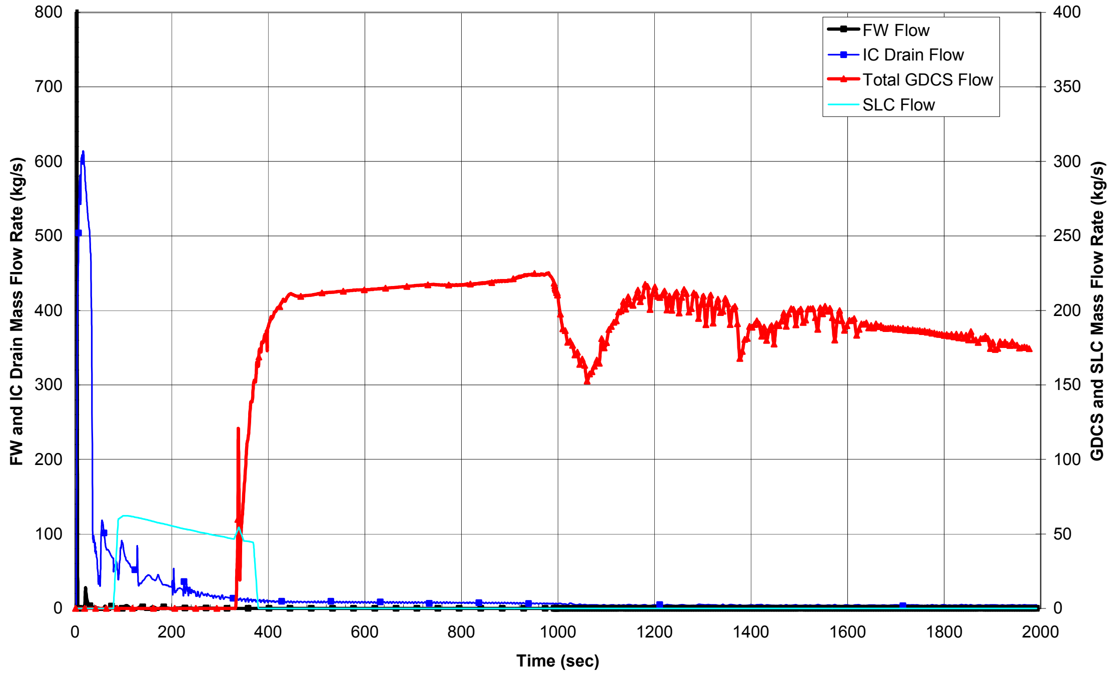

Figure 6.3-37. Flows Into Vessel, GDCS Injection Line Break, 1 GDCS Valve Failure

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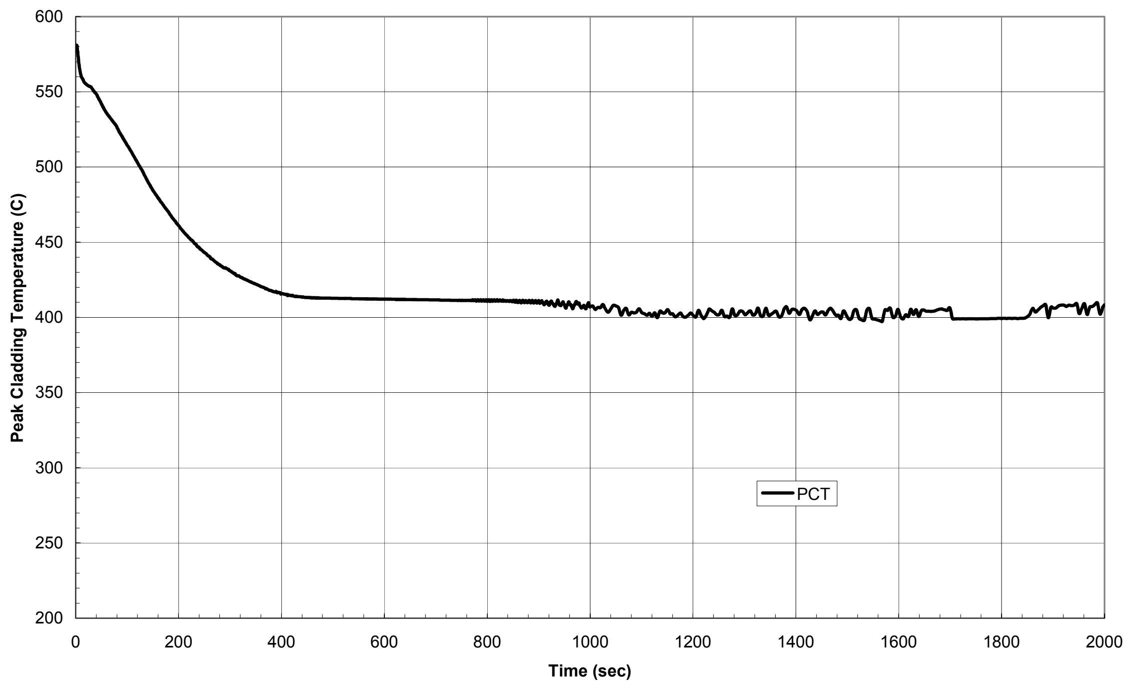

Figure 6.3-38. PCT, GDCS Injection Line Break, 1 GDCS Valve Failure

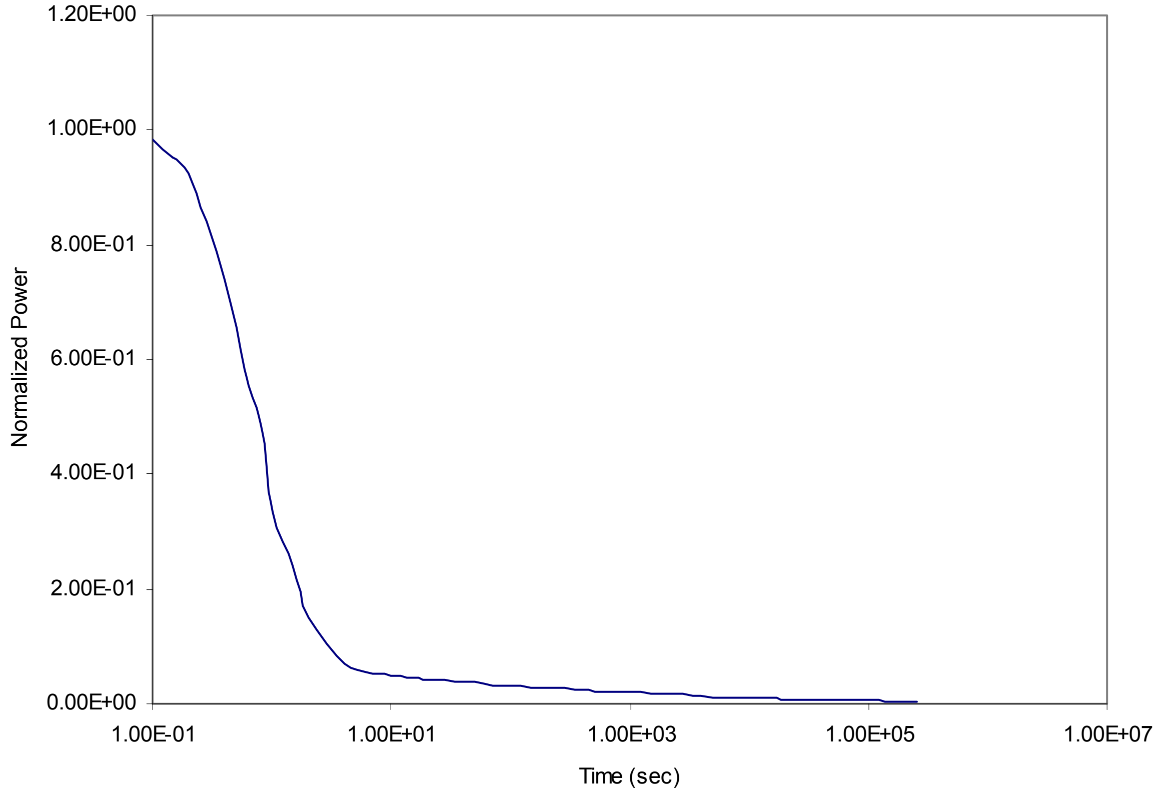

Figure 6.3-39. Normalized Shutdown Power

6.4 CONTROL ROOM HABITABILITY SYSTEMS

ESBWR design features are provided to ensure that the control room operators can remain in the control room and take actions to safely operate the plant under normal conditions and to maintain it in a safe condition under accident conditions.

These habitability features include missile protection, radiation shielding, radiation monitoring, air filtration and ventilation systems, lighting, personnel and administrative support, and fire protection.

The design bases and descriptions of the various habitability features are contained in the following sections:

Conformance with NRC General Design Criteria	Section 3.1
Wind and Tornado Loading	Section 3.3
Water Level (Flood) Design	Section 3.4
Missile Protection	Section 3.5
Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping	Section 3.6
Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	Section 3.10
Environmental Qualification of Mechanical and Electrical Equipment	Section 3.11
Radiation Protection	Section 12.3
Control Room Area Ventilation System	Subsection 9.4.1
Fire Protection System	Subsection 9.5.1
Lighting System	Subsection 9.5.3
Electrical Power	Chapter 8
Leak Detection and Isolation System	Subsection 7.3.3
Process Radiation Monitoring System	Subsection 7.5.3
Area Radiation Monitoring System	Subsection 7.5.4
Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	Subsection 11.5

Equipment and systems are discussed in this section only as necessary to describe their connection with control room habitability. References to other sections are made where appropriate.

The term "Control Room" includes the plant area in which actions can be taken to operate the plant safely under normal conditions and to maintain the reactor in a safe condition during accident situations. It includes the Main Control Room (MCR) area and areas adjacent to the Main Control Room containing operator facilities.

The CRHA contains the following features:

- main control consoles and associated equipment;

- shielding and area radiation monitoring;
- provisions for emergency food, water, storage and air supply systems;
- kitchen and sanitary facilities; and
- provision for protection from airborne radioactive contaminants.

Relevant to ESBWR control room habitability systems, this subsection addresses or refers to other DCD locations that address the applicable requirements of General Design Criteria (GDC) 4, 5 and 19 discussed in Standard Review Plan (SRP) 6.4 Draft R3. See Subsection 9.4.1 for additional description of how GDC 4, 5, 19 and other habitability requirements are met.

The ESBWR:

- Meets GDC 4, as it relates to accommodating the effects of and being compatible with postulated accidents, including the effects of the release of toxic gases.
- Meets the intent of GDC 5, because each ESBWR unit at a multi-unit site has a separate control room for each unit. Thus the ability to perform safety functions including an orderly shutdown and cool down of any remaining unit(s) is not impaired.
- Meets GDC 19, as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation.

6.4.1 Design Bases

Criteria for the selection of design bases are found within Section 1.2.

The CRHA is contained inside a Seismic Category I structure (the Control Building) and is protected from wind and tornado effects as discussed in Section 3.3, from external floods and internal flooding as discussed in Section 3.4, from external and internal missiles as discussed in Section 3.5, and from the dynamic effects associated with the postulated rupture of piping as discussed in Section 3.6. The seismic qualification of electrical and mechanical components as discussed in Section 3.10 and environmental design is discussed in Section 3.11. Radiation exposure to control room personnel during postulated accidents is described in Chapter 15.

6.4.1.1 Safety Design Basis

- The CRHA includes all instrumentation and controls necessary for safe shutdown of the plant and is limited to those areas requiring operator access during and after a design basis accident (DBA).
- The Control Room Habitability Area Heating, Ventilation, and Air Conditioning System (CRHAHVS) is a sub-system of the Control Building HVAC system (CB HVAC). CRHAHVS instrumentation detects and automatically protects the CRHA habitability upon detection of high airborne radioactivity, toxic gases, or smoke.
- MCR habitability requirements are satisfied without the need for individual breathing apparatus and/or special protective clothing.
- The CRHA Emergency Breathing Air System (EBAS), the CRHA envelope structures, doors, components and the CRHA isolation dampers including supporting ductwork and associated controls are safety-related and Seismic Category I.

- Nonsafety-related pipe, ductwork, or other components located in the control room are designed as necessary to ensure that they do not adversely affect safety-related components or the plant operators during an SSE.
- The EBAS is designed with sufficient redundancy to ensure operation under emergency conditions.
- The EBAS is operable during loss of preferred power, loss of onsite AC power or Station blackout (SBO).
- The EBAS operates during an emergency to ensure the safety of the control room operators and the integrity of the control room by maintaining a minimum positive differential pressure inside the CRHA as noted in Table 6.4-1.
- The CRHA envelope is sufficiently leak tight to maintain positive differential pressure with the EBAS in operation.

6.4.1.2 Power Generation Design Bases

- The CRHAHVS is designed to provide a controlled environment for personnel comfort and for the proper operation and integrity of equipment when AC power is available.
- Provisions for periodic inspection, testing and maintenance of the principal components of both the EBAS and the CRHAHVS are incorporated in the design.

6.4.2 System Design

Figure 6.4-1 provides a schematic diagram of the EBAS. The CRHAHVS description is provided in Subsection 9.4.1.

The EBAS is a redundant safety-related system that supplies stored, compressed air to the CRHA for breathing and for pressurization to minimize inleakage. The EBAS is automatically initiated. There are multiple trains of EBAS to provide the required redundancy. Each train of EBAS consists of compressed breathing air tanks and associated piping and components as shown on Figure 6.4-1. EBAS has been sized to provide sufficient breathing quality air to maintain a positive pressure in the CRHA for a minimum of 72 hours.

6.4.3 Control Room Habitability Area

The CRHA envelope includes the following areas:

- Main Control Room
- Shift Supervisor Office
- Shift Supervisor Conference Room
- Operator's Area
- Shift Technical Advisor Office
- Main control Room Storage Room
- Electrical Panel Board Room

- Restroom A
- Restroom B

These areas constitute the operation control area, which can be isolated and remain habitable for 72 hours if such is required by the existence of a LOCA or high radiation condition. Also potential sources of danger such as steam lines, pressurized piping, pressure vessels, CO_2 fire fighting containers, etc. are located outside of the CRHA.

Radiation Protection

Description of control room instrumentation for monitoring of radioactivity is given in Sections 11.5 and 12.3.

Shielding Design

The control room shielding design is based upon protecting personnel from radiation resulting from a design basis LOCA. The radioactive sources existing under normal operating conditions are not determining factors for the shielding design.

Fire Protection

A description of the smoke detectors is in Subsection 9.5.1. Smoke removal is described in Section 6.4.4 and Section 9.4.1.

Layout

The layout of the CRHA, which includes the MCR, is shown within Figure 9A.2-3.

Release Points

Radiological release parameters are described in Section 15.4

Leak Tightness

The CRHA boundary envelope structures are designed with low leakage construction. The access doors are designed with self-closing devices, which close and latch the doors automatically following use. There are double door air locks for access and egress during emergencies.

Interaction With Other Zones and Pressure-Containing Equipment

During normal operation the CRHA is heated, cooled, ventilated, and pressurized by a recirculating air system using filtered outdoor air for ventilation and pressurization purposes. See Subsection 9.4.1 for a complete description of the CRHAHVS.

In a radiological event concurrent with SBO the EBAS maintains a positive pressure in the CRHA to minimize infiltration of airborne contamination. Interlocked double-vestibule type doors maintain the positive pressure, thereby minimizing infiltration when a door is opened.

The CRHA remains habitable during emergency conditions. To make this possible, potential sources of danger such as steam lines, pressure vessels, CO_2 fire fighting containers, etc. are located outside of the CRHA.

6.4.4 System Operation Procedures

The CRHAHVS operates during all modes of normal power plant operation, including startup and shutdown. For a detailed description of the CRHAHVS operation see Subsection 9.4.1. It is not required to operate during an SBO except for the EBAS. During an SBO concurrent with a radiological event the EBAS operates for 72 hours. Also, the MCR air handling units/condensing units (AHUs/CDUs) operates for the first two hours of an SBO; otherwise certain nonsafety-related loads are automatically de-energized. The CRHA isolation dampers fail closed on a loss of AC power or instrument air. Rooms containing safety-related equipment have passive cooling features. These features limit the temperature rise to temperature limits listed in Table 6.4-1 for the first 72 hours of an SBO.

6.4.5 Design Evaluations

Radiological protection assumptions used in the generation of post-LOCA radiation source terms are described fully in Section 15.

Smoke and toxic gas protection are discussed in Subsection 9.4.1 and evaluated in Subsection 9.5.1. The use of noncombustible construction and heat- and flame-resistant materials wherever possible throughout the plant minimizes the likelihood of fire and consequential fouling of the control room atmosphere with smoke or noxious vapor introduced into the control room air. In the smoke removal mode, the purge flow through the control room provides 100% outside airflow.

Conditions such as toxic gas or high radiation causes automatic changeover to the operating modes described in Section 6.4.4 and in Section 9.4.1.2. EBAS automatically starts to provide CRHA breathing air and pressurization during SBO concurrent with a radiological event. Local, audible alarms warn the operators to shut the self-closing doors, if for some reason they are open.

Redundant EBAS components are provided to ensure CRHA pressurization upon a radiological event concurrent with SBO.

The EBAS is designed in accordance with Seismic Category I requirements. The failure of components (and supporting structures) of any system, equipment or structure, which is not Seismic Category I, does not result in loss of a required function of the EBAS.

Potential site-specific toxic or hazardous materials that may affect control room habitability will be defined by the COL applicant as stated in Subsection 6.4.9.

6.4.6 Life Support

In addition to the supply and recirculation of vital air, food, water and sanitary facilities are provided.

6.4.7 Testing and Inspection

Routine testing of components of the CBHVS are conducted in accordance with routine power plant requirements for demonstrating system and component operability and integrity.

Periodic surveillance testing of safety-related CRHA isolation dampers and the EBAS components are carried out per IEEE-338. Safety-related CRHA isolation dampers and the EBAS are operational during plant normal and abnormal operating modes. The EBAS pressure-retaining components (tanks, piping and valves) are designed to meet the in-service inspection (ISI) requirements of ASME Section XI.

The CRHAHVS filtration components are periodically tested in accordance with ANSI/ASME N509, Nuclear Power Plant Air Cleaning Units and Components, and ANSI/ASME N510, Testing of Nuclear Air Cleaning Systems. HEPA filters are tested periodically with dioctyl phthalate (DOP), and the charcoal filters are periodically tested for bypasses.

Testing to demonstrate the integrity of the CHRA envelope is in accordance with Regulatory Guide 1.197 and ASTM E741.

6.4.8 Instrumentation Requirements

A description of the required instrumentation is given in Subsection 9.4.1.5. Alarms for the following CRHA/CRHAHVS conditions are provided in the MCR:

- Low airflow (each fan and air handling unit)
- High filter pressure drop
- High space temperatures
- Low space temperatures
- Low coil entering air temperature
- Low CRHA differential pressure
- Smoke detected
- Toxic gas detected
- High and low humidity in the CRHA
- CRHA airlock doors are open during an SBO

6.4.9 COL Information

The COL applicant is responsible for defining site adequacy in regard to neighboring toxic or hazardous materials shipping, handling, or storage (Subsection 6.4.5).

6.4.10 References

None.

Table 6.4-1

Design Parameters for CRHAHVS and EBAS

Operation periods:	Normal plant operation, plant startup, and plant shutdown
Outside Air Design Conditions:	
For CRHAHVS and EBAS (0% Exceedance values)	Summer: 46.1°C (115°F) Dry Bulb 26.7°C (80°F) Wet Bulb
	Winter: -40.0°C (-40°F) Dry bulb
Inside Design temperatures and humidity:	
CRHA (normal operation)	22.8°C (73°F) to 25.6°C (78°F) and 25% to 60% relative humidity (RH)
CRHA (SBO)	Maximum 8.3°C (15°F) rise above normal operating temperature for the first 72 hours into the event, RH not controlled
Pressurization:	> atmospheric pressure
EBAS	
Operation period:	Emergency mode
Breathing air supply capacity:	9.5 l/s/person for 5 persons (47.5 l/s total) for 72 hours
Pressurization capability:	31 Pa positive differential

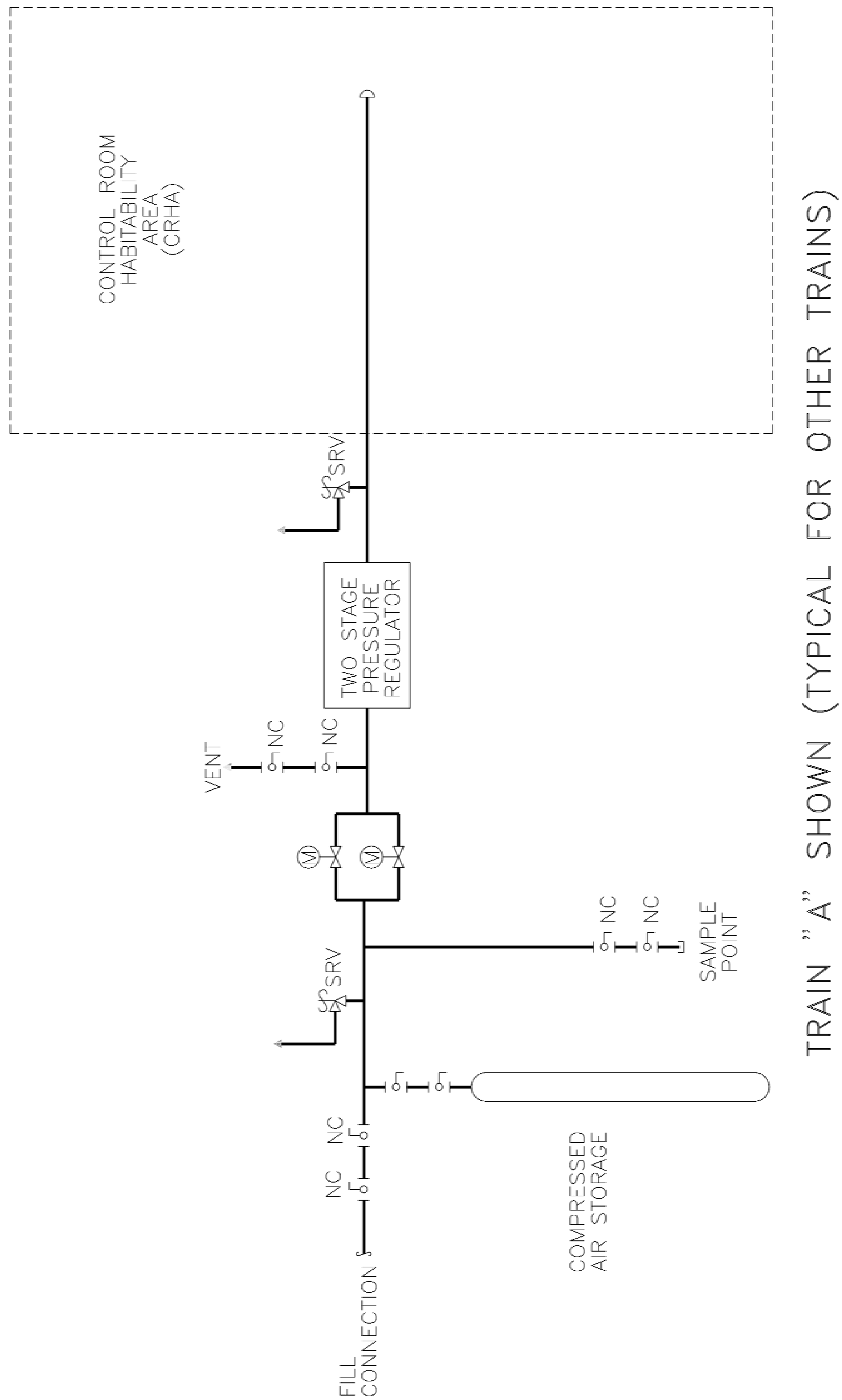

Figure 6.4-1. Emergency Breathing Air System Schematic Diagram

6.5 ATMOSPHERE CLEANUP SYSTEMS

The ESBWR does not need, and thus has no filter system that performs a safety-related function following a design basis accident. The control room is provided with self contained bottled air to maintain a safe control room atmosphere following a design basis accident as discussed in Section 6.4. Therefore, the acceptance criteria in Standard Review Plan 6.5.1 are not applicable to the ESBWR.

6.5.1 Containment Spray Systems

The ESBWR contains drywell containment sprays, which can be initiated manually 72 hours after a LOCA to cooldown the containment to aid in post-accident recovery or to mitigate the effects of a beyond design basis severe accident. The spray system is nonsafety-related, and no credit is taken for removal of fission products in design basis accident evaluations due to this spray system. Therefore, the acceptance criteria in Standard Review Plan 6.5.2 are not applicable to the ESBWR.

6.5.2 Fission Product Control Systems and Structures

The ESBWR is provided with a number of fission product control systems to contain and mitigate potential releases of radionuclides to the plant areas and the environment. These systems are described below by functional area and in terms of the conditions under which each system may be applied. In addition, the systems are classified into "active" systems that employ chemical or physical trapping and treatment, or "passive" ESF systems which remove or hold-up (e.g., radioactive decay) radionuclides through natural processes.

6.5.2.1 General

The ESBWR is functionally divided into three distinct buildings, the Reactor Building that contains the bulk of the radioactive inventory (see Section 12.2), the radwaste building for processing of liquid and solid radioactive waste streams (Chapter 11), and the turbine building which contains one liquid treatment system and the offgas gaseous waste treatment system (Chapter 11). The Reactor Building is further divided into three distinct areas, the containment and clean and controlled radiological areas. Each area is described below along with its function in controlling potential releases.

6.5.2.2 Containment

The containment is a stepped cylindrical steel-lined reinforced concrete structure. This structure is designed to be periodically tested to meet specific criteria for leak tightness under designed pressure and temperature conditions (see Subsection 6.2.6). The primary ESF function of this structure is to provide a passive fission product barrier for events, where core fission products are released to the containment air space. The containment design, which is described in Subsection 6.2.1, is subdivided into a suppression chamber and drywell, with the drywell being further divided into upper and lower drywell regions. The lower portion of the suppression chamber region is filled with water, or a suppression pool. The overall design of the containment channels steam releases from a break in any location in the drywell from the drywell airspace through a series of downcomers, which exhaust into the suppression pool water. During such an event, releases would then be subject to suppression pool scrubbing. (See Subsection 6.5.5 for a discussion of suppression pool scrubbing.) The ESBWR containment with the suppression pool, GDCS pool, and PCCS heat exchangers serves to (a) remove decay heat from the reactor core, (b) suppress and remove steam release from the vessel into the pools or to the drywell itself, and (c) provide passive removal pathways for the mitigation of potential fission product releases by means of hold up, plate out and physical (water) removal processes.

Structural design requirements for the containment are described in Section 3.8. During power operations, the containment is nitrogen inerted. Under design basis conditions the containment is isolated as described in Subsection 6.2.4 and hydrogen releases from metal water reactions are controlled as described in Subsection 6.2.5.

6.5.2.3 Reactor Building

The Reactor Building completely surrounds the containment and is divided into clean and contaminated radiological zones (see Section 12.3). Under normal conditions, air flow is maintained from clean to contaminated areas and then routed via the Reactor Building HVAC system to the plant stack. To accomplish this air flow routing, over pressurization is maintained in the central corridor on each floor with the flow then directed into individual equipment cubicles to the inside of each area and then via penetrations to the Reactor Building HVAC system stack. Under high radiation conditions, the air flow is rerouted to the HEPA filter train or shut down to provide a hold up and plate out volume. Under high energy release conditions such as a high energy line break, the overpressure is routed to the Reactor Building operating floor in which blow out panels in the upper sections relieve the overpressure to the environment. The Reactor Building HVAC system performs no ESF/safety-related function, but credit is taken for hold up and plate out in the Reactor Building because the building is sealed during isolation and if AC power is available, internal recirculation is active.

The controlled area of the Reactor Building surrounds most of the containment and provides a barrier for airborne leakage of fission products resulting from containment leakage including containment penetrations. Toward this end, most penetrations into the containment (with the exception of the main steam lines, the feedwater lines, the isolation condensers (ICs), and miscellaneous other penetrations) terminate in this volume. The second isolation valves on all GDC 54 lines (with the exception of the Isolation Condenser containment isolation valves) are found in this volume such that any potential valve leakage as well as penetration leakage collects in here. The Reactor Building under accident conditions is automatically isolated or passively sealed (e.g., water loop seals) to provide a hold up and plate out barrier. When isolated, the Reactor Building can be serviced by the Reactor Building HVAC system through a HEPA filtration system (Refer to Subsection 9.4.6). With low leakage and stagnant conditions, hold up and plate out mechanisms perform the basic mitigating functions.

Leakage through the containment drywell head is minimized by a pool of water, located in the reactor well, which is approximately 2.4 meters (8 feet) deep. Any drywell head leakage which exceeds the pool's hydrostatic water pressure is scrubbed by the large pool of water prior to be released to the operating floor.

Leakage from the isolation condensers is also subject to pool scrubbing in transit through the IC/PCC pool.

Leakage through the MSIVs is routed through the main steamline drain lines to the main condenser. These large volumes and surface areas are effective mechanisms to hold up and plate out the relatively low leakage flow. (See Section 15.4) The feedwater lines are large pipes that are flooded with water. The water acts as a seal to resist leakage and scrub any leakage that does occur.

The miscellaneous other penetrations that are based within the Reactor Building (e.g., RWCU/SDC, FAPCS, RCCWS, etc.) are protected from excess leakage by one of the following methods: (1) water inventories acting as seals to resist leakage and scrub entrained fission products, (2) redundant automatic isolation valves, or (3) closed loop piping systems qualified to maintain their pressure boundary function during the event.

6.5.2.4 Radwaste Building

The radwaste building is designed to contain any liquid releases by locating all high activity tanks in water-tight rooms designed to contain the maximum liquid release for that room. Airborne releases are routed by the Radwaste Building HVAC system through a HEPA filter to the Reactor Building plant stack. Under loss of power conditions, the Radwaste Building HVAC system is isolated providing hold up of potential releases. The Radwaste Building HVAC system performs no ESF/safety-related function.

6.5.2.5 Turbine Building

The turbine building contains two major process systems that remove fission products: the condensate filters and deep-bed demineralizers (with backwash tank), and the Offgas System with its charcoal adsorber beds. The activities in the filter/demineralizer system and in the Offgas System are relatively fixed, and in the event of breach of the system, would not result in a significant release of fission products to the environment. The condensate filter backwash receiving tank is located in a water-tight room which would contain any liquid release for treatment by the radwaste system. Airborne releases are routed via the Turbine Building HVAC system to the plant stack. The Turbine Building HVAC system performs no ESF/safety-related function.

6.5.3 Ice Condenser as a Fission Product Control System

The ESBWR does not use any kind of an ice condenser feature as a fission product control system.

6.5.4 Suppression Pool as a Fission Product Cleanup System

The ESBWR design incorporates isolation condensers, passive containment cooling condensers, and a suppression pool to condense steam under transients, accidents or unplanned reactor isolation conditions. In the event of an accident condition involving the direct release of fission products from the reactor core to either the reactor vessel or, the release of fission products directly to the drywell airspace, fission products blown into the suppression pool are entrained as they pass through water. This is effective in removing particulate and elemental forms of fission products. The ESBWR suppression pool is designed and complies with GDCs 41, 42, and 43 and provides water submergence and relief valve discharge quenchers similar to existing Mark III containments. The design of the ESBWR quenchers are similar (X-quenchers) to a Mark III design and the submergence depth of the downcomers and quenchers are also similar to the Mark III design. Consequently, a decontamination factor of 10 is assumed for any particulate species or elemental iodine species that traverses the pool based upon Standard Review Plan 6.5.5.

6.5.5 COL Information

None.

6.5.6 References

None.

6.6 PRESERVICE AND INSERVICE INSPECTION AND TESTING OF CLASS 2 AND 3 COMPONENTS AND PIPING

The ESBWR meets requirements for periodic inspection and testing of Class 2 and 3 systems in General Design Criteria (GDC) 36, 37, 39, 40, 42, 43, 45 and 46, as specified in part in 10 CFR Section 50.55a, and as detailed in Section XI of the ASME Code. Compliance with the preservice and inservice examinations of 10 CFR 50.55a, as detailed in Section XI of the Code, satisfies in part the requirements of GDC 36, 37, 39, 40, 42, 43, 45 and 46. ESBWR meets SRP 6.6, Revision 1 acceptance criteria by meeting the ISI requirements of these GDC and 10 CFR 50.55a for the areas of review described in Subsection I of the SRP.

This subsection describes the preservice and inservice inspection and system pressure test programs for Quality Groups B and C, i.e., ASME Code Class 2 and 3 items35, respectively, as defined in Table 3.2-3. This section describes those programs implementing the requirements of ASME B&PV Code, Section XI, Subsections IWC and IWD.

The development of the preservice and inservice inspection program plans will be the responsibility of the COL licensee, and is based on the ASME Code, Section XI, Edition and Addenda specified in accordance with 10 CFR 50.55a. The COL licensee shall specify the Edition of ASME to be used, based on the date of issuance of the construction permit or license, per 10 CFR 50.55a. The requirements presented in this section are provided for information, and are based on the 2001 Edition of ASME Section XI with 2003 Addenda of the ASME Code.

6.6.1 Class 2 and 3 System Boundaries

The Class 2 and 3 system boundaries for both preservice and inservice inspection programs and the system pressure test program include item boundaries include all or part of the following:

- Reactor Pressure Vessel (RPV) system
- Nuclear Boiler System (NBS)
- Isolation Condenser System (ICS)
- Control Rod Drive (CRD) system
- Standby Liquid Control (SLC) system
- Gravity Driven Cooling System (GDCS)
- Fuel and Auxiliary Pools Cooling System (FAPCS)
- Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system
- Reactor Component Cooling Water System (RCCWS)
- Passive Containment Cooling System (PCCS).

35 Items used herein are products constructed under a Certificate of Authorization (NCA-3120) and Material (NCA-1220). See Section III, NCA-1000, footnote 2.

6.6.1.1 Class 2 System Boundary Description

Those portions of the systems listed in Subsection 6.6.1 within the Class 2 boundary, based on Regulatory Guide 1.26, for Quality Group B (QGB) are as follows:

- Portions of the Reactor Coolant Pressure Boundary as defined within Subsection 3.2.2.1, but which are excluded from the Class 1 boundary pursuant to Subsection 3.2.2.2.
- Safety-related systems or portions of systems that are designed for reactor shutdown or residual heat removal.
- Portions of the steam system extending from the outermost containment isolation valve up to but not including the turbine stop and bypass valves and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation.
- Systems or portions of systems that are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is normally closed or capable of automatic closure.
- Safety-related systems or portions of systems that are designed for (A) emergency core cooling, (B) post accident containment heat removal, or (C) post accident fission product removal.

The above describes the Class 2 boundary only and is not related to exemptions from inservice examinations under ASME Section XI Code rules. The Class 2 components exempt from the inservice examinations are described in ASME Section XI, IWC-1220.

6.6.1.2 Class 3 System Boundary Description

Those portions of the systems listed in section 6.6.1 within the Class 3 boundary, based on Regulatory Guide 1.26 for Quality Group C (QGC) are as follows:

- Safety-related cooling water systems or portions of cooling water systems that are designed for emergency core cooling, post-accident containment heat removal, post-accident containment atmosphere cleanup, or residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems). Portions of these systems that are required for their safety functions and that do not operate during any mode of normal operation and cannot be tested adequately, however, are included with the Class 2 portion of the system.
- Cooling water and seal water systems or portions of these systems that are designed to maintain functioning of safety-related components and systems.
- Systems or portions of systems that are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves each of which is normally closed or capable of automatic closure.
- Systems, other than radioactive waste management systems, not covered by the above three paragraphs, that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (reference

Regulatory Guide 1.183), that exceed 0.5 rem to the whole body or its equivalent to any part of the body.

The above four paragraphs describe the Class 3 boundary only and are not exemptions from inservice examinations under ASME Section XI Code rules. The Class 3 components exempt from inservice examinations are described in ASME Section XI, IWD-1220.

6.6.2 Accessibility

All items within the Class 2 and 3 boundaries are designed to provide access for the examinations required by IWC-2500 and IWD-2500. Responsibility for designing components for accessibility for preservice and inservice inspection is the responsibility of the COL applicant.

Class 2 Piping, Pumps, Valves and Supports

Physical arrangement of piping pumps and valves provide personnel access to each weld location for performance of ultrasonic and surface (magnetic particle or liquid penetrant) examinations and sufficient access to supports for performance of visual (VT-3) examination. Working platforms are provided in some areas to facilitate servicing of pumps and valves. Removable thermal insulation is provided on welds and components, which require frequent access for examination or are located in high radiation areas. Welds are located to permit ultrasonic examination from at least one side, but where component geometry permits, access from both sides is provided.

Restrictions: For piping systems and portions of piping systems subject to volumetric examination, the following piping designs are generally not used:

- Valve to valve
- Valve to reducer
- Valve to tee
- Elbow to elbow
- Elbow to tee
- Nozzle to elbow
- Reducer to elbow
- Tee to tee
- Pump to valve

Straight sections of pipe and spool pieces shall be added between fittings. The minimum length of the spool piece has been determined by using the formula L=2T + 152 mm (2T + 6 inches), where L equals the length of the spool piece (not including weld preparation) and T equals the pipe wall thickness. Where such straight sections are not added or where less than the minimum straight section length is used, an evaluation shall be performed to demonstrate that sufficient access exists to perform the required examinations.

6.6.3 Examination Categories and Methods

6.6.3.1 Examination Categories

The examination category of each item in accordance with ASME Section XI, IWC-2500 and IWD-2500 will be listed in the preservice and inservice programs prepared by the COL licensee. The items will be listed by system and component description; or line number where available. The preservice and inservice inspection programs will also state the method of examination for each item.

For preservice examination, all of the items selected for inservice examination shall be performed once in accordance with ASME Section XI, IWC-2200 and IWD-2200, with the exception of the examinations specifically excluded by ASME Section XI from preservice requirements, such as the visual VT-2 examinations for Category C-H and D-A.

6.6.3.2 Examination Methods

6.6.3.2.1 Visual Examination

Visual Examination Methods, VT-2 and VT-3, shall be conducted in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations shall also meet the requirements of IWA-5240.

At locations where leakages are normally expected and leakage collection systems are located (e.g., valve stems and pump seals), the visual VT-2 examination shall verify that the leakage collection system is operative.

Piping runs shall be clearly identified and laid out such that insulation damage, leaks and structural distress are evident to a trained visual examiner.

Surface Examination

Magnetic Particle and Liquid Penetrant examination techniques shall be performed in accordance with ASME Section XI, IWA-2221 and IWA-2222 respectively. For direct examination access for magnetic particle (MT) and liquid penetrant (LP) examination, at least 610 mm (24 inches) of clear space is provided where feasible, for the head and shoulders of a man within a working man's arm length 508 mm (20 inches) of the surface to be examined. In addition, access shall be provided as necessary to enable physical contact with the item as necessary to perform the examination. Remote MT and PT generally are not appropriate as a standard examination process, however, bore scopes and mirrors can be used at close range to improve the angle of vision. As a minimum, insulation removal shall expose the area of each weld plus at least 152 mm (6 inches) from the toe of the weld on each side. Generally, insulation is removed 406 mm (16 inches) on each side of the weld.

6.6.3.2.2 Volumetric-Manual Ultrasonic Examination

Volumetric ultrasonic direct examination shall be performed in accordance with ASME Section XI, IWA-2232 and Appendix I. . In order to perform the examination, visual access to place the head and shoulder within 508 mm (20 inches) of the area of interest shall be provided where feasible. If there is free access on each side of the pipes, then 229 mm (9 inches) between adjacent pipes is sufficient spacing. The transducer dimension has been considered: a 38-mm

(1.5-inch) diameter cylinder, 76 mm (3 inches) long placed with the access at a right angle to the surface to be examined. The ultrasonic examination instrument has been considered as a rectangular box 305 x 305 x 508 mm (12 x 12 x 20 inches) located within 3.7 meters (12 feet) of the transducer. Space for a second examiner to monitor the instrument shall be provided if necessary.

Insulation removal for inspection is to allow sufficient room for the ultrasonic transducer to scan the examination area. A distance of 2T plus 152 mm (6 inches), where T is the pipe thickness, is the minimum required on each side of the examination area. The insulation design generally leaves 406 mm (16 inches) on each side of the weld, which exceeds minimum requirements.

6.6.3.2.3 Alternative Examination Techniques

As provided by ASME Section XI, IWA-2240, alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified for a given item in this section, provided that they are demonstrated to be equivalent or superior to the specified method. This provision allows for the use of newly developed examination methods, techniques, etc., which may result in improvements in examination reliability and reductions in personnel exposure. IWA-2240 as written in the 1997 Addenda of ASME Section XI must be used when applying these provisions.

6.6.3.2.4 Data Recording

Manual data recording is performed where manual ultrasonic examinations are performed. If automated systems are used, electronic data recording and comparison analysis are to be employed with the automated ultrasonic examination equipment. Signals from each ultrasonic transducer would be fed into a data acquisition system in which the key parameters of any reflectors are recorded. The data to be recorded for manual and automated methods are:

- Location;
- Position;
- Depth below the scanning surface;
- Length of the reflector;
- Transducer data including angle and frequency; and
- Calibration data.

The data for recorded indications shall be compared with the results of subsequent examinations to determine the behavior of the reflector.

6.6.3.2.5 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems shall be qualified in accordance with an industry accepted program for implementation of ASME Section XI, Appendix VIII.

6.6.4 Inspection Intervals

Class 2 Systems

The inservice inspection intervals for Class 2 systems conform to Inspection Program B as described in Section XI, IWC-2412. Except where deferral is permitted by Table IWC-2500-1, the percentages of examinations completed within each period of the interval shall correspond to Table IWC-2412-1. Inspection Program B provides for Inspection Intervals of a nominal length of 10 years with allowance for up to a year variation to coincide with refueling outages.

Class 3 Systems

The inservice inspection intervals for Class 3 systems conform to Inspection Program B as described in Section XI, IWD-2412. Except where deferral is permitted by Table IWD-2500-1, the percentages of examinations completed within each period of the interval shall correspond to Table IWD-2412-1. Inspection Program B provides for Inspection Intervals of a nominal length of 10 years with allowance for up to a year variation to coincide with refueling outages.

6.6.5 Evaluation of Examination Results

Examination results are evaluated in accordance with ASME Section XI, IWC-3000 for Class 2 components, with repairs based on the requirements of IWA-4000 and IWC4000. Examination results are evaluated in accordance with ASME Section XI, IWD-3000 for Class 3 components, with repairs based on the requirements of IWA-4000 and IWD-4000.

6.6.6 System Pressure Tests

6.6.6.1 System Leakage Test

As required by Section XI, IWC-2500 for category C-H and by IWD-2500 for category D-B a system leakage test shall be performed in accordance with IWC-5220 on Class 2 systems, and IWD-5220 on Class 3 systems The test shall include all Class 2 or 3 pressure retaining components and piping within the boundaries defined by IWC-5222 and IWD-5222. The test shall be performed once during each inspection period as defined in Tables IWC-2412-1 and IWD-2412-1 for Program B. The system leakage test shall include a VT-2 examination in accordance with IWA-5240. The system leakage test is conducted at the pressure during system operation or the test pressure used for systems that are not required to function during normal operation. The system hydrostatic test, when performed is acceptable in lieu of the system leakage test.

6.6.6.2 Hydrostatic Pressure Tests

A system hydrostatic test may be performed in lieu of a system leakage test, and when required for repairs, replacements, and modifications per IWA-4540. The test shall include all Class 2 or 3 pressure retaining components and piping within the boundaries defined by IWC-5222 and IWD-5222 or the boundary of a repair or replacement as applicable.

6.6.7 Augmented Inservice Inspections

High Energy Piping

High energy piping (defined within Subsection 3.6.2 and associated tables) between the containment isolation valves is subject to the following additional inspection requirements.

Circumferential welds shall be 100 percent volumetrically examined each inspection interval as defined within Subsections 6.6.3.2 and 6.6.4. Accessibility, examination requirements, and procedures shall be as discussed in Subsections 6.6.2, 6.6.3 and 6.6.5, respectively. Piping in these areas shall be seamless, thereby eliminating longitudinal welds.

Erosion-Corrosion

Piping systems determined to be susceptible to single-phase erosion-corrosion shall be subject to a system program of nondestructive examinations to verify the system structural integrity. The examination schedule and examination methods shall be determined in accordance with the NUMARC Program (or equally effective program, submitted by the COL applicant) as discussed in NRC Generic Letter 89-08, and applicable rules of ASME Section XI.

6.6.8 Code Exemptions

As provided in ASME Section XI, IWC-1220 and IWD-1220, certain portions of Class 2 and 3 systems are exempt from the volumetric, surface and visual examination requirements of IWC-2500 and IWD-2500.

6.6.9 Code Cases

As applicable, the provisions of the Code Cases listed in Table 5.2-1 may be used for preservice and inservice inspections, evaluations, and repair and replacement activities.

6.6.10 COL Information

Plant Specific PSI/ISI Program Information

Applicants will provide plant specific information concerning the PSI and ISI Programs for NRC review and approval. The information will need to:

- The development of the preservice and inservice inspection program plans will be the responsibility of the COL applicant/licensee and will be based on the ASME Code, Section XI, Edition and Addenda specified in accordance with 10 CFR 50.55a. The COL applicant shall specify the Edition of ASME to be used, based on the date of issuance of the construction permit or license, per 10 CFR 50.55a. The requirements presented in this section are provided for information, and are based on the 1989 Edition of ASME Section XI.
- Identify the specific areas where the applicable ASME Code requirements cannot be met after the initial examinations are performed and provide supporting technical justification for this request for relief.
- Include references to the edition and addenda of ASME Section XI Code that will be used for the selection of components for examination, lists of the components subject to examination, and a description of the components exempt from examination.

Information will be included in sufficient detail such that the Program and Inspection Plans form a complete and comprehensive document.

6.6.11 References

None.