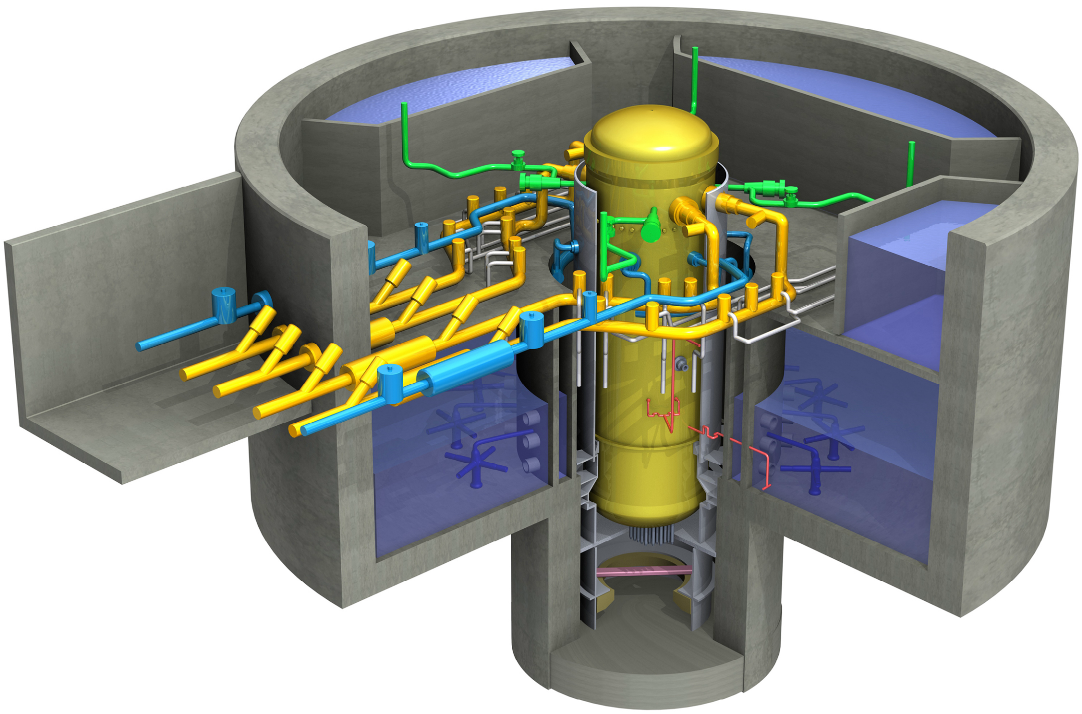
GE-Hitachi Nuclear Energy

26A6642AT
Revision 4
September 2007



ESBWR Design Control Document
Tier 2

Chapter 6
Engineered Safety Features

Contents

List of Tables

List of Illustrations

Global Abbreviations And Acronyms List

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
ABWR	Advanced Boiling Water Reactor
AC	Air Conditioning
ADS	Automatic Depressurization System
AHU	Air Handling Units
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
ASA	American Standards Association
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing Methods
B&PV	Boiler and Pressure Vessel
BiMAC	Basemat-Internal Melt Arrest Coolability
CB	Control Building
CFR	Code of Federal Regulations
CMS	Containment Monitoring System
COL	Combined Operating License
COLR	Core Operating Limits Report
CONAVS	Reactor Building Contaminated Area HVAC Subsystem
CPR	Critical Power Ratio
CR	Control Rod
CRD	Control Rod Drive
CRHA	Control Room Habitability Area
CRHAVS	Control Room Habitability Area HVAC Subsystem
CS	Containment System
CS / CST	Condensate Storage Tank
CWS	Chilled Water System
DBA	Design Basis Accident
DCD	Design Control Document
DCS	Drywell Cooling System
DPV	Depressurization Valve
DW	Drywell
ECCS	Emergency Core Cooling System
EFU	Emergency Filter Unit

Global Abbreviations And Acronyms List

Term	Definition
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
FAPCS	Fuel and Auxiliary Pools Cooling System
FMCRD	Fine Motion Control Rod Drive
FPS	Fire Protection System
FW	Feedwater
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GEH	GE-Hitachi Nuclear Energy
GEEN	General Electric Energy Nuclear
GENE	GE Nuclear Energy
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air/Absolute
HVAC	Heating, Ventilation and Air Conditioning
IC	Ion Chamber
IC	Isolation Condenser
ICS	Isolation Condenser System
IEEE	Institute of Electrical and Electronic Engineers
IGSCC	Intergranular Stress Corrosion Cracking
ILRT	Integrated Leak Rate Test
LD	Logic Diagram
LDW	Lower Drywell
LD&IS	Leak Detection and Isolation System
LOCA	Loss-of-Coolant-Accident
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MLHGR	Maximum Linear Heat Generation Rate
MOV	Motor-Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSL	Main Steamline
MT	Main Transformer
NBS	Nuclear Boiler System
NOV	Nitrogen Operated Valve
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission

Global Abbreviations And Acronyms List

Term	Definition
PCC	Passive Containment Cooling
PCT	Peak Cladding Temperature
PL	Parking Lot
PM	Preventive Maintenance
POV	Power Operated Valve
PSWS	Plant Service Water System
PWR	Pressurized Water Reactor
RB	Reactor Building
RCCWS	Reactor Component Cooling Water System
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RHX	Regenerative Heat Exchanger
RMS	Root Mean Square
RMS	Radiation Monitoring Subsystem
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSW	Reactor Shield Wall
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
SA	Severe Accident
SAM	Severe Accident Management
SC	Suppression Chamber
S/P	Suppression Pool
SBO	Station Blackout
SDC	Shutdown Cooling
SER	Safety Evaluation Report
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SP	Setpoint
SPC	Suppression Pool Cooling
SRP	Standard Review Plan
ADS-SRV	Safety Relief Valve
SS	Sub-scale
SSE	Safe Shutdown Earthquake
TAF	Top of Active Fuel
TEMA	Tubular Exchanger Manufacturers' Association
TMI	Three Mile Island
USNRC	United States Nuclear Regulatory Commission
WW	Wetwell

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) fission product containment and containment cooling systems; (2) emergency core cooling systems; and (3) control room habitability systems. The systems in each general group are:

(1) Fission Product Containment and Containment Cooling 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. Control Room Habitability Area (CRHA) HVAC Subsystem (CRHAVS)

6.1 ENGINEERED SAFETY FEATURE MATERIALS

Materials used in the 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. Only metallic insulation is used inside the containment. All nonmetallic thermal insulation employed outside the containment 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 Design Control Document (DCD) locations that address the relevant requirements of General Design Criteria (GDC) 1, 4, 14, 31, 35, and 41, Title 10, Code of Federal Regulations (10 CFR) Section 50.55a and Appendix B as discussed in SRP 6.1.1. 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 Loss-of-Coolant Injection (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, 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 including PCCS, and the Emergency Core Cooling Systems (ECCS.)

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 to any ESF materials from allowable containment levels in the ESBWR core cooling water and containment sprays. 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 including cleaning in accordance with Regulatory Guide 1.37 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, which 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. Because impurity levels in the water are controlled and 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, equipment and pipe supports 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 ASTM 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 and applicable reference standards including ASTM D 5144.

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.

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. (See Subsection 6.1.3.1 for COL items.)

6.1.3 COL Information

6.1.3-1-A Protective Coatings and Organic Materials

The COL Applicant will:

- Describe the approach to be taken to identify and quantify all organic materials that exist within the containment in significant amounts that do not meet the requirements of ASTM D 5144 and Regulatory Guide 1.54 as per section 6.1.2.
- Provide the milestone when evaluations will be complete to determine the generation rate, as a function of time, of combustible gases that can be formed from these unqualified organic materials under DBA conditions.
- As part of these evaluations, provide the technical basis and assumptions used.

Table 6.1-1

Containment System Including PCCS, and ECCS Component Materials

Component	Applicable ASME Code Section III,	Form	Material	Specification (ASTM/ASME)
Containment				
Containment Vessel Liner [1]	Div 2, Subsection CC	Plate ≤ 64 mm	Carbon Steel	SA-285 Gr ASA-516 Gr 60 or Gr 70
	Div 2, Subsection CC	Plate > 64 mm	Carbon Steel	SA-516 Gr 60 or Gr 70
	Div 2, Subsection CC	Plate	Stainless Steel	SA-240 Type 304L
Penetrations	Div 1, Subsection NE	Plate	Carbon Steel	SA-516 Gr 60 or Gr 70 SA-537 Class 1
	Div 1, Subsection NE	Pipe	Carbon Steel	SA-333 Gr 6
GDCS and Suppression Pool Liner	Div 2, Subsection CC	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	Div 1, Subsection NE	Shapes	Carbon Steel	A 36, A 572 Gr 50
Vent Pipe	Div 1, Subsection NE	Plate	Stainless Steel	SA-240 Gr 304L
PCCS				
Condenser	Div 1, Subsection NC	Forging	Stainless Steel	SA-182 Gr F304L
		Tube	Stainless Steel	SA-213 Gr TP304L
		Pipe	Stainless Steel	SA-312 Gr TP304L
Piping	Div 1, Subsection NC	Pipe	Stainless Steel	SA-312 Gr TP304L
Flanges	Div 1, Subsection NC	Forging	Stainless Steel	SA-182 Gr F304L

Table 6.1-1

Containment System Including PCCS, and ECCS Component Materials

Component	Applicable ASME Code Section III,	Form	Material	Specification (ASTM/ASME)
Nuts and Bolts	Div 1, Subsection NC	Bar	Stainless Steel	SA-194 Gr 8, SA-193 Gr B8
ADS				
DPV Body	See Table 5.2-4			
SRV Body	See Table 5.2-4			
SRV Discharge Piping Outside Suppression Pool	Div 1, Subsection ND	Pipe	Carbon Steel	SA-106 Gr B
SRV Discharge Piping Inside Suppression Pool	Div 1, Subsection ND	Pipe	Stainless Steel	SA-312 Gr TP316L[2]
GDCS				
Piping downstream of check valve	Div 1, Subsection NB	Pipe	Stainless Steel	SA-376 Gr TP304L or TP316L [2] SA-312 Gr TP304L or TP316L [2] SA-358 Gr TP304L or TP316L [2]
Piping-upstream of check valve	Div 1, Subsection NC			
Fittings	Same as mating pipe	Forging	Stainless Steel	SA-182 Gr F304L or F316L [2] SA-403 WP 304L or WP 316L [2]
Flanges	Same as mating pipe	Forging	Stainless Steel	SA-182 Gr F304L or F316L [2]
Valves (Gate, Squib, Check)				
Body	Div 1, Subsection NB	Forging	Stainless Steel	SA-182 Gr F304L or F316L [2]
		Casting	Stainless Steel	SA-351 Gr CF3 or CF3M
Bolts	Div 1, Subsection NB	Bar	Low Alloy Steel	SA-193 Gr B7 or B7M

Table 6.1-1

Containment System Including PCCS, and ECCS Component Materials

Component	Applicable ASME Code Section III,	Form	Material	Specification (ASTM/ASME)
Nuts	Div 1, Subsection NB	Bar	Low Alloy Steel	SA-194 Gr 7 or 7M
ICS				
Condenser	Div 1, Subsection NC	Tube	Alloy Steel	SB-163 (Inconel 600)
		Header	Alloy Steel	SB-564 (Inconel 600)
Steam Piping	Div 1, Subsection NB	Pipe	Carbon Steel	SA-333 Gr 6
Condensate Piping	Div 1, Subsection NB	Pipe	Stainless Steel	SA-376 Gr TP304L/316L [2] SA-312 Gr TP304L/316L [2] SA-358 Gr TP304L/316L [2]
SLC				
Accumulator	Div 1, Subsection NC	Plate Forging	Low Alloy Steel with Stainless Steel Cladding	SA-533 Gr B Cl 2 SA-508 Gr 3 Cl 1
Piping- downstream of injection valve	Div 1, Subsection NB	Pipe	Stainless Steel	SA-312 Gr TP316L [2]
Piping- upstream of injection valve	Div 1, Subsection NC	Pipe	Stainless Steel	SA-312 Gr TP316L [2]
Weld Filler Metals				
Carbon Steel P1, G1	Same as the component being welded	Covered Electrodes or Filler Wire	SFA-5.1 SFA-5.18	E7018 ER70s-2 ER70S-3 ER70S-6
Carbon Steel P1, G2	Same as the component being welded	Covered Electrodes or Filler Wire	SFA-5.1 SFA-5.18 SFA-5.28	E7018 ER70S-2 ER80S-D2

Table 6.1-1

Containment System Including PCCS, and ECCS Component Materials

Component	Applicable ASME Code Section III,	Form	Material	Specification (ASTM/ASME)
Low Alloy Steel P3, G3	Same as the component being welded	Covered Electrodes or Filler Wire	SFA-5.5 SFA-5.1 SFA-5.28 SFA-5.18	E8018-C3 E8018-G E7018 ER80S-D2 ER70S-2
Low Alloy Steel P5A, G1 (2-1/4Cr, 1Mo)	Same as the component being welded	Covered Electrodes or Filler Wire	SFA-5.5 SFA-5.1 SFA-5.28 SFA-5.18	E9016-B3 E9018-B3 E9018-B3L E7018 ER90S-B3 ER90S-B3L ER70S-2
Low Alloy Steel P5C, G1 (2-1/4Cr, 1Mo)	Same as the component being welded	Covered Electrodes or Filler Wire	SFA-5.5 SFA-5.1 SFA-5.28 SFA-5.18	E9016-B3 E9018-B3 E9018-B3L E7018 ER90S-B3 ER90S-B3L ER70S-2
Stainless Steel Filler	Same as the component being welded	Covered Electrode or Filler Wire	SFA-5.4 SFA-5.9	E308L-16 E309L-16 E316L-16 ER308L ER309L ER316L

Table 6.1-1

Containment System Including PCCS, and ECCS Component Materials

Component	Applicable ASME Code Section III,	Form	Material	Specification (ASTM/ASME)
Nickel Alloy Filler	Same as the component being welded	Filler Wire	SFA-5.14	ERNiCr-3

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 GDC 4, 16, 50, and 53 discussed in Standard Review Plan (SRP) 6.2.1.1.C. 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 (for example, 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 LOCA. A 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 RB surrounding the containment.

- The containment structure and RB, with concurrent operation of containment isolation function (isolation of 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 50.67.

- 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 RPV 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 RB surrounding the containment. The refueling bellows seal extends from the lower flange of the reactor vessel to the interior of the reactor cavity. This extension is also not part of the containment boundary, however, it provides a Seismic Category I seal between the upper drywell and reactor well during a refueling outage.

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, GDCS pools and piping, PCCS piping, 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 structure 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 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 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.

Vacuum breakers are provided between the DW and WW. The vacuum breaker is a process-actuated valve, similar to a check valve. 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-powered solenoid-controlled and spring-operated backup valve designed to fail-close is provided. The vacuum breaker is illustrated in Figure 6.2-28. During a LOCA, when the vacuum breaker opens and allows the flow of gas from WW to DW 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 backup 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 of vacuum breaker, that is, failure to open or failure to close when required. The design DW-to-WW pressure difference and the vacuum breaker full open differential pressure are given in Table 6.2-1.

The vacuum breaker valves are protected from pool swell loads by structural shielding designed for pool swell loads determined based on the Mark II/III containment design.

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 9.4.8.

Protection against the dynamic effects from the piping systems 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 RB 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. 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.

The spillover function provides drywell to wetwell connection for limiting suppression pool drawdown and the holdup volume in the drywell by transferring water from the drywell annulus to the suppression pool. Spillover is accomplished by twelve (12) horizontal holes (200 mm nominal diameter, (8 inch nominal diameter) which are built into the vent wall connecting the drywall annulus with each vertical vent module. Each spillover hole is horizontally oriented with with an elevation as shown in Figure 3G.1-57. If water, ascending through the drywell annulus following a postulated LOCA, reaches the spillover holes, it will flow into the suppression pool via the vertical/horizontal vent modules. Once in the suppression pool, the water can be used to maximum advantage for accident mitigation (that is, by restoration of RPV inventory). Figure 5.2-3 shows the location of the spillover holes.

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 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 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 analyses of a postulated instantaneous guillotine rupture of a feedwater line, a main steam line, a GDCS injection line, and a bottom head drain line. For plant operation with nominal feedwater temperature, the analysis results are discussed in this subsection. For plant operation with feedwater temperature maneuvering (increase and reduction), the limiting breaks were evaluated and results are discussed in Reference 6.2-7.

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-Hitachi Nuclear Energy (GEH) computer code TRACG (Reference 6.2-1). NRC has reviewed and approved the application of TRACG to ESBWR LOCA analyses, per the application methodology outlined in the report. The confirmatory items in the Staff's Safety Evaluation Report (SER) (Reference 6.2-1) concerning the TRACG computer code are addressed and provided in References 6.2-3 and 6.2-4. TRACG is applicable to LOCAs covering the complete spectrum of pipe break sizes, from a small break accident to a 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.

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.

Containment design basis calculations are performed for a spectrum of possible pipe break sizes and the results show that the Double-Ended Guillotine (DEG) pipe break is limiting. Results of DEG pipe break analyses at 4 different locations show that an instantaneous guillotine rupture of a main steam line with failure of one Depressurization Vavle (DPV) produces the most limiting responses for the containment pressure evaluation. The second limiting case is an instaneous guillotine rupture of a feedwater line with failure of one SRV. Table 6.2-5 summarizes the results of these DEG pipe break calculations. Subsections 6.2.1.1.3.1 through 6.2.1.1.3.5 discuss the results of these calculations.

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 utilized in the calculations. 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. Figure 6.2-8a shows the nodalization for the ESBWR isolation condenser system. Figure 6.2-8b shows the nodalization of the ESBWR feedwater line system.

This analysis follows the application methodology outlined in Reference 6.2-1. The TRACG nodalization approach in this analysis is similar to that used in Reference 6.2-1. However, this nodalization includes some additional features and details. Some of these features are implemented to address the confirmatory items listed in the Safety Evaluation Report of Reference 6.2-1. Other features are implemented due to design changes. Table 6.2-6a summarizes the list of these changes in the TRACG nodalization. The details of the TRACG application procedure of Reference 6.2-1 have been re-evaluated for the present configuration. Results of this evaluation show that the overall philosophy of the TRACG application procedure remains the same. Appendix 6A summarizes the details of this evaluation. Appendix 6B provides the justification for the use of the DCD nodalization (similar to that in Reference 6.2-1, as outlined in the first row of Table 6A-1), including the results of the tie-back calculations between these nodalizations.

The combined nodalization that integrates the responses between the containment and the reactor vessel is used for both the containment analyses (Subsection 6.2.1.1.3) and the ECCS analyses (Subsection 6.3.3). The impact of containment back pressure on the ECCS performance has been evaluated and the results show that the minimum chimney collapsed level is not sensitive for a wide range of change in the containment back pressure. Appendix 6C summarizes the details of this evaluation.

The analysis considers the contribution of radiolytic hydrogen and oxygen generation following the line break. Subsection 6.2.1.1.3.2 provides additional details on the gas generation. The analysis also considers one vacuum breaker and one IC out of service, that is, two vacuum breakers and 3 ICs are available. In addition, the analysis only models the initial water inventory in the 3 ICs, and conservatively takes no credit for the heat transfer in any of the ICs.

Table 6.2-7 shows the sequence of events for this analysis. Figures 6.2-9a1 through 6.2-9d3 show the pressure, temperature, PCCS and DW and GDCS airspace pressure responses for this analysis. Table 6.2-5 summarizes the results of this calculation. The calculated maximum drywell pressure during the 72 hours following a LOCA for the nominal case is below the containment design pressure.

6.2.1.1.3.2 Main Steam 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. The analysis considers the contribution of radiolytic hydrogen and oxygen generation following the line break. The generation rate of radiolytic gas depends on the reactor decay power profile, whether the reactor coolant is boiling, and the amount of fission products released to the coolant. Appendix A of SRP Section 6.2.5 provides a conservative 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 Regulatory Guide (RG) 1.7, except that radiolysis from fission products transported to sump water was not included because fuel failure does not occur in this accident sequence. Also production of hydrogen from fuel cladding metal-water reaction is not a contributor, because cladding temperatures remain below the point at which metal-water reaction occurs.

Table 6.2-7a shows the sequence of events for this analysis. Figures 6.2-10a1 through 6.2-10d3 show the pressure, temperature, PCCS and DW and GDCS airspace pressure responses for this analysis. Table 6.2-5 summarizes the results of this calculation. The calculated maximum drywell pressure during the 72 hours following a LOCA for the nominal case is below the containment design pressure.

6.2.1.1.3.3 GDCS Line Break and Bottom Drain Line Break – Nominal Analysis

These analyses initialize the RPV and containment at the base conditions shown in the Nominal Value column of Table 6.2-6. Table 6.2-7b shows the sequence of events for the GDCS line break analysis. Figures 6.2-11a1 through 6.2-11d3 show the pressure, temperature, PCCS and DW and GDCS airspace pressure responses for the GDCS line break analysis. Table 6.2-7c shows the sequence of events for the bottom drain line break analysis. Figures 6.2-12a1 through 6.2-12d3 show the pressure, temperature and PCCS responses for the bottom drain line break analysis. Table 6.2-5 summarizes the results of these calculations. The calculated maximum drywell pressures during the 72 hours following a LOCA for these nominal cases are below the containment design pressure.

6.2.1.1.3.4 Feedwater Line Break – Bounding Analysis

This analysis initializes the RPV and containment at the base conditions shown in the Bounding Value column of Table 6.2-6. Table 6.2-7d shows the sequence of events for their analysis. 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 6.2-13a1 through 6.2-13d3 show the pressure, temperature, PCCS and DW and GDCS airspace pressure responses for this analysis. Table 6.2-5 summarizes the results of this calculation. The calculated maximum drywell pressure during the 72 hours following a LOCA for the bounding case is below the containment design pressure.

6.2.1.1.3.5 Main Steam Line Break – Bounding Analysis

This analysis initializes the RPV and containment at the base conditions shown in the Bounding Value column of Table 6.2-6. Table 6.2-7d shows the sequence of events for their analysis. 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 6.2-14a1 through 6.2-14d3 show the pressure, temperature, PCCS and DW and GDCS airspace pressure responses for this analysis. Table 6.2-5 summarizes the results of this calculation. The calculated maximum drywell pressure during the 72 hours following a LOCA 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 RB 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, and so forth) 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 MSL break analysis show that the containment does not reach negative pressure relative to the RB, and the maximum wetwell-drywell differential pressure is within the design capability. This calculation assumes one available wetwell-drywell vacuum breaker with an area of 0.2 m^2, (2.16 ft^3), which is conservative with respect to the planned installed vacuum breaker area. An evaluation of the effect of drywell spray on containment integrity for a main steam line break and a feedwater line break was performed to determine the maximum negative differential pressures (drywell to wetwell, and drywell to reactor building). This evaluation assumed that a drywell spray flow rate of 454 m^3/hr (2000 gpm) at a temperature of 293ºK is initiated at the worst possible moment (i.e., when the drywell pressure has peaked just prior to the opening of the drywell-wetwell vacuum breakers), and verified that the maximum negative differential pressures remain within the design criteria. For additional conservatism and to account for uncertainties in the design of the drywell spray piping system, including drywell spray flow limiting design features, a value of 227 m^3/hr (1000 gpm) has been established as the maximum design operating limit (see Subsection 9.1.3).

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 (1.08E-03 ft^2), ($A/\sqrt{K}$). Table 6.2-5 shows this results in acceptable containment pressures. Additional bounding design basis accident calculations show also that with a bypass leakage assumption of 2 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$) the containment pressures continue to be below the design pressure and with a bypass leakage assumption of 14 cm^2 (1.51E-02 ft^2), ($A/\sqrt{K}$) the containment pressures remain below the ultimate pressure capability of the drywell head (1.204 MPag) (see reference 6.2-6) with ample margin.

6.2.1.1.5.2 Suppression Pool Bypass During Severe Accidents

See Chapter 19 for discussion on Suppression Pool Bypass During Severe Accidents.

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.

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 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, monthly testing is not performed for these vacuum breakers. However, the vacuum breakers are tested for free movement and leakage during each outage.

6.2.1.1.5.4 Bypass Leakage Tests and Surveillance

There is 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 a pre-operational and periodic local leak rate testing of vacuum breakers and, a periodic visual inspection of drywell to wetwell penetrations.

6.2.1.1.5.4.1

Deleted.

6.2.1.1.5.4.2 Local Leak Rate Testing of Drywell to Wetwell

A pre-operational and post-operational visual inspection of drywell to wetwell penetrations and local leak rate testing of vacuum breakers is performed to detect leakage from the drywell to wetwell. This test is performed at each refueling outage. A low-pressure test is not conducted since the vacuum breakers are the only credible source for bypass leakage; other existing penetrations are pipe connections that are welded and cannot physically leak. The acceptance criteria are specified in Subsection 6.2.1.1.5.4.3.

6.2.1.1.5.4.3 Acceptance Criteria for Leakage Tests

NUREG-0800, 6.2.1.1.c Draft 1996, Appendix A, Steam Bypass, specifies acceptance criteria for drywell/wetwell steam bypass testing for Mark I, II and III containments. It states that alternative criteria can be proposed for review by the NRC staff. For ESBWR an alternate criteria is proposed, to:

- Provide a drywell/wetwell interface, sufficiently leak tight, to assure the containment performs the intended function of containment of radioactivity.
- Provide flexibility for the licensee in conducting tests.
- Account for degradation in performance between tests.
- Account the uncertainties in test measurement.

The criteria specified for Mark II and III containments is a fraction of the analytical leakage capability. The fraction is judged small enough to cover degradation in performance between tests and uncertainties in test measurement. For ESBWR, an alternate acceptance criteria will be applied. The ability of the containment to tolerate degraded (increased) leakage up to ultimate strength has been determined to be more than a factor of 5 above the design capability (see Subsection 6.2.1.1.5.1). This adequately bounds potential degradation between test intervals. The uncertainty in the test measurement will be quantified and applied to the acceptance criteria. The acceptance criteria will be the leakage analytically required to keep the containment below design pressure, 2 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$). The uncertainties associated with the specific test procedure and equipment applied will be determined by the licensee and added to the measured leakage prior to comparison against the acceptance criteria.

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 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.

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 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 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 Subsection 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 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 (for example, 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, and
- 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) supports both reactor water level and pressure control and 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 the setpoint for 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.

The RPV depressurizes in the event of a break in the primary system or after ADS actuation. Furthermore, the ESBWR design does not require the operation of the ICs to prevent containment pressurization and containment pressure control function is served by the Passive Containment Cooling System (PCCS).

(2) Automatic Depressurization 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 RB volume is reduced by relocating selected equipment and systems to areas outside of the RB. The major portion of this relocation is to remove non-safety items from the Seismic Class 1 structure and to place them in other structures that are classified as Non-Seismic. Along with other system design simplifications and the above described relocation of non-safety items, a compact containment design is achieved with the characteristic of having a minimum number of penetrations. This reduces the leakage potential 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)

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 deluge 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 multiple 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 deluge flooder lines from the GDCS pools provide sufficient water to quench all core debris. The deluge lines originating from the GDCS provide water to the Basemat-Internal Melt Arrest and Coolability (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 GDCS pools. 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 Configuration

The GDCS pools are placed above the RPV with their air space connected to the drywell. A line with normally closed valves connects the GDCS pools 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 Lower Drywell (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 relevant 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 (for example, 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, that is, 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, for example, 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.
  - 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. The as-built calculated differential pressure is not expected to be substantially different from the design value. 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. At least 15% margin above the analytically determined pressures is applied for structural analysis.

6.2.1.2.2 Design Features

The DW and WW subcompartments are described in Subsection 6.2.1.1. The remaining containment subcompartments are 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 6.2-1. 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

FWL or RWCU line break within the Reactor Shield Annulus are identified to be the accident with most severe consequences. Mass and Energy releases from the postulated pipe breaks are based on the reactor operating condition prior to the break. It was assumed that the reactor is operating at full power and the containment is filled with dry air at atmospheric pressure and 100°C when the postulated pipe break occurs. The mass release rates are determined with Moody's Frictionless Critical Flow Model (Reference 6.2-5). Analyzed with TRACG, the peak subcompartment pressure responses were found to be below the design pressure for 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;
  - 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; and
  - 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 to the containment.
- Break Size and Location
  - The choice of break locations and types is discussed in Subsection 6.2.1.1.3;
  - Of several breaks postulated on the basis stated above, the break selected as the reference case yields the highest containment pressure consistent with the criteria for establishing the break location and area; and
  - 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

  Following the procedure, documented in Reference 6.2-1, calculations of the mass and energy release rates for a LOCA are performed in a manner that conservatively establishes the containment internal design pressure (that is, maximizes the post-accident containment pressure).

A spectrum of breaks was considered and analyzed using GEH-developed and approved computer codes described in Reference 6.2-1. The summary of this evaluation is discussed in Subsection 6.2.1.1.3.

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 RB 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-related design; that is, having suitable redundancy of components and features, and interconnections, that ensures that for a loss of offsite 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; and
- 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.

6.2.2.1 Design Basis

Functions

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 without makeup to the Isolation Condenser/Passive Containment Cooling (IC/PCC) pools, dryer/separator pool, and reactor well.

The PCCS is an 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 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 102°C (216°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 Tubular Exchanger

Manufacturers Association (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 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 drywell and WW vacuum breaker must fully close after each demand to support the PCCS operation. If the vacuum breaker does not close, a backup isolation valve will close.

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, 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 102°C (216°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 expansion 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 a pneumatic powered or equivalent Power Operated Valve (POV) 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 expansion pool.

Cooling and cleanup of IC/PCC pool water is performed by the FAPCS (Subsection 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 RB, 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 valves, 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; and
- 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 it's 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 (for example, 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 an essentially leak-tight barriers against the uncontrolled release of radioactive material to the environment;
- GDC 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; and
- 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 RB design. The RB structure encloses 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 RB HVAC system through a High

Efficiency Particulate Air/Absolute (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 RB 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 RB 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;
- 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 that 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 RB is a reinforced concrete structure that forms an envelope completely surrounding the containment (except the basemat). The boundary of the clean areas and the RB are shown in Figure 6.2-17.

During normal operation, the RB potentially contaminated areas are maintained at a slightly negative pressure relative to adjoining areas by the Contaminated Area HVAC Subsystem (CONAVS) portion of the RB 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. RB 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 RB is through interlocked doors. The RB HVAC system is designed and tested for isolation under accident conditions.

High-energy line breaks in any of the RB 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 (Subsection 7.4.3). 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 RB. 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 RB operating floor. Figure 6.2-18 shows the model of the RB 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. Figures 6.2-19 through 6.2-27 provide the pressure profiles due to all postulated RWCU/SDC system break cases for each individual room/region. The envelope profile represents the calculated maximum pressure response values for the given room/region due to all postulated RWCU/SDC system pipe breaks. No margin is included in these pressure profiles. Table 6.2-12b provides the mass and energy release data for the break cases analyzed.

Isolation Condenser (IC) System

The isolation condensers are located in the RB at the 27000 mm elevation. The IC steam supply line is connected directly to the RPV. 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 RB main steam tunnel is located between the primary containment vessel and the turbine building. The limiting break is a main steam line longitudinal break. The main steam lines originate at the RPV 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 RB, 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 RB 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 pressures, which include a 10% margin, are listed in Table 6.2-12a, out of which the maximum is 32.6 kPag which is below the RB compartment pressurization design requirement 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. Mass and energy blowdown data are given in Table 6.2-12b.

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 RB 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, and subcompartment nodal description are given in Table 6.2-12a. Blow out panels are passive, and blow out pressure listed in Table 6.2-12 is the upper bound. Heat sinks are credited and the characteristics are given in Table 6.2-12c.

The selected nodalization maximizes differential pressure. Owing to the geometry of the regions, each room-region was assigned to a node of the model. No simple or artificial divisions of rooms were considered to evaluate the sensitivity of the model to nodalization. A sensitivity study of pressure response was performed to select the time step. Additional sensitivity studies were performed to evaluate the impact of the heat sinks, dropout, and inertia term. Modeling follows the recommendations given by SMSAB-02-04, "CONTAIN Code Qualification Report/User Guide for Auditing Subcompartment Analysis Calculations"

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 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. Containment isolation signals result from diverse sources of sensory inputs. Subsections 5.2.5 and 7.3.3.2 describe the parameters used to initiate these signals.

Relevant to the containment isolation function, this subsection addresses or references to other DCD locations that address the applicable requirements of 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. Regulatory Guide 1.141 and ANS 56.2 are used as guidance documents for the design of containment isolation provisions for fluid systems. The plant meets the relevant requirements of:

- 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 (for example, earthquakes) without loss of capability to perform their safety functions; and systems being designed to accommodate postulated environmental conditions and protected against dynamic effects (for example, 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 they relate 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. A simple check valve may not be used as the automatic isolation valve outside containment; or
  - One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the 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. This valve is to be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.
- 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 50.34(a)(1) limits. Leak-tightness of the valves shall be verified by Type C test.
- Capability for rapid closure or isolation of pipes or ducts that penetrate the containment is performed by means or devices that provide a containment barrier 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. Exemptions from GDCs are listed in Table 1.9-6.
- 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 functions for automatic containment isolation valves ensures that resetting the isolation signal shall not result in the automatic reopening of containment isolation valves, and,
- Penetrations with trapped liquid volume between the isolation valves have adequate relief for thermally-induced pressurization.

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 powered or equivalent containment isolation POVs 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 Tables 6.2-16 through 6.2-42.

Penetration piping is evaluated for entrapped liquid subject to thermally-induced pressurization following isolation. The preferred pressure relief method is through a self-relieving penetration by selection and orientation of an inboard isolation valve that permits excess fluid to be released inward to the containment. Use of a separate relief valve to provide penetration piping overpressure protection is permissible on a case-by-case basis when no other isolation valve selection option is available.

The criteria for the design of the 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 RCPB influent lines are identified in Table 6.2-13, and the RCPB effluent lines are identified in Table 6.2-14. Table 6.2-15 through 6.2-42 show the pertinent data for the containment isolation valves. (Refer to COL item in section 6.2.8). 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 indicating switches in the control room to show whether the valve is open or closed. 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 POVs with geared or bi-directional actuators (motorized or fluid-powered) remain in their last position upon failure of valve power. All POVs with fluid-operated/spring-return actuators (not applicable to air-testable check valves) close on loss of fluid pressure or power supply. To support the inerted containment design, pneumatic actuators for valves located inside containment are supplied with pressurized nitrogen gas, whereas pneumatic actuators for valves located outside of containment are generally supplied compressed air.

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 process-actuated check valves, are identified in the containment isolation valve information Tables 6.2-13 through 6.2-42.

Systems containing penetrations that support or provide a flow path for emergency operation of ESF systems are not automatically isolated. The penetrations supporting ESF systems include some of the Fuel and Auxiliary Pool Cooling system (FAPCS) penetrations. Those FAPCS penetrations required for emergency operation include remote manual isolation valves or check valves. In addition, the Standby Liquid control Sysstem (SLC) and Isolation condenser System (ICS) are ESF systems that have fluid paths through containment penetrations. The SLC penetrations are not automatically isolated and do not contain remote manual isolation valves. Instead, the SLC penetrations are isolated if necessary by process-actuated check valves, but only after the SLC flow into the reactor pressure vessel/containment has ceased following an accident. The ICS penetrations listed in Tables 6.2-23 through 6.2-30 consist of various system process lines, all of which may be open or required to be opened following an accient in order to perform the required ESF function. The ICS penetration flow paths contain remote manual isolation valves, process-actuated flow control valves, or automatic isolation valves that only close for the applicable ICS train if leakage outside of containment is detected through IC/Passive Containment Cooling (PCC) pool high radiation or IC lines high flow.

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 CFR50.67. 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. However the design values of closure times for power-operated valves is more conservative than the above requirement. For valves above 80 mm (3 inches) up to and including 300 mm (12 inches) in diameter, the closure time is at least within a time determined by dividing the nominal valve diameter by 300 mm (12 inches) per minute. Valves 80 mm (3 inches) and less generally close within 15 seconds. All valves larger than 300 mm (12 inches) in diameter close within 60 seconds unless an accident radiation dose calculation is performed to show that the longer closure time does not result in a significant increase in off-site dose.

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, a manually-operated isolation valve just outside the containment followed by an excess flow check valve. The instrument line is designed such that the instrument response time is acceptable with the presence of the orifice, and that the flow restriction is not plugged.

6.2.4.2.3 Compliance with General Design Criteria and Regulatory Guides

In general, all requirements of General Design Criteria 54, 55, 56, 57 and Regulatory Guides 1.11 and 1.141 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 RB. 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.

Section 3.11 presents a discussion of the environmental conditions, both normal and accidental, for which the containment isolation valves and pipe are designed. Containment isolation valves and associated pipes are designed to withstand the peak calculated temperatures and pressures during postulated design basis accidents to which they would be exposed. 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 and meet at least Group B quality standards, as defined in RG 1.26. 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 single active failure of any kind should prevent 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 Guides 1.11 and 1.141.

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 (for example, magnetic fields, high radiation, high temperature and high humidity).

Administrative controls will be applied by the plant operators by using established procedures and checklist for all non-powered containment isolation valves to ensure that their position 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 RCPB, as defined in 10 CFR 50, Section 50.2, consists of the RPV, pressure-retaining appurtenances attached to the vessel, valves and pipes which extend from the RPV 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 RPV. It has two containment isolation valves with process-actuated closure to isolate the line in the event of an outboard feedwater pipe rupture (feedwater HELB). Additionally, two valves with automatic power-actuated closure, including the outboard containment isolation valve, isolate the line in the event of an inboard feedwater pipe rupture (feedwater LOCA). The isolation valve inside the containment is a check valve, located as close as practicable to the containment wall. Outside the containment is a non-simple check valve located as close as practicable to the containment wall. The non-simple check valve outside containment is provided with powered actuation that, upon an automatic or remote manual signal from the main control room, provides closure force to the valve disk for isolation. An additional POV with automatic closure is provided upstream of the outboard containment isolation valve as a redundant backup valve for feedwater isolation for the LOCA event.

Isolation Condenser Condensate and Venting Lines

The isolation condenser condensate lines penetrate the containment and connect directly to the RPV. The isolation condenser venting lines extend from the isolation condenser through the containment and connect together downstream of two tandem installed normally-closed stop valves. 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 condensate return line isolating shutoff 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. 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. The combination of an already closed loop outside the containment plus the two series automatic isolation valves inside the containment comply with the requirements 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 SLC system line penetrates the containment to inject directly into the RPV. In addition to a simple check valve inside the containment, a check valve, together with two parallel squib-activated 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-activated 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 RPV 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 shutoff valves, one inside and one just outside the containment. The main steam line isolation valves (MSIVs) are described in Subsection 5.4.5.

Isolation Condenser Steam Supply Lines

The isolation condenser steam supply lines penetrate the containment and connect directly to the RPV. Two isolation shutoff 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 functions 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 consists of two independent trains. Each train takes its suction from the RPV mid-vessel region as well as from the RPV bottom region. The suction lines of each train are isolated by one automatic pneumatic-operated valve inside and one automatic pneumatic-operated valve outside the containment. The reactor bottom suction line has a sampling line isolated by one automatic solenoid-operated valve inside and one automatic solenoid-operated valve outside the containment. The details regarding these valves are shown in Table 6.2-31. 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 ensure 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

Tables 6.2-33 through 6.2-42 identifies 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 GDCS pools. In each of these lines there is one pnuematic-operated or equivalent– shutoff valve outside and one check valve inside the containment. Only the GDCS pool return line pneumatic-operated or equivalent shutoff valve is automatically closed on a containment isolation signal.

Subsection 9.1.3.3 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 non safety-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 pneumatic-operated or equivalent shutoff valve outside and a pneumatic-operated or equivalent shutoff inside the containment.

Containment Inerting System

The penetration of the Containment Inerting System consists of two tandem quarter-turn or equivalent shutoff valves (normally closed) in parallel with two tandem stop or shutoff valves. All isolation valves on these lines are outside of the containment to provide accessibility to the valves. Both containment isolation valves are located as close as practical to the containment. The valve nearest to the containment is provided with a capability of detection and termination of a leak. The piping between the containment and the first isolation valve and the piping between the two isolation valves are designed as per requirements of SRP 3.6.2. These piping are also designed to:

- Meet Safety Class 2 design requirements;
- Withstand the containment design temperature;
- Withstand internal pressure from containment structural integrity test;
- Withstand loss-of-coolant accident transient and environment;
- Meet Seismic Category I design requirements; and
- Are protected against a high energy line break outside of containment when needed for containment isolation.

High Pressure Nitrogen Supply System

The High Pressure Nitrogen Supply System penetrates the containment at two places. Each line has one air-operated shutoff valve outside and one check valve inside the containment.

Passive Containment Cooling System

The PCCS does not have isolation valves as the heat exchanger modules and piping are designed as extensions of the safety-related containment. The design pressure of the PCCS is greater than twice the containment design pressure and the design temperature is same as the drywell design temperature.

6.2.4.3.2.2 Effluent Lines from Containment

Tables 6.2-33 through 6.2-42 identify 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 shutoff valves, one pneumatic-operated or equivalent inside and one pneumatic-operated or equivalent outside the containment.

Before it exits containment, the FAPCS suction line from the suppression pool branches into two parallel lines, each of which penetrate the containment boundary. Once outside, each parallel flow path contains two pneumatic isolation valves in series after which the lines converge back into a single flow path. Because the penetration can be under water under certain accident conditions, there can be no isolation valve located inside the containment. The valves are located as close as possible to the containment.

Subsection 9.1.3.3 contains additional information about the containment isolation design for FAPCS

Chilled Water System

The CWS effluent lines penetrating the containment each has a pneumatic-operated or equivalent shutoff valve outside containment and a pneumatic-operated or equivalent shutoff valve inside the containment.

Containment Inerting System

The penetration of the Containment Inerting System consists of two tandem quarter-turn shutoff valves (normally closed) in parallel with tandem stop or shutoff valves. All isolation valves on these lines are outside of the containment to provide accessibility to the valves. Both containment isolation valves are located as close as practical to the containment. The valve nearest to the containment is provided with a capability of detection and termination of a leak. The piping between the containment and the first isolation valve and the piping between the two isolation valves are designed as per requirements of SRP 3.6.2. These piping are also designed to:

- meet Safety Class 2 design requirements;
- withstand the containment design temperature;
- withstand internal pressure from containment structural integrity test;
- withstand loss-of-coolant accident transient and environment;
- meet Seismic Category I design requirements; and

- are protected against a high energy line break outside of containment when needed for containment isolation.

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 tandem stop or shutoff valves. One valve is a manual-operated valve used for maintenance and is located close to the containment. The other two valves are pneumatic, solenoid or equivalent power operated 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.

Passive Containment Cooling System

The PCCS does not have isolation valves as the heat exchanger modules and piping are designed as extensions of the safety-related containment. The design pressure of the PCCS is greater than twice the containment design pressure and the design temperature is same as the drywell design temperature.

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 are within the scope of GDC 57.

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 (for example, 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. If a component, such as an electrically-operated valve, is designed to not receive a signal to change state (open or close) in a safety scheme, it is accounted as a single failure if the system component does change state. 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 remote-manually actuated from the main control room. Therefore, a single failure in any electrical or mechnical system is analyzed, regardless of whether the loss of a safety-related function results from a component failing to perform a requisite mechanical motion or a component performing an unnecessary mechanical motion. Each of the power operated containment isolation valves for any given penetration is powered from different divisions in order to meet the single failure criteria.

6.2.4.4 Test and Inspections

The automatic functions of the Containment Isolation Valves (CIVs) are periodically tested by ensuring actuation to the isolation position on an actual or simulated isolation signal. 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.

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.

A function, for reducing pressure post 72 hours design basis accident, will be added to containment. 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 as described in Subsection 6.2.5.5.

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

In addition to inerting containment, the Flammability Control System (FCS), a nonsafety-related system, is provided as defense-in-depth protection against the potential buildup of combustible gases generated by the radiolytic decomposition of water post 72 hours of a LOCA. The FCS is designed for long-term continuous operation. It performs its function by controlled reaction of hydrogen with oxygen at low volumetric concentrations of whichever of these two gaseous constituents is limiting the progress of the reaction. The FCS consists of passive autocatalytic recombiners (PARs) strategically located throughout the WW gas space and DW.

PARs capacity is sized with ample margin to recombine the production of radiolytic decomposition of water post 72 hours of LOCA and with sufficient margin to account for any recombination prior 72 hours. In addition, PARs capacity shall have minimal loss of catalytic efficiency under the foreseeable threat spectrum involving the generation of common catalytic poisons released.

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 As Low As Reasonably Achievable (ALARA) principles for occupational exposure. The piping, valves and controls in the Containment Inerting System can be used to control containment pressure (that is, purge the containment), and can reliably be isolated under accident conditions.

6.2.5.2 Containment Inerting System

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.

6.2.5.2.1 Design Bases

Safety (10 CFR 50.2) Design Bases

The Containment Inerting System (CIS) does not perform any safety-related function. Therefore, the CIS has no safety design bases other than provision for safety-related containment penetrations and isolation valves, as described in Subsection 6.2.4.

Power Generation Design Bases

- The CIS is designed to establish an inert atmosphere (i.e., less than 4% oxygen by volume) throughout the containment in less than 4 hours and less than 2% oxygen by volume in the next 8 hours following an outage.
- The CIS is designed to maintain the containment oxygen concentration below the maximum permissible limit (3%) during normal, abnormal, and accident conditions to assure an inert atmosphere.
- The CIS is designed to maintain a positive pressure in the primary containment during normal, abnormal, and accident conditions to prevent air (oxygen) in-leakage into the inerted spaces from the Reactor Building. The CIS nitrogen gas makeup supply line is designed for the normal daily operating capacity to maintain approximately 4.8 kPaG (0.7 psig) positive pressure within the containment. The system has the capability to replenish containment atmosphere leakage at a design value of 0.5% per day based on containment operating pressure.
- The inerting auxiliary steam vaporizer is sized to provide at least 2.5 times the containment (wetwell and drywell) free volume of nitrogen within the allotted 4 hours. The temperature of the injected nitrogen is within the range of 10°C (50°F) to 65°C (150°F).
- The CIS is designed to permit de-inerting the containment for safe operator access without breathing apparatus in less than 12 hours.

- The CIS is designed to perform continuous containment leakage rate monitoring and detect gross leakage of containment atmosphere during normal reactor operation.
- The CIS is also designed to release containment pressure before uncontrolled containment failure could occur. (See Subsection 6.2.5)

6.2.5.2.2 System Description

Summary Description

The CIS establishes and maintains an inert atmosphere within the primary containment during all plant operating modes except during plant shutdown for refueling or maintenance and during limited periods of time to permit access for inspection during reactor low power operation. The purpose of the system is to provide an inert containment atmosphere (≤ 3% oxygen) during normal operation to minimize hydrogen burn inside the containment.

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.

A simplified CIS system diagram is shown in Figure 6.2-29 and subsection 9.4.10 describes component information.

Detailed System Description

The CIS consists of a pressurized liquid nitrogen storage tank, a steam-heated main vaporizer for large nitrogen flow, electric heater for vaporizing makeup flow, two injection lines, an exhaust line, a bleed line and associated valves, controls and instrumentation. All CIS components are located inside the Reactor Building except the liquid nitrogen storage tank and the steam-heated main vaporizer that are located in the yard.

The first of the injection lines is used only for makeup. It includes an electric heater to vaporize the nitrogen and to regulate the nitrogen temperature to acceptable injection temperatures. Remotely operated valves, together with a control valve, enable the operator to accomplish low rates of nitrogen injection into the drywell and suppression pool airspace.

The second injection line is used for the inerting function where larger flow rates of nitrogen are required. This line provides the flow path for vaporized nitrogen at an appropriate temperature from the steam-heated main vaporizer to be injected into the containment through remotely operated valves and a control valve to injection points common with the makeup supply. The inerting and makeup lines converge to common injection points in the upper drywell and suppression pool airspace.

The CIS includes an exhaust line from the lower drywell on the opposite side of containment from the injection points. The discharge line connects to the Reactor Building HVAC system exhaust before being diverted to the plant stack. The Reactor Building HVAC system is discussed in Subsection 9.4.6.

A small bleed line bypassing the main exhaust line is also provided for manual pressure control of the containment during normal reactor operation.

Redundant containment isolation valves provided in the inerting, makeup, exhaust and bleed lines close automatically upon receipt of an isolation signal from the Leak Detection and Isolation System (LD&IS). Discussion of these signals is provided in Subsection 7.3.3.

Upstream of the pressure-reducing valve in the makeup line, a small branch line is provided and connected to the High Pressure Nitrogen Supply System (HPNSS) (see Subsection 9.3.8).

System Operation

During plant startup, large flow rates of nitrogen from the liquid nitrogen storage tank are vaporized by the steam-heated vaporizer and injected into the drywell and the wetwell. The exhaust line is kept open to displace containment resident atmosphere with nitrogen. Once the desired concentration of oxygen is reached, the exhaust line is closed. When the required inerted containment operating pressure is attained, the nitrogen supply shutoff valve and the inerting isolation valves are closed to terminate the inerting process. The system is capable of inerting the containment to $\leq$ 4% oxygen by volume within four hours. The CIS is capable of establishing a more completely inert atmosphere, equal to or less than 2% in containment with the next eight hours after reaching 4% conditions.

Containment pressure is maintained automatically after manually aligning the nitrogen makeup subsystem. Low flows of liquid nitrogen are vaporized and heated to the desired temperature and injected into the drywell and the wetwell to makeup for the nitrogen out-leakage. The containment atmosphere is kept constant at a positive pressure relative to the Reactor Building to preclude air (oxygen) in-leakage. In response to a change in containment pressure, the pressure control valve modulates (opens or closes) to provide nitrogen makeup and thereby maintaining the containment pressure. The flow integrator monitors the nitrogen makeup to compensate for leakage during normal containment pressure control and to the HPNSS. Large makeup flow indicates gross or excessive leakage and is annunciated in the MCR. Manual venting through the exhaust bleed line controls increases in containment pressure greater than the normal operating range.

During plant shutdown, the containment atmosphere is de-inerted to allow safe personnel access inside the containment. Breathable air from the Reactor Building HVAC system is injected into the drywell and wetwell air space through the inerting injection line. The incoming air displaces containment gases (mostly nitrogen) into the exhaust line. The Reactor Building HVAC system exhaust fans, filters, and radiation detectors remove the vented gases and then they are diverted to the plant stack. The CIS is capable of reaching a volumetric oxygen concentration of $\geq$ 19% within 12 hours after de-inerting begins.

6.2.5.2.3 Safety Evaluation

The CIS has no safety-related function except the containment isolation function, which is discussed in Subsection 6.2.4. Failure of the CIS does not compromise any safety-related system or component, nor does it prevent a safe shutdown of the plant.

6.2.5.2.4 Testing and Inspection Requirements

CIS containment penetrations, including isolation valves, undergo routine in-service inspection and testing as required by ASME Code, Section XI.

Permanently installed instrumentation inside containment is maintained, tested, and calibrated during every refueling outage.

6.2.5.2.5 Instrumentation Requirements

CIS instrumentation requirements are discussed in Subsection 7.7.7.

This instrumentation conforms to GDC 13. Refer to Subsection 3.1.2 for a general discussion of the GDC.

6.2.5.2.6 HVAC Codes

The applicable HVAC codes and standards are shown in Table 9.4-17.

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 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 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 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 containment design features that will reduce the likelihood of combustible gas deflagrations resulting from localized buildup of combustible gases during degraded core accidents are listed in Section 19.3.

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 containment's limiting component is higher than the pressure that 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 flow paths, each with tandem-paired containment isolation valves . One de-inerting flow path receives flow from the suppression pool airspace. For containment overpressure protection during severe accident conditions, only this line will be used. Having the release point in the suppression pool airspace forces evacuated atmpsphere through the suppression pool to scrub out fission products.

6.2.5.4.2 Containment Structural Integrity

See Appendix 19B.

6.2.5.5 Post Accident Radiolytic Oxygen Generation

For a design basis LOCA in the ESBWR, the ADS would depressurize the reactor vessel and the 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 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 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%; and
- 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 10 CFR 50 Appendix J, Option A or Option B as per Regulatory Guide 1.163, and GDC 52, 53 and 54. The leakage rate testing capability is consistent with the testing requirements of ANS-56.8. 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 bellows; and
- 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_a. Pressurization takes approximately 8 hours.
- Pressure Stabilization Phase:
  - 10 CFR 50 Appendix J, Option A - 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.
  - 10 CFR 50 Appendix J, Option B - The containment atmosphere stabilization criteria given in Section 5.6 of ANS 56.8 shall be implemented.
- 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.

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. As an exemption from the definition of L_a in 10 CFR 50 Appendix J, the maximum allowable leakage rate (L_a) is redefined as Containment Leakage Rate given in Table 6.2-1 which excludes the Main Steam Isolation Valve (MSIV) leakage rate. The treatment of MSIV leakage pathway separately in radiological dose analysis in Section 15.4.4.5.2 justifies this exemption..
- 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 $\pm 0.25L_a$. 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 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.
  - For 10 CFR 50 Appendix J Option A, the sum of the local leakage rates and the UCL shall be less than 0.75 L_a. Local leakage rates shall not be subtracted from the Type A test results to determine acceptability of the test.
  - For 10 CFR 50 Appendix J, Option B, the acceptance criteria shall be based on a calculated performance leakage rate which is defined as the sum of Type A UCL and As-left Minimum Pathway Leakage Rate (MNPLR) for all Type B and Type C pathways that were in service, isolated or not lined up in their test position (that is, drained and vented to containment atmosphere) prior to performing the Type A test. In addition, any leakage pathways that were isolated during performance of the test shall be factored into the performance determination. If the leakage can be determined by a local leak rate test, the As-left MNPLR for that leakage path must also be added to the Type A UCL. If the leakage cannot be determined by local leak rate testing, the performance criteria for the Type A test is not met.

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. 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 (for example, 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 intervals as 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, periodic ILRTs will be performed at intervals depending on whether Option A or Option B of 10 CFR 50 Appendix J is selected by the COL Holder. In case Option A is selected, the ILRTs will be performed at least three (3) times during each 10-year service period. In case Option B is selected, the test interval will be as per Regulatory Guide 1.163. 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 10 CFR 50 Appendix J Option A is followed and 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. 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.

If 10 CFR 50 Appendix J Option B is followed and if the ILRT results are not acceptable, then a determination should be performed to identify the cause of unacceptable performance and determine appropriate corrective actions. Once the cause determination and corrective actions have been completed, acceptable performance should be reestablished by performing an ILRT within 48 months following the unsuccessful ILRT test. Following a successful ILRT, the surveillance frequency may be returned to one per 10 years.

The following additional Criteria will be met for Integrated Leakage Rate Tests if 10 CFR 50 Appendix J Option A is implemented:

- 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 rupture as a result of LOCA.
- All systems not designed to remain filled with fluid (for example, 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; and
- 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.

The following additional Criteria will be met for Integrated Leakage Rate Tests if 10 CFR 50 Appendix J Option B is implemented:

All Appendix J pathways must be properly drained and vented during the performance of ILRT, with the following exceptions:

- Pathways in systems that are required for proper conduct of the ILRT or to maintain the plant in a safe shutdown condition during the ILRT;
- Pathways in systems that are normally filled with fluid and operable under post-accident condition;
- Portion of pathways outside primary containment that are designed to Seismic Category I and at least Safety Class 2; and
- For planning and scheduling purpose, or ALARA considerations, pathways, which are Type B or C tested within the previous 24 calendar months, need not be vented or drained during the ILRT.

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_a, 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_a. The rate of decay of pressure of the known 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 or nitrogen, 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.

In accordance with 10 CFR 50 Appendix J, Type B tests are performed at intervals depending on whether Option A or Option B of 10 CFR 50 Appendix J is selected on a unit-specific basis. In case Option A is selected, Type B tests (except for air-locks) will be performed during each reactor shutdown for major fuel reloading, or other convenient intervals, but in no case at intervals greater than two years. Under this option 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. The acceptance criteria for air-lock is a leakage rate of less than or equal to 0.05 L_a, when tested at pressure greater than or equal to P_a. As an exemption from 10 CFR 50 Appendix J, Section III.D.2.(b)(ii), can be satisfied by testing at the end of periods when containment integrity is not required by the plant's Technical Specifications at a lower test pressure specified in the technical specification applied between the door seals with an acceptable maximum measured leakage rate of 0.01 La. Air-locks are tested at initial fuel loading, and at least once every 6 months thereafter. In case Option B is selected, the test interval will be as per Regulatory Guide 1.163. Air-locks that are allowed to be opened during power operation may be tested at power operation so as to avoid shutting down.

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 (if pressure-decay method is used) or flow (if flowmeter method is used) is 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 or flow 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.

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 Option A or Option B. Containment isolation valves subject to Type C tests are listed within Tables 6.2-16 through 6.2-42.

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_a. As an exemption from 10 CFR 50 Appendix J, Section III.D.2.(b)(ii), can be satisfied by testing at the end of periods when containment integrity is not required by the plant's Technical Specifications at a lower test pressure specified in the Technical Specification applied between the door seals with an acceptable maximum measured leakage rate of 0.01 La. The rate of decay of pressure of the known 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 (for valves served by seal system or valves equivalent to a valve served by a seal system) 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_a.

Valves which are sealed with a fluid from a sealed system or valves not provided with a seal system and may be justified to be equivalent to valves with seal system shall be tested in accordance with 10 CFR 50 Appendix J Option A or Option B as given below. A valid justification for equivalency of such valves is that they are in lines designed to be, or remain, filled with water for at least 30 days subsequent to a LOCA.

Option A of Appendix J - Valves sealed with a seal system shall be pressurized with the seal system fluid to a pressure not less than 1.10 P_a. Valves not provided with a seal system, which may be justified to be equivalent to valves with seal system, will be leakage rate tested with water (exemption from 10 CFR 50 Appendix J) to a pressure not less than 1.10 P_a. In both cases the measured leakage may not be converted to equivalent air leakage and may be excluded when determining the combined leakage rate of components subject to Type B and Type C tests provided that:

- Such valves have been demonstrated to have fluid leakage rates that do not exceed those specified in the Technical Specifications or associated bases, and
- For sealed valves, the installed isolation valve seal-water system fluid inventory is sufficient to assume the sealing function for at least 30 days at a pressure of 1.10 P_a.

Option B of Appendix J – As per ANS 56.8 Section 3.4, valves sealed with a fluid from a qualified seal system are not required to be leak tested and the testing of seal system shall be as per Section 3.4 of ANS 56.8. However valves not provided with a seal systems, which have been justified to be equivalent to valves with seal systems, may be leakage rate tested with water to a pressure not less than 1.10 P_a. The measured leakage may not be converted to equivalent air leakage, and may be excluded when determining the combined leakage rate of components subject to Type B and Type C tests provided that:

- Such valves have been demonstrated to have fluid leakage rates that do not exceed those specified in the Technical Specifications or associated bases, and
- For sealed valves the installed isolation valve seal-water system fluid inventory is sufficient to assume the sealing function for at least 30 days at a pressure of 1.10 P_a.

The following exemptions from 10 CFR 50 Appendix J Option A or Option B will be taken for Type C test for MSIVs:

- For Y-globe MSIVs only, testing will be preformed at a pressure less than P_a as specified in Technical Specifications. This is justified because the design of the Y-globe MSIVs is such that the test pressure that is applied between two MSIVs in the same line is in the reverse direction for the upstream MSIV. Normat test pressure tends to unseat the upstream valve disc and results in a meaningless test. Also, leakage testing at a lower differential pressure across the Y-globe MSIV is more severe and conservative than at higher differential pressure. The Y-globe MSIV seat and disc are made of steel for which the lower differential pressure will not have enough force to deform the seat and disc interface to seal off the micro-openings between the two parts. A lower test pressure will drive the air across the opening whereas a higher differential pressure may actually seal some of the leakage paths.
- The measured leakage rate of MSIV in a Type C test will be excluded when determining the combined leakage rate of components subject to Type B and Type C tests. The justification for this exemption from 10 CFR 50 Appendix J requirement is because it is excluded from L_a which is redefined in Subsection 6.2.6.1.1.

All test connections, vent lines, or drain lines consisting of double or multiple barriers (for example, 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 and are 25.4-mm (1-inch) or less in size 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.

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 may be conducted at any time during normal plant operations or during shutdown periods, with test intervals as per Option A or Option B of 10 CFR 50 Appendix J. 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.

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

Deleted.

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

6.2-1-H The COL holder shall provide the missing information indicated in Tables 6.2-16 through 6.2-42. (Subsection 6.2.4.2)

6.2.9 References

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

6.2-3 GE letter from David H. Hinds to U.S. Regulatory Commission, TRACG LOCA SER Confirmatory Items (TAC # MC 8168), Enclosure 2, Reactor pressure Vessel (RPV)

Level Response for the Long Term PCCS Period, Phenomena Identification and Ranking Table, and Major Design Changes from Pre-Application Review Design to DCD Design, MFN 05-105, October 6, 2005

6.2-4 GE letter from David H. Hinds to U.S. Regulatory Commission, Revised Response – GE Response to Results of NRC Acceptance Review for ESBWR Design Certification Application – Item 2, MFN 06-094, March 28, 2006.

6.2-5 Moody, F .J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Trans. ASME, Series C, Vol. 87, P 134, February 1965.

6.2-6 Deleted.

6.2-7 GE-Hitachi Nuclear Energy, “ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis”, NEDO-33338, scheduled September 2007.

Table 6.2-1

Containment Design Parameters

Design Conditions:	
Upper and Lower Drywell	
Design Pressure	310 kPa(g) [45 psig] / 414 kPa [60 psia]
Design Temperature	171°C (340°F)
Internal minus External Differential Pressure	-20.7 kPa(d) [-3.0 psid]
Drywell minus 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 Weight of Containment Free Volume per 24 hours at Pressure 310 kPa(g) [45 psig] and Standard Temperature 20°C (68°F)
Vacuum Breakers Between Drywell and Wetwell	
Number of Vacuum Breakers	Three (3)
Vacuum Breaker Opening Differential Pressure (Wetwell Pressure minus Drywell Pressure)	3.07 kPa [0.445 psi]
Vacuum Breaker Closing Differential Pressure (Wetwell Pressure minus Drywell Pressure)	2.21 kPa [0.320 psi]

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 ft)
GDCS Pools	
Total Water Volume (per pool for pools at 90 and 270 degrees) at Normal water level	560 m^3 (~19776 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)	58.1 m^3 (2051.8 ft^3)
Non-Drainable Water Volume (for pool at 180 degrees)	77.8 m^3 (2747.5 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 Containment-LOCA Performance Analyses

Break Location	Break Size+ m^2 (ft^2)	Single Failure	Maximum DW Pressure* kPa,a (psia)	Maximum DW Pressure* kPa,gauge (psig)	Margin** to Design Pressure of 45.3 psig (%)	Short-term DW Temperature (ºC)	Long-term DW Temperature (ºC)	Long-term WW Temperature (ºC)	Long term Suppression Pool Temperature (ºC)
Based on standard TRACG evaluation model:									
Steam Line Inside Containment+++	0.09832 (1.058)	1 DPV	354.8kPa (51.5psia)	253.4kPa (36.8 psig)	18.9%	175.4°C (347.7°F)	140.2°C (284.4°F)	113.8°C (236.8°F)	75.8°C (168.4°F)
Feedwater Line++	0.07420 (0.7986)	1 DPV	326.0 kPa (47.3 psia)	224.6 kPa (32.6 psig)	28.0%	176.3°C (349.3°F)	149.7°C (301.5°F)	111.1°C (232.0°F)	70.1°C (158.2°F)
GDCS Injection Line	0.004561 (0.04910)	1 DPV	300.1 kPa (43.5 psia)	198.7 kPa (28.8 psig)	36.4%	168.4°C (335.1°F)	133.8°C (272.8°F)	108.0°C (226.3°F)	60.5°C (141.0°F)
Bottom Head Drain Line	0.004052 (0.04361)	1 DPV	303.6 kPa (44.0 psia)	202.2 kPa (29.3 psig)	35. 2%	171. 6°C (340. 9°F)	134. 1°C (273. 3°F)	108.3°C (226. 9°F)	60. 9°C (141. 6°F)
Based on bounding values:									
Steam Line Inside Containment+++	0.09832 (1.058)	1 DPV	384.6 kPa (55.8 psia)	283.3 kPa (41.1 psig)	9.3%	171.3°C (340.3°F)	142.9°C (289.2°F)	116. 3°C (241. 3°F)	76.7°C (170.1°F)
Feedwater Line++	0.07420 (0.7986)	1 DPV	351.7 kPa (51.0 psia)	250.4 kPa (36.3 psig)	19.9%	177.4°C (351. 3°F)	151.7°C (305.1°F)	112.9°C (235.2°F)	70. 4°C (158. 7°F)

+ The break area is from the RPV side of the break.

++ For the feedwater line break, the total break area from the turbine building side is limited at the two parallel ventrui sections, with flow area of 0.04997 m^2 each. The analysis assumes feedwater lines are completely isolated in 52 seconds after the LOCA initiation (isolation valves start to close at 42 s with closure time of 10s).

+++ Main Stream Line Break, at level 34, 2GDCS vent paths.

* Maximum DW pressure calculated during the 72 hours following a LOCA.

** Minimum pressure margin calculated during the 72 hours following a LCOA.

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	43.3°C (110°F)	43.3°C (110°F)
5	DW Pressure	101.3 kPa (14.7 psia)	110.3 kPa (16.0 psia)
6	DW Temperature	46.1°C (115°F)	46.1°C (115°F)
7	WW Pressure	101.3 kPa (14.7 psia)	110.3 kPa (16.0 psia)
8	WW Temperature	43.3°C (110°F)	43.3°C (110°F)
9	Suppression pool Temp.	43.3°C (110°F)	43.3°C (110°F)
10	GDCS pool temperature	46.1°C (115°F)	46.1°C (115°F)
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	7.17 MPa (1040 psia)	7.274 MPa (1055 psia)
15	RPV Water Level	NWL	NWL+0.3m

Table 6.2-6a

Summary of ESBWR TRACG Nodalization Changes

(From the Design in Ref. 6.2-1 to the DCD Design)

Item #	Description	Change	Due to Design Change	Addressing Ref. 6.2-1 SER conditions
1	Core Power	4000 MW to 4500 MW	✓	
2	Number of bundles	1020 to 1132	✓	
3	Core shroud OD	+ 0.328 m	✓	
4	Number of CRDs	121 to 269	✓	
5	GDCS pool and air space location	Connection changed from WW to DW; Eliminated the GDCS air space vent pipes to WW	✓	✓
6	GDCS pool air space and DW connection	For bounding calculation, two pipes are used to simulate the connection between the GDCS pool air space and the DW, to purge residual noncondensable gases in this air space.	✓	
7	Total PCCS capacity	4x13.5 MW to 6x11 MW	✓	
8	Total IC capacity	4x30 MW to 4x33.75 MW	✓	
9	Pressure relief system	12 ADS valves to 10 ADS valves + 8 S/R	✓	
10	Containment vents	10 to 12	✓	
11	Spill-over connection (DW annulus to vertical vent module)	Changed from 10 horizontal holes to 12 horizontal holes; hole inlet elevation raised to approximately 2 m above the suppression pool normal water level	✓	
12	SLCS activated on ADS	Yes for the DCD design	✓	
13	Credit for water added by HCUs during scram	Yes for the DCD design	✓	
14	Credit for IC inventory for RPV analysis	Yes for the DCD design	✓	

Table 6.2-6a

Summary of ESBWR TRACG Nodalization Changes

(From the Design in Ref. 6.2-1 to the DCD Design)

Item #	Description	Change	Due to Design Change	Addressing Ref. 6.2-1 SER conditions
15	Integrated TRACG input deck	Combined the RPV and containment input decks into one consistent, detailed deck		✓
16	Chimney modeling	Changed from 1 chimney in containment P&T nodalization and 3 chimneys in RPV water level nodalization, to a total of 5 chimneys in the common nodalization, to calculate the minimum water level; Two of these simulate the individual chimney partition		✓
17	Air gap and reactor shield wall	Modeled in the combined nodalization		✓
18	Lower drywell	Changed the modeling from TEE component cells to VSSL component cells (improved nodalization)		✓
19	Drywell head air space	Modeled in the combined nodalization (improved nodalization)		✓
20	Top of wetwell airspace	One additional axial level added at an elevation near the top of wetwell (improved nodalization)		✓
21	IC/PCC pools	Separate regions to model the expansion pool and the dryer/separator storage pool.	✓	
22	FW system	Included FW mass outside of containment in LOCA evaluation model of FW line break.		✓
23	PCCS condensate return	PCCS condensate drains directly to the GDCS pools in the DCD design; the PCCS drain tanks are eliminated.	✓	

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. Loss of power generation bus initiates signals for scram and IC.
<1	High Drywell pressure setpoint reached, Scram signal from high drywell pressure is not credited in this analysis.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion started at 0.25 second later.
Vent clearing time	Top Vent: 1.8 sec, Middle Vent: 2.3 sec, Bottom Vent: 3.0 sec.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves start to open at 15 seconds later.
4	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
12	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
13	Level 2 is reached (Level 2 MSIV closure and IC initiation signals are not credited in this analysis).
16	Reactor isolated on low MSL pressure setpoint.
263	Level 1 is reached.
273	Level 1 signal confirmed; ADS/GDCS/SLCS timer initiated; SRV actuated.
323	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLCS flow starts on DPV actuation.
423	GDCS timer (150 seconds after confirmed Level 1 signal) timed out. GDCS injection valves open.
450	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
623	SLCS flow depleted.
135827 (~ 37.7 hrs)	PCC pool drops below the elevation of 29.6 m; top ¼ portion of the PCC tube length becomes uncovered; connection valves open to allow the water from the Dryer/Separator storage pool to flow into the IC/PCC expansion pools.

Table 6.2-7

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

Time (sec)	**Events**
From ~450 to 259000 (72 hrs)	RPV water level remains higher than Level 0.5. Therefore, equalizing line valves do not open for this event.
~259000 (72 hrs)	DW pressure rises to 326.0 kPa (47.3 psia).

Table 6.2-7a

Operational Sequence of ECCS for a Main Steam Line Break with Failure of One DPV (Nominal Case)

Time (sec)	**Events**
0	Guillotine break of main steam line inside containment; normal auxiliary power assumed to be lost; feedwater is lost. Loss of power generation bus initiates signals for scram and IC.
<1	High Drywell pressure setpoint reached, scram signal from high drywell pressure is not credited in this analysis.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion starts 0.25 second later.
Vent clearing time	Top Vent: 1.7 sec, Middle Vent: 2.1 sec, Bottom Vent: 2.8 sec.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves start to open 15 seconds later.
6	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
8	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
10	Reactor isolated on low MSL pressure setpoint.
17	Level 2 is reached (Level 2 MSIV closure and IC initiation signals are not credited in this analysis).
462	Level 1 is reached.
472	Level 1 signal confirmed; ADS/GDCS/SLCS timer initiated; SRV actuated.
522	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLCS flow starts on DPV actuation.
622	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDCS injection valves open.
644	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
822	SLCS flow depleted.
140195 (~ 38.9 hrs)	PCC pool drops below the elevation of 29.6 m; top ¼ portion of the PCC tube length becomes uncovered; connection valves open to allow the water from the Dryer/Separator storage pool to flow into the IC/PCC expansion pools.

Table 6.2-7a

Operational Sequence of ECCS for a Main Steam Line Break with Failure of One DPV

(Nominal Case)

Time (sec)	**Events**
From ~450 to 259000 (72 hrs)	RPV water level remains higher than Level 0.5. Therefore, equalizing line valves do not open for this event.
~259000 (72 hrs)	DW pressure rises to 354.8 kPa (51.5 psia).

Table 6.2-7b

Operational Sequence of ECCS for a GDCS Line Break with Failure of One DPV (Nominal Case)

Time (sec)	**Events**
0	Guillotine break of GDCS line inside containment; normal auxiliary power assumed to be lost; feedwater is lost. Loss of power generation bus initiates signals for scram and IC.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion started at 0.25 second later.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves started to open at 15 seconds later.
4	High Drywell pressure setpoint reached, Scram signal from high drywell pressure is not credited in this analysis.
6	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
Vent clearing time	Top Vent: 9.4 sec, Middle Vent: 209 sec, Bottom Vent: never cleared.
15	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
16	Level 2 is reached (Level 2 MSIV closure and IC initiation signals are not credited in this analysis).
19	Reactor isolated on low MSL pressure set point.
144	Level 1 is reached.
154	Level 1 signal confirmed; ADS/GDCS/SLCS timer initiated; SRV actuated.
204	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLCS flow starts on DPV actuation.

Table 6.2-7b

Operational Sequence of ECCS for a GDCS Line Break with Failure of One DPV

(Nominal Case)

Time (sec)	**Events**
304	GDCS timer (150 seconds after confirmed Level 1 signal) timed out. GDCS injection valves open.
436	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
504	SLCS flow depleted.
133380 (~ 37.05 hrs)	PCCS pool drops below the elevation of 29.6 m; top ¼ portion of the PCCS tube length becomes uncovered; connection valves open to allow the water from the Dryer/Separator storage pool to flow into the PCCS pools.
From ~800 to 259000 (72 hrs)	RPV water level remains higher than Level 0.5. Therefore equalizing line valves do not open for this event.
~259000 (72 hrs)	DW pressure rises to 300.1 kPa (43.5 psia).

Table 6.2-7c

Operational Sequence of ECCS for a Bottom Drain Line Break with Failure of One DPV (Nominal Case)

Time (sec)	**Events**
0	Guillotine break of bottom drain line inside containment; normal auxiliary power assumed to be lost; feedwater is lost. Loss of power generation bus initiates signals for scram and IC.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion started at 0.25 second later.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves started to open at 15 seconds later.
5	High Drywell pressure setpoint reached, Scram signal from high drywell pressure is not credited in this analysis.
6	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
Vent clearing time	Top Vent: 349 sec, Middle Vent: never cleared, Bottom Vent: never cleared.
15	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
16	Level 2 is reached (Level 2 MSIV closure and IC initiation signals are not credited in this analysis).
19	Reactor isolated on low MSL pressure set point.
285	Level 1 is reached.
295	Level 1 signal confirmed; ADS/GDCS/SLCS timer initiated; SRV actuated.
345	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLCS flow starts on DPV actuation.
445	GDCS timer (150 seconds after confirmed Level 1 signal) timed out. GDCS injection valves open.

Table 6.2-7c

Operational Sequence of ECCS for a Bottom Drain Line Break with Failure of One DPV (Nominal Case)

Time (sec)	**Events**
577	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
647	SLCS flow depleted.
136484 (~ 37.9 hrs)	PCCS pool drops below the elevation of 29.6 m; top ¼ portion of the PCCS tube length becomes uncovered; connection valves open to allow the water from the Dryer/Separator storage pool to flow into the PCCS pools.
From ~800 to 259000 (72 hrs)	RPV water level remains higher than Level 0.5. Therefore equalizing line valves do not open for this event.
~259000 (72 hrs)	DW pressure rises to 303.6 kPa (44.0 psia).

Table 6.2-7d

Operational Sequence of ECCS for a Feedwater Line Break with Failure of One DPV (Bounding Case)

Time (sec)	**Events**
0	Guillotine break of feedwater line inside containment; normal auxiliary power assumed to be lost; feedwater is lost. Loss of power generation bus initiates signals for scram and IC.
<1	High Drywell pressure setpoint reached, scram signal from high drywell pressure is not credited in this analysis.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion starts 0.25 second later.
Vent clearing time	Top Vent: 2.0 sec, Middle Vent: 2.4 sec, Bottom Vent: 3.3 sec.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves starts to open 15 seconds later.
4	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
13	Level 2 is reached (Level 2 MSIV closure and IC initiation signals are not credited in this analysis).
13	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
17	Reactor isolated on low MSL pressure setpoint.
284	Level 1 is reached.
294	Level 1 signal confirmed; ADS/GDCS/SLCS timer initiated; SRV actuated.
344	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLCS flow starts on DPV actuation.
444	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDCS injection valves open.
507	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
644	SLCS flow depleted.

Table 6.2-7d
Operational Sequence of ECCS for a Feedwater Line Break with Failure of One DPV (Bounding Case)

Time (sec)	**Events**
122913 (~ 34.1 hrs)	PCC pool drops below the elevation of 29.6 m; top ¼ portion of the PCC tube length becomes uncovered; connection valves open to allow the water from the Dryer/Separator storage pool to flow into the IC/PCC expansion pools.
From ~450 to 259000 (72 hrs)	RPV water level remains higher than Level 0.5. Therefore, equalizing line valves do not open for this event.
~259000 (72 hrs)	DW pressure rises to 351.7 kPa (51.0 psia).

Table 6.2-7e

Operational Sequence of ECCS for a Main Steam Line Break with Failure of one DPV (Bounding Case)

Time (sec)	**Events**
0	Guillotine break of main steam line inside containment; normal auxiliary power assumed to be lost; feedwater is lost. Loss of power generation bus initiates signals for scram and IC.
<1	High Drywell pressure setpoint reached, scram signal from high drywell pressure is not credited in this analysis.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion starts 0.25 second later.
Vent clearing time	Top Vent: 1.9 sec, Middle Vent: 2.3 sec, Bottom Vent: 3.0 sec.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves start to open 15 seconds later.
7	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
8	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
11	Reactor isolated on low MSL pressure setpoint.
19	Level 2 is reached (Level 2 MSIV closure and IC initiation signals are not credited in this analysis).
499	Level 1 is reached.
509	Level 1 signal confirmed; ADS/GDCS/SLCS timer initiated; SRV actuated.
559	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLCS flow starts on DPV actuation.
659	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDCS injection valves open.
722	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
860	SLCS flow depleted.
125714 (~ 34.9 hrs)	PCC pool drops below the elevation of 29.6 m; top ¼ portion of the PCC tube length becomes uncovered; connection valves open to allow the water from the Dryer/Separator storage pool to flow into the IC/PCC expansion pools.

Table 6.2-7e

Operational Sequence of ECCS for a Main Steam Line Break with Failure of one DPV (Bounding Case)

Time (sec)	**Events**
From ~450 to 259000 (72 hrs)	RPV water level remains higher than Level 0.5. Therefore, equalizing line valves do not open for this event.
~259000 (72 hrs)	DW pressure rises to 384.6 kPa (55.8 psia).

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}

* In Reference 6.2-1, the limiting case is the Main Steam Line Break and the value for PIRT84 is –2 sigma (or 0.81).

** Free surface to vapor heat transfer in wetwell.

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 Depressurization 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 configuration	Prevention /Mitigation	Increases drywell 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
5	Break in RWCU/SDC Filter/Demineralizer Room

Table 6.2-12

Subcompartment Vent Path Designation

Flow Path No.	Type	Cell From	Cell To	P (m)	DH (m)	L/DH	T	K FORW	K REVER	K AVERA	K CONTAIN	Flow Condition	Flow Direction	Blow Out Pressure (k Pa)	Comments
1	DOOR	1	2	8.00	2.00	1.00	0.24	1.56	1.61	1.58	0.79	SUBSONIC	BOTH	NO	TWO WAY PATH
2	DOOR	2	3	8.00	2.00	0.50	0.97	1.51	1.24	1.38	0.69	SUBSONIC	FORWARD	10.34	
3	DOOR	2	3	8.00	2.00	0.50	0.97	1.52	1.26	1.39	0.70	SUBSONIC	FORWARD	10.34	
4	DOOR	3	4	8.00	2.00	0.35	1.13	1.25	1.24	1.24	0.62	SUBSONIC	FORWARD	10.34	
5	DOOR	3	5	8.00	2.00	0.25	1.19	1.31	1.32	1.31	0.66	SUBSONIC	FORWARD	10.34	
6	DOOR	6	7	8.00	2.00	1.00	0.24	1.56	1.61	1.58	0.79	SUBSONIC	FORWARD	NO	TWO WAY PATH
7	DOOR	7	5	8.00	2.00	0.50	0.97	1.52	1.26	1.39	0.70	SUBSONIC	FORWARD	10.34	
8	DOOR	7	5	8.00	2.00	0.50	0.97	1.51	1.24	1.38	0.69	SUBSONIC	FORWARD	10.34	
9	DOOR	8	4	8.00	2.00	1.00	0.24	1.43	1.47	1.45	0.72	SUBSONIC	FORWARD	10.34	
10	DOOR	9	10	8.00	2.00	1.00	0.24	1.49	1.48	1.49	0.74	SUBSONIC	FORWARD	10.34	
11	DOOR	10	5	8.00	2.00	0.35	1.13	1.25	1.24	1.24	0.62	SUBSONIC	FORWARD	10.34	
12	DOOR	10	4	8.00	2.00	0.25	1.19	1.24	1.24	1.24	0.62	SUBSONIC	FORWARD	10.34	
13	DELETED														
14	OPEN SPACE	12	16	10.00	2.00	0.50	0.97	0.90	0.47	0.69	0.34	SUBSONIC	BOTH	NO	TWO WAY PATH
15	OPEN SPACE	13	16	10.00	2.00	0.50	0.97	0.90	0.47	0.69	0.34	SUBSONIC	BOTH	NO	TWO WAY PATH
16	OPEN SPACE	14	16	10.00	2.00	0.50	0.97	0.93	0.48	0.71	0.35	SUBSONIC	BOTH	NO	TWO WAY PATH
17	OPEN SPACE	15	16	10.00	2.00	0.50	0.97	0.93	0.48	0.70	0.35	SUBSONIC	BOTH	NO	TWO WAY PATH
18	OPEN SPACE	3	12	10.00	2.00	0.50	0.97	0.47	0.90	0.69	0.34	SUBSONIC	BOTH	NO	TWO WAY PATH

Table 6.2-12

Subcompartment Vent Path Designation

Flow Path No.	Type	Cell From	Cell To	P (m)	DH (m)	L/DH	T	K FORW	K REVER	K AVERA	K CONTAIN	Flow Condition	Flow Direction	Blow Out Pressure (k Pa)	Comments
19	OPEN SPACE	5	14	10.00	2.00	0.50	0.97	0.48	0.93	0.71	0.35	SUBSONIC	BOTH	NO	TWO WAY PATH
20	OPEN SPACE	10	15	10.00	2.00	0.50	0.97	0.48	0.93	0.70	0.35	SUBSONIC	BOTH	NO	TWO WAY PATH
21	OPEN SPACE	4	13	10.00	2.00	0.50	0.97	0.47	0.90	0.69	0.34	SUBSONIC	BOTH	NO	TWO WAY PATH
22	BLOW-OUT PANEL	16	17	16.00	4.00	0.50	0.97	2.46	2.44	2.45	1.23	SUBSONIC	FORWARD	10.34	TO ATMOSPHERE
23	HATCH	11	18	13.60	3.40	0.29	1.16	2.09	1.41	1.75	0.87	SUBSONIC	FORWARD	10.34	
24	DOOR	18	19	8.00	2.00	0.30	1.16	1.97	1.97	1.97	0.98	SUBSONIC	FORWARD	10.34	
25	DOOR	19	20	8.00	2.00	0.15	1.25	2.07	2.07	2.07	1.04	SUBSONIC	FORWARD	10.34	
26	DOOR	20	5	8.00	2.00	1.00	0.24	1.22	0.82	1.02	0.51	SUBSONIC	FORWARD	10.34	
27	DOOR	20	21	8.00	2.00	0.15	1.25	2.07	2.07	2.07	1.04	SUBSONIC	FORWARD	10.34	
28	DOOR	21	22	8.00	2.00	0.30	1.16	1.97	1.97	1.97	0.98	SUBSONIC	FORWARD	10.34	
29	DOOR	18	22	8.00	2.00	0.15	1.25	2.07	2.07	2.07	1.04	SUBSONIC	FORWARD	10.34	
30	DOOR	22	23	8.00	2.00	1.00	0.24	1.49	1.63	1.56	0.78	SUBSONIC	FORWARD	10.34	
31	DOOR	23	24	8.00	2.00	1.00	0.24	1.59	1.48	1.53	0.77	SUBSONIC	FORWARD	10.34	

Table 6.2-12 (Continued)

Subcompartment Vent Path Designation

Flow Path No.	Cell From	Volume (m^3)	Cell To	Volume (m^3)	F1 (m^2)	L1 (m)	F0 (m^2)	Lp (m).	F2 (m^2)	L2 (m)	A/L per ANSI/ANS-56.10-1982	A/L per SMSAB-02-04	EL. In (m)	EL. Out (m)
1	1	348	2	272	106.03	2.00	4.00	2.00	54.38	2.50	1.77	0.62	-10.50	-10.50
2	2	271	3	334	9.64	5.00	4.00	1.00	19.37	5.20	0.96	0.62	-10.50	-10.50
3	2	271	3	334	9.84	5.50	4.00	1.00	19.37	3.20	1.03	0.62	-10.50	-10.50
4	3	334	4	472	10.40	11.00	4.00	0.70	10.63	11.00	0.44	0.58	-10.50	-10.50
5	3	334	5	342	10.97	9.50	4.00	0.50	10.73	12.00	0.47	0.58	-10.50	-10.50
6	6	353	7	271	106.03	2.00	4.00	2.00	54.38	2.50	1.77	0.62	-10.50	-10.50
7	7	271	5	342	9.84	5.50	4.00	1.00	19.37	5.00	0.94	0.62	-10.50	-10.50
8	7	271	5	342	9.64	5.50	4.00	1.00	19.37	4.00	0.97	0.62	-10.50	-10.50
9	8	151	4	472	43.79	1.00	4.00	2.00	32.92	2.50	1.67	0.75	-10.50	-10.50
10	9	151	10	519	42.50	1.00	4.00	2.00	45.79	2.50	1.73	0.75	-10.50	-10.50
11	10	519	5	342	10.40	11.00	4.00	0.70	10.63	9.50	0.47	0.57	-10.50	-10.50
12	10	519	4	472	10.25	11.50	4.00	0.50	10.25	11.50	0.42	0.51	-10.50	-10.50
13	DELETED													
14	12	197	16	26163	5.00	19.50	5.00	1.00	95.86	2.00	0.24	0.86	33.00	34.00
15	13	197	16	26163	5.00	19.50	5.00	1.00	98.13	2.00	0.24	0.86	33.00	34.00
16	14	197	16	26163	5.00	19.50	5.00	1.00	148.93	2.00	0.24	0.86	33.00	34.00
17	15	197	16	26163	5.00	19.50	5.00	1.00	135.41	2.00	0.24	0.86	33.00	34.00
18	3	334	12	197	95.86	2.00	5.00	1.00	5.00	19.50	0.24	0.86	-7.40	-6.40
19	5	342	14	197	148.93	2.00	5.00	1.00	5.00	19.50	0.24	0.86	-7.40	-6.40

Table 6.2-12 (Continued)

Subcompartment Vent Path Designation

20	10	519	15	197	135.41	2.00	5.00	1.00	5.00	19.50	0.24	0.86	-7.40	-6.40
21	4	472	13	197	98.13	2.00	5.00	1.00	5.00	19.50	0.24	0.86	-7.40	-6.40
22	16	26163	17	1.00E+08	1739.00	9.00	16.00	1.00	99999.00	100.00	14.56	0.54	51.70	51.70
23	11	94	18	458	25.00	2.20	11.56	1.00	1078.65	1.80	5.68	2.55	-2.00	-1.00
24	18	458	19	458	23.00	12.00	4.00	0.60	23.00	12.00	0.84	0.52	0.00	0.00
25	19	458	20	153	24.27	8.00	4.00	0.30	24.27	1.10	2.22	0.75	0.00	0.00
26	20	153	5	342	8.00	2.80	4.00	2.00	42.22	1.25	1.14	0.75	-10.50	-10.50
27	20	153	21	458	24.27	1.10	4.00	0.30	24.27	8.00	2.22	0.75	0.00	0.00
28	21	458	22	458	23.00	12.00	4.00	0.60	23.00	12.00	0.84	0.52	0.00	0.00
29	18	458	22	458	24.27	8.00	4.00	0.30	24.27	8.00	1.36	0.52	0.00	0.00
30	22	458	23	122	206.40	3.30	4.00	2.00	35.80	1.25	1.82	0.81	0.00	0.00
31	23	122	24	29000	35.80	1.25	4.00	2.00	107.00	3.85	1.75	0.81	0.00	0.00

NOTES:
F0 – Path Area
F1 - Cross-Section Area of From Cell
F2 - Cross-Section Area of To Cell
P – Path Perimeter
Lp – Passage Length
DH - Hydraulic Diameter
T - Flow Coefficient per Diagram 4-11 of Idel'Chik Handbook
KFORWARD - Direct Loss Pressure Coefficient per Diagram 4-11 of Idel'Chik Handbook
KREVERSE - Inverse Loss Pressure Coefficient per Diagram 4-11 of Idel'Chik Handbook
KAVERAGE - (KFORWARD+KREVERSE)/2
KCONTAIN - KAVERAGED/2
Inertia Term - A/L per ANSI/ANS-56.10-1982 has been used in the pressurization analyses.
Water entrainment - Dropout is activated since volumetric power is less than 5 MW/m^3 (according to Section 4.2.1 of SMSAB-02-04, CONTAIN Code Qualification Report/User Guide for Auditing Subcompartment Analysis Calculations.

Table 6.2-12a

Subcompartment Nodal Description

Figure	Cell Number	Postulated Break (See Table 6.2-11 for Break Case Description)	Description	Room No.	Net Volume (m^3)	Calculated Peak Pressure (kPa g) (1)	Initial Conditions		
							Pressure (Pa a)	Temperature (ºC)	Relative Humidity (%)
6.2-18	1	CASE 1 CASE 3	RWCU /Shutdown Cooling Heat Exchanger Room A	1151	348	25.2	1.013e5	43	0
6.2-18	2	CASE 2	RWCU /Shutdown Cooling Valve Room A	1150	271	15.3	1.013e5	43	0
6.2-18	3	NO	Corridor A El. -11500 mm	1100	334	13.5	1.013e5	43	0
6.2-18	4	NO	Corridor B El. -11500 mm	1101	472	12.4	1.013e5	43	0
6.2-18	5	NO	Corridor D El. -11500 mm	1103	342	21.7	1.013e5	43	0
6.2-18	6	CASE 1 CASE 3	RWCU /Shutdown Cooling Heat Exchanger Room B	1161	353	32.6	1.013e5	43	0
6.2-18	7	CASE 2	RWCU /Shutdown Cooling Valve Room B	1160	271	24.2	1.013e5	43	0
6.2-18	8	CASE 4	RWCU /Shutdown Cooling Pump Room A	1152	151	12.5	1.013e5	43	0
6.2-18	9	CASE 4	RWCU /Shutdown Cooling Pump Room B	1162	151	12.6	1.013e5	43	0
6.2-18	10	NO	Corridor C El. -11500 mm	1102	519	12.4	1.013e5	43	0
6.2-18	11	CASE 5	RWCU /Shutdown Cooling Filter/Demin. Vault A1	1251	94	13.7	1.013e5	43	0
6.2-18	12	NO	Non-Divisional Commodity Chase A	1293	197	12	1.013e5	43	0
6.2-18	13	NO	Non-Divisional Commodity Chase B	1294	197	11.9	1.013e5	43	0
6.2-18	14	NO	Non-Divisional Commodity Chase D	1296	197	14.2	1.013e5	43	0

Table 6.2-12a

Subcompartment Nodal Description

Figure	Cell Number	Postulated Break (See Table 6.2-11 for Break Case Description)	Description	Room No.	Net Volume (m^3)	Calculated Peak Pressure (kPa g) (1)	Initial Conditions		
							Pressure (Pa a)	Temperature (°C)	Relative Humidity (%)
6.2-18	15	NO	Non-Divisional Commodity Chase C	1295	197	12	1.013e5	43	0
6.2-18	16	NO	UPPER PLENUM	-----	26163	11.4	1.013e5	43	0
6.2-18	17	NO	Atmosphere	-----	1.0E8	-----	1.013e5	40	0
6.2-18	18	NO	Filter/Demin Access Room	1306	458	13.2	1.013e5	43	0
6.2-18	19	NO	RWCU /Shutdown Cooling Heat Exchanger Access Room A	1304	458	12.6	1.013e5	43	0
6.2-18	20	NO	Interior Stairwell A	1195	153	12.2	1.013e5	43	0
6.2-18	21	NO	RWCU /Shutdown Cooling Heat Exchanger Access Room B	1305	458	12.2	1.013e5	43	0
6.2-18	22	NO	Control Rod Drive Pump Access Room	1307	458	12.9	1.013e5	43	0
6.2-18	23	NO	Controlled Equipment Removal Access Room	1308	122	12.6	1.013e5	43	0
6.2-18	24	NO	FUEL BUILDING	-----	29000	12.4	1.013e5	43	0

(1) Includes a 10% margin

Table 6.2-12b

Mass and Energy Release Rate

TIME (sec)	MASS FLOW (kg/sec)	ENTHALPY (kJ/kg)	ENERGY (W)
Break Case 1- RWCU/SDC System Break in NRHX Room.			
0	782.6	1224.41	9.58E+08
5.14	782.6	1224.41	9.58E+08
5.14	617.2	1224.41	7.55E+08
7.83	617.2	1224.41	7.55E+08
7.83	337.7	976.69	3.29E+08
42.7	337.7	976.69	3.29E+08
42.7	337.7	888.76	3.0E+08
53.61	337.7	888.76	3.0E+08
53.61	225.9	1224.41	2.77E+08
64.05	225.9	1224.41	2.77E+08
76	0	1224.41	0.0
Break Case 2- RWCU/SDC System Break in NRHX Valve Room.			
0	503.1	1058.10	5.32E+08
12.97	503.1	1058.10	5.32E+08
12.97	204.4	815.03	1.67E+08
60.29	204.4	815.03	1.67E+08
60.29	92.6	1224.41	1.13E+08
64.05	92.6	1224.41	1.13E+08
76	0.0	1224.41	0
Break Case 3- RWCU/SDC System Break in RHX Room			
0	782.6	1224.41	9.58E+08
5.14	782.6	1224.41	9.58E+08
5.14	617.2	1224.41	7.56E+08
7.83	617.2	1224.41	7.56E+08
7.83	337.7	976.69	3.3E+08
42.7	337.7	976.69	3.3E+08
42.7	337.7	888.76	3E+08

Table 6.2-12b

Mass and Energy Release Rate

TIME (sec)	MASS FLOW (kg/sec)	ENTHALPY (kJ/kg)	ENERGY (W)
53.61	337.7	888.76	3E+08
53.61	225.9	1224.41	2.77E+08
64.05	225.9	1224.41	2.77E+08
76	0	1224.41	0
Break Case 4- RWCU/SDC System Break in RWCU/SDC Pump Rooms.			
0	223.6	1224.41	2.74E+08
17.02	223.6	1224.41	2.74E+08
17.02	111.8	1224.41	1.37E+08
34.69	111.8	1224.41	1.37E+08
34.69	111.8	1224.41	1.37E+08
36.77	111.8	1224.41	1.37E+08
36.77	391.3	1224.41	4.79E+08
49.73	391.3	1224.41	4.79E+08
49.73	68.5	1224.41	8.39E+07
64.05	68.5	1224.41	8.39E+07
76	0	1224.41	0.00E+00
Break Case 5- RWCU/SDC System Break in Filter/Demineralizer Room			
0	503.1	1167.42	5.87E+08
9.9	503.1	1167.42	5.87E+08
30.55	503.1	1167.42	5.87E+08
30.55	180.3	1065.54	1.92E+08
64.05	180.3	1065.54	1.92E+08
64.05	180.3	1065.54	1.92E+08
76	111.8	968.31	1.08E+08
136.42	111.8	968.31	1.08E+08
136.42	0	0	0

Table 6.2-12c

Heat Sink Descriptions

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (ºC)
1	1	roof of 600 mm	roof1	roof	slab	77.33	0.6	concrete	43
	2	M1 wall of 2000 mm (25º)	S1-1	wall	slab	37.31	1	concrete	43
	3	M2 wall of 2000 mm (47º)	S2-1	wall	slab	70.18	1	concrete	43
	4	M3 of 1400 mm	S3-1	wall	slab	18.90	1.4	concrete	43
	5	M4 of 1400 mm	S4-1	wall	slab	20.70	1.4	concrete	43
	6	M5 Containment of 1400 mm	S5-1	wall	slab	75.57	1.4	concrete	43
	7	floors CELL 1	floor1	floor	slab	101.82	4	concrete	43
2	1	roof of 1000 mm	roof2	roof	slab	77.74	1	concrete	43
	2	wall of 1000 mm common with cell 3	S1-2	wall	slab	90.83	0.5	concrete	43
	3	wall of 2.0 m common with V1 (arc of 26º)	S2-2	wall	slab	38.80	1	concrete	43
	4	floors CELL 2	floor2	floor	slab	77.74	4	concrete	43
3	1	roof of 1000 mm	roof3	roof	slab	95.87	1	concrete	43
	2	wall of 2000 mm with outside	S1-3	wall	slab	191.70	2	concrete	43
	3	wall of 1000 mm common with cell 2	S2-3	wall	slab	102.96	0.5	concrete	43
	4	wall of 2.0 m common with V1 (arc of sum 39º)	S3-3	wall	slab	53.02	1	concrete	43

Table 6.2-12c

Heat Sink Descriptions

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (ºC)
	5	floors CELL 3	floor3	floor	slab	95.87	4	concrete	43
4	1	roof of 1000 mm	roof4	roof	slab	121.16	1	concrete	43
	2	wall of 2000 mm with outside	S1-4	wall	slab	74.70	2	concrete	43
	3	wall of 2000 mm with fuel building	S2-4	wall	slab	86.10	2	concrete	43
	4	wall of 2000 mm common with cell 8	S3-4	wall	slab	27.19	1	concrete	43
	5	wall of 2000 mm common with sump pumps	S4-4	wall	slab	33.99	2	concrete	43
	6	wall of 750 mm common with valve room	S5-4	wall	slab	81.16	0.75	concrete	43
	7	floors CELL 4	floor4	floor	slab	135.41	4	concrete	43
5	1	roof of 1000 mm	roof5	roof	slab	83.64	1	concrete	43
	2	wall of 2000 mm with outside	S1-5	wall	slab	181.31	2	concrete	43
	3	wall of 1000 mm with cell 7	S2-5	wall	slab	103.79	0.5	concrete	43
	4	wall of 2000 mm common with cell 6	S3-5	wall	slab	53.02	1	concrete	43
	5	Floors CELL 5	floor5	wall	slab	98.13	4	concrete	43
6	1	roof of 600 mm	roof6	roof	slab	67.69	0.6	concrete	43
	2	M1 wall of 2000 mm (arc of 25º)	S1-1	wall	slab	107.44	1	concrete	43

Table 6.2-12c

Heat Sink Descriptions

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (ºC)
	3	M2 wall of 900 mm	S2-6	wall	slab	18.90	0.9	concrete	43
	4	M3 of 1000 mm	S3-6	wall	slab	20.75	1	concrete	43
	5	M5 CONTAINMENT of 1400 mm	S4-6	wall	slab	83.06	1.4	concrete	43
	6	floors CELL 6	floor6	floor	slab	92.17	4	concrete	43
7	1	roof of 1000 mm	roof7	roof	slab	77.74	1	concrete	43
	2	wall of 1000 mm common with cell 5	S1-7	wall	slab	90.83	0.5	concrete	43
	3	wall of 2.0 m common with cell 2 (arc of 26º)	S2-7	wall	slab	38.80	1	concrete	43
	4	Floors Cell 7	floor7	floor	slab	77.74	4	concrete	43
8	1	roof of 1000 mm	roof8	roof	slab	43.43	1	concrete	43
	2	wall of 250 mm	S1-8	wall	slab	43.13	0.25	concrete	43
	3	wall of 2.0 m common with V5 (arc of 20º)	S2-8	wall	slab	27.19	1	concrete	43
	4	wall of 2000 mm with internal room (arc of 11º)	S3-8	wall	slab	14.95	2	concrete	43
	5	Containment of 600 mm	S4-8	wall	slab	34.38	0.6	concrete	43
	6	floors CELL 8	floor8	floor	slab	43.43	4	concrete	43
9	1	roof of 1000 mm	roof9	roof	slab	43.43	1	concrete	43

Table 6.2-12c

Heat Sink Descriptions

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (ºC)
	2	wall of 250 mm	S1-9	wall	slab	43.13	0.25	concrete	43
	3	wall of 2.0 m common with cell 5 (arc of 20º)	S2-9	wall	slab	27.19	1	concrete	43
	4	wall of 2000 mm with internal room (arc of 11º)	S3-9	wall	slab	14.95	2	concrete	43
	5	CONTAINMENT WALL of 600 mm	S4-9	wall	slab	34.38	0.6	concrete	43
	6	floors CELL 9	floor9	floor	slab	43.43	4	concrete	43
10	1	roof of 1000 mm	roof10	roof	slab	148.93	1	concrete	43
	2	wall of 2000 mm with outside and stairs	S1-10	wall	slab	93.19	2	concrete	43
	3	wall of 2000 mm with fuel building	S2-10	wall	slab	77.70	2	concrete	43
	4	wall of 2000 mm common with cell 9/11	S3-10	wall	slab	57.10	1	concrete	43
	5	wall of 750 mm common with valve room	S4-10	wall	slab	82.44	0.75	concrete	43
	6	Floors CELL 10	floor10	floor	slab	148.93	4	concrete	43
11	1	M1 wall of 2000 mm	S1-11	wall	slab	28.48	2	concrete	43
	2	M2 wall of 700 mm	S2-11	wall	slab	19.36	0.7	concrete	43
	3	M3 wall of 1500 mm	S3-11	wall	slab	21.51	1.5	concrete	43
	4	M4 of 1000 mm	S4-11	wall	slab	19.36	1	concrete	43

Table 6.2-12c

Heat Sink Descriptions

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (°C)
	5	floors CELL 11	floor11	floor	slab	25.00	1	concrete	43
12	1	wall of 2000 mm with outside	S1-16	wall	slab	85.64	2	concrete	43
	2	wall of 1500 mm with outside	S2-16	wall	slab	195.54	1.5	concrete	43
	3	wall of 1000 mm with outside	S3-16	wall	slab	48.00	1	concrete	43
13	1	wall of 2000 mm with outside	S1-13	wall	slab	85.64	2	concrete	43
	2	wall of 1500 mm with outside	S2-13	wall	slab	195.54	1.5	concrete	43
	3	wall of 1000 mm with outside	S3-13	wall	slab	48.00	1	concrete	43
14	1	wall of 2000 mm with outside	S1-14	wall	slab	85.64	2	concrete	43
	2	wall of 1500 mm with outside	S2-14	wall	slab	195.54	1.5	concrete	43
	3	wall of 1000 mm with outside	S3-14	wall	slab	48.00	1	concrete	43
15	1	wall of 2000 mm with outside	S1-15	wall	slab	85.64	2	concrete	43
	2	wall of 1500 mm with outside	S2-15	wall	slab	195.54	1.5	concrete	43
	3	wall of 1000 mm with outside	S3-15	wall	slab	48.00	1	concrete	43
16	No heat sinks								
17	No heat sinks								

Table 6.2-12c

Heat Sink Descriptions

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (°C)
18	1	roof of 2000 mm	roof18	roof	slab	147.76	2	concrete	43
	2	wall of 2000 mm	S1-18	wall	slab	95.54	2	concrete	43
	3	wall of 600 mm	S2-18	wall	slab	68.31	0.6	concrete	43
	4	floors CELL 18	floor18	floor	slab	147.76	0.6	concrete	43
19	1	roof of 2000 mm	roof19	roof	slab	147.76	2	concrete	43
	2	wall of 2000 mm	S1-19	wall	slab	95.54	2	concrete	43
	3	wall of 600 mm	S2-19	wall	slab	68.31	0.6	concrete	43
	4	Floors CELL 19	floor19	floor	slab	147.76	0.6	concrete	43
20	1	roof of 2000 mm	roof20	roof	slab	14.74	2	concrete	43
	2	wall of 1400 mm	S1-20	wall	slab	114.24	0.7	concrete	43
	3	wall of 300 mm	S2-20	wall	slab	48.91	0.15	concrete	43
	4	wall of 2000 mm	S3-20	wall	slab	30.47	2	concrete	43
	5	Floors CELL 20	floor20	floor	slab	12.32	4.3	concrete	43
21	1	roof of 2000 mm	roof21	roof	slab	147.76	2	concrete	43
	2	wall of 2000 mm	S1-21	wall	slab	95.54	2	concrete	43

Table 6.2-12c

Heat Sink Descriptions

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (ºC)
	3	wall of 600 mm	S2-21	wall	slab	68.31	0.6	concrete	43
	4	Floors CELL 21	floor21	floor	slab	147.76	0.6	concrete	43
22	1	roof of 2000 mm	roof22	roof	slab	147.76	2	concrete	43
	2	wall of 2000 mm	S1-22	wall	slab	95.54	2	concrete	43
	3	wall of 600 mm	S2-22	wall	slab	68.31	0.6	concrete	43
	4	Floors CELL 22	floor22	floor	slab	147.76	0.6	concrete	43
23	1	roof of 1000 mm	roof23	roof	slab	30.92	1	concrete	43
	2	wall of 2000 mm common with other rooms	S1-23	wall	slab	31.15	1	concrete	43
24	No heat sinks								

Table 6.2-13

Reactor Coolant Pressure Boundary Influent Lines Penetrating Drywell

Influent Line		Inside Drywell	Outside Drywell
1	Feedwater	CV or equivalent	(1) CV/POV combination (1) POV
2	IC Condensate	(1) NMOV or equivalent (1) NOV or equivalent	None (closed loop outside containment)
3	Standby liquid control	CV or equivalent	(1) CV or equivalent (2) SQUIB (parallel)
4	IC Purge Line	(1) CV (1) NOV or equivalent	None (closed loop outside containment)

CV = Check valve or equivalent process flow isolated valve
POV = Power-operated valve

NOV = Nitrogen-operated valve
SQUIB = Squib-activated valve, normally closed with solid metal isolation barrier
NMOV = Nitrogen motor operated valve or equivalent with fail as-is actuator

Table 6.2-14

Reactor Coolant Pressure Boundary Effluent Lines Penetrating Drywell

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

AOV = Air-operated valve or equivalent with fail-closed actuator
MOV = Motor-operated valve or equivalent with fail as-is actuator
NOV = Nitrogen-operated valve or equivalent with fail-closed actuator
NMOV = Nitrogen motor operated valve or equivalent with fail as-is actuator

Table 6.2-15

Legend For Tables 6.2-16 through 6.2-42

(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[1]:

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
PM	=	Process-medium 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	Process actuated
R	Local manual (By Hand)

[1] The operator types listed embody certain functional characteristics, such as those that fail-safe vs. fail as-is, or those that have a stored energy source (spring, fluid accumulator) to permit completion of function or repeat performance of functions upon loss of normal power supply. The actuator type listed for any valve application is generally based on historical BWR design. Alternate valve-&-operator combinations that provide equivalent functional capability and performance are permissible.

S High radiation in DW sump line
T High HVAC radiation exhaust from refueling area or from Reactor Building.
U Feedwater lines differential flow

(e) Valve Types[2]:

OS&Y Outside stem and yoke, typical of gate and globe valve designs that have an externally exposed rising or non-rising stem that connects a yoke-mounted actuator (any type) to the internal disk assembly, and includes a stem sealing gland (with or without a hermetic disk-to-stem internal seal such as a metal bellows or diaphragm).

Gate (GT) Any of several styles of valve where the disk is formed as a plate which transits the fluid flow stream with an orthogonal motion. The seating surface of the valve body is also manufactured to be at a slight angle to or set orthogonal to the flow stream. The disk can be wedge-shaped in either solid or split/flexible form, or as two plates mounted back-to-back or similar form (e.g., parallel-slide or double-disk gate), matching the seat configuration. Additional variants include shutter type and rotating-slide type gate valves.

Globe (GB) Any of several styles and configurations of valves where the disk is formed either as a truncated cone or curved section (spherical, elliptical, parabolic, etc.) with or without a following structure to support and guide the disk-&-stem motion. The body seat is centered around the flow stream and the disk-&-stem motion axis is perpendicular to the seat (i.e., axially concentric with the flow stream at the seat orifice plane). Body variations are based on the angle of the inlet-to-outlet nozzles and/or the angle of the stem to the inlet or outlet nozzle. Stem and disk assembly may be unconnected to permit a combined check and stop valve function (floating-disk stop, non-return check, etc.).

Quarter-turn (QT) Any of various types of butterfly (QBF) and ball (QBL) valves where the stem/shaft is mounted across the flow stream and the pallet, ball or plug (disk) is rotated through a 90 degree arc from full-closed to full-open. The actuator mechanism is typically mounted directly to the valve bonnet and there may be no exposed stem. The butterfly valve pallet remains in the flow stream when the valve is open whereas the plug and ball valves provide either a reduced or full pipe diameter flow orifice and shield the valve and disk seating surfaces when opened.

Axial-flow (AF) A variant of globe valves with the valve bonnet and disk-stem assembly rotated to be completely internalized and concentric with the fluid flow axis through the valve. There may be no external exposed stem or sealing gland, depending on design function(s) and selected actuation option. Based on specific product design, the flow path is typically formed as either an annular nozzle in a wafer-style body or as annular venturi in a teardrop-style body.

Check (CK) A valve operated by process flow (opens on forward flow, closes on reverse flow and gravity) with a pallet style disk that is connected by a hinge bracket or arm to a

[2] Valve type(s) listed for each containment isolation valve number in Tables 6.2-16 through 6.2-42 indicates either the specific design characteristics of a type or the range of types with suitable equivalent design characteristics capable of performing the intended function(s) for each application. The first type listed is generally based on historical selection from previous BWR designs.

shaft (or hinge end pins). The shaft is aligned in the horizontal plane with its rotation center typically set above the main fluid flow path so that the pallet swings up and out of the main flow on valve opening. A variant is the tilting disk pattern wherein the shaft is set closer to flow center and the hinge point is mounted directly behind the pallet (similar to a butterfly valve). Check valves may have spring-return closure (closure-assist) either internally or externally mounted. Globe and axial-flow valve variants are also designed to perform the check valve function.

Relief (RV) A variant of globe valves operated by process pressure most commonly built in spring-closed pressure-under-seat/pressurize-to-lift pattern (also referred to as direct-acting). There are also piloted relief valve versions, using a piston and process pressure to move the main disk off its seat, that are either depressurize-to-operate or pressurize-to-operate designs. The control pilot that operates the piston is typically a small version of direct-acting pressure relief valve.

Table 6.2-16

Containment Isolation Valve Information for the Nuclear Boiler System

Main Steam Line A

Penetration Identification	B21-MPEN-0001		
Valve No.	F001A	F002A	F016A
Applicable Basis	GDC 55	GDC 55	GDC 55
Tier 2 Figure	5.1-2	5.1-2	5.1-2
ESF	No	No	No
Fluid	Steam	Steam	Steam/Water
Line Size	700 mm	700 mm.	50 mm.
Type C Leakage Test	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide
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[(e)]	GB. GT, AF, QT	GB. GT, AF, QT	GT, GB, QBL
Operator[(c)]	NO/Acc	AO/Acc	NMO
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)	3.0-5.0	3.0-5.0	15
Power Source	Div. 1, 2	Div. 1, 2	Div. 1, 2, 3

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	B21-MPEN-0002		
Valve No.	F001B	F002B	F016B
Applicable Basis	GDC 55	GDC 55	GDC 55
Tier 2 Figure	5.1-2	5.1-2	5.1-2
ESF	No	No	No
Fluid	Steam	Steam	Steam/Water
Line Size	700 mm.	700 mm	50 mm
Type C Leakage Test	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide
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[e]	GB, GT, AF, QT	GB, GT, AF, QT	GT, GB, QBL
Operator[c]	NO/Acc	AO/Acc	NMO
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)	3.0-5.0	3.0-5.0	15
Power Source	Div. 1, 2	Div. 1, 2	Div. 1, 2, 3

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	B21-MPEN-0003		
Valve No.	F001C	F002C	F016C
Applicable Basis	GDC 55	GDC 55	GDC 55
Tier 2 Figure	5.1-2	5.1-2	5.1-2
ESF	No	No	No
Fluid	Steam	Steam	Steam/Water
Line Size	700 mm	700 mm	50 mm
Type C Leakage Test	Yes	Yes	Yes
Pipe Length from cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide
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[e]	GB, GT, AF, QT	GB, GT, AF, QT	GT, GB, GBL
Operator[c]	NO/Acc	AO/Acc	NMO
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)	3.0-5.0	3.0-5.0	15
Power Source	Div. 1, 2	Div. 1, 2	Div. 1, 2, 3

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	B21-MPEN-0004		
Valve No.	F001D	F002D	F016D
Applicable Basis	GDC 55	GDC 55	GDC 55
Tier 2 Figure	5.1-2	5.1-2	5.1-2
ESF	No	No	No
Fluid	Steam	Steam	Steam/Water
Line Size	700 mm	700 mm	50 mm
Type C Leakage Test	Yes	Yes	Yes
Pipe Length from cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide
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[(e)]	GB, GT, AF, QT	GB, GT, AF, QT	GT, GB, QBL
Operator[(c)]	NO/Acc	AO/Acc	NMO
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)	3.0-5.0	3.0-5.0	15
Power Source	Div. 1, 2	Div. 1, 2	Div. 1, 2, 3

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	**B21-MPEN-0005**	
Valve No.	F010	F011
Applicable Basis	GDC 55	GDC 55
Tier 2 Figure	5.1-2	5.1-2
ESF	No	No
Fluid	Steam/water	Steam/water
Line Size	80 mm	80 mm
Type C Leakage Test	Yes	Yes
Pipe Length from cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide
Leakage Through Packing[a]	N/A	(a_2)
Leakage Past Seat[b]	(b_3)	(b_3)
Location	Inboard	Outboard
Valve Type[e]	GT, QBL, GB	GT, QBL, GB
Operator[c]	NO	AO
Normal Position	Open	Open
Shutdown Position	Open	Open
Post-Acc Position	Closed	Closed
Power Fail Position	Closed	Closed
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, 4	Div. 1, 3

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	**B21-MPEN-0006**	
Valve No.	F102A	F101A
Applicable Basis	GDC 55	GDC 55
Tier 2 Figure	5.1-2	5.1-2
ESF	No	No
Fluid	Water	Water
Line Size	550 mm	550 mm
Type C Leakage Test	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide
Leakage Through Packing[a]	N/A	(a_2)
Leakage Past Seat[b]	N/A	(b_3)
Location	Inboard	Outboard
Valve Type[e]	CK, AF, GB	GB, AF
Operator[c]	N/A	AO, PM
Normal Position	Open	Open
Shutdown Position	N/A	Closed
Post-Acc Position	N/A	Closed
Power Fail Position	N/A	N/A
Cont. Iso. Signal[d]	Q	Q, or U, B+H
Primary Actuation	Flow	Flow to open/close or Auto-closed
Secondary Actuation	N/A	Remote manual
Closure Time (sec)	N/A	N/A on reverse-flow, < 40 sec on auto-isolation
Power Source	N/A	Div. 1, 3

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	B21-MPEN-0007	
Valve No.	F102B	F101B
Applicable Basis	GDC 55	GDC 55
Tier 2 Figure	5.1-2	5.1-2
ESF	No	No
Fluid	Water	Water
Line Size	550 mm	550 mm
Type C Leakage Test	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	N/A	(a_2)
Leakage Past Seat[(b)]	N/A	(b_3)
Location	Inboard	Outboard
Valve Type[(e)]	CK, AF, GB	GB, AF
Operator[(c)]	N/A	AO, PM
Normal Position	Open	Open
Shutdown Position	N/A	Closed
Post-Acc Position	N/A	Closed
Power Fail Position	N/A	N/A
Cont. Iso. Signal[(d)]	Q	Q, or U, B+H
Primary Actuation	Flow	Flow to open/close or Auto-closed
Secondary Actuation	N/A	Remote manual
Closure Time (sec)	N/A	N/A on reverse flow, < 40 sec on auto-isolation
Power Source	N/A	Div. 1, 3

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

Table 6.2-23

Containment Isolation Valve Information for the Isolation Condenser System Loop A

Penetration Identification	B32-MPEN-0001[3]		B32-MPEN-0005	
Valve Number	F001A	F002A	F003A	F004A
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Tier 2 Figure	5.1-3	5.1-3	5.1-3	5.1-3
ESF	Yes	Yes	Yes	Yes
Fluid	Steam	Steam	Condensate	Condensate
Line Size	350mm	350mm	200mm	200mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[4] b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard
Valve Type[(e)]	QBL, GT	QBL, GT	QBL, GT	QBL, GT
Operator[(c)]	NMO/Acc	NO	NO	NMO/Acc
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Acc Position	Open[5]	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

[3]Two in series valves
[4]Piping of IC Quality Group B Design
[5]Except on IC pipe or tube failure

Table 6.2-23

Containment Isolation Valve Information for the Isolation Condenser System Loop A

Penetration Identification	**B32-MPEN-0001[3]**		**B32-MPEN-0005**	
Valve Number	F001A	F002A	F003A	F004A
Power Source	Div. 1, 3	Div. 2, 4	Div. 2, 4	Div. 1, 3

* With respect to meeting the requirements of US NRC 10 CFR 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. The combination of an already closed loop outside the containment plus the two series automatic isolation valves inside the containment comply with the requirement of the isolation guidelines of 10 CFR50, App.A, Criterion 55 and 56.

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

Table 6.2-24

Containment Isolation Valve Information for the Isolation Condenser System Loop A

Penetration Identification	B32-MPEN-0009[6]		B32-MPEN-0013[7]				B32-MPEN-0017[4]	
Valve Number	F007A	F008A	F009A	F010A	F011A	F012A	F013A	F014A
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*
Tier 2 Figure	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3
ESF	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
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
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[8]b6	b6	b6	b6	b6	b6	b6	b6

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

Table 6.2-24

Containment Isolation Valve Information for the Isolation Condenser System Loop A

Penetration Identification	**B32-MPEN-0009[6]**		**B32-MPEN-0013[7]**				**B32-MPEN-0017[4]**	
Valve Number	F007A	F008A	F009A	F010A	F011A	F012A	F013A	F014A
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard
Valve Type	GB, QBL	GB, QBL	GB, QBL	GB, QBL	QBF, QBL, GT	QBF, QBL, GT	QBF, QBL, GT	Excess-CK
Operator[(c)]	SO	SO	SO	SO	SO	SO	NO	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
Power Fail Position	Closed	Closed	Closed	Closed	Closed	Closed	Closed	As is
Cont. Iso. Signal[(d)]	P	P	P	P	P	P	I	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Automatic	Diff Pressure
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A	Remote Manual	N/A
Closure Time (sec)	< 30	< 30	< 30	< 30	< 30	< 30	< 30	< 30
Power Source	Div. 1	Div. 1	Div. 2, 4	Div. 2, 4	Div. 1	Div. 1	Div. 1, 2, 3	N/A

* The piping and valve arrangement for these lines meet the requirement of 10 CFR50, 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.

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	B32-MPEN-0002[9]		B32-MPEN-0006[7]	
Valve Number	F001B	F002B	F003B	F004B
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Tier 2 Figure	5.1-3	5.1-3	5.1-3	5.1-3
ESF	Yes	Yes	Yes	Yes
Fluid	Steam	Steam	Condensate	Condensate
Line Size	350mm	350mm	200mm	200mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[10] b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard
Valve Type[(e)]	QBL, GT	QBL, GT	QB, GT	QBL, GT
Operator[(c)]	NMO/Acc	NO	NO	NMO/Acc
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Acc Position	Open[11]	Open[9]	Open[9]	Open[9]
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

[9] Two in series valves
[10] Closed barrier outside containment (Piping of IC Quality Group B Design)
[11] Except on IC pipe or tube failure

Table 6.2-25

Containment Isolation Valve Information for the Isolation Condenser System Loop B

Penetration Identification	B32-MPEN-0002[9]		B32-MPEN-0006[7]	
Closure Time (sec)	< 60	< 60	< 35	< 35
Power Source	Div. 1, 3	Div. 2, 4	Div. 2, 4	Div. 1, 3

* With respect to meeting the requirements of US NRC 10 CFR 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. The combination of an already closed loop outside the containment plus the two series automatic isolation valves inside the containment comply with the requirements of the isolation guidelines of 10 CFR50, App. A, Criterion 55 and 56.

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

Table 6.2-26

Containment Isolation Valve Information for the Isolation Condenser System Loop B

Penetration Identification	B32-MPEN-0010[12]		B32-MPEN-0014[13]				B32-MPEN-0018[10]	
Valve Number	F007B	F008B	F009B	F010B	F011B	F012B	F013B	F014B
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*
Tier 2 Figure	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3
ESF	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
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
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[14]b6	b6	b6	b6	b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard

[12]Two in series valves
[13]Two in series valves (F009/F010) in parallel with two in series valves (F011/F012)
[14]Closed barrier outside containment

Table 6.2-26

Containment Isolation Valve Information for the Isolation Condenser System Loop B

Penetration Identification	**B32-MPEN-0010[12]**		**B32-MPEN-0014[13]**				**B32-MPEN-0018[10]**	
Valve Number	F007B	F008B	F009B	F010B	F011B	F012B	F013B	F014B
Valve Type(e)	GB, QBL	GB, QBL	GB, QBL	GB, QBL	, QBF, QBL, GT	, QBF, QBL, GT	, QBF, QBL, GT	Excess-CK
Operator(c)	SO	SO	SO	SO	SO	SO	NO	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
Power Fail Position	Closed	Closed	Closed	Closed	Closed	Closed	Closed	As is
Cont. Iso. Signal(d)	P	P	P	P	P	P	I	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Automatic	Diff Pressure
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A	Remote Manual	N/A
Closure Time (sec)	< 30	< 30	< 30	< 30	< 30	< 30	< 30	< 30
Power Source	Div. 2	Div. 2	Div. 1, 3	Div. 1, 3	Div. 2	Div. 2	Div. 2, 3, 4	N/A

* The piping and valve arrangement for these lines meet the requirements of 10 CFR50, 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.

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

Table 6.2-27

Containment Isolation Valve Information for the Isolation Condenser System Loop C

Penetration Identification	**B32-MPEN-0003**[15]		**B32-MPEN-0007**[13]	
Valve Number	F001C	F002C	F003C	F004C
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Tier 2 Figure	5.1-3	5.1-3	5.1-3	5.1-3
ESF	Yes	Yes	Yes	Yes
Fluid	Steam	Steam	Condensate	Condensate
Line Size	350 mm.	350 mm.	200 mm	200 mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[16] b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard
Valve Type[(e)]	QBL, GT	QBL, GT	QB, GT	QBL, GT
Operator[(c)]	NMO/Acc	NO	NO	NMO/Acc
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Acc Position	Open[17]	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, 3	Div. 2, 4	Div. 2, 4	Div. 1, 3

* With respect to meeting the requirements of US NRC 10 CFR 50, Appendix A, General Design Criteria 55, the closed loop safety-related IC loop outside the containment is a

[15]Two in series valves
[16]Closed barrier outside containment (Piping of IC Quality Group B Design)
[17]Except on IC pipe or tube failure

"passive" substitute for an open "active" valve outside the containment. The combination of an already closed loop outside the containment plus the two series automatic isolation valves inside the containment comply with the intent of the isolation guidelines of 10 CFR 50, App.A, Criterion 55 and 56.

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

Table 6.2-28

Containment Isolation Valve Information for the Isolation Condenser System Loop C

Penetration Identification	**B32-MPEN-0011**[18]		**B32-MPEN-0015**[19]				**B32-MPEN-0019**[16]	
Valve Number	F007C	F008C	F009C	F010C	F011C	F012C	F013C	F014C
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*
Tier 2 Figure	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3
ESF	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
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
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[20]b6	b6	b6	b6	b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard
Valve Type[(e)]	GB, QBL	GB, QBL	GB, QBL	GB, QBL	QBF, QBL,	QBF, QBL,	QBF, QBL,	Excess-CK

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

Table 6.2-28

Containment Isolation Valve Information for the Isolation Condenser System Loop C

Penetration Identification	B32-MPEN-0011[18]		B32-MPEN-0015[19]				B32-MPEN-0019[16]	
Valve Number	F007C	F008C	F009C	F010C	F011C	F012C	F013C	F014C
					GT	GT	GT	
Operator(c)	SO	SO	SO	SO	SO	SO	NO	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
Power Fail Position	Closed	Closed	Closed	Closed	Closed	Closed	Closed	As is
Cont. Iso. Signal(d)	P	P	P	P	P	P	I	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Automatic	Diff Pressure
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A	Remote Manual	N/A
Closure Time (sec)	< 30	< 30	< 30	< 30	< 30	< 30	< 30	< 30
Power Source	Div. 3	Div. 3	Div. 2, 4	Div. 2, 4	Div. 3	Div. 3	Div. 3, 4, 1	N/A

* The piping and valve arrangement for these lines meet the requirements of 10 CFR 50, 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.

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

Table 6.2-29

Containment Isolation Valve Information for the Isolation Condenser System Loop D

Penetration Identification	B32-MPEN-0004[21]		B32-MPEN-0008[19]	
Valve Number	F001D	F002D	F003D	F004D
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55*	GDC 55*	GDC 55*	GDC 55*
Tier 2 Figure	5.1-3	5.1-3	5.1-3	5.1-3
ESF	Yes	Yes	Yes	Yes
Fluid	Steam	Steam	Condensate	Condensate
Line Size	350 mm	350 mm	200 mm	200 mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	[22] b6	B6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard
Valve Type[(e)]	QBL, GT	QBL, GT	QBL, GT	QBL, GT
Operator[(c)]	NMO/Acc	NO	NO	NMO/Acc
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Acc Position	Open[23]	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, 3	Div. 2, 4	Div. 2, 4	Div. 1, 3

[21]Two in series valves
[22]Closed barrier outside containment (Piping of IC Quality Group B Design)
[23]Except on IC pipe or tube failure

* With respect to meeting the requirements of US NRC 10 CFR 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. The combination of an already isolated loop outside the containment plus the two series automatic isolation valves inside the containment comply with the requirements of the isolation guidelines of 10 CFR 50, App. A, Criterion 55 and 56.

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

Table 6.2-30

Containment Isolation Valve Information for the Isolation Condenser System Loop D

Penetration Identification	B32-MPEN-0012[24]		B32-MPEN-0016[25]				B32-MPEN-0020[22]	
Valve Number	F007D	F008D	F009D	F010D	F011D	F012D	F013D	F014D
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*
Tier 2 Figure	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3	5.1-3
ESF	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
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
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing(a)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat(b)	[26] b6	b6	b6	b6	b6	b6	b6	b6
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard
Valve Type(e)	GB, QBL	GB, QBL	GB, QBL	GB, QBL	QBF, QBL,	QBL, QBF,	QBL, QBF,	Excess-CK

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

Table 6.2-30

Containment Isolation Valve Information for the Isolation Condenser System Loop D

Penetration Identification	B32-MPEN-0012[24]		B32-MPEN-0016[25]				B32-MPEN-0020[22]	
Valve Number	F007D	F008D	F009D	F010D	F011D	F012D	F013D	F014D
					GT	GT	GT	
Operator(c)	SO	SO	SO	SO	SO	SO	NO	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
Power Fail Position	Closed	Closed	Closed	Closed	Closed	Closed	Closed	As is
Cont. Iso. Signal(d)	P	P	P	P	P	P	I	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Automatic	Diff Pressure
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A	Remote Manual	N/A
Closure Time (sec)	< 30	< 30	< 30	< 30	< 30	< 30	< 30	< 30
Power Source	Div. 4	Div. 4	Div. 1, 3	Div. 1, 3	Div. 4	Div. 4	Div. 4, 1, 2	N/A

* The piping and valve arrangement for these lines meet the requirements of 10 CFR50, 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.

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	**G31-MPEN-0001**		**G31-MPEN-0003**		**G31-MPEN-0002**		**G31-MPEN-0004**	
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
Tier 2 Figure	5.1-4	5.1-4	5.1-4	5.1-4	5.1-4	5.1-4	5.1-4	5.1-4
ESF	No	No	No	No	No	No	No	No
Fluid	Water	Water	Water	Water	Water	Water	Water	Water
Line Size	250 mm	250 mm.	150 mm	150 mm	250 mm	250 mm.	150 mm	150 mm
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	N/A	(a_1)	N/A	(a_1)	N/A	(a_1)	N/A	(a_1)
Leakage Past Seat[(b)]	(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[(e)]	GT, QBL, AF	GT, QBL, AF	GT, QBL, AF	GT, QBL, AF	GT, QBL, AF	GT, QBL, AF	GT, QBL, AF	GT, QBL, AF
Operator[(c)]	NO	AO	NO	AO	NO	AO	NO	AO
Normal Position	O/C	O/C	O/C	O/C	O/C	O/C	OC	OC
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)]	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

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

Penetration Identification	**G31-MPEN-0001**		**G31-MPEN-0003**		**G31-MPEN-0002**		**G31-MPEN-0004**	
Valve No.	F002A	F003A	F007A	F008A	F002B	F003B	F007B	F008B
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)	<20	<20	<15	<15	<20	<20	<15	<15
Power Source	Div. 2, 4	Div. 1, 3	Div. 2, 4	Div 1, 3	Div. 2, 4	Div. 1, 3	Div. 2, 4	Div. 1, 3

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

Table 6.2-31a

Containment Isolation Valve Information for the Reactor Water Cleanup/Shutdown Cooling System

Penetration Identification	**G31-MPEN-0005**		**G31-MPEN-0006**	
Valve No.	F038A	F039A	F038B	F039B
Applicable Basis	GDC 55	GDC 55	GDC 55	GDC 55
Tier 2 Figure	5.1-4	5.1-4	5.1-4	5.1-4
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size	20 mm.	20 mm	20 mm.	20 mm.
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	N/A	(a_1)	N/A	(a_1)
Leakage Past Seat[(b)]	b7	b7	b7	b7
Location	Inboard	Outboard	Inboard	Outboard
Valve Type[(e)]	GB, QBL	GB, QBL	GB, QBL	GB, QBL
Operator[(c)]	SO	SO	SO	SO
Normal Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Shutdown Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Post-Acc Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Power Fail Position	Closed	Closed	Closed	Closed
Cont. Iso. Signal[(d)]	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
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)	<15	<15	<15	<15
Power Source	Div. 2, 4	Div. 1, 3	Div. 2, 4	Div 1, 3

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

Table 6.2-32a

Containment Isolation Valve Information for the Standby Liquid Control System

Penetration Identification	C41-MPEN-0001			
Valve No.	F005A	F004A	F003A	F003C
Applicable Basis	GDC 55	GDC 55	GDC 55	GDC 55
Tier 2 Figure	9.3-1	9.3-1	9.3-1	9.3-1
ESF	Yes	Yes	Yes	Yes
Fluid	Boron/Water	Boron/Water	Boron/Water	Boron/Water
Line Size	80 mm	80 mm	80 mm	80 mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing(a)	N/A	(a1)	(a1)	(a1)
Leakage Past Seat(b)	(b5)	(b5)	(b5)	(b5)
Location	Inboard	Outboard	Outboard	Outboard
Valve Type	CK, GB, AF	CK, GB, AF	GT*	GT*
Operator(c)	N/A	N/A	N/A**	N/A**
Normal Position	Closed	Closed	Closed	Closed
Shutdown Position	Closed	Closed	Closed	Closed
Post-Acc Position	Operable	Operable	Open	Open
Power Fail Position	N/A	N/A	As is	As is
Cont. Iso. Signal(d)	Q	Q	N/A**	N/A**
Primary Actuation	Flow	Flow	N/A**	N/A**
Secondary Actuation	N/A	N/A	N/A**	N/A**
Closure Time (sec)	N/A	N/A	N/A**	N/A**
Power Source	N/A	N/A	N/A**	N/A**

* The disk/inlet-fitting cap is hermetically sealed and when valve is actuated, the cap is sheared to permanently open the flow path.
**Not relevant to the valve isolation function.
Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-32b

Containment Isolation Valve Information for the Standby Liquid Control System

Penetration Identification	C41-MPEN-0002			
Valve No.	F005B	F004B	F003B	F003D
Applicable Basis	GDC 55	GDC 55	GDC 55	GDC 55
Tier 2 Figure	9.3-1	9.3-1	9.3-1	9.3-1
ESF	Yes	Yes	Yes	Yes
Fluid	Boron/Water	Boron/Water	Boron/Water	Boron/Water
Line Size	80 mm	80 mm	80 mm	80 mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing(a)	N/A	(a1)	(a1)	(a1)
Leakage Past Seat(b)	(b5)	(b5)	(b5)	(b5)
Location	Inboard	Outboard	Outboard	Outboard
Valve Type	CK, GB, AF	CK, GB, AF	GT*	GT*
Operator(c)	N/A	N/A	N/A**	N/A**
Normal Position	Closed	Closed	Closed	Closed
Shutdown Position	Closed	Closed	Closed	Closed
Post-Acc Position	Operable	Operable	Open	Open
Power Fail Position	N/A	N/A	As is	As is
Cont. Iso. Signal(d)	Q	Q	N/A**	N/A**
Primary Actuation	Flow	Flow	N/A**	N/A**
Secondary Actuation	N/A	N/A	N/A**	N/A**
Closure Time (sec)	N/A	N/A	N/A**	N/A**
Power Source	N/A	N/A	N/A**	N/A**

* The disk/inlet-fitting cap is hermetically sealed and when valve is actuated, the cap is sheared to permanently open the flow path.
**Not relevant to the valve isolation function.
Note: For explanation of codes, see legend on Table 6.2-15.

Table 6.2-33a

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

Penetration Identification	G21-MPEN-0005		G21-MPEN-0002	
Valve No.	F321A	F322A	F306A	F307A
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	9.1-1	9.1-1	9.1-1	9.1-1
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size	250 mm.	250 mm.	250 mm.	250 mm.
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[a]	(a_1)	(a_1)	(a_1)	N/A
Leakage Past Seat[b]	b6	b6	b6	b6
Location	Outboard	Outboard	Outboard	Inboard
Valve Type	GT, QT, AF	GT, QT, AF	GT, QT, AF	CK, AF
Operator[c]	NMO	NMO	NMO	N/A
Normal Position	Closed[27]	Closed[27]	Closed[27]	N/A
Shutdown Position	Closed[27]	Closed[27]	Closed[27]	N/A
Post-Acc Position	Closed[28]	Closed[28]	Closed[29]	N/A
Power Fail Position	As-is	As-is	As-is	N/A
Cont. Iso. Signal[d]	P	P	P	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Flow
Secondary Actuation	Local manual	Local manual	Local manual	N/A

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

Table 6.2-33a

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

Penetration Identification	**G21-MPEN-0005**		**G21-MPEN-0002**	
Valve No.	F321A	F322A	F306A	F307A
Closure Time (sec)	<30	<30	<30	N/A
Power Source	Div. 1, 3	Div. 1, 3	Div. 1, 3	N/A

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

Table 6.2-33b

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

Penetration Identification	G21-MPEN-0007		G21-MPEN-0006	
Valve No.	F321B	F322B	F306B	F307B
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	9.1-1	9.1-1	9.1-1	9.1-1
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size	250 mm.	250 mm.	250 mm.	250 mm.
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	(a_1)	(a_1)	(a_1)	N/A
Leakage Past Seat[(b)]	b6	b6	b6	b6
Location	Outboard	Outboard	Outboard	Inboard
Valve Type	GT, QT, AF	GT, QT, AF	GT, QT, AF	CK, AF
Operator[(c)]	NMO	NMO	NMO	N/A
Normal Position	Closed[30]	Closed[30]	Closed[30]	N/A
Shutdown Position	Closed[31]	Closed[31]	Closed[31]	N/A
Post-Acc Position	Closed[32]	Closed[32]	Closed[33]	N/A
Power Fail Position	As-is	As-is	As-is	N/A
Cont. Iso. Signal[(d)]	P	P	P	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Self

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

Table 6.2-33b

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

Penetration Identification	**G21-MPEN-0007**		**G21-MPEN-0006**	
Valve No.	F321B	F322B	F306B	F307B
Secondary Actuation	Local manual	Local manual	Local manual	N/A
Closure Time (sec)	<30	<30	<30	N/A
Power Source	Div. 2, 4	Div. 2, 4	Div. 2, 4	N/A

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

Table 6.2-34

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

Penetration Identification	**G21-MPEN-0004**		**G21-MPEN-0003**	
Valve No.	F323	F324	F303	F304
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	9.1-1	9.1-1	9.1-1	9.1-1
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size	250 mm.	250 mm	250 mm.	250 mm.
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
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	GT, QT, AF	GT, QT, AF	GT, QT, AF	CK, AF
Operator[(c)]	NO	AO	AO	N/A
Normal Position	Closed[34]	Closed[34]	Closed[34]	N/A
Shutdown Position	Closed	Closed	Closed	N/A
Post-Acc Position	Closed	Closed	Closed	N/A
Power Fail Position	Closed	Closed	Closed	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)	<30	<30	<30	N/A
Power Source	Div. 2, 4	Div. 1, 3	Div. 1, 2, 3	N/A

[34]The valve is open occasionally for GDCS pools cooling and cleanup function.

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

Table 6.2-35

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

Penetration Identification	G21-MPEN-0001	
Valve No.	F309	F310
Applicable Basis	GDC 56	GDC 56
Tier 2 Figure	9.1-1	9.1-1
ESF	No	No
Fluid	Water	Water
Line Size	250 mm.	250 mm.
Type C Leakage Test	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	(a_1)	N/A
Leakage Past Seat[(b)]	b6	b6
Location	Outboard	Inboard
Valve Type	GB, AF	CK, AF
Operator[(c)]	AO	N/A
Normal Position	Closed	N/A
Shutdown Position	Closed	N/A
Post-Acc. Position	Closed[35]	N/A
Power Fail Position	Closed	N/A
Cont. Iso. Signal[(d)]	P	Q
Primary Actuation	Electrical	Flow
Secondary Actuation	Remote manual	N/A
Closure Time(sec)	<35	N/A
Power Source	Div. 1, 2, 3	N/A

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

[35]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	**T31-MPEN-0004**		**T31-MPEN-0003**[36]			
Valve No.	F012	F011	F010	F011	F014	F015
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	9.4-14	9.4-14	9.4-14	9.4-14	9.4-14	9.4-14
ESF	No	No	No	No	No	No
Fluid	Air/N_2	Air/N_2	Air/N_2	Air/N_2	Air/N_2	Air/N_2
Line Size	350 mm	500 mm	400 mm	500 mm	25 mm.	25mm
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	(a_1)	(a_1)	(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)	(b_2/b_5)	(b_2/b_5)
Location	Outboard	Outboard	Outboard	Outboard	Outboard	Outboard
Valve Type	QBF	QBF	QBF	QBF	GB, QBL, GT	GB,QBL, GT
Operator[(c)]	AO	AO	AO	AO	AO	AO
Normal Position	Closed	Closed	Closed	Closed	Closed[37]	Closed[31]
Shutdown Position	Closed[32]	Closed[32]	Closed[38]	Closed[32]	Closed	Closed
Post-Acc Position	Closed	Closed	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed	Closed	Closed
Cont. Iso. Signal[(d)]	B,C,H,T	B,C,H,T	B,C,H,T	B,C,H,T	B,C,H,T	B,C,H,T
Primary Actuation	Automatic	Automatic	Automatic	Automatic	Automatic	Automatic

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

Table 6.2-36

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	T31-MPEN-0004		T31-MPEN-0003[36]			
Valve No.	F012	F011	F010	F011	F014	F015
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)	< 30	< 30	< 30	< 30	< 5	< 5
Power Source	Div. 2, 4	Div. 1, 3	Div. 2, 4	Div. 1, 3	Div. 2, 4	Div. 1, 3

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

Table 6.2-37

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	T31-MPEN-0002[39]			
Valve No.	F008	F007	F024	F023
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	9.4-14	9.4-14	9.4-14	9.4-14
ESF	No	No	No	No
Fluid	Air/N_2	Air/N_2	Air/N_2	Air/N_2
Line Size	500 mm	350 mm	25 mm	25 mm.
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
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	QBF	QBF	GB, QBL, GT	GB, QBL, GT
Operator[(c)]	AO	AO	AO	AO
Normal Position	Closed	Closed	Open	Open
Shutdown Position	Open	Open	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

[39]Valve F008 in series with F007, valve F024 in series with F023.

Table 6.2-37

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	T31-MPEN-0002[39]			
Valve No.	F008	F007	F024	F023
Closure Time (sec.)	< 30	< 30	< 5	< 5
Power Source	Div. 1, 3	Div. 2, 4	Div. 2, 4	Div. 1, 3

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

Table 6.2-38

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	T31-MPEN-0001[40]			
Valve No.	F025	F023	F008	F009
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	9.4-14	9.4-14	9.4-14	9.4-14
ESF	No	No	No	No
Fluid	Air/N_2	Air/N_2	Air/N_2	Air/N_2
Line Size	25 mm	25 mm	500 mm	350 mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
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	GB, QBL, GT	GB, QBL, GT	QBF	QBF
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

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

Table 6.2-38

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	T31-MPEN-0001[40]			
Valve No.	F025	F023	F008	F009
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)	< 5	< 5	< 30	< 30
Power Source	Div. 2, 4	Div. 1, 3	Div. 1, 3	Div. 2, 4

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

Table 6.2-39

Containment Isolation Valve Information for the Chilled Water System Train A

Penetration Identification	P25-MPEN-0001		P25-MPEN-0002	
Valve No.	F023A	F024A	F025A	F026A
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	9.2-3	9.2-3	9.2-3	9.2-3
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size	150 mm	150 mm	150 mm	150 mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[(a)]	(a1)	N/A	N/A	(a1)
Leakage Past Seat[(b)]	(b2)	(b2)	(b2)	(b2)
Location	Outboard	Inboard	Inboard	Outboard
Valve Type	GB, QT, GT	GB, QT, GT	GB, QT, GT	GB, QT, GT
Operator[(c)]	AO	NO	NO	AO
Normal Position	Open	Open	Open	Open
Shutdown Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Post-Acc Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Cont. Iso. Signal[(d)]	C,H	C,H	C,H	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

Table 6.2-39

Containment Isolation Valve Information for the Chilled Water System Train A

Penetration Identification	P25-MPEN-0001		P25-MPEN-0002	
Valve No.	F023A	F024A	F025A	F026A
Power Source	Div. 2, 4	Div. 1, 3	Div. 1, 3	Div. 2, 4

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

Table 6.2-39a
Containment Isolation Valve Information for the Chilled Water System Train B

Penetration Identification	P25-MPEN-0003		P25-MPEN-0004	
Valve No.	F023B	F024B	F025B	F026B
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	9.2-3	9.2-3	9.2-3	9.2-3
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size	150 mm	150 mm	150 mm	150 mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[a]	(a1)	N/A	N/A	(a1)
Leakage Past Seat[b]	(b2)	(b2)	(b2)	(b2)
Location	Outboard	Inboard	Inboard	Outboard
Valve Type	GB, QT, GT	GB, QT, GT	GB, QT, GT	GB, QT, GT
Operator[c]	AO	NO	NO	AO
Normal Position	Open	Open	Open	Open
Shutdown Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Post-Acc Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Cont. Iso. Signal[d]	C,H	C,H	C,H	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. 2, 4	Div. 1, 3	Div. 1, 3	Div. 2, 4

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 Gas Supply System

Penetration Identification	P54-MPEN-0001		P54-MPEN-0002	
Valve No.	F0026	F027	F009	F010
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	N/A	N/A	N/A	N/A
ESF	No	No	No	No
Fluid	Air/N_2	Air/N_2	N_2	N_2
Line Size	50 mm	50 mm	50 mm	50 mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	COL holder to provide	COL holder to provide	COL holder to provide	COL holder to provide
Leakage Through Packing[a]	(a_1)	N/A	(a_1)	N/A
Leakage Past Seat[b]	(b_2)	(b_2)	(b_2)	(b_2)
Location	Outboard	Inboard	Outboard	Inboard
Valve Type	GB, QT	CK	GB, QT	CK
Operator[c]	AO	PM	AO	PM
Normal Position	Open	Open/Closed	Open	Open/Closed
Shutdown Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Post-Acc Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Cont. Iso. Signal[d]	C,H	Q	C,H	Q
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Process Actuated	Remote manual	Process Actuated
Closure Time (sec.)	< 30	N/A	< 30	N/A
Power Source	Div. 2, 4	N/A	Div. 2, 4	N/A

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

Table 6.2-41

(Not used)

Table 6.2-42

Containment Isolation Valve Information for the Process Radiation Monitoring System

Penetration Identification	D11-MPEN-0001[41]		D11-MPEN-0002[42]	
Valve No.	F001	F002	F003	F004
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	N/A	N/A	N/A	N/A
ESF	No	No	No	No
Fluid	Air/N_2	Air/N_2	Air/N_2	Air/N_2
Line Size	25 mm	25 mm	25 mm	25 mm
Type C Leakage Test	Yes	Yes	Yes	Yes
Pipe Length from Cont. to Inboard/Outboard Isolation Valve	See Sub-section 6.2.4.2	See Sub-section 6.2.4.2	See Sub-section 6.2.4.2	See Sub-section 6.2.4.2
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	GB, QT	GB, QT	GB, QT	GB, QT
Operator[(c)]	SO	SO	SO	SO
Normal Position	Open	Open	Open	Open
Shutdown Position	Closed	Closed	Closed	Closed
Post-Acc Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Cont. Iso. Signal[(d)]	C,H,T	C,H,T	C,H,T	C,H,T
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual

[41] Valve F001 in series with F002.
[42] Valve F003 in series with F004.

Table 6.2-42

Containment Isolation Valve Information for the Process Radiation Monitoring System

Penetration Identification	D11-MPEN-0001[41]		D11-MPEN-0002[42]	
Valve No.	F001	F002	F003	F004
Closure Time (sec.)	< 5	< 5	< 5	< 5
Power Source	Div. 2, 4	Div. 1, 3	Div. 2, 4	Div. 1, 3

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

Table 6.2-43

(Not used)

Table 6.2-44

(Not used)

Table 6.2-45

(Not used)

Table 6.2-46

(Not used

Table 6.2-47

Containment Penetrations Subject To Type A, B, and C Testing

Penetration Number (1)	Description	Location (3)/Room #	RCCV Sector	Penetration Type (4)	Leak Test Type (5)
Piping Penetrations					
B21: Nuclear Boiler System (NBS)					
B21-MPEN-0001	Main Steam Line A	UD / ST	I	A	A
B21-MPEN-0002	Main Steam Line B	UD / ST	I	A	A
B21-MPEN-0003	Main Steam Line C	UD / ST	IV	A	A
B21-MPEN-0004	Main Steam Line D	UD / ST	IV	A	A
B21-MPEN-0006	Feedwater Line A	UD / ST	I	A	A
B21-MPEN-0007	Feedwater Line B	UD / ST	IV	A	A
B21-MPEN-0005	Main Steam Drain Header	UD /ST	TBD	A	A
B21-MPEN-TBD	RPV Water Level	UD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	UD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	UD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	UD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	UD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	UD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	UD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	UD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	LD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	LD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	LD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	LD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	LD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	LD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	LD	TBD	I	A
B21-MPEN-TBD	RPV Water Level	LD	TBD	I	A
B21-MPEN-TBD	Main Steam Line A Flow Restrictor Instr Line 1	UD	TBD	I	A
B21-MPEN-TBD	Main Steam Line A Flow Restrictor Instr Line 2	UD	TBD	I	A
B21-MPEN-TBD	Main Steam Line B Flow Restrictor Instr Line 1	UD	TBD	I	A
B21-MPEN-TBD	Main Steam Line B Flow Restrictor Instr Line 2	UD	TBD	I	A
B21-MPEN-TBD	Main Steam Line C Flow Restrictor Instr Line 1	UD	TBD	I	A
B21-MPEN-TBD	Main Steam Line C Flow Restrictor Instr Line 2	UD	TBD	I	A
B21-MPEN-TBD	Main Steam Line D Flow Restrictor Instr Line 1	UD	TBD	I	A
B21-MPEN-TBD	Main Steam Line D Flow Restrictor Instr Line 2	UD	TBD	I	A

Table 6.2-47

Containment Penetrations Subject To Type A, B, and C Testing

Penetration Number (1)	Description	Location (3)/Room #	RCCV Sector	Penetration Type (4)	Leak Test Type (5)
B21-MPEN-TBD	Feedwater Line A Instrumentation	UD / ST	TBD	I	A
B21-MPEN-TBD	Feedwater Line B Instrumentation	UD / ST	TBD	I	A
B21-MPEN-TBD	RPV Flange Seal Leakage Monitor	UD / ST	TBD	I	A
B21-MPEN-TBD	RPV Top Head Vent Instrument Line	UD / ST	TBD	I	A
B32: Isolation Condenser System (ICS)					
B32-MPEN-0001	Train A Steam Supply Line	TS	I	B	A
B32-MPEN-0017	Train A Purge Line From Steam Supply Line	TS	I	B	A
B32-MPEN-0005	Train A Condensate Return	TS	I	B	A
B32-MPEN-0009	Train A Vent Line A From Upper Header (1CA)	TS	I	B	A
B32-MPEN-0013	Train A Vent Line A From Lower Header (1CA)	TS	I	B	A
B32-MPEN-TBD	Train A Steam Line Flowrate Instrumentation	UD	I	I	A
B32-MPEN-TBD	Train A Steam Line Flowrate Instrumentation	UD	I	I	A
B32-MPEN-TBD	Train A Steam Line Flowrate Instrumentation	UD	I	I	A
B32-MPEN-TBD	Train A Steam Line Flowrate Instrumentation	UD	I	I	A
B32-MPEN-TBD	Train A Condensate Line Flowrate Instrumentation	UD	I	I	A
B32-MPEN-TBD	Train A Condensate Line Flowrate Instrumentation	UD	I	I	A
B32-MPEN-TBD	Train A Condensate Line Flowrate Instrumentation	UD	I	I	A
B32-MPEN-TBD	Train A Condensate Line Flowrate Instrumentation	UD	I	I	A
B32-MPEN-0002	Train B Steam Supply Line	TS	III	B	A
B32-MPEN-0018	Train B Purge Line From Steam Supply Line	TS	III	B	A
B32-MPEN-0006	Train B Condensate Return	TS	III	B	A
B32-MPEN-0010	Train B Vent Line A From Upper Header (1CB)	TS	III	B	A
B32-MPEN-0014	Train B Vent line A From Lower Header (1CB)	TS	III	B	A
B32-MPEN-TBD	Train B Steam Line Flowrate Instrumentation	UD	III	I	A

Table 6.2-47

Containment Penetrations Subject To Type A, B, and C Testing

Penetration Number (1)	Description	Location (3)/Room #	RCCV Sector	Penetration Type (4)	Leak Test Type (5)
B32-MPEN-TBD	Train B Steam Line Flowrate Instrumentation	UD	III	I	A
B32-MPEN-TBD	Train B Steam Line Flowrate Instrumentation	UD	III	I	A
B32-MPEN-TBD	Train B Steam Line Flowrate Instrumentation	UD	III	I	A
B32-MPEN-TBD	Train B Condensate Line Flowrate Instrumentation	UD	III	I	A
B32-MPEN-TBD	Train B Condensate Line Flowrate Instrumentation	UD	III	I	A
B32-MPEN-TBD	Train B Condensate Line Flowrate Instrumentation	UD	III	I	A
B32-MPEN-TBD	Train B Condensate Line Flowrate Instrumentation	UD	III	I	A
B32-MPEN-0003	Train C Steam Supply Line	TS	II	B	A
B32-MPEN-0019	Train C Purge Line From Steam Supply Line	TS	II	B	A
B32-MPEN-007	Train C Condensate Return	TS	II	B	A
B32-MPEN-0011	Train C Vent Line A From Upper Header (1CC)	TS	II	B	A
B32-MPEN-0015	Train C Vent Line A From Lower Header (1CC)	TS	II	B	A
B32-MPEN-TBD	Train C Steam Line Flowrate Instrumentation	UD	II	I	A
B32-MPEN-TBD	Train C Steam Line Flowrate Instrumentation	UD	II	I	A
B32-MPEN-TBD	Train C Steam Line Flowrate Instrumentation	UD	II	I	A
B32-MPEN-TBD	Train C Steam Line Flowrate Instrumentation	UD	II	I	A
B32-MPEN-TBD	Train C Condensate Line Flowrate Instrumentation	UD	II	I	A
B32-MPEN-TBD	Train C Condensate Line Flowrate Instrumentation	UD	II	I	A
B32-MPEN-TBD	Train C Condensate Line Flowrate Instrumentation	UD	II	I	A
B32-MPEN-TBD	Train C Condensate Line Flowrate Instrumentation	UD	II	I	A
B32-MPEN-0004	Train D Steam Supply Line	TS	IV	B	A
B32-MPEN-0020	Train D Purge Line From Steam Supply Line	TS	IV	B	A
B32-MPEN-0008	Train D Condensate Return	TS	IV	B	A
B32-MPEN-0012	Train D Vent Line A From Upper Header (1CD)	TS	IV	B	A

Table 6.2-47

Containment Penetrations Subject To Type A, B, and C Testing

Penetration Number (1)	Description	Location (3)/Room #	RCCV Sector	Penetration Type (4)	Leak Test Type (5)
B32-MPEN-0016	Train D Vent Line A From Lower Header (1CD)	TS	IV	B	A
B32-MPEN-TBD	Train D Steam Line Flowrate Instrumentation	UD	IV	I	A
B32-MPEN-TBD	Train D Steam Line Flowrate Instrumentation	UD	IV	I	A
B32-MPEN-TBD	Train D Steam Line Flowrate Instrumentation	UD	IV	I	A
B32-MPEN-TBD	Train D Steam Line Flowrate Instrumentation	UD	IV	I	A
B32-MPEN-TBD	Train D Condensate Line Flowrate Instrumentation	UD	IV	I	A
B32-MPEN-TBD	Train D Condensate Line Flowrate Instrumentation	UD	IV	I	A
B32-MPEN-TBD	Train D Condensate Line Flowrate Instrumentation	UD	IV	I	A
B32-MPEN-TBD	Train D Condensate Line Flowrate Instrumentation	UD	IV	I	A
C12: Control Rod Drive System (CRDS)					
C12-MPEN-TBD	FMCRD: 23 Hydraulic Lines (2)	LD /1110	I	M	A
C12-MPEN-TBD	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE	LD /1110	I	M	A
C12-MPEN-TBD	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE	LD /1110	I	M	A
C12-MPEN-TBD	FMCRD: 23 Hydraulic Lines (2)	LD /1120	II	M	A
C12-MPEN-TBD	FMCRD: 23 Hydraulic Lines (2)	LD /1120	II	M	A
C12-MPEN-TBD	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE	LD /1120	II	M	A
C12-MPEN-TBD	FMCRD: 23 Hydraulic Lines (2)	LD /1130	III	M	A
C12-MPEN-TBD	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE	LD /1130	III	M	A
C12-MPEN-TBD	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE	LD /1130	III	M	A
C12-MPEN-TBD	FMCRD: 23 Hydraulic Lines (2)	LD /1140	IV	M	A
C12-MPEN-TBD	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE	LD /1140	IV	M	A
C12-MPEN-TBD	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE	LD /1140	IV	M	A
C41: Standby Liquid Control System (SLCS)					
C41-MPEN-0001	Borated Liquid Injection (Train A)	UD	TBD	B	A
C41-MPEN-0002	Borated Liquid Injection (Train B)	UD	TBD	B	A
D11: Process Radiation Monitoring System (PRMS)					

Table 6.2-47

Containment Penetrations Subject To Type A, B, and C Testing

Penetration Number (1)	Description	Location (3)/Room #	RCCV Sector	Penetration Type (4)	Leak Test Type (5)
D11-MPEN-0001	Fission Product Rad Monitoring Extraction Line	UD	TBD	I	A
D11-MPEN-0002	Fission Product Rad Monitoring Return Line	UD	TBD	I	A
E50: Gravity Driven Cooling System (GDCS)					
E50-MPEN-TBD	GDCS Pool A Water Level	UD	TBD	I	A
E50-MPEN-TBD	GDCS Pool A Water Level	UD	TBD	I	A
E50-MPEN-TBD	GDCS Pool B/C Water Level	UD	TBD	I	A
E50-MPEN-TBD	GDCS Pool B/C Water Level	UD	TBD	I	A
E50-MPEN-TBD	GDCS Pool D Water Level	UD	TBD	I	A
E50-MPEN-TBD	GDCS Pool D Water Level	UD	TBD	I	A
G21: Fuel and Auxiliary Pools Cooling System (FAPCS)					
G21-MPEN-0001	Drywell Spray Discharge Line	UD	TBD	C	A
G21-MPEN-0002	Suppression Pool Return Line A	UD	TBD	C	A
G21-MPEN-0003	GDCS Pool Return Line	UD	TBD	C	A
G21-MPEN-0004	Suction Line from GDCS Pool	UD	TBD	C	A
G21-MPEN-0005	Suction Line A from Suppression Pool	LD	TBD	C	A
G21-MPEN-0006	Suppression Pool Return Line B	UD	TBD	C	A
G21-MPEN-0007	Suction Line B from Suppression Pool	LD	TBD	C	A
G21-MPEN-TBD	Reactor Well Drain Line	TS	TBD	C	A
G31: Reactor Water Cleanup and Shutdown Cooling System (RWCU/SDCS)					
G31-MPEN-0001	RPV Mid-Vessel Line (Train A)	LD	TBD	A	A
G31-MPEN-0002	RPV Mid-Vessel Line (Train B)	LD	TBD	A	A
G31-MPEN-0003	RPV Bottom Drain Line (Train A)	LD	TBD	B	A
G31-MPEN-0004	RPV Bottom Drain Line (Train B)	LD	TBD	B	A
G31-MPEN-0005	Sample Line (Train A)	LD	TBD	B	A
G31-MPEN-0006	Sample Line (Train B)	LD	TBD	B	A
P10: Makeup Water System (MWS)					
P10-MPEN-0001	Demin Water Drywell Distribution	TBD	TBD	C	A
P25: Chilled Water System (CWS)					
P25-MPEN-0001	CWS Supply Line Train A	UD	TBD	B	A
P25-MPEN-0003	CWS Supply Line Train B	UD	TBD	B	A
P25-MPEN-0002	CWS Return Line Train A	UD	TBD	B	A
P25-MPEN-0004	CWS Return Line Train B	UD	TBD	B	A
P51: Service Air System (SAS)					
P51-MPEN-TBD	Service Air Supply	UD	TBD	C	A
P51-MPEN-TBD	Breathing Air Supply	UD	TBD	C	A
P54: High Pressure Nitrogen Supply System (HPNSS)					

Table 6.2-47

Containment Penetrations Subject To Type A, B, and C Testing

Penetration Number (1)	Description	Location (3)/Room #	RCCV Sector	Penetration Type (4)	Leak Test Type (5)
P54-MPEN-0001	Supply to MSIV Accumulators	UD	TBD	B	A
P54-MPEN-0002	Supply to ADS and ICIV Accumulators	UD	TBD	B	A
T11: Containment Vessel: Equipment & Personnel Access Hatches					
T11-SPEN-TBD	LD Equipment Hatch	LD/1206	II/III	Hatch	B
T11-SPEN-TBD	LD Personnel Airlock	LD /1205	I/IV	Air Lock	B
T11-SPEN-TBD	Wetwell Access Hatch	WA/1600	III	Hatch	B
T11-SPEN-TBD	UD Equipment Hatch	UD /1740	IV	Hatch	B
T11-SPEN-TBD	UD Personnel Airlock	UD/1710	I	Air Lock	B
T11: Containment Vessel: Temporary Services During Outages & Spare Penetrations					
T11-MPEN-TBD	Temporary Services During Outages	LD	TBD	TBD	B
T11-MPEN-TBD	Temporary Services During Outages	LD	TBD	TBD	B
T11-MPEN-TBD	Temporary Services During Outages	UD	TBD	TBD	B
T11-MPEN-TBD	Temporary Services During Outages	UD	TBD	TBD	B
T11-MPEN-TBD	Temporary Services During Outages	WA	III	TBD	B
T11-MPEN-TBD	Spare Mechanical Penetration	TBD	TBD	S	A
T11-MPEN-TBD	Spare Mechanical Penetration	TBD	TBD	S	A
T11-MPEN-TBD	Spare Mechanical Penetration	TBD	TBD	S	A
T11-EPEN-TBD	Spare Electrical Penetration	TBD	I	E	B
T11-EPEN-TBD	Spare Electrical Penetration	TBD	II	E	B
T11-EPEN-TBD	Spare Electrical Penetration	TBD	III	E	B
T11-EPEN-TBD	Spare Electrical Penetration	TBD	IV	E	B
T11-EPEN-TBD	Spare Electrical Penetration	TBD	I	E	B
T11-EPEN-TBD	Spare Electrical Penetration	TBD	II	E	B
T11-EPEN-TBD	Spare Electrical Penetration	TBD	III	E	B
T11-EPEN-TBD	Spare Electrical Penetration	TBD	IV	E	B
T15: Passive Containment Cooling System (PCCS)					
T15-MPEN-0001	Condenser Steam Inlet Line A (6)	TS	I	B	A
T15-MPEN-0007	Condenser Condensate + Vent Line A1 (6)	TS	I	B	A
T15-MPEN-0008	Condenser Condensate + Vent Line A2 (6)	TS	I	B	A
T15-MPEN-0002	Condenser Steam Inlet Line B	TS	I /III	B	A
T15-MPEN-0009	Condenser Condensate + Vent Line B1	TS	I /III	B	A
T15-MPEN-0010	Condenser Condensate + Vent Line B2	TS	I /III	B	A

Table 6.2-47

Containment Penetrations Subject To Type A, B, and C Testing

Penetration Number (1)	Description	Location (3)/Room #	RCCV Sector	Penetration Type (4)	Leak Test Type (5)
T15-MPEN-0003	Condenser Steam Inlet Line C (6)	TS	III	B	A
T15-MPEN-0011	Condenser Condensate + Vent Line C1 (6)	TS	III	B	A
T15-MPEN-0012	Condenser Condensate + Vent Line C2 (6)	TS	III	B	A
T15-MPEN-0004	Condenser Steam Inlet Line D	TS	II	B	A
T15-MPEN-0013	Condenser Condensate + Vent Line D1	TS	II	B	A
T15-MPEN-0014	Condenser Condensate + Vent Line D2	TS	II	B	A
T15-MPEN-0005	Condenser Steam Inlet Line E	TS	II / IV	B	A
T15-MPEN-0015	Condenser Condensate + Vent Line E1	TS	II / IV	B	A
T15-MPEN-0016	Condenser Condensate + Vent Line E2	TS	II / IV	B	A
T15-MPEN-0006	Condenser Steam Inlet Line F	TS	IV	B	A
T15-MPEN-0017	Condenser Condensate + Vent Line F1	TS	IV	B	A
T15-MPEN-0018	Condenser Condensate + Vent Line F2	TS	IV	B	A
T31: Containment Inerting System (CIS)					
T31-MPEN-0001	Upper Drywell Injection Line	UD	TBD	C	A
T31-MPEN-0002	Suppression Pool Airspace Injection Line	WA	TBD	C	A
T31-MPEN-0003	Main Exhaust Line (Lower Drywell)	LD	TBD	C	A
T31-MPEN-0004	Second Exhaust Line (Suppression Pool Airspace)	UD	TBD	C	A
T31-MPEN-TBD	Containment Pressure Test (GDCS Pool)	UW	TBD	C	A
T31-MPEN-TBD	Containment Pressure Test (Lower Drywell)	WA	TBD	C	A
T62: Containment Monitoring System (CMS)					
T62-MPEN-TBD	H2-O2 & Drywell Gas Sample Line From Upper Drywell (Loop A)	UD	TBD	C	A
T62-MPEN-TBD	H2-O2 & Drywell Gas Sample Return Line to Upper Drywell (Loop A)	UD	TBD	C	A
T62-MPEN-TBD	H2-O2 & Drywell Gas Sample Line From Wetwell Airspace (Loop A)	WA	TBD	C	A

Table 6.2-47

Containment Penetrations Subject To Type A, B, and C Testing

Penetration Number (1)	Description	Location (3)/Room #	RCCV Sector	Penetration Type (4)	Leak Test Type (5)
T62-MPEN-TBD	H2-O2 & Drywell Gas Sample Return Line to Wetwell Airspace (Loop A)	WA	TBD	C	A
T62-MPEN-TBD	H2-O2 & Drywell Gas Sample Line From Upper Drywell (Loop B)	UD	TBD	C	A
T62-MPEN-TBD	H2-O2 & Drywell Gas Sample Return Line to Upper Drywell (Loop B)	UD	TBD	C	A
T62-MPEN-TBD	H2-O2 & Wetwell Gas Sample Line From Wetwell Airspace (Loop B)	WA	TBD	C	A
T62-MPEN-TBD	H2-O2 & Wetwell Gas Sample Return Line to Wetwell Airspace (Loop B)	WA	TBD	C	A
T62-MPEN-TBD	Suppression Pool Water Level Monitoring-Wide Range	WP	TBD	I	A
T62-MPEN-TBD	Suppression Pool Water Level Monitoring-Wide Range	WP	TBD	I	A
T62-MPEN-TBD	Suppression Pool Water Level Monitoring-Wide Range	WP	TBD	I	A
T62-MPEN-TBD	Suppression Pool Water Level Monitoring-Wide Range	WP	TBD	I	A
T62-MPEN-TBD	Suppression Pool Water Level Monitoring-Narrow Range	WP	TBD	I	A
T62-MPEN-TBD	Suppression Pool Water Level Monitoring-Narrow Range	WP	TBD	I	A
T62-MPEN-TBD	Suppression Pool Water Level Monitoring-Narrow Range	WP	TBD	I	A
T62-MPEN-TBD	Suppression Pool Water Level Monitoring-Narrow Range	WP	TBD	I	A
T62-MPEN-TBD	Drywell Pressure Monitoring-Wide Range (Post-Accident Monitoring)	UD	TBD	I	A
T62-MPEN-TBD	Drywell Pressure Monitoring-Wide Range (Post-Accident Monitoring)	UD	TBD	I	A
T62-MPEN-TBD	Drywell Pressure Monitoring-Wide Range (Diverse Protection System)	UD	TBD	I	A
T62-MPEN-TBD	Drywell Pressure Monitoring-Wide Range (Diverse Protection System)	UD	TBD	I	A

Table 6.2-47

Containment Penetrations Subject To Type A, B, and C Testing

Penetration Number (1)	Description	Location (3)/Room #	RCCV Sector	Penetration Type (4)	Leak Test Type (5)
T62-MPEN-TBD	Drywell Pressure Monitoring- Wide Range (Diverse Protection System)	UD	TBD	I	A
T62-MPEN-TBD	Drywell Pressure Monitoring- Wide Range (Diverse Protection System)	UD	TBD	I	A
T62-MPEN-TBD	Drywell Pressure Monitoring – Narrow Range	UD	TBD	I	A
T62-MPEN-TBD	Drywell Pressure Monitoring – Narrow Range	UD	TBD	I	A
T62-MPEN-TBD	Drywell Pressure Monitoring – Narrow Range	UD	TBD	I	A
T62-MPEN-TBD	Drywell Pressure Monitoring – Narrow Range	UD	TBD	I	A
T62-MPEN-TBD	Wetwell Vapor Pressure Monitoring	WA	TBD	I	A
T62-MPEN-TBD	Wetwell Vapor Pressure Monitoring	WA	TBD	I	A
T62-MPEN-TBD	Drywell/Wetwell Differential Pressure Monitoring (UD)	UD	TBD	I	A
T62-MPEN-TBD	Drywell/Wetwell Differential Pressure Monitoring (WA)	WA	TBD	I	A
T62-MPEN-TBD	Drywell/Wetwell Differential Pressure Monitoring (LD)	LD	TBD	I	A
T62-MPEN-TBD	Drywell/Wetwell Differential Pressure Monitoring (WA)	WA	TBD	I	A
T62-MPEN-TBD	Lower Drywell Post-LOCA Water Level Monitoring Line A	LD	TBD	I	A
T62-MPEN-TBD	Lower Drywell Post-LOCA Water Level Monitoring Line B	LD	TBD	I	A
T62-MPEN-TBD	Upper Drywell Post-LOCA Water Level Monitoring Line A	LD	TBD	I	A
T62-MPEN-TBD	Upper Drywell Post-LOCA Water Level Monitoring Line B	LD	TBD	I	A
U50: Equipment and Floor Drain System (EFDS)					
U50-MPEN-TBD	Drywell LCW Sump Discharge Line (3)	LD	TBD	B	A
U50-MPEN-TBD	Drywell HCW Sump Discharge Line (3)	LD	TBD	B	A
Electrical Penetrations					
R31: Raceway System					
R31-EPEN-TBD	Div 1 Electrical Penetration	LD /1312	I	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	LD /1300	I	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	LD /1300	I	E	B

Table 6.2-47

Containment Penetrations Subject To Type A, B, and C Testing

Penetration Number (1)	Description	Location (3)/Room #	RCCV Sector	Penetration Type (4)	Leak Test Type (5)
R31-EPEN-TBD	Div 2 Electrical Penetration	LD /1322	II	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	LD/1302	II	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	LD/1302	II	E	B
R31-EPEN-TBD	Div 3 Electrical Penetration	LD /1332	III	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	LD/1301	III	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	LD/1301	III	E	B
R31-EPEN-TBD	Div 4 Electrical Penetration	LD /1342	IV	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	LD/1303	IV	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	LD/1303	IV	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	WP/1600	I	E	B
R31-EPEN-TBD	Div 1 Electrical Penetration	WP/1610	I	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	WP/1600	II	E	B
R31-EPEN-TBD	Electrical Penetration	WP/1600	II	E	B
R31-EPEN-TBD	Div 2 Electrical Penetration	WP/1620	II	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	WP/1600	III	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	WP/1600	III	E	B
R31-EPEN-TBD	Div 3 Electrical Penetration	WP/1630	III	E	B
R31-EPEN-TBD	Non-Div Electrical Penetration	WP/1600	IV	E	B
R31-EPEN-TBD	Div 4 Electrical Penetration	WP/1640	IV	E	B
R31-EPEN-TBD	Div 1 Electrical Penetration	UD/1711	I	E	B
R31-EPEN-TBD	Div 1 Electrical Penetration	UD/1711	I	E	B
R31-EPEN-TBD	Div 2 Electrical Penetration	UD/1721	II	E	B
R31-EPEN-TBD	Div 2 Electrical Penetration	UD/1721	II	E	B
R31-EPEN-TBD	Div 3 Electrical Penetration	UD/1731	III	E	B
R31-EPEN-TBD	Div 3 Electrical Penetration	UD/1731	III	E	B
R31-EPEN-TBD	Div 4 Electrical Penetration	UD/1741	IV	E	B
R31-EPEN-TBD	Div 4 Electrical Penetration	UD/1741	IV	E	B

Notes:

(1) Penetration numbering:
EPEN = Electrical Penetrations
MPEN = Mechanical penetrations
SPEN = Structural penetration, Hatch, Equip or Personnel

(2) Estimation is based on 269 FMCRD hydraulic lines and 12 sleeves

(3) UD – UPPER DRYWELL
ST – STEAM TUNNEL
TS – TOP SLAB
LD – LOWER DRYWELL
WA – WETWELL AIRSPACE
WP – WETWELL POOL
TBD - TO BE DETERMINED
HCW - HIGH CONDUCTIVITY WASTE
LCW - LOW CONDUCTIVITY WASTE

(4) Penetration type:
Type A = Penetration with thermal sleeve for High Energy Pipelines; (Main Steam & Feed Water Lines) (Fig. 3.8-6)
Type B = Penetration with thermal sleeve for Low / High Energy Flow (DCD, Rev.3 Fig. 3.8-6 and 3.8-7)
Type C = Embedded penetration without thermal sleeve (Cold Type for flow Tmax<93°C(200°F)) (Fig. 3.8-8)
Type E = Penetration with flanges (Electrical, Maintenance, etc) (Fig. 3.8-10)
Type I = Instrumentation and Radiation Monitoring. (TBD)
Type M = Multiple penetration with sleeve (Fig. 3.8-9)
Type S = Spare Mechanical Penetration (TBD)

(5) All penetration will be subject to the Type A, Integrated Leak Rate Test (ILRT)
All penetrations excluded from Type B testing are welded penetrations and do not include any resilient seals in their design.

(6) PCCS Pool designations are subject to change

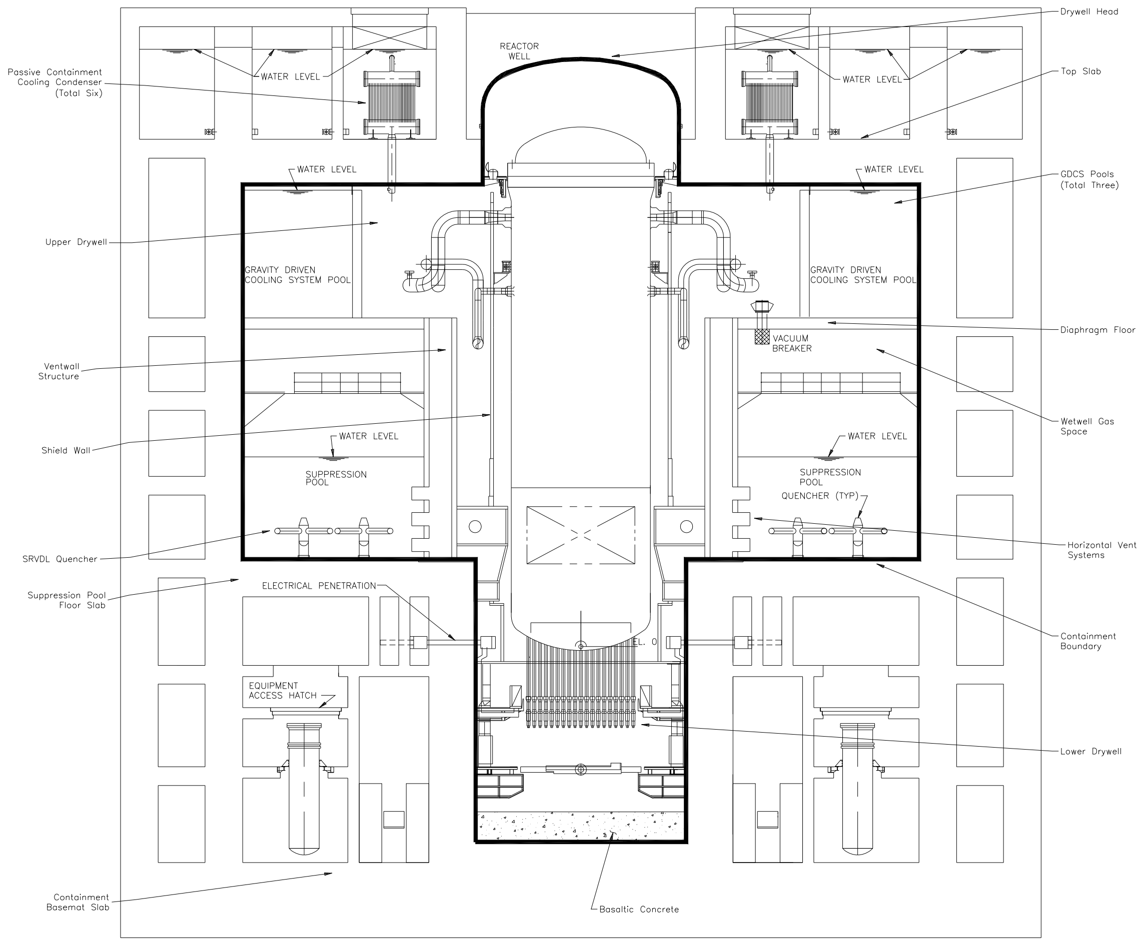


Figure 6.2-1. Containment System

Figure 6.2-2. IC/PCC Pools Configuration

{{{Security-Related Information - Withhold Under 10 CFR 2.390.}}}

Figure 6.2-3. GDCS Pools Configuration

{{{Security-Related Information - Withhold Under 10 CFR 2.390}}}

Figure 6.2-4. [Not Used]

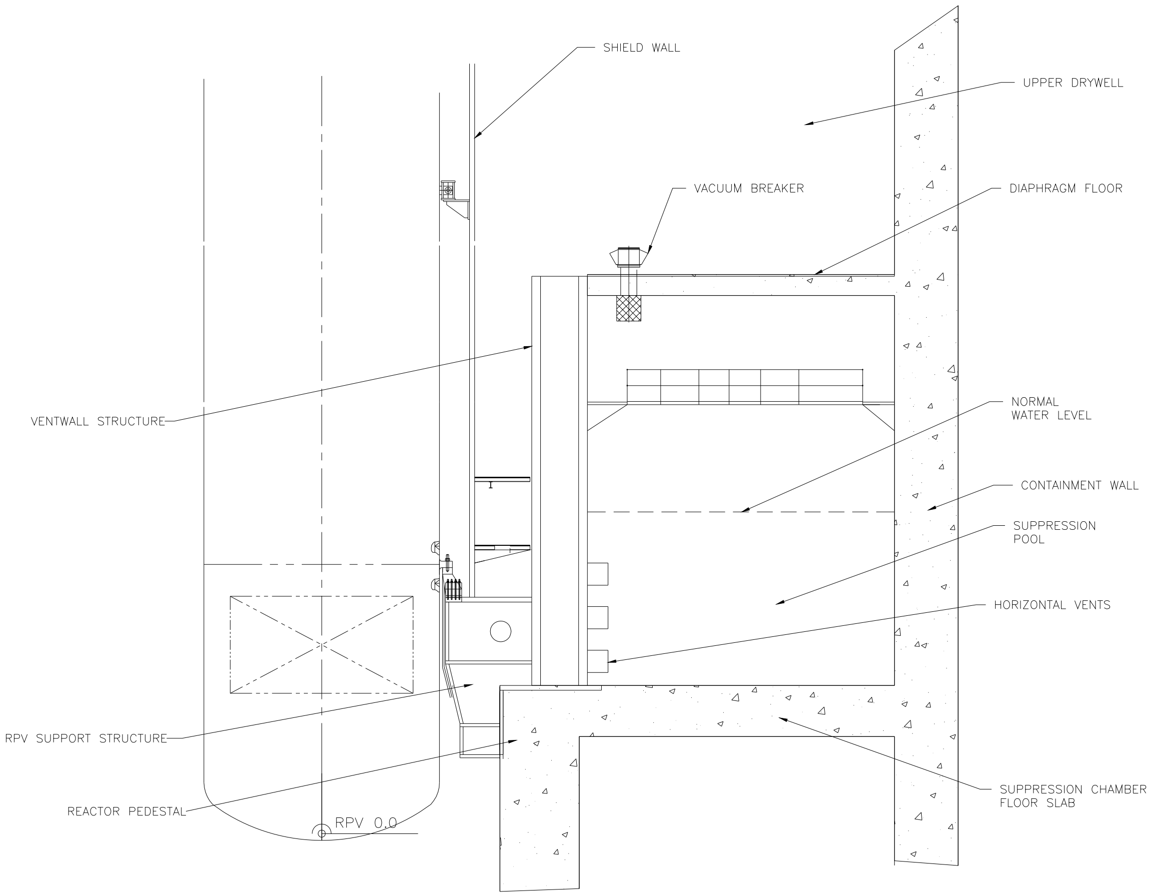


Figure 6.2-5. Horizontal Vent System Configuration

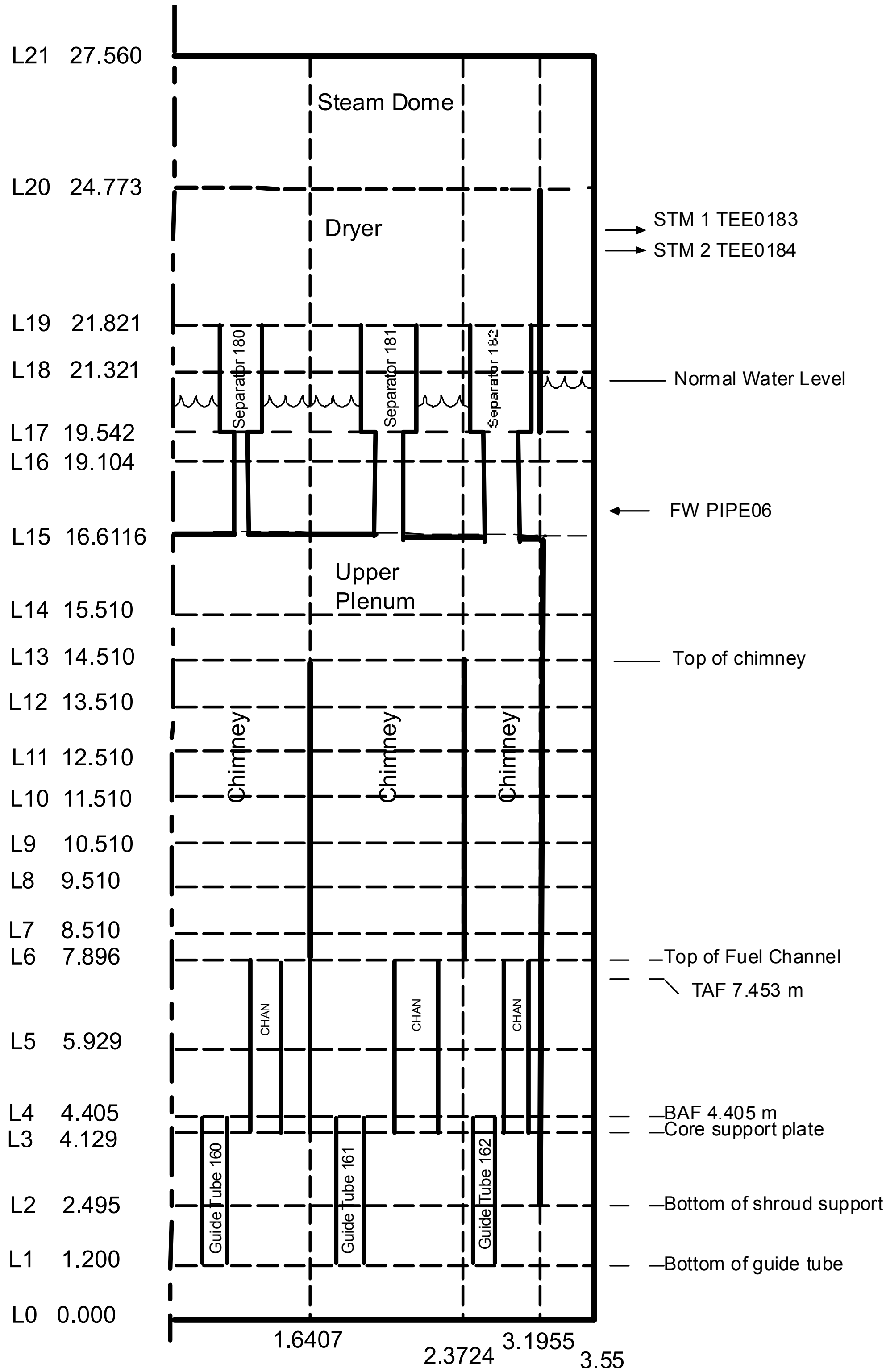


Figure 6.2-6. TRACG Nodalization of the ESBWR RPV

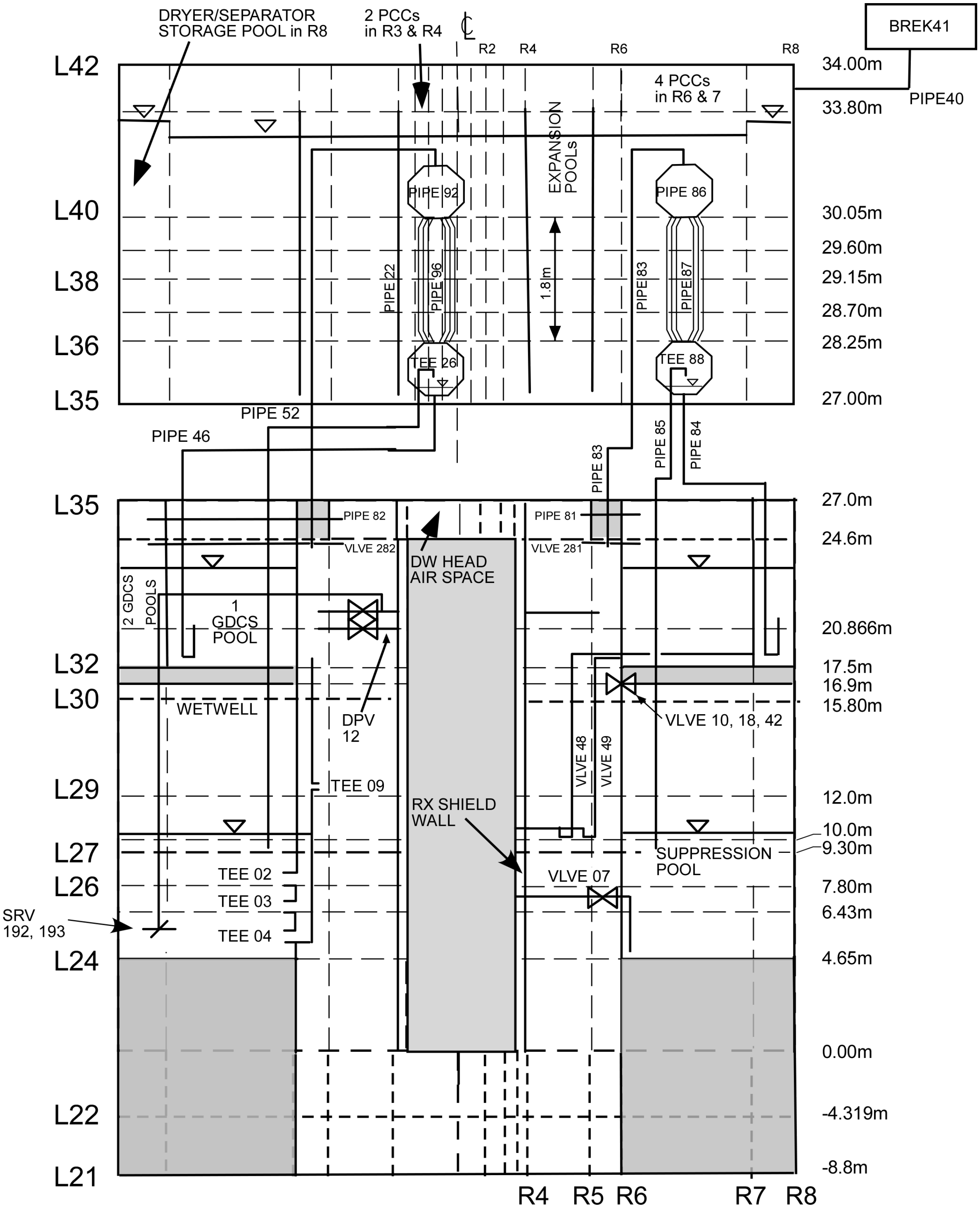


Figure 6.2-7. TRACG Nodalization of the ESBWR Containment

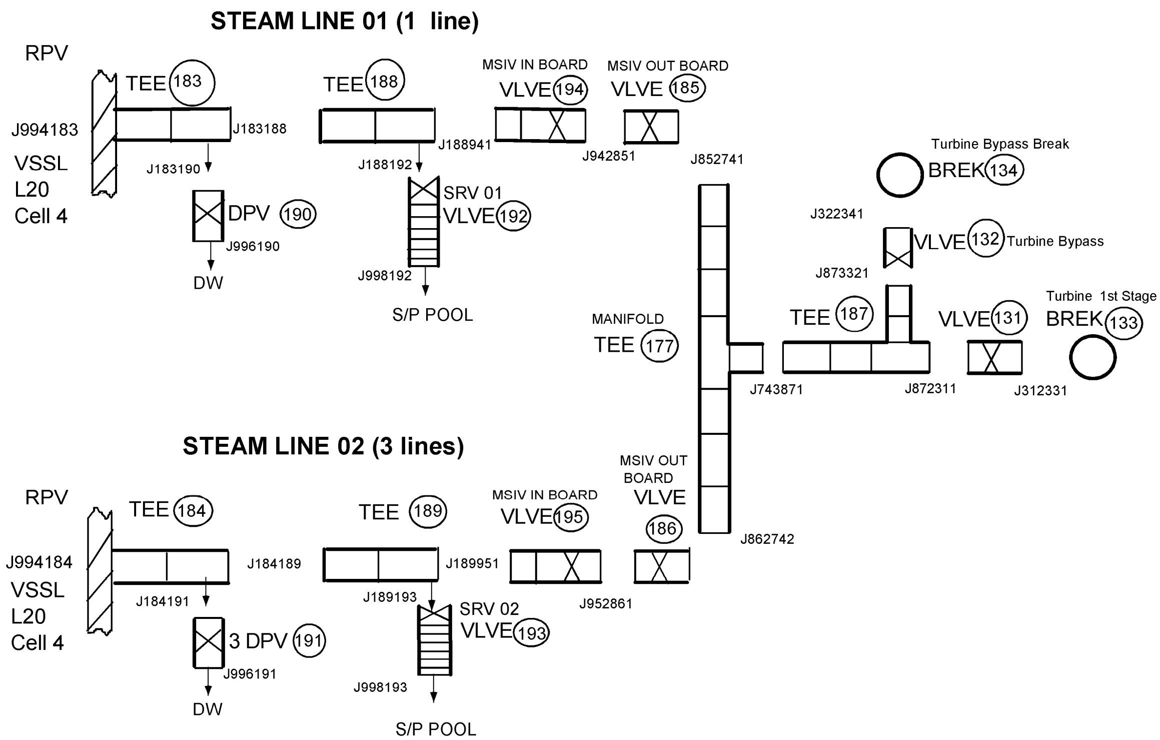
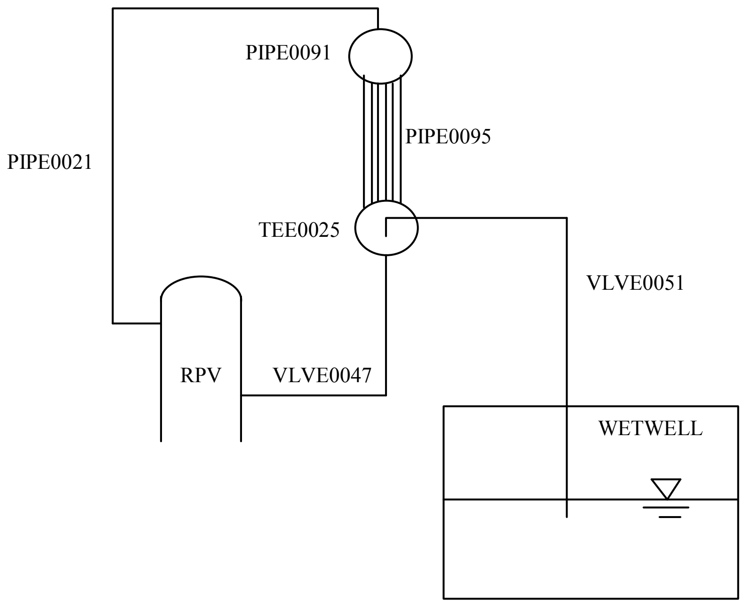


Figure 6.2-8. TRACG Nodalization of the ESBWR Main Steam Lines



Component	Description
PIPE0021	IC Steam Supply Line
PIPE0091	IC Steam Box
PIPE0095	IC Tube
TEE0025	IC Water Box
VLVE0051	IC Vent Line
VLVE0047	IC Drain Tank and Drain Line

Figure 6.2-8a. TRACG Nodalization of the ESBWR Isolation Condenser System

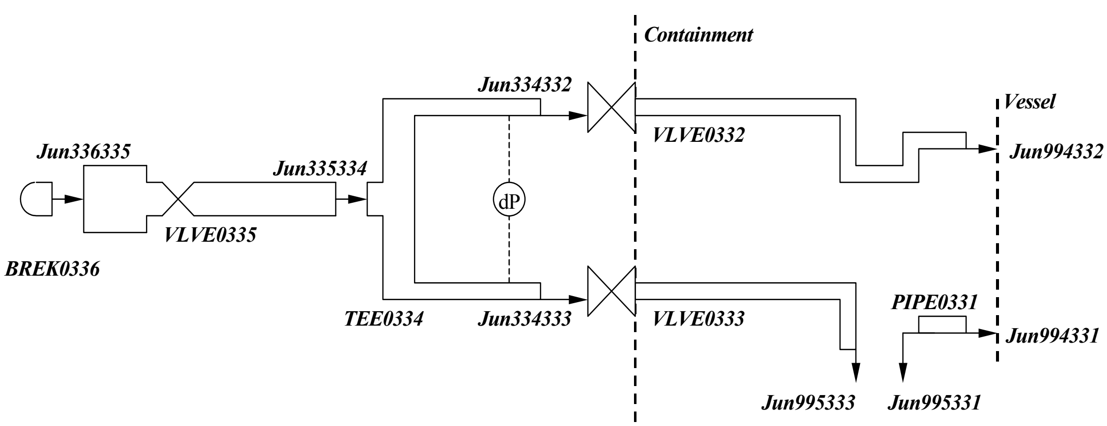


Figure 6.2-8b. TRACG Nodalization of the ESBWR Feedwater Line System

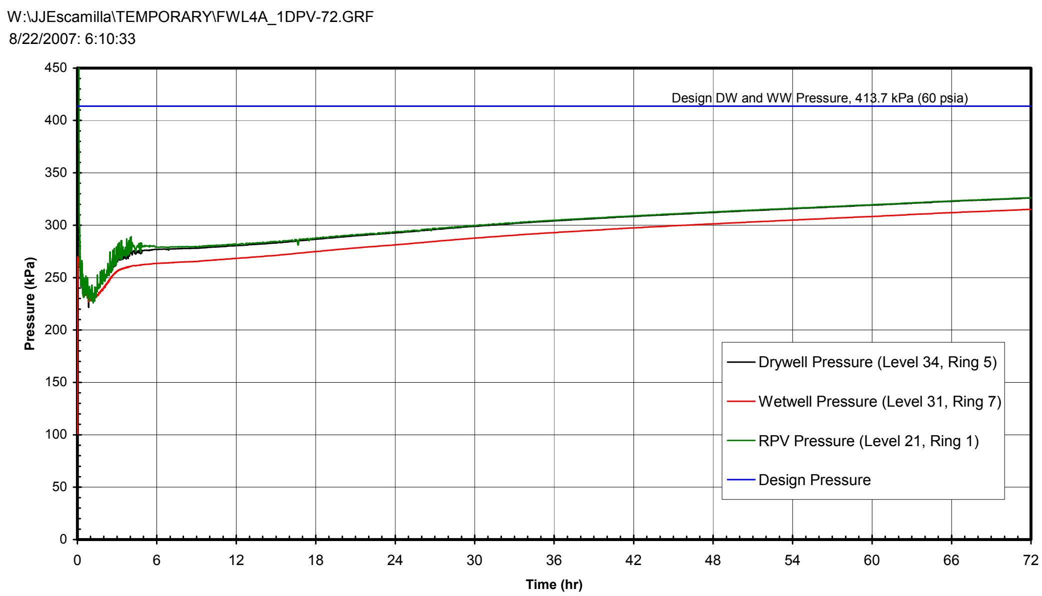


Figure 6.2-9a1. Feedwater Line Break (Nominal Case) – Containment Pressures (72 hrs)

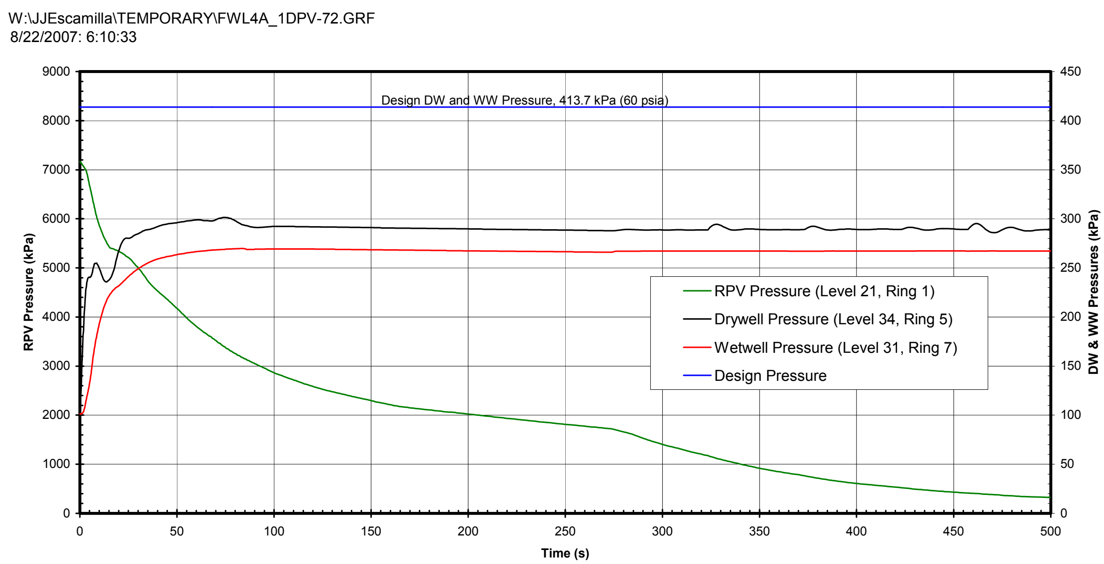


Figure 6.2-9a2. Feedwater Line Break (Nominal Case) – Containment Pressures (500 s)

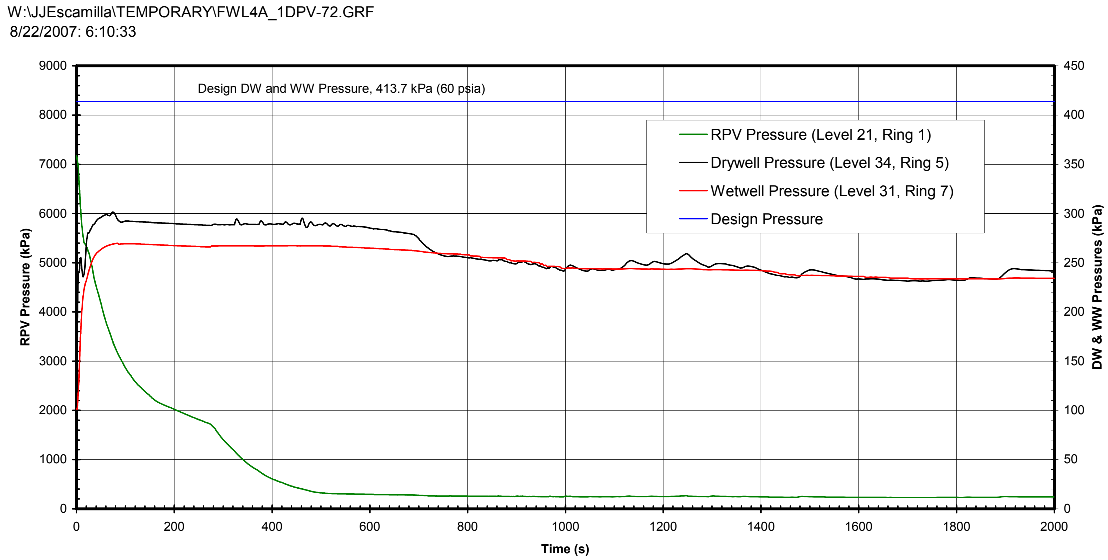


Figure 6.2-9a3. Feedwater Line Break (Nominal Case) – Containment Pressures (2000 s)

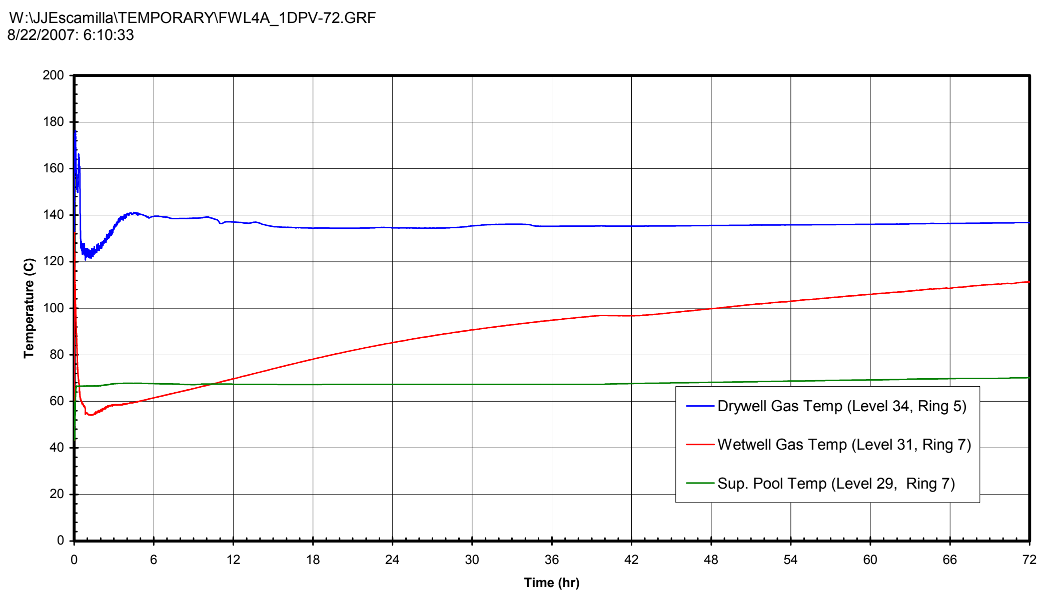


Figure 6.2-9b1. Feedwater Line Break (Nominal Case) – Containment Temperatures (72hrs)

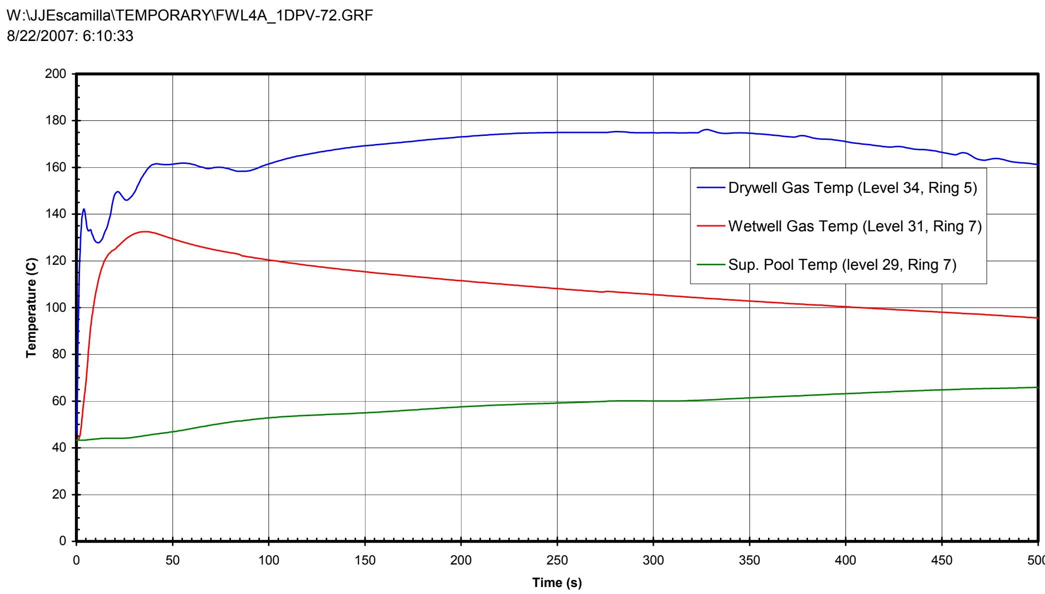


Figure 6.2-9b2. Feedwater Line Break (Nominal Case) – Containment Temperatures (500 s)

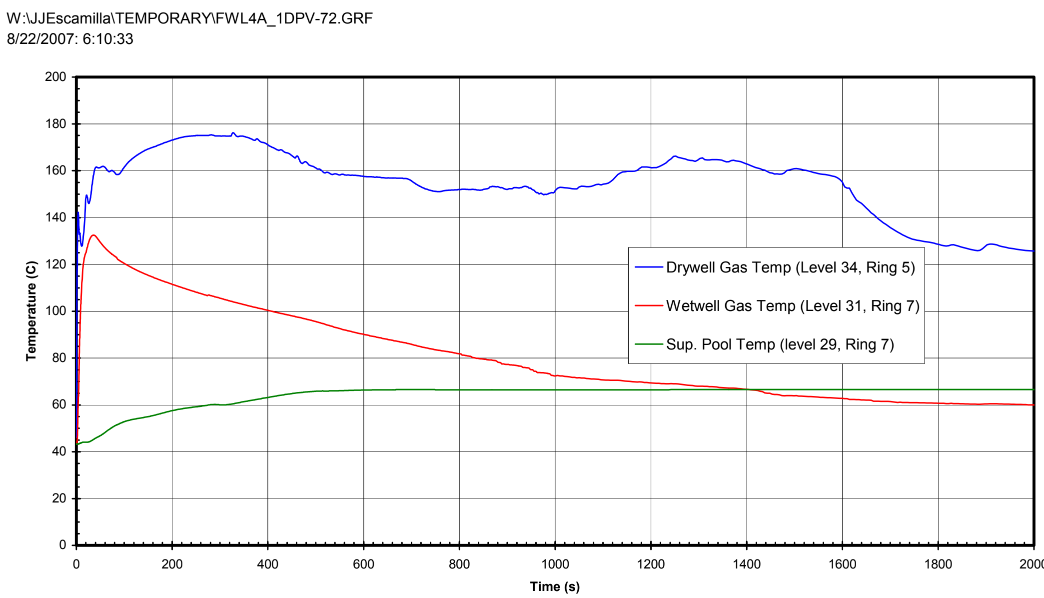


Figure 6.2-9b3. Feedwater Line Break (Nominal Case) – Containment Temperatures (2000 s)

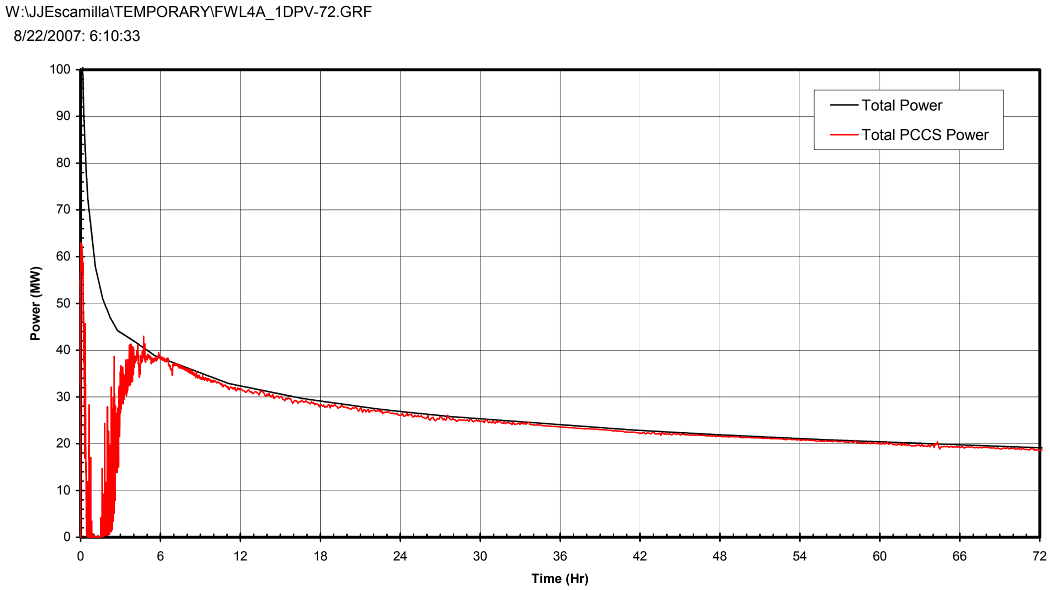


Figure 6.2-9c1. Feedwater Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

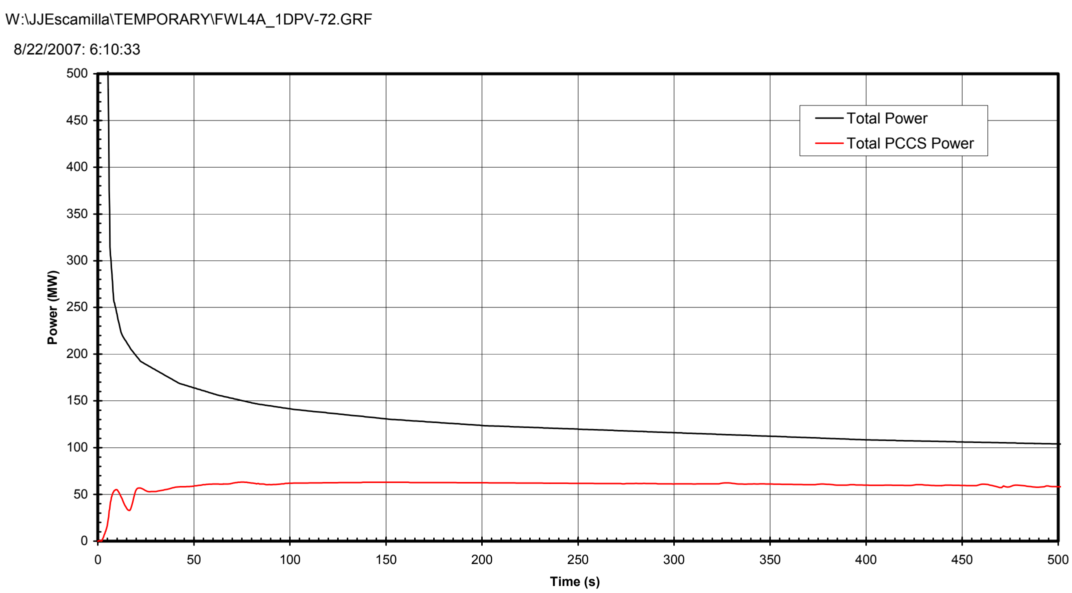


Figure 6.2-9c2. Feedwater Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (500 s)

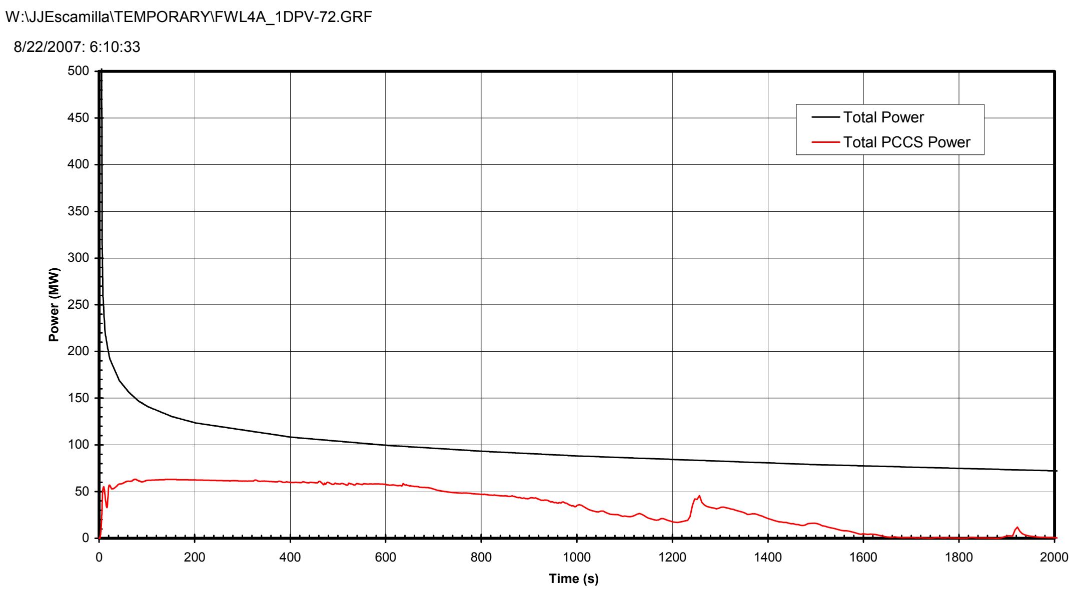


Figure 6.2-9c3. Feedwater Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (2000 s)

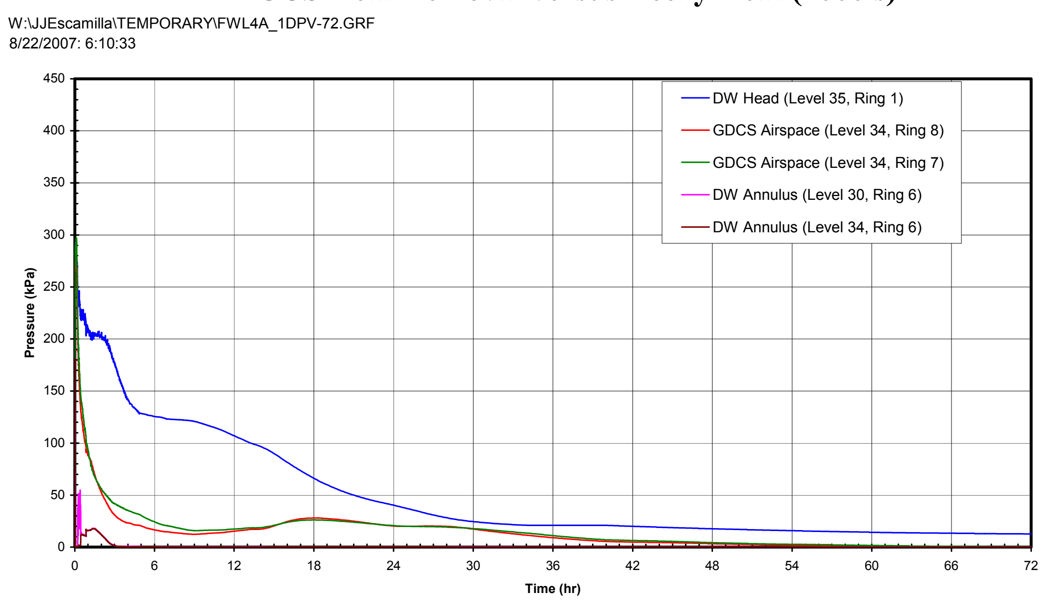


Figure 6.2-9d1. Feedwater Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (72 hrs)

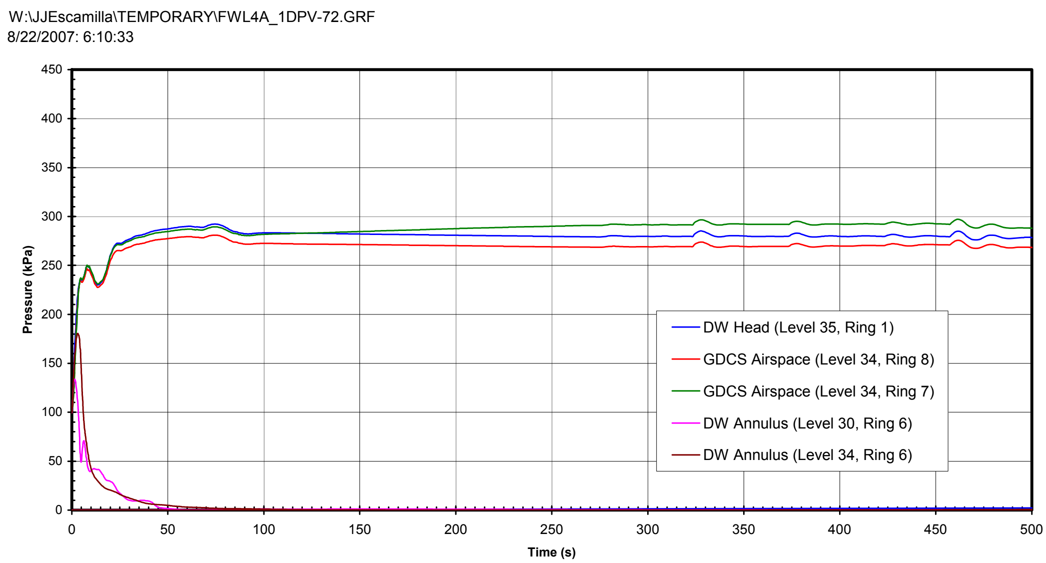


Figure 6.2-9d2. Feedwater Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (500 s)

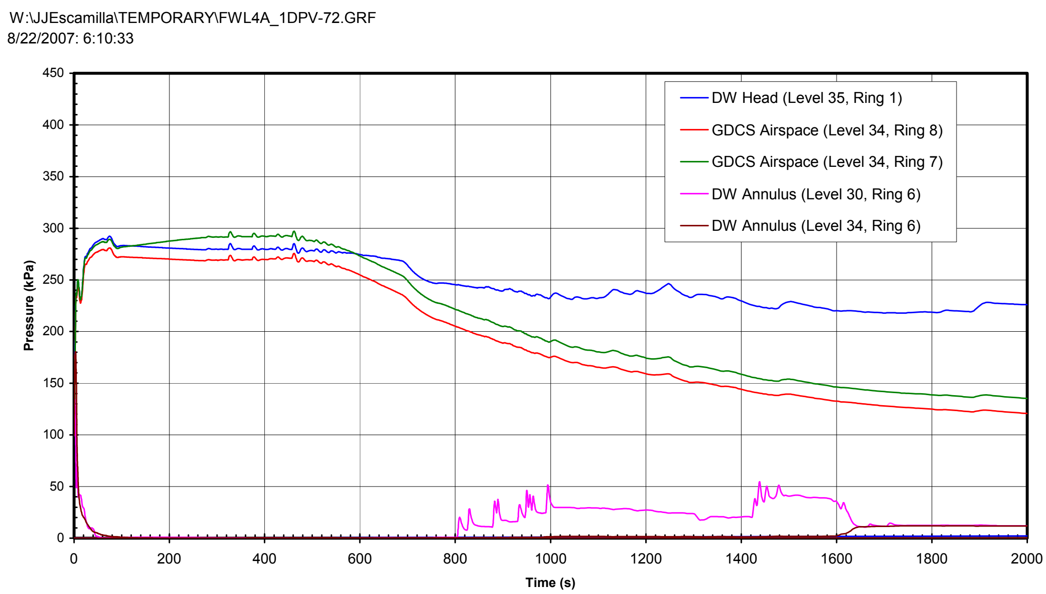


Figure 6.2-9d3. Feedwater Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (2000 s)

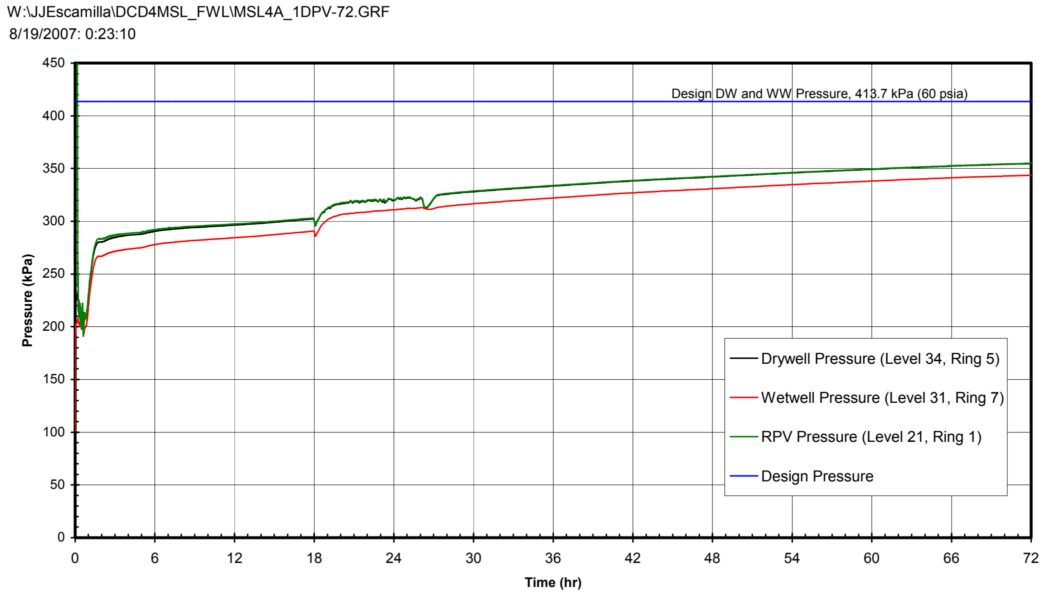


Figure 6.2-10a1. Main Steam Line Break (Nominal Case) – Containment Pressures (72 hrs)

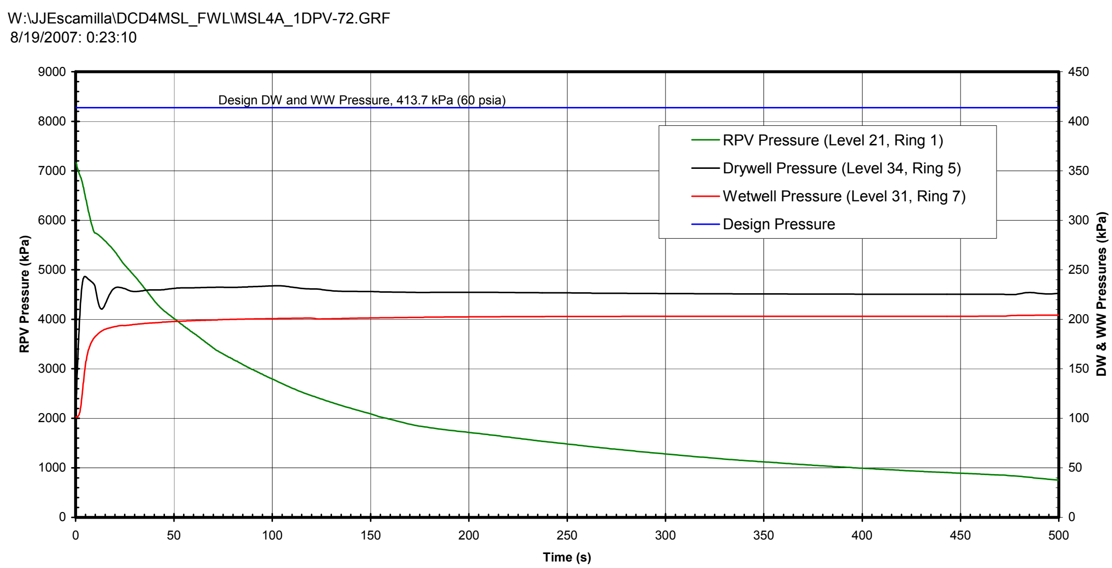


Figure 6.2-10a2. Main Steam Line Break (Nominal Case) – Containment Pressures (500 s)

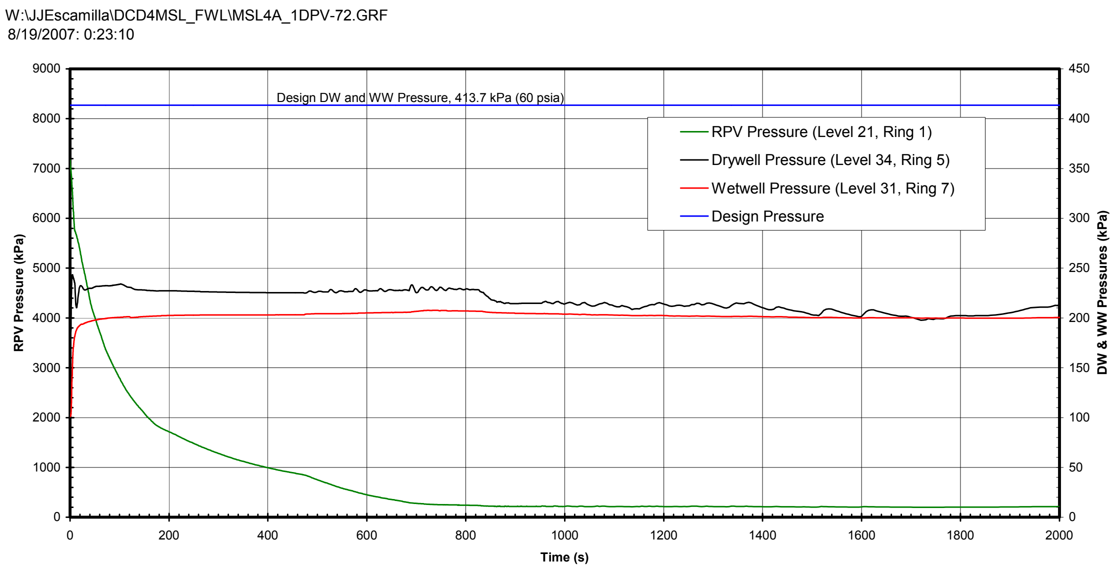


Figure 6.2-10a3. Main Steam Line Break (Nominal Case) – Containment Pressures (2000 s)

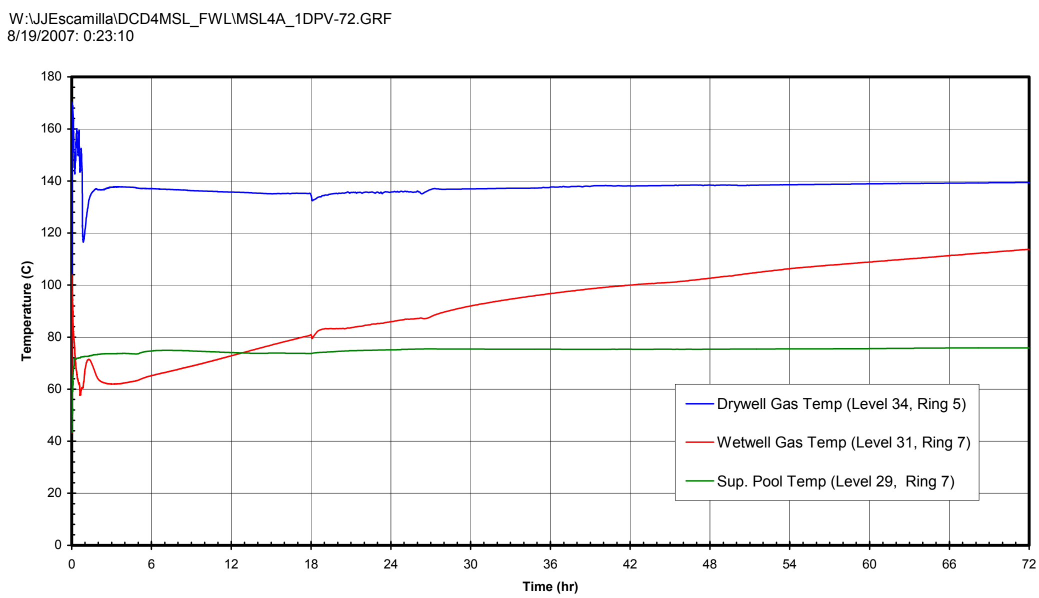


Figure 6.2-10b1. Main Steam Line Break (Nominal Case) – Containment Temperatures (72 hrs)

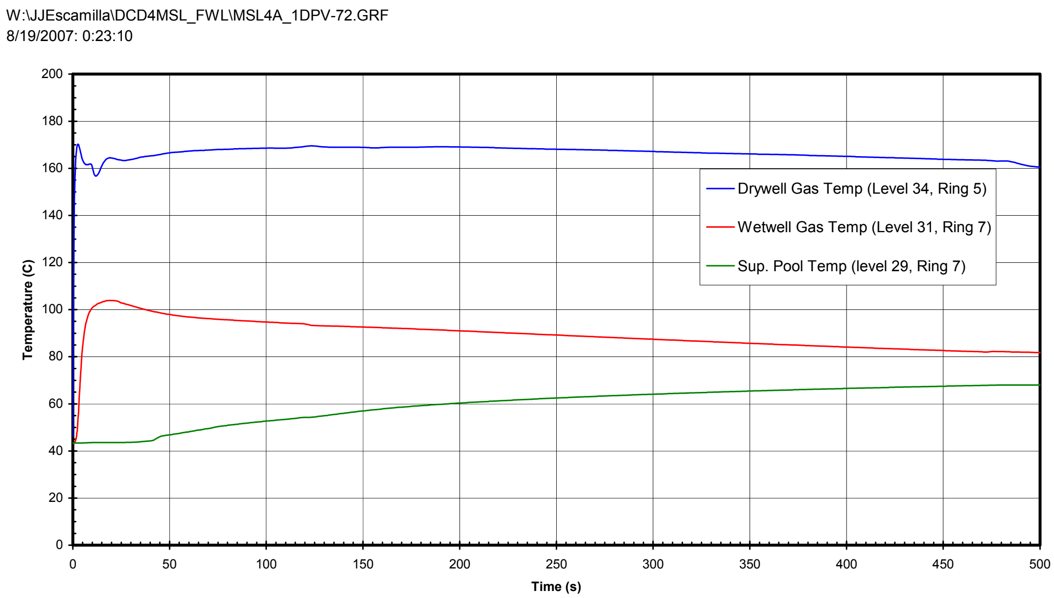


Figure 6.2-10b2. Main Steam Line Break (Nominal Case) – Containment Temperatures (500 s)

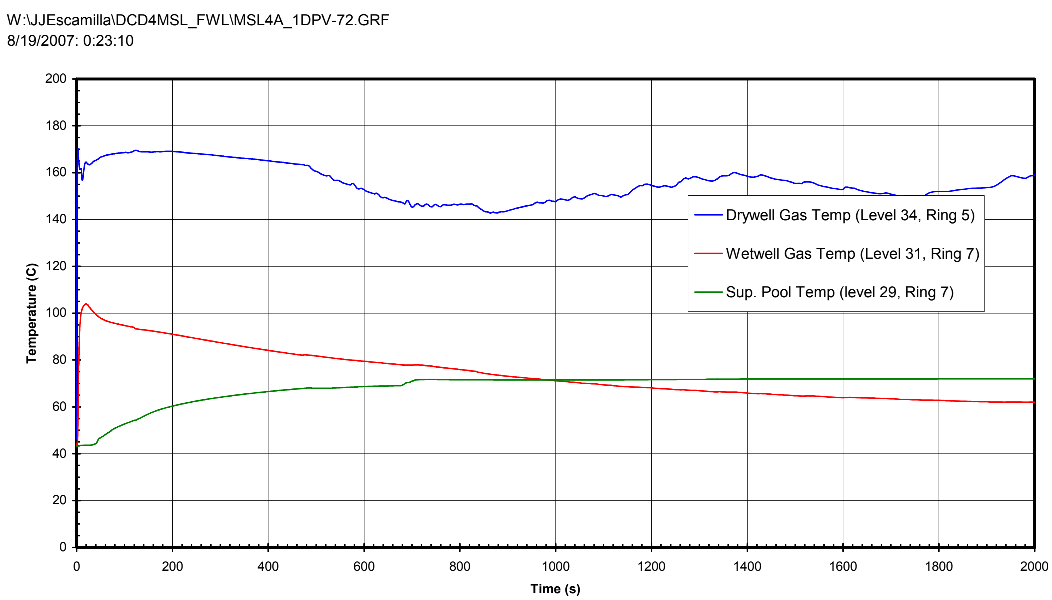


Figure 6.2-10b3. Main Steam Line Break (Nominal Case) – Containment Temperatures (2000 s)

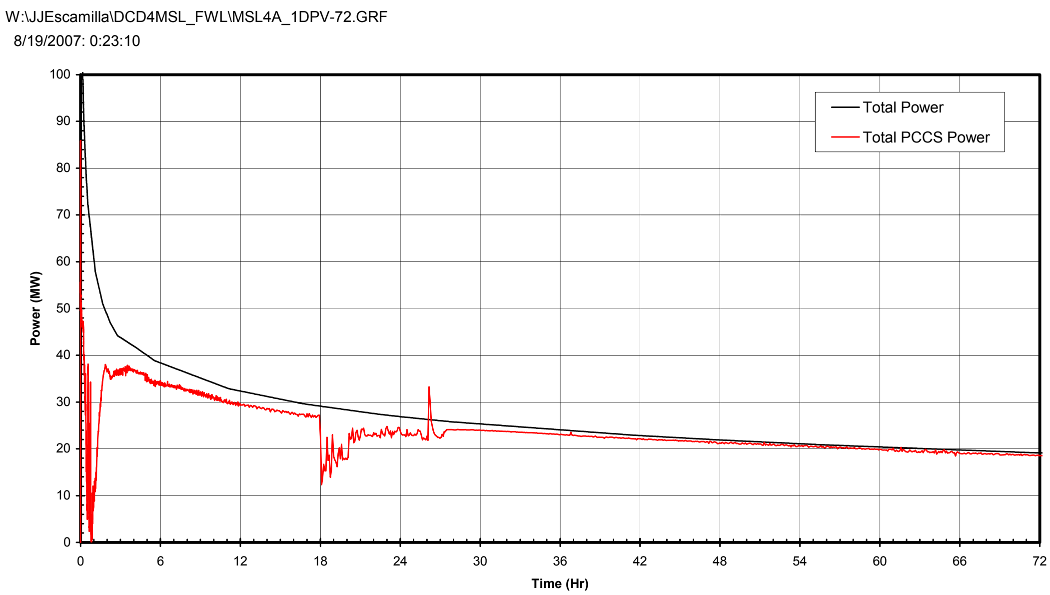


Figure 6.2-10c1. Main Steam Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

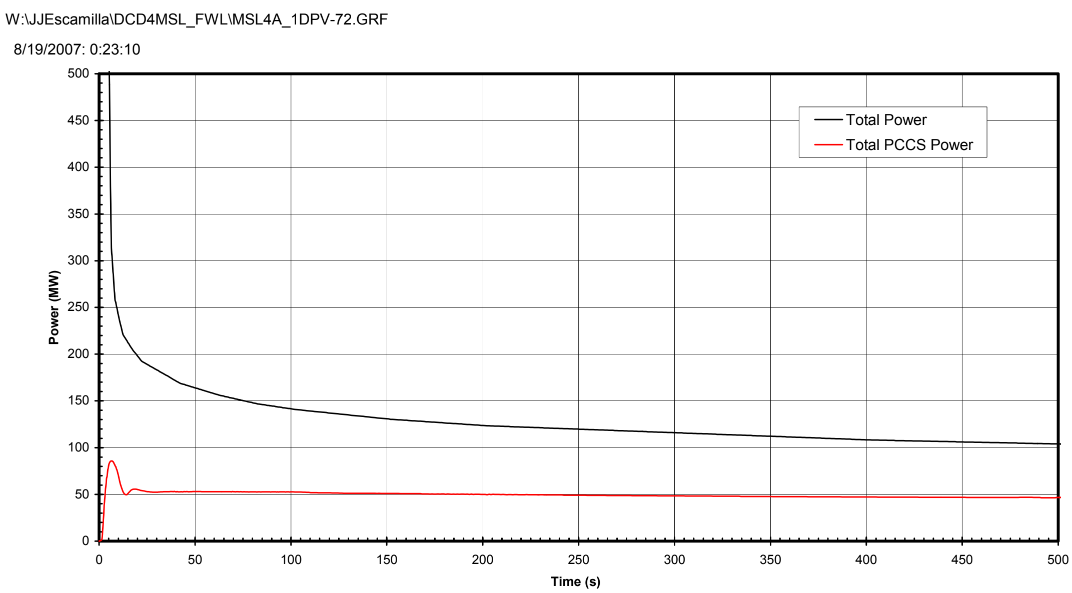


Figure 6.2-10c2. Main Steam Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (500 s)

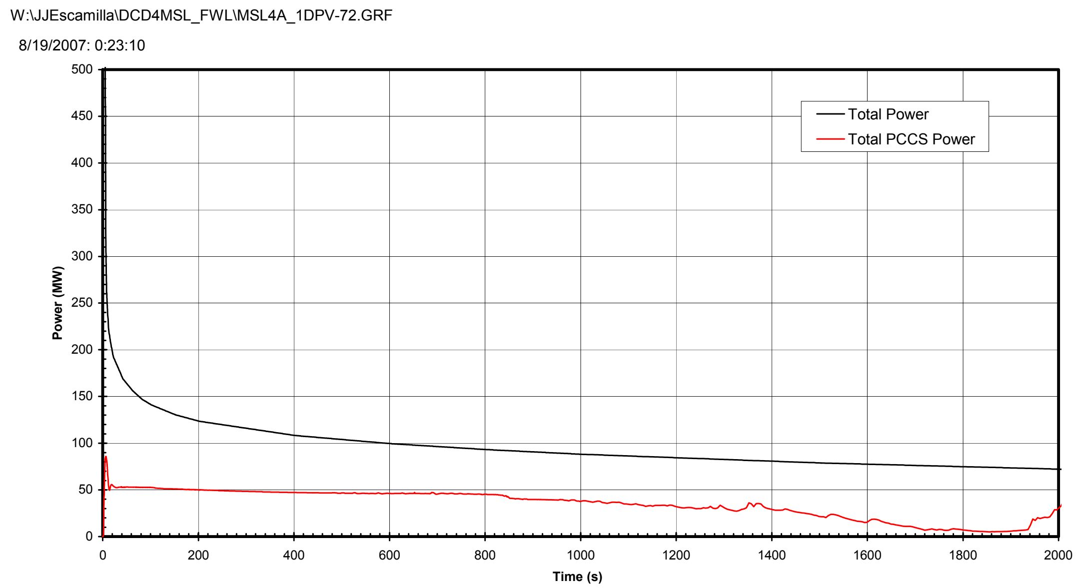


Figure 6.2-10c3. Main Steam Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (2000 s)

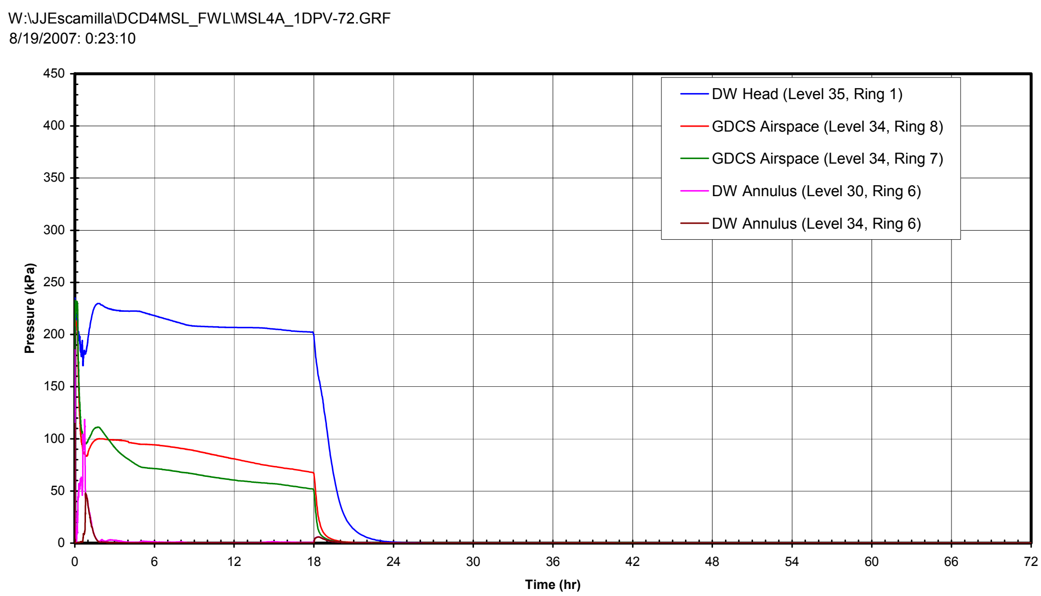


Figure 6.2-10d1. Main Steam Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (72 hrs)

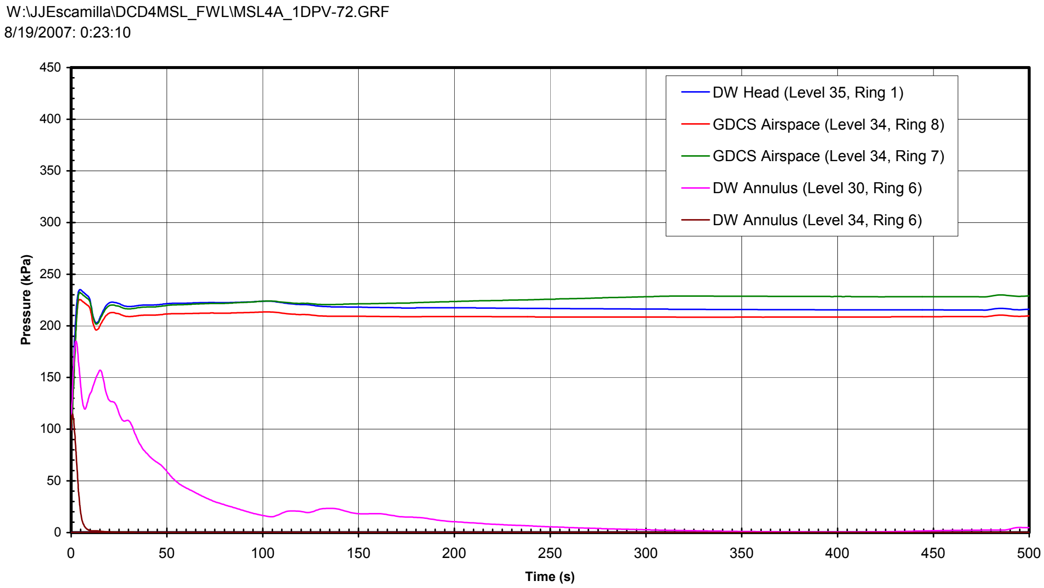


Figure 6.2-10d2. Main Steam Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (500 s)

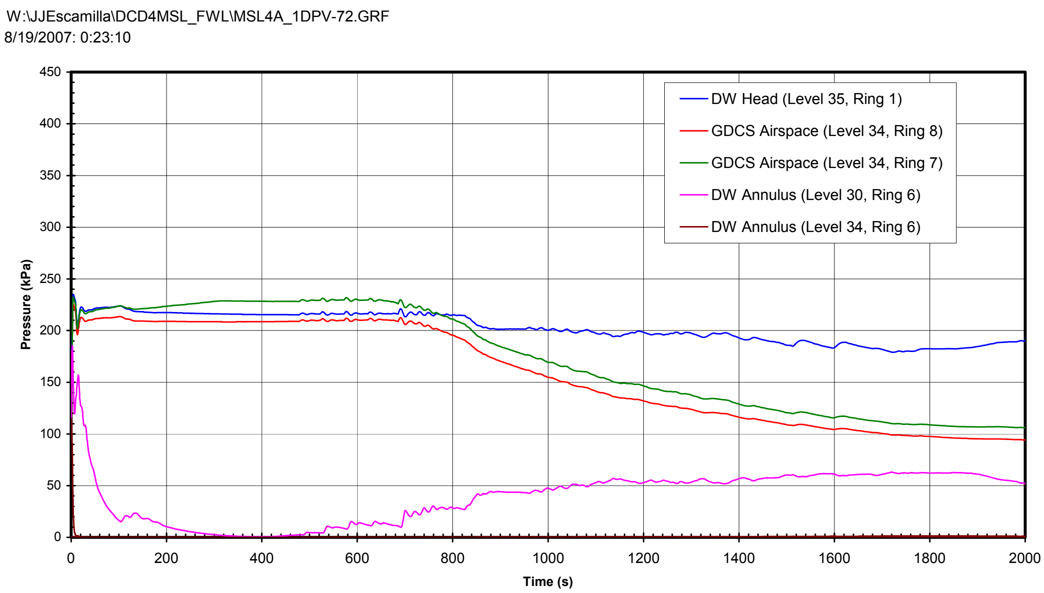


Figure 6.2-10d3. Main Steam Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (2000 s)

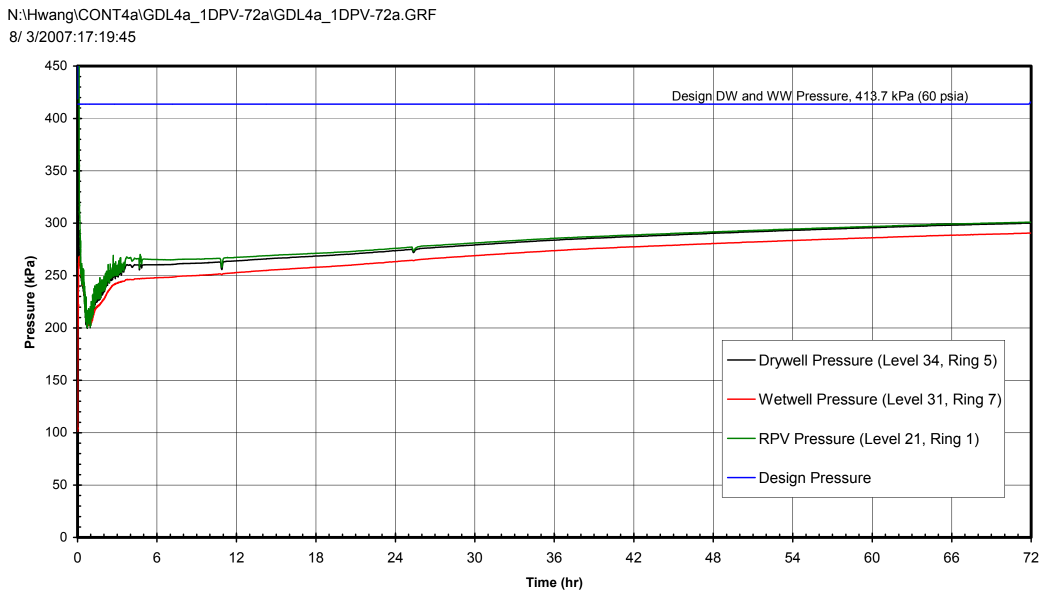


Figure 6.2-11a1. GDCS Line Break (Nominal Case) – Containment Pressures (72 hrs)

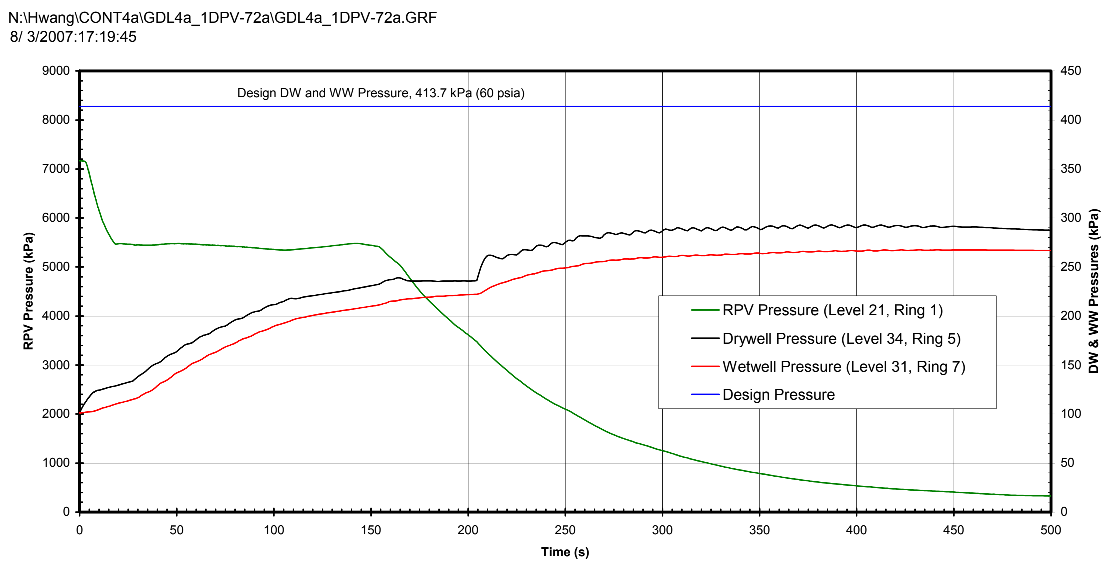


Figure 6.2-11a2. GDCS Line Break (Nominal Case) – Containment Pressures (500 s)

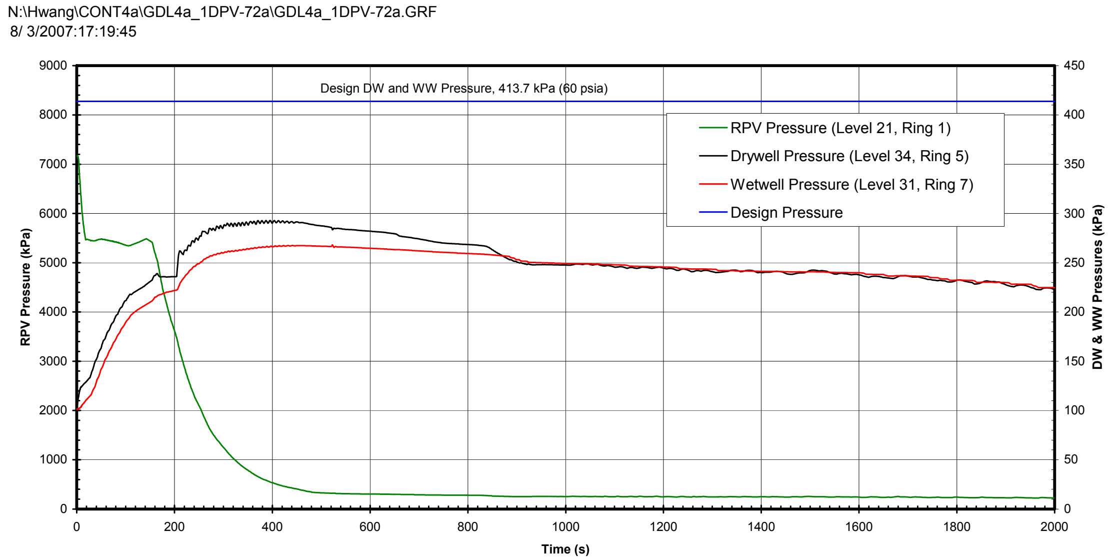


Figure 6.2-11a3. GDCS Line Break (Nominal Case) – Containment Pressures (2000 s)

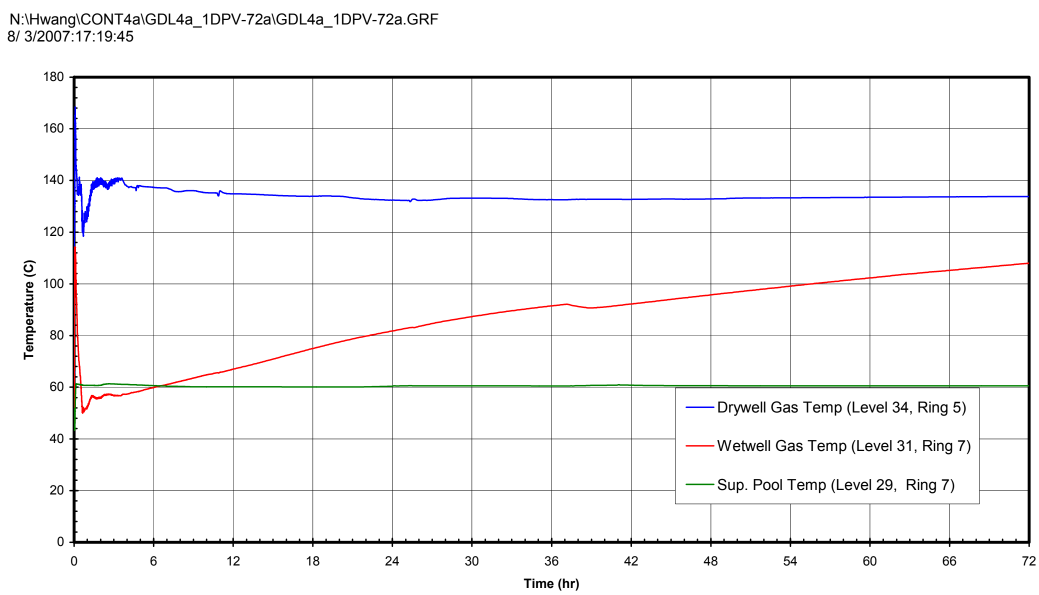


Figure 6.2-11b1. GDCS Line Break (Nominal Case) – Containment Temperatures (72 hrs)

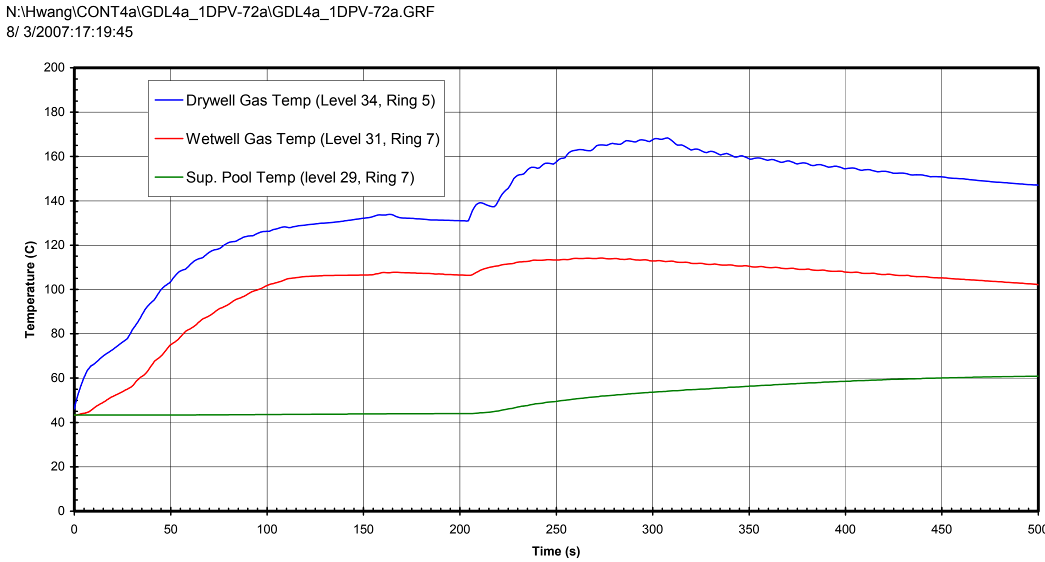


Figure 6.2-11b2. GDCS Line Break (Nominal Case) – Containment Temperatures (500 s)

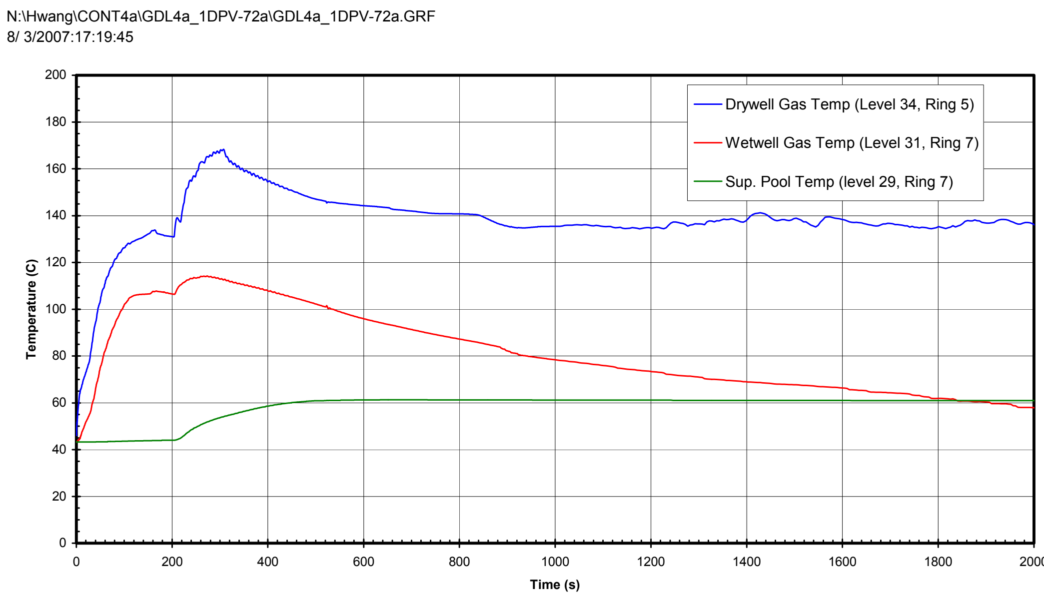


Figure 6.2-11b3. GDCS Line Break (Nominal Case) – Containment Temperatures (2000 s)

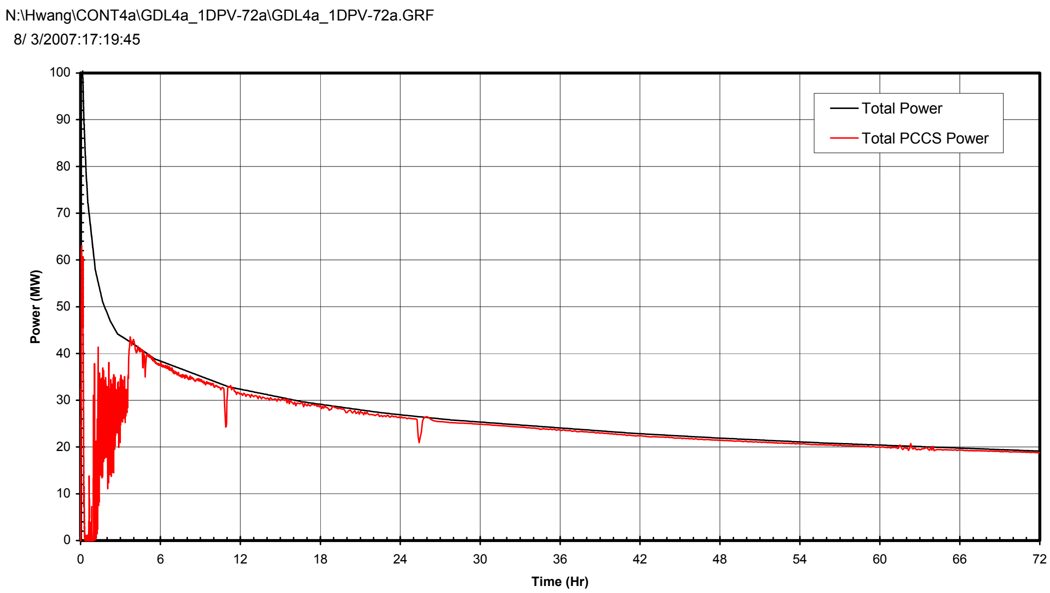


Figure 6.2-11c1. GDCS Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

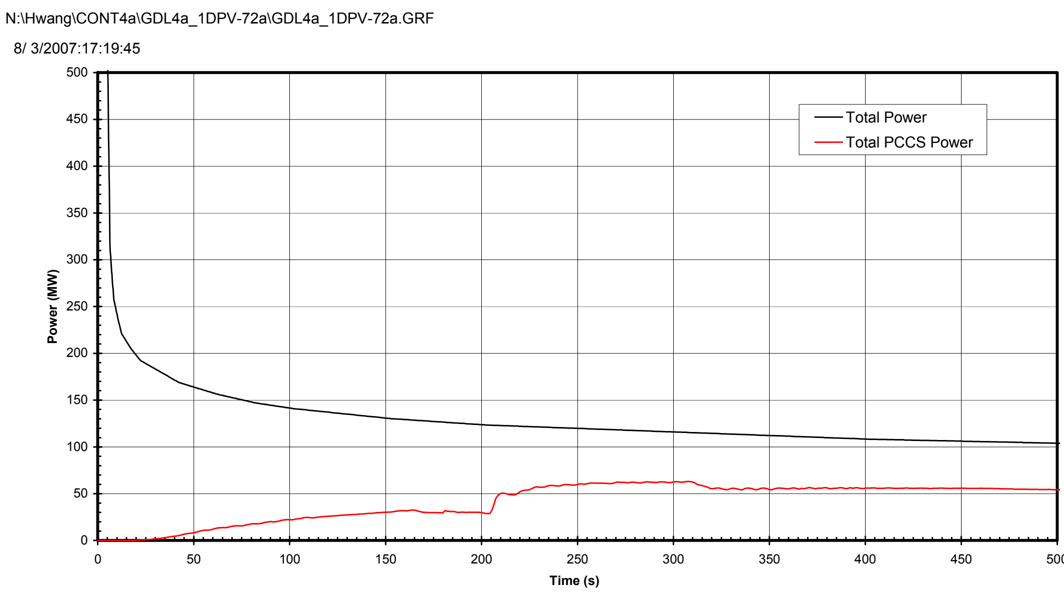


Figure 6.2-11c2. GDCS Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (500 s)

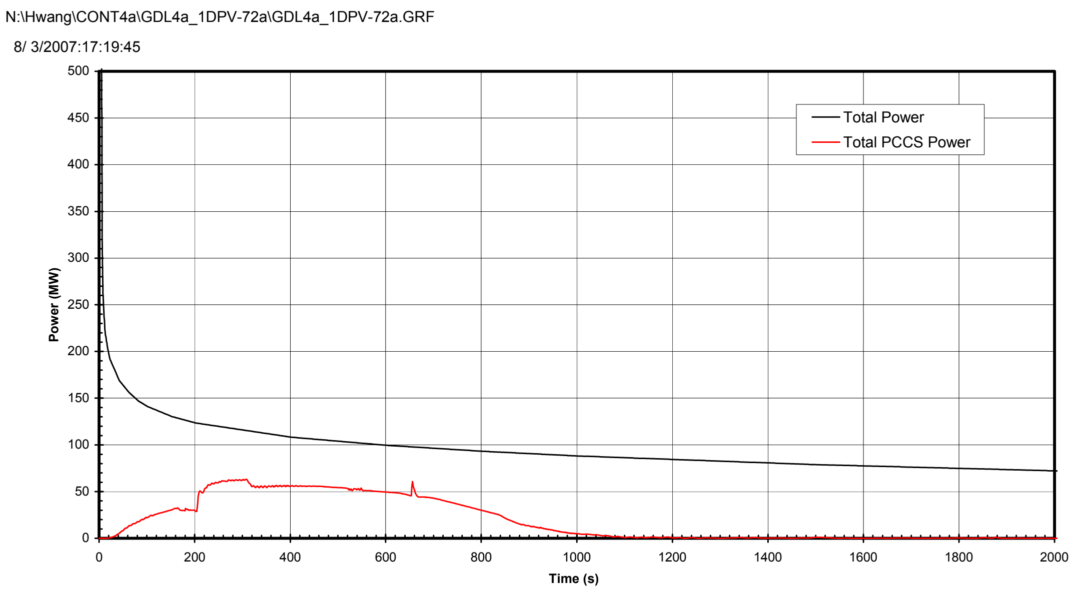


Figure 6.2-11c3. GDCS Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (2000 s)

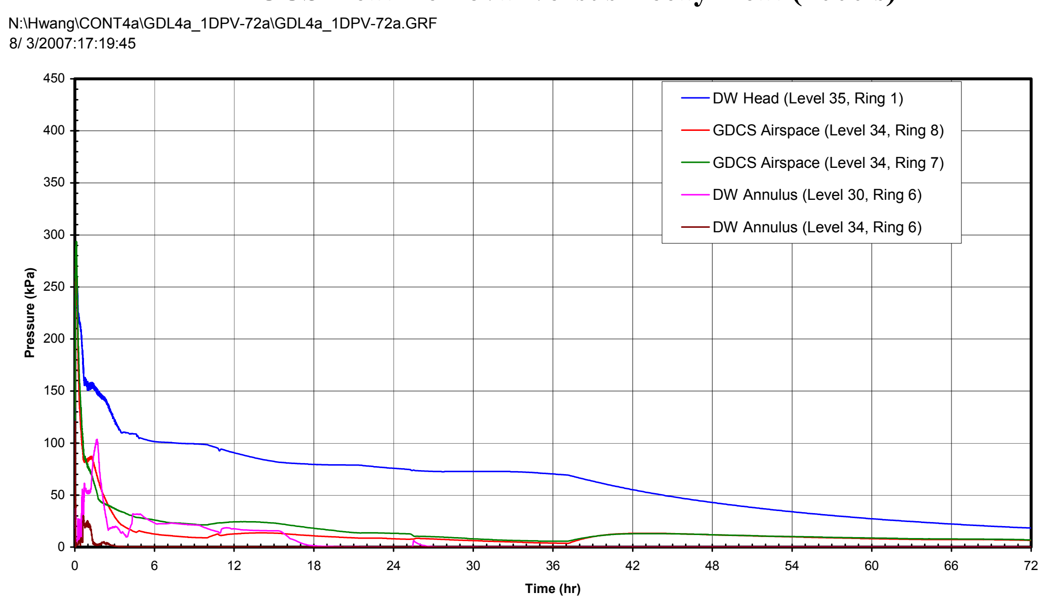


Figure 6.2-11d1. GDCS Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (72 hrs)

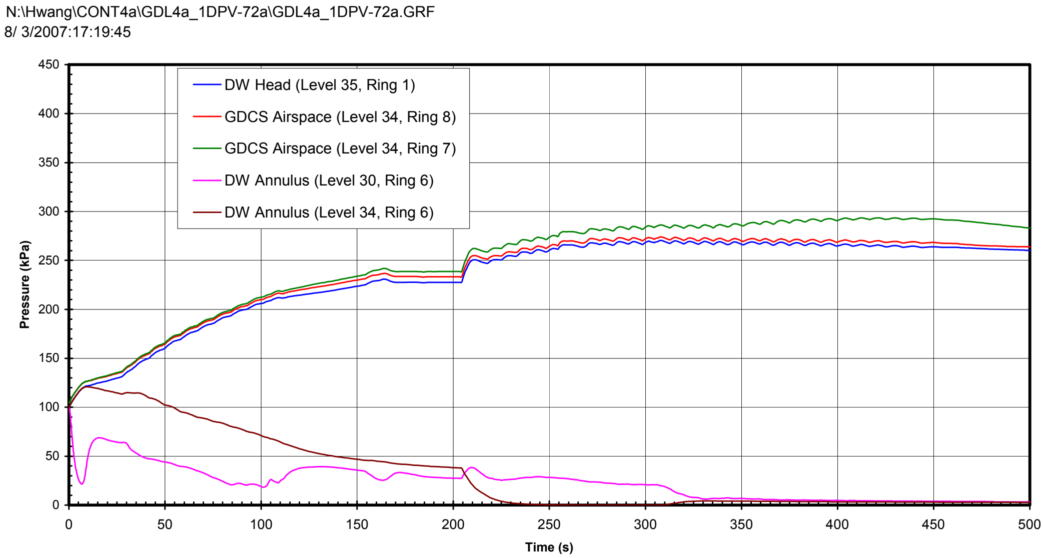


Figure 6.2-11d2. GDCS Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (500 s)

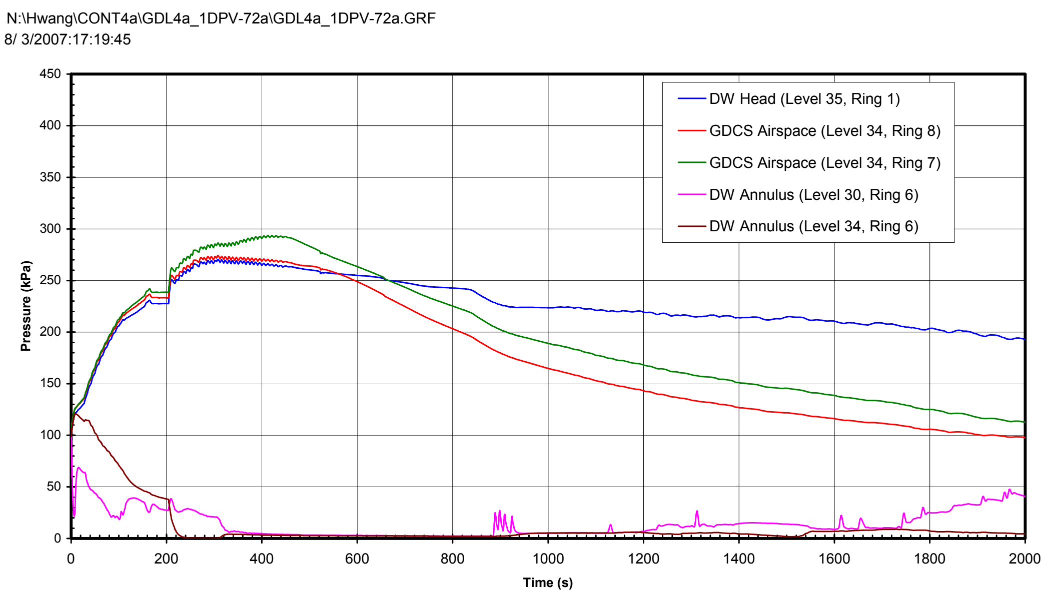


Figure 6.2-11d3. GDCS Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (2000 s)

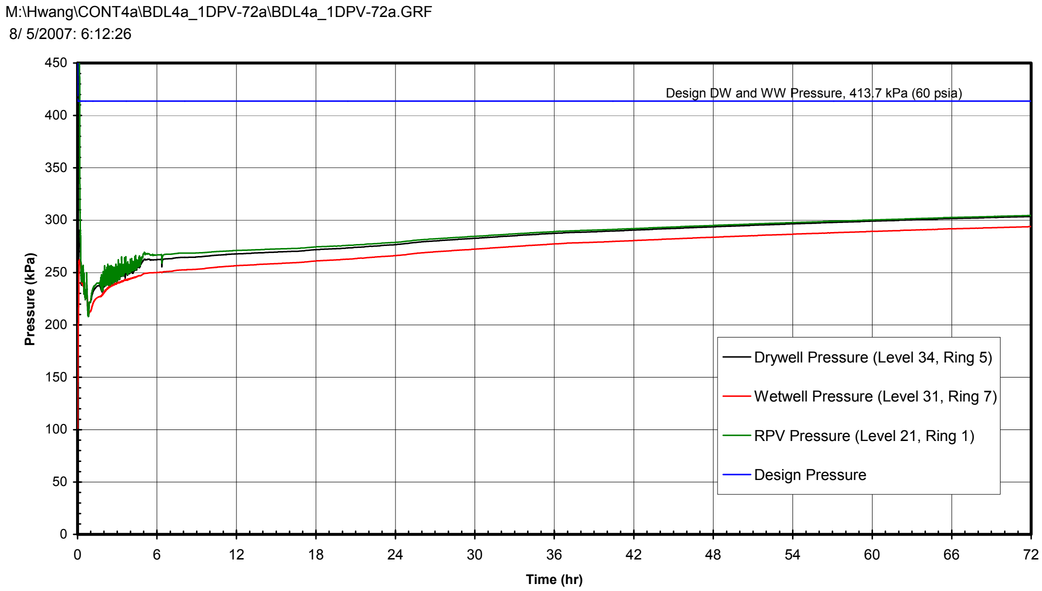


Figure 6.2-12a1. Bottom Drain Line Break (Nominal Case) – Containment Pressures (72 hrs)

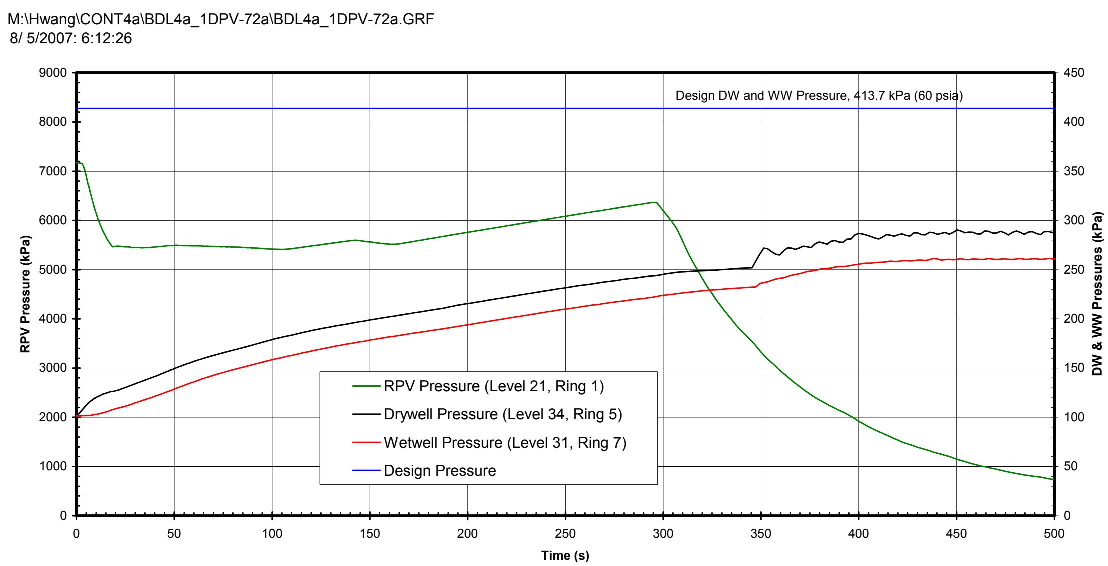


Figure 6.2-12a2. Bottom Drain Line Break (Nominal Case) – Containment Pressures (500 s)

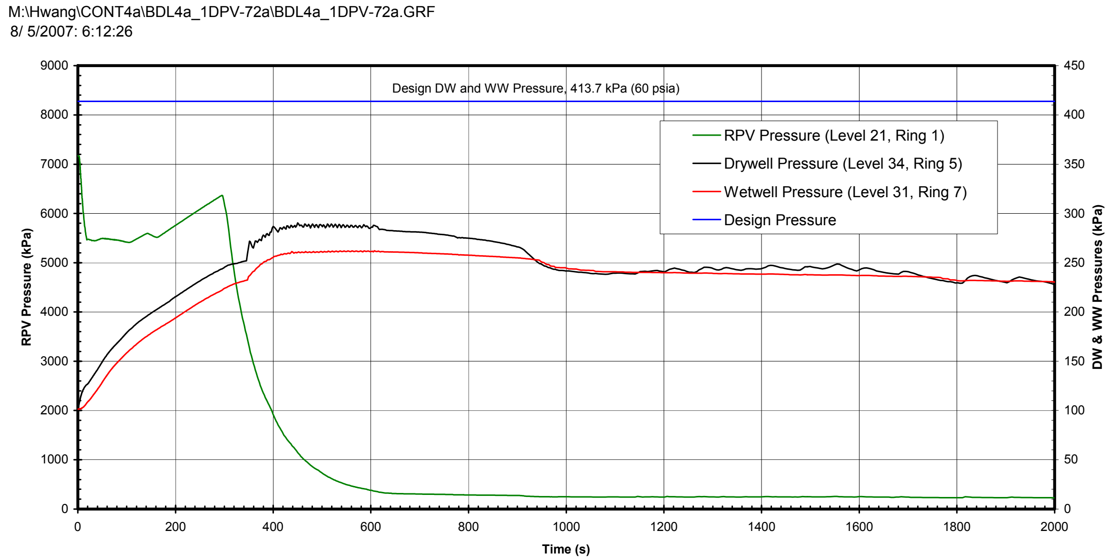


Figure 6.2-12a3. Bottom Drain Line Break (Nominal Case) – Containment Pressures (2000 s)

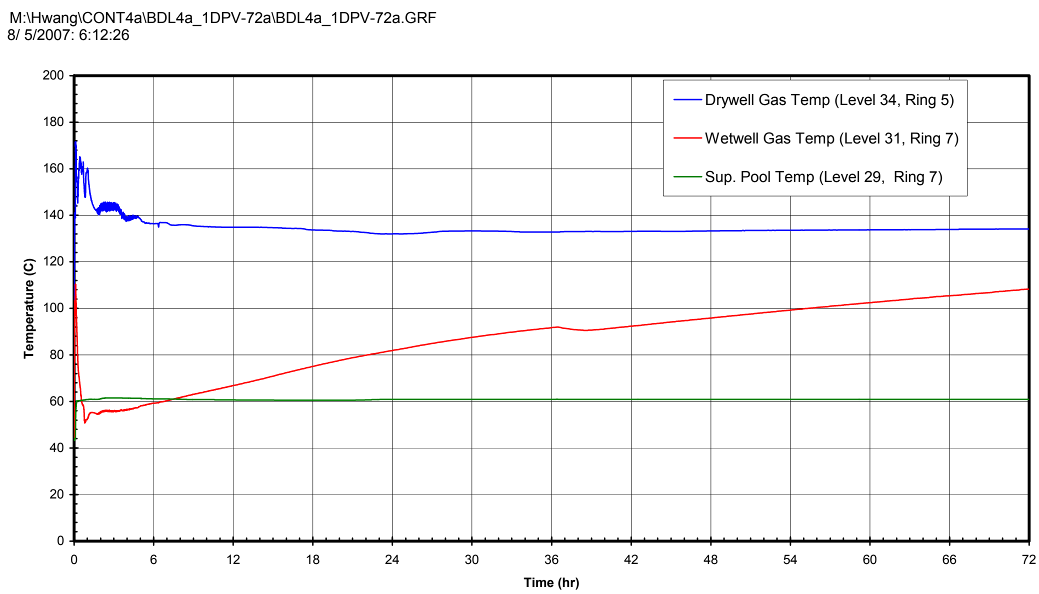


Figure 6.2-12b1. Bottom Drain Line Break (Nominal Case) – Containment Temperatures (72 hrs)

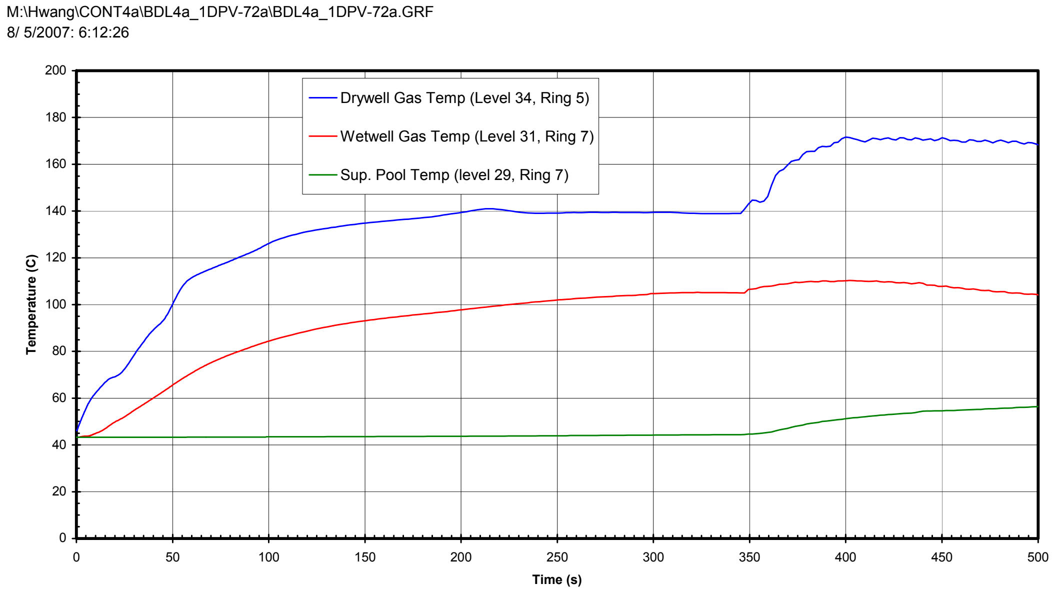


Figure 6.2-12b2. Bottom Drain Line Break (Nominal Case) – Containment Temperatures (500 s)

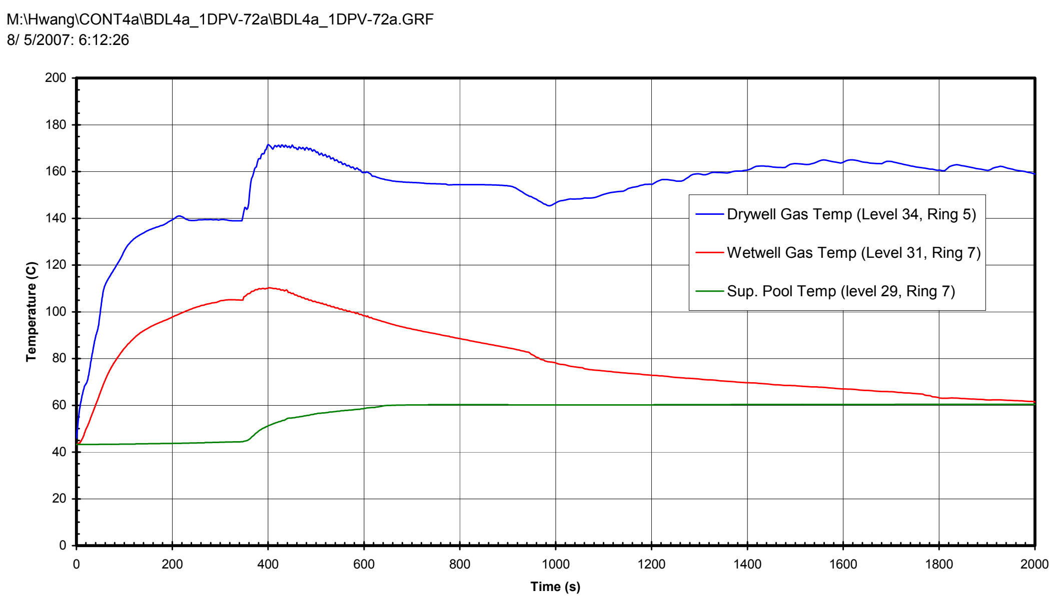


Figure 6.2-12b3. Bottom Drain Line Break (Nominal Case) – Containment Temperatures (2000 s)

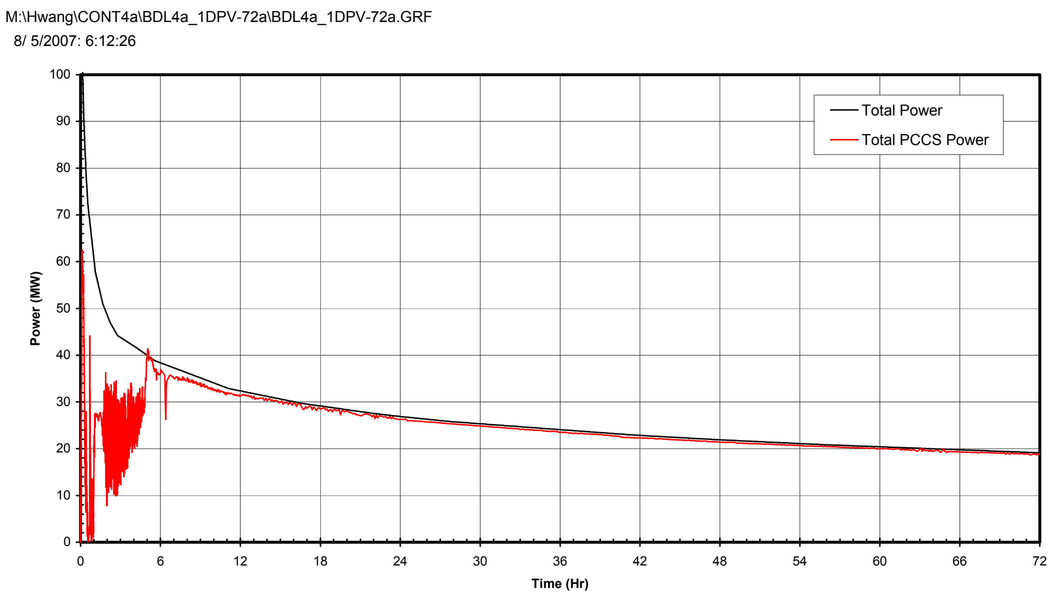


Figure 6.2-12c1. Bottom Drain Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

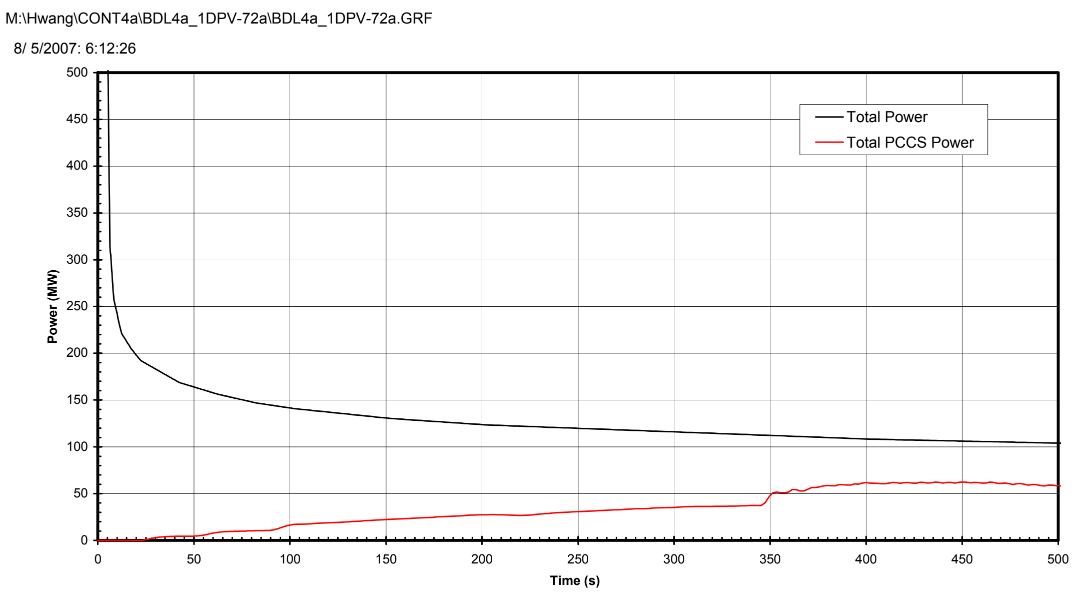


Figure 6.2-12c2. Bottom Drain Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (500 s)

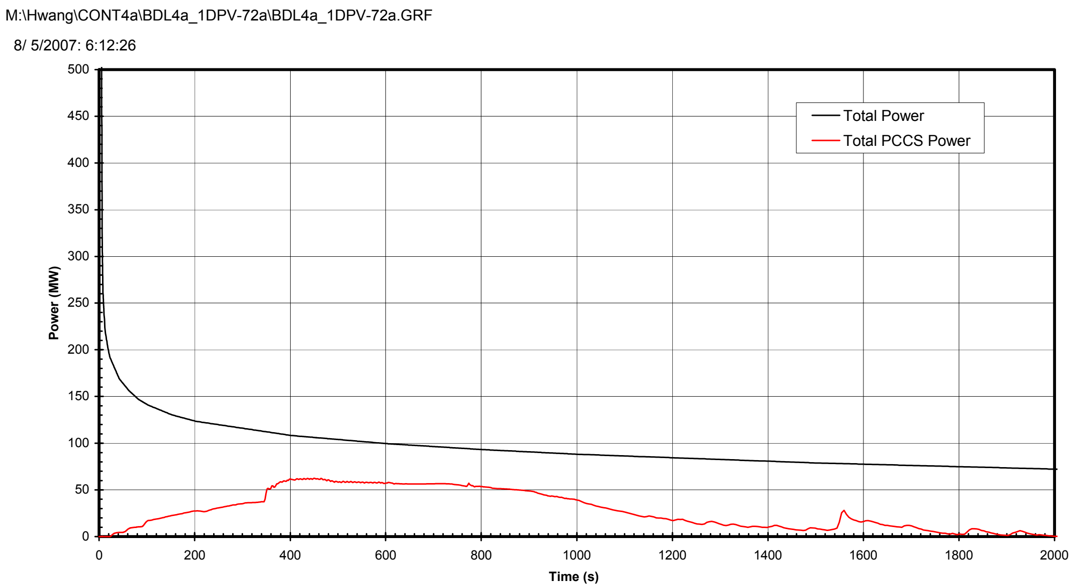


Figure 6.2-12c3. Bottom Drain Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (2000 s)

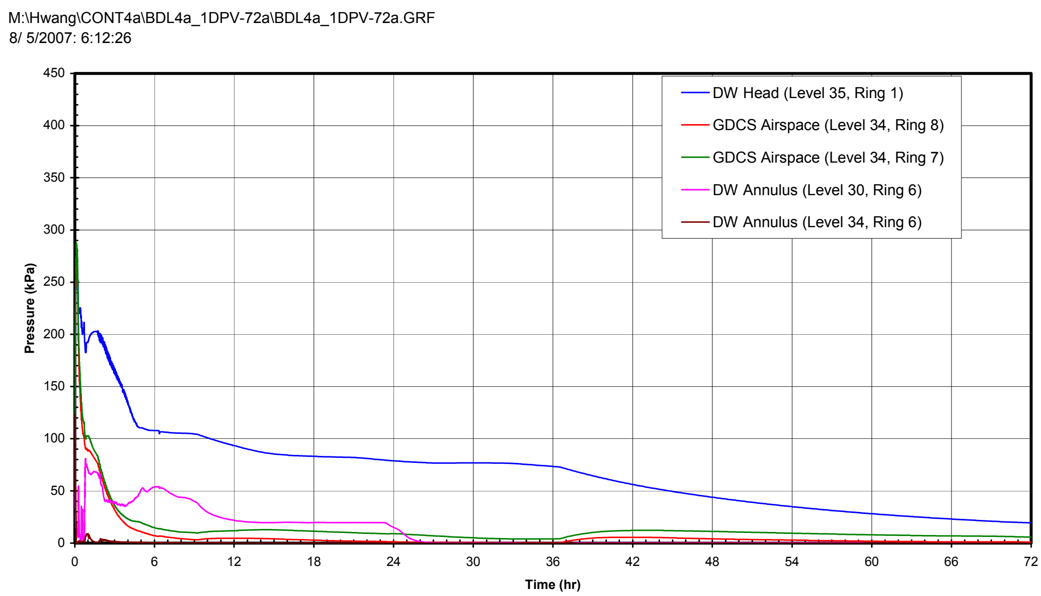


Figure 6.2-12d1. Bottom Drain Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (72 hrs)

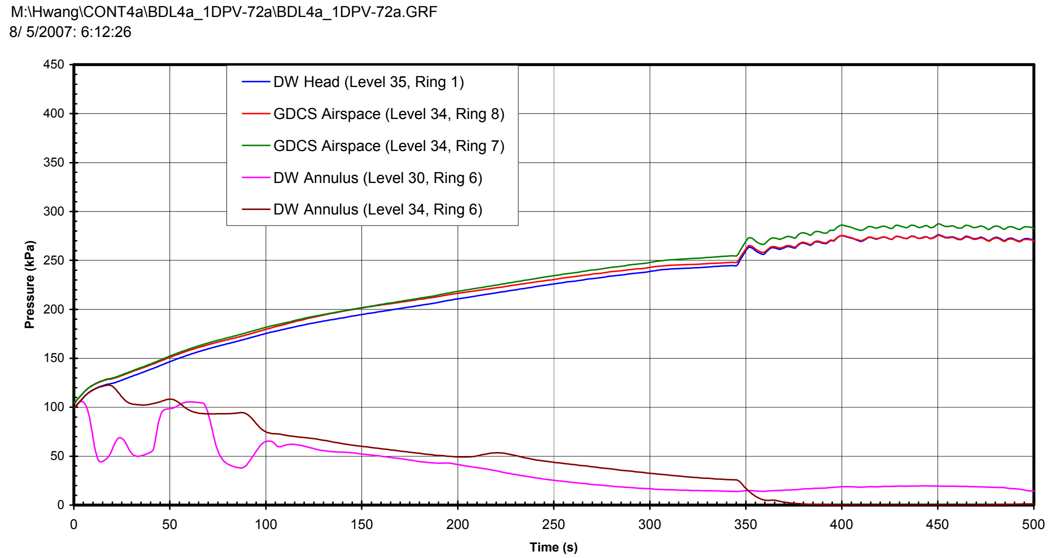


Figure 6.2-12d2. Bottom Drain Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (500 s)

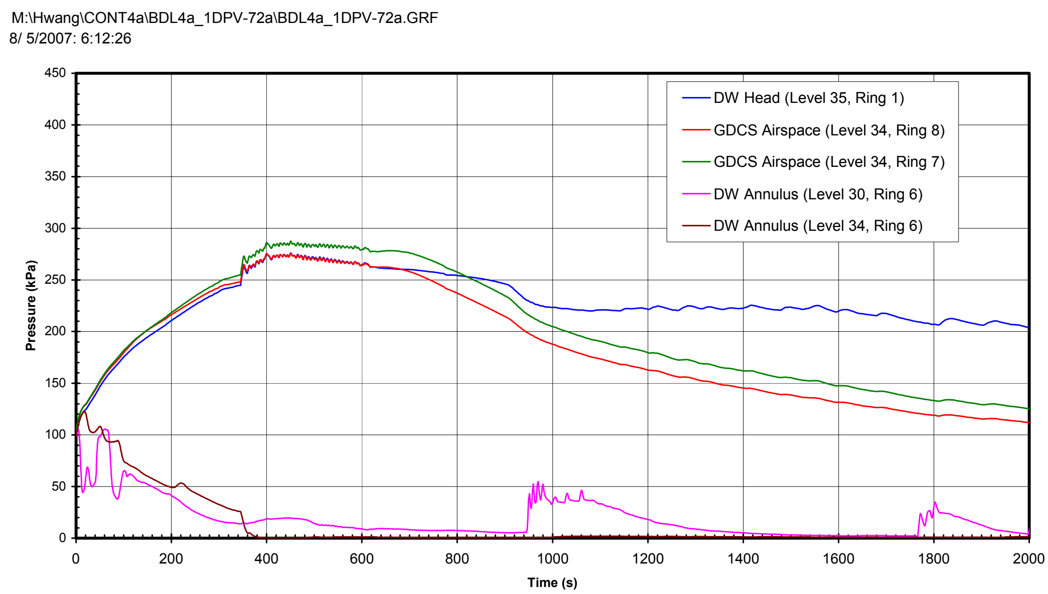


Figure 6.2-12d3. Bottom Drain Line Break (Nominal Case) - Drywell and GDCS NC Gas Pressures (2000 s)

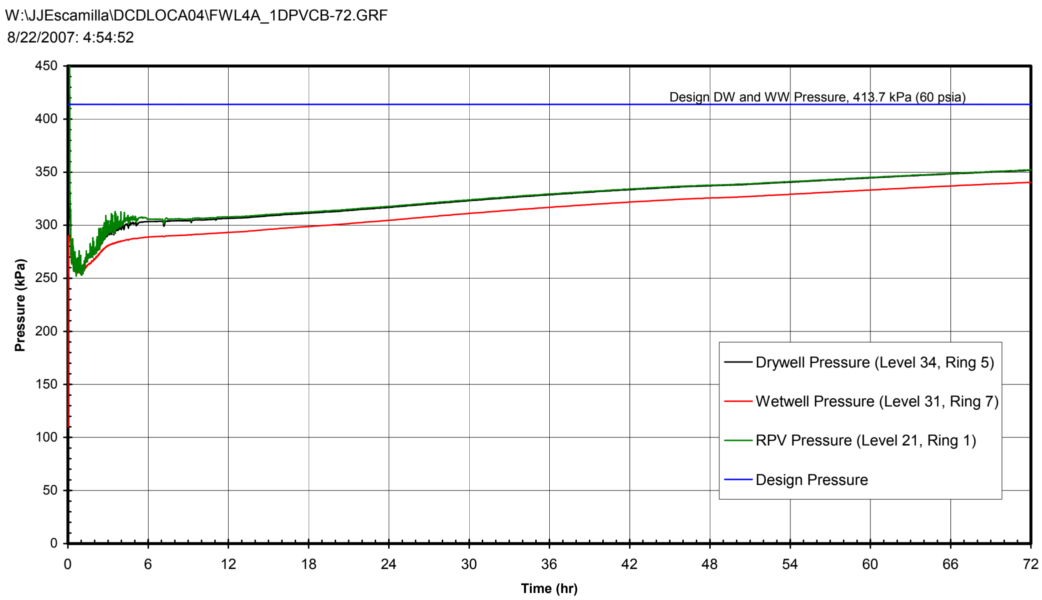


Figure 6.2-13a1. Feedwater Line Break (Bounding Case) – Containment Pressures (72 hrs)

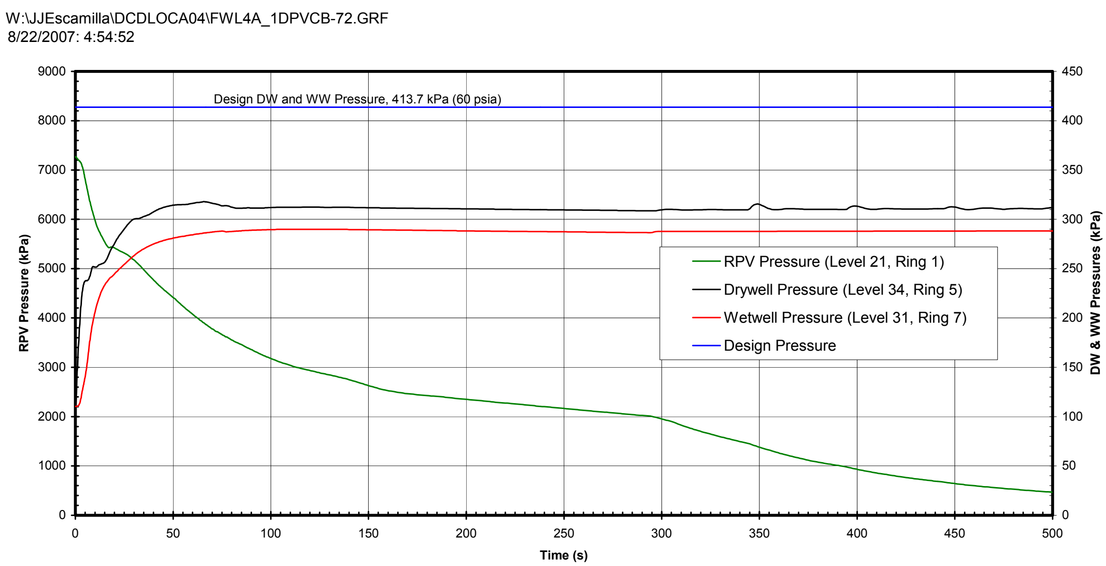


Figure 6.2-13a2. Feedwater Line Break (Bounding Case) – Containment Pressures (500 s)

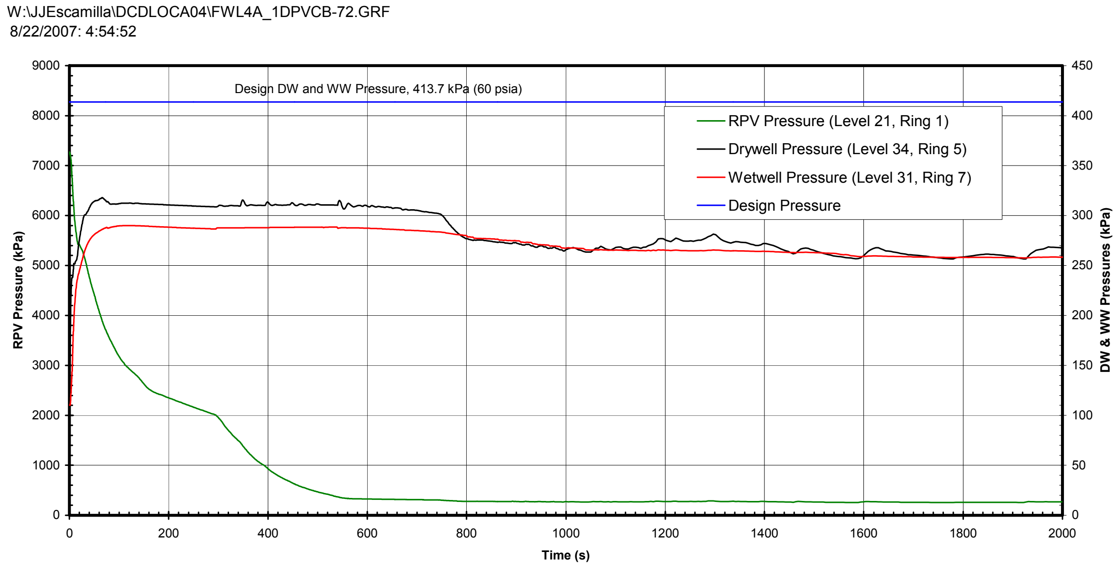


Figure 6.2-13a3. Feedwater Line Break (Bounding Case) – Containment Pressures (2000 s)

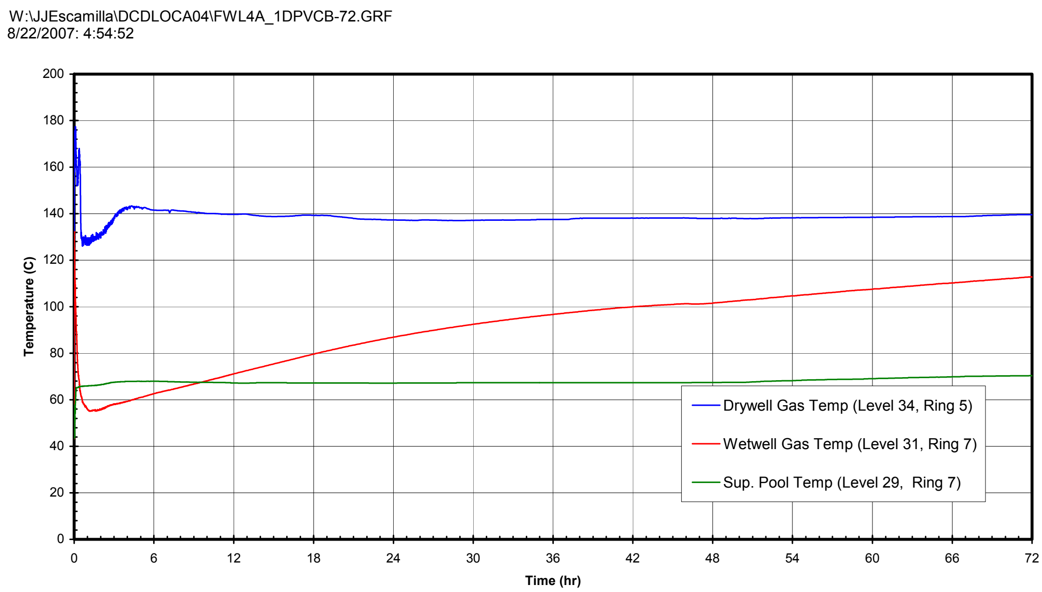


Figure 6.2-13b1. Feedwater Line Break (Bounding Case) – Containment Temperatures (72 hrs)

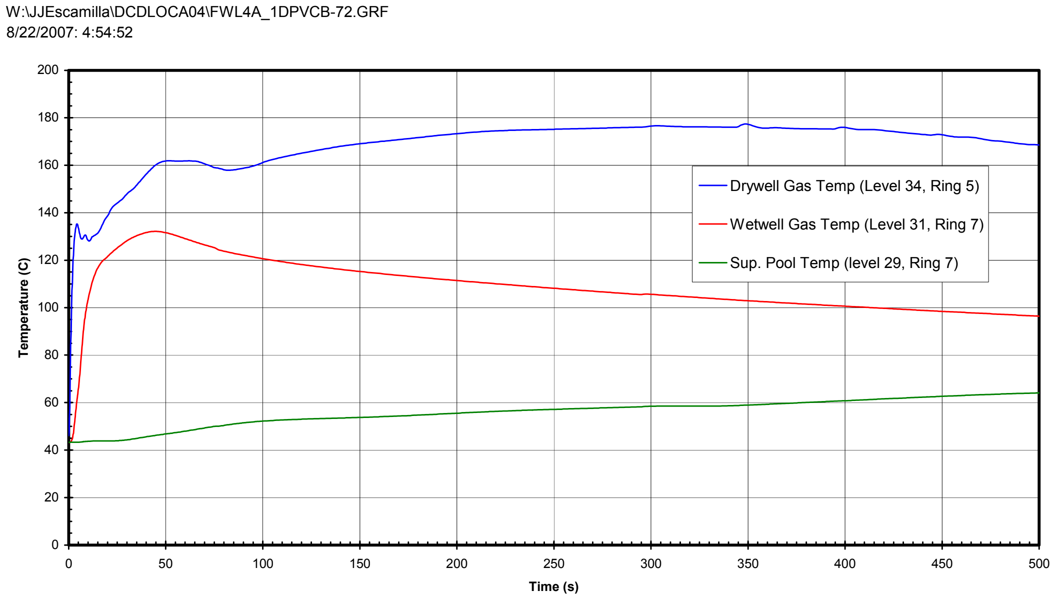


Figure 6.2-13b2. Feedwater Line Break (Bounding Case) – Containment Temperatures (500 s)

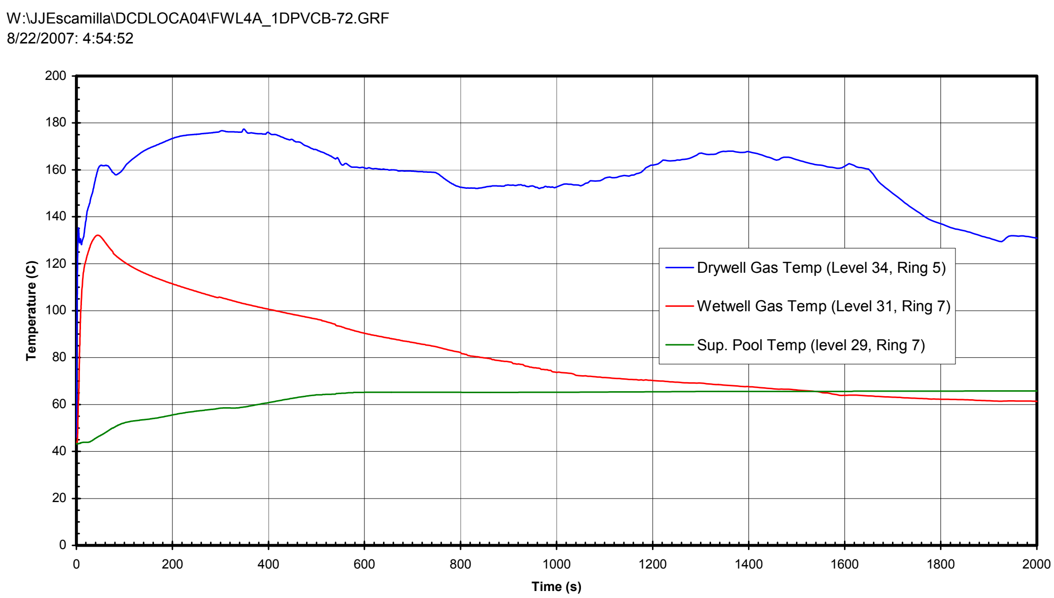


Figure 6.2-13b3. Feedwater Line Break (Bounding Case) – Containment Temperatures (2000 s)

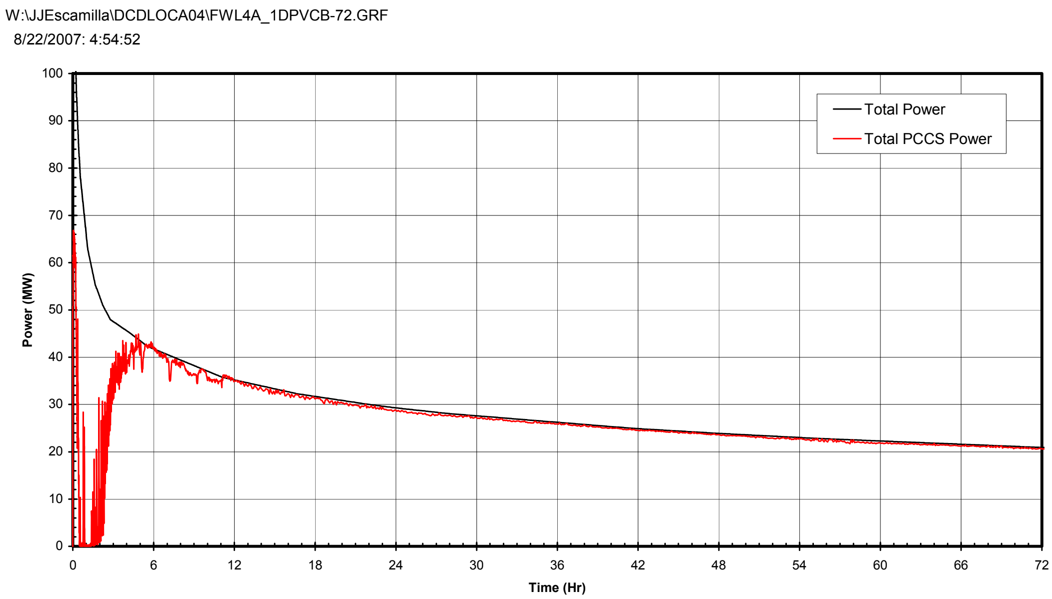


Figure 6.2-13c1. Feedwater Line Break (Bounding Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

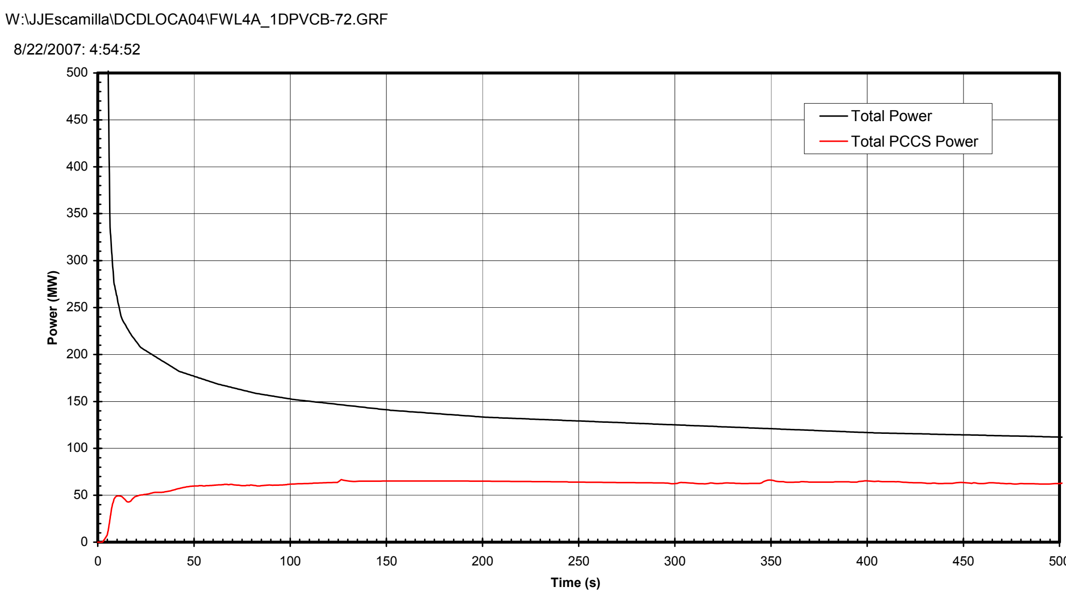


Figure 6.2-13c2. Feedwater Line Break (Bounding Case) – PCCS Heat Removal versus Decay Heat (500 s)

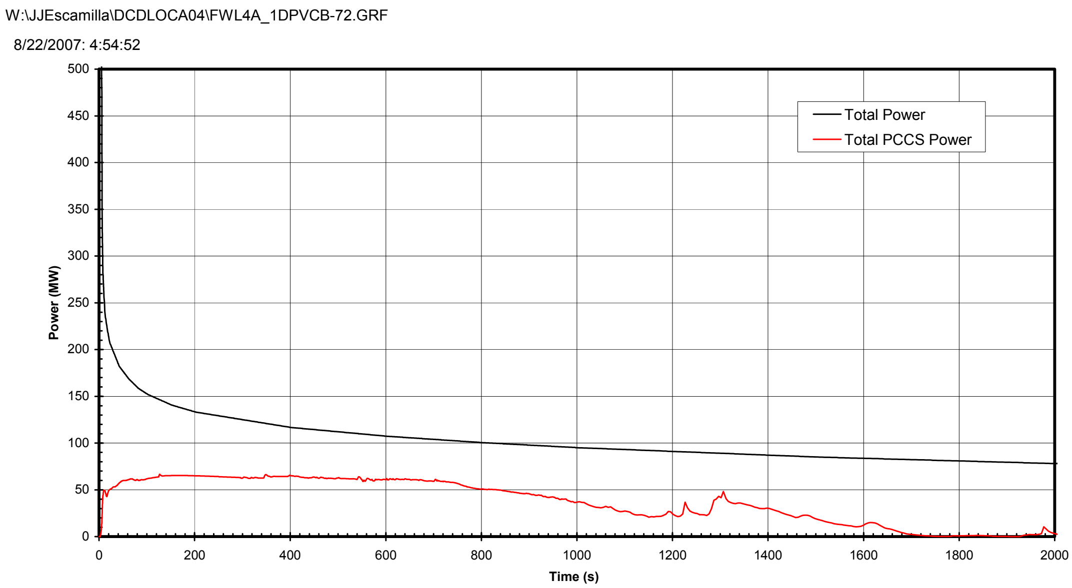


Figure 6.2-13c3. Feedwater Line Break (Bounding Case) – PCCS Heat Removal versus Decay Heat (2000 s)

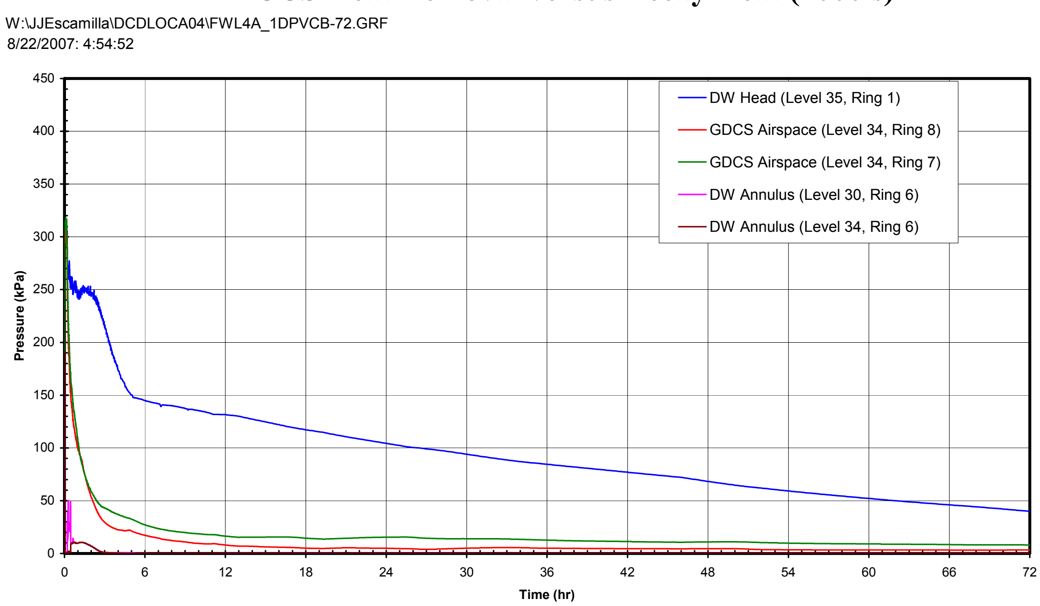


Figure 6.2-13d1. Feedwater Line Break (Bounding Case) - Drywell and GDCS NC Gas Pressures (72 hrs)

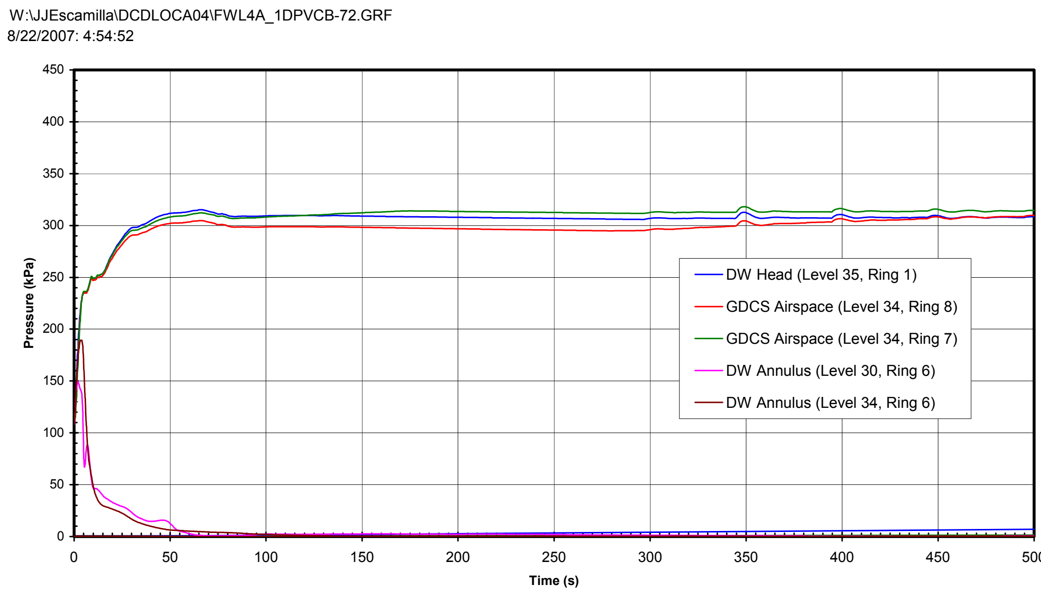


Figure 6.2-13d2. Feedwater Line Break (Bounding Case) - Drywell and GDCS NC Gas Pressures (500 s)

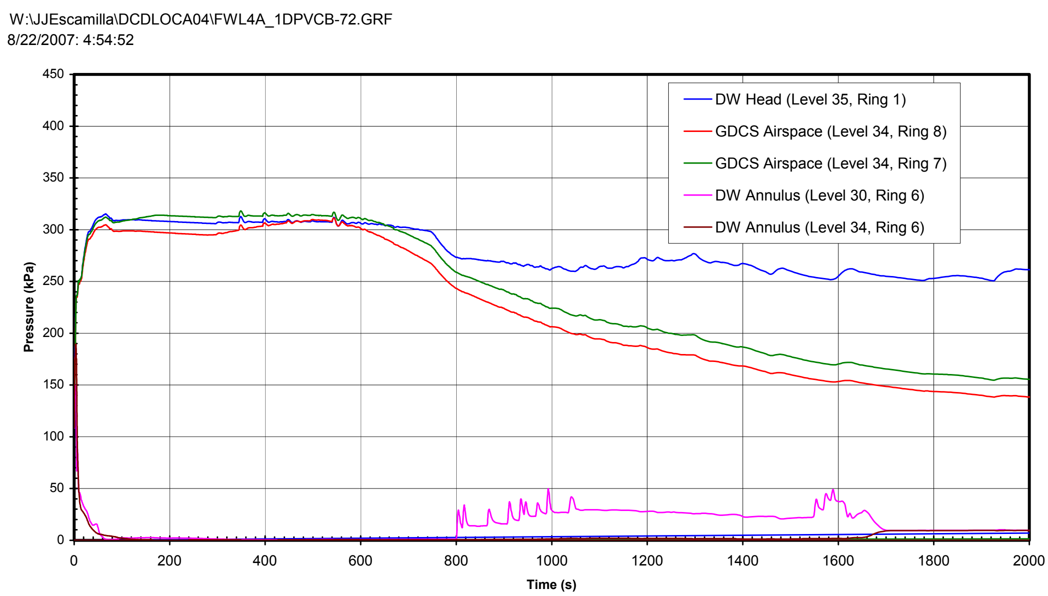


Figure 6.2-13d3. Feedwater Line Break (Bounding Case) - Drywell and GDCS NC Gas Pressures (2000 s)

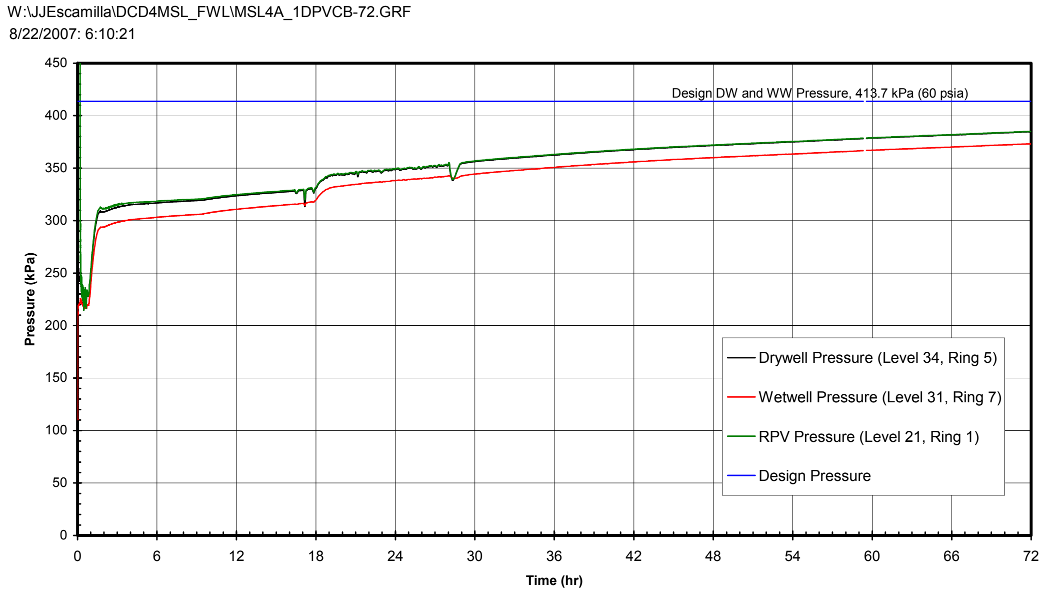


Figure 6.2-14a1. Main Steam Line Break (Bounding Case) Containment Pressures (72 hrs)

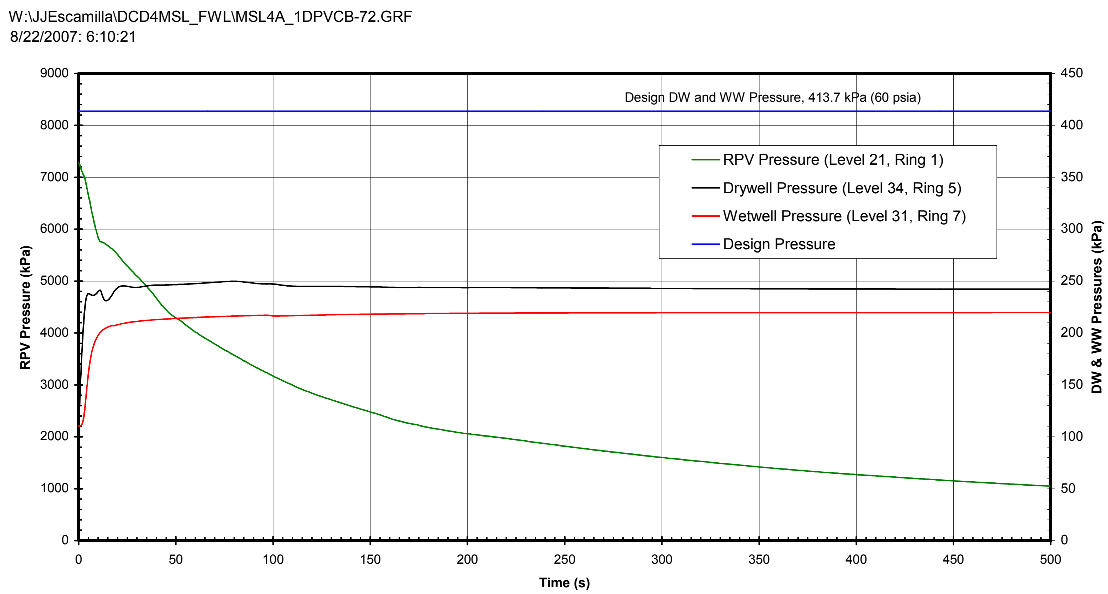


Figure 6.2-14a2. Main Steam Line Break (Bounding Case) – Containment Pressures (500 s)

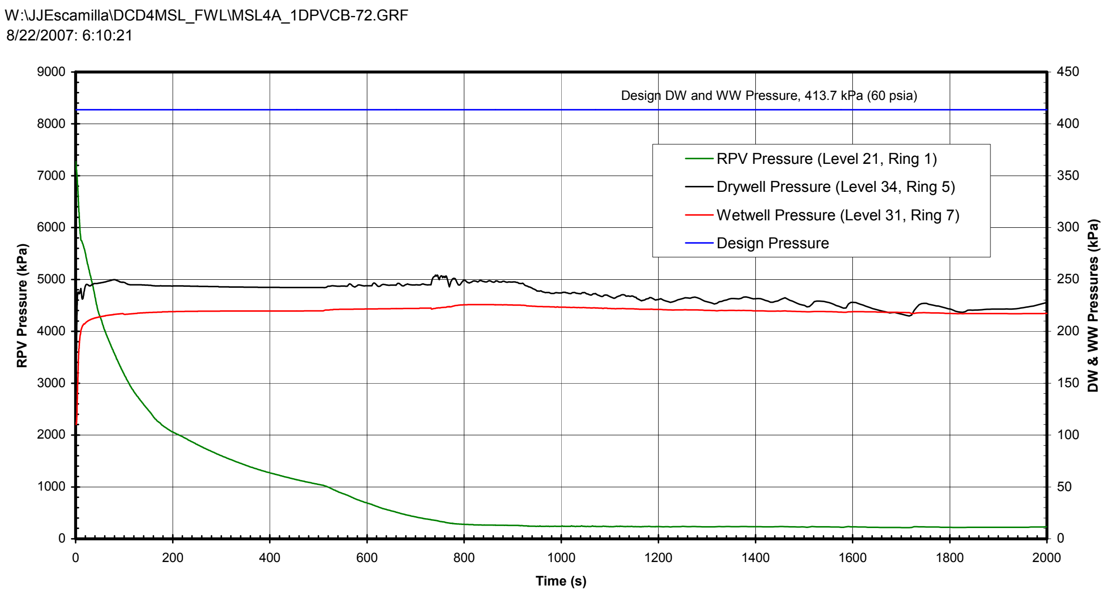


Figure 6.2-14a3. Main Steam Line Break (Bounding Case) – Containment Pressures (2000 s)

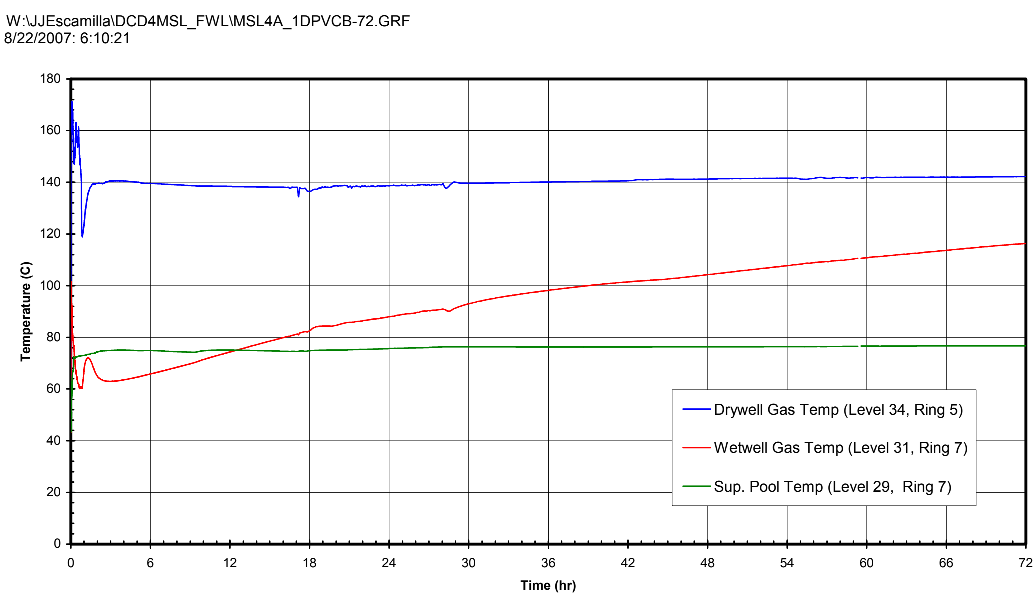


Figure 6.2-14b1. Main Steam Line Break (Bounding Case) – Containment Temperatures (72 hrs)

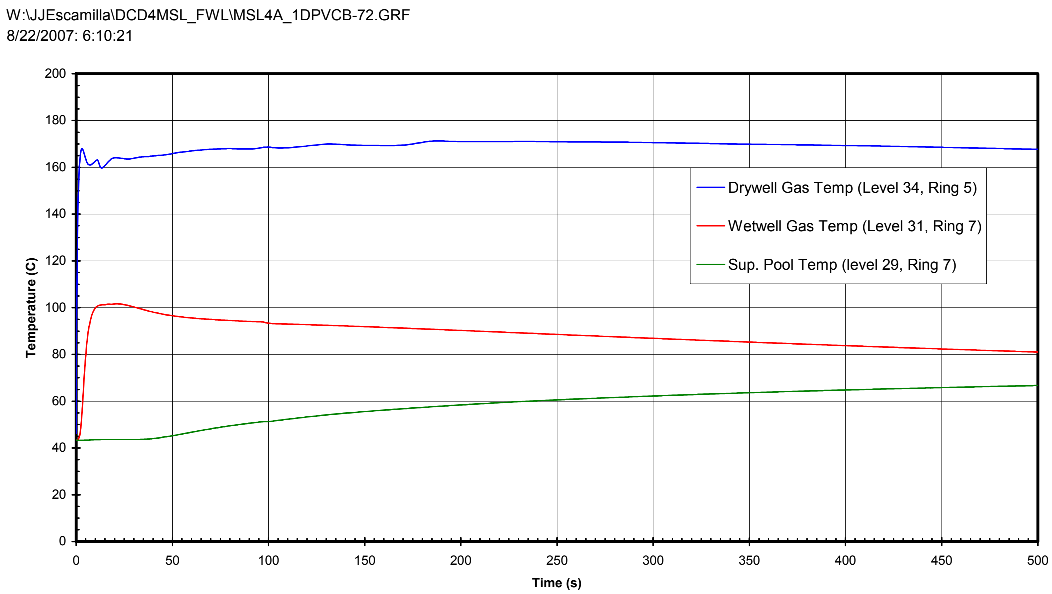


Figure 6.2-14b2. Main Steam Line Break (Bounding Case) – Containment Temperatures (500 s)

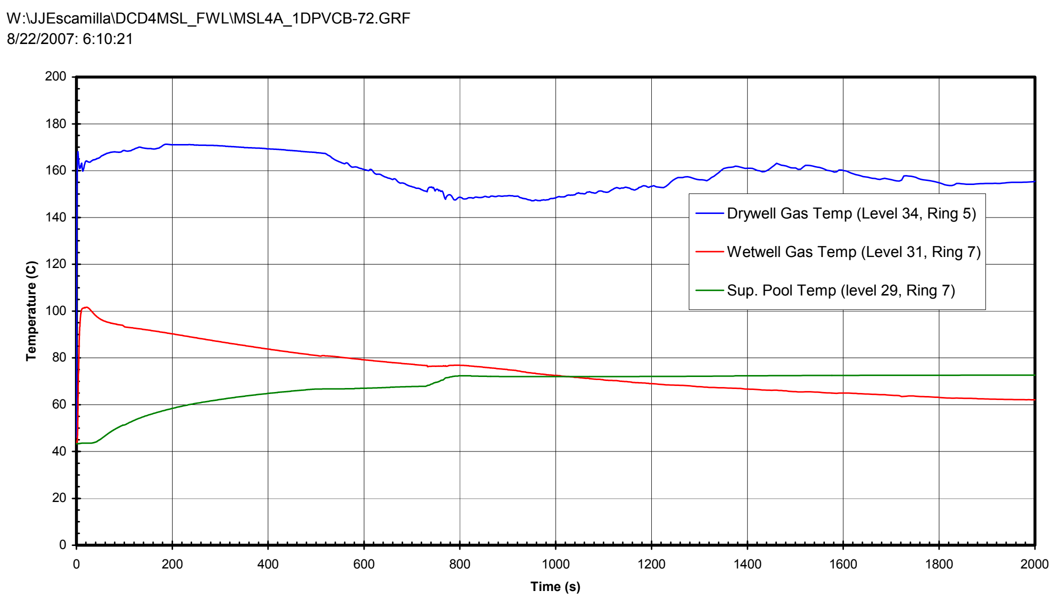


Figure 6.2-14b3. Main Steam Line Break (Bounding Case) – Containment Temperatures (2000 s)

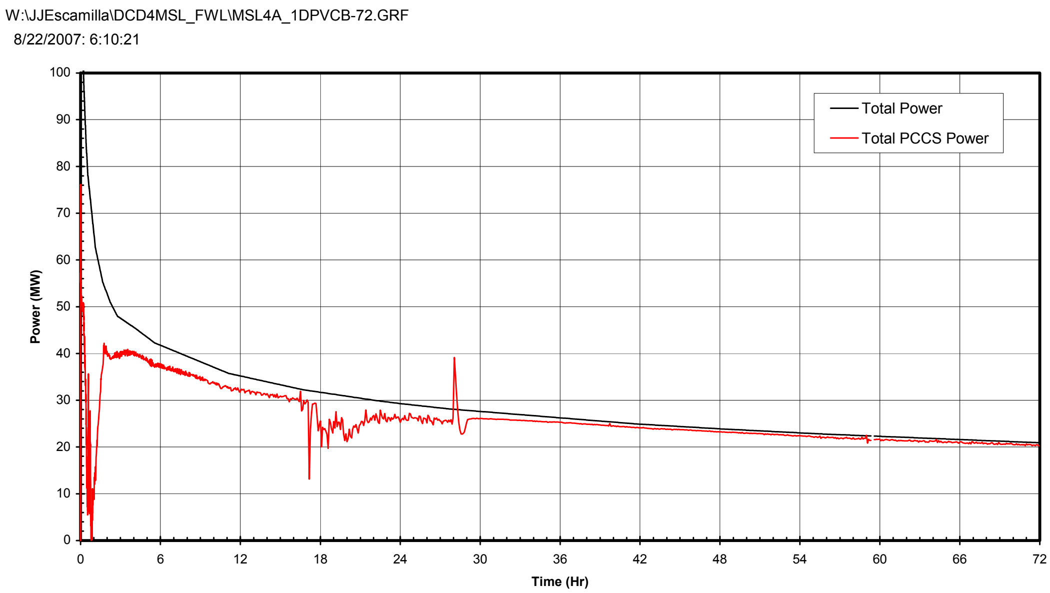


Figure 6.2-14c1. Main Steam Line Break (Bounding Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

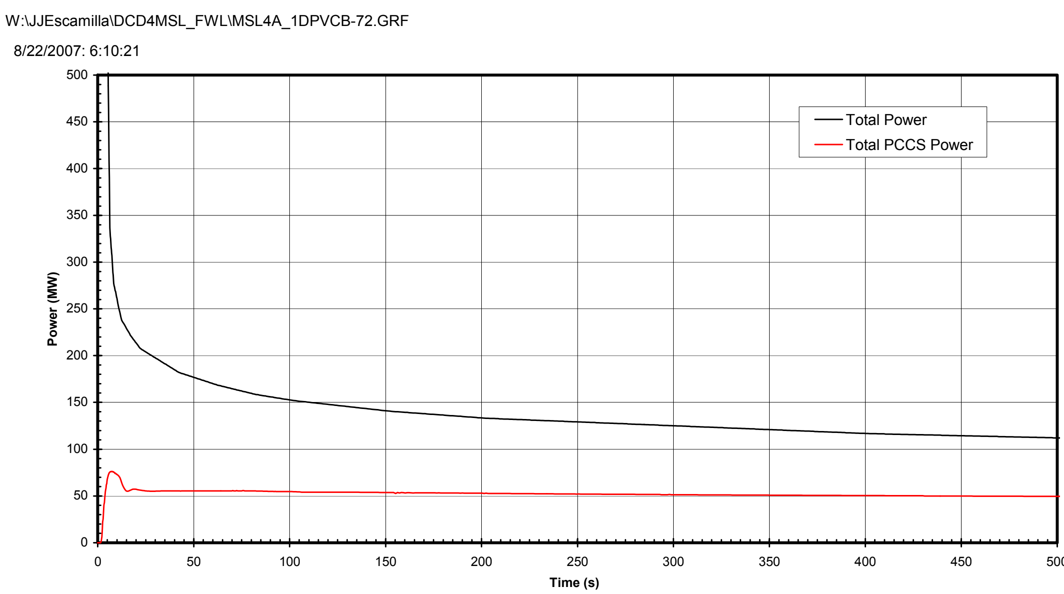


Figure 6.2-14c2. Main Steam Line Break (Bounding Case) – PCCS Heat Removal versus Decay Heat (500 s)

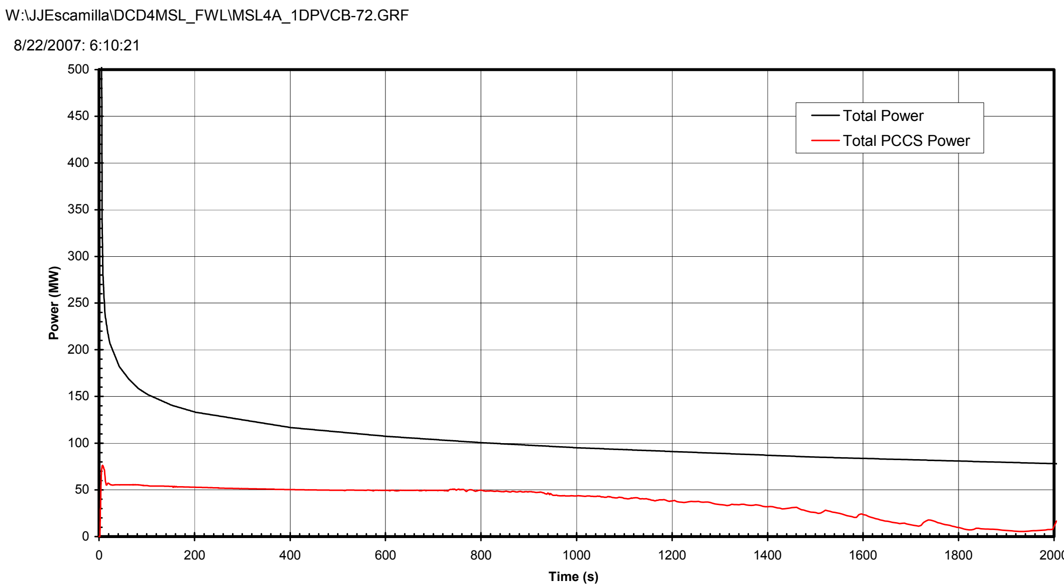


Figure 6.2-14c3. Main Steam Line Break (Bounding Case) – PCCS Heat Removal versus Decay Heat (2000 s)

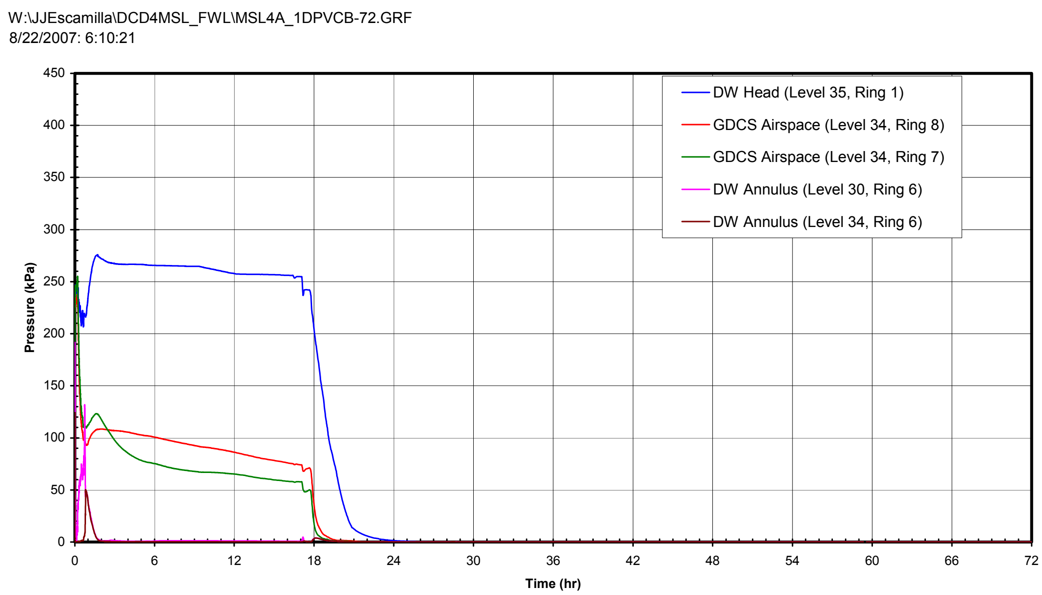


Figure 6.2-14d1. Main Steam Line Break (Bounding Case) - Drywell and GDCS NC Gas Pressures (72 hrs)

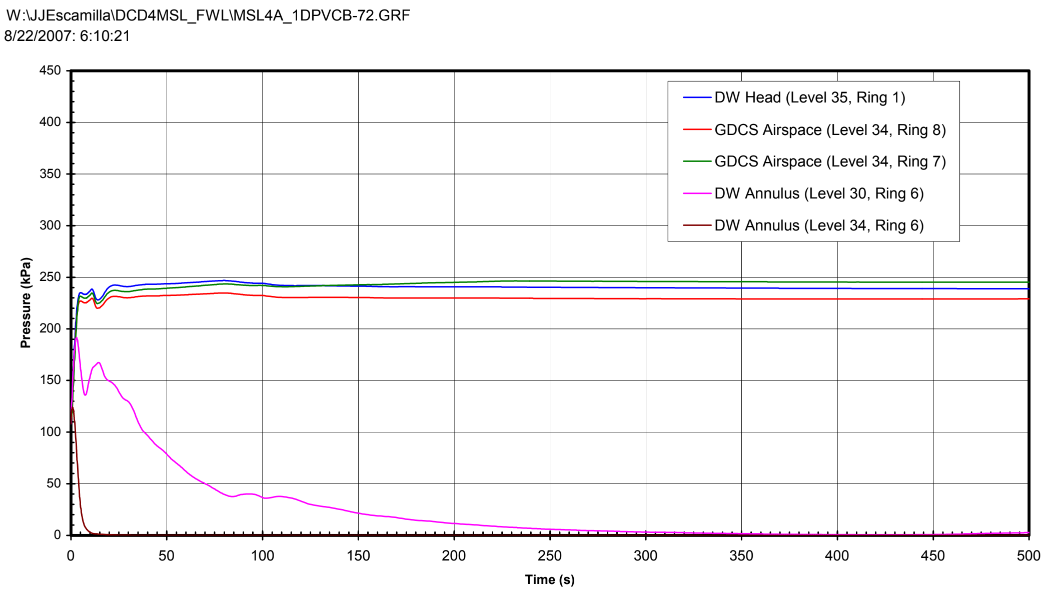


Figure 6.2-14d2. Main Steam Line Break (Bounding Case) - Drywell and GDCS NC Gas Pressures (500 s)

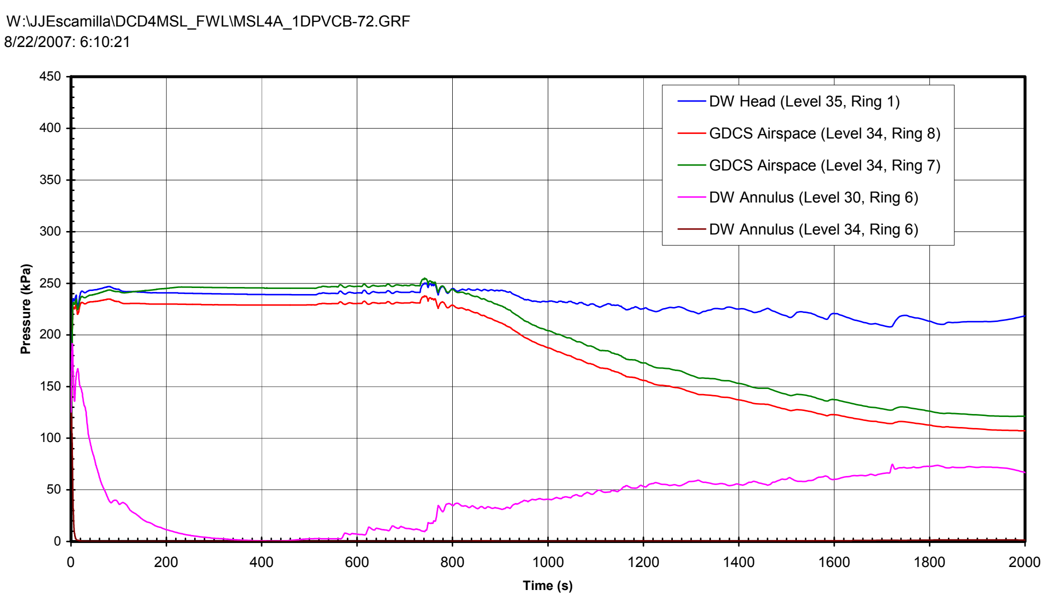


Figure 6.2-14d3. Main Steam Line Break (Bounding Case) - Drywell and GDCS NC Gas Pressures (2000 s)

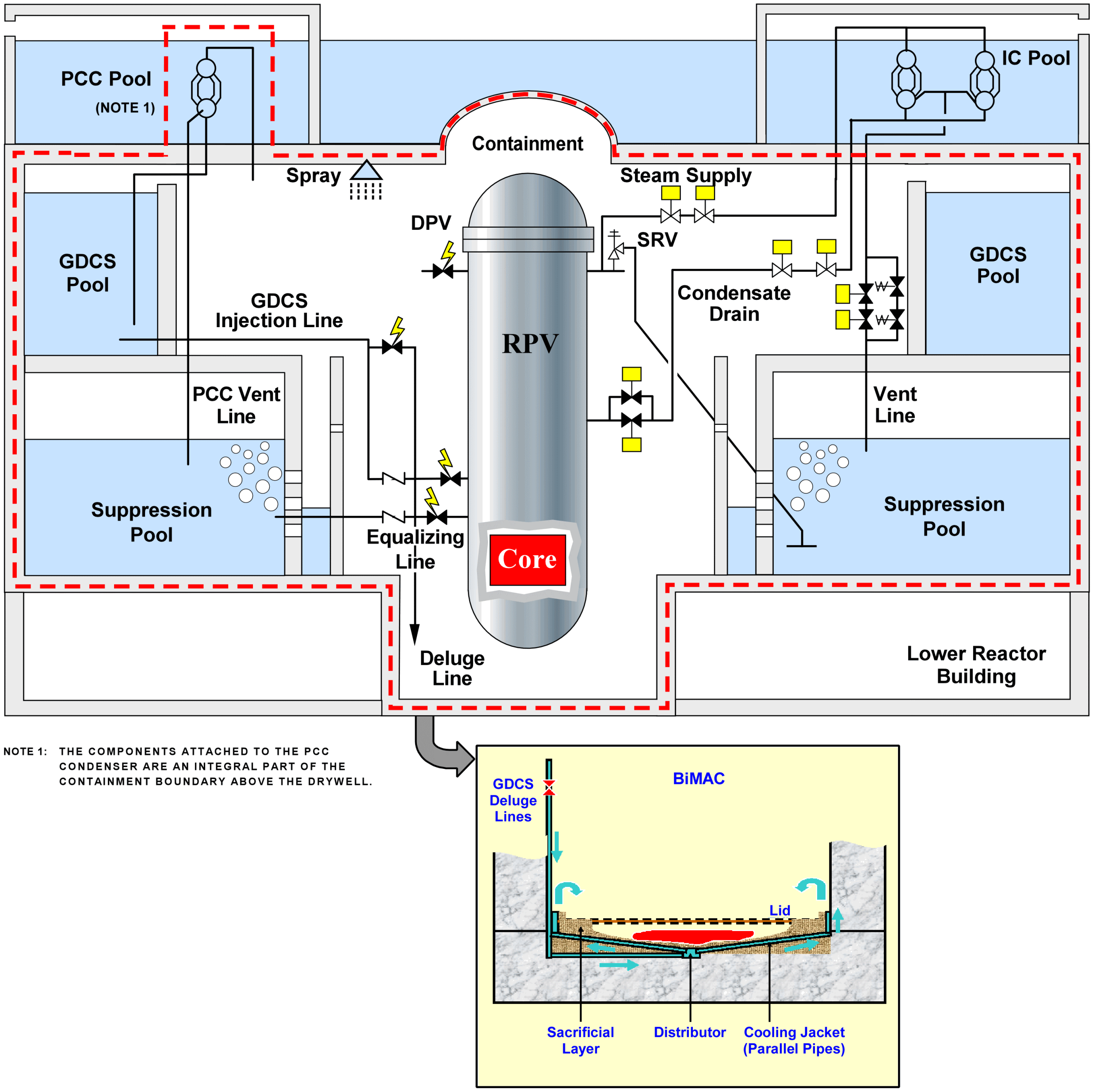


Figure 6.2-15. Summary of Severe Accident Design Features

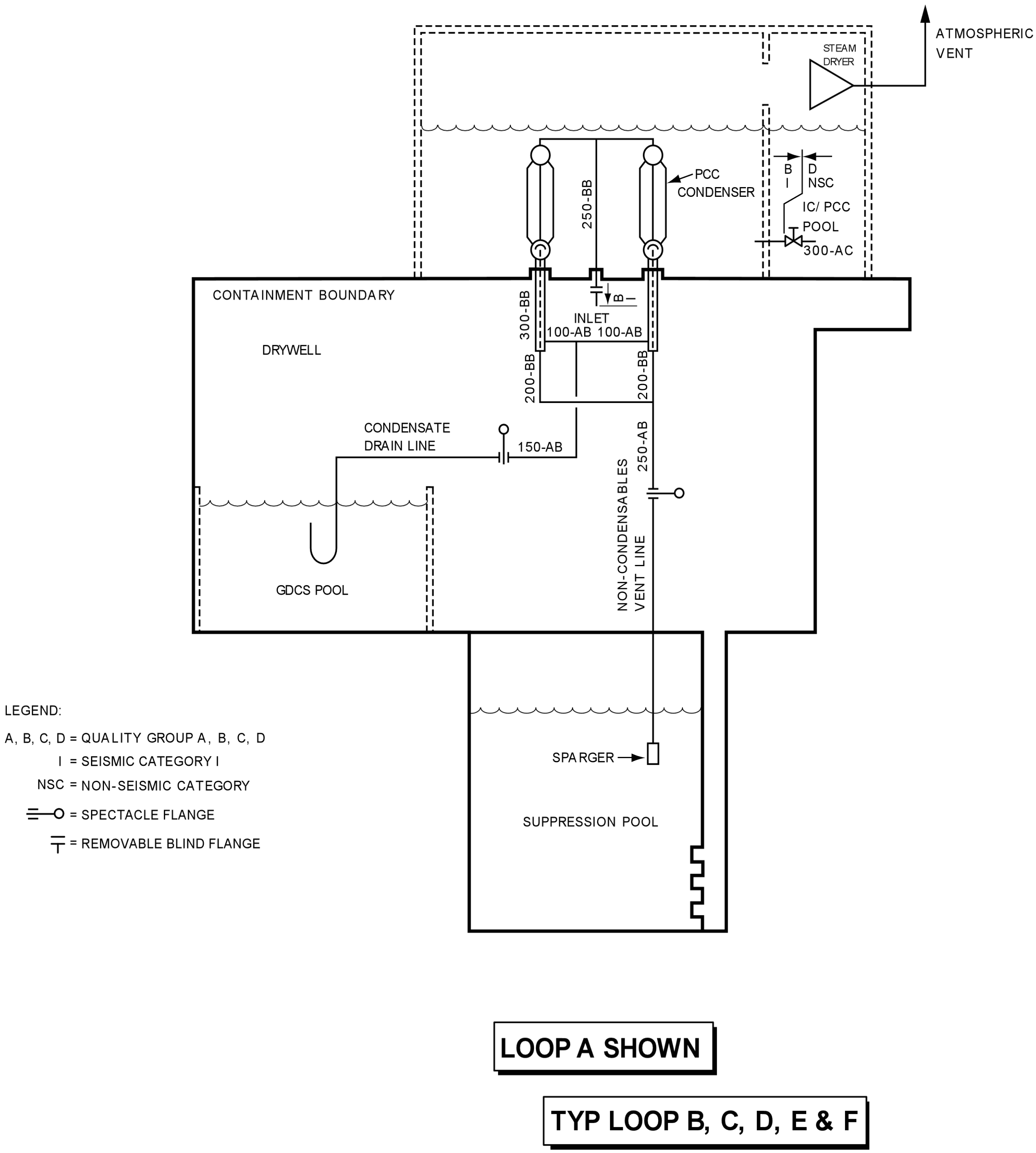


Figure 6.2-16. PCCS Schematic Diagram

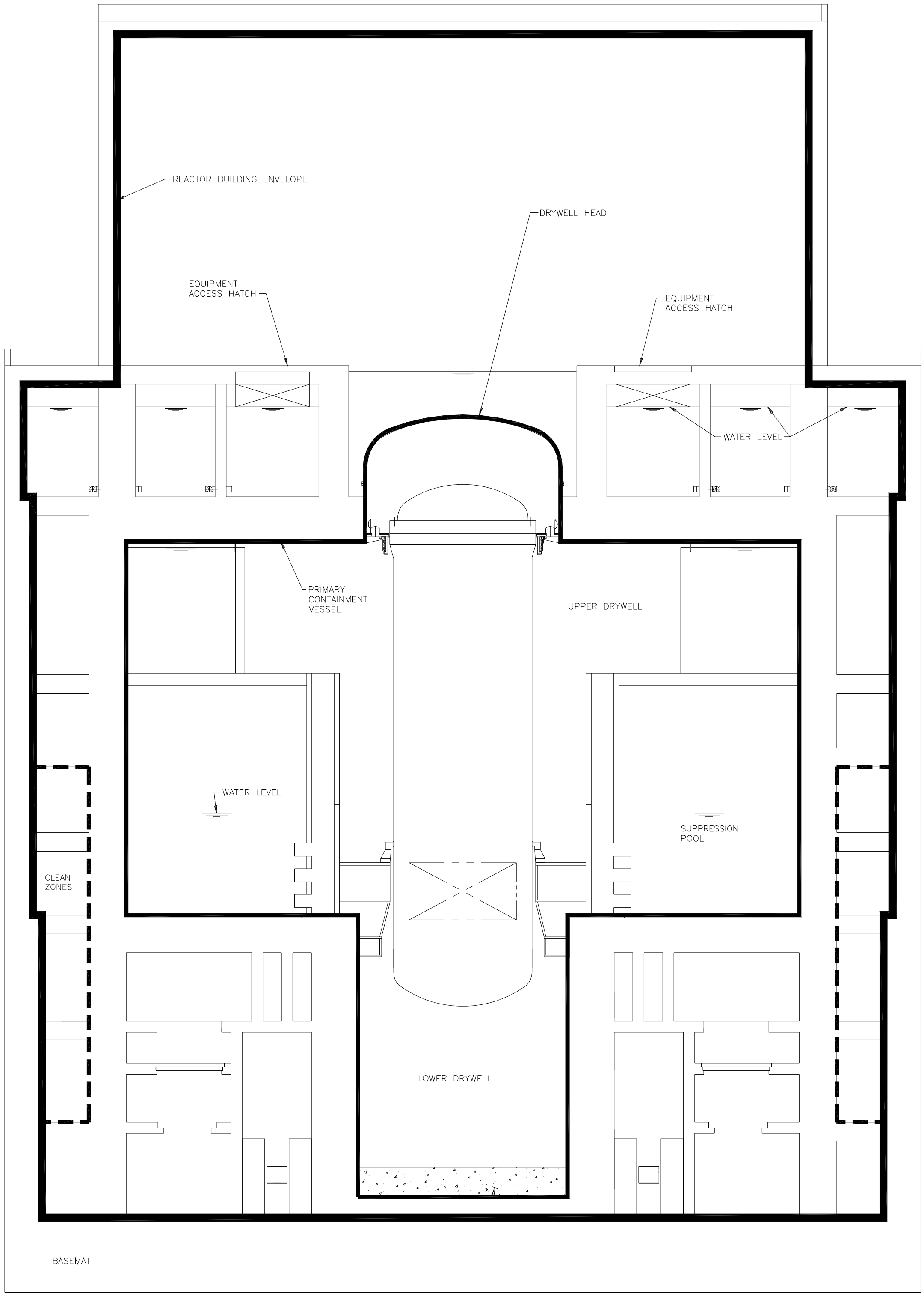


Figure 6.2-17. Reactor Building Envelope

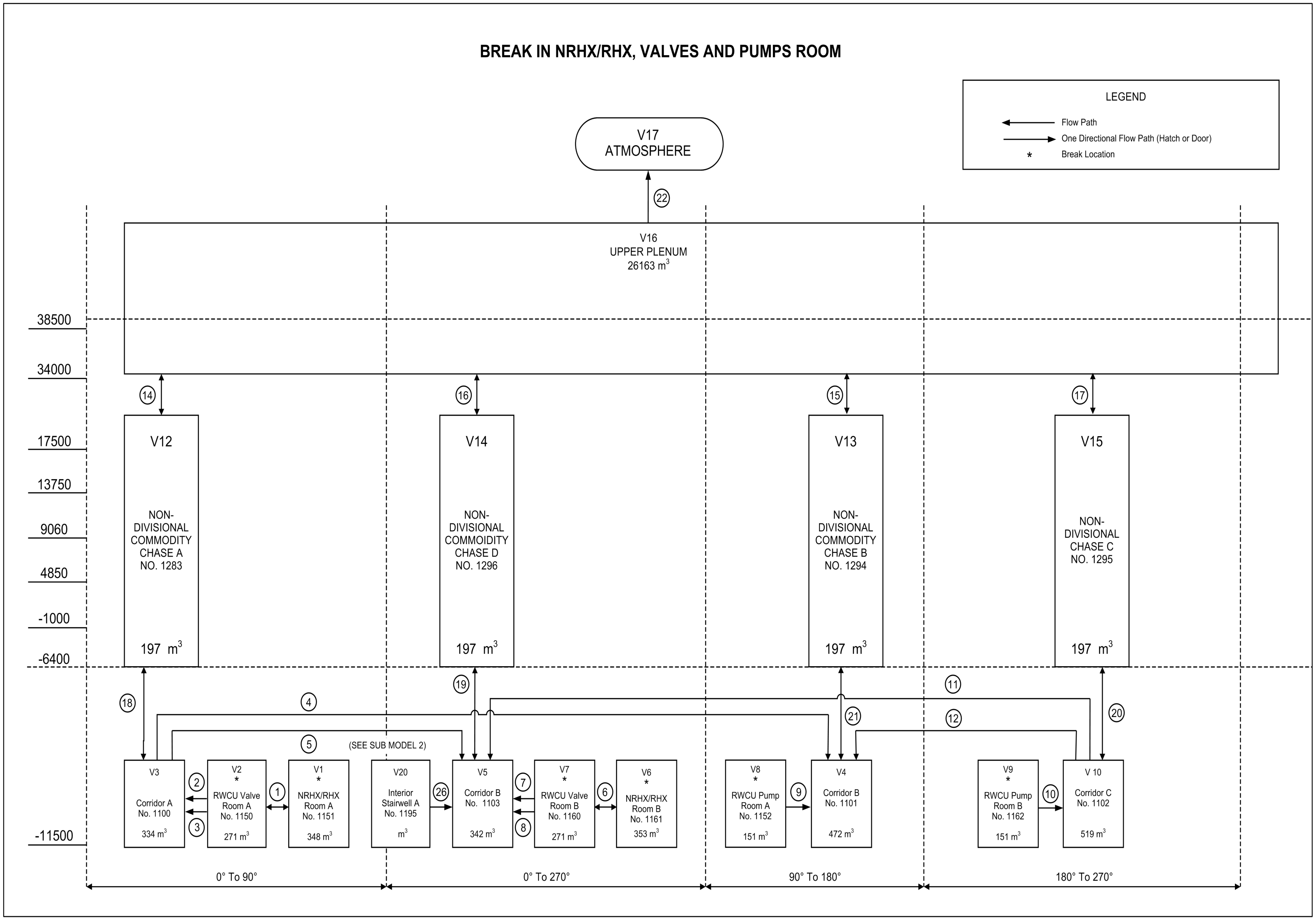


Figure 6.2-18. RWCU System Subcompartment Pressurization Analysis

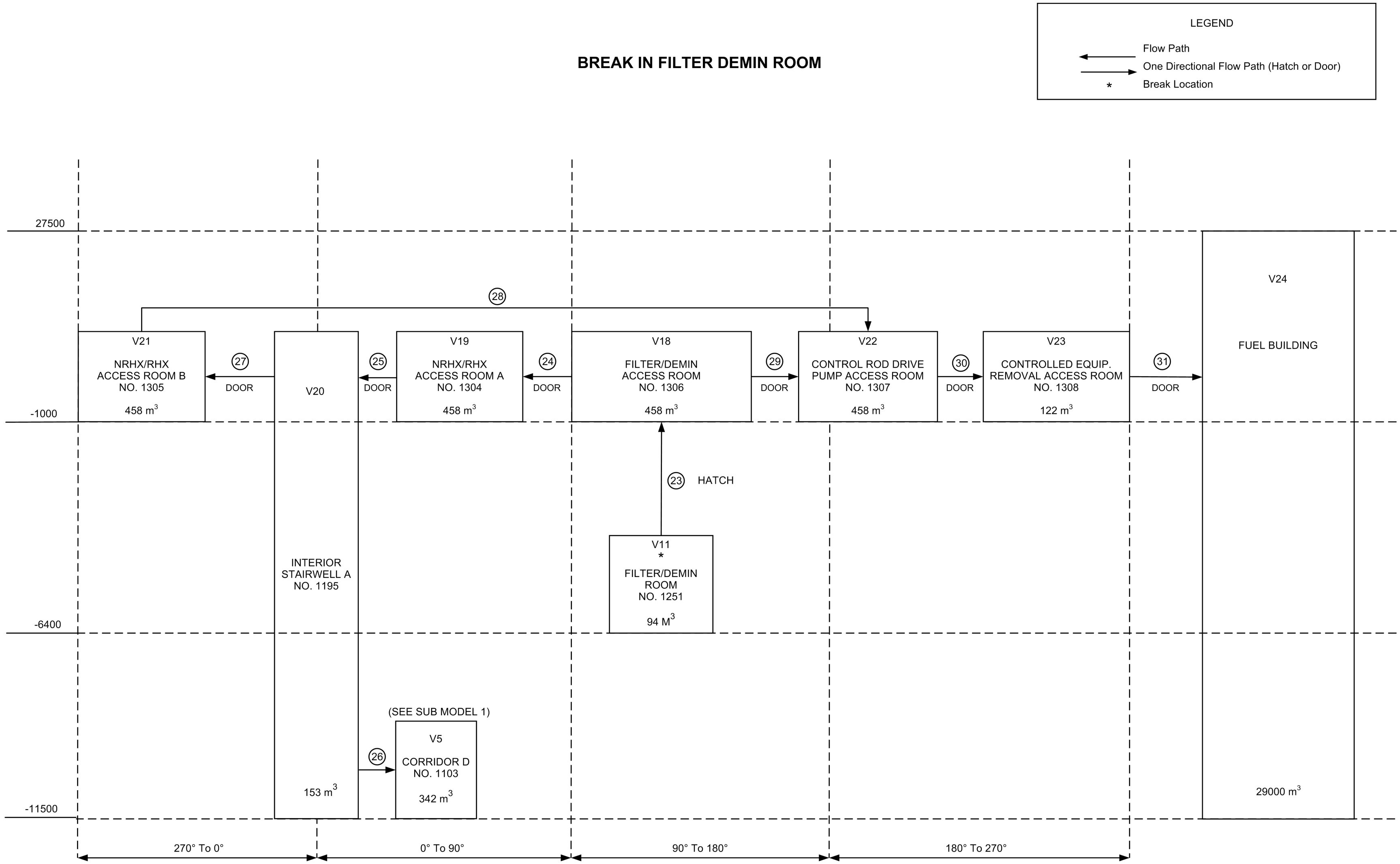


Figure 6.2-18. (continued) RWCU/SDC System Subcompartment Pressurization Analysis (Sub-Model 2)

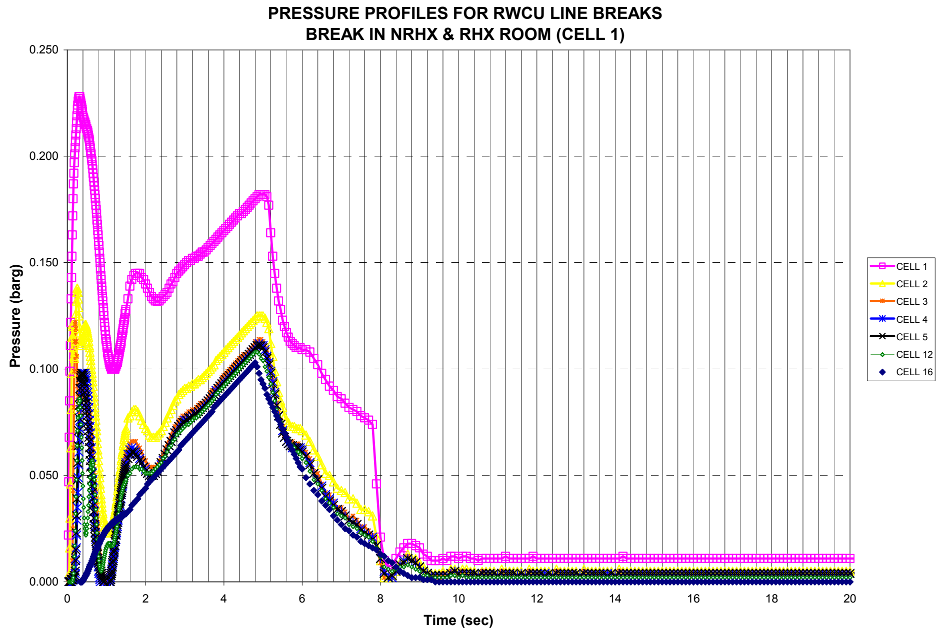


Figure 6.2-19. Pressure Histories due to Break Case 1 in Cell 1 (Sub-Model 1)

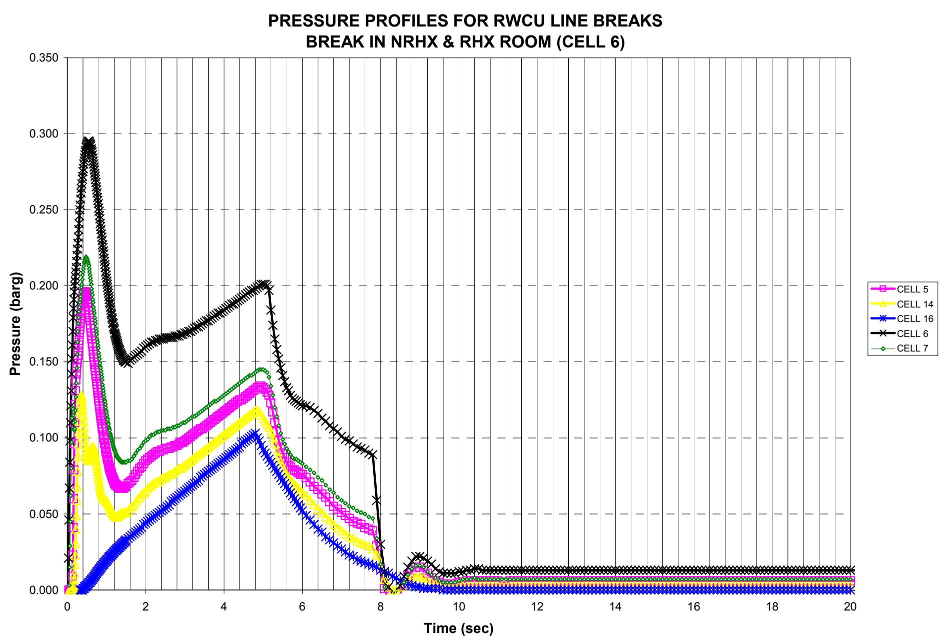


Figure 6.2-20. Pressure Histories due to Break Case 1 in Cell 6 (Sub-Model 1)

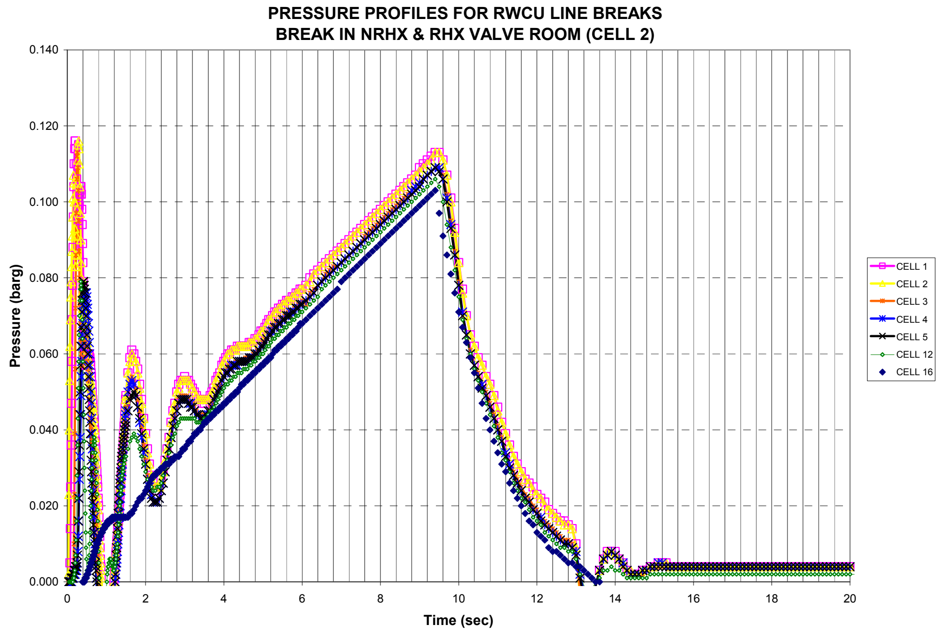


Figure 6.2-21. Pressure Histories due to Break Case 2 in Cell 2 (Sub-Model 1)

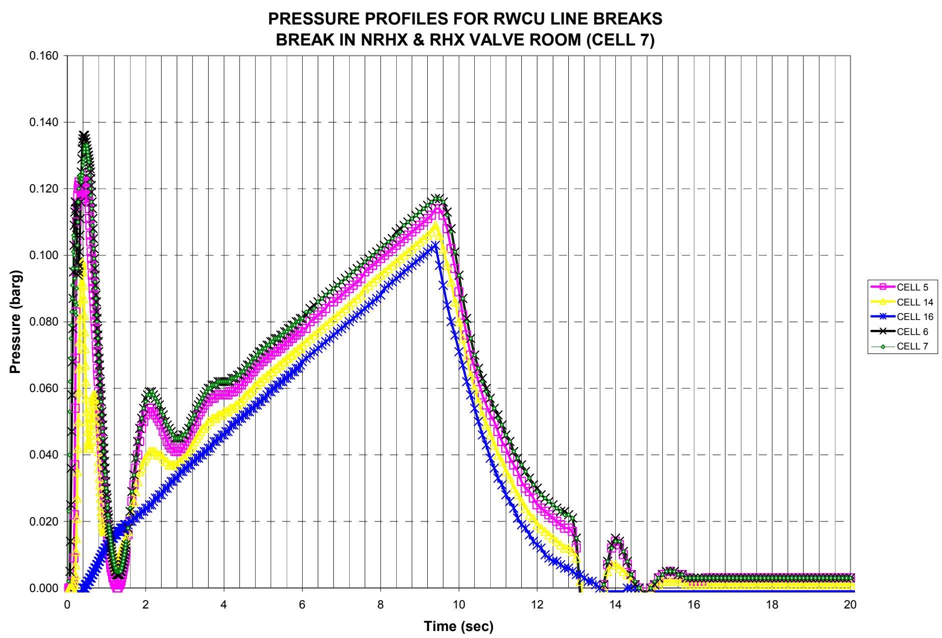


Figure 6.2-22. Pressure Histories due to Break Case 2 in Cell 7 (Sub-Model 1)

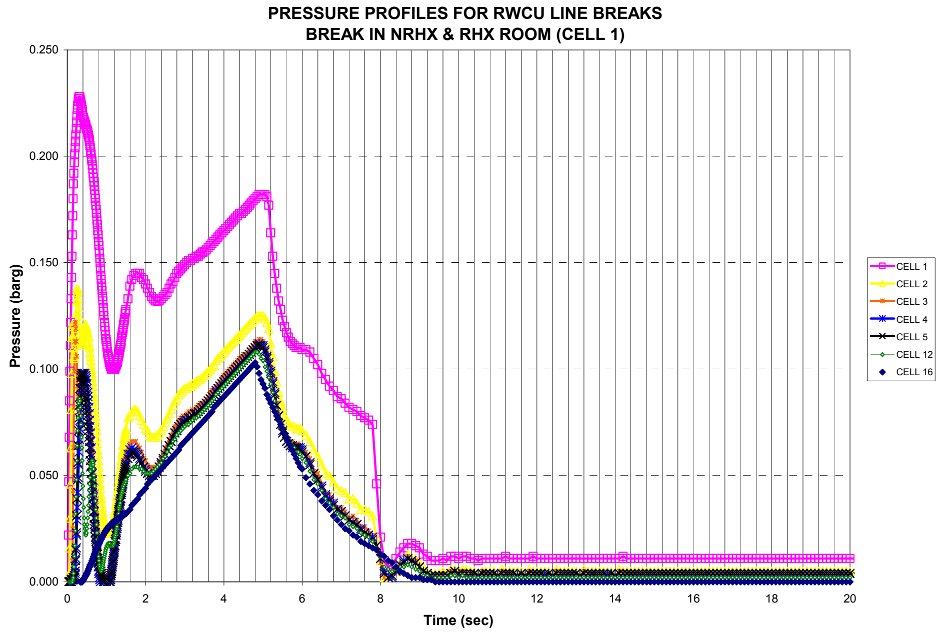


Figure 6.2-23. Pressure Histories due to Break Case 3 in Cell 1 (Sub-Model 1)

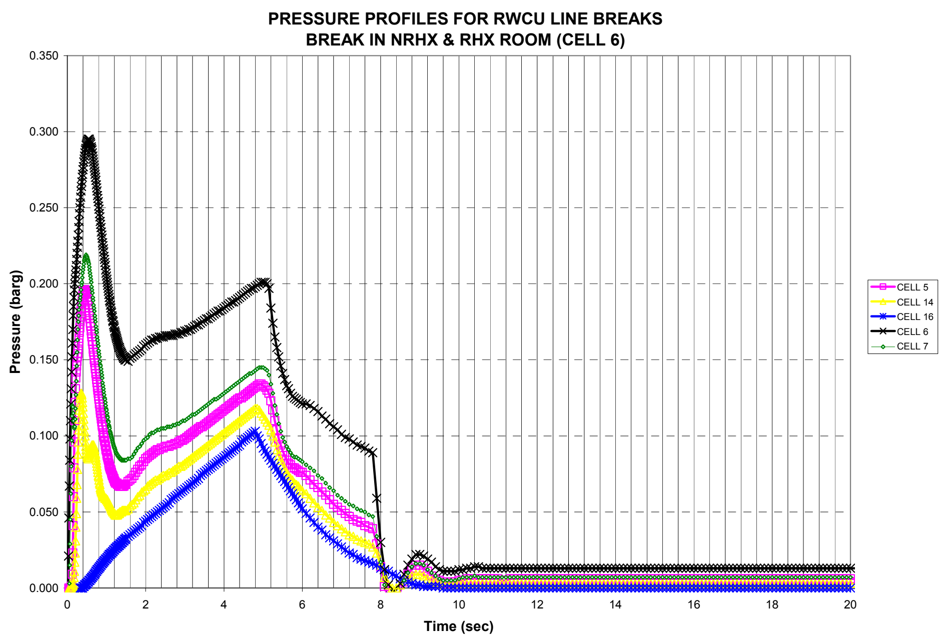


Figure 6.2-24. Pressure Histories due to Break Case 3 in Cell 6 (Sub-Model 1)

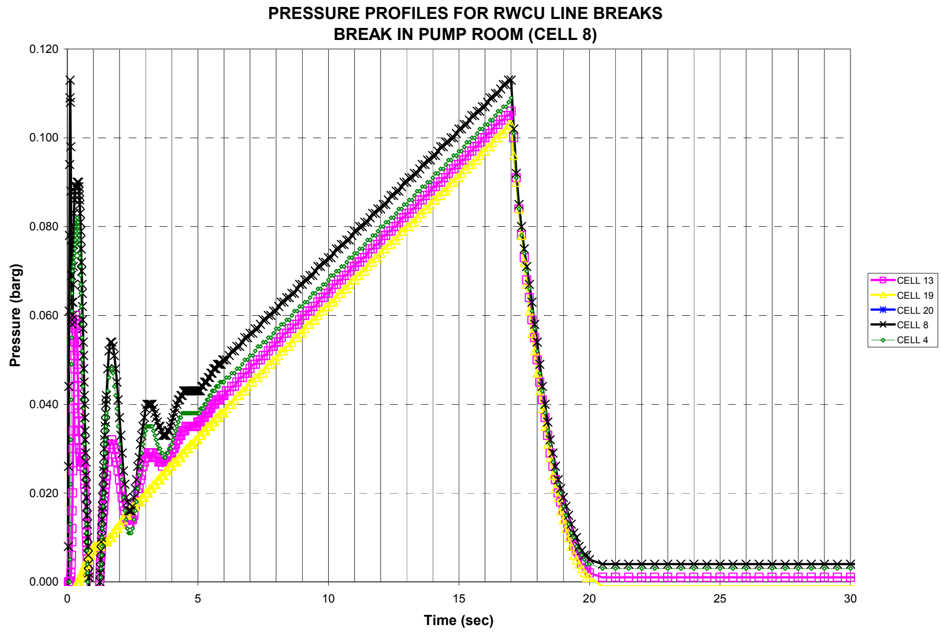


Figure 6.2-25. Pressure Histories due to Break Case 4 in Cell 8 (Sub-Model 1)

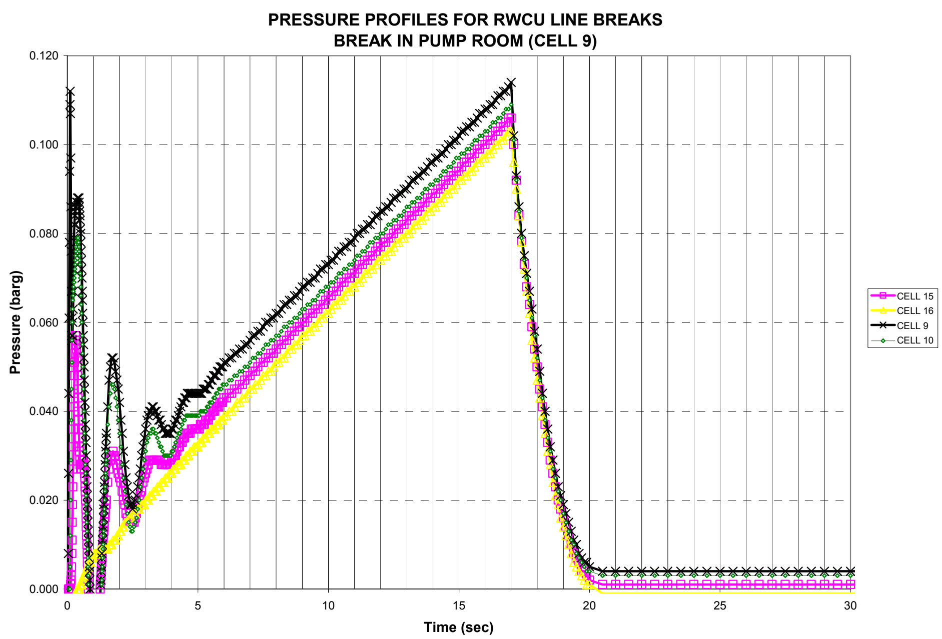


Figure 6.2-26. Pressure Histories due to Break Case 4 in Cell 9 (Sub-Model 1)

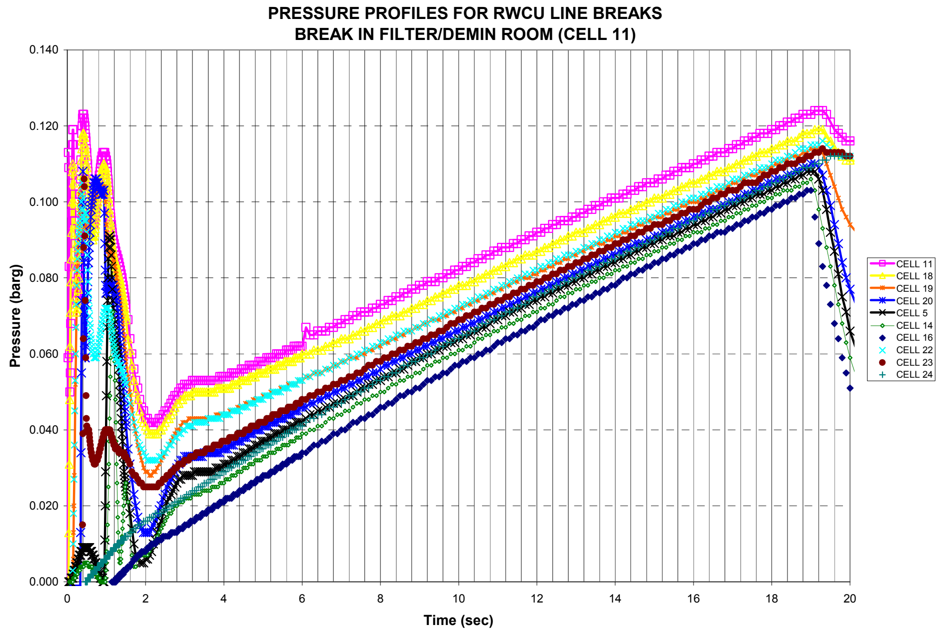


Figure 6.2-27. Pressure Histories due to Break Case 5 in Cell 11 (Sub-Model 2)

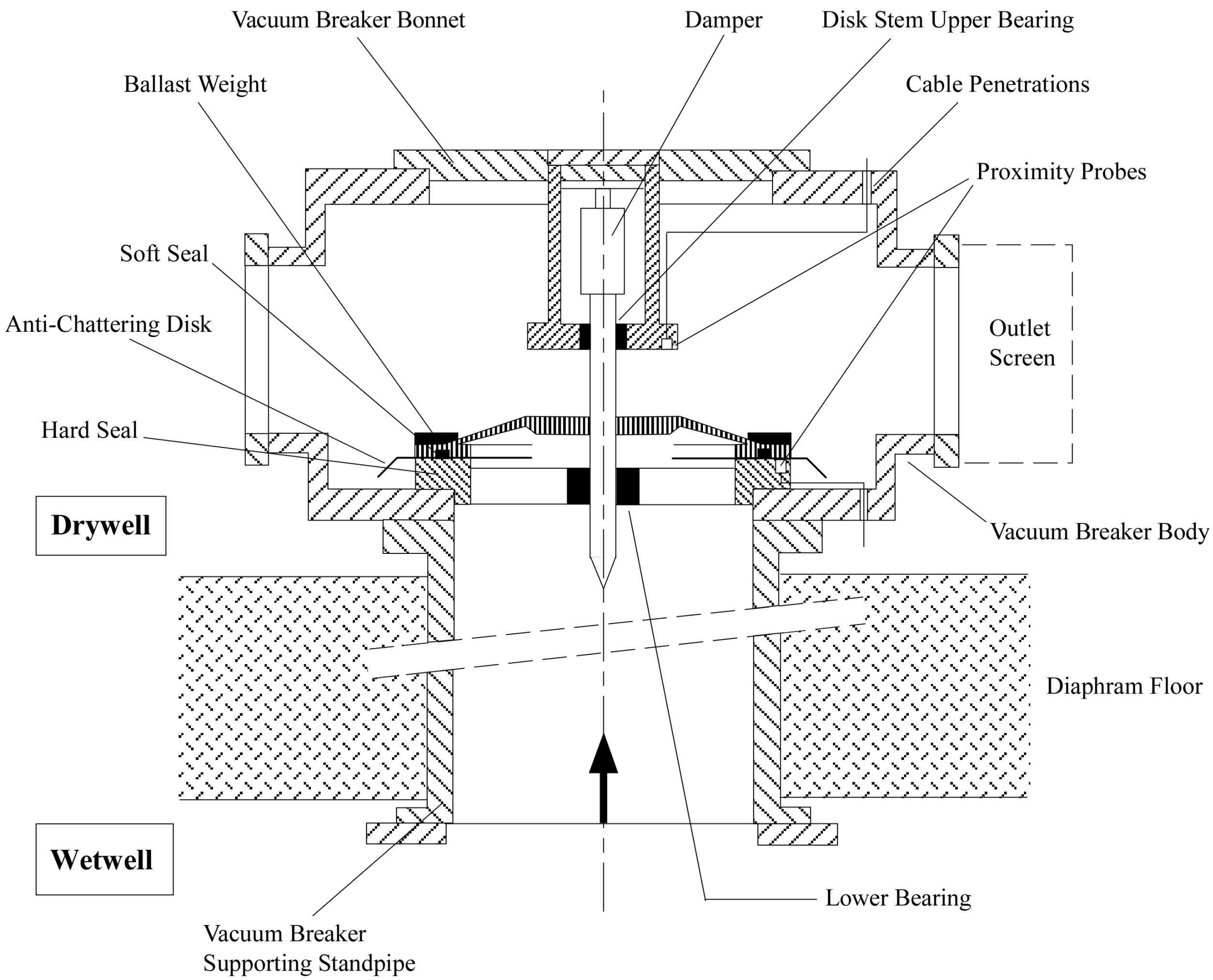


Figure 6.2-28. Vacuum Breaker Design Features

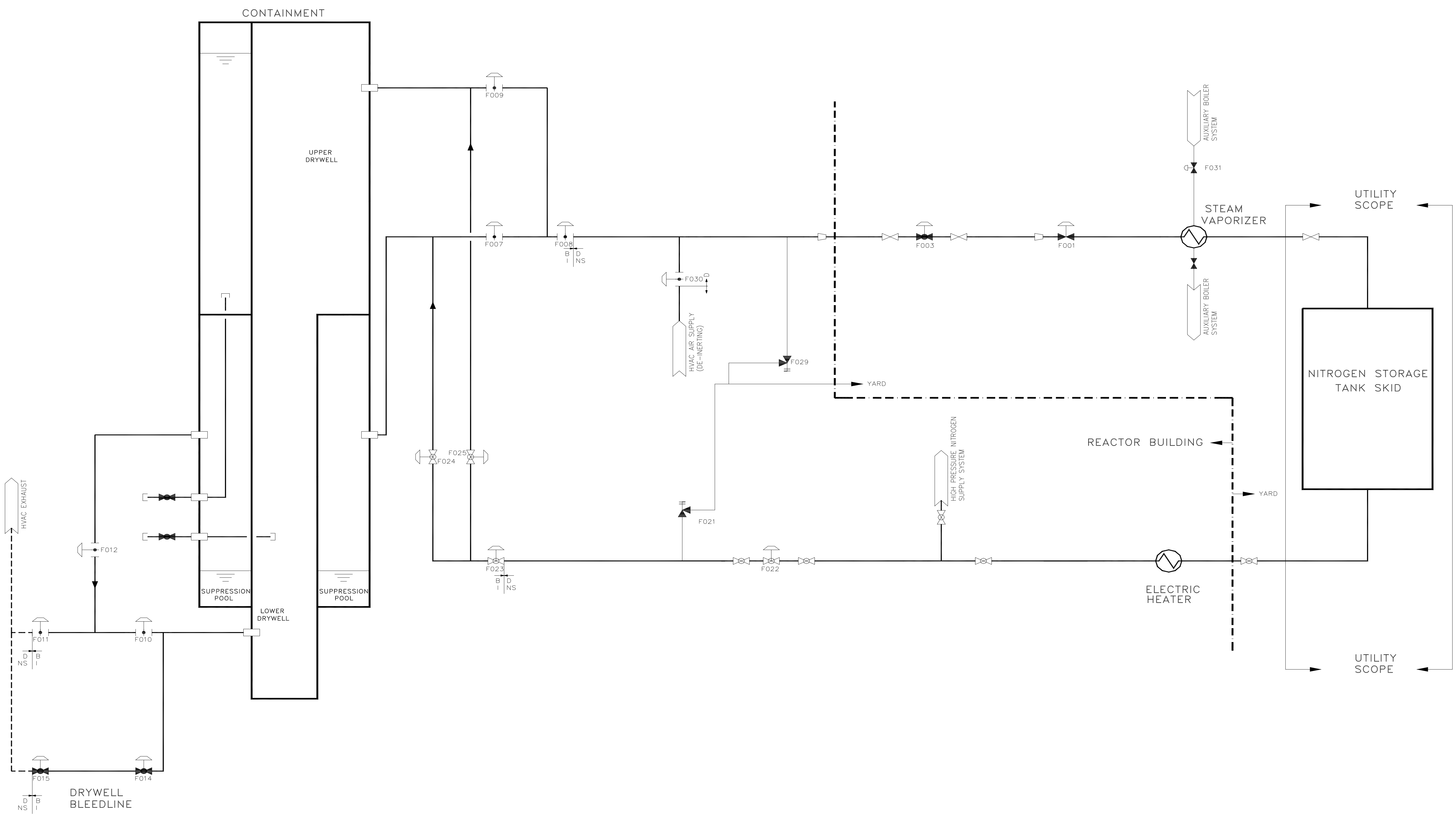


Figure 6.2-29. CIS Simplified System Diagram

6.3 EMERGENCY CORE COOLING SYSTEMS

Relevant to ESBWR ECCS, this subsection addresses or references to other DCD locations that address the applicable requirements of GDC 2, 4, 5, 13, 17, 19, 27, 35, 36 and 37, 10 CFR 50.46, Three Miles Island (TMI) action plan items in 10 CFR 50.34(f), discussed in Standard Review Plan (SRP) 6.3.

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 (for example, 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. Further discussion on GDC 17 is given in Sections 8.1.5.2.4 and 8.3.1.2.1.

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 LOCA modeling code has been reviewed and approved by the NRC, and the ECCS performance analysis results demonstrate that the 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 Gravity-Driven Coolant System (GDCS) and Standby Liquid Control (SLC) system are initiated via the use of squib valves 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, and depressurization valves the 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 is the Gravity-Driven Cooling System (GDCS), Isolation Condenser System (ICS), The SLC system, and the ADS function of the NBS.

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 (for example, 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 6.3-6.
- 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 provided 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 RB 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 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 Subsection 5.4.6 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 Subsection 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 through 6.3.2.10.

6.3.2.1 Equipment and Component Descriptions

The starting signal for the ECCS comes from independent and redundant sensors as per Table 6.3-1, item B.1. 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.

For equipment and component description detail, see individual system Subsections 6.3.2.7 through 6.3.2.10.

Because GDCS flow is gravity driven, Net Positive Suction Head (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 that are essential to the core cooling function. Institute of Electrical and Electronic Engineers (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, two control switches of the "arm/fire" type located in the MCR are to be both actuated. Inadvertent manual actuation is prevented by four deliberate operator actions (two for "arm" and two for "fire").

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 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 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. Since inadvertent actuation of the automatic logic circuits could result in loss of GDCS pool inventory and consequent unavailability of water for injection into the reactor vessel on a valid GDCS actuation signal, a set of safety-related temperature switches are used to inhibit deluge actuation as long as the drywell temperature is less than a preset value.

The GDCS requires no external electrical power source or operator intervention. The GDCS initiation signal is the receipt of a confirmed ECCS initiation signal from the NBS (see Subsection 7.3.1.2). 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 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 safety-related. 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 three 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 designated as Divisions A, B, C, and D. Electrical separation and mechanical train separation between the divisions is provided. The mechanical trains A and D draw water from independent pools designated as A and D and trains B and C draw water from a common pool designated as B/C. Physical separation is ensured between divisions by locating each train in a different area of the reactor containment. 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. GDCS typical process flows are shown in Figure 6.3-1a.

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 Suppression Pool (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, provided another set of dedicated thermocouples also sense the drywell temperature to be higher than a preset value. 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[1] 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 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

[1] Temporary strainer will be removed after initial flushing of GDCS injection lines.

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.). 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 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.

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 enough 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 close upon reverse impulse cuased by spurious GDCS squib valve operation to protect the lower pressure piping 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 located on 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 fully open when zero differential pressure exists across the valve. 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 the DW will be covered by a perforated steel plate to prevent debris from entering pool and potentially blocking the coolant flow through the fuel. Protection against the dynamic effects associated with postulated pipe ruptures is described Section 3.6. The maximum hole diameters in the perforated steel plate are are 38 mm (1.5 inch) A splash guard is provided at 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 and the GDCS squibs are powered by divisionally separated safety-related power. The deluge valve initiation circuitry is powered nonsafety-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 boil-off 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 boil-off 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 is divided into 30 cells, with two thermocouples (channel A and B) installed in each cell, to sense the presence of molten fuel on the lower drywell floor. Temperature greater than setpoint sensed by channel A thermocouples in any two adjacent cells, coincident with channel B thermocouples also sensing temperature greater than setpoint in any two adjacent cells, initiates deluge line flow. Inadvertent actuation is prevented by the presence of an inhibit signal if another set of dedicated safety-related thermocouples monitoring the lower drywell temperature do not sense the temperature to be greater than a preset value. 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 function in the closed position to maintain RPV backflow leaktight and maintain 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 Figure 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. To minimize the probability of common mode failure, the injection line squib valve pyrotechnic booster charge is from a different batch than from the batch used in equalizing line squib valves. 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 are made from heat treated material for necessary strength.

GDCS Check Valve

The GDCS check valves are designed such that the check valve is fully open when zero differential pressure is applied across the check valve. The full open position is accomplished by valve design and installation. The check valve is a long duration submersible valve. The valve meets the minimum flow requirements for a valve stuck in the open position. The check valve is classified as Quality Group A, Seismic Category I, and ASME Section III Class 1.

Remote check valve position indication is provided in the main control room by position-indication instrumentation.

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 check valve are 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 check valve, and all piping and valves (including supports) connecting GDCS pool to lower drywell are classified as follows:

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

The electrical design is classified safety-related.

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 is 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.

To confirm functional capability, the ignitors and booster subassemblies of the GDCS squib valves are removed from the valves and tested in sequential sets during the refueling and maintenance outage at the end of each plant operating cycle. The initial qualified life for the boosters and ignitors is four (4) years. Replacement is done without any opening of the 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) is designed to:

- 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;
- Ensure a single failure of ADS does not render more than one NBS 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 to maintain function during a SSE (cannot open a closed valve or cause an open valve to close); and
- Be capable of initiating vessel depressurization over the full range of reactor vessel pressures and reactor vessel-to-drywell differential pressures down to and including a differential pressure of zero.

Nonsafety, Power Generation Design Bases

The ADS is 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.

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 a SSLC/ESF that uses the NBS instrumentation signals, and the SRVs and DPVs and their associated instrumentation and controls.

Detailed System Description

The ADS interfaces with the NBS controls for the 10 SRVs and 8 DPVs. The SRVs are described in Section 5.2.2. The DPVs are described in Section 5.4.13.

System Operation

The ADS automatically actuates in response to the ECCS initiation signals in Table 6.3-1. A two-out-of-four Level initiation logic is used to activate the SRVs and DPVs. The 10-second time delay to confirm Level initiation 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 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.

6.3.2.8.4 Testing and Inspection Requirements

See Subsection 7.3.5.4 for ADS logic testing requirements.

6.3.2.8.5 Instrumentation Requirements

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 demonstrate the adequacy of ESBWR ECCS network performance for the entire spectrum of postulated break sizes.

The analyses are based upon the bundle design shown within Section 4.3 and were performed with the TRACG model. For plant operation with nominal feedwater temperature, the analysis results are discussed in Subsection 6.3.3.7. For plant operation with feedwater temperature maneuvering (increase and reduction), the limiting breaks were evaluated and results are discussed in Reference 6.3-3.

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 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 and 6.3.3.9. Limits on minimum suppression pool water level are discussed in Subsection 6.2.1.1.2 and Table 6.2-3.

*6.3.3.2 Acceptance Criteria for ECCS Perfo*rmance

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).

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.

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. The ESBWR ECCS maintains the water level in the vessel above the core for a period of greater than 30 days following a LOCA.

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 1 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
- 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 delay time to confirm the presence of a low water level (Level 1) 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 6.3-9 (GDCS Injection Line Break with failure of one GDCS Injection Valve).

Operator action is not required for 72 hours, except as a monitoring function, following any LOCA.

6.3.3.5 Use of Dual Function Components for ECCS

The ECCS systems ADS and GDCS are designed to accomplish only one function, to cool the reactor core following a LOCA. The ECCS system SLCS is designed to be used during an ATWS, and the ECCS system ICS is designed to avoid unnecessary use of other ESFs for residual heat removal. Both, SLCS and ICS, provide additional liquid inventory upon actuation. 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. Figures 6.2-6 to 6.2-8 show the TRACG nodalization of the RPV, the containment, and the steam line system. Refer to Subsection 6.2.1.1.3.1 for the discussion of the TRACG nodalization.

6.3.3.7.2 Accident Description

The sequence of events for the 4 representative 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 6.3-6.

Conformance to the 10 CFR 50.46 acceptance criteria [PCT ≤ 1204°C (2200°F), local oxidation ≤ 17% and core-wide metal-water reaction ≤ 1%] is demonstrated for the fuel parameters listed in Table 6.3-1. For each bundle design in a plant, comformance is reconfirmed for the limiting break . 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 valves, 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 equalization line break, SLC injection line break, the GDCS injection line break and the bottom head drain line break were analyzed. Results show that the GDCS injection line break and the bottom drain line break bound the other small line breaks. Important variables from these two 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. Because the isolation condenser system is part of the ECCS network, once the break is isolated, the isolation condensers, High Pressure Control Rod Drive flow and/or the ADS/GDCS systems 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, that is, 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). The results for maximum GDCS injection line break with 1 GDCS valve failure and maximum inside steam line break with 1 DPV failure are slightly more limiting than the other LOCA cases.

For each bundle design in a plant, comformance is reconfirmed by the limiting break.

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 significant plant parameters in the conservative direction simultaneously. The maximum inside steam line break cases (refer to Subsection 6.3.3.7.8) and the GDCS injection line break (the most limiting break location, refer to Table 6.3-5) were evaluated. The results of these calculations are given in Table 6.3-5. The GDCS injection line break with a GDCS injection valve failure results 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

The ECCS-LOCA perforemance analyses are performed according to the key parameters listed in Table 6.3-11. Results of these analyses demonstrate the compliance with all the applicable acceptance criteria. It is concluded that the ECCS would perform its function in an acceptable manner.

6.3.4 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.

6.3.4.1 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)

All ECCS safety-related valves are tested during plant initial power ascension per Regulatory Guide 1.68, Appendix A, except that the mechanical components of the ECCS squib type valves will be fully tested by the manufacturer prior to delivery to the site.

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 Technical Specifications. Components inside the drywell can be visually inspected only during periods of access to the drywell.

6.3.5 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 ECCS initiating signals are shown in Table 6.3-1.

6.3.6 COL Information

6.3-1-H ECCS Testing Requirements (Deleted)

6.3-2-H Limiting Break Results (Deleted)

6.3.7 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 and NEDO-33083-A, Class I (non-proprietary), October 2005

6.3-3 GE-Hitachi Nuclear Energy, “ESBWR Feedwater Temperature Operating Domain Transient and Accident Analysis”, NEDO-33338, scheduled October 2007.

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]
Scram Initiating Signal	---	Loss of power to two feedwater pump buses
Maximum Sensor Response Time	sec	2

B. Emergency Core Cooling System Parameters

B.1 ECCS Initiation Signal

Variable	Units	Value
Initiating Signal: Level 1	meters (above TAF) [ft] (above TAF)	4.05 [13.28]
Maximum Allowable Time Delay to Confirm ECCS-LOCA Signal	sec	10

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

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

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]	13.88 [490.1]
B.4 Standby Liquid Control System		
Variable	**Units**	**Value**
Initiating Signal	—	DPV actuation (See B.5)
Liquid Volume per Tank	m^3 [ft^3]	7.8 [275.4]
B.5 Automatic Depressurization Subsystem		
Variable	**Units**	**Value**
Initiating Signal	—	ECCS-LOCA confirmed initiating signal (See B.1)

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

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]	5.18×10^5 8.618 [1.14×10^4] [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]
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]

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

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)	—	1.10

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
Total inventory (for 3 GDCS pools) at GDCS pool low water level	1830 m^3 (64626 ft^3)
Minimum total drainable inventory (for 3 GDCS pools) at low water level of 6.5 meters	1636 m^3 (57775 ft^3)
Minimum elevation of GDCS pool surfaces above the RPV nozzles, at GDCS pool low water level	13.5 m (44.3 ft)
Minimum long-term core cooling flow delivered by the GDCS equalizing lines for a minimum ΔP of 9.12 kPa (1.32 psid) across the equalizing lines	22.7 m^3/hr (100 gpm)
Minimum flow through the deluge lines required to flood the lower drywell region	256 m^3/hr (1127 gpm)*
Minimum available suppression pool water inventory 1 meter above TAF with 1.0 m of equalizing line driving head	799 m^3 (28216 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

Inservice Testing and Maintenance

Component	Type of Test	Description of Test
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 (Not used)

Table 6.3-5
Summary of ECCS-LOCA Performance Analyses

Break Location	Break Size+ m^2 (ft^2)	Minimum Chimney Static Head* Level with reference to 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:											
Steam Line Inside Containment	0.09832 (1.058)	9.04 (29.65)	9.06 (29.74)	8.98 (29.46)	No heatup	<1.0	7.38 (24.22)	7.39 (24.25)	7.59 (24.89)	Increases	Decreases
Feedwater Line++	0.07420 (0.7986)	9.24 (30.32)	8.84 (29.00)	9.13 (29.97)	No heatup	<1.0	6.30 (20.67)	6.40 (21.01)	6.56 (21.51)	Increases	Decreases
GDCS Injection Line	0.004561 (0.04910)	9.09 (29.84)	8.82 (28.95)	9.09 (29.83)	No heatup	<1.0	6.22 (20.39)	6.07 (19.91)	6.41 (21.02)	Increases	Decreases
Bottom Head Drain Line	0.004052 (0.04361)	9.00 (29.51)	8.90 (29.21)	9.01 (29.57)	No heatup	<1.0	5.89 (19.32)	6.21 (20.37)	5.98 (19.61)	Increases	Decreases
Based on bounding values:											
Steam Line Inside Containment	0.09832 (1.058)	9.25 (30.34)	9.28 (30.46)	9.30 (30.52)	No heatup	<1.0	6.97 (22.88)	7.06 (23.15)	7.04 (23.10)	Increases	Decreases
GDCS Injection Line	0.004561 (0.04910)	8.69 (28.51)	8.50 (27.89)	8.92 (29.26)	No heatup	<1.0	5.81 (19.08)	5.90 (19.37)	5.94 (19.48)	Increases	Decreases

+ The break area is from the RPV side of the break.

++ For the feedwater line break, the total break area from the turbine building side is limited at the two parallel ventrui sections, with flow area of 0.04997 m^2 each.

* Chimney static head level with reference to vessel zero is calculated by adding the equivalent height of water corresponding to the static head of the two-phase mixture inside the chimney to the elevation (7.896 m) of bottom of chimney.

** 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, 3 ICs***, 2 SLC accumulators and 4 GDCS with 8 Injection Lines
One Safety/Relief Valve	9 SRVs, 8 DPVs, 3 ICs***, 2 SLC accumulators and 4 GDCS with 8 Injection Lines
One GDCS Injection Valve	10 SRVs, 8 DPVs, 3 ICs***, 2 SLC accumulators 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.

*** Assuming only 3 ICs available at the start of the LOCA event, the 4^{th} IC is assumed to be out of service and not available.

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. Loss of power generation bus initiates signals for scram and IC.
~ 1	High Drywell pressure setpoint reached; Scram signal from high drywell pressure is not credited in this analysis.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion started at 0.25 second later.
Vent Clearing Time	Top vent: 1.9 sec, Middle vent: 2.3, Bottom vent: 3.0 sec.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves started to open at 15 seconds later.
4	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
12	Low MSL pressure setpoint reached, MSIV closure initiated in 0.7 seconds later.
13	Level 2 is reached (Level 2 MSIV closure and IC initiation signals are not credited in this analysis).
16	Reactor is isolated on low MSL pressure setpoint.
266	Level 1 is reached.
276	Level 1 confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
326	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLCS flow starts on DPV actuation.
426	GDCS timer (150 seconds after confirmed Level 1 signal) runs out; GDCS injection valves open.
446	Vessel pressure decreases below the maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.

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**
566	Minimum chimney water level (8.84 m) is reached.
627	SLCS flow depleted.
From 600 to 2000	RPV water level remains higher than Level 0.5. Therefore, equalizing line valves do not open for this event.

Table 6.3-8

Operational Sequence of ECCS For a Main Steam Line Break with Failure of One GDCS Injection Valve (Nominal Calculation)

Time (sec)	**Events**
0	Guillotine break of main steam line inside containment; normal auxiliary power assumed to be lost; Feedwater is lost. Loss of power generation bus initiates signals for scram and IC.
< 1	High Drywell pressure setpoint reached, scram signal from high drywell pressure is not credited in this analysis.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion started at 0.25 second later.
Vent clearing time	Top vent: 1.7 sec. Middle vent: 2.1 sec. Bottom vent: 2.8 sec.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time; drain valves start to open at 15 seconds later.
6	Low MSL pressure setpoint reached; MSIV closure initiated at 0.7 second later.
8	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
10	Reactor is isolated on low MSL pressure setpoint.
17	Level 2 is reached (Level 2 MSIV closure and IC initiation signals are not credited in this analysis).
463	Level 1 is reached.
473	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
523	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC flow starts on DPV actuation.
623	GDCS timer timed out; (150 seconds after confirmed Level 1 signal); GDCS injection valves open.

Table 6.3-8

Operational Sequence of ECCS For a Main Steam Line Break with Failure of One GDCS Injection Valve (Nominal Calculation)

Time (sec)	**Events**
636	Vessel pressure decreases below the maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
823	SLCSflow depleted
945	Minimum chimney water level (9.14 m) is reached.
From 900 to 2000	RPV water level remains higher than Level 0.5. Therefore, equalizing line valves do not open for this event.

Table 6.3-9

Operational Sequence of ECCS For a GDCS Line Break with Failure of One GDCS Injection Valve (Nominal Calculation)

Time (sec)	**Events**
0	Guillotine break of GDCS line inside containment; normal auxiliary power assumed to be lost; feedwater is lost. Loss of power generation bus initiates signals for scram and IC.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion started at 0.25 second later.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves started to open at 15 seconds later.
4	High drywell pressure setpoint reached; Scram signal from high drywell pressure is not credited in this analysis.
6	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
Vent clearing time	Top Vent: 9.4 sec, Middle Vent; 208 sec, Bottom Vent never cleared.
15	Level 2 is reached (Level 2 MSIV closure and IC initiation signals are not credited in this analysis).
16	Low MSL pressure setpoint reached; MSIV closure initiated in 0.7 second later.
20	Reactor isolated on low MSL pressure setpoint
144	Level 1 is reached.
154	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated. SRV actuated.
204	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC flow starts on DPV actuation.
304	GDCS timer (150 seconds after confirmed Level 1signal) timed out; GDCS injection valves open.
419	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start

Table 6.3-9

Operational Sequence of ECCS For a GDCS Line Break with Failure of One GDCS Injection Valve (Nominal Calculation)

Time (sec)	**Events**
	to rise.
504	SLCS flow depleted.
525	Minimum chimney water level (8.82 m) is reached.
From 900 to 2000	RPV water level remains higher than Level 0.5. Therefore equalizing line valves do not open for this event.

Table 6.3-10

Operational Sequence of ECCS For a Bottom Drain Line Break with Failure of One GDCS Injection Valve (Nominal Calculation)

Time (sec)	**Events**
0	Guillotine break of Bottom Drain Line inside containment; normal auxiliary power assumed to be lost; Feedwater is lost. Loss of power generation bus initiates signals for scram and 1C.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion started at 0.25 second later.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves started to open at 15 seconds later.
5	High drywell pressure setpoint reached; Scram signal not credited in this analysis.
6	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
Vent Clearing time	Top Vent: 293 sec, Middle Vent: 348, Bottom Vent never cleared.
15	Low main steamline pressure setpoint reached; MSIV closure initiated in 0.7 seconds later.
16	Level 2 is reached (Level 2 MSIV closure and IC initiation signals are not credited in this analysis).
19	Reactor is isolated on low MSL pressure setpoint
286	Level 1 is reached.
295	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
346	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLCS flow starts On DPV actuation.
445	GDCS timer (150 seconds after confirmed Level 1 signal) timed out. GDCS injection valves open.

Table 6.3-10

Operational Sequence of ECCS For a Bottom Drain Line Break with Failure of One GDCS Injection Valve (Nominal Calculation)

Time (sec)	**Events**
558	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
647	SLCS flow depleted.
685	Minimum chimney water level (8.9 m) is reached.
From 900 to 2000	RPV water level remains higher than Level 0.5. Therefore equalizing line valves do not open for this event.

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	1994 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, Table 2.5-2.

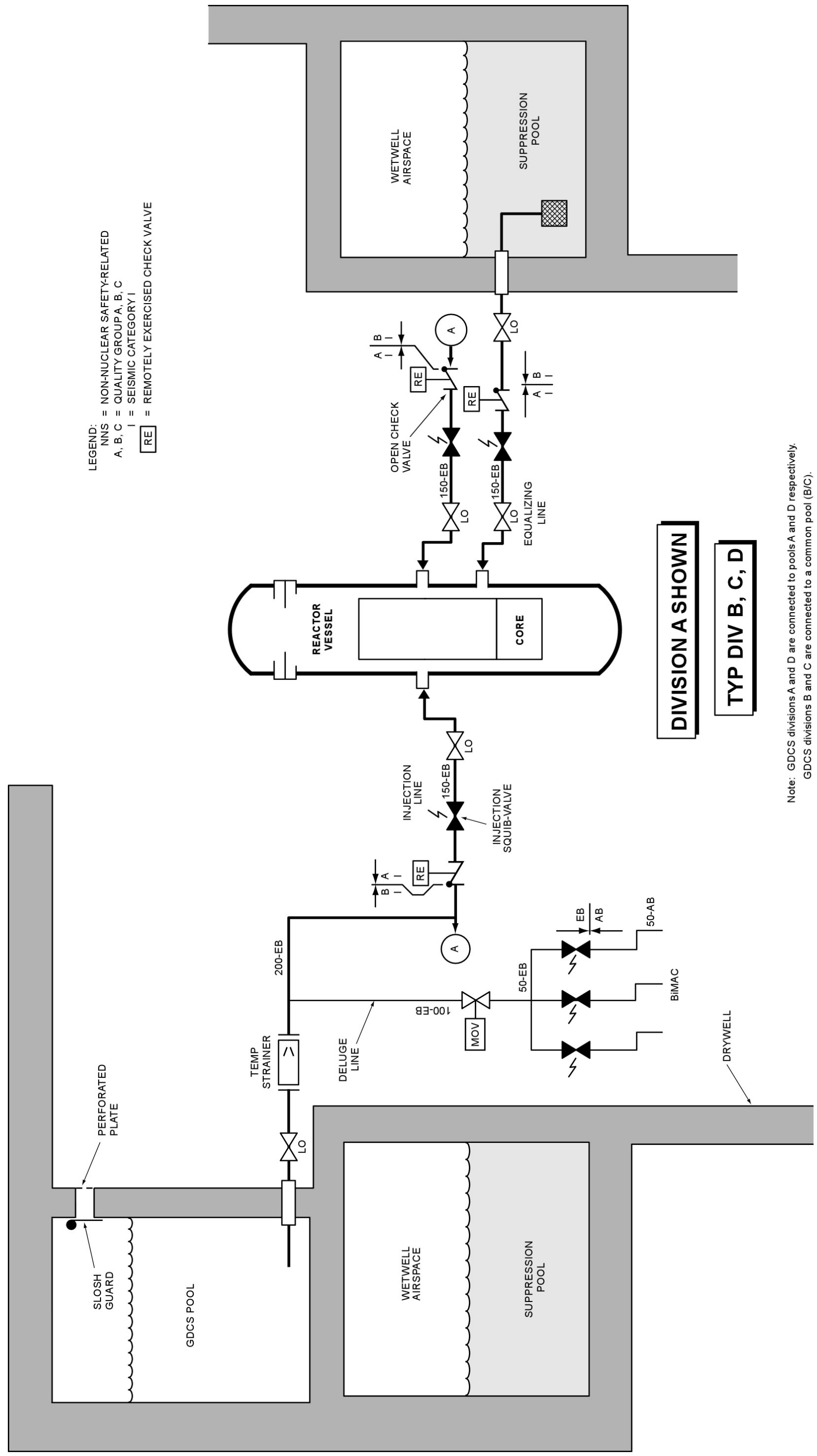
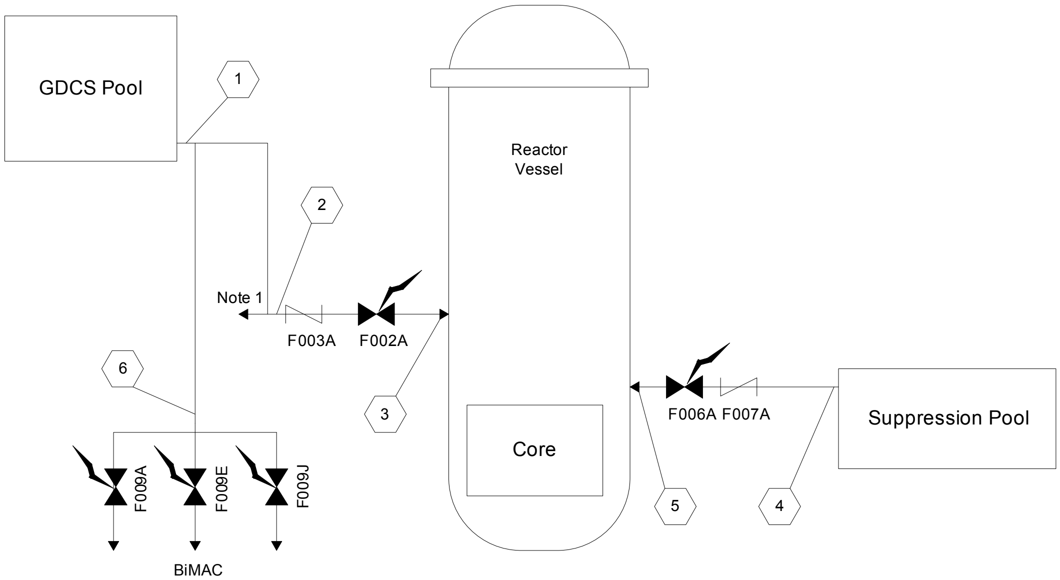


Figure 6.3-1. GDCS Configuration



Mode A: Standby						
Position ⬡	1	2	3	4	5	6
Temperature °C (°F)	43 (110)	43 (110)	288 (550)	43 (110)	288.0 (550)	43.4 (110)
Pressure kPa(d) (psid)	162 (23.5)	256 (37.1)	7241 (1050.7)	16.2 (2.3)	7256 (1052.4)	162 (23.5)

Mode B: Design Basis (Note 2)						
Position ⬡	1	2	3	4	5	6
Flow kg/s (lb/hr)	115 (912704)	57.5 (456352)	57.5 (456352)	27.8 (201623)	27.8 (201623)	0 0
Temperature °C (°F)	47 (117)	47 (117)	47 (117)	61 (142)	61 (142)	43.4 (110)
Pressure kPa(d) (psid)	318 (46.2)	386 (56.0)	329 (47.8)	223 (32.3)	225 (32.6)	161.988 (23.5)

Mode C: Severe Accident	
Position ⬡	6
Flow kg/s (lb/hr)	Note 3
Temperature °C (°F)	Note 3
Pressure kPa(d) (psid)	Note 3

Valve Position	Mode A	Mode B	Mode C
F002A	C	O	C
F003A	O	C/O	O
F006A	C	O	C
F007A	O	C/O	O
F009A,E,J	C	C	O

O=Open, C=Closed, C/O=Initially closes then opens based on system back pressure

Notes:

1. To parallel injection line nozzle with equal flow rate at node 3.
2. Flow rates vary with drywell, wetwell, and reactor vessel pressures which are time dependent and vary with the type of postulated design basis accident.
3. Severe accident parameters are discussed in DCD Tier 2, Chapter 21.

Figure 6.3-1a. GDCS Typical Process Flows

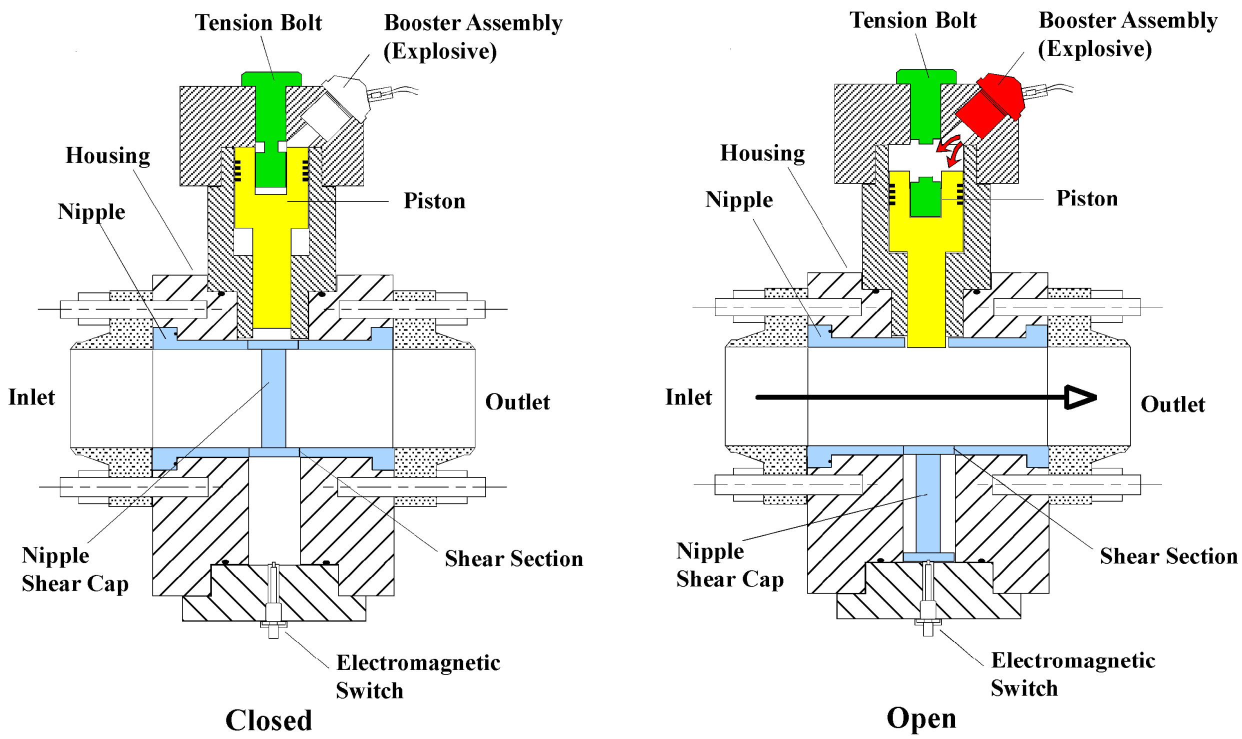


Figure 6.3-2. Typical GDCS Squib Valve

Figure 6.3-3.
(Not Used)

Figure 6.3-4.
(Not Used)

Figure 6.3-5.
(Not Used)

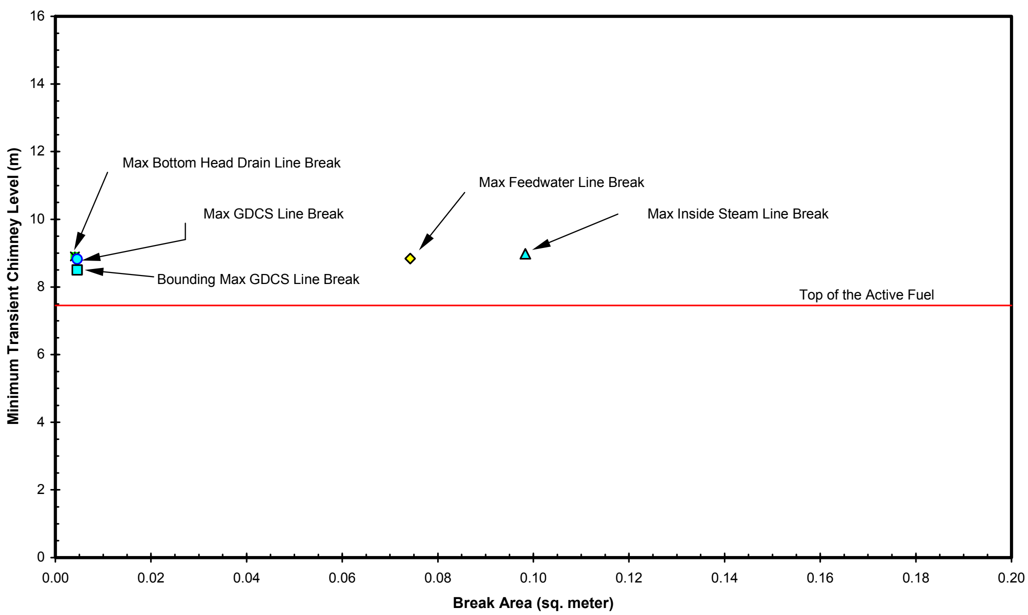


Figure 6.3-6. Minimum Transient Chimney Water Level vs. Break Area

Y:\Hwang\FWL2_1INJ\FWL2_1INJ.GRF
10/ 3/2006:15:32:27

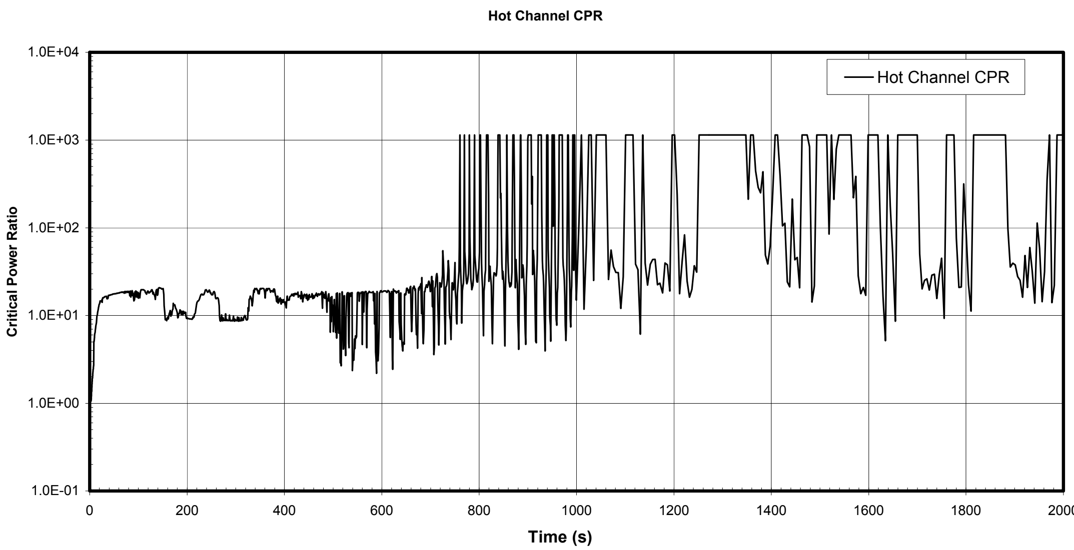


Figure 6.3-7a. MCPR, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

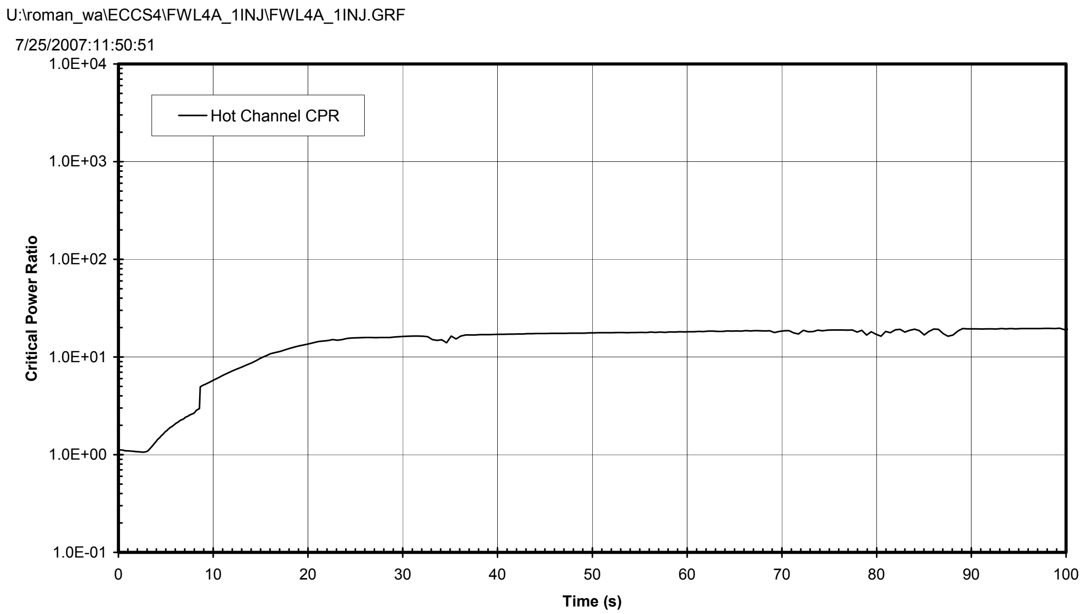


Figure 6.3-7b. MCPR, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

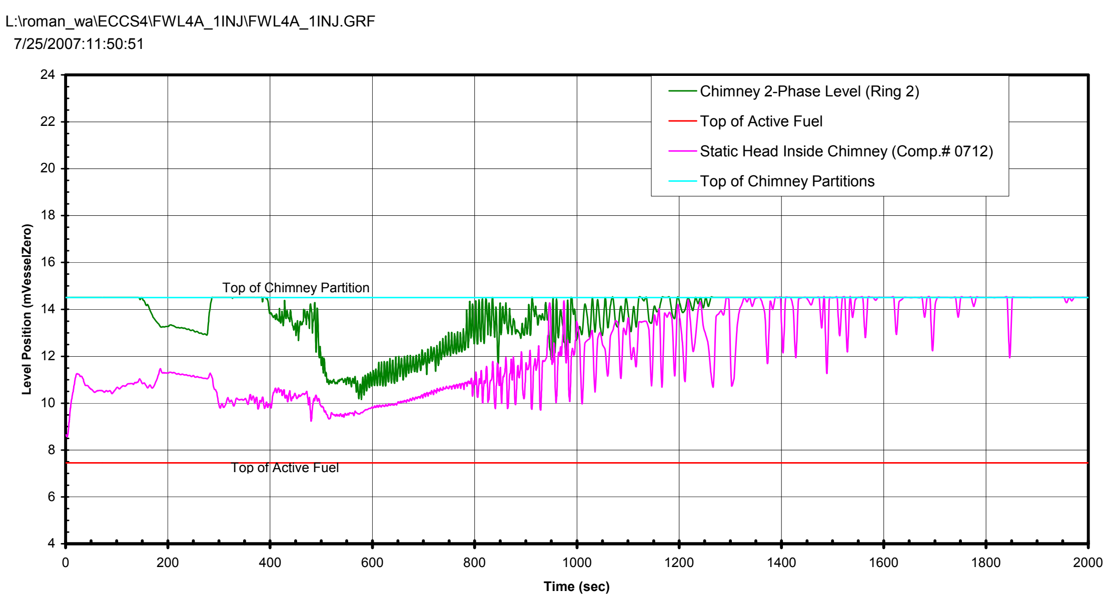


Figure 6.3-8a. Chimney Water Level, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

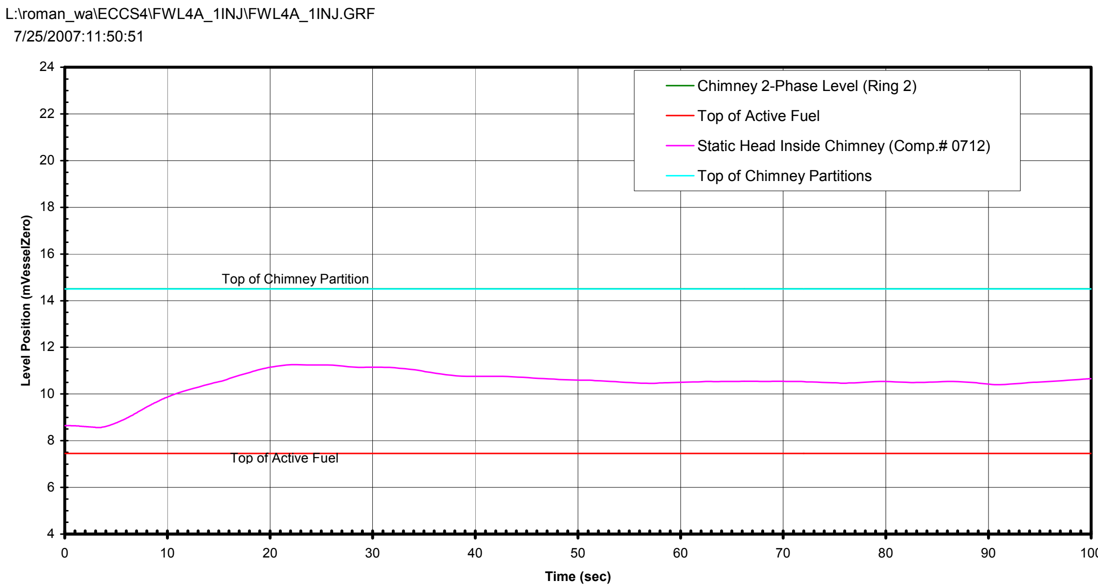


Figure 6.3-8b. Chimney Water Level, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

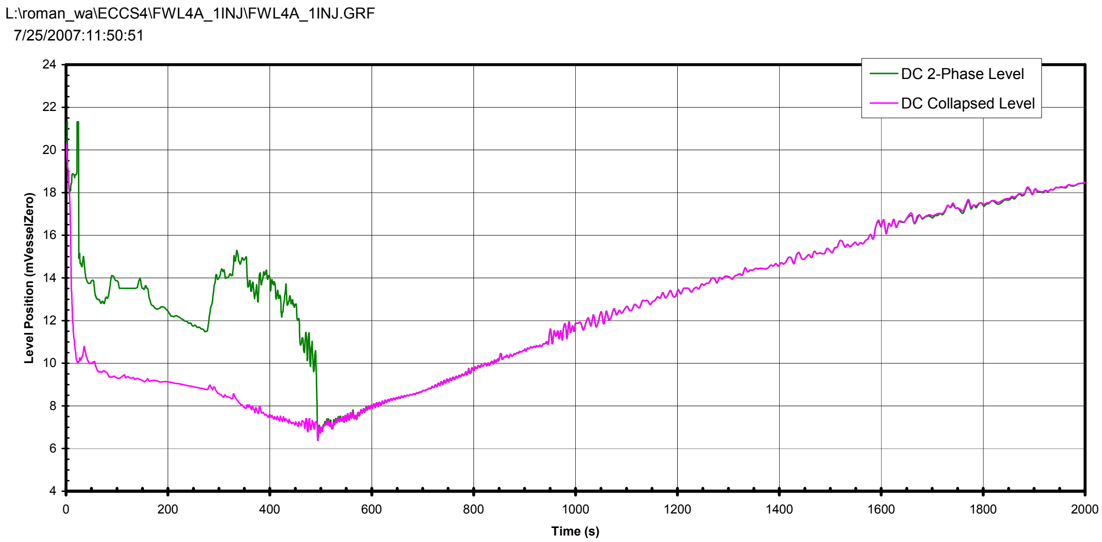


Figure 6.3-9a. Downcomer Water Level, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

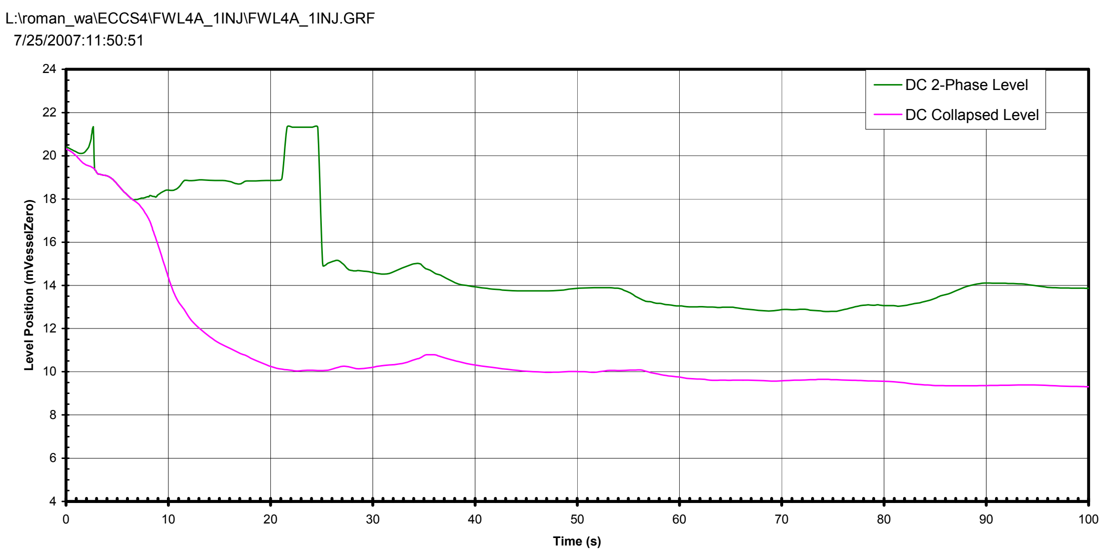


Figure 6.3-9b. Downcomer Water Level, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

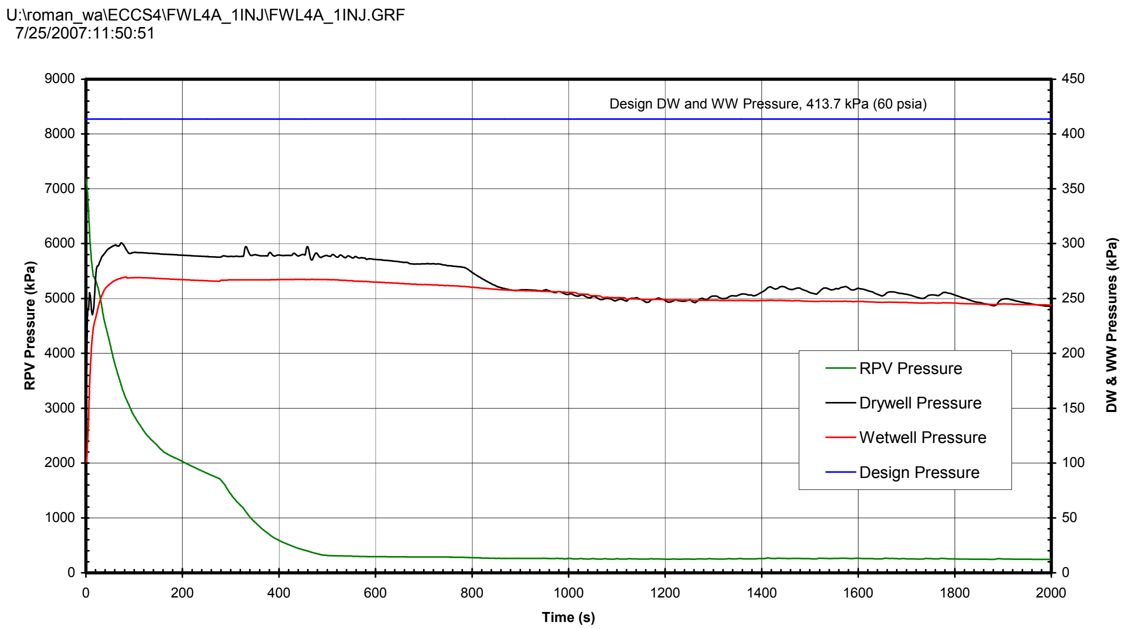


Figure 6.3-10a. System Pressures, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

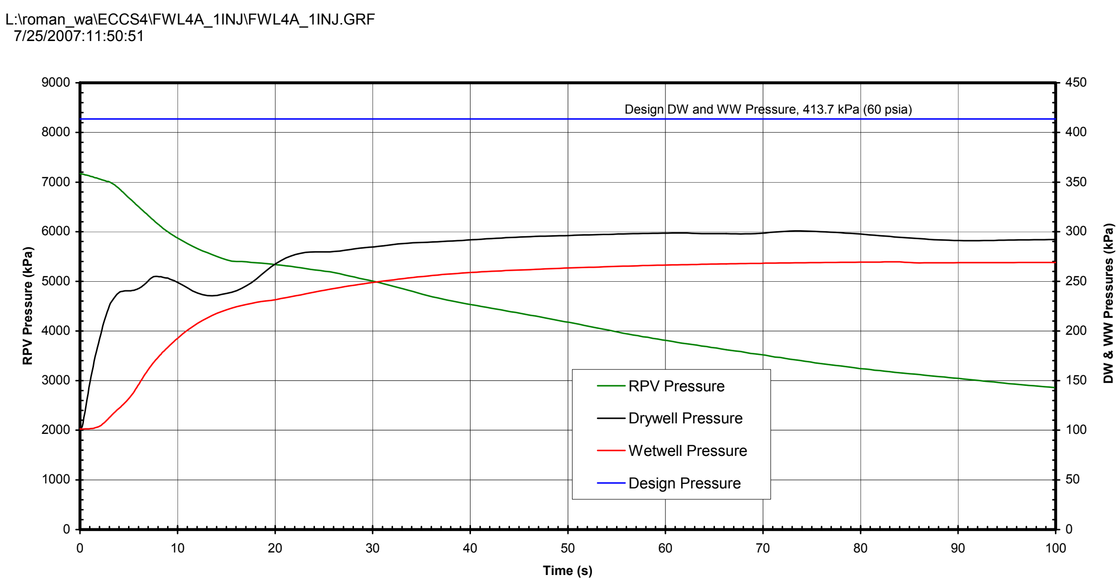


Figure 6.3-10b. System Pressures, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

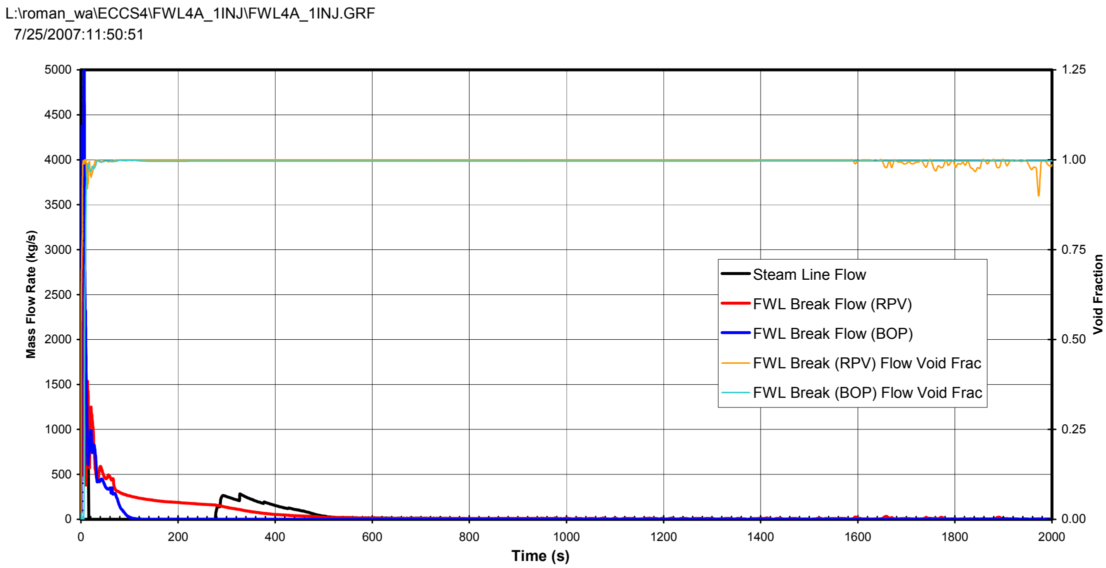


Figure 6.3-11a. Steam Line and Break Flows, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

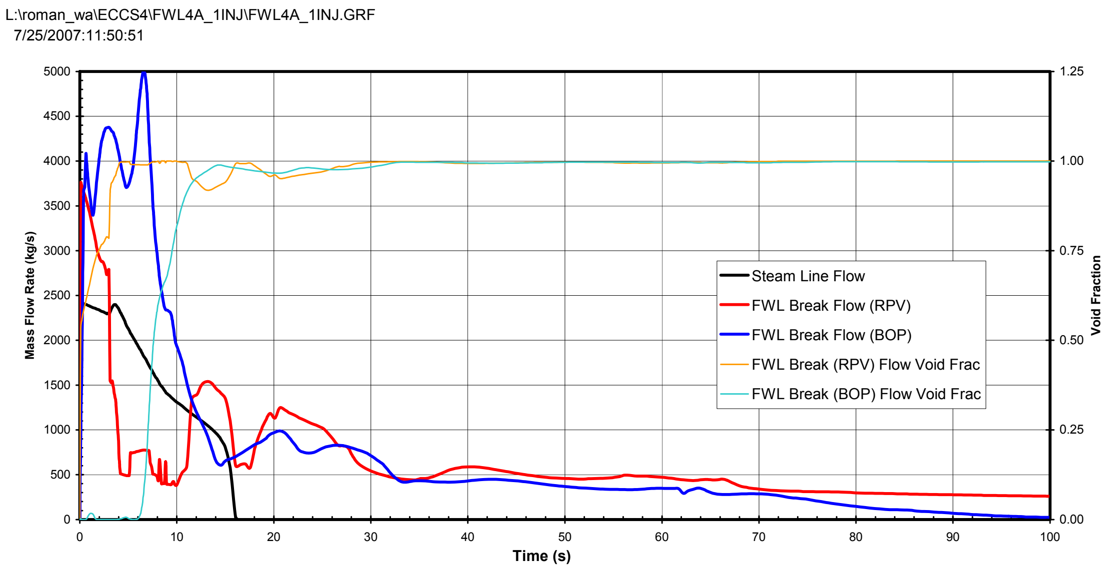


Figure 6.3-11b. Steam Line and Break Flows, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

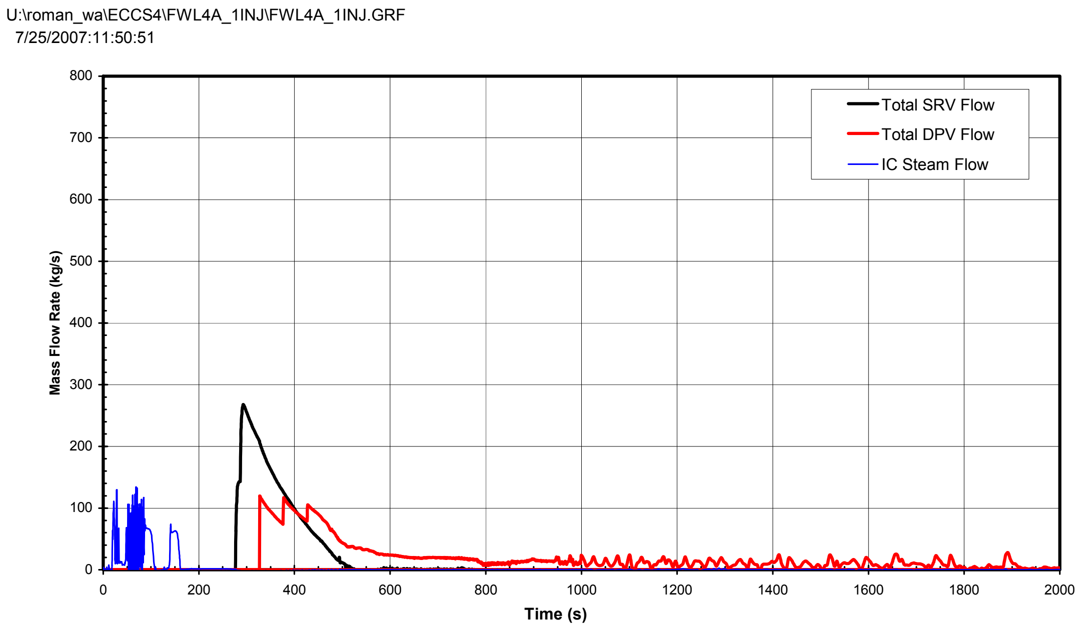


Figure 6.3-12a. ADS Flows, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

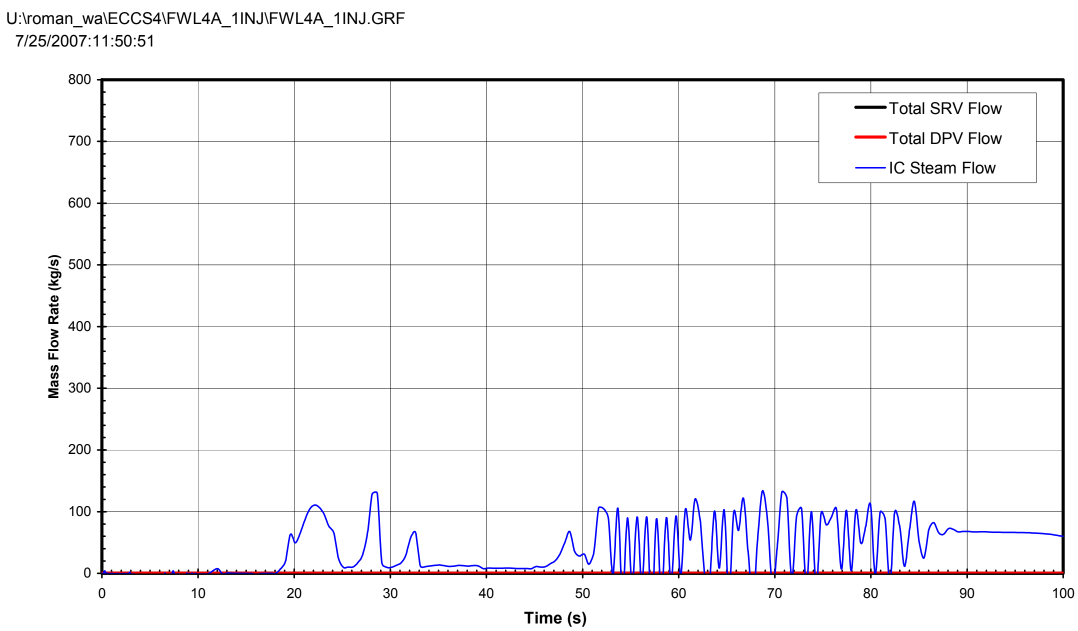


Figure 6.3-12b. ADS Flows, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

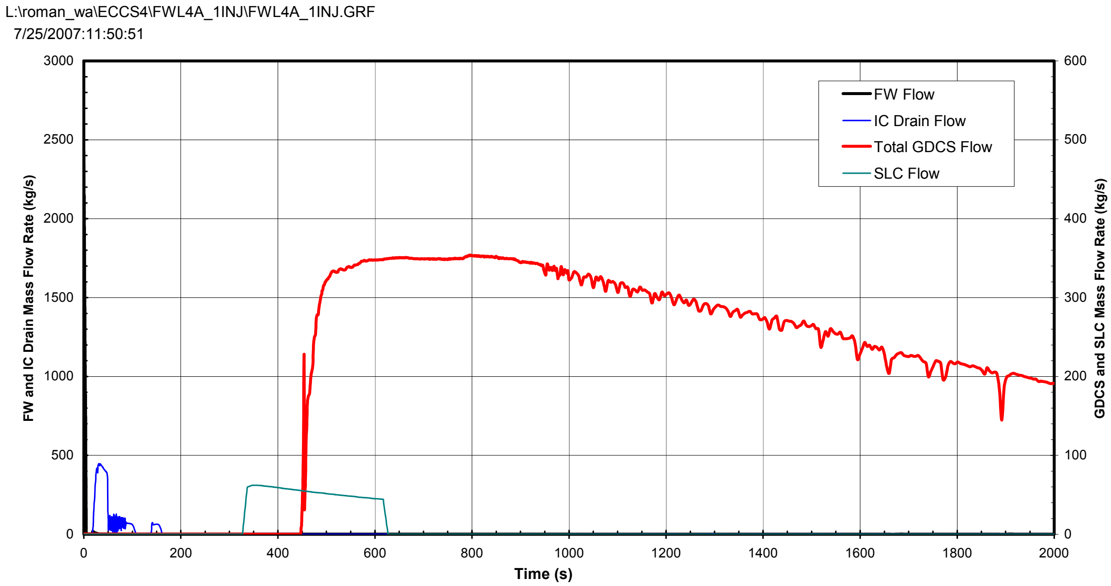


Figure 6.3-13a. Flows Into Vessel, Feedwater Line Break(Nominal Case), 1 GDCS Valve Failure (2000 s)

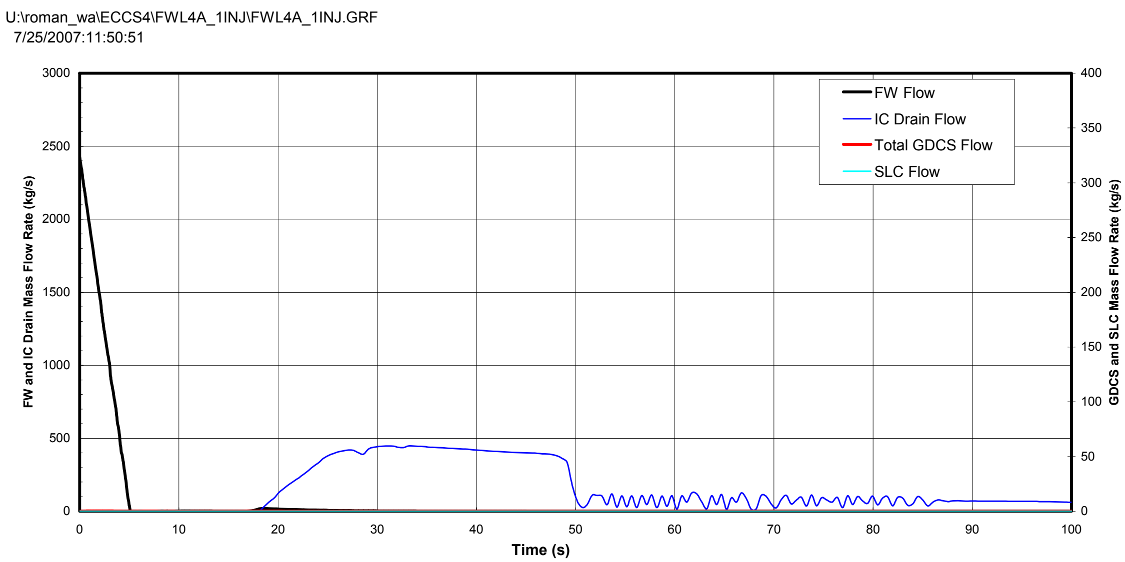


Figure 6.3-13b. Flows Into Vessel, Feedwater Line Break(Nominal Case), 1 GDCS Valve Failure (100 s)

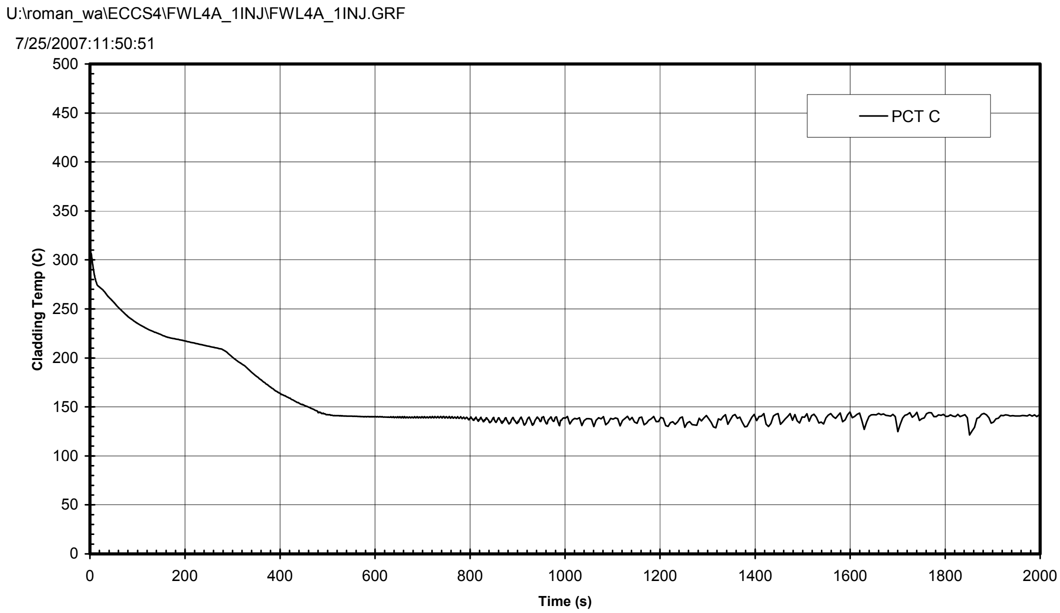


Figure 6.3-14a. PCT, Feedwater Line Break(Nominal Case), 1 GDCS Valve Failure (2000 s)

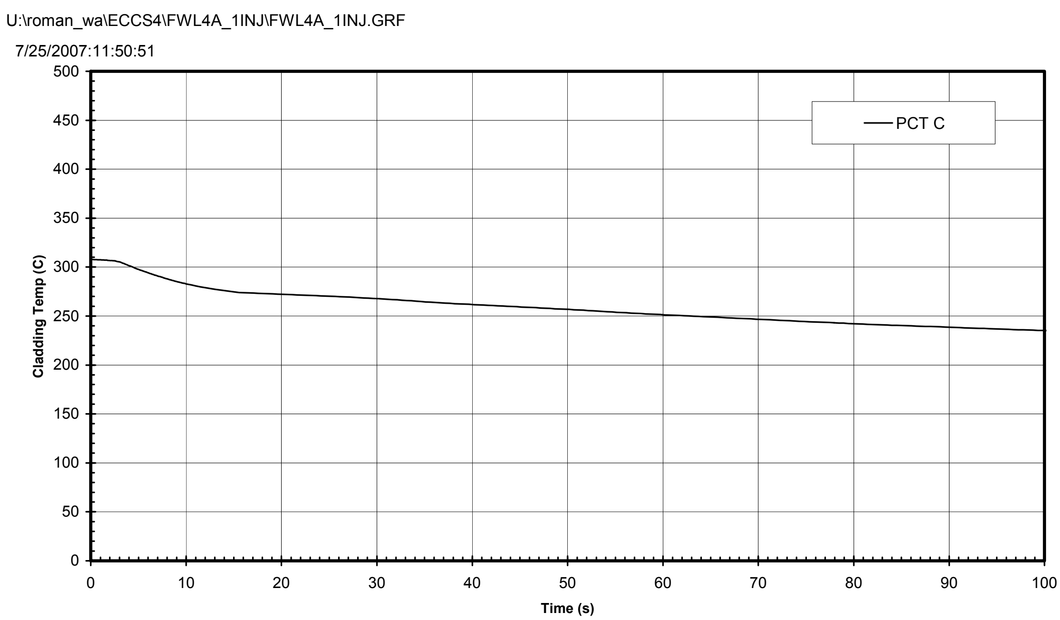


Figure 6.3-14b. PCT, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

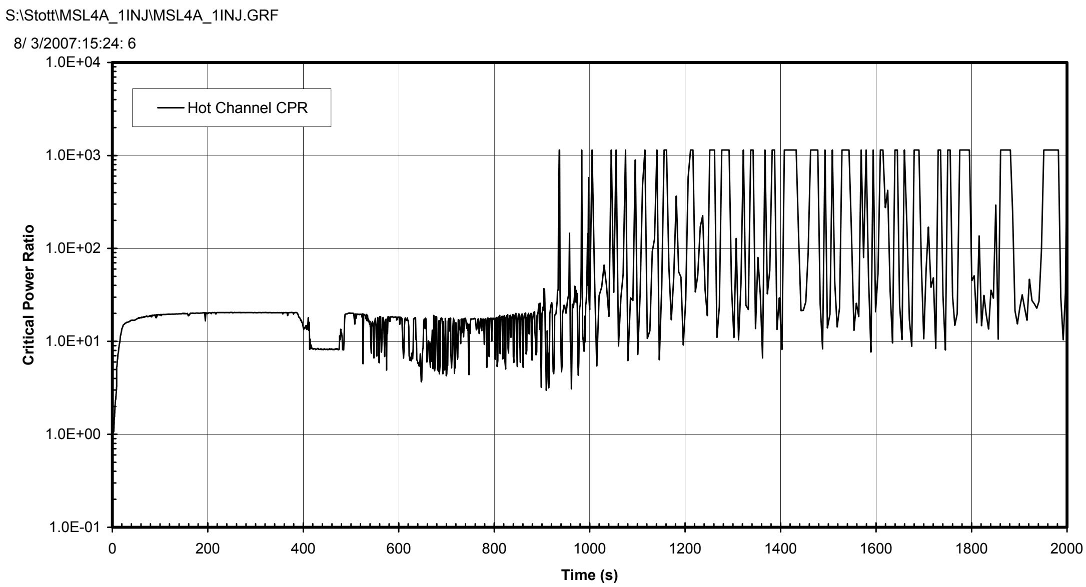


Figure 6.3-15a. MCPR, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

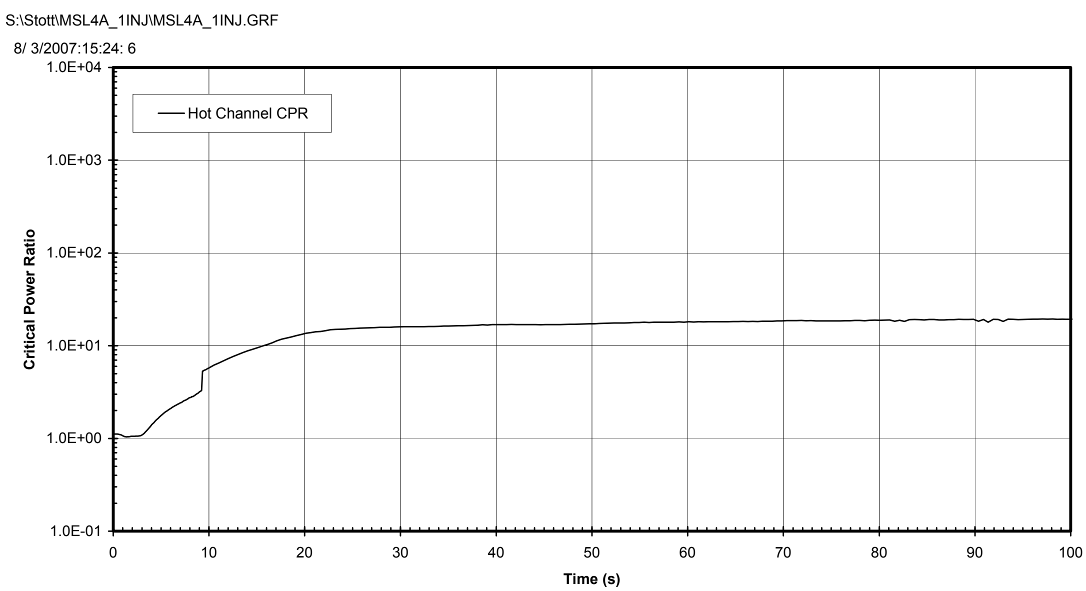


Figure 6.3-15b. MCPR, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

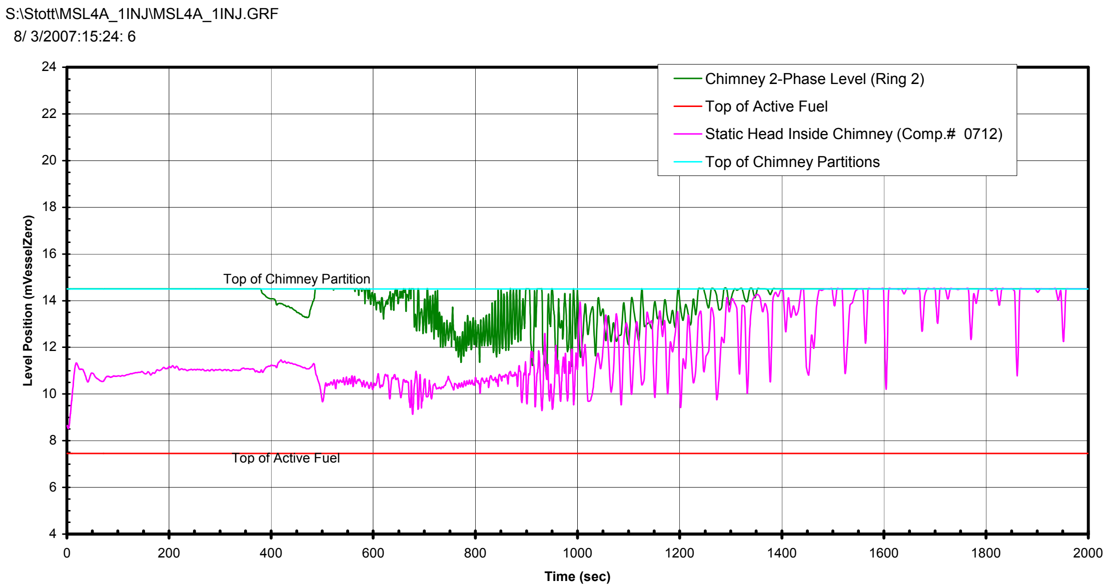


Figure 6.3-16a. Chimney Water Level, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

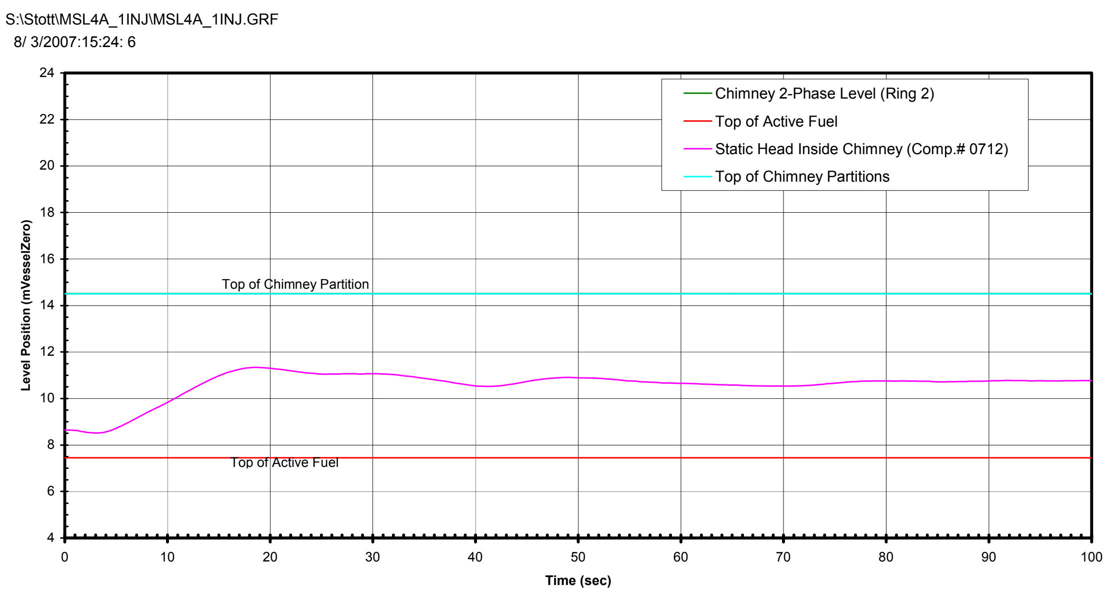


Figure 6.3-16b. Chimney Water Level, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

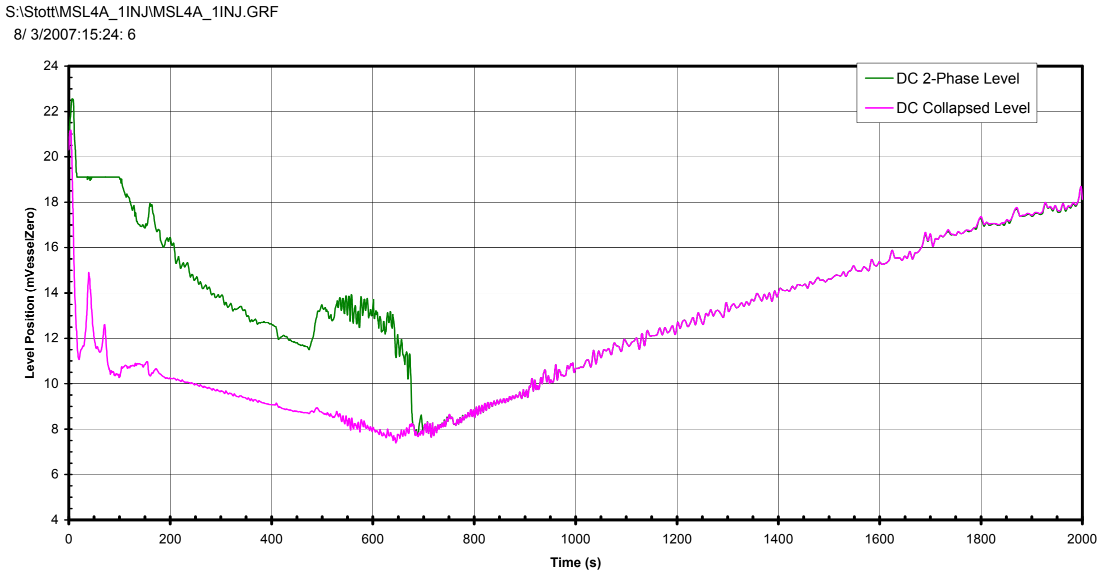


Figure 6.3-17a. Downcomer Water Level, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

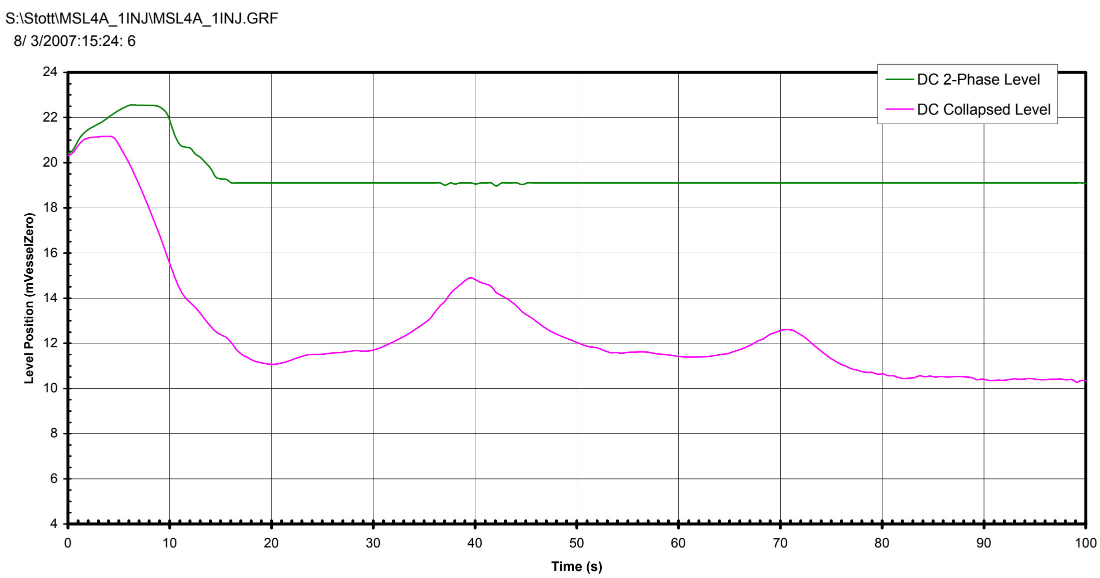


Figure 6.3-17b. Downcomer Water Level, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

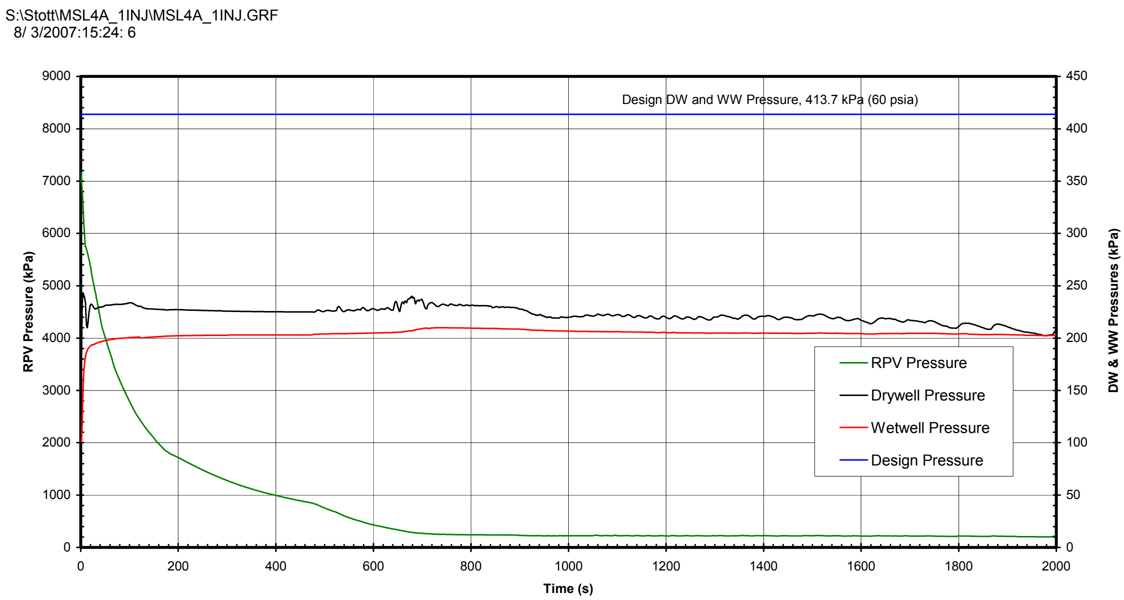


Figure 6.3-18a. System Pressures, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

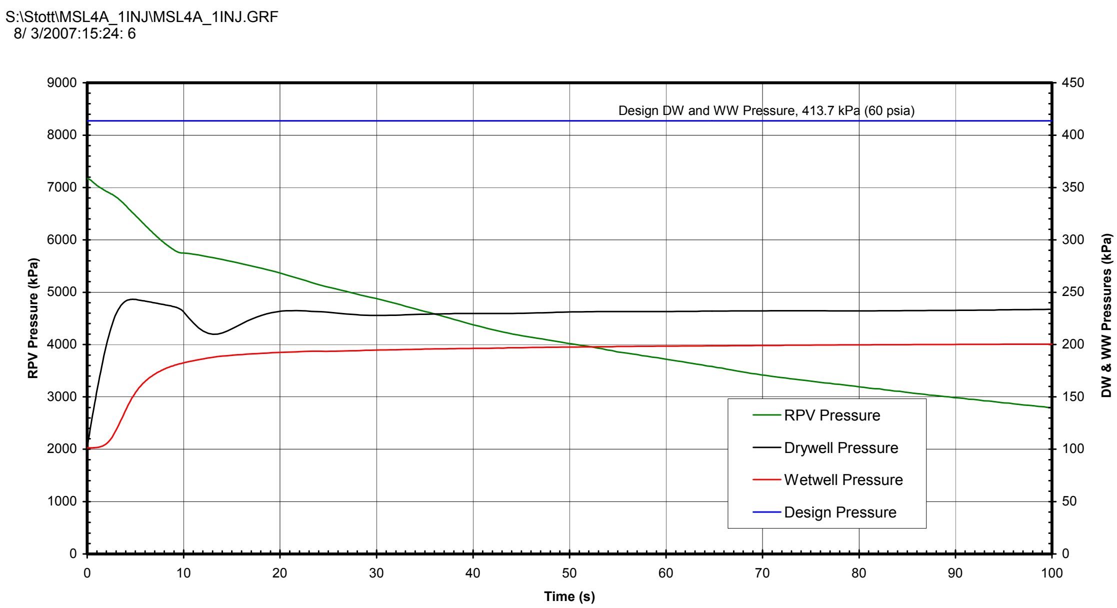


Figure 6.3-18b. System Pressures, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

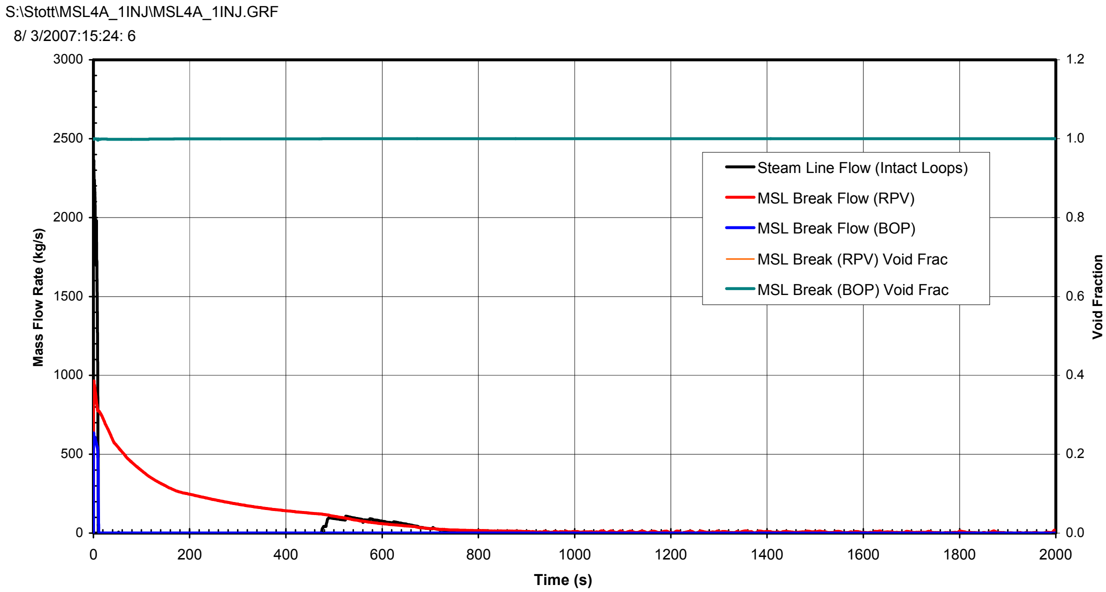


Figure 6.3-19a. Steam Line and Break Flows, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

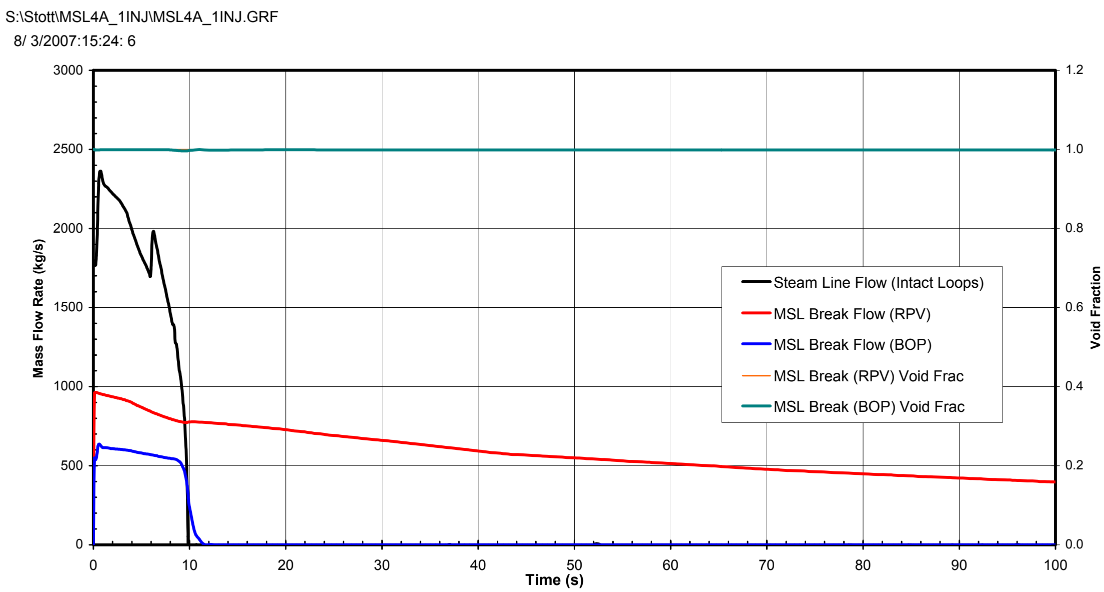


Figure 6.3-19b. Steam Line and Break Flows, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

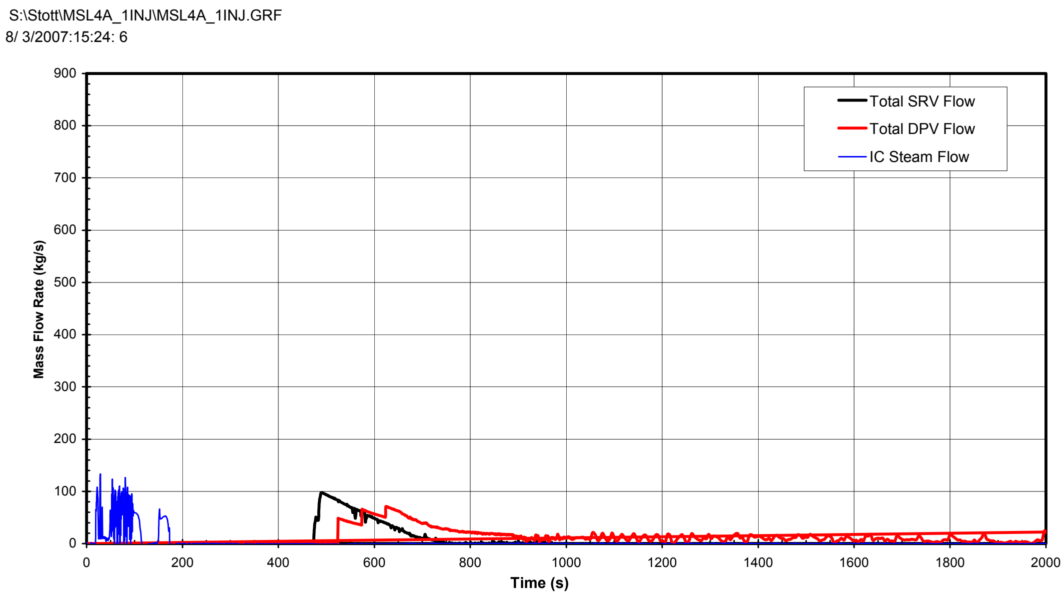


Figure 6.3-20a. ADS Flows, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

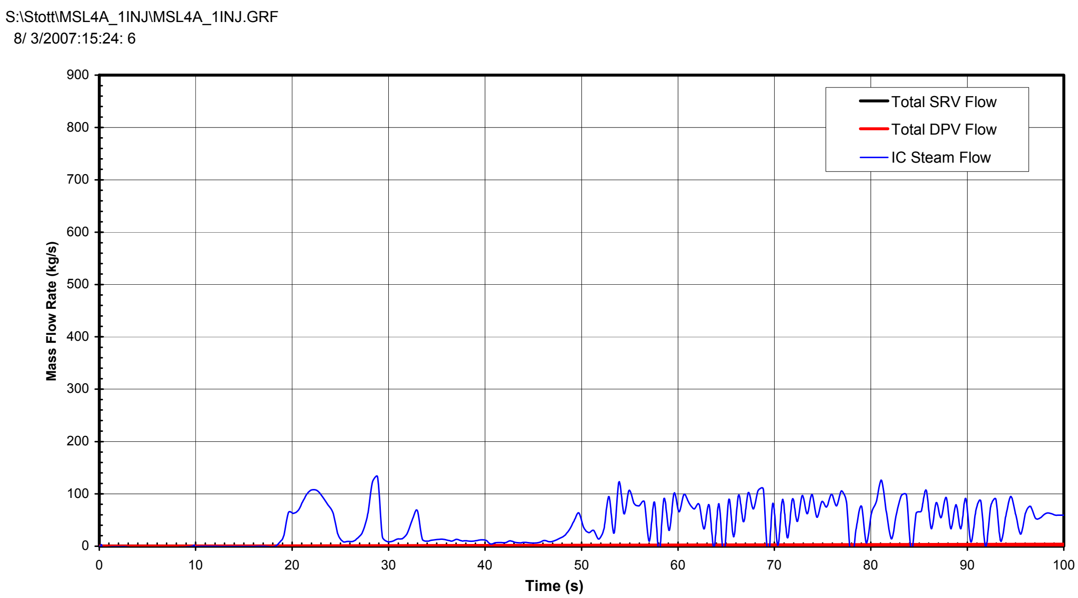


Figure 6.3-20b. ADS Flows, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

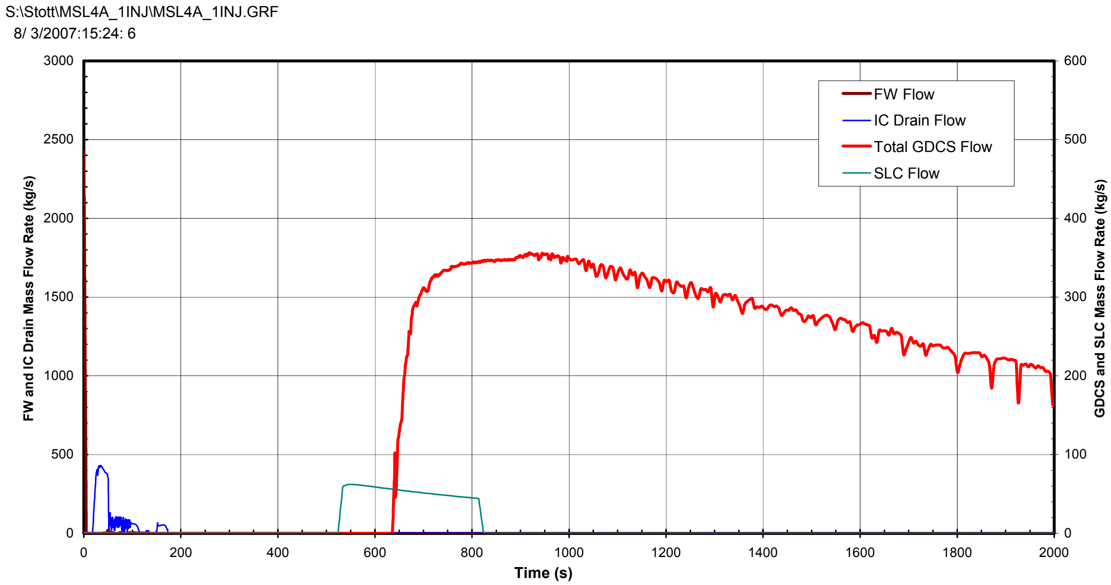


Figure 6.3-21a. Flows Into Vessel, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

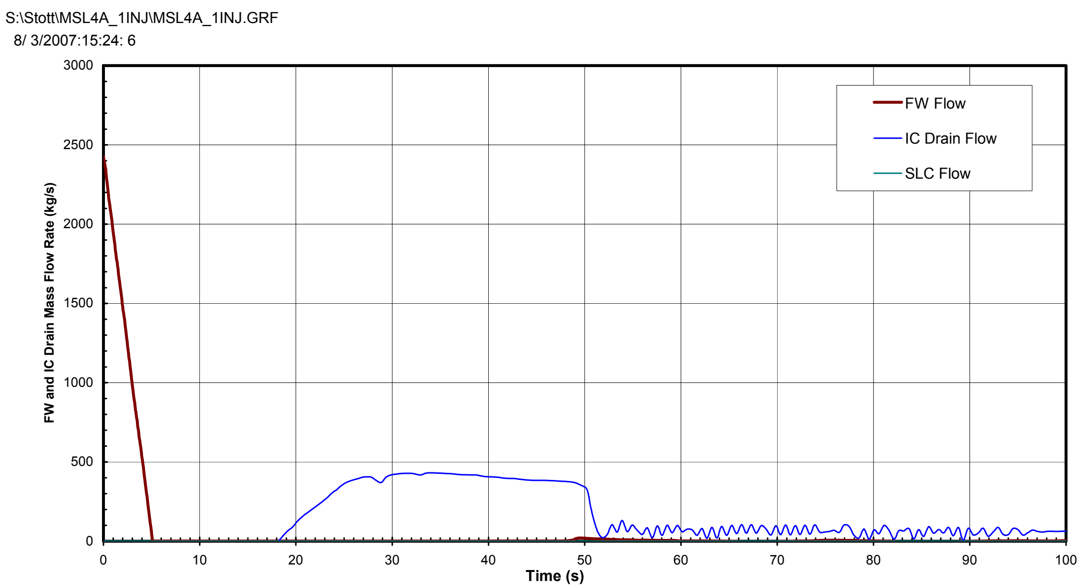


Figure 6.3-21b. Flows Into Vessel, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

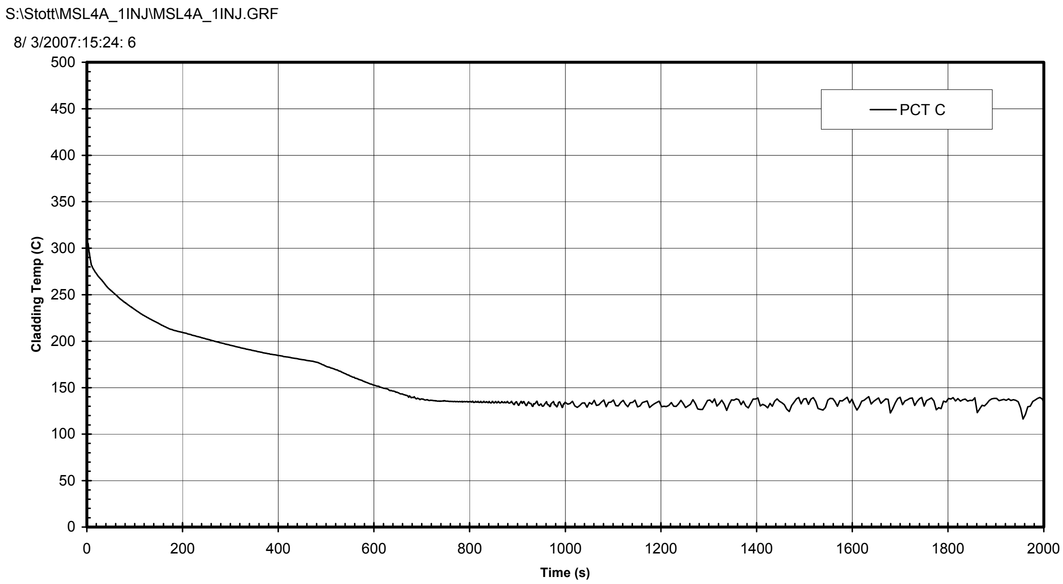


Figure 6.3-22a. PCT, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

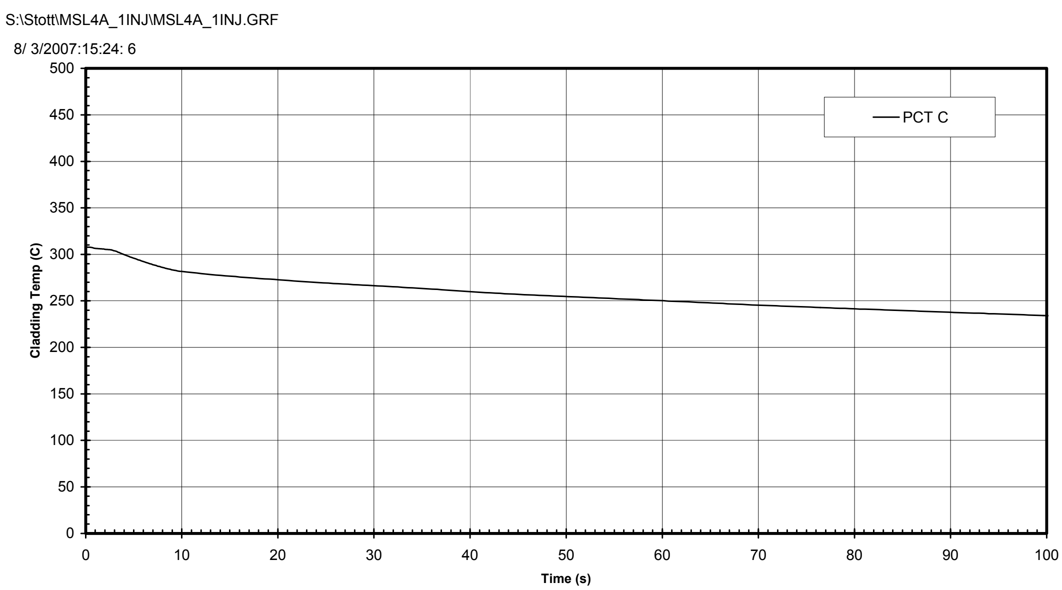


Figure 6.3-22b. PCT, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

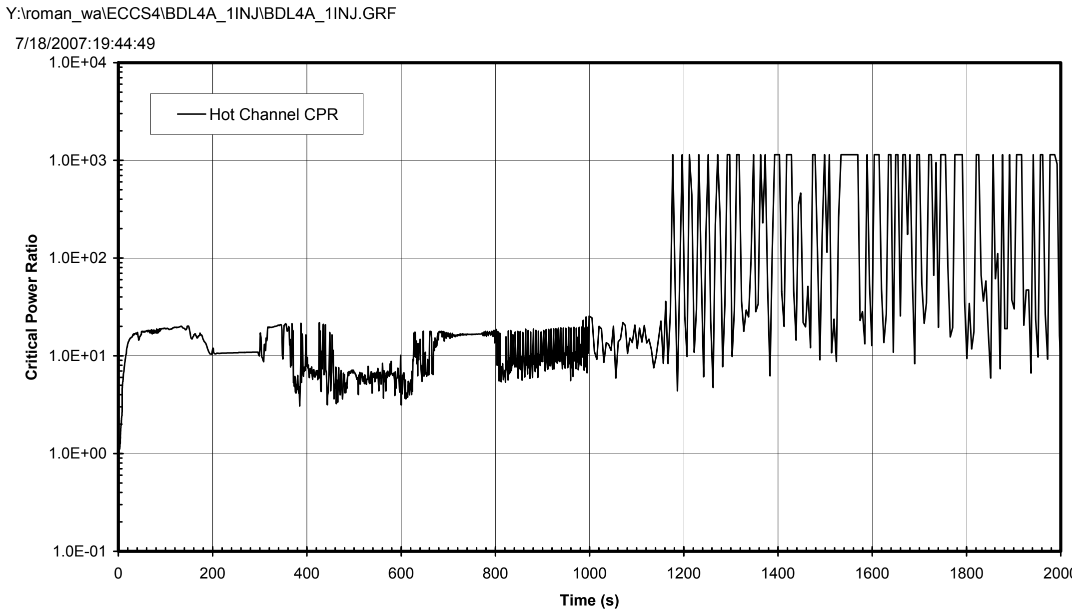


Figure 6.3-23a. MCPR, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

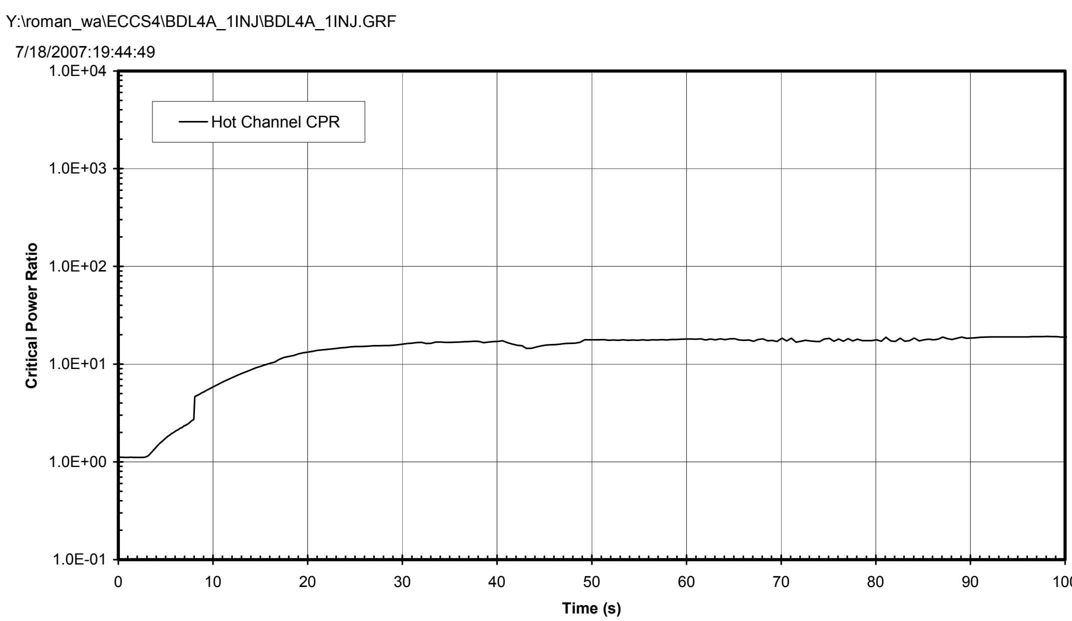


Figure 6.3.3-23b. MCPR, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

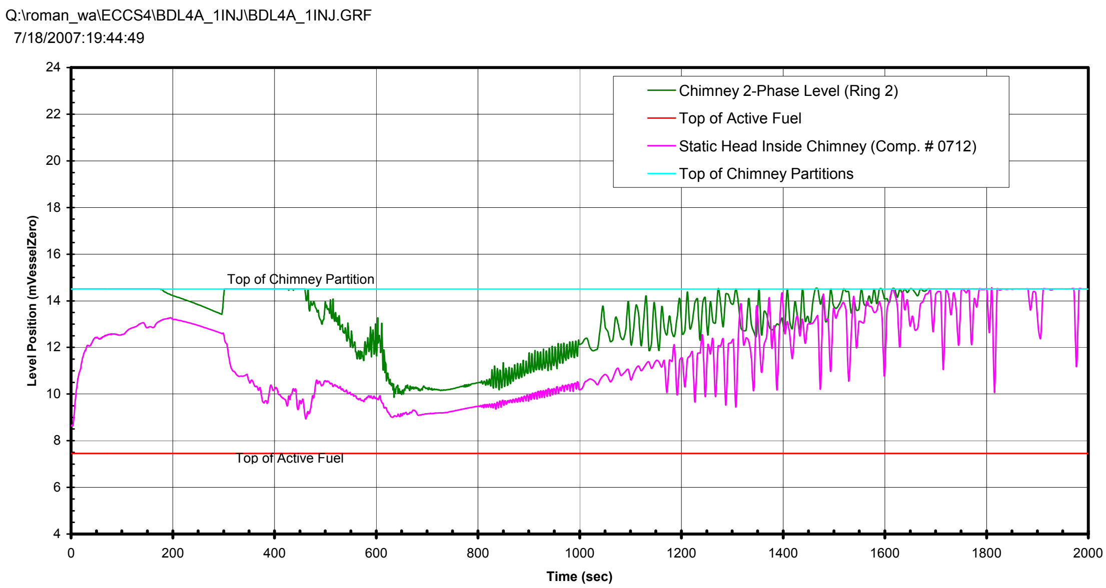


Figure 6.3-24a. Chimney Water Level, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

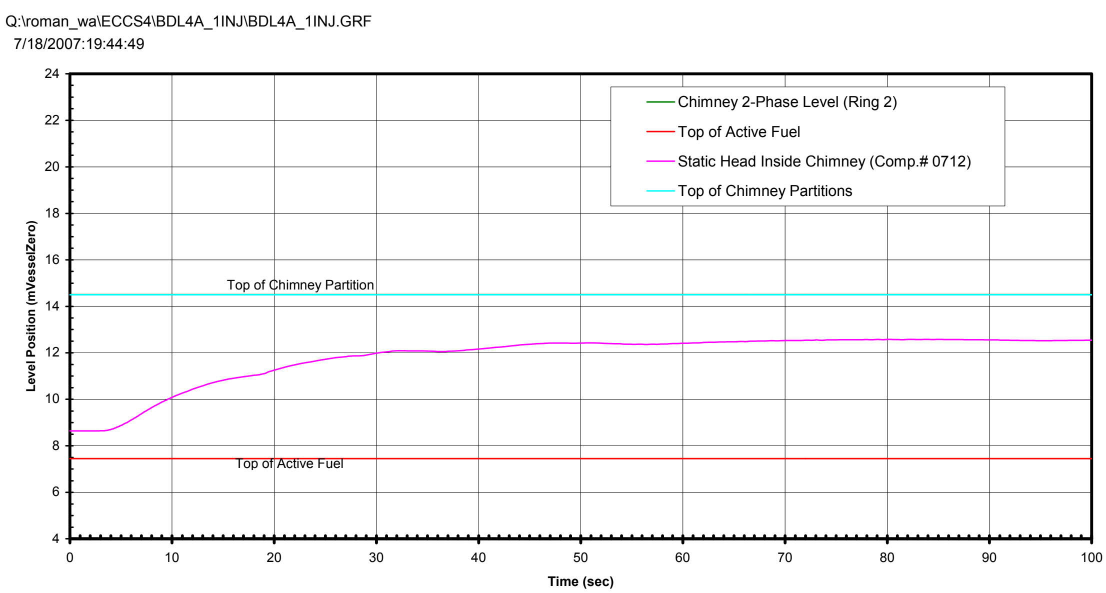


Figure 6.3-24b. Chimney Water Level, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

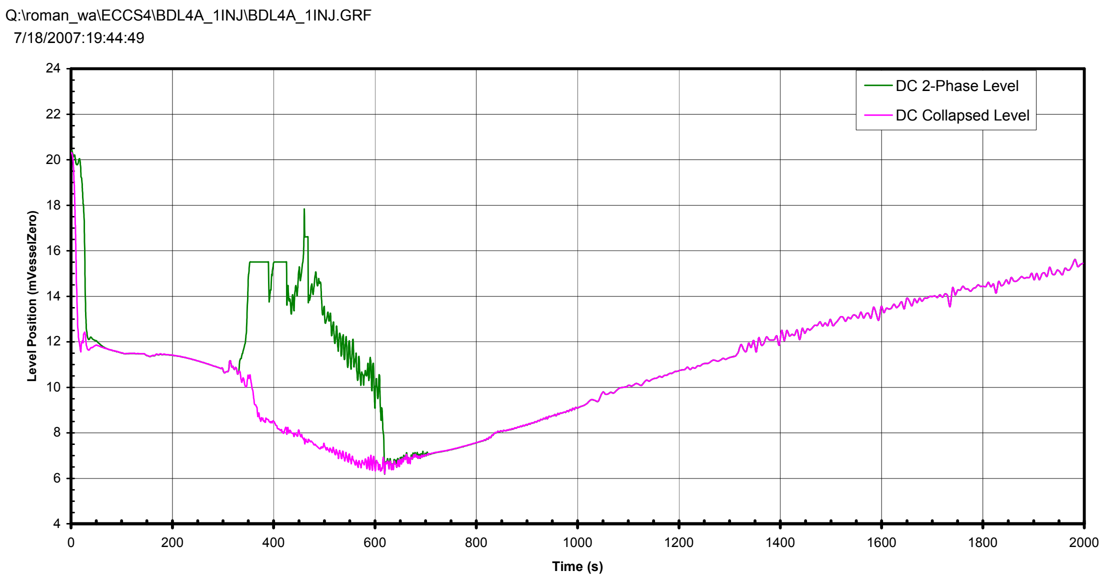


Figure 6.3-25a. Downcomer Water Level, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

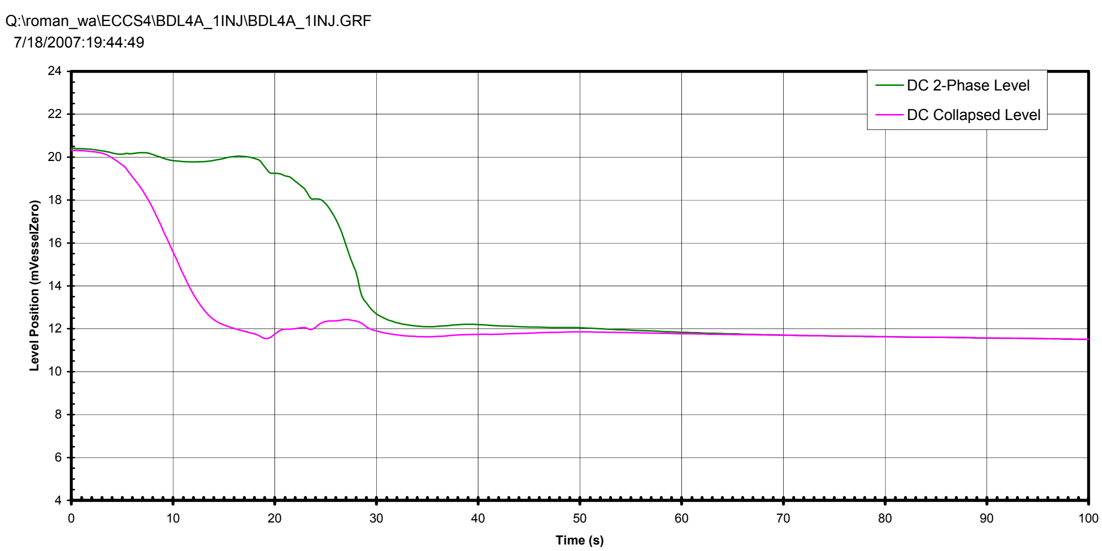


Figure 6.3-25b. Downcomer Water Level, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

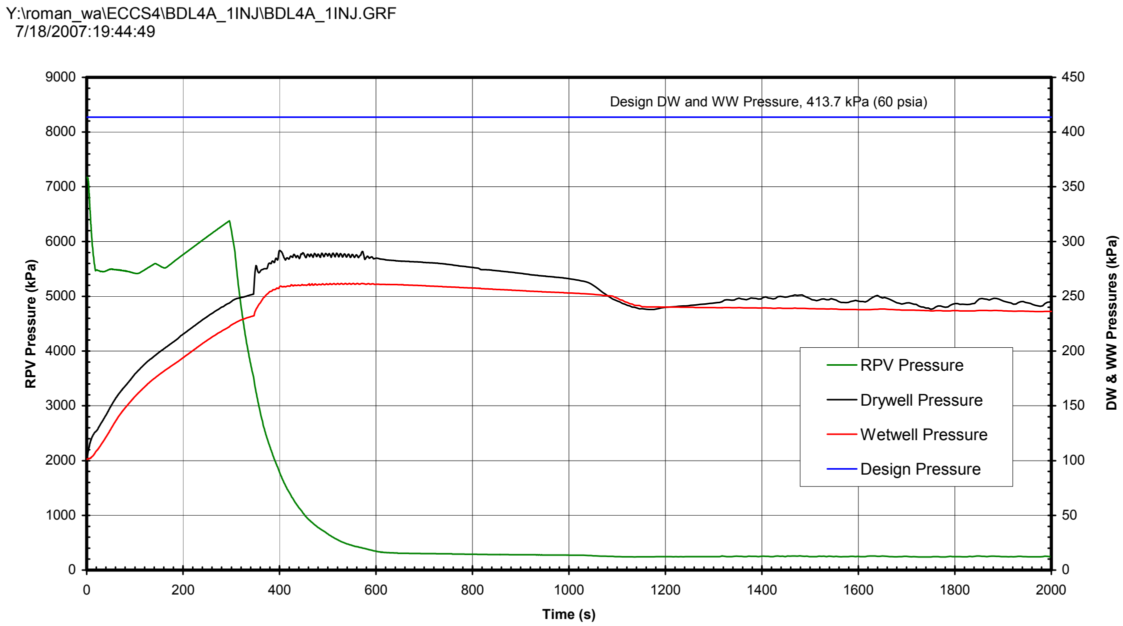


Figure 6.3-26a. System Pressures, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

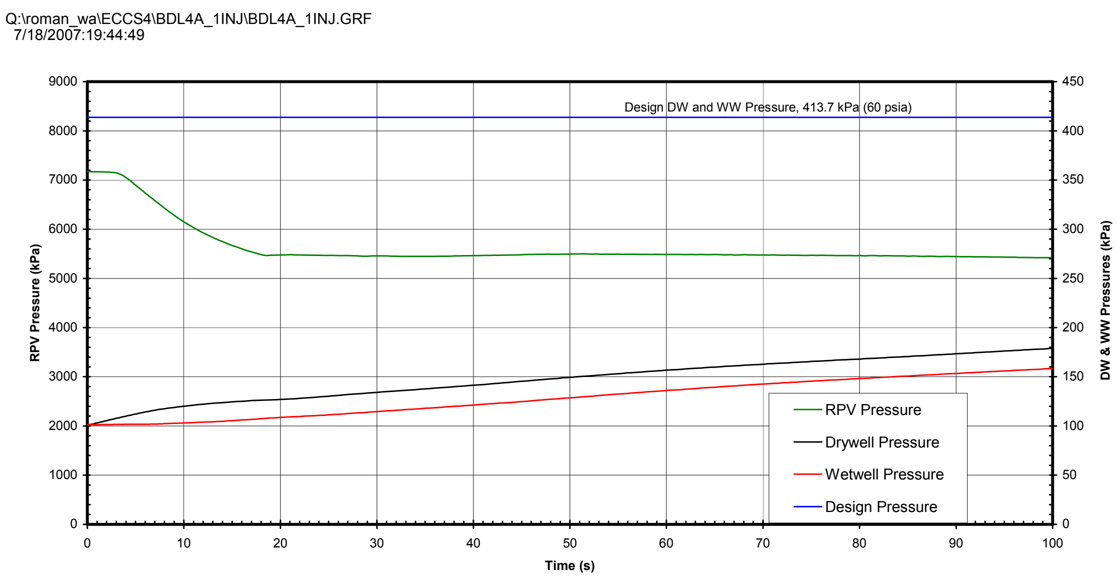


Figure 6.3-26b. System Pressures, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

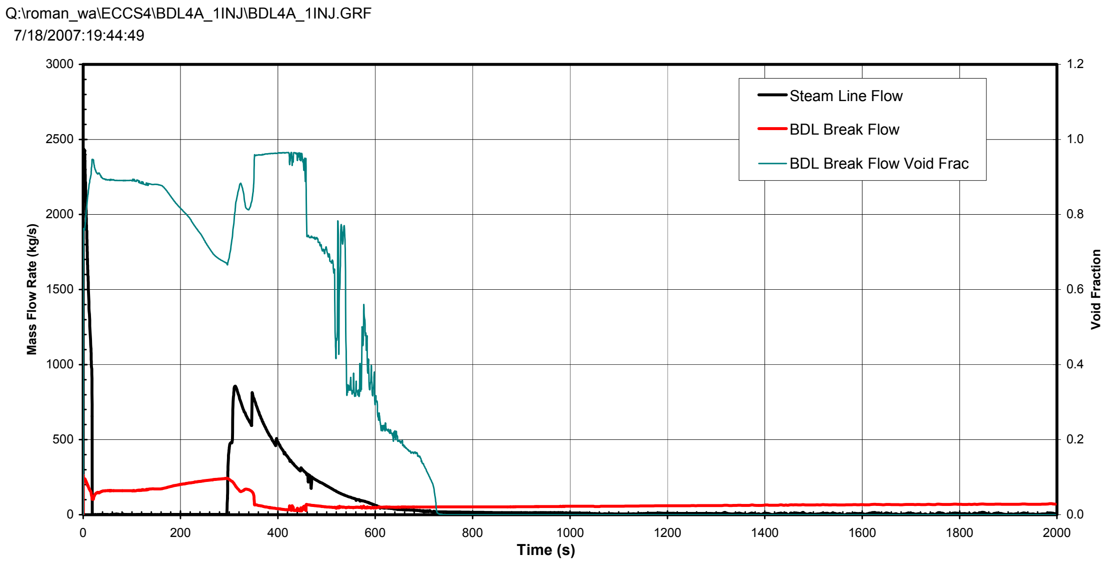


Figure 6.3-27a. Steam Line and Break Flows, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

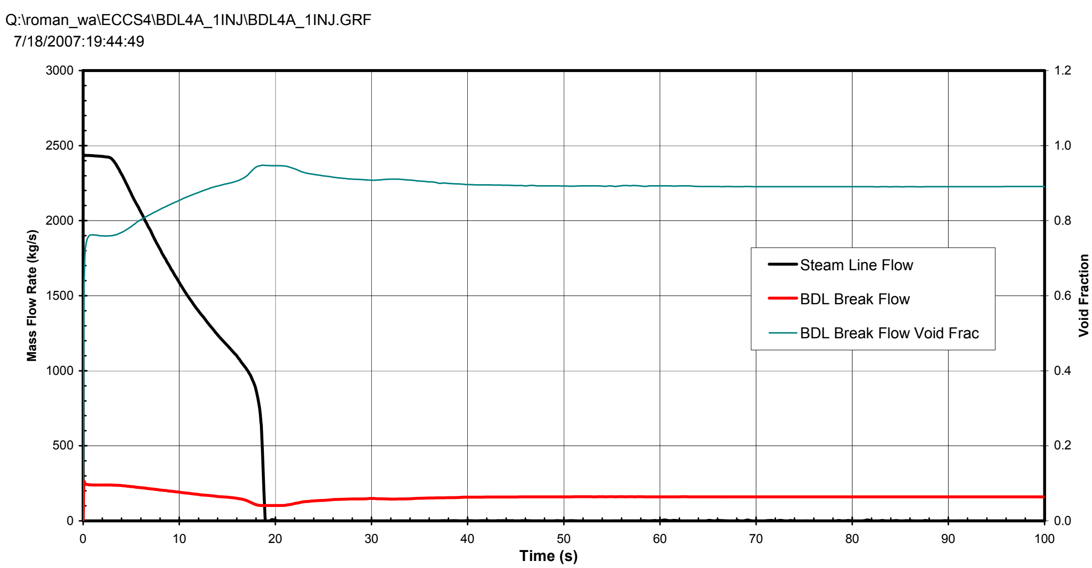


Figure 6.3-27b. Steam Line and Break Flows, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

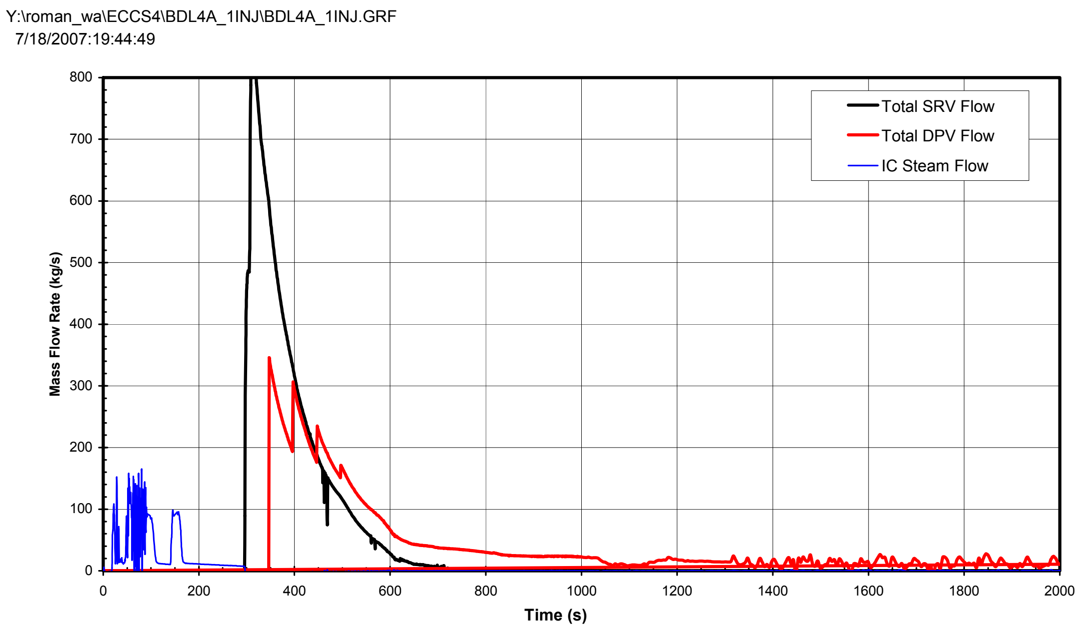


Figure 6.3-28a. ADS Flows, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

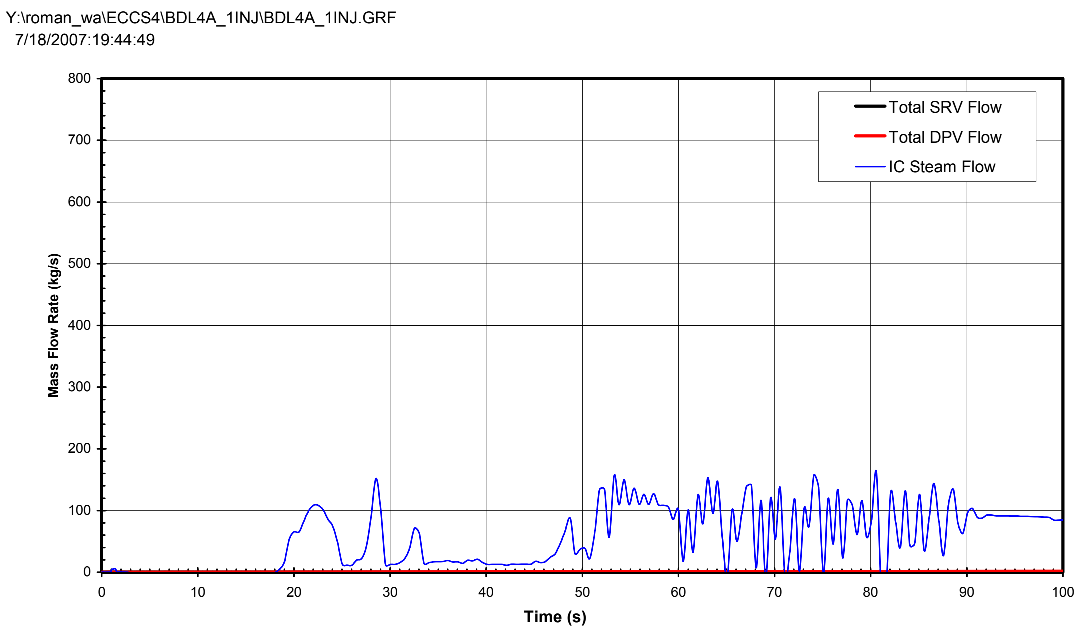


Figure 6.3-28b. ADS Flows, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

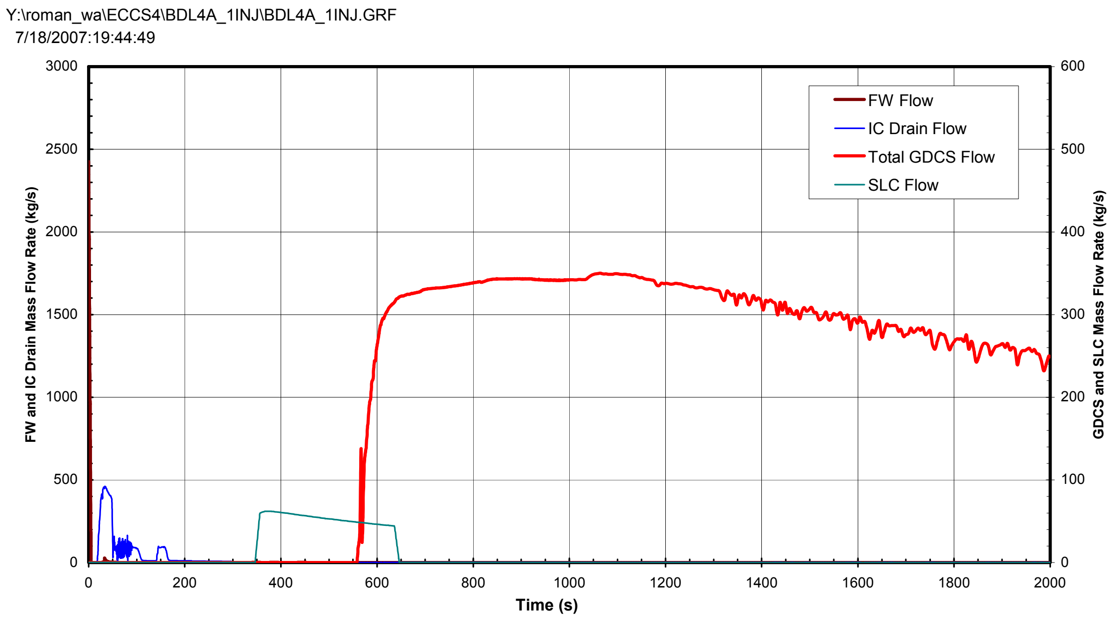


Figure 6.3-29a. Flows Into Vessel, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

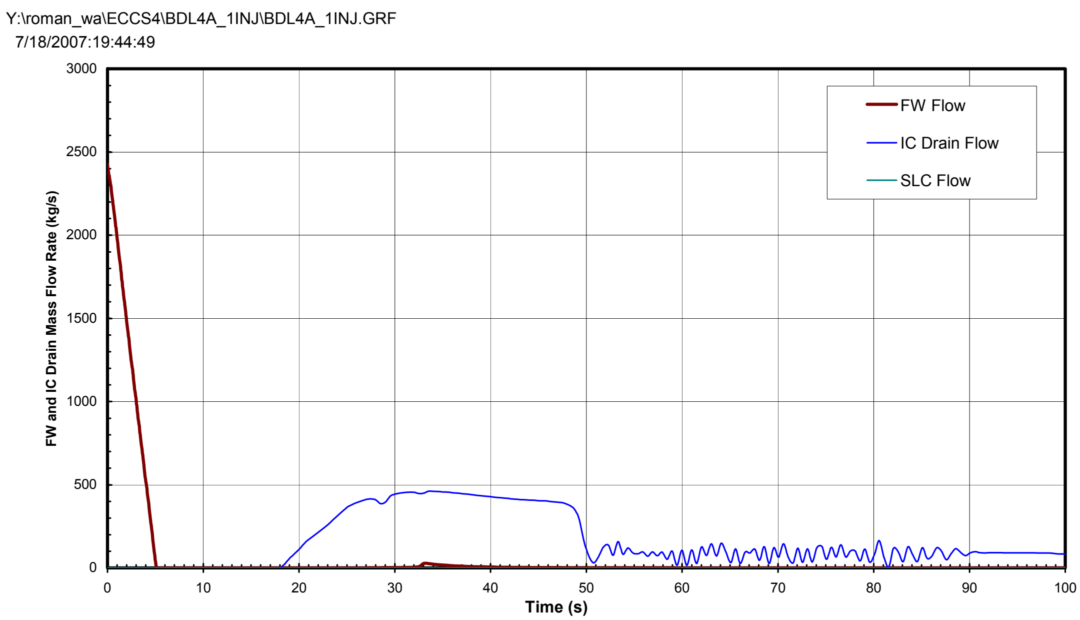


Figure 6.3-29b. Flows Into Vessel, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

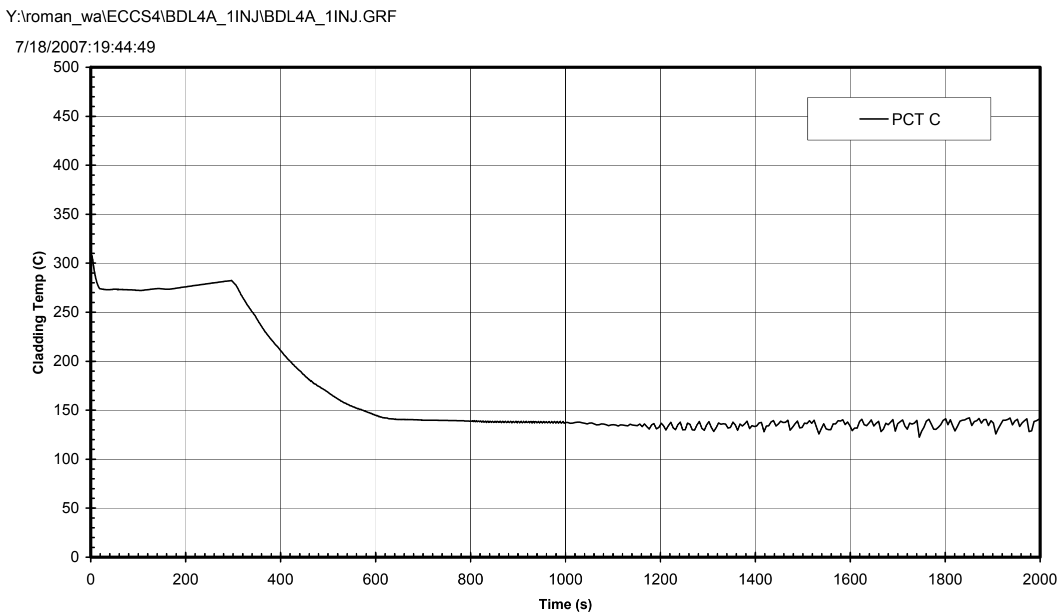


Figure 6.3-30a. PCT, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

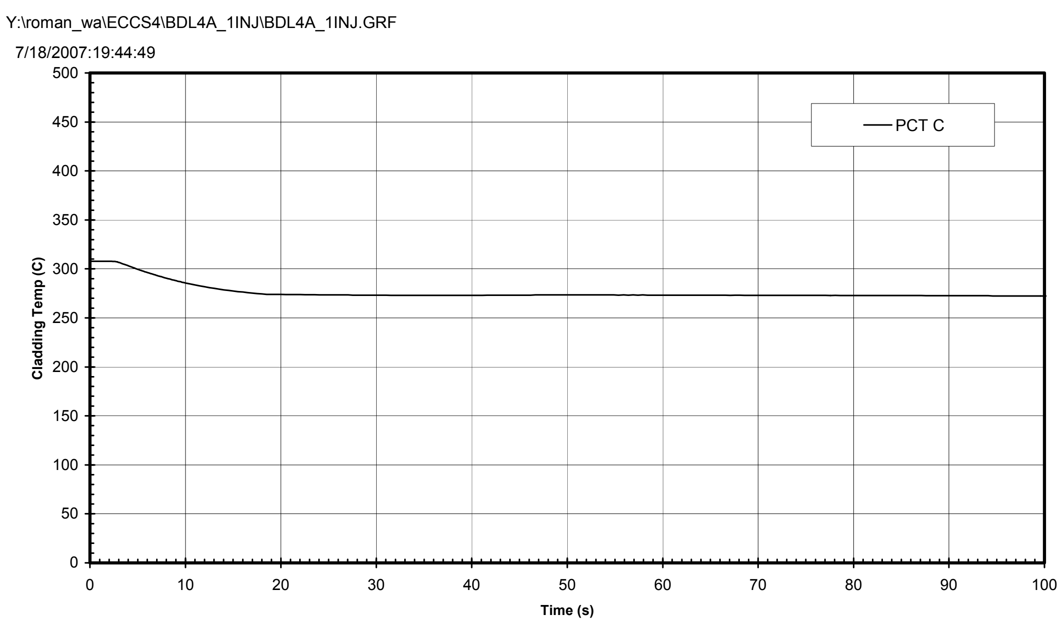


Figure 6.3-30b. PCT, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

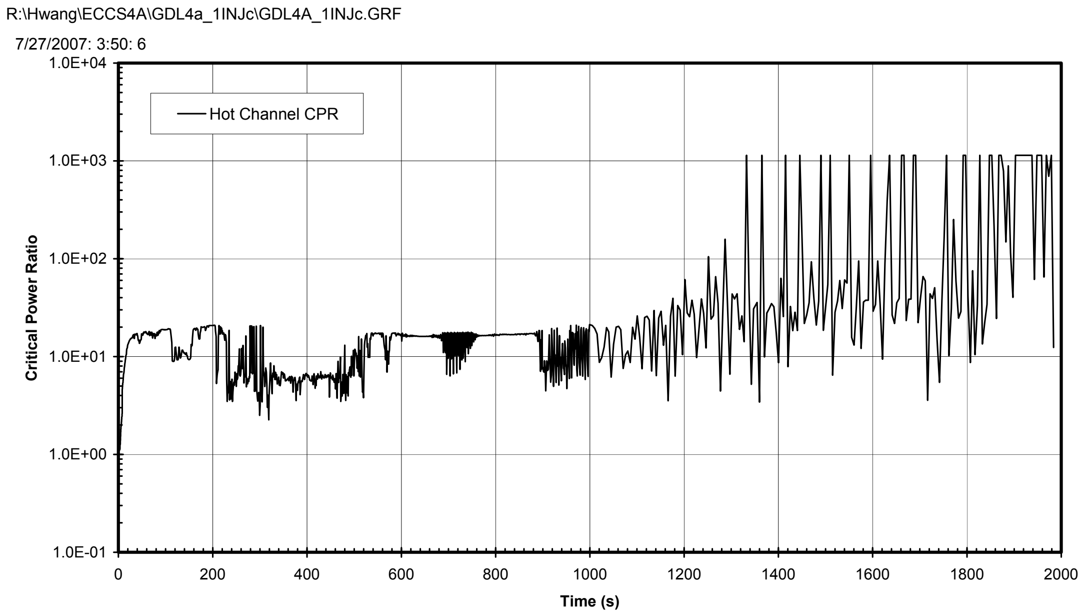


Figure 6.3-31a. MCPR, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

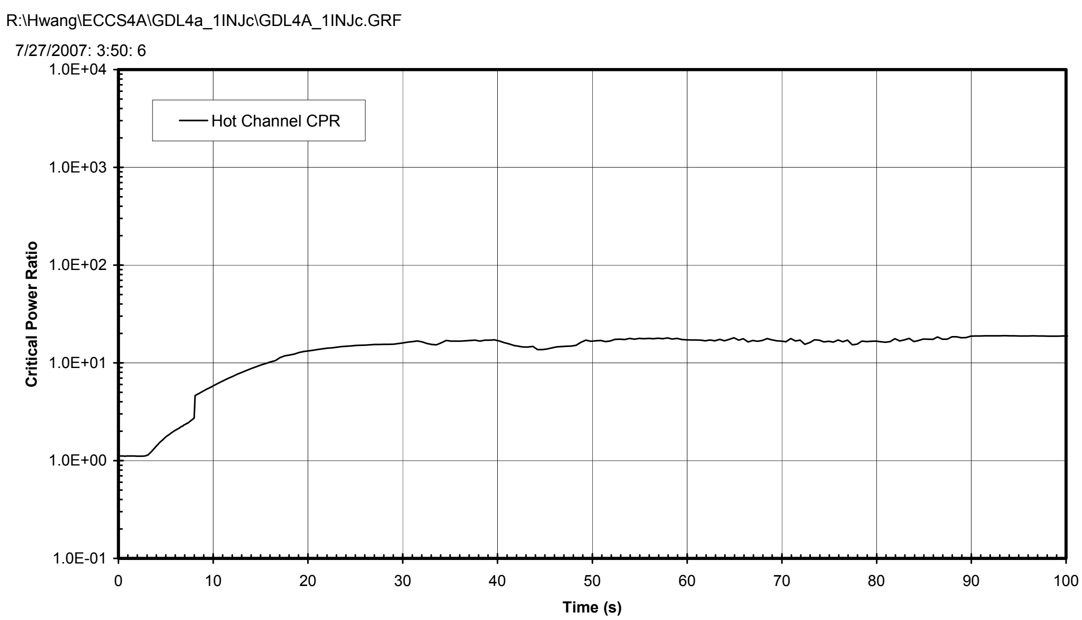


Figure 6.3-31b. MCPR, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

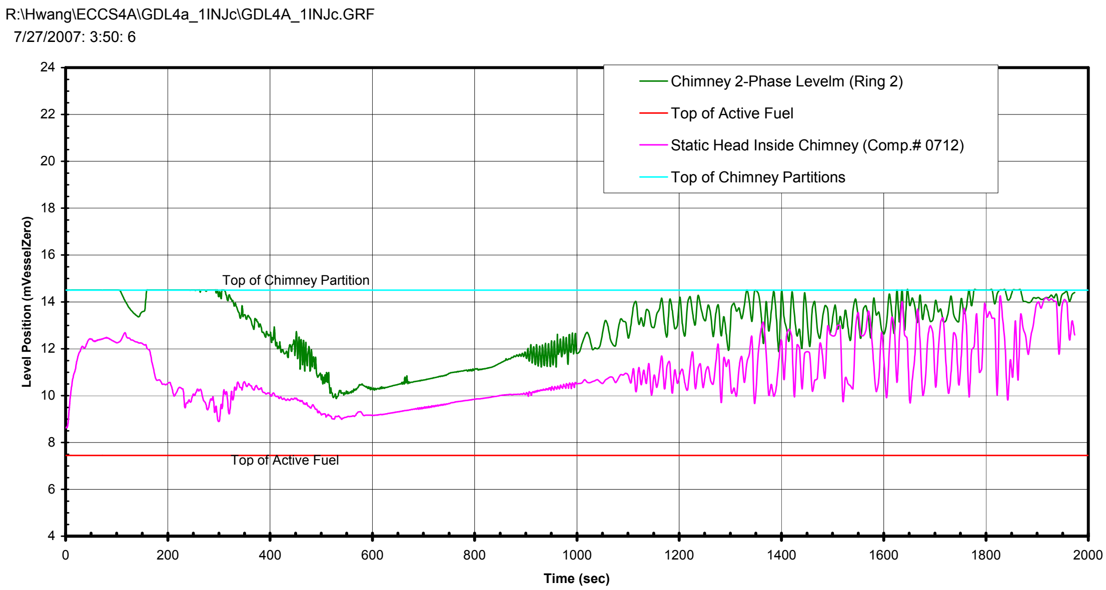


Figure 6.3-32a. Chimney Water Level, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

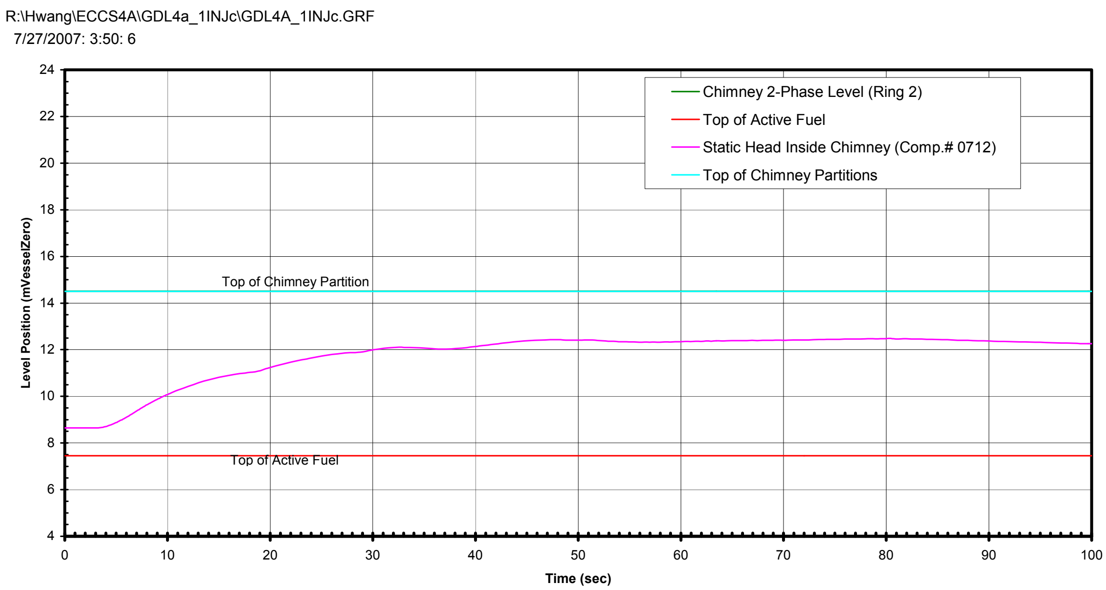


Figure 6.3-32b. Chimney Water Level, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

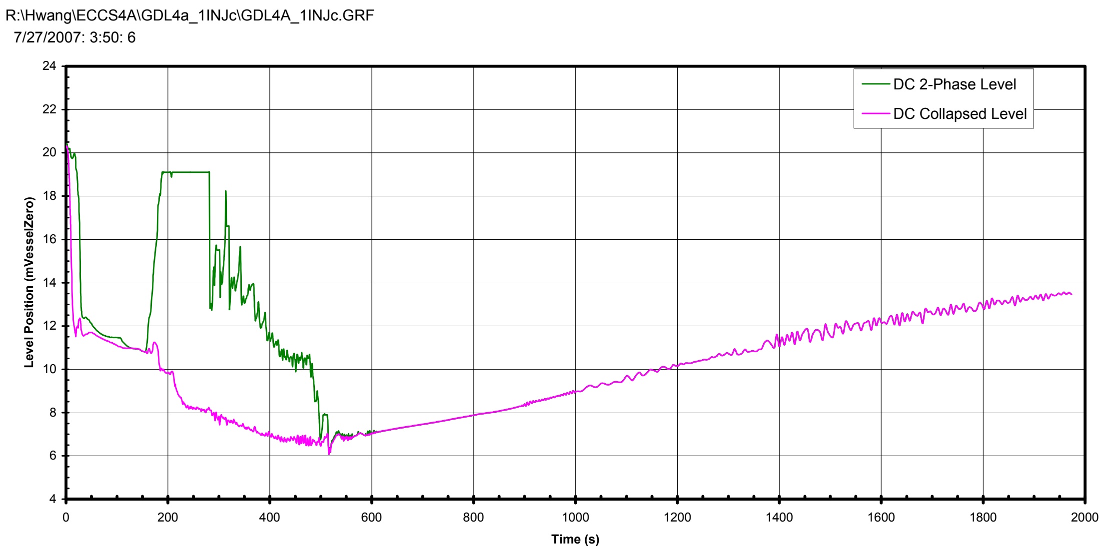


Figure 6.3-33a. Downcomer Water Level, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

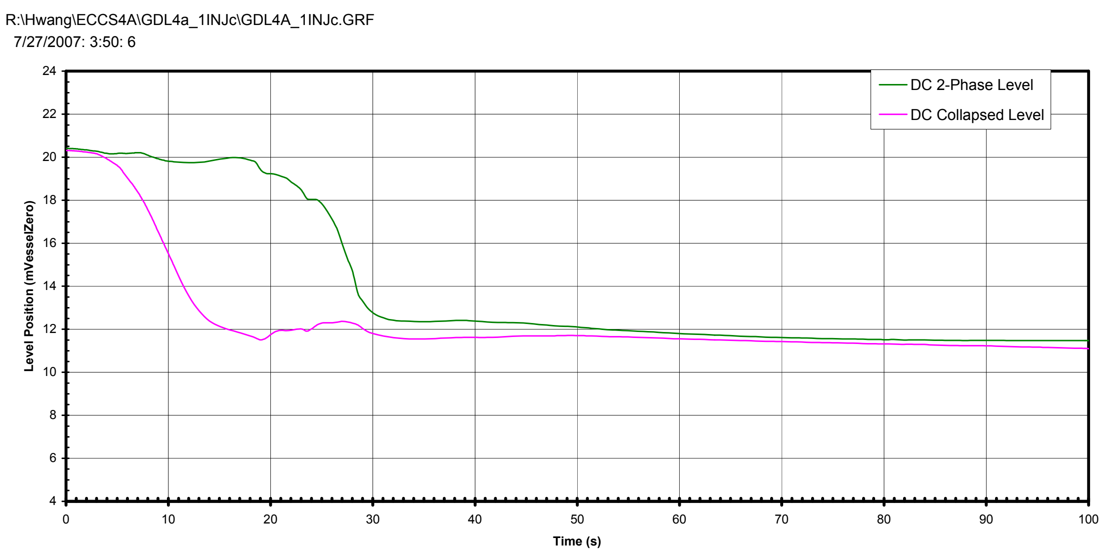


Figure 6.3-33b. Downcomer Water Level, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

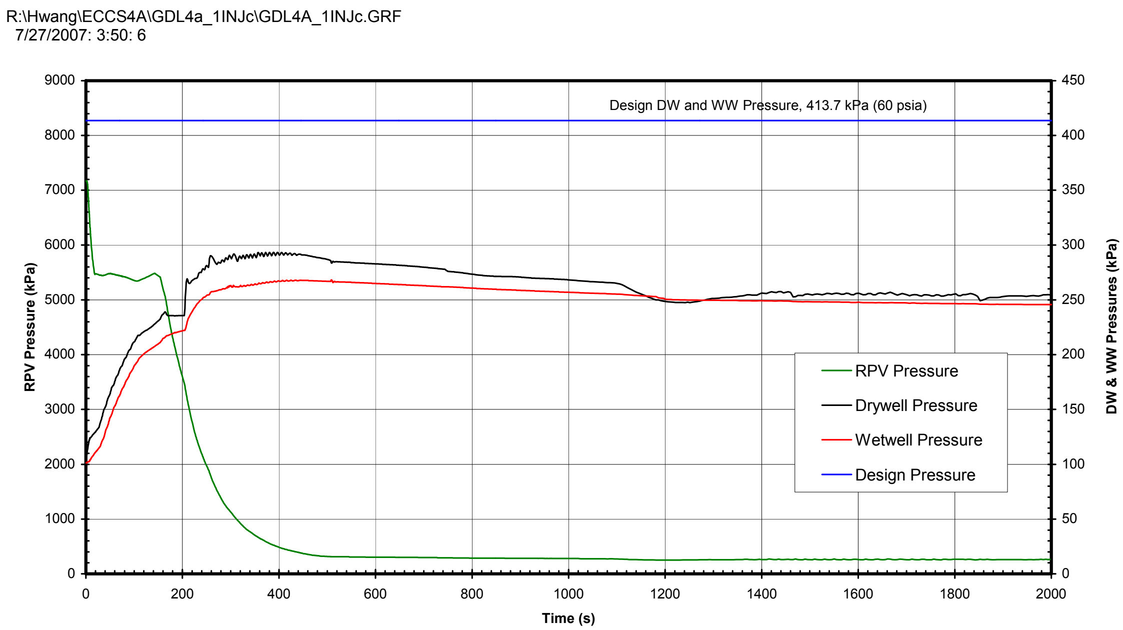


Figure 6.3-34a. System Pressures, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

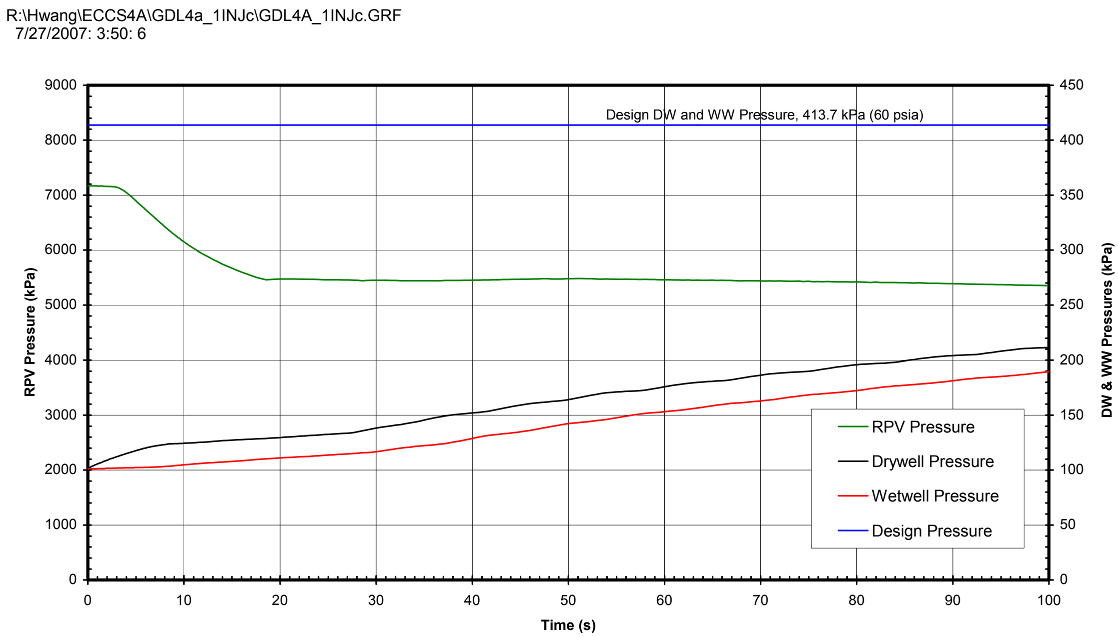


Figure 6.3-34b. System Pressures, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

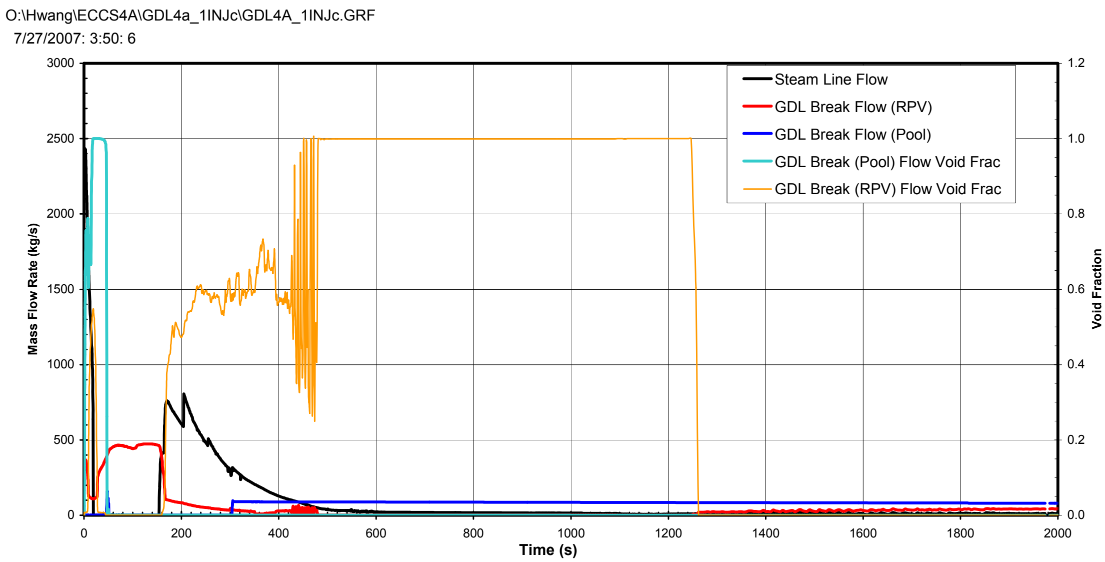


Figure 6.3-35a. Steam Line and Break Flows, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

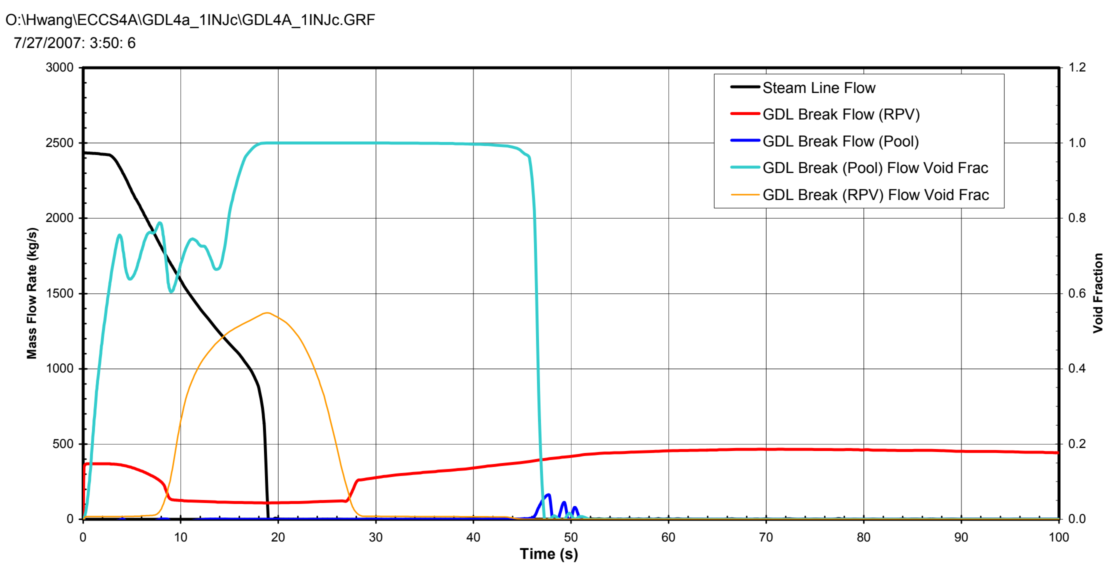


Figure 6.3-35b. Steam Line and Break Flows, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

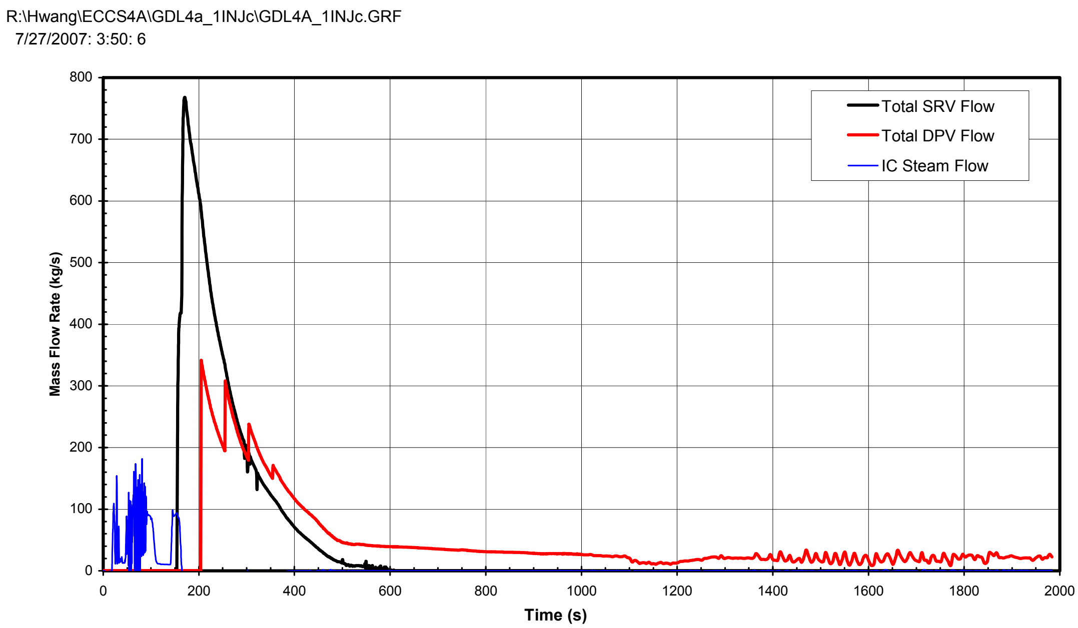


Figure 6.3-36a. ADS Flows, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

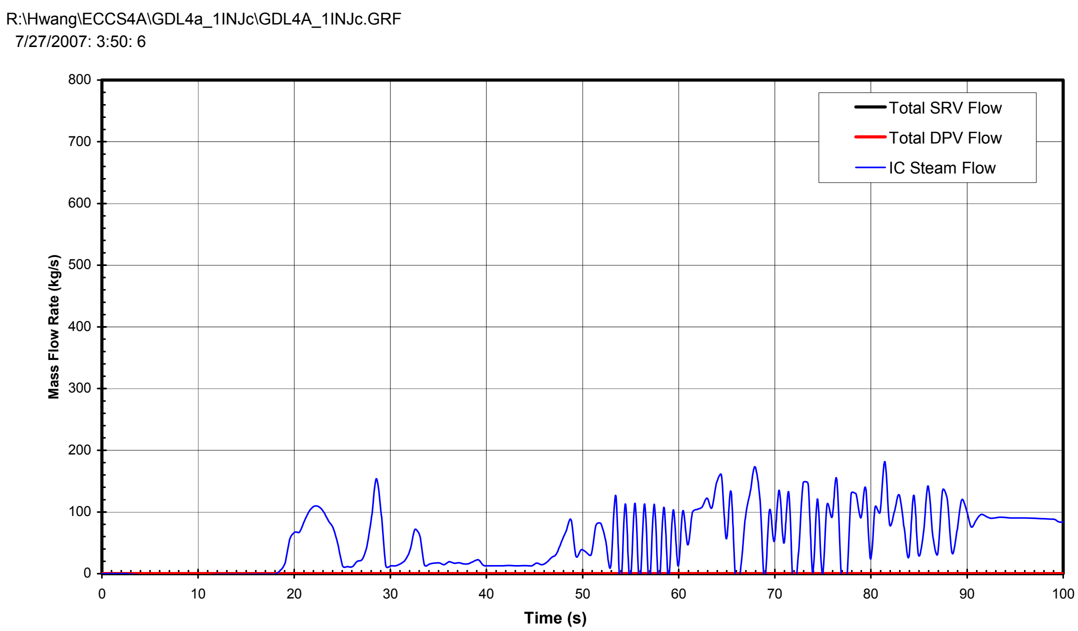


Figure 6.3-36b. ADS Flows, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

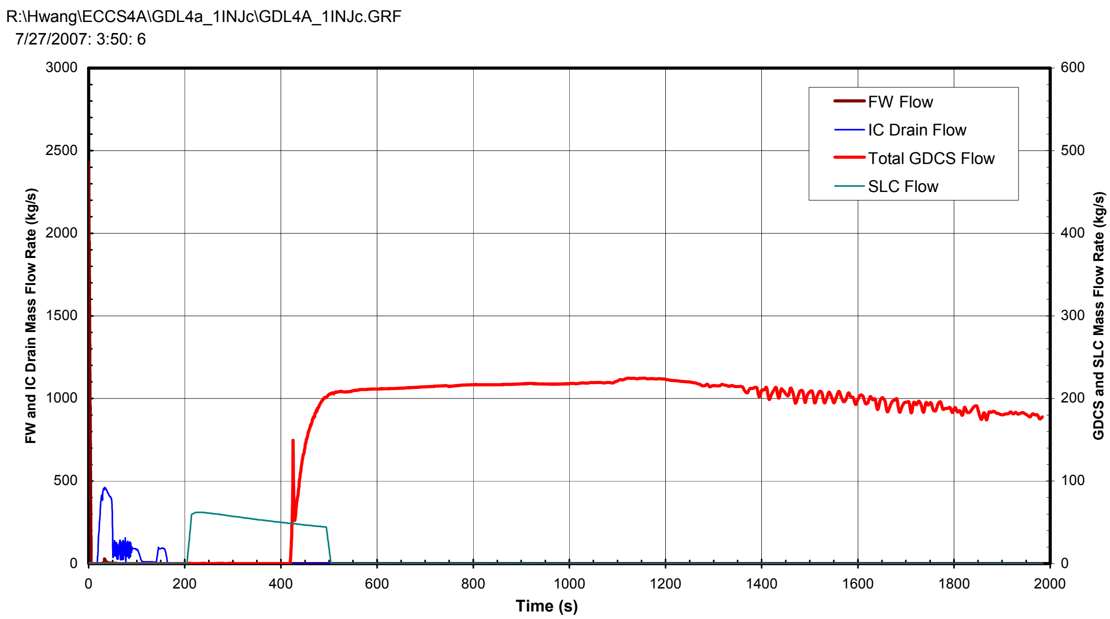


Figure 6.3-37a. Flows Into Vessel, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

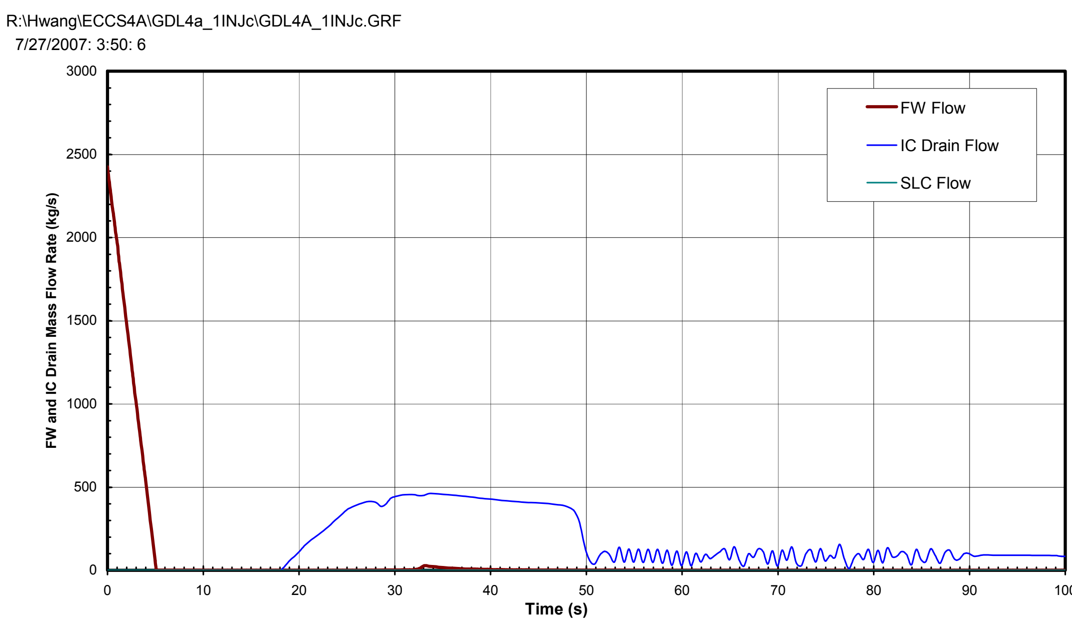


Figure 6.3-37b. Flows Into Vessel, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

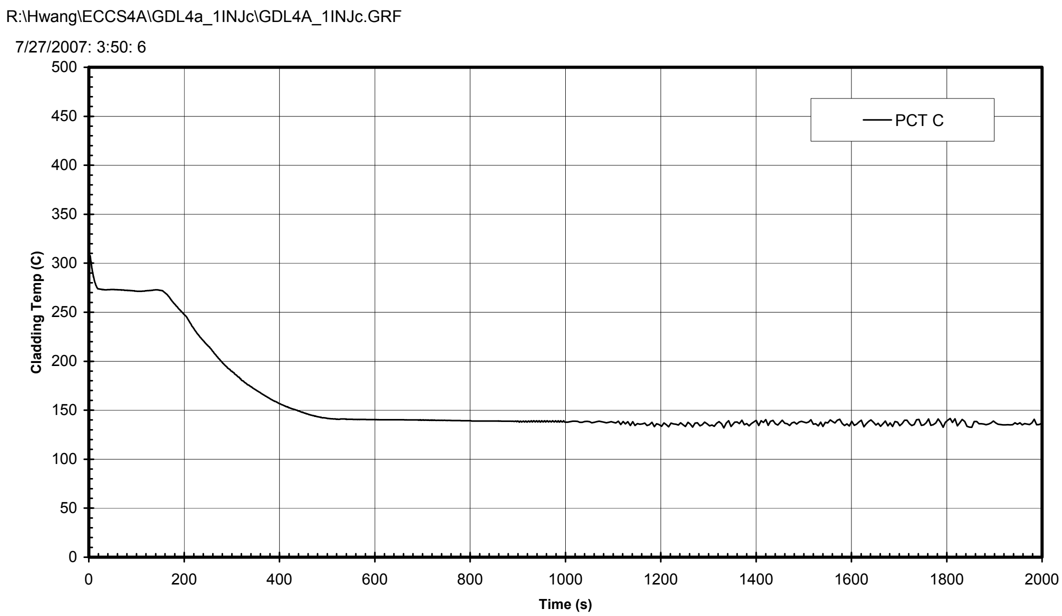


Figure 6.3-38a. PCT, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

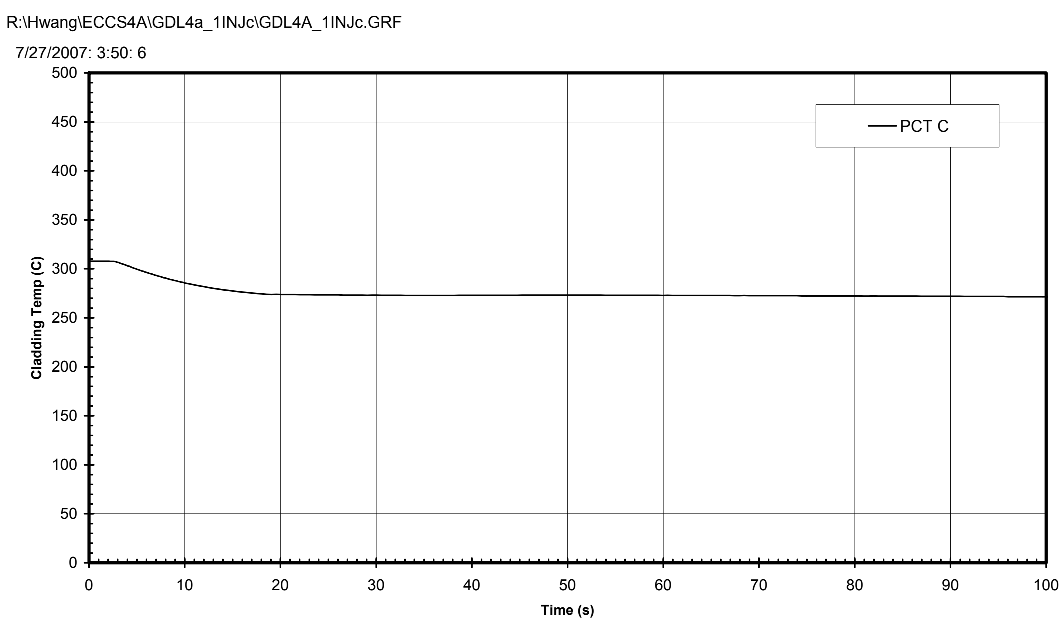


Figure 6.3-38b. PCT, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

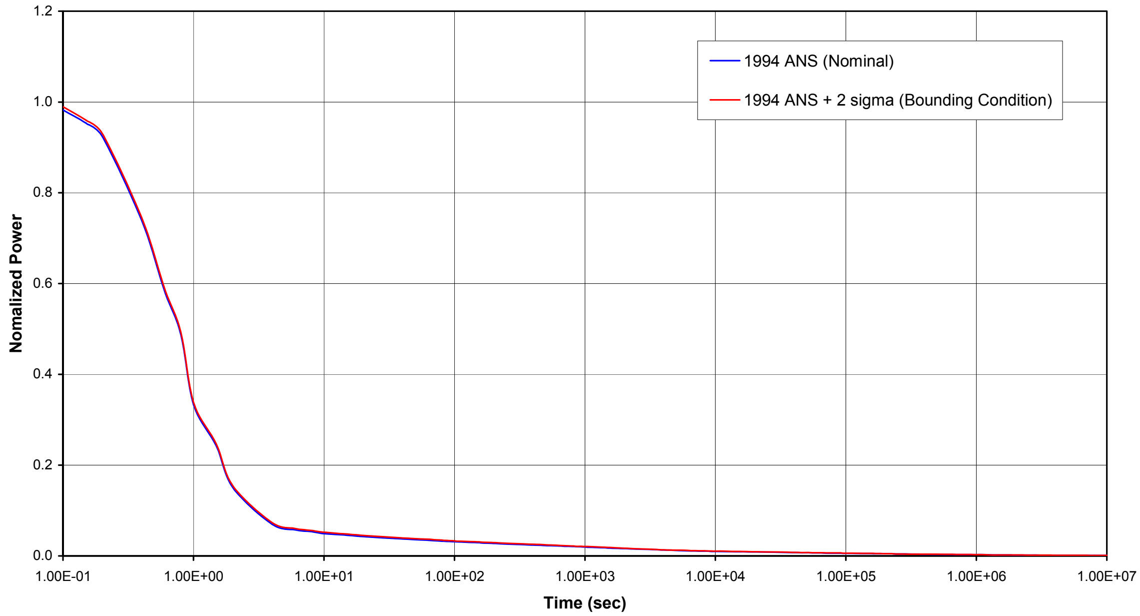


Figure 6.3-39. Normalized Shutdown Power

6.4 CONTROL ROOM HABITABILITY SYSTEMS

The Control Room Habitability Area (CRHA) is served by a combination of individual systems that collectively provide the habitability functions. The systems that make up the habitability systems are the:

- CRHA HVAC Subsystem (CRHAVS)
- Radiation Monitoring Subsystem (RMS)
- Lighting System
- Fire Protection System (FPS)

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 Habitability Area HVAC Subsystem	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
Control Room Habitability System	Subsection 7.3.4
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.

When AC power is available, the CRHAVS provides normal and abnormal HVAC service to the CRHA as described in Subsection 9.4.1. When AC power is unavailable for an extended time, or if high radioactivity is detected by the RMS in the CRHA outside air supply duct, the CRHA normal air supply is automatically isolated and the habitability requirements are then met by the operation of an Emergency Filter Unit (EFU). Two trains of EFUs, consisting of two (2) 100% fans each, including HEPA and carbon filters, serve the CRHA envelope. Redundant fans are provided for each EFU to allow continued system operability during maintenance of electrical power supplies. The EFUs provide emergency ventilation and pressurization for the CRHA. When AC power is unavailable, the CRHA is passively cooled by the CRHA passive heat sink.

The RMS provides radiation monitoring of the CRHA environment and outside air intake. The FPS provides smoke detection and fire damper isolation. Emergency lighting is provided by the Lighting System. Storage capacity is provided in the main control room for personnel support equipment. Manual hose stations outside the CRHA and portable fire extinguishers provide fire suppression in the CRHA.

The CRHA 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 MCR 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 GDC 4, 5 and 19 discussed in 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 habitability systems maintain the main control room environment suitable for prolonged occupancy throughout the duration of the postulated accidents. Chapter 15 discusses dose protection requirement following a postulated radioactive release. Refer to Sections 3.1, 15.0, and Subsections 6.4.5 and 9.4.1 for discussions on conformance with GDC 19, and to Section 1.11 for a discussion on conformance with Generic Issues B-36 and B-66.

- The main control room is designed to withstand the effects of an SSE and a design-basis tornado as described in Section 3.8.
- The radiation exposure of main control room personnel throughout the duration of the postulated limiting faults discussed in Chapter 15 does not exceed the limits set by GDC 19.
- The emergency habitability system maintains the American Society of Heating, Refrigeration, and Air Conditioning Engineers fresh air requirements for up to 21 main control room occupants. (Ref. ASHRAE Standard 62).
- The habitability systems detect and protect main control room personnel from external fire, smoke, and airborne radioactivity.
- Automatic actuation of the individual systems that perform a habitability systems function is provided. Smoke detectors, radiation detectors, and associated control equipment are installed at various plant locations as necessary to provide the appropriate operation of the systems.
- The CRHA includes all instrumentation and controls necessary during safe shutdown of the plant and is limited to those areas requiring operator access during and after a DBA.
- CRHA habitability requirements are satisfied without the need for individual breathing apparatus and/or special protective clothing.
- The CRHA EFUs and associated fans and ductwork, the CRHA envelope structures, the CRHA heat sink, doors, isolation dampers and/or valves, including supporting ductwork/piping, 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 EFU trains are designed with sufficient redundancy to ensure operation under emergency conditions.
- The EFUs are operable during loss of preferred power, loss of onsite AC power or Station Blackout (SBO).
- The EFUs operate 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 one EFU in operation.
- Electrical power for safety-related equipment including EFUs, dampers, valves and associated instrumentation and controls is supplied from the safety-related uninterruptible power supply. Active safety-related components are redundant and their power supply is divisionally separated such that the loss of any two electrical divisions does not render the component function inoperable.

6.4.1.2 Power Generation Design Bases

- The CRHAVS 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 EFUs and the CRHAVS are incorporated in the design.

6.4.2 System Design

Only the habitability portion of the CRHAVS is discussed in this subsection. The remaining systems are described only as necessary to define their functions in meeting the safety-related design bases of the habitability systems. Descriptions of the CRHAVS, FPS, Lighting System, and RMS are found in subsections 9.4.1, 9.5.1, 9.5.3, and Section 11.5, respectively. Figure 6.4-1 provides a schematic diagram of the CRHAVS.

The EFUs are redundant safety-related components that supply filtered air to the CRHA for breathing and pressurization to minimize inleakage. The EFUs and their related components form a safety-related subset of the CRHAVS. The EFU portion of the system and the associated components are designed, constructed, and tested as a safety-related nuclear air filtration system in accordance with ASME AG-1 requirements. An EFU is automatically initiated. There are two redundant EFU trains to provide protection against a single failure. Each train consists of an air intake, two (2) 100% capacity fans, filtration housing, ductwork, and dampers as shown in Figure 6.4-1. The EFUs have been sized to provide sufficient breathing quality air and to maintain a positive pressure in the CRHA with respect to the adjacent areas.

6.4.3 Control Room Habitability Area

The CRHA boundary is located on elevation –2000mm in the Control Building. The layout of the CRHA, which includes the MCR, is shown within Figure 1.2-3. The CRHA envelope includes the following areas:

- Main Control Room (Room 3275)

- Shift Supervisor Office (Room 3272)
- Shift Supervisor Conference Room (Room 3273)
- Operator's Area (Room 3270)
- Shift Technical Advisor Office (Room 3271)
- Main control Room Storage Room (Room 3204)
- Electrical Panel Board Room (Room 3205)
- Restroom A (Room 3201)
- Restroom B (Room 3202)

These areas constitute the operation control area, which can be isolated and remain habitable for 72 hours without AC power if high radiation conditions exist. Potential sources of danger such as steam lines, pressurized piping, pressure vessels, CO_2 fire fighting containers, etc. are located outside of the CRHA.

Heat Sink

The function of providing a passive heat sink for the CRHA is part of the CRHA emergency habitability system. The heat sink for each room is designed to limit the temperature rise inside each room during the 72-hour period following a loss of CRHAVS operation. The heat sink consists of the thermal mass of the concrete that makes up the ceilings and walls of these rooms.

Radiation Protection

Description of control room instrumentation for monitoring of radioactivity is given in Sections 11.5 and 12.3.

Shielding Design

The design basis LOCA dictates the shielding requirements for the CRHA. Main control room shielding design bases are discussed in Section 12.3. Descriptions of the design basis LOCA source terms, main control room shielding parameters, and evaluation of doses to main control room personnel are presented in Section 15.4. The main control room location in the plant with respect to designated radiation zones is shown in Figure 12.3-3.

Fire Protection

A description of the smoke detectors is in Subsection 9.5.1. Smoke removal is described in Subsection 9.4.1.

Layout

The layout of the CRHA, which includes the MCR, is shown within Figures 1.2-3 and 9A.2-3.

Release Points

Radiological release parameters are described in Section 15.4

Component Descriptions

The EFU outside air supply portion of the CRHAVS is safety-related and Seismic Category I. Single active failure protection is provided by the use of two trains, which are physically and electrically redundant and separated. In the event of failure in one train, the failed train is isolated and the alternate train is automatically initiated. Both trains are 100% capacity and capable of supplying 99% credited efficiency filtered air to the CRHA pressure boundary at the required flow rate.

- EFUs

The EFU design utilizes a Pre-filter, HEPA filter, Carbon Filter, and Post-filter to provide radiological protection of the CRHA outside air supply. The units, including the housings, internal components, ductwork, dampers, fans and controls are designed, constructed, and tested in accordance with ASME AG-1 to meet the requirements of Regulatory Guide 1.52. Each EFU design incorporates 2 (two) 100% capacity upstream fans powered by the respective divisional power supply to maintain the entire filtration sequence and air delivery duct to the CRHA under positive pressure. See Table 6.4-1 for detailed filter efficiency requirements.

- EFU Fans

EFU fans are designed and rated in accordance with ANSI/AMCA 210, 301, 302, 303, and 410.

- Isolation Dampers and Valves

The CRHA pressure boundary includes penetrations, dampers and/or valves, interconnecting duct or piping, and related test connections and manual valves. The isolation dampers and/or valves are classified as Safety Class 3 (Table 3.2-1) and Seismic Category I. The dampers and/or valves have spring return actuators that fail close on a loss of electrical power. Isolation dampers are constructed, qualified, and tested in accordance with ANSI/AMCA 500-D or ASME AG-1, Section DA. Isolation valves are qualified to provide a leak tight barrier for the CRHA envelope pressure. The boundary isolation function of isolation dampers and/or valves will be demonstrated by pressure testing of the CRHA and in-leakage testing in accordance with ASTM E741.

- Tornado Protection Dampers

Tornado protection dampers are a split wing or equivalent type designed to close automatically. The tornado protection dampers are designed to mitigate the effect of a design basis tornado.

- Shutoff, Balancing, and Backdraft Dampers

All shutoff, balancing, and backdraft dampers in the EFU outside air delivery path are constructed, qualified, and tested in accordance with ASNI/AMCA 500-D or ASME AG-1, Section DA. Backdraft dampers meet the Leakage Class II requirements of ASME AG-1. Remotely operated two-position type shutoff dampers are designed for the maximum fan static pressure.

- Ductwork and Related Components

Ductwork, duct supports, and accessories are constructed of galvanized or stainless steel, or of carbon or stainless steel if standard pipe is utilized. Ductwork subject to fan shutoff pressures is structurally designed to accommodate fan shutoff pressures. The EFU related ductwork, including the EFUs and the related ductwork outside the CRHA boundary, are designed in accordance with ASME AG-1, Article SA-4500, to provide low leakage components necessary to maintain the CRHA habitability.

- Control Room Access Doors

Two sets of doors, with a vestibule between them that acts as an airlock, are provided at each access to the main control room.

Leak Tightness

The CRHA boundary envelope structures are designed with low leakage construction. The CRHA is located in an underground portion of the Control Building (CB). The boundary walls are adjacent to underground fill or underground internal areas of the CB. The construction consists of cast-in-place reinforced concrete walls and slabs, and is constructed to minimize leakage through joints and penetrations. The following features are applied as required to achieve the leak tightness objective:

- The EFU filter train is located downstream of the EFU fan. This maintains the filter train and delivery ductwork to the CRHA at a positive pressure precluding any unfiltered in-leakage into the system.
- The access doors are designed with self-closing devices, which close and latch the doors automatically. There are double door air locks for access and egress during emergencies.
- The outside surface of penetration sleeves in contact with concrete is sealed with epoxy or equivalent sealant. Piping and electrical cable penetrations are sealed with a qualified pressure resistant material compatible with penetration materials and/or cable jacketing.
- Inside surfaces of penetrations and sleeves in contact with commodities are sealed.
- Penetration sealing materials are designed to withstand at least a ¼ inch water gauge pressure differential. The bulk penetration sealing material is gypsum cement or equivalent, with epoxy or equivalent sealants applied to compliment penetration sealing.
- The CRHA utilizes internal recirculation Air Handling Units (AHUs) that preclude any AHU ductwork external to the CRHA envelope.

Interaction With Other Zones and Pressure-Containing Equipment

During normal operation the CRHA is heated, cooled, ventilated, and pressurized by either of a redundant set of recirculating air handling units and either of a redundant set of outside air intake fans for ventilation and pressurization purposes. See Figure 6.4-1 and Subsection 9.4.1 for a complete description of the CRHAVS.

During a radiological event or an SBO, the EFU 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 CRHA emergency habitability portion of the CRHAVS is not required to operate during normal conditions. The normal operation of the CRHAVS maintains the air temperature of the CRHA within a predetermined temperature range. This maintains the CRHA emergency habitability system passive heat sink at or below a predetermined temperature. The normal operation portion of the CRHAVS operates during all modes of normal power plant operation, including startup and shutdown. For a detailed description of the CRHAVS operation see Subsection 9.4.1.

The COL Applicant will verify procedures and training for control room habitability address the applicable aspects of NRC Generic Letter 2003-01 and are consistent with the intent of Generic Issue 83 (6.4-1-A).

Emergency Mode

Operation of the emergency habitability portion of the CRHAVS is automatically initiated by either of the following conditions:

- High radioactivity in the main control room supply air duct
- Extended Loss of AC power

Operation can also be initiated by manual actuation. Upon receipt of a high radiation level in the main control room supply air duct exceeding the setpoint, the normal outside air intake and restroom exhaust are isolated from the CRHA pressure boundary by automatic closure of the isolation dampers in the system ductwork. At the same time, one of the EFU automatically starts and begins to deliver filtered air from one of the two unique safety-related outside air intake locations. A constant air flow rate is maintained and this flow rate is sufficient to pressurize the CRHA boundary to at least 31 Pa (1/8-inch water gauge) positive differential pressure with respect to the adjacent areas. The EFU system air flow rate is also sufficient to supply the fresh air requirement of 9.5 l/s (20 cfm) per person for up to 21 occupants. (Ref. ASHRAE Standard-62).

With a source of AC power available, the EFU can operate indefinitely. In the event that AC power is not available, the safety-related battery power supply is sized to provide the required power to the operating EFU fan for 72 hours of operation. For longer-term operation, from after 72 hrs to 7 days, a small portable AC power generator that is kept on the plant site can power the EFU fan system. The temperature and humidity in the CRHA pressure boundary following a loss of the normal portion of the CRHAVS remain within the limits for reliable human performance (References 6.4.10-1 and 6.4.10-2) over a 72-hour period. The CRHA isolation dampers fail as is on a loss of AC power or instrument air.

Backup power to the safety-related Control Room (CR) EFU fans (post 72 hours) is provided by a small portable dedicated electrical generator. This generator is required to support operation of the Control Room EFU beyond 72 hours through 7 days after an accident. This function is a nonsafety-related function that satisfies the significance criteria for Regulatory Treatment of NonSafety Systems. This AC power generator is kept on the plant site to power the EFU fan system. For a period between 7 days out to 30 days, the EFU can be powered from either off-site power, onsite diesel generator powered PIP bus, or by continued use of the small AC power generator. The requirements for the CRHAVS portable generator are described in Appendix 19A.

Upon a loss of preferred power or SBO, the initial ranges of temperature/relative humidity in the CRHA are 22.8 – 25.6°C (73 - 78°F) and 25% - 60% RH. During the first two hours of an SBO, most of the equipment in the MCR remains powered by the nonsafety-related battery supply. After two hours, the nonsafety-related batteries are exhausted, and only a small amount of safety-related equipment remains powered. During the first two hours the environmental conditions are maintained within the normal ranges listed above. This is accomplished via the continued operation of a CRHA recirculation AHU and chilled water pump powered from the same nonsafety-related battery supply that powers the non-safety MCR equipment. Chilled water from a chilled water thermal storage tank is utilized as the heat sink. The cooling function for this two hour period is not a safety function, if this cooling function is lost, the nonsafety- related equipment and their associated heat loads are automatically de-energized.

If power remains unavailable beyond two hours, the remaining CRHA safety-related equipment heat loads are dissipated passively to the CRHA heat sink. The CRHA heat sink limits the temperature rise to that listed in Table 6.4-1. The CRHA is passively cooled by conduction into the walls and ceiling. Sufficient thermal mass is provided in the walls and ceiling of the main control room to absorb the heat generated by the equipment, lights, and occupants.

These actions discussed above protect the main control room occupants from a potential radiation release and maintain the CRHA as a safe and habitable environment for continued operator occupancy.

6.4.5 Design Evaluations

System Safety Evaluation

Doses to main control room personnel are calculated for the accident scenario where the EFU provides filtered air to pressurize the CRHA. Doses are calculated for the following accident:

Loss Of Coolant Accident Table 15.4-9

The dose analyses are performed in accordance with the requirements of Regulatory Guides 1.194 and 1.196. For all events, the dose is within the dose acceptance limit of 5.0 rem TEDE. The details of the analytical assumptions for modeling the doses to the main control room personnel are delineated in Chapter 15. No radioactive material storage areas are located adjacent to the main control room pressure boundary. As discussed and evaluated in subsection 9.5.1, the use of noncombustible construction and heat and flame resistant materials throughout the plant reduces the likelihood of fire and consequential impact on the main control room atmosphere. Operation of the CRHAVS in the event of a fire is discussed in Subsection 9.4.1. The exhaust stacks of the onsite standby power diesel generators are located in excess of 48 m (157 ft) away from the fresh air intakes of the main control room. The onsite standby power system fuel oil storage tanks are located in excess of 55 m (180 ft) feet from the main control room fresh air intakes. These separation distances reduce the possibility that combustion fumes or smoke from an oil fire would be drawn into the main control room.

Typical sources of onsite chemicals are listed in Table 6.4-2, and their locations are shown on Figure 1.1-1. Analysis of these sources are in accordance with Regulatory Guide 1.78 and the methodology in NUREG-0570, “Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release” is to be performed on a site specific basis (See Subsection 6.4.9).

During emergency operation, the CRHA emergency habitability system passive heat sink is designed to limit the temperature rise inside the CRHA to 8.3°C (15°F). This maintains the CRHA within the limits for reliable human performance (References 6.4.10-1 and 6.4.10-2) over 72 hours. The walls and ceiling that act as the passive heat sink contain sufficient thermal mass to accommodate the heat sources from equipment, personnel, and lighting for 72 hours. The EFU portion of the CRHAVS nominally provides 200 l/s (424 scfm) of ventilation air to the main control room and is sufficient to pressurize the control room to at least a positive 1/8-inch water gauge differential pressure with respect to the adjacent areas. This flowrate also supplies the recommended fresh air supply of 9.5 l/s (20 cfm) per person for a maximum occupancy of 21 persons. (Reference ASHRAE Standard-62). Automatic isolation of the normal air intake and transfer of outside air supply to the EFU is initiated by either the following conditions:

- High radioactivity in CRHA normal air supply duct
- Extended Loss of AC power

The airborne fission product source term in the reactor containment following the postulated LOCA is assumed to leak from the containment. The concentration of radioactivity is evaluated as a function of the fission product decay constants, the containment leak rate, and the meteorological conditions assumed. The assessment of the amount of radioactivity within the CRHA takes into consideration the radiological decay of fission products and the infiltration/exfiltration rates to and from the CRHA pressure boundary. Specific radiological protection assumptions used in the generation of post-LOCA radiation source terms are described fully in Chapter 15.

Smoke protection is 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, a dedicated fan, intake, and exhaust path are utilized to purge the control room with a high volume of outside airflow.

A high radiation condition causes automatic changeover to the operating modes described in Section 6.4.4 and in Section 9.4.1.2. The EFU automatically starts to provide CRHA breathing air and pressurization during an 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 EFU components are provided to ensure CRHA pressurization upon a radiological event concurrent with SBO.

The EFUs are 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 EFUs.

Potential site-specific toxic or hazardous materials that may affect control room habitability will be identified by the COL applicant. The COL Applicant will identify potential site specific toxic or hazardous materials that may affect control room habitability in order to meet the requirements of TMI Action Plan III.D.3.4 and GDC 19 (6.4-2-A).

6.4.6 Life Support

In addition to the supply of vital air, food, water and sanitary facilities are provided.

6.4.7 Testing and Inspection

A program of preoperational and post operational testing requirements is implemented to confirm initial and continued system capability. The CRHAVS is tested and inspected at appropriate intervals consistent with plant technical specifications. Emphasis is placed on tests and inspections of the safety-related portions of the habitability systems.

Preoperational Inspection and Testing

Preoperational testing of the CRHAVS is performed to verify that the minimum air flow rate of 200 l/s (424 cfm) is sufficient to maintain pressurization of the main control room envelope of at least 31 Pa (1/8" wg) with respect to the adjacent areas. The positive pressure within the main control room is confirmed via the differential pressure transmitters within the control room. The installed flow meters are utilized to verify the system flow rates. The pressurization of the control room limits the ingress of radioactivity to maintain operator dose limits below regulatory limits. Air quality within the CRHA environment is confirmed to be within the guidelines of ASHRAE Standard 62 requirements for continued occupancy via meeting the fresh air supply requirement of 9.5 l/s (20 cfm) per person for the type of occupancy expected in the CRHA. The capacity of the safety-related battery is verified to ensure it can power an EFU fan for a minimum of 72 hours. Heat loads within the CRHA are verified to be less than the specified values. Preoperational testing of the CRHAVS isolation dampers is performed to verify the leaktightness of the dampers. Preoperational testing for CRHA inleakage during EFU operation is conducted in accordance with ASTM E741 Testing and inspection of the radiation monitors is discussed in Section 11.5. The other tests noted above are discussed in Chapter 14.

Inservice Testing

Inservice testing of the CRHAVS includes operational testing of the EFU fans and filter unit combinations, EFU filter performance testing automatic actuation testing of the CRHA isolation dampers and EFU fans , and unfiltered air inleakage testing of the CRHA envelope boundary . The CRHA boundary is Pressure Tested (PT) periodically to verify leakage tightness on the envelope walls, doors, and boundaries. Testing to demonstrate the integrity of the CRHA envelope is performed in accordance with Regulatory Guide 1.197 and ASTM E741.

Nuclear Air Filtration Unit Testing

The EFU filtration components are periodically tested in accordance with ASME AG-1, Code On Nuclear Air And Gas Treatment, to meet the requirements of Regulatory Guide 1.52.

Periodic surveillance testing of safety-related CRHA isolation dampers and the EFU components are carried out per IEEE-338. Safety-related CRHA isolation dampers and the EFU are operational during plant normal and abnormal operating modes.

6.4.8 Instrumentation Requirements

A description of the required instrumentation is given in Subsection 9.4.1.5. Instrumentation required for actuation of the CRHAVS emergency habitability system, a description of initiating circuits, logic, periodic testing requirements, and redundancy of instrumentation relating to the habitability systems is provided in Section 7.3. Details of the radiation monitors used to provide the main control room indication of actuation of a CRHA isolation and EFU initiation are given in Section 11.5. Alarms for the following CRHA/CRHAVS conditions are provided in the MCR:

- Low airflow (each EFU fan, recirculation AHU, and Outside Air Intake Fan);
- High filter pressure drop (each EFU and normal Outside Air Intake filters);
- High space temperatures;
- Low space temperatures;
- Low recirculation AHU entering air temperature;
- Low CRHA differential pressure;
- Smoke detected;
- High and low humidity in the CRHA;
- CRHA airlock doors are open during an SBO;
- Area high radiation in the CRHA; and
- High radiation in the Outside Air Intake duct.

6.4.9 COL Information

6.4-1-A **CRHA Procedures and Training**

The COL Applicant will verify procedures and training for control room habitability address the applicable aspects of NRC Generic Letter 2003-01 and are consistent with the intent of Generic Issue 83.

6.4-2-A **Toxic Gas Analysis**

The COL Applicant will identify potential site specific toxic or hazardous materials that may affect control room habitability in order to meet the requirements of TMI Action Plan III. D.3.4 and GDC 19.

6.4.10 References

6.4-1 MIL-HDBK-759C, Human Engineering Design Guidelines

6.4-2 MIL-STD-1472E, Human Engineering

6.4-3 A Priotization of Generic Safety Issues, NUREG-0933, October 2006.

Table 6.4-1

Design Parameters for CRHAVS

Operation periods:	Normal plant operation, plant startup, and plant shutdown
Outside Air Design Conditions:	
For CRHAVS (0% Exceedance values)	Summer: 47.2°C (117°F) Dry Bulb 26.7°C (80°F) Wet Bulb(Coincident) 31.1°C (88°F) wb (non-coincident)
	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 and Accident Conditions)	Maximum 8.3°C (15°F) rise above normal operating temperature for the first 72 hours into the event, RH not controlled
Pressurization	> 31 Pa (1/8" wg) positive differential
CRHAVS EFUs	
CRHAVS Breathing air supply capacity	9.5 l/s/ (20 cfm) person for up to 21 persons (200 l/s total) (424 cfm total) for 72 hours
Quantity	2 - 100% capacity each
Capacity	Flow – 200 l/s per unit (424 cfm)
Type	Metal housing containing medium efficiency pre-filter, HEPA filter, carbon filter, and post filter
Medium efficiency filter minimum ASHRAE efficiency	40%
HEPA filter minimum efficiency	99.97% DOP
Post-filter minimum efficiency	95% DOP
Carbon Adsorber Requirements	minimum 4" bed depth minimum air residence time of 0.5 seconds decontamination efficiency of 99%

Table 6.4-2

Typical Onsite Chemicals and Typical Locations

Chemical	System	Location
Carbon Dioxide	Generator Gas Control System	CO_2 Storage Area - Outside the Turbine Building, West side
Hydrogen	Generator Gas Control System	H_2 Storage Area - Outside the Turbine Building, West side
Nitrogen	Nitrogen Supply and Misc. Gas System	N_2 Storage Area - Outside the Reactor Building, West side
Boiler deposit control	Chemical Addition System	Aux. Boiler Building
Sodium Sulfite	Chemical Addition System	Aux. Boiler Building
Boiler alkylinity control, Sodium Hydroxide	Chemical Addition System	Aux. Boiler Building
Sodium Hypochlorite	Potable Water System	Make-up Water Treatment Bldg.
Sulfuric Acid	Make-up Water System	Make-up Water Treatment Bldg.
Caustic Soda	Make-up Water System	Make-up Water Treatment Bldg.
Floculant	Make-up Water System	Make-up Water Treatment Bldg.
Sodium Hypochlorite	Circulating/Service Water Systems	Pump House
Sulfuric Acid	Circulating/Service Water Systems	Pump House
Scale/Corrosion Inhibitor	Circulating/Service Water Systems	Pump House
UREA Dry Power aqua solution 40% (NH2)2CO	Diesel Generator Exhaust	Diesel Generator Building
25% solution liquid ammonia aqua solution NH_3	Diesel Generator Exhaust	Diesel Generator Building

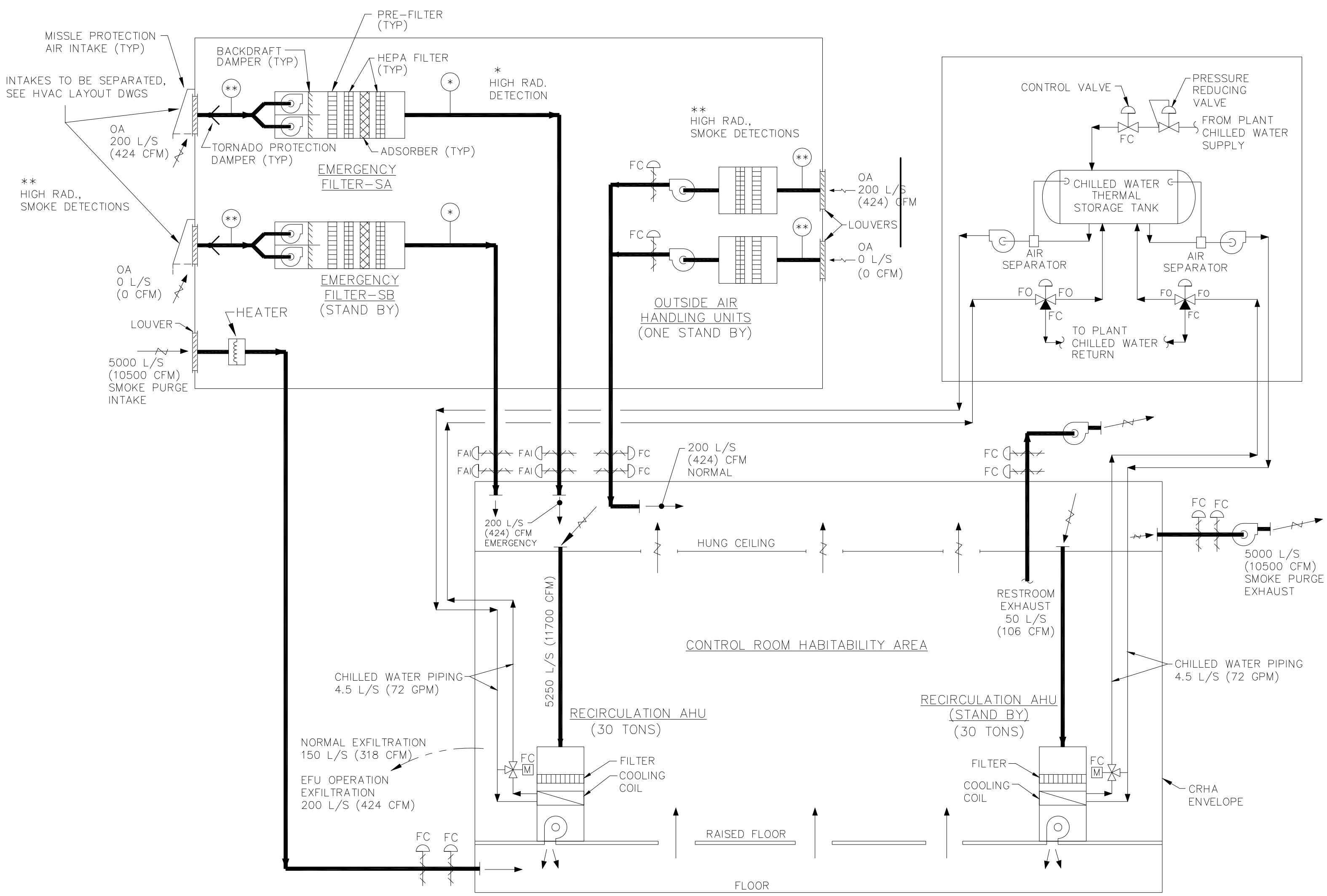


Figure 6.4-1. CRHAVS Schematic Diagram

6.5 ATMOSPHERE CLEANUP SYSTEMS

The ESBWR utilizes a safety-related filter system to protect the main control room environment following a design basis accident as discussed in Section 6.4. The system meets the acceptance criteria in Standard Review Plan 6.5.1.

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 non safety-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 (for example, radioactive decay) radionuclides through natural processes.

6.5.2.1 General

The ESBWR is functionally divided into three distinct buildings, the RB 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 RB 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 design 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 6.2.1.1.2 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 RB 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 RB 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 stack. Under high radiation conditions, the ventilation shuts down to provide a hold up volume. The affected area may be routed to the Purge Exhaust Filter Unit to ensure building and affected area negative pressure is maintained, while controlling contamination. Under high energy release conditions such as a high energy line break, the overpressure is routed to the RB operating floor in which blow out panels in the upper sections relieve the overpressure to the environment. The RB HVAC system performs no safety-related function, other than the building ventilation isolation function, but credit is taken for hold up in the RB as discussed in Subsection 15.4.4.5.2.

The controlled area of the RB 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 ICs, and miscellaneous other penetrations) terminate in this volume. The second isolation valves on all GDC 54 lines (with the exception of the IC containment isolation valves) are found in this volume such that any potential valve leakage as well as penetration leakage collects in here. The RB under accident conditions is automatically isolated or passively sealed (for example, water loop seals) to provide a hold up barrier. When isolated, the RB can be serviced by the HEPA and charcoal filtration systems of the RB Purge Exhaust Filter Unit (Refer to Subsection 9.4.6). With low leakage and stagnant conditions, hold up mechanisms perform the basic mitigating functions.

Leakage from the PCCS heat exchangers through the IC/PCC pool is discussed in Subsection 15.4.4.5.2.

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 RB (for example, 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 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 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 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. Suppression pool compliance with GDCs 42 and 43 is addressed in Subsection 3.8.1.7. 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.

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 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, that is, ASME Code Class 2 and 3 items, respectively, as defined in Table 3.2-3. This section describes those programs implementing the requirements of ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Subsections IWC and IWD.

A preservice and inservice inspection program for Class 2 and 3 components and piping is based on the ASME code, Section XI, Edition and Addenda specified in accordance with 10 CFR 50.55a subject to limitations and modifications found therein. Additionally, 10 CFR 50.55a provides an allowance to request alternatives to or relief from ASME Section XI Code requirements. The COL applicant will provide a description of this program and its implementation. (See section 6.6.11.)

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 item boundaries include all or part of the following:

- (Deleted)
- 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
- Chilled Water System (CWS)
- Passive Containment Cooling System (PCCS).
- Containment Inerting System
- High Pressure Nitrogen Supply System
- Process Radiation Monitoring System

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.

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.

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.

Class 2 and Class 3 Piping, Pumps, Valves and Supports

Physical arrangement of piping pumps and valves provide personnel access to each weld location for performance of volumetric and surface (magnetic particle or liquid penetrant) examinations (Class 2 only), and sufficient access to supports for performance of visual VT-1 and VT-3 examinations in accordance with Subsection IWF. 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 100% volumetric 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; and
- Pump to valve.

Straight sections of pipe and spool pieces are added between fittings. The minimum length of the spool piece is 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 is 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 is in compliance with ASME Section XI, IWC-2500 and IWD-2500, and is listed in the preservice and inservice program. The items are listed by system and component description; or line number where available. The preservice and inservice inspection programs states the method of examination for each item.

For preservice examination, all of the items selected for inservice examination are 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 Examinations Methods VT1, VT-2 and VT-3 are used in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations meet the requirements of IWA-5240.

At locations where leakages are normally expected and leakage collection systems are located (for example, valve stems), the visual VT-2 examination verifies that the leakage collection system is operative.

Piping runs are 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 are 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 an arm's length (508 mm (20 inches)) of the surface to be examined. In addition, access is provided as necessary to enable physical contact with the item as necessary to perform the examination. Remote MT and Pressure Transmitter (PT) generally are not appropriate as a standard examination process, however, borescopes and mirrors can be used at close range to improve the angle of vision. As a minimum, insulation removal exposes 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 from approximately 406 mm (16 inches) on each side of the weld.

6.6.3.2.2 Volumetric Examination

Ultrasonic examination is 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 is 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 considered is 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 is 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 Radiographic Examination

ASME Section XI, IWA-2230 includes radiographic examination as a volumetric examination method. Section XI requires that the requirements of Article 2 of Section V be used for methodology. Radiography may be accomplished with x-rays or gamma rays and has historically been performed using film as the recording media. Due to ALARA, personnel access limitations in the work area when radiography is performed, radiography is not used as often as ultrasonic examination for Inservice Inspection. Use of computed and digital radiographic systems can result in greater latitude and reduced overall exposure times and make radiography a more practical examination method for Inservice Inspection. For the ESBWR, radiography may be used alone as a volumetric method or it may be used to supplement ultrasonic examination to improve coverage of the required examination volume.

6.6.3.2.4 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. In accordance with 10 CFR 50.55a(b)(2)(xix), IWA-2240 as written in the 1997 Addenda of ASME Section XI is used when applying these provisions.

6.6.3.2.5 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 employed with the automated ultrasonic examination equipment. Signals from each ultrasonic transducer are fed into a data acquisition system in which the key parameters of any reflectors are recorded. The data 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 are compared with the results of subsequent examinations to determine the behavior of the reflector.

6.6.3.2.6 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations are qualified in accordance with ASME Section XI, Appendix VII. Ultrasonic examination systems are 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 corresponds 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 corresponds 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

The evaluation of Class 2 component examination results is consistent with ASME Section XI, IWA-2000. 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. 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.

Class 2 components containing flaws or relevant conditions and accepted for continued service in accordance with the requirements of IWC-3122.3 or IWC-3132.3, and Class 3 components in accordance with IWD-3000 requirements, are subjected to successive period examinations in accordance with the requirements of IWC/IWD-2420. Examinations of Class 2 and 3 components that reveal flaws or relevant conditions exceeding Table IWC-3410-1 or IWD-3000 acceptance standards, respectively, are extended to include additional examinations in accordance with the requirements of IWC/IWD-2430.

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 is performed in accordance with IWC-5220 on Class 2 systems, and IWD-5220 on Class 3 systems The test include all Class 2 or 3 pressure retaining components and piping within the boundaries defined by IWC-5222 and IWD-5222. The test is performed once during each inspection period as defined in Tables IWC-2412-1 and IWD-2412-1 for Program B. The system leakage test includes a VT-2 examination in accordance with IWA-5240. The system leakage test is conducted at the system pressure during 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 includes 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 are 100 percent volumetrically examined each inspection interval as defined within Subsections 6.6.3.2 and 6.6.4. Accessibility, examination requirements, and procedures are as discussed in Subsections 6.6.2, 6.6.3 and 6.6.5, respectively. Piping in these areas is seamless, thereby eliminating longitudinal welds.

Erosion-Corrosion

Piping systems, ASME Section III Code Class 1, 2, 3 and nonsafety-related piping and components as described in NRC Generic Letter 89-08, determined to be susceptible to erosion-corrosion are subject to a program of nondestructive examinations to verify system structural integrity. The examination schedule and examination methods are determined in accordance with the Electric Power Research Institute (EPRI) guidelines in NSAC-202L-R2, which satisfies NRC Generic Letter 89-08, or the latest revision approved by NRC (or equally effective program), 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

Section XI requirements can be modified by invoking approved Section XI Code Cases. Approved Code Cases for inservice inspection are listed in Regulatory Guide 1.147. As applicable, the provisions of the Code Cases listed in Table 5.2-1 may be used for preservice and inservice inspections, pressure tests, evaluations, and repair and replacement activities.

6.6.10 Plant Specific PSI/ISI Program Information

6.6.10.1 Relief Requests

The specific areas where the applicable ASME Code requirements cannot be met are identified after the examinations are performed. Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(g)(5). The relief requests include appropriate justifications and proposed alternative inspection methods.

6.6.10.2 Code Edition

The ASME Section XI edition and addenda for this program description are as specified in Table 1.9-22. The COL Holder will define the applicable addition and addenda of the ASME Code in the plant specific ISI program.

6.6.11 COL Information

6.6-1-A ***PSI/ISI Program Description***

The COL applicant will provide a description of the PSI/ISI programs for Class 2 and 3 components and piping. The COL applicant will also provide a milestone for full program implementation.

6.6.12 References

None

6A. TRACG APPLICATION FOR CONTAINMENT ANALYSIS

NRC approved the application of TRACG for ESBWR containment analysis in NEDC-33083P-A. Since the approval, there have been some configuration changes in the ESBWR, primarily the location of the GDCS pools in the drywell space rather than being a part of the wetwell volume. None of these invalidate the PIRTs, applicability of TRACG models or the TRACG qualification basis for containment analysis. However, the details of the application procedure have been re-evaluated for the current configuration.

The TRACG application procedure consists of the following categories of inputs:

- Nodalization;
- Phenomena for which bounding models are used;
- Model inputs – key model inputs are chosen to be conservative; and
- Plant parameter inputs – key initial conditions are chosen to be conservative.

The overall philosophy of the application procedure remains the same. The details have been re-evaluated as shown in Table 6A-1.

Table 6A-1.

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A	Implementation in ESBWR Design Certification
Nodalization			
1	Fluid Volumes Combined model of Reactor Pressure Vessel (RPV) and containment.	The nodalization diagram for containment analysis is shown in Figure 3.7-1. The Upper Drywell (UDW) is modeled with the VSSL component with 11 levels and 2 rings. The wetwell is modeled with the VSSL component with 2 levels and 2 rings. The Lower Drywell (LDW) is modeled with a 1D TEE component.	The nodalization diagram is shown in Figure 6.2-7 of the DCD. The same nodalization is used for ECCS and containment analysis. The Upper Drywell is modeled with the VSSL component with 12 levels and 2 rings. The wetwell is modeled with the VSSL component with 3 levels and 2 rings. The Lower Drywell is modeled as part of VSSL component with 2 levels and 6 rings. The new nodalization results in a quicker and more complete clearing of non-condensable gas from the lower drywell.
2	Heat Slabs	Only the vertical heat structures representing the inner and outer wetwell walls are modeled for the containment. The area associated with the horizontal diaphragm floor between the wetwell (WW) and drywell (DW) is lumped in with the inner wetwell wall.	Same, see Figure 6A-1.

Table 6A-1.

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A	**Implementation in ESBWR Design Certification**
3	RPV heat structures	Internal heat structures are modeled; the RPV vessel wall is modeled as 1-D heat slab between the RPV and the DW.	Internal heat structures are modeled; the RPV vessel wall is modeled as lumped heat slab, see Figure 6A-2.
Bounding Models			
4	Suppression Pool (SP) Stratification: Region above highest source of mass and energy to pool is assumed to be stratified.	Section 3.3.1.1.1	Same
5	Wetwell gas space: Top level assumed to be stratified due to Vacuum Breaker (VB) leakage	Section 3.3.1.1.2	Same
6	Noncon - densable gases (NC) transport to WW: All NC moved to WW; rate of transfer varied to maximize containment pressure	Section 3.3.1.1.3 Hideout volume combined with LDW. Rate of release varied. Results insensitive to release rate.	With the current nodalization, NC in LDW quickly moved over to WW. Initial Hideout volumes are cleared. For bounding the calculation, two pipes are used to simulate the connection between the GDCS pool gas space and the drywell to purge residual NC in this space

Table 6A-1.

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A	**Implementation in ESBWR Design Certification**
7	Location of MSL break elevation – vary to obtain highest containment pressure	Section 3.3.1.1.3	Same. Break location in the upper portion of the UDW is slightly more limiting.
Model Inputs			
8	Critical flow at break, ADS-SRV, DPV: -2σ	Table 3.4-1	Same (Table 6.2-8) (PIRT84 = 0.81)
9	Decay heat: + 2σ	Table 3.4-1	Same (Table 6.2-8) (DH + 2σ)
10	SP surface heat transfer, lower bound	Table 3.4-1	Same (Table 6.2-8) (PIRT07 = 1.0)
11	PCC inlet loss: +2σ	Table 3.4-1	Same (Table 6.2-8) (1585 m^{-4})
12	PCC heat transfer: -2σ	Table 3.4-1	Same (Table 6.2-8) (PIRT78 = 0.902)
13	VB loss: +2σ	Table 3.4-1	Same (Table 6.2-8) (211.4 m^{-4})
14	PCC tube crud factor of 4.5e-5 m^2-K/W	Section 3.7.2	Same
15	NC gas properties	Air properties are used.	Nitrogen properties are used

Table 6A-1.

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A	**Implementation in ESBWR Design Certification**
16	Wall condensation heat transfer option	KSPw = Kuhn-Shrock-Peterson laminar film correlation with no shear enhancement for application to wall; BURM = Burmeister turbulent film correlation; Re_l = Reynolds number; Wall condensation HT = (NC gas degradation factor) * (KSPw for Re_l <1000 smoothed into 0.55*BURM for Re_l > 2000).	Same
17	Other parameters shown to have low sensitivity: nominal values	Table 3.4-1	Same
18	Radiolytic gas generation	Consistent with SRP 6.2-5 and RG 1.7.	Same
19	Isolation Condensers (ICs)	No credit is assumed for both the initial water inventory and the heat transfer in the ICs.	The initial water inventory in the ICs is modeled in the analysis; No credit is assumed for the heat transfer in the ICs.
Model Input for Numerical Stability			
20	Interfacial heat transfer at level in vertical portion of main vent	Nominal	Set to zero to prevent non-physical spikes in interfacial heat transfer due to level movement – effect of this change is of the order of ~ 1 psi in peak pressure
Plant Parameter Inputs			

Table 6A-1.

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A	Implementation in ESBWR Design Certification
21	RPV Power: 102%	Table 3.5-1	Same (Table 6.2-6)
22	WW relative humidity: 100%	Table 3.5-1	Same (Table 6.2-6)
23	PCC pool level: NWL	Table 3.5-1	Same (Table 6.2-6)
24	PCC pool temperature: 316.5K	Table 3.5-1	Same (Table 6.2-6)
25	DW pressure: 110.3kPa	Table 3.5-1	Same (Table 6.2-6)
26	DW temperature: 319.3K	Table 3.5-1	Same (Table 6.2-6)
27	WW pressure: 110.3kPa	Table 3.5-1	Same (Table 6.2-6)
28	WW temperature: 316.5K	Table 3.5-1	Same (Table 6.2-6)
29	SP temperature: 316.5K	Table 3.5-1	Same (Table 6.2-6)
30	GDCS pool temperature:	Table 3.5-1 (316.5K)	319.3K, (Configuration changed, GDCS connected to DW) (Table 6.2-6)
31	SP level: NWL+0.05m	Table 3.5-1 (5.50m)	Same (5.50m) (Table 6.2-6)
32	GDCS pool level:	Table 3.5-1 (NWL+0.05m = 6.75m, to minimize the draindown volume and the effective WW volume)	NWL 6.60m (Configuration changed, GDCS connected to DW)

Table 6A-1.

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A	Implementation in ESBWR Design Certification
33	DW relative humidity: 20%	Table 3.5-1	Same
34	RPV initial pressure: 7.274 MPa	Table 3.5-1	Same
35	RPV initial water level: NWL+0.3m	Table 3.5-1	Same

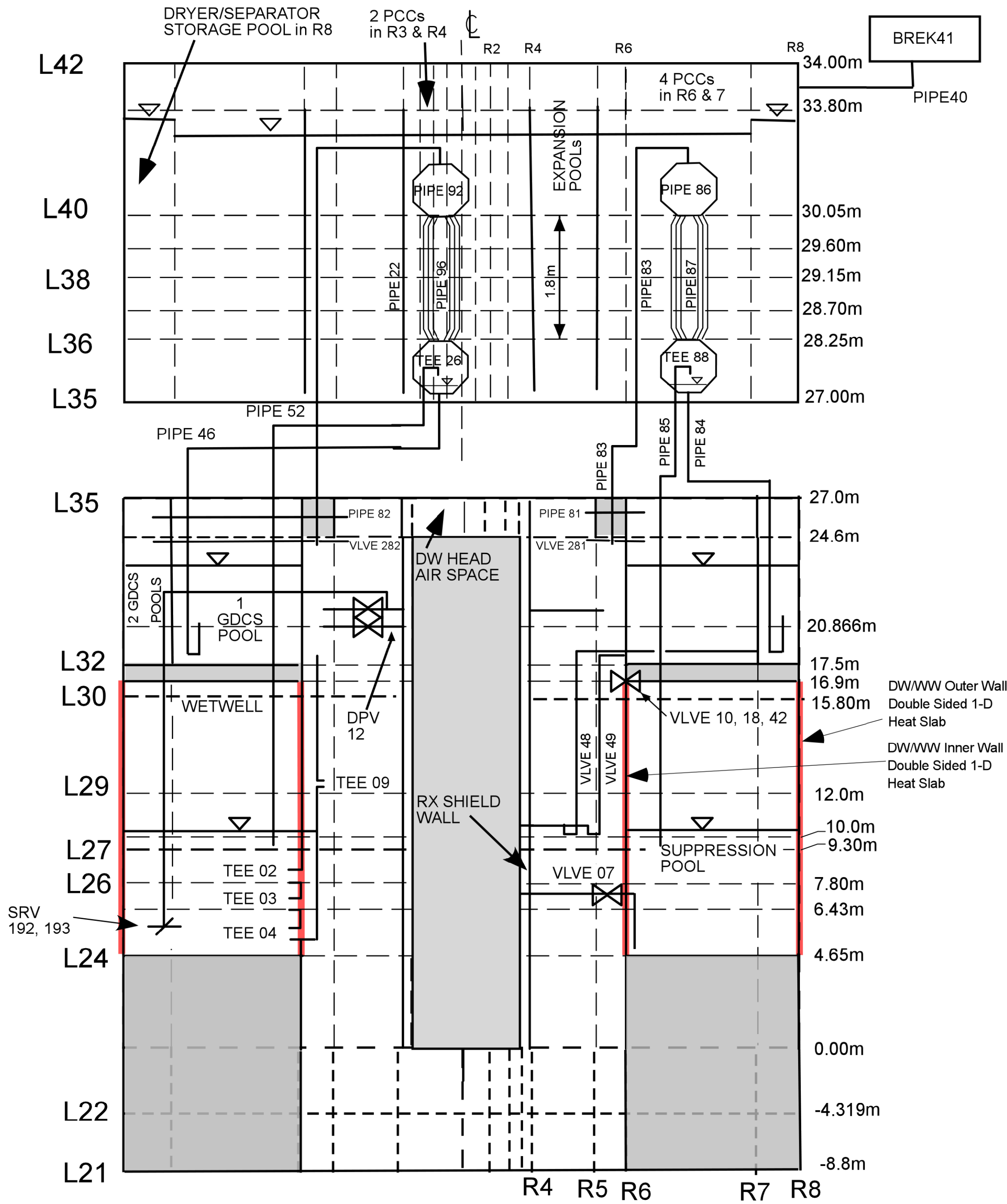


Figure 6A-1 TRACG Nodalization showing Containment Heat Slabs

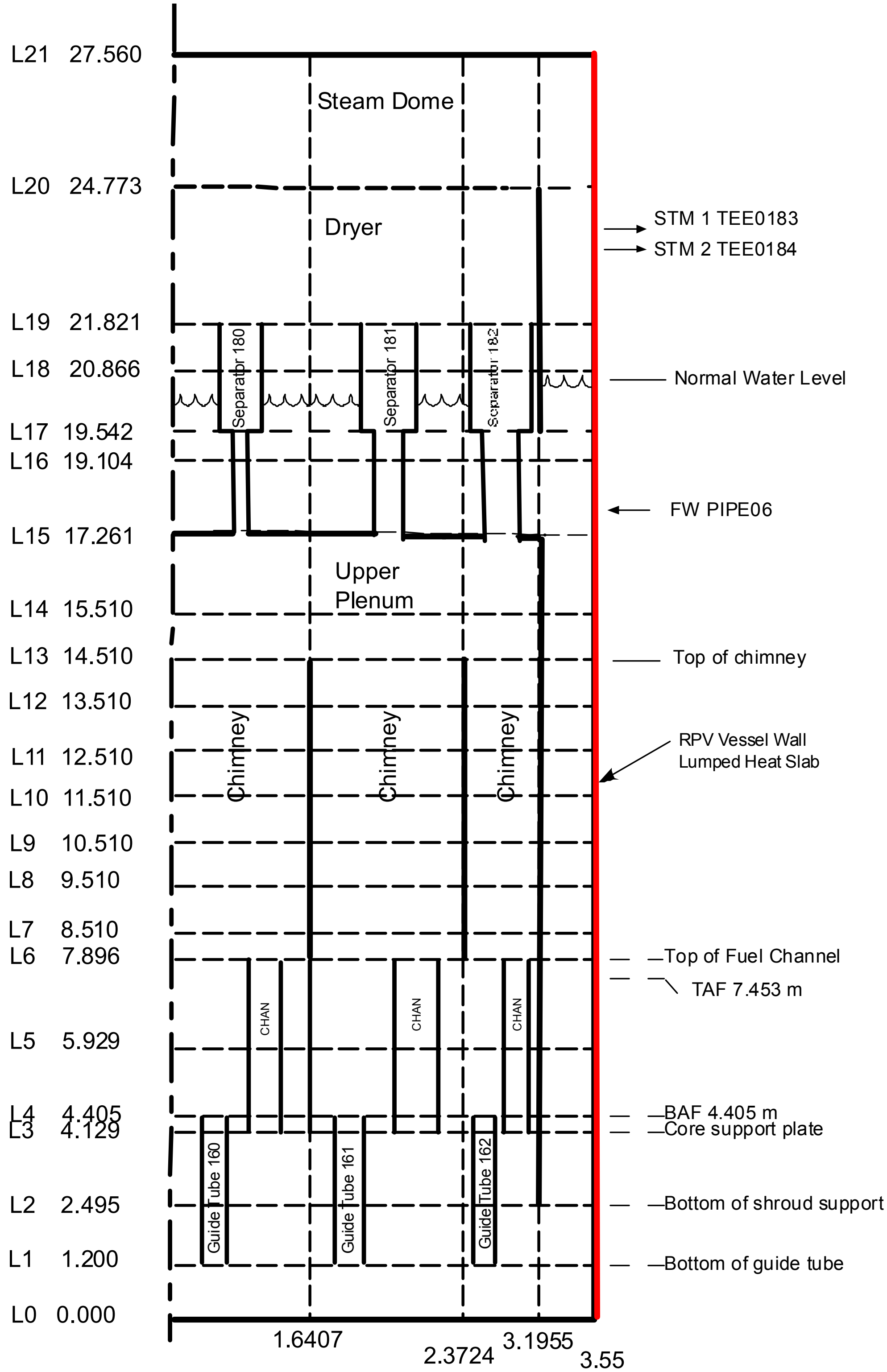


Figure 6A-2 RPV Nodalization showing Vessel Wall Heat Slab

6B. EVALUATION OF THE TRAGG NODALIZATION FOR THE ESBWR LICENSING ANALYSIS

The analysis for the ESBWR containment evaluation follows the application methodology outlined in Reference 6B-1. The TRACG nodalization approach in the licensing analysis is similar to that used in Reference 6B-1. However, this licensing nodalization includes some additional features and details. Some of these features are implemented to address the confirmatory items listed in the Safety Evaluation Report of Reference 6B-1. Other features are implemented due to design changes. Table 6.2-6a summarizes the list of these changes in the TRACG nodalization.

This appendix provides the justification for the use of the DCD nodalization (DCD Tier 2, Figures 6.2-6 and 6.2-7), including the results of the tie-back calculations using the DCD nodalization and the nodalizations presented in Reference 6B-1.

Tie-back calculations were performed with the combined TRACG nodalization similar to that presented in the DCD Tier 2, Figures 6.2-6 and 6.2-7. The results for these calculations were compared with those using the TRACG nodalizations presented in Reference 6B-1. These tie-back calculations include a total of three cases used in Reference 6B-1, two for the ECCS analysis (short-term calculation) and one for the containment analysis (long-term calculation). Results of this comparison show that the calculations using the combined TRACG nodalization compared well with those from the base cases in Reference 6B-1, and the impacts due to nodalization changes on the minimum chimney static head level (+0.1 to -0.16 m) and on the long-term drywell pressure (< 0.3 psia) are judged to be small by comparing to the margins.

The following paragraphs discuss the results of these tie-back calculations, including the effect on the non-condensable (NC) gases holdup, mixing and stratification.

<u>Combined Nodalization</u>

In the TRACG Application Report (Reference 6B-1), two nodalizations are used for the calculations, Figure 6B-1 (from Figure 2.7-1 in Reference 6B-1) for the ECCS analyses (short-term) and Figure 6B-2 (from Figure 3.7-1 in Reference 6B-1) for the Containment (long-term) analyses. In the SER for Reference 6B-1, several subjects regarding these two nodalizations are included as Confirmatory Items in the report. A combined nodalization for use in both the short-term and the long-term calculations is desirable for the ESBWR licensing analyses. This combined nodalization addresses the confirmatory items in the NEDC-33083P-A report (Reference 6B-1). (DCD Tier 2, Table 6.2-6a).

The combined nodalization merges the detailed RPV model in Figure 6B-1 with the detailed containment model in Figure 6B-2. For comparison purpose, the combined nodalization is developed in two steps. In the first step, the component TEE35 is used to model the lower drywell. And in the second step, component TEE35 is replaced with additional vessel levels and cells to model the lower drywell.

Model COMB-5 (with TEE35 to model the Lower Drywell)

Figure 6B-3 shows the combined nodalization in the first step (Model COMB-5, with TEE35 as the Lower Drywell). The vessel component in this model has a total of 39 axial levels and 8 rings.

Cells in VSSL Levels 1 to 21, Rings 1 to 4 in Model COMB-5 are used to model the RPV, identical to that as shown in Figure 6B-1 (RPV portion). Cells in Levels 1 to 21, Rings 5 to 8 in Model COMB-5 are dummy cells or dead cells. These cells are not participating in the calculation.

Cells in VSSL Levels 22 to 32, Rings 1 to 4 in Model COMB-5 are dummy cells or dead cells. Cells in Levels 22 to 32, Rings 5 to 8 in Model COMB-5 are used to model the drywell, suppression pool, wetwell, and GDCS pool. These VSSL cell inputs are converted from those containment cells in Levels 1 to 11, Rings 3 to 6, shown in Figure 6B-2.

Cells in VSSL Levels 33 to 39, Rings 1 to 2 in Model COMB-5 are dummy cells or dead cells. Cells in Levels 33 to 39, Rings 3 to 8 in Model COMB-5 are used to model the PCC/IC pool. These VSSL cell inputs are converted from those containment cells in Levels 12 to 18, Rings 1 to 6, shown in Figure 6B-2.

The one-dimensional TRACG components in COMB-5 are converted from those in Figure 6B-1 and Figure 6B-2. The drain tanks (TEE62 and TEE46) in Figure 6B-2 are not shown in Figure 6B-3.

Model COMB-6 (with VSSL Cells to model the Lower Drywell)

Figure 6B-4 shows the combined nodalization in the second step (Model COMB-6), replacing component TEE35 with additional vessel levels and cells to model the lower drywell. The vessel component in this model has a total of 42 axial levels and 8 rings, 3 more axial levels than that in Model COMB-5.

VSSL Levels 1 to 21 in COMB-6 are identical to those in COMB-5. VSSL Levels 22 and 23, Rings 1 to 6, are used to model the lower drywell. The total volume in these cells is calculated to be the same as that in TEE035. Cells in Levels 24 to 34, Ring 5 to 8 in Model-6 are used to model the drywell, wetwell, suppression pool, and GDCS pools. These cells are converted from those in COMB-5, Levels 21 to 31.

In COMB-6, one additional axial (Level 31) is added at a location close to the top of wetwell. The main purpose of this additional axial layer is to simulate the thin layer near the top of the wetwell. The NC gases trapped inside this thin layer (between the I-beams) will have restricted flow paths and therefore more thermal stratification. Axial Level 35, Cells 1 to 4 are used to model the DW head airspace. Axial Level 35, Cells 7 to 8 are used to model a small airspace above the GDCS pool. The connection PIPE81 and PIPE82 connect this level and Level 31 (top of WW).

Cells in Levels 36 to 42, Rings 3 to 8 in Model COMB-6 are used to model the PCC/IC pool, same as those in Level 33 to 39 in Model COMB-5.

The one-dimensional TRACG components in COMB-6 are same as those in COMB-5. The drain tanks (TEE62 and TEE46) in Figure 6B-2 are not shown in Figure 6B-4.

Tie-Back Calculations

The baseline case (nominal main steam line break, MSLB-NL2_V40) discussed in Section 3.7.2 in Reference 6B-1, is calculated using the combined nodalizations COMB-5 and COMB-6. The initial conditions and thermo-hydraulic conditions in the cells (such as volume, pressure, temperature, etc.) are consistent among these cases: MSLB-NL2_V40, MSLB-CB5 and MSLB-CB6. All these cases are run using the same TRACG04 Version. Results from these calculations are discussed and compared in the following paragraphs.

Comparison between MSLB-CB5 and MSLB-NL2_V40 (0 – 72 hr)

Figure 6B-5 shows the GDCS pool and downcomer water levels for Case “MSLB-CB5”, and Figure 6B-6 shows the GDCS pool and downcomer water levels for Case “MSLB-NL2_V40”. For Case “MSLB-NL2_V40”, the RPV is modeled by 11 axial levels and 2 rings (Figure 6B-2). The downcomer level response for this Case (Figure 6B-6) is more exaggerated because of the coarse nodalization for the RPV. The downcomer level response for Case “MSLB-CB5” (Figure 6B-5) is milder and smoother due to finer nodalization for the RPV (21 axial levels and 4 rings).

Figure 6B-7 compares the total PCC condensation powers between these two Cases. For “MSLB-NL2_V40”, because of the exaggerated DC level response, the condensation power shows spiky drops at the time of the spiky increases in the DC level (Figure 6B-6). The additional subcooled water in the downcomer consumes part of the decay energy and reduces the production of steam that is feeding the PCCS. For the first 9 hrs, the total condensation power for Case “MSLB-CB5” is slightly less than that for Case “MSLB-NL2_V40”. The difference in energy between the decay heat and the total PCC power is discharged into the suppression pool and heats up the pool water. In this case, Case “MSLB-CB5” discharges more energy to the suppression pool. This is the consequence of more accurate nodalization of the RPV. The total PCC condensation power is greater than the decay heat after 9 ~ 10 hours. From that time on, the added energy to the suppression pool water due the movement of NC gases from the DW to the WW is not significant because of the over-capacity in the PCCS.

Figure 6B-8 compares the suppression pool water temperature at the surface. For both cases, the pool surface temperatures increase during the first 9 hrs. However, the pool surface temperature for Case “MSLB-CB5” is higher than that for Case “MSLB-NL2_V40” by about 8 ºF at the end of 9 hr (because the PCCS condensation power is lower for the Case “MSLB-CB5”).

Figure 6B-9 shows the Drywell partial NC gas pressures for Case “MSLB-CB5”, and Figure 6B-10 shows the Drywell partial NC gas pressures for Case “MSLB-NL2_V40”. During the first 18 hours, Case “MSLB-CB5” retains more NC gases in the DW than that for Case “MSLB-NL2-V40”. It takes 54 hours to purge all DW NC gases for Case “MSLB-CB5”, while it takes 48 hours for Case “MSLB-NL2-V40”. The long-term NC gas distribution depends on the NC gas circulation pattern, which depends on the DW annulus geometry and the strength of the steam source. However the difference in purged timing has no significant impact on the long-term DW pressure (Figure 6B-11).

Figure 6B-11 compares the DW pressures between these two Cases. During the first 9 hours, the DW pressure for Case “MSLB-NL2_V40” is higher than that for Case “MSLB-CB5”, due to lesser NC gases remaining in the DW. After all the NC gases have been purged into the WW (48 hrs for Case “MLSB-NL2_V40”, 54 hrs for Case “MSLB-CB5”), the DW pressures reach the maximum value for the rest of the transient. The maximum DW pressure for Case “MSLB-

CB5" is 1.3 psia higher than that for Case "MSLB-NL2-V40", corresponding to the higher suppression pool surface temperature in Case "MSLB-CB5" (Figure 6B-8).

Comparison between MSLB-CB6 and MSLB-CB5 (0 – 72 hr)

Figure 6B-9 shows the Drywell partial NC gas pressures for Case "MSLB-CB5", and Figure 6B-12 shows the Drywell partial NC gas pressures for Case "MSLB-CB6". During the first 18 hours, Case "MSLB-CB5" retains more NC gases in the DW than that for Case "MSLB-CB6". It takes 54 hours to purge all DW NC gases for Case "MSLB-CB5", while it takes 21 hours for Case "MSLB-CB6". The DW pressure reaches the maximum value when all the NC gases have been purged in the WW (Figure 6B-14). The difference in the timing for purging all NC gases shows no impact on the peak value.

Figure 6B-13 compares the RPV, DW and WW pressures for Case "MSLB-CB6". After all the NC gases have been purged into the WW (21 hrs for Case "MSLB-CB6"), the DW pressure reaches the maximum value for the rest of the transient. There are 12 VB openings between 21 and 72 hours, as shown in Figures 6B-12 and 6B-13. Small amount of NC gases are cycling back and forth between the DW and WW. These VB openings have no impact on the peak DW pressure (Figure 6B-14).

Figure 6B-14 compares the DW pressures between Cases "MSLB-CB6", "MSLB-CB5" and "MSLB-NL2_V40". After all the NC gases have been purged into the WW (48 hrs for Case "MSLB-NL2_V40", 54 hrs for Case "MSLB-CB5", and 21 hrs for Case "MSLB-CB6"), the DW pressures reach the maximum value for the rest of the transient. The maximum DW pressure for Case "MLSB-CB5" is higher than that for Case "MSLB-NL2_V40", corresponding to the higher suppression pool surface temperature in Case "MSLB-CB5" (Figure 6B-8). The maximum DW pressure for Case "MSLB-CB6**"** is 0.3 psia lower than that for Case "MSLB-CB5", when all NC gases have been purged into the WW.

Figure 6B-15 shows the suppression pool temperatures for Case "MSLB-CB6", and Figure 6B-16 shows the suppression pool temperatures for Case "MSLB-CB5". The peak pool temperature for Case "MSLB-CB6" is about 1 ºF lower than that for Case "MSLB-CB5".

Figure 6B-17 shows the PCC condensation power for Case "MSLB-CB6", and Figure 6B-18 shows the PCC condensation power for Case "MSLB-CB5". For Case "MSLB-CB6", there are 12 spikes in the total PCC condensation power between 21 and 72 hours, corresponding to the VB openings. The PCC condensation powers between Cases "MSLB-CB5" and "MSLB-CB6" agree well as shown in these figures.

Summary

Main steam line breaks were simulated using the combined nodalization (with and without the TEE35 as the lower DW).

The downcomer level response for the cases using the combined nodalization is milder and more accurate compared to the base case, which is exaggerated because of the coarse nodalization for the RPV. The suppression pool surface temperature is about 8 ºF higher than that for the base case, due to slightly more energy discharging to the suppression pool during the first 9 hrs. The long-term DW pressure is about 1.3 psia higher than that in the base case, due to higher suppression pool surface temperature.

Modeling the lower DW with VSSL cells leads to shorter time period (21 hours for VSSL cells versus 54 hours for TEE35) for purging all drywell NC gases into the WW. However, the effect of purging time period on the peak, long-term DW pressure is small and is less than 0.3 psia (Figure 6B-14). The relatively small effect is due to that the total PCC condensation power is greater than the decay heat after 9 ~ 10 hours. From that time on, the added energy to the suppression pool water due the movement of NC gases from the DW to the WW is not significant because of the over-capacity in the PCCS. The DW pressure reaches the maximum value when all the NC gases have been purged in the WW (Figure 6B-14). The difference in the timing for purging all NC gases and the subsequent V/B openings show no significant impact on the peak value.

Results of this comparison show that the calculations using the combined TRACG nodalization compared well with those from the base cases, and the impacts due to nodalization changes on the minimum chimney static head level (+0.1 to -0.16 m) and on the long-term drywell pressure (< 0.3 psia) are judged to be small by comparing to the margins.

Reference

6B-1. GE Nuclear Energy, "TRAGG Application for ESBWR," NEDC-33083P-A, Class III, (Proprietary), March 2005, and NEDO-33083-A, Class I (non-proprietary), October 2005.

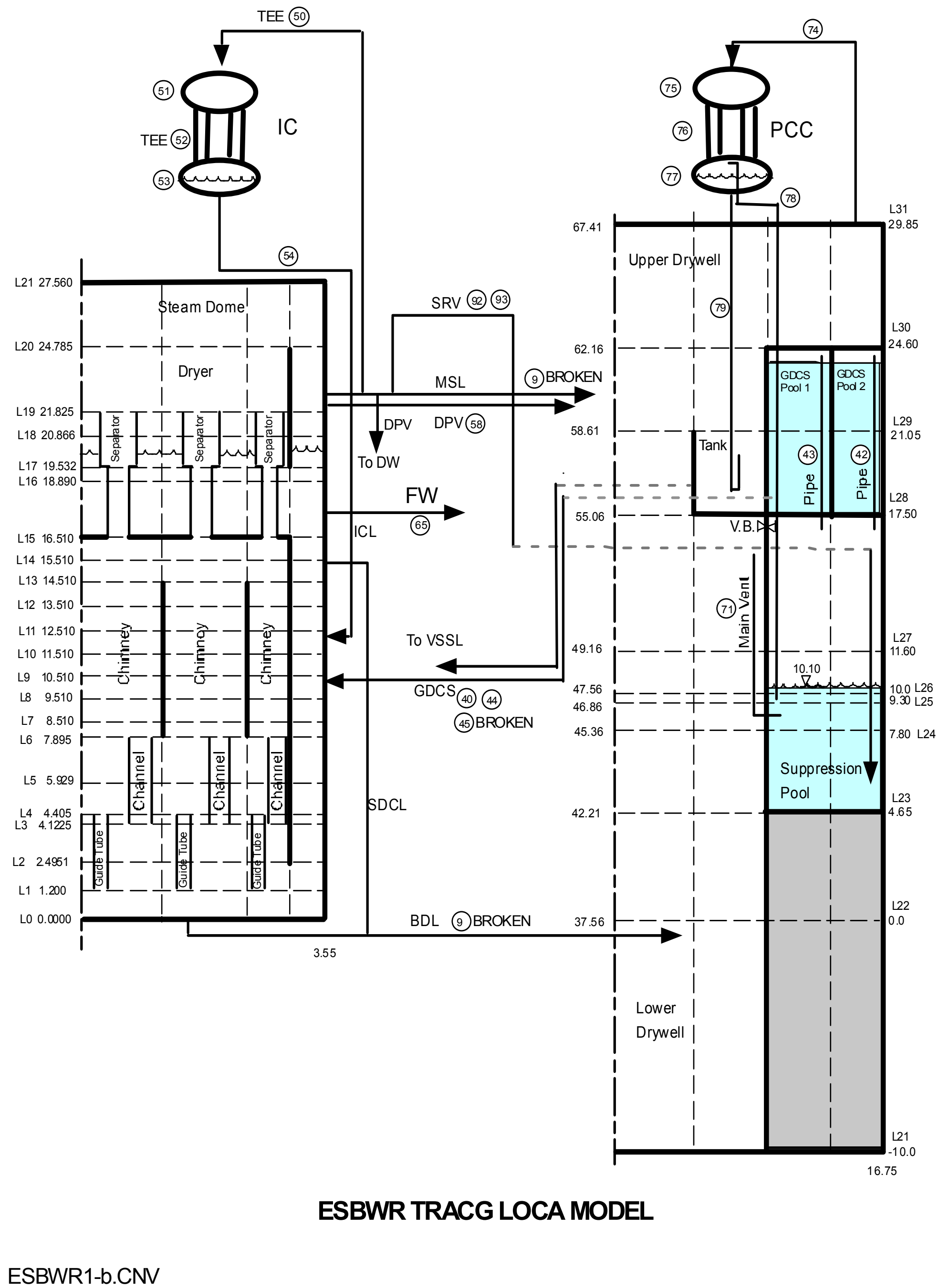


Figure 6B-1 TRACG Nodalization for ESBWR ECCS/LOCA Analysis (NEDC-33083P-A)

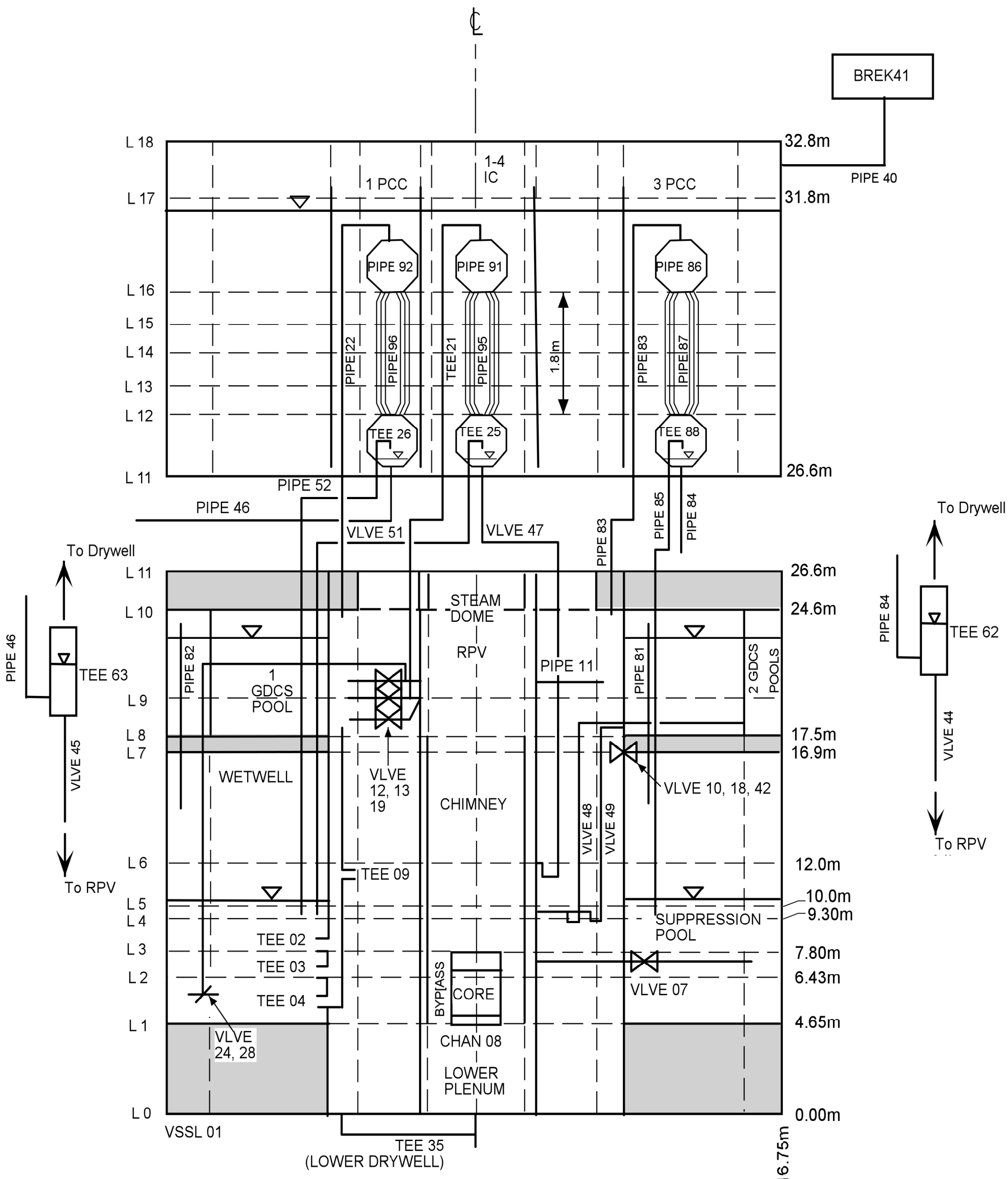


Figure 6B-2 TRACG Nodalization for ESBWR Containment Analysis (NEDC-33083P-1A)

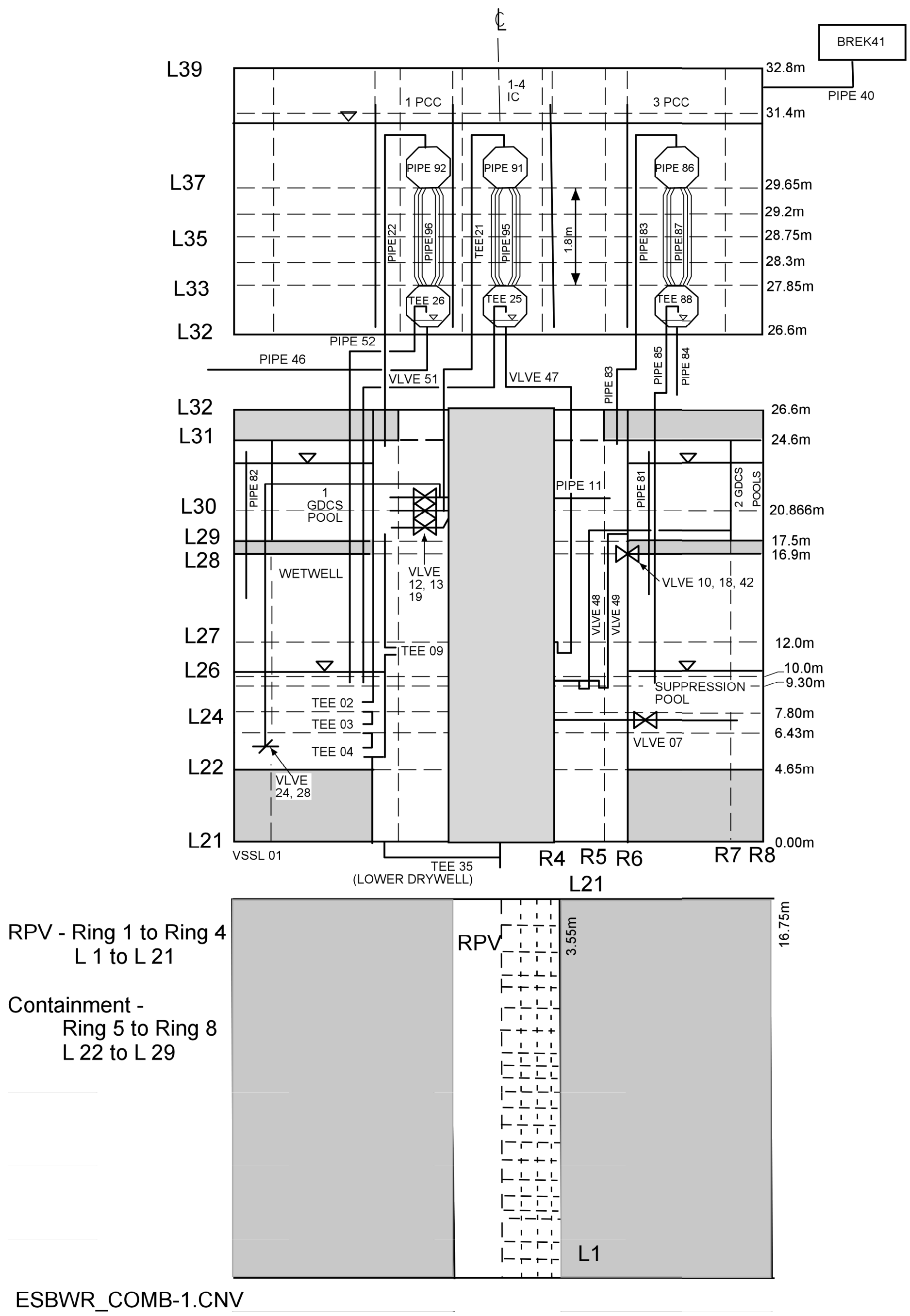


Figure 6B-3 TRACG Combined Nodalization (Model COMB-5, with TEE35 as Lower Drywell)

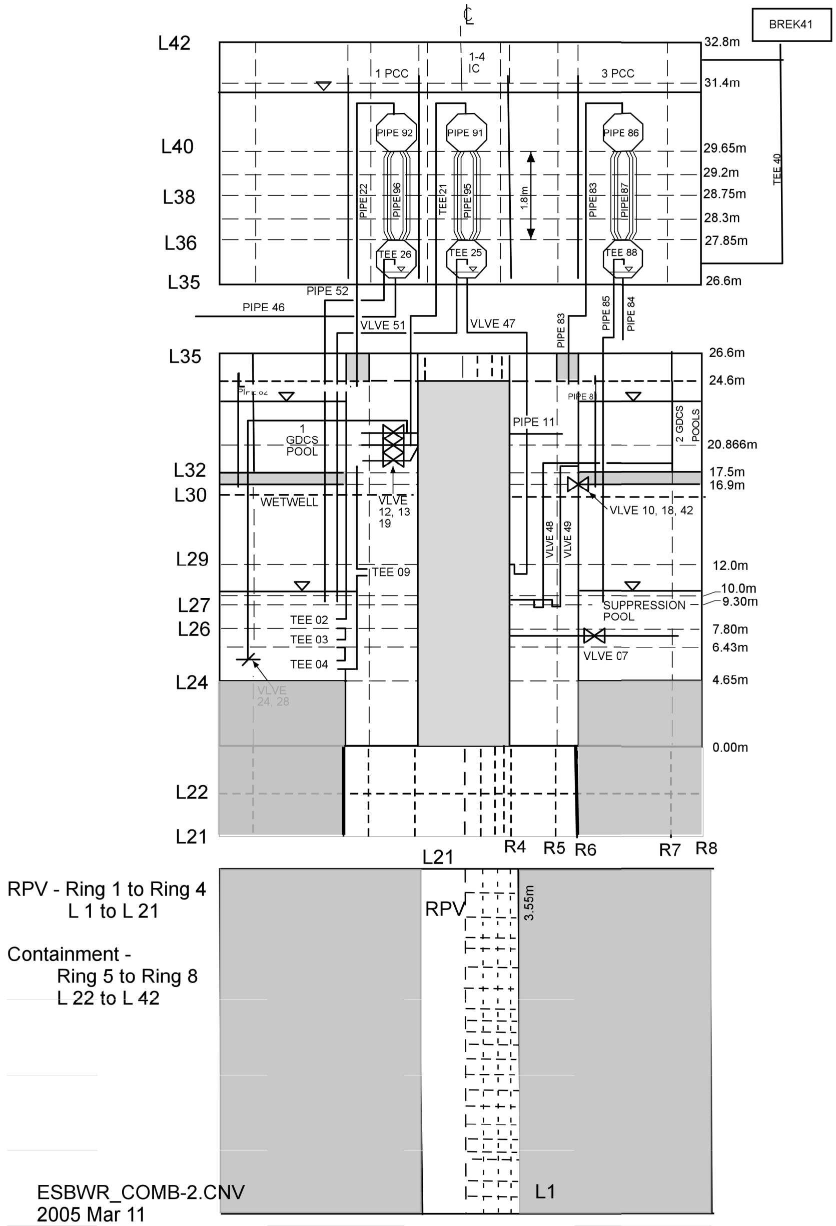


Figure 6B-4 TRACG Combined Nodalization (Model COMB-6, without TEE35 as Lower Drywell)

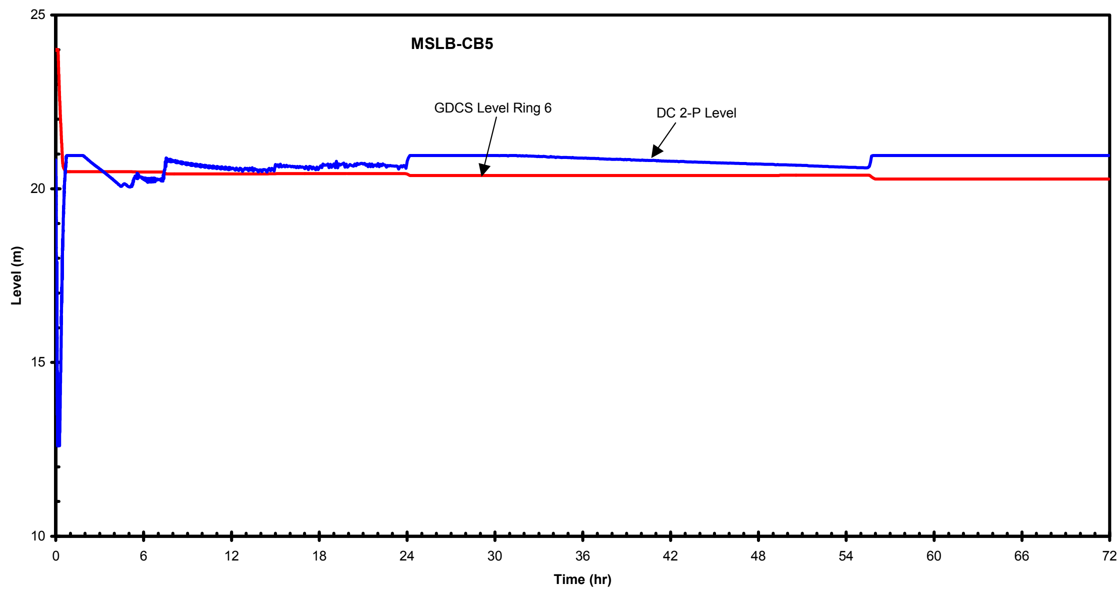


Figure 6B-5 MSLB-CB5 – GDCS and DC Water Levels

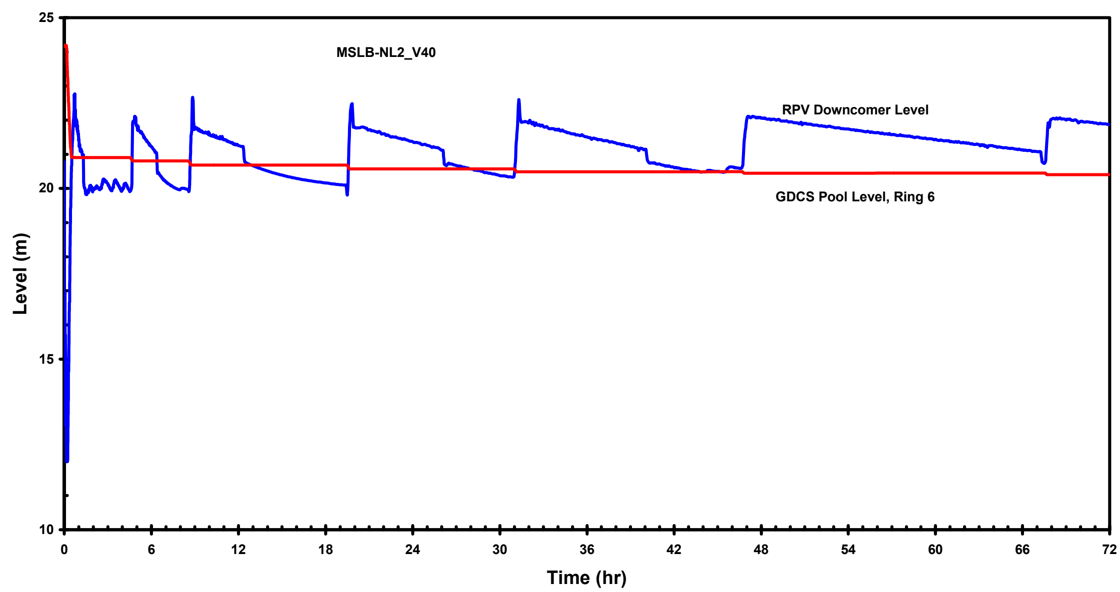


Figure 6B-6 MSLB-NLW_V40 – GDCS and DC Water Levels

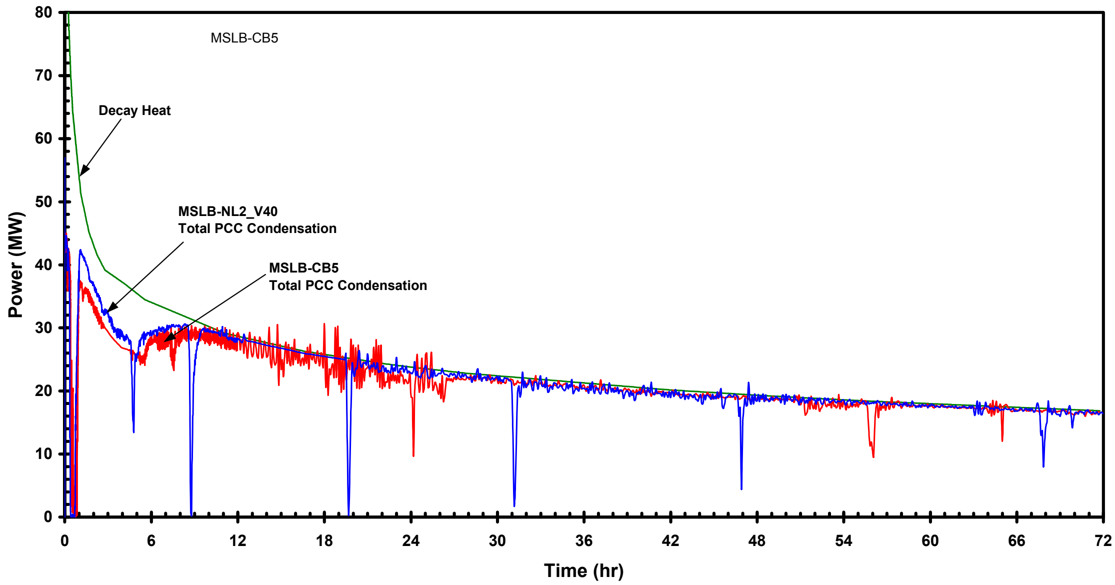


Figure 6B-7 MSLB-CB5 vs. MSLB-NLW_V40 – Total PCC Condensation Powers

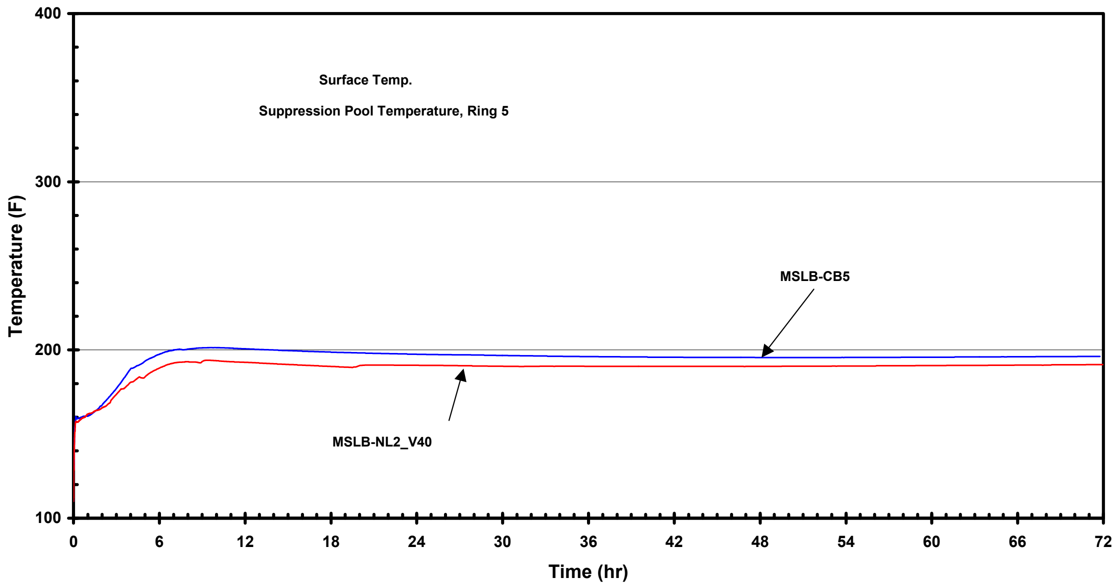


Figure 6B-8 MSLB-CB5 vs. MSLB-NLW_V40 – Suppression Pool Surface Temperatures

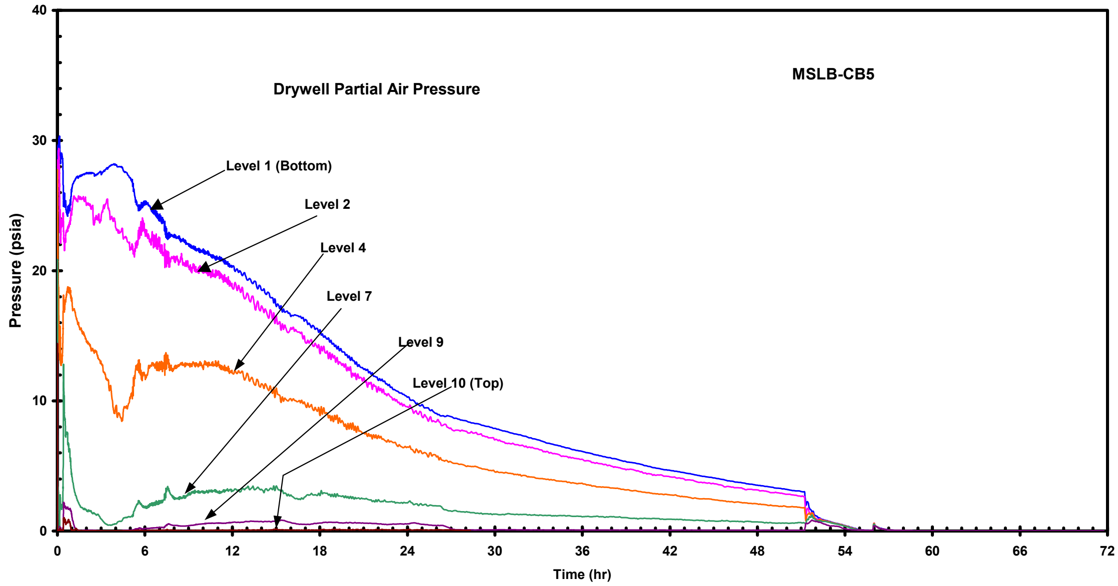


Figure 6B-9 MSLB-CB5 – Drywell Partial NC Gas Pressures

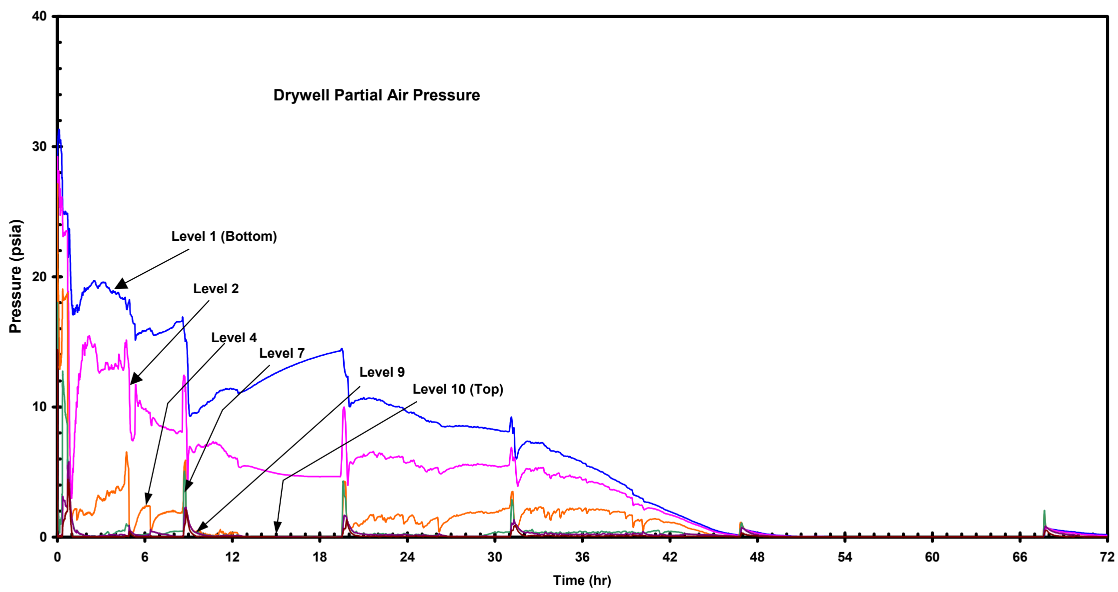


Figure 6B-10 MSLB-NL2_V40 – Drywell Partial NC Gas Pressures

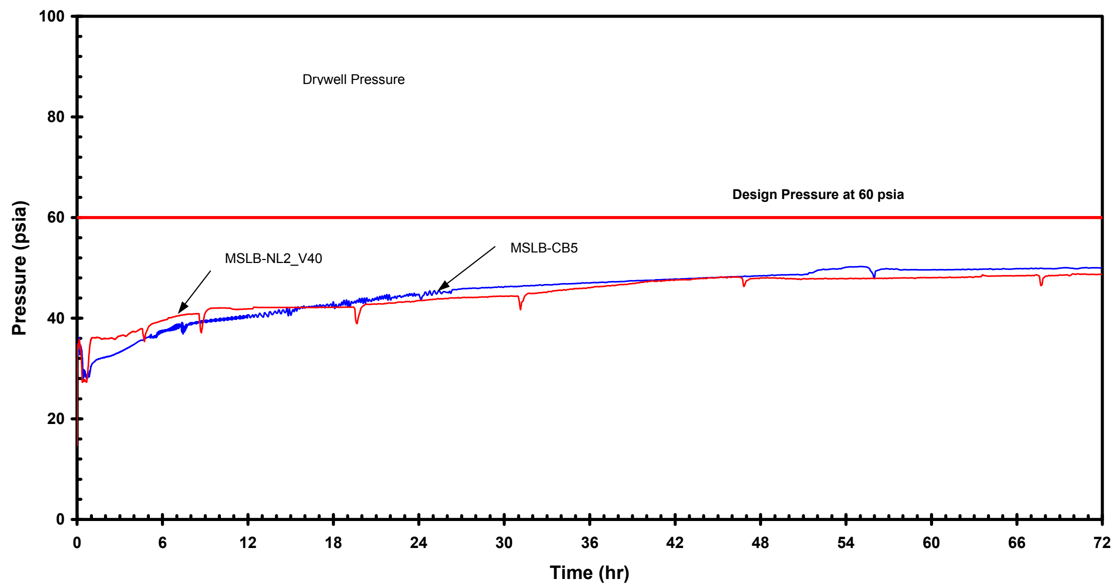


Figure 6B-11 MSLB-CB5 vs MSLB-NLW_V40 – Drywell Pressures

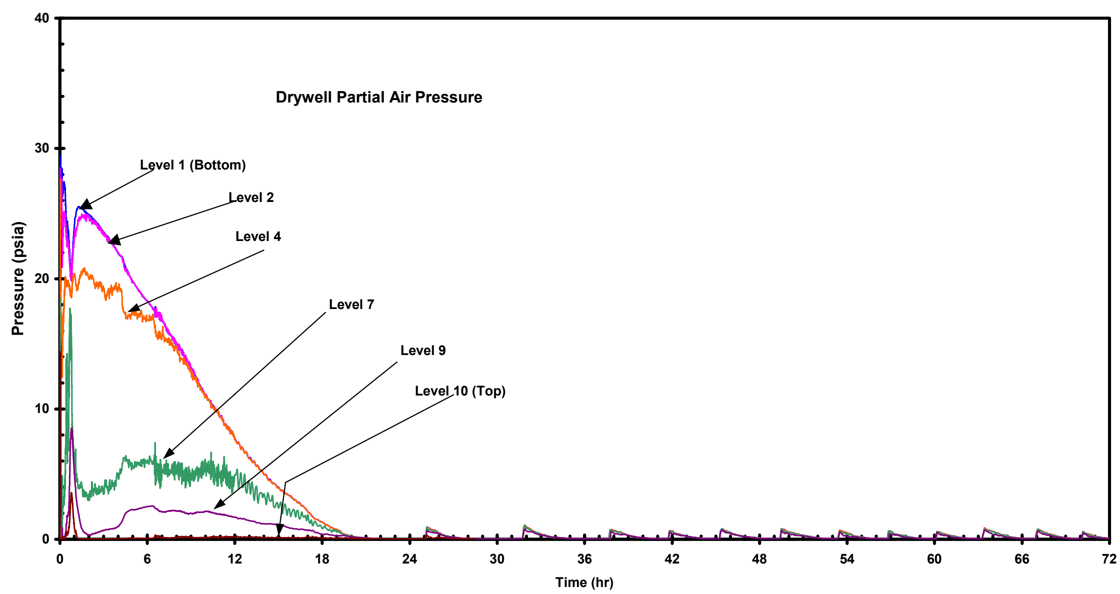


Figure 6B-12 MSLB-CB6 – Drywell Partial NC Gas Pressures

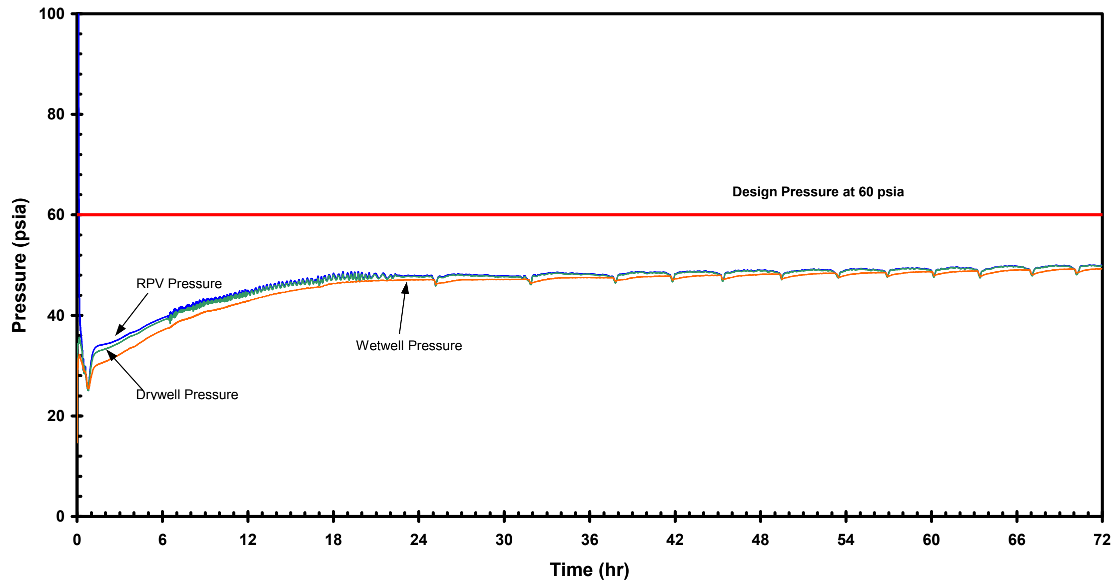


Figure 6B-13 MSLB-CB6 – RPV, Drywell and Wetwell Pressures

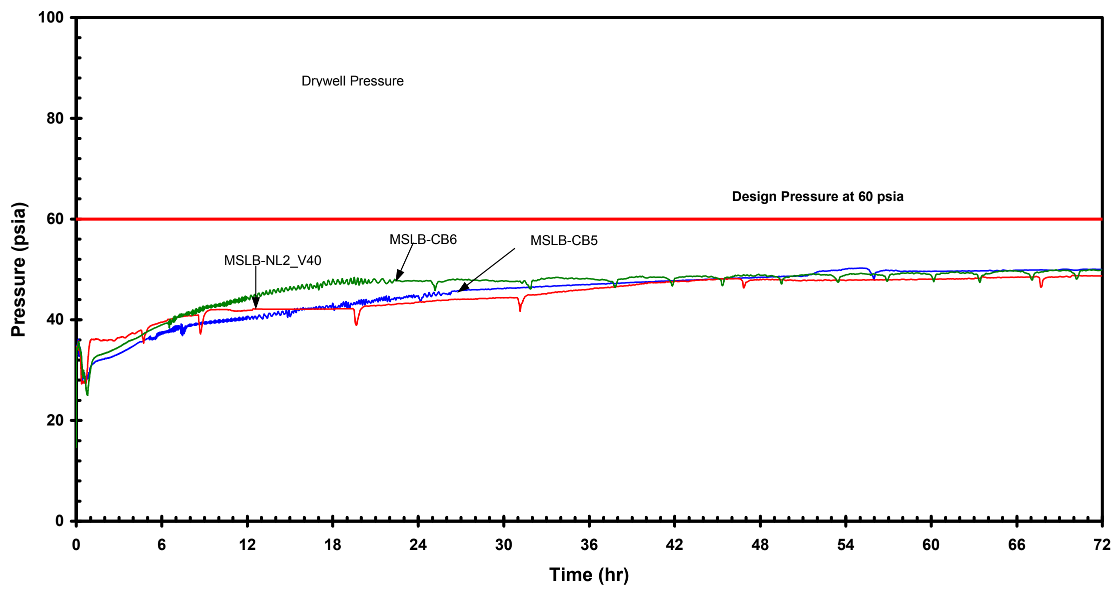


Figure 6B-14 Drywell Pressure Comparison – MSLB-CB6, MSLB-CB5 and MSLB-NL2_V40

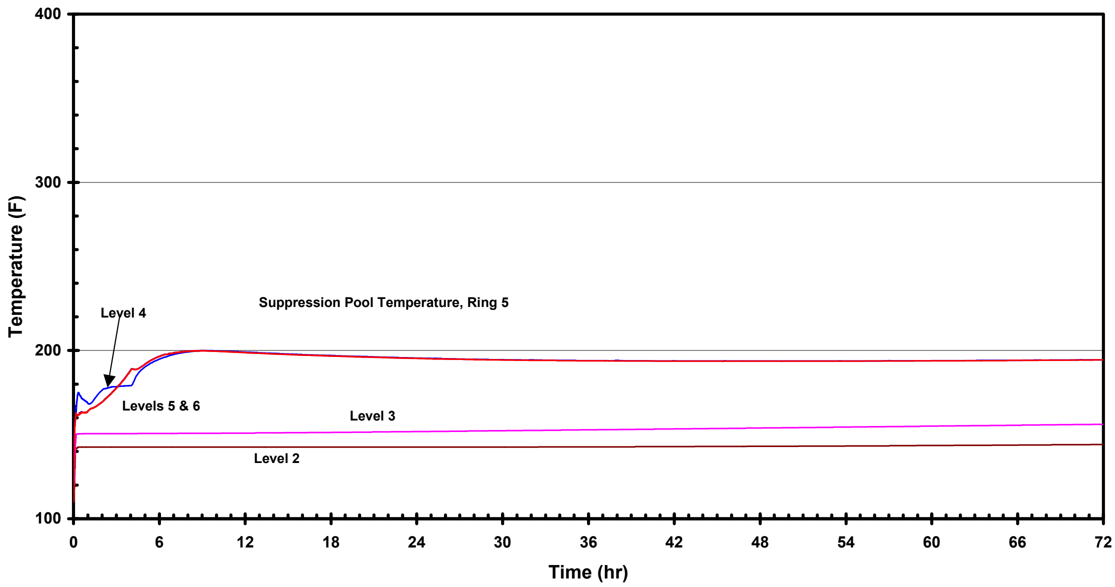


Figure 6B-15 MSLB-CB6 – Suppression Pool Temperatures

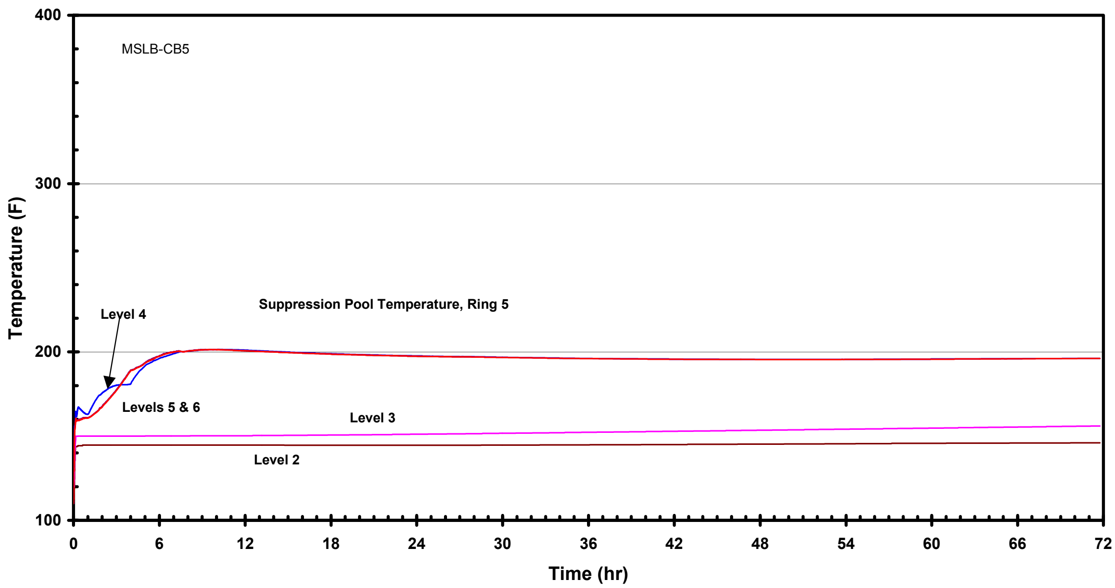


Figure 6B-16 MSLB-CB5 – Suppression Pool Temperatures

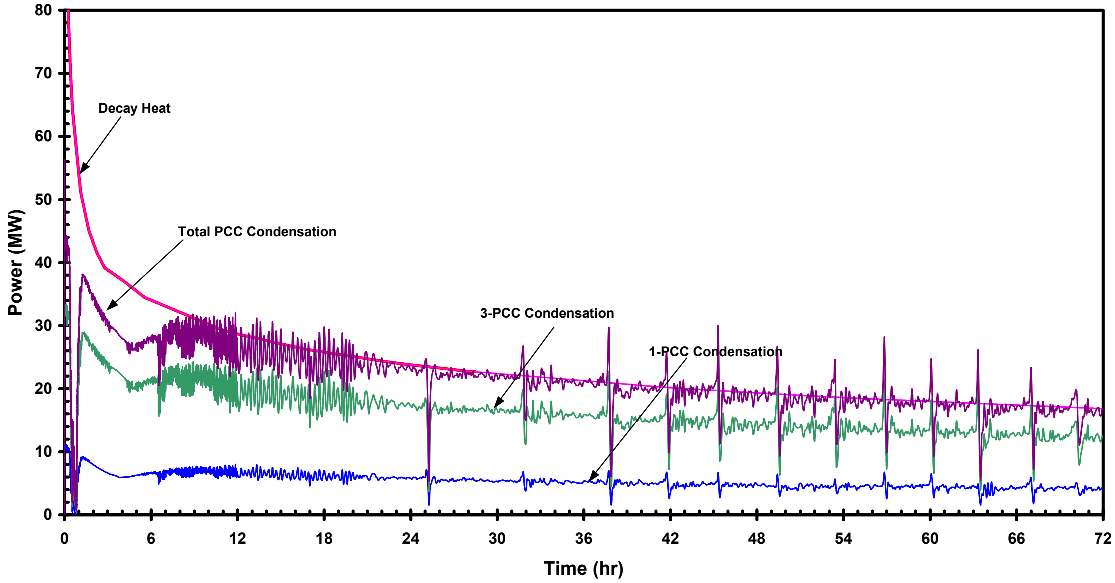


Figure 6B-17 MSLB-CB6 – PCC Condensation Power

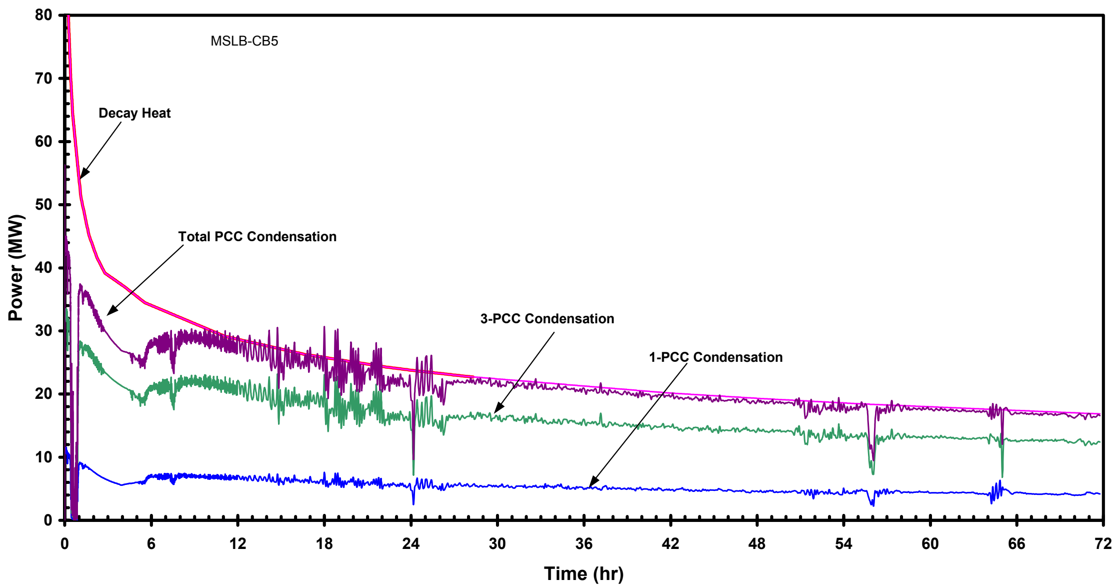


Figure 6B-18 MSLB-CB5 – PCC Condensation Power

6C. EVALUATION OF THE IMPACT OF CONTAINMENT BACK PRESSURE ON THE ECCS PERFORMANCE

This Appendix provides the justification for the use of the combined TRACG nodalizaton, that integrates the responses between the containment and the reactor vessel, for the ECCS performance analyses. This evaluation considers the Standard Review Plan (SRP) Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," Revision 2, July 1981 and Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." The following paragraphs and figures summarize the details of this evaluation.

The ESBWR analyses (presented in DCD Tier 2, Section 6.3.3) model the containment back pressure in the calculation of the minimum chimney collapsed level following a LOCA. The input information for the model, active heat sinks and passive heat sinks affect the containment back pressure (Standard Review Plan Section 6.2.1.5), which could affect the ECCS performance. However, the GDCS, the drywell/wetwell and the RPV are interconnected in the ESBWR design. And the GDCS performance depends mainly on the gravity head difference between the GDCS pool and the RPV, and not on the containment back pressure. Results of parametric cases, which conservatively model the additional passive and active heat sinks, shows that the impacts on the containment back pressure are less than 20 kPa. The effects of this change in containment back pressure on the minimum chimney collapsed level are less than 0.1 m (6% of the level margin to the top of the active fuel). Results also show that the minimum chimney collapsed level is not sensitive for a wider range of change in the containment back pressure.

Parametric cases have been performed to assess the effect of containment back pressure on the minimum chimney collapsed level. For evaluation purpose, the GDCS line break with failure of one injection valve (nominal case) is selected as the base case. One case includes conservative modeling of additional passive heat sinks (containment walls and internal structures), and another case includes conservative modeling of both the additional passive heat sinks and active heat sink (1000 gpm drywell spray). Three additional parametric cases are performed to bound the impact of change in model input information on the containment back pressure. In these three cases, the wetwell and suppression pool volumes are artificially increased by a factor of 1.5, 2.0 and 3.0. Larger wetwell volume results in smaller drywell pressurization rate and therefore lower containment back pressure at the time of GDCS flow initiation. These three cases provide the bounding estimate of the containment back pressure and therefore the bounding effect on the minimum chimney collapsed level.

The key parameters, for comparison purpose, are the maximum wetwell pressure at around the time of GDCS flow initiation, the GDCS flow initiation timing, and the minimum chimney collapsed level. The results show that the maximum wetwell pressure decreases with the additional drywell steam condensing heat sinks such as the drywell spray and additional drywell heat structures, it also decreases with larger wetwell volume. With the additional passive and active heat sinks, the maximum wetwell pressure decreases by 17 kPa from the base case value.

Figure 6C_1 shows the effect of the maximum wetwell pressure on the GDCS injection timing. The initiation time for the GDCS flow increases linearly as the wetwell pressure reduces. With the additional passive and active heat sinks, the initiation time increases by 7 seconds from the base case value.

Figure 6C_2 shows the effect of the maximum wetwell pressure on the minimum chimney collapsed level. This figure shows that the minimum chimney collapsed level is not sensitive to the change in the containment back pressure. With the additional passive and active heat sinks, the minimum chimney collapsed level reduces by less than 0.1 m from the base case value. This reduction in minimum chimney collapsed level corresponds to 6% of the level margin to the top of the active fuel.

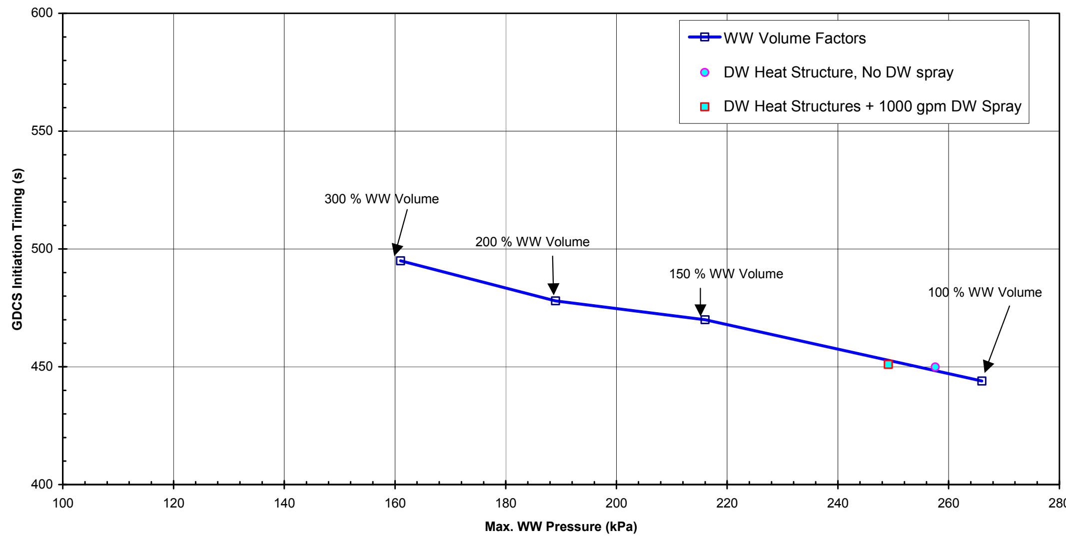


Figure 6C_1 Effect of Wetwell Pressure on the GDCS Initiation Timing (GDCS Injection Line Break, 1 GDCS Injection Valve Failure)

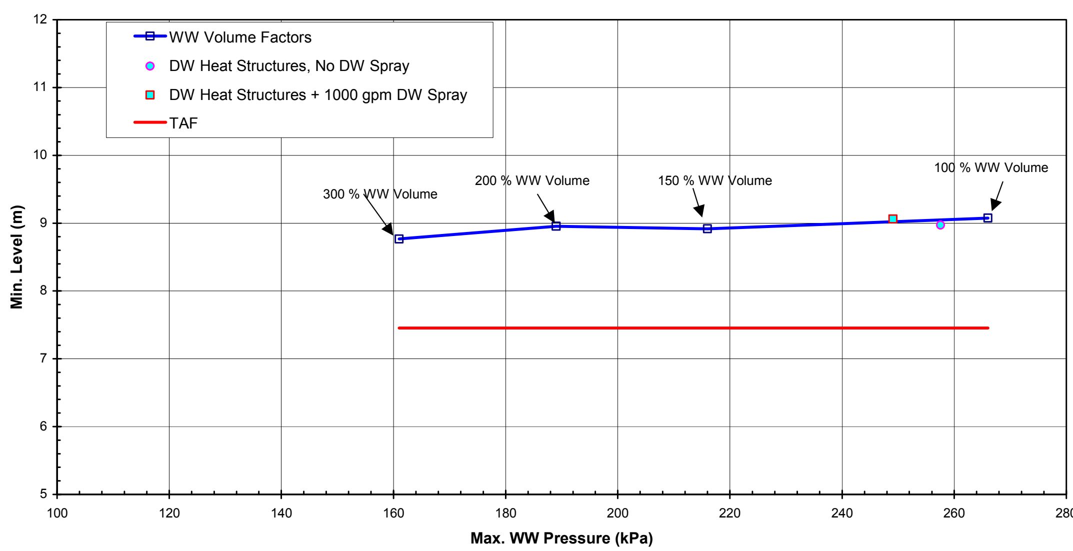


Figure 6C_2 Effect of Wetwell Pressure on the Minimum Chimney Collapsed Level (GDCS Injection Line Break, 1GDCS Injection Valve Failure)